Nebraska Public Power District
Nebraska's Energy Leader

NLS2002014
February 26, 2002

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

Subject: License Condition 2.C.(6) Seismic Evaluation
Cooper Nuclear Station
NRC Docket No. 50-298, DPR-46

Reference:

1. Letter to J. Swailes (Nebraska Public Power District) from L. Burkhart (U. S. Nuclear Regulatory Commission), dated April 7, 2000, "Cooper Nuclear Station - Issuance of Amendment on Design-Basis Accident Radiological Assessment Calculational Methodology Revision (TAC No. MA7758)."

2. Letter to D. Wilson (Nebraska Public Power District) from M. Thadani (U. S. Nuclear Regulatory Commission), dated October 23, 2001, "Cooper Nuclear Station - Issuance of Amendment Regarding Revised Radiological Dose Assessment and Technical Specification Changes (TAC No. MB1419)."

3. Letter to U. S. Nuclear Regulatory Commission from J. Swailes (Nebraska Public Power District), dated February 28, 2001, "Proposed License Amendment Related to the Design Basis Accident Radiological Assessment Calculational Methodology."

The purpose of this letter is to submit the Seismic Evaluation required by Cooper Nuclear Station License Condition 2.C.(6) and to request its review and approval by the Nuclear Regulatory Commission (NRC). In Reference 1, the NRC issued License Amendment 183 which included License Condition 2.C.(6). This Condition states:

> No later than 8 weeks after the Cooper Nuclear Station (CNS) Cycle 21 startup, the licensee shall submit a request for the staff to review and approve a seismic evaluation to ensure the structural integrity of the main steam line piping from the main steam isolation valves (MSIV) to the main turbine condenser, the main turbine condenser, and the turbine building. The evaluation will be performed to assess the ability of the aforementioned main steam piping and main turbine condenser to remain sufficiently intact to direct main

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> steam leakage from the MSIVs to the main turbine condenser, consistent with the leakage assumptions in the design-basis accident dose calculations during and after a Safe Shutdown Earthquake. This seismic evaluation will employ an analytical methodology acceptable to the staff and will identify any modifications necessary to support the evaluation. The licensee's approved request shall be fully implemented, including the completion of modifications, within 12 months of approval or prior to CNS Cycle 22 startup, whichever is later.

In conformance with License Condition 2.C.(6), the Nebraska Public Power District (NPPD) hereby submits the enclosed Engineering Evaluation, and requests NRC review and approval of this enclosure consistent with the License Condition by May 30, 2002. Attachment 1 summarizes the correspondence history related to the issuance of License Condition 2.C.(6) and describes how the Engineering Evaluation meets the submittal requirements of the License Condition. The Engineering Evaluation discusses the full scope of potential plant modifications that may be required to support the seismic evaluation. However, further engineering analysis is required to determine the final flow path to be relied on and to resolve the outliers identified in the attached evaluation. Following NRC approval of the enclosed Engineering Evaluation, NPPD will submit the finalized description of modifications needed to configure the MSIV leakage pathway. That submittal will take into account the resolution of identified NRC concerns, if any, and will be made by July 30, 2002.

In Reference 3, NPPD requested NRC approval of revised design basis accident radiological assessment calculational methodologies. In Reference 2, the NRC approved the radiological dose assessment methodology for the Fuel Handling Accident, and directed interim approval of the Loss-of-Coolant-Accident, the Control Rod Drop Accident, and the Main Steam Line Break Accident for one operating cycle. The NRC further conveyed that the deferred review of the remaining methodologies would continue on a preapplication basis, pending NPPD's submittal of the seismic evaluation of the adequacy of the Main Steam piping, the Main Turbine Condenser and the Turbine Building. With the submittal of the enclosed seismic evaluation, which supplements the previous license amendment request of Reference 3, NPPD requests that the NRC complete their review of the remaining dose calculational methodologies, with a target date of January 30, 2003 for issuance of the License Amendment.

Should you have any questions regarding this matter, please contact David F. Kunsemiller at (402) 825-5236.

Sincerely,

David L. Wilson

David L. Wilson
Vice President of Nuclear Energy

/wrv

Attachment
Enclosure

cc: Regional Administrator w/attachment and enclosure
USNRC Region IV

Senior Project Manager w/attachment and enclosure
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector w/attachment and enclosure
USNRC

NPG Distribution w/attachment w/o enclosure

Records w/attachment and enclosure

STATE OF NEBRASKA)
)
NEMAHA COUNTY)

David L. Wilson, being first duly sworn, deposes and says that he is an authorized representative of the Nebraska Public Power District, a public corporation and political subdivision of the State of Nebraska; that he is duly authorized to submit this correspondence on behalf of the Nebraska Public Power District; and that the statements contained herein are true to the best of his knowledge and belief.

David L. Wilson

David L. Wilson

Subscribed in my presence and sworn to before me this 26 day of Feb, 2002.

Wilma M Werner

NOTARY PUBLIC

Wilma M. Werner
General Notary
State of Nebraska
My Commission Expires Oct. 26, 2002

ATTACHMENT 1

INTRODUCTION

The purpose of this Attachment is four-fold. First, the Attachment provides a background of License Condition 2.C.(6), focusing on NRC concerns with crediting Structures, Systems, and Components (SSCs) for accident mitigation that were not established as able to withstand the loadings of a Safe Shutdown Earthquake (SSE). Second, the Attachment summarizes the enclosed Engineering Evaluation (the Evaluation) to better enable the NRC to focus on information being provided by NPPD which supports its position regarding the level of seismic protection provided for the Main Steam Isolation Valve (MSIV) leakage pathway. In this regard, this summary will: a) provide the means used to define the leakage pathway, b) describe the efforts taken to date to seismically analyze the credited SSCs, and c) demonstrate that the Evaluation methodology used meets NRC-approved criteria based on previous applications. Third, the Attachment summarizes the proposed modifications to configure the main leakage pathway, and the proposed approach to resolving seismic "outlier" issues. Fourth, the Attachment requests NRC review and approval of the Evaluation, and completion of the deferred design basis accident dose calculation reviews.

BACKGROUND

In a letter dated December 22, 1999 [Reference A-1], NPPD submitted a License Amendment Request to revise the design basis accident radiological dose methodology. As discussed in that letter, the accident dose calculations were being revised to incorporate more recent site specific meteorological data, to reflect plant specific system operating parameters and design, to utilize more widely accepted accident assumptions for a facility of CNS's vintage, to incorporate the Technical Information Document (TID-14844) source term (to be consistent with the accident assumptions used), to update fuel parameter considerations to include higher burnup fuel designs, and to utilize generic and updated calculational and software methodologies to perform the analysis. In this submittal, NPPD took credit for iodine plateout in the Main Turbine Condenser (located in the Turbine Building) for the Loss-of-Coolant-Accident (LOCA) and Control Rod Drop Accident (CRDA).

In Question 6 of an NRC Request for Additional Information, dated March 6, 2000 [Reference A-2], the NRC indicated that they were unaware of any licensing precedent where "non-seismic" steam line piping and condensers were credited for iodine removal. The NRC further noted that an approved methodology for crediting iodine removal after an SSE was contained in NEDC-31858P-A, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems." The NRC requested justification as to why this standard methodology was not being used for crediting iodine plateout in the Main Turbine Condenser, and for NPPD to discuss the methodology being used in lieu of this guidance.

NPPD provided a preliminary response in a letter dated March 20, 2000 [Reference A-3], which stated that crediting iodine removal via the existing Main Turbine Condenser design was already a part of the CNS licensing basis for radiological assessment calculation accident mitigation. References to relevant CNS licensing basis documents were also provided. However, NPPD further stated that it was appropriate to evaluate the Main Steam piping from the MSIVs up to and including the Main Turbine Condenser to confirm that these components would remain structurally intact following an SSE. NPPD committed to submit a letter that would both describe the structural robustness of the existing configuration, and would address the low probability of having to rely on this configuration for accident mitigation. Additionally, NPPD committed to propose a License Condition that would address when additional information would be provided to the NRC regarding the ability of the Main Steam piping and Main Turbine Condenser configuration to remain functional during and after an SSE.

In a letter dated March 24, 2000 [Reference A-4], NPPD addressed the issues committed to in the March 20, 2000 correspondence. The credited Main Steam piping, the Main Turbine Condenser, and the Turbine Building were qualitatively demonstrated by NPPD to remain intact following an SSE. This position was further supplemented in a subsequent correspondence dated March 29, 2000 [Reference A-7]. Also, a Probabilistic Safety Assessment summary was provided which evaluated core damage LOCA events in combination with seismic spectra above the Operating Basis Earthquake, but less than the SSE (acting either concurrently or up to 30 days post-LOCA). NPPD also proposed wording for a License Condition pertaining to a future seismic evaluation submittal (including proposed modifications).

The NRC issued License Amendment 183 on April 7, 2000 [Reference A-5]. In the associated Safety Evaluation, the NRC stated that NPPD had provided sufficient information to justify operability of the Main Steam piping and Main Turbine Condenser following an SSE so that iodine removal could be accomplished. However, the NRC also indicated that its long-term acceptance of the current CNS configuration would require the completion of a more technically detailed analysis. License Condition 2.C.(6) was issued (based on the wording previously provided by NPPD) to ensure, in part, that no changes to this commitment would be made without prior NRC approval in accordance with 10CFR50.90.

In summary, NPPD agrees that for purposes of the proposed LOCA radiological assessment calculational methodology [Reference A-9] credit for iodine plateout in the Main Turbine Condenser should only be taken if there is reasonable assurance that the MSIV leakage pathway will remain intact following the loadings of an SSE. NPPD further recognizes the need for a detailed seismic evaluation to ensure the long-term crediting of this pathway.

DESCRIPTION OF THE SEISMIC EVALUATION

The enclosed Engineering Evaluation (the Evaluation) summarizes detailed analyses that have been performed to confirm the seismic robustness of the Main Steam piping, the Main Turbine Condenser, and the Turbine Building. The analyses are based on General Electric (GE) Topical Report, NEDC-31858P, Revision 2, "BWROG Report for Increasing MSIV Leakage Limits and

Elimination of Leakage Control Systems," September 1993. The NRC has endorsed use of this Topical Report in a Safety Evaluation dated March 3, 1999 [Reference A-6]. The NPPD Evaluation, which is consistent with the NRC's Safety Evaluation and previous position on this matter, also describes modifications needed to configure the leakage pathway and to resolve those instances where SSCs could not be credited long-term without modification or more detailed evaluation (referred to herein as "outlier" issues). The Evaluation concludes that following the resolution of certain outliers, including the implementation of appropriate modifications, the subject SSCs will conform with the portions of the NRC's Safety Evaluation for NEDC-31858P relating to seismic robustness.

Pathway Identification- Section 4.2 of the Evaluation discusses the method used to identify the MSIV leakage pathway, and to establish the boundaries of this pathway. The MSIV leakage pathway boundaries for CNS are supported by calculations and are shown on the drawing provided in Attachment 9.1 of the Evaluation.

CNS Seismic Criteria- Section 4.3 of the Evaluation describes the basis for the acceptability of applying the NEDC-31858P methodology and the Seismic Qualification Utility Group-Generic Implementation Procedure (SQUG-GIP) earthquake experience database as a surrogate for similar equipment at CNS. A comparison was made between the ground motions of the earthquakes associated with that database, and the historical earthquakes that formed the basis for the CNS SSE. It was concluded that the CNS SSE ground demand motion was below the seismic ground motion experienced at the facilities credited in the earthquake experience database, and that use of the database was therefore acceptable for establishing the seismic robustness of CNS equipment.

Section 4.4 of the Evaluation describes the CNS Seismic Demand. A conservative value of 2.0 times the SSE ground response spectrum was used in this analysis for piping and equipment located at elevations of the Turbine Building and Reactor Building up to and including 932'-6". The SSE ground spectrum was used for the applied seismic demand for the Main Turbine Condenser, which is located below grade at the lowest level of the Turbine Building.

Turbine Building Seismic Capacity- Section 4.5.2 of the Evaluation discusses the analyses that demonstrate ability of the Turbine Building to withstand the seismic loadings of an SSE. In summary, the increase in design loadings from the original seismic Class II criteria to the postulated SSE condition does not result in stresses that exceed the allowable limits applicable to the SSE load case. This is primarily due to the fact that the increased seismic loading for the SSE load case is offset by an increase in allowable stresses.

Main Turbine Condenser Seismic Capacity- Section 4.5.3 of the Evaluation provides a description of the Main Turbine Condenser and summarizes the evaluation of the Main Turbine Condenser structure and anchorages. This section subsumes information previously provided to the NRC in a letter dated September 4, 2001 [Reference A-8]. The Main Turbine Condenser structure is bounded by existing earthquake experience data, thereby rendering it highly unlikely that a failure and significant breach of the pressure boundary would occur during an SSE. NPPD has calculated that the Main Turbine Condenser anchorages will withstand postulated SSE loadings.

MSIV Leakage Pathway Seismic Analysis- Section 4.5.4 of the Evaluation describes the means used to seismically analyze the MSIV leakage piping pathway per NEDC-31858P. Walkdowns by experienced SQUG-GIP engineering personnel and associated analyses evaluated the seismic capacity of the subject piping system. If the piping and/or piping supports could not be demonstrated to be seismically rugged, these SSCs were designated as outliers, which would require either modification or more detailed analysis to address. Three piping systems shown on Attachment 9.1 were selected for supplementary detailed computer analysis (as required by NEDC-31858P): a) Main Steam system (including the bypass piping) (Pathway P8), b) the primary leakage pathway (Pathway P1), and c) the alternate leakage pathway (Pathway P2). The loads from these analyses were used to conduct detailed evaluations of the associated pipe supports. Table 6-10 of Evaluation Attachment 9.2 provide a summary of supports subjected to detailed analytical reviews and the basis of these reviews. These selected supports represent over 30% of the support population in the MSIV leakage path. In addition, these supports are most susceptible to failure during a design basis seismic event. By demonstrating the acceptability of these supports, it is reasonable to assume that the remaining supports for the MSIV leakage pathway have adequate seismic capacity. MSIV leakage pathway equipment was evaluated to establish whether the GIP screening criteria was met. The outliers of this effort are summarized in Section 4.6 of the Evaluation.

NRC Methodology Approval (Precedence)- As previously discussed, in Reference A-2, the NRC cited NEDC-31858P-A as a previously approved methodology for crediting iodine removal after a Safe Shutdown Earthquake. The CNS MSIV leakage pathway was analyzed using the methodology described in this GE Topical Report. The NRC has previously reviewed this methodology and acknowledged its acceptance of NEDC-31858P, Revision 2, in a Safety Evaluation dated March 3, 1999 [Reference A-6][1]. NPPD has reviewed this document and found that the methodology used for the CNS seismic analysis is consistent with the subject Safety Evaluation. Moreover, in Section 6 of the Safety Evaluation, the NRC identified nine limitations on the use of the GE Topical Report. NPPD has reviewed these limitations with respect to their applicability in Section 4.1 of the Evaluation and provided appropriate disposition for each. NPPD concludes that the Evaluation is in conformance with these limitations, and thus allows the use of NEDC-31858P, Rev. 2 at CNS.

MODIFICATIONS

The following discussion identifies modifications presently under evaluation by NPPD. Additional analyses to complete the design for these modifications is under way and will be finalized following receipt of the NRC's approval of the enclosed Evaluation.

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1 . NEDC-31858P-A, August 1999, incorporates NEDC-31858P, Rev. 2, the associated NRC Safety Evaluation (including limitations), and related regulatory correspondence.

Modifications to the MSIV Leakage Pathway Configuration- The purpose of these modifications is to ensure that the MSIV leakage propagates to the Main Turbine Condenser in a manner assumed by the previously submitted design basis accident dose calculations. In Section 4.6.1 of the Evaluation, two groups of modifications have been identified:

1) Five manual isolation valves to be installed on Main Steam branch lines in order to limit the amount of piping to be credited for the MSIV leakage flowpath (and hence, maintained as seismically robust). Post-accident Operator action will be required to close these valves (which will be located in the Turbine Building).

2) To direct the MSIV leakage to the Main Turbine Condenser, two options are being pursued:

    - Install new Main Steam bypass piping isolated by two normally closed manual valves (requiring post-accident operator action to open), or
    - Modify the motive power of the existing Main Steam bypass valves to allow them to be opened in a post-accident condition.

    Crediting local operator action to open the proposed new Main Steam bypass piping valves meets the intent of Section 5.3 of Reference A-6 with respect to the reliability of the Alternate Leakage Treatment pathway. The final design decision will be communicated to the NRC in a future correspondence, as discussed in the cover letter to this submittal.

Outlier Resolution- Section 4.6.2 of the Evaluation itemizes the outliers resulting from the seismic analysis. Performing modifications to resolve these issues represents the most conservative engineering outcome. NPPD realistically expects to be able to perform more detailed analyses that will disposition many of these outliers as acceptable.

CONCLUSIONS

Based on discussions herein, it is concluded that the enclosed Evaluation meets the submittal requirements of License Condition 2.C.(6). Accordingly, NPPD requests that the NRC review and approve the Evaluation. NPPD recognizes that the final scope of the necessary modifications is subject to the design decisions and supplemental analyses previously described. Accordingly, as committed to in the cover letter, NPPD will submit a finalized description of those modifications that configure the MSIV leakage pathway following receipt of NRC approval of the enclosed Evaluation.

In the Safety Evaluation to License Amendment 183 (received on April 7, 2000), the NRC identified three issues that warranted either the use of compensatory measures or the limitation of that amendment to one fuel cycle in order for the NRC to find the design basis accident dose calculations acceptable[2]:

1. The first issue concerned the effects on the TID-14844 source term with respect to the LOCA and CRDA analyses given the higher burnup fuel design of GE-14 fuel. The proposed resolution of this issue was discussed with the NRC staff in a meeting held in NRC Headquarters on October 4, 2000, and revised calculations were submitted to the NRC to resolve this issue in a letter dated February 28, 2001.

2. The second issue involved the need for submitting a technically detailed analysis of the seismic adequacy of the Main Steam piping, Main Turbine Condenser, and Turbine Building following an SSE so that iodine plate-out in the Main Turbine Condenser could be credited. With the submittal of the enclosed Evaluation, NPPD has performed this action.

3. The third issue involved the consideration of fumigation conditions with respect to the elevated release path for the first 30 minutes following an accident. The proposed resolution of this issue was discussed with the NRC staff in a meeting held in NRC Headquarters on October 4, 2000, and revised calculations were submitted to the NRC in a letter dated February 28, 2001 which incorporated the 30 minute post-accident fumigation conditions.

With the resolution of these three outstanding issues, NPPD requests that the NRC resume their review of the LOCA, CRDA, and Main Steam Line Break Accident dose calculations (previously submitted in Reference A-9), in parallel with their review of the seismic evaluation. NPPD requests issuance of that License Amendment by January 30, 2003.

REFERENCES

A-1 Letter to NRC from J. Swailes (NPPD), dated December 22, 1999, "Design Basis Accident Radiological Assessment Calculational Methodology Revision."

A-2 Letter to J. Swailes (NPPD) from L. Burkhart (NRC), dated March 6, 2000, "Cooper Nuclear Station – Request For Additional Information (TAC NO. MA7758)."

A-3 Letter to the NRC from J. Swailes (NPPD), dated March 20, 2000, "Design Basis Accident Radiological Assessment Calculational Methodology – Response to Request for Additional Information."

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2. In License Amendment 187, dated October 23, 2001, the NRC extended their interim approval of the LOCA, CRDA, and Main Steam Line Break Accident dose calculations for one additional cycle. The NRC also approved the dose calculation for the Fuel Handling Accident, which had been deferred in License Amendment 183.

A-4 Letter to the NRC from J. Swailes (NPPD), dated March 24, 2000, "Design Basis Accident Radiological Assessment Calculational Methodology – Response to Request For Additional Information (Question #6)."

A-5 Letter to J. Swailes (NPPD) from L. Burkhart (NRC), dated April 7, 2000, "Cooper Nuclear Station – Issuance of Amendment on Design Basis Accident Radiological Assessment Calculational Methodology Revision (TAC NO. MA7758)."

A-6 Letter to T. Green (GE) from F. Akstulewicz (NRC), dated March 3, 1999, "Safety Evaluation of GE Topical Report, NEDC-31858P, Revision 2, 'BWROG Report For Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems,' September 1993."

A-7 Letter to NRC from J. Swailes (NPPD), dated March 29, 2000, "Design Basis Accident Radiological Assessment Calculational Methodology – Supplemental Seismic Information."

A-8 Letter to NRC from J. Swailes (NPPD), dated September 4, 2001, "Design Basis Accident Radiological Assessment Calculation Methodology – Supplemental Information, Main Condenser Seismic Evaluation."

A-9 Letter to NRC from J. Swailes (NPPD), dated February 28, 2001, "Proposed License Amendment Related to the Design Basis Accident Radiological Assessment Calculational Methodology.

ENCLOSURE 1

Engineering Evaluation 01-147, "Summary of Main Steam Isolation Valve (MSIV) Leakage Pathway to the Condenser Seismic Qualification"

w/Attachment 9.1- Drawing CNS-MS-43, Rev. 1, "Leakage Paths from Outboard MSIVs Cooper Nuclear Station"
w/Attachment 9.2- Selected Figures and Tables from Attachment 10.4

w/o Attachments 10.1, 10.2, 10.3, 10.4

ATTACHMENT 1 ENGINEERING EVALUATION COVER SHEET

ENGINEERING EVALUATION

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SECTION I, BACKGROUND INFORMATION

TITLE: Summary of Main Steam Isolation Valve (MSIV) Leakage Pathway to the Condenser Seismic Qualification

SOURCE DOCUMENT(S): SAP Network ID No.: 6003941
(e.g., Notification, Work Order, DWG, etc.)

PLANT SYSTEM(S): MS, RF, BLDG

PLANT COMPONENT(S): ______ ID No.: ______

EE TYPE: [ ] Use-As-Is [X] Assessment [ ] Facility Modification [ ] Configuration Control

[ ] Other: ______

SECTION II, EVALUATION:

Document and attach responses to the following items:

a. Issue Description: (Describe in sufficient detail to explain the need for the evaluation)

b. Applicable Design Basis: (Inputs, assumptions, industrial design function, nuclear safety function, key features)

c. Objective: (Concise statement which defines the purpose and scope)

d. Design Evaluation: (Impact on the plant design basis and assurance that the basis has not been adversely affected)

e. Conclusions: (Technically supported and meets the stated objective)

f. References: (Those required to support the evaluation)

g. Recommendations: (Additional or planned recommended actions and reference to the "stand-alone" controlling document or Notification Number)

III: PREPARATION / APPROVAL SIGNATURES SECTION

Prepared By: Perry K. Adelung Perry K. Adelung Date: 2-4-02

Independent Design Verified By: Ronald L. Yantz Ronald L. Yantz Date: 2/4/02

Supervisor: ALI BACHA Date: 02/05/2002

SORC Review required? [X] YES; [ ] NO Bmk 2/21/02

SORC Meeting Number: ~~S2002-011~~ S2002-013 Date: Bmk 2/12/02

ENGINEERING EVALUATION

REVISION SUMMARY SHEET

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Reason for Change: ____________________

Page(s) Affected	Section(s) Affected	Description of Change

Prepared By: ____________________ Date: __________

Independent Design Verified By: ____________________ Date: __________

Supervisor: ____________________ Date: __________

SORC Review Required? [ ] YES; [ ] NO

SORC Meeting Number: __________ Date: __________

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Item No.	Design Inputs	Rev. No.	Pending Changes to Design Inputs
1	NEDC 92-143, "Turbine Building Main Steam Pipe Analysis"	1	None
2	NEDC 92-151, "Code Qualification of Pipe Supports for the Turbine Building Main Steam System"	2	None
3	NEDC 2000-026, "Evaluation of the Seismic Capability of the Condenser Anchorage for Safe Shutdown Earthquake Loading"	2	Yes
4	NEDC 2000-027, "Evaluation of the Seismic Capability of the Turbine Building for Safe Shutdown Earthquake Loading"	0	Yes
5	NEDC 00-028, "Structural Integrity Walkdowns of the MSIV Leakage Paths to the Condenser"	0	None
6	NEDC 00-029, "Post-LOCA MSIV Leakage Path to Main Condenser"	1	Yes
7	NEDC 00-029A, "Post-LOCA MSIV Leakage Evaluation"	0	Yes
8	NEDC 00-078, "Review of Stevenson and Associates Report No. 00C4152-R01, Assessment of the Seismic Capacity of the MSIV Leakage Pathway for the Cooper Nuclear Station"	0	None
9	NEDC 01-006, "MSIV Leakage Pathway Piping System Walkdown and Evaluation Packages"	0	None
10	NEDC 01-007, "Seismic Qualification of Main Steam Drain Lines"	0	None
11	NEDC 01-008, "Evaluation of the Seismic Capability of the MSIV Equipment for Safe Shutdown Earthquake Loading"	0	None

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Item No.	Affected Documents	Rev. No.	Change Required	Action Item Tracking Number (If change is required)
1	NEDC 2000-026, "Evaluation of the Seismic Capability of the Condenser Anchorage for Safe Shutdown Earthquake Loading"	2	Revise to Status 1	4224752
2	NEDC 2000-027, "Evaluation of the Seismic Capability of the Turbine Building for Safe Shutdown Earthquake Loading"	0	Revise to Status 1	4224752
3	NEDC 00-029, "Post-LOCA MSIV Leakage Path to Main Condenser"	1	Revise to Status 1	4224752
4	NEDC 00-029A, "Post-LOCA MSIV Leakage Evaluation"	0	Revise to Status 1	4224752
5				
6				
7				
8				
9				

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1 ISSUE DESCRIPTION

By letter dated December 22, 1999, Nebraska Public Power District (NPPD) submitted revised design basis accident radiological assessment calculational methodology for NRC review and approval. The NRC subsequently provided a request for additional information (RAI) to NPPD. NPPD's response to those RAIs was later submitted in a letter dated March 20, 2000.

NPPD committed to perform additional evaluations in response to RAI Question 6. Question 6 requested justification for crediting iodine removal in the Class II main turbine condenser. In a follow-up response letter to the NRC dated March 24, 2000, NPPD stated that *"While the District believes that crediting iodine removal in the existing main turbine condenser design is already a part of the Cooper Nuclear Station (CNS) licensing basis for radiological assessment calculation accident mitigation, the District committed to provide a description of the structural robustness of the existing main steam line piping from the main steam isolation valves (MSIVs) to the main turbine condenser and the main turbine condenser."* This statement was based on section 6.2.2 of the original plant license Safety Evaluation Report (OL-SER) where the NRC provided its original evaluation and acceptance of the basis for MSIV leakage considerations following a postulated LOCA. The OL-SER states: *"Limitations of the dose from the leakage through the closed main steamline isolation valves (MSLIV) following a postulated LOCA presently relies on the low leakage characteristic of the valves. The MSLIV leakage was not identified as a concern at the time of the construction permit review for the Cooper Nuclear Station. On the basis of the staff's calculations, we have concluded that the leakage of fission products from the containment after a LOCA through the MSLIV at the previously accepted Technical Specification leakage of 11.5 scf/hr from each of the valves is acceptable provided there is not a concurrent failure to the steamlines outside of the containment or to the turbine-condenser [these are not designed to Category I (seismic) criteria on the CNS]."*

Furthermore, NPPD's response to FSAR Question 12.42, *"Evaluate the capability of the main steam line and all branch lines connected to it over 2-1/2" up to and including the turbine stop valve to withstand the design basis earthquake. Identify the modifications, if any, which would be required to render such lines capable of satisfying seismic Class I criteria."*, explicitly stated that the Turbine Building and the piping in it are Class II structures for which the seismic load for such structures does not include the effects of the maximum possible earthquake (DBE).

NPPD previously provided the above noted follow-up response document, dated March 24, 2000, which was supplemented by letter dated March 29, 2000, for crediting iodine plateout in the condenser to support proposed revisions to the dose calculations for the postulated Loss of Coolant Accident (LOCA). These documents summarized the seismic and structural design of the Cooper Nuclear Station (CNS) main turbine condenser, the MSIV leakage pathway (piping) to the main turbine condenser, and the Turbine Building (TB) structure. This information established the interim operability of these Class II components and structure following a postulated Safe Shutdown Earthquake (SSE).

The NRC subsequently approved a **license condition** (Amendment 183) to NPPD's operating license that states:

"No later than 8 weeks after the Cooper Nuclear Station (CNS) Cycle 21 startup, the licensee shall submit a request for the staff to review and approve a seismic evaluation to ensure the structural integrity of the main steam line piping from the main steam isolation valves (MSIV) to the main turbine condenser, the main turbine condenser, and the turbine building. The evaluation will be performed to assess the ability of the aforementioned main steam piping and main turbine condenser to remain sufficiently intact to direct main steam leakage from the MSIVs to the main turbine condenser, consistent with the leakage assumptions in the design-basis accident dose calculations during and after a Safe Shutdown Earthquake. This seismic evaluation will employ an analytical methodology acceptable to the staff and will identify any modifications necessary to support the evaluation. The licensee's approved request shall be fully implemented, including the completion of modifications, within 12 months of approval or prior to CNS Cycle 22 startup, whichever is later."

In the NRC's Safety Evaluation (SE) for License Amendment 183, the NRC also stated:

"The staff believes that justification of the capability to direct MSIV leakage from the MSIVs to the main turbine condenser is necessary for crediting dose consequence mitigation by iodine plateout on the condenser. An acceptable method for providing full qualification could follow the pertinent guidelines contained in the staff's safety evaluation dated March 3, 1999, "Safety Evaluation of GE Topical Report, NEDC-31858P, Revision 2, 'BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems,' September 1993." The staff recognizes that the technical nature of this evaluation and the identification of plant modifications that may be necessary to support this evaluation may require significant NPPD resources. The staff believes that this justification is necessary to support the long-term acceptability of the main turbine condenser and MSIV leakage pathway (piping) to perform the dose consequence mitigation function. In a letter dated March 24, 2000, NPPD committed to provide this evaluation in a timely manner. The staff believes that this commitment is of such importance to safety that no change should be made without prior staff approval in accordance with 10 CFR 50.90. Consequently, License No. DPR-46 is amended with additional condition 2.(C).(6) (See section 3.5). Absent this long-term evaluation, and using engineering judgement (supported by simplified calculations) along with the guidance of Generic Letter 91-18, NPPD, by letter dated March 24, 2000, submitted sufficient information to demonstrate the operability of the main steam line piping exiting from the MSIVs to the main turbine condenser, the main turbine condenser, and the TB in the event of an SSE...".

The **purpose of this EE** is to:

1) provide a summary of the activities completed to evaluate the MSIV leakage pathway to the condenser seismic qualification in order for NPPD to meet the license condition applicable to this seismic qualification;
2) document the acceptance of Stevenson and Associates report AR-001, Rev. 0, "Seismic Evaluation of MSIV Leakage Pathway at Cooper Nuclear Station", and;

3) provide design control authorization for all calculations prepared to support the seismic qualification evaluations.

2 APPLICABLE DESIGN BASIS

2.1 Load Combination Assumptions

2.1.1 A Safe Shutdown Earthquake can occur during and after the postulated LOCA (reference section 2.7 of the Plant Unique Analysis Report (PUAR) and the BWROG Report).

2.1.2 Maximum allowable leakage will occur past the MSIVs following a postulated Loss Of Coolant Accident (LOCA).

2.2 Safety Design Basis

The Safety Design Basis of the MSIV leakage pathway to the condenser will be to direct MSIV leakage from the MSIVs to the main turbine condenser. This allows crediting the dose consequence mitigation assumptions related to leakage holdup and the resulting iodine plateout within the condenser following a postulated LOCA. The pathway, and its associated components and equipment, must also remain sufficiently intact to direct main steam leakage from the MSIVs to the main turbine condenser, consistent with the leakage assumptions in the design basis accident dose calculations, during and after a postulated Safe Shutdown Earthquake (SSE).

2.3 CNS License Condition

The license condition (Amendment 183) to NPPD's operating license states that:

"No later than 8 weeks after the Cooper Nuclear Station (CNS) Cycle 21 startup, the licensee shall submit a request for the staff to review and approve a seismic evaluation to ensure the structural integrity of the main steam line piping from the main steam isolation valves (MSIV) to the main turbine condenser, the main turbine condenser, and the turbine building. The evaluation will be performed to assess the ability of the aforementioned main steam piping and main turbine condenser to remain sufficiently intact to direct main steam leakage from the MSIVs to the main turbine condenser, consistent with the leakage assumptions in the design-basis accident dose calculations during and after a Safe Shutdown Earthquake. This seismic evaluation will employ an analytical methodology acceptable to the staff and will identify any modifications necessary to support the evaluation. The licensee's approved request shall be fully implemented, including the completion of modifications, within 12 months of approval or prior to CNS Cycle 22 startup, whichever is later."

2.4 Design Basis Requirements and Specifications

The proposed Design Basis requirements, specifications, and methodology for the MSIV leakage pathway to the condenser seismic qualification are established and authorized herein by this EE. The information in this EE will be submitted to the NRC for review and approval as required by the License Condition.

3 OBJECTIVE

The **objective and purpose of this EE** is to:

3.1 provide a summary of the activities completed to evaluate the MSIV leakage pathway to the condenser seismic qualification in order for NPPD to meet the license condition applicable to this seismic qualification;

3.2 document the acceptance of Stevenson and Associates report AR-001, Rev. 0, "Seismic Evaluation of MSIV Leakage Pathway at Cooper Nuclear Station", and;

3.3 provide design control authorization for all calculations listed below which were prepared to support the seismic qualification evaluations:

3.3.1 NEDC 92-143, Rev. 1, "Turbine Building Main Steam Pipe Analysis"
3.3.2 NEDC 92-151, Rev. 2, "Code Qualification of Pipe Supports for the Turbine Building Main Steam System"
3.3.3 NEDC 2000-026, Rev. 2, "Evaluation of the Seismic Capability of the Condenser Anchorage for Safe Shutdown Earthquake Loading"
3.3.4 NEDC 2000-027, Rev. 0, "Evaluation of the Seismic Capability of the Turbine Building for Safe Shutdown Earthquake Loading"
3.3.5 NEDC 00-028, Rev. 0, "Structural Integrity Walkdowns of the MSIV Leakage Paths to the Condenser"
3.3.6 NEDC 00-029, Rev. 1, "Post-LOCA MSIV Leakage Path to Main Condenser"
3.3.7 NEDC 00-029A, Rev. 0, "Post-LOCA MSIV Leakage Evaluation"
3.3.8 NEDC 00-078, Rev. 0, "Review of Stevenson and Associates Report No. 00C4152-R01, Assessment of the Seismic Capacity of the MSIV Leakage Pathway for the Cooper Nuclear Station"
3.3.9 NEDC 01-006, Rev. 0, "MSIV Leakage Pathway Piping System Walkdown and Evaluation Packages"
3.3.10 NEDC 01-007, Rev. 0, "Seismic Qualification of Main Steam Drain Lines"
3.3.11 NEDC 01-008, Rev. 0, "Evaluation of the Seismic Capability of the MSIV Equipment for Safe Shutdown Earthquake Loading"

4 DESIGN EVALUATION

Evaluation of the Structural Integrity of the Main Steam Isolation Valve Leakage Pathway to the Condenser Following a Design Basis Earthquake

4.1 Approach

When crediting dose consequence mitigation due to iodine plateout in the main turbine condenser, NPPD has evaluated the ability of the piping pathway, main turbine condenser, and the Turbine Building (TB) to remain structurally intact following a Safe Shutdown Earthquake (SSE). Full qualification for the purpose of dose consequence mitigation has been demonstrated by satisfactory completion of a technically detailed seismic evaluation of the ability of these SSCs to maintain sufficient structural integrity during and after an SSE.

An acceptable method, previously approved by the NRC, for providing full qualification is to follow the pertinent guidelines and limitations contained in the NRC staff's safety evaluation dated March 3, 1999, "Safety Evaluation of GE Topical Report, NEDC-31858P, Revision 2, 'BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems,' September 1993." NPPD has addressed the limitations itemized in section 6.0 of the SE as follows:

1. A detailed description of the drain path and the basis for its functional reliability are provided in NEDC 00-029 and NEDC 00-029A (references 6.10 and 6.11). The general description of the maintenance and testing program for the active components of the drain path is described in section 4.7 of this EE.
2. CNS-specific information for piping design parameters are summarized in section 4.5.4.4 of this EE to demonstrate that they are enveloped by those associated with the earthquake experience database.
3. NPPD has completed a detailed comparison of the CNS condenser with the earthquake experience database under NEDC 2000-026 (reference 6.7) to demonstrate that the CNS condenser design falls within the bounds of design characteristics found in the database. NEDC 2000-026 also includes an analysis of the CNS condenser anchorage to verify and demonstrate that the condenser has adequate anchorage. This information is summarized in section 4.5.3 of the EE.
4. CNS-specific seismic evaluations for pipe and equipment supports and anchorage associated with affected piping and the condenser have been completed and are summarized in section 4.5 of this EE.
5. NPPD has confirmed that the CNS condenser will not fail due to seismic II/I type of interaction. Details are provided in section 4.5 of this EE.
6. NPPD has performed both detailed analyses and "bounding"/limited analytical reviews of piping and has discussed the basis for selecting the piping as described in section 4.5.4 of the EE.
7. NPPD has used a methodology and criteria for the analytical evaluations which are either in compliance with the existing CNS design basis methodology and criteria or have been previously approved by the NRC for other plants (see section 4.5.4 of the EE).

8. The facility ground motion estimates shown in the SE were used to verify the seismic adequacy of equipment at CNS as described in section 4.3 of the EE.
9. NPPD has used a sufficient quantity of data that was referenced in the experience database for concluding the acceptability of the BWROG methodology for evaluating the applicable equipment at CNS.

Detailed analyses have been performed by NPPD to meet the requirements of the BWROG Report or to identify "outliers" that are required to be resolved for full qualification. "Outliers" are SSCs that do not meet all criteria for full qualification as specified in "Safety Evaluation of GE Topical Report, NEDC-31858P, Revision 2, 'BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems,' September 1993." These "outliers" will be resolved through modifications to the plant and/or further detailed analyses. The "outliers" are itemized in section 4.6 below.

In the interim, NPPD will continue to credit iodine plateout in the main turbine condenser as previously accepted by the NRC. The NRC previously acknowledged that CNS would require modifications to meet the new acceptance criteria of the BWROG report by stating that "*This seismic evaluation will employ an analytical methodology acceptable to the staff and will identify any modifications necessary to support the evaluation*".

4.2 MSIV Leakage Pathway Identification

NPPD has identified, documented, and evaluated the boundaries of the MSIV leakage pathway to the condenser in calculations NEDC 00-029 and NEDC 00-029A to ensure compliance with the applicable guidelines of the "Safety Evaluation [SE] of GE Topical Report, NEDC-31858P, Revision 2, "BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems", September 1993." The MSIV leakage pathway boundaries for CNS are shown on Attachment 9.1.

In order to limit the scope of evaluations, limit modifications to existing piping, and to reduce the scope of future required maintenance/surveillance activities, and yet still fully meet all the applicable guidelines of the SE for the BWROG Report, NPPD has elected to modify the existing leakage flow path boundaries and/or change positions of specific flow path valves following a LOCA.

The proposed modifications under consideration are: 1) the addition of new isolation valves on various existing leakage path branches; and, 2) the addition of a new Main Steam bypass piping system or changes in the motive power source of the existing Main Steam bypass valves. These changes are proposed to establish a more direct leakage pathway from the MSIVs to the condenser than that which is currently provided. The proposed valves and additional piping will be installed in the future. Post-accident operator action will be required for some of the system isolations. The simplified system flow diagram showing the proposed piping and valves is included as Attachment 9.1.

4.3 Earthquake Ground Motion

This section contains a review of the earthquake data to assure that the vibratory ground motion, experienced at each of the facilities with equipment being used as a surrogate for similar equipment at Cooper Nuclear Station, did indeed exceed the Cooper Nuclear Station SSE. The ideal case, for this type of comparison, is to have actual recordings of the earthquake ground motion made at each of the facilities. This evaluation relies on the ground motion estimates in the experience data base from actual instrument recordings at or near the facility sites to verify the adequacy of the MSIV leakage path piping and equipment.

The ground motion from an earthquake at a particular site is a function of the earthquake source characteristics such as the magnitude, focal mechanism, radiation pattern, stress drop, location of asperities and fault rupture history, and depth and orientation of the fault. It is also a function of the distance of the facility to the fault and the propagation properties of the ground material between them. The geology immediately under the facility site can also have a large effect on the amplitude and frequency content of the ground motion. Two of the more appropriate methods of estimating earthquake ground motion where there are no nearby recordings involve the use of (1) calibrated numerical modeling of the fault rupture and wave propagation process, and (2) empirical attenuation relationships obtained from the statistical analysis of large sets of earthquake data.

Per Section II-5.2.3 of the CNS USAR [7.6], the horizontal safe shutdown earthquake (SSE) for CNS is based on the N69W component of the 1952 Kern County earthquake recorded at Taft, California, scaled to a peak ground acceleration (PGA) of 0.20g. The SSE ground response spectrum is shown in Figure 4.1 of Attachments 9.2 and 10.4.

The earthquake experience data that is directly being used for comparison to the Cooper Nuclear Station piping is obtained from the following site-earthquake pairs. This data is taken from Reference [7.11].

- El Centro Steam Plant - Imperial Valley 1979 earthquake.
- Valley Steam Plant - San Fernando 1971 earthquake.
- Moss Landing Power Plant - Loma Prieta 1989 earthquake.
- Humboldt Bay Power Plant - Ferndale 1975 earthquake.

For other equipment included in the scope of the leakage path review, the Bounding Spectrum in the SQUG GIP-2 is used to verify seismic adequacy. Figure 4.2 of Attachments 9.2 and 10.4 contains a comparison of each of these records and the Cooper Nuclear Station SSE. The following paragraphs discuss each of these earthquakes and make a comparison to the Cooper Nuclear Station SSE.

The ground motion estimate at the El Centro Steam Plant from the Imperial Valley 1979 earthquake was based on a recording made at a U.S. Geological Survey (USGS) strong ground motion station about 1 kilometer from the facility. Because of the density of seismic recordings in that area and the distribution of the ground motion it can be concluded that the estimate for the site is significantly larger than the Cooper Nuclear Station SSE and is appropriate for use in verifying the seismic adequacy of the Cooper Nuclear Station equipment, in the MSIV leakage pathway, similar to that in the El Centro Steam Plant.

The ground motion estimate developed by EQE [Earthquake Engineering], Inc. at the Valley Steam Plant from the San Fernando 1971 earthquake, for use in the USI A-46 program, was based on an extrapolation of data from a relatively distant location. In 1988, the USGS performed studies to estimate the ground motion at selected sites from the San Fernando 1971 earthquake in support of the USNRC's resolution of USI A-46. The USGS estimate of the ground motion at the Valley Steam Plant is lower than the EQE, Inc. estimate.

As in the resolution of the A-46 ground motion issue, the USNRC still considers the USGS estimate to be the characterization of the ground motion at the Valley Steam Plant from the San Fernando 1971 earthquake. This estimate is appropriate for use in verifying the seismic adequacy of the Cooper Nuclear Station equipment, in the MSIV leakage pathway, similar to that in the Valley Steam Plant.

The ground motion estimate at the Moss Landing Steam Plant from the Loma Prieta 1989 earthquake is based on a study performed by Pacific Gas and Electric Company (PG&E) the owner of the Moss Landing Steam Plant at the time of the earthquake. A copy of the report of the study was provided by PG&E to the USNRC. The analysis performed by PG&E was technically sound and comprehensive. It shows a thorough understanding of the problem and it can be concluded that their estimate of the ground motion is appropriate for use in verifying the seismic adequacy of the Cooper Nuclear Station equipment, in the MSIV leakage pathway, similar to that in the Moss Landing Steam Plant.

The strong ground motion recordings and response spectra at Humboldt Bay from the Ferndale 1975 earthquake are available and accepted by the USNRC. The ground elevation vibratory motions experienced at the plant were all larger than the Cooper Nuclear Station SSE ground motions. Therefore, the ground motion at the Humboldt Bay Plant from the Ferndale 1975 earthquake is appropriate for use in verifying the seismic adequacy of the Cooper Nuclear Station equipment, in the MSIV leakage pathway, similar to that in the Humboldt Bay Plant.

In 1988 as part of the USNRC review of the SQUG earthquake data base for use in the resolution of USI A-46, the USNRC asked the USGS to make independent ground motion estimates for the data base earthquake facility pairs. In general, depending on the site and the frequency, the USGS estimates exceeded or were less than the SQUG estimates. The average of the USGS estimated ground motion was compared to the GIP-2 Reference Spectrum and the GIP-2 spectrum was found to exceed the USGS estimates at all frequencies. On this basis, the USNRC concluded that the GIP-2 Reference Spectrum and the GIP-2 Bounding Spectrum (lower bound of the SQUG data base spectra) were acceptable for verifying the seismic adequacy of the equipment in the SQUG data base facilities. The GIP-2 Bounding Spectrum is significantly higher than the Cooper Nuclear Station SSE ground motion spectrum. Therefore it is appropriate to use the GIP-2 Bounding Spectrum in verifying the seismic adequacy of the Cooper Nuclear Station equipment, in the MSIV leakage pathway, similar to that in the SQUG GIP-2 data base.

Based on the analysis of the earthquake experience database as discussed in References [7.1] and [7.11], the Cooper Nuclear Station SSE ground demand motion is below the seismic ground motion which was experienced at all the facilities discussed above. Consequently, the use of the BWROG's and SQUG-GIP database to verify the seismic adequacy of the equipment and piping in the MSIV leakage pathway is acceptable for the Cooper Nuclear Station.

4.4 Cooper Nuclear Station (CNS) Seismic Demand

4.4.1 Safe Shutdown Earthquake (SSE)

Per Section II-5.2.3 of the CNS USAR [7.6], the horizontal safe shutdown earthquake (SSE) for CNS is based on the N69W component of the 1952 Kern County earthquake recorded at Taft, California, scaled to a peak ground acceleration (PGA) of 0.20g. The SSE ground response spectrum is shown in Figure 4.1 of Attachments 9.2 and 10.4.

4.4.2 Location of Equipment and Piping

The majority of the piping and equipment in the scope of the MSIV leakage pathway evaluation at Cooper Nuclear Station (CNS) is in the Turbine Building (TB); the balance is in the Reactor Building's (RB) steam tunnel and torus compartment. The relevant building elevations are as follows (all elevations are "top of slab"):

Bldg. / Elev.	Description
TB 877'-6"	Turbine Building foundation mat (condenser support elevation)
TB 882'-6"	Turbine Building basement
TB 903'-6"	Turbine Building mezzanine
TB 909'-6"	Turbine Building heater bay
TB 932'-6"	Turbine Building operating deck
RB 859'-9"	Reactor Building foundation mat (Torus support elevation)
RB 902'-6"	Steam Tunnel floor / Torus compartment roof
RB 916'-6"	Steam Tunnel upper level
RB 932'-6"	Steam Tunnel roof

4.4.3 Turbine Building Seismic Demand

Per Section XII-2.1 of the USAR, the Turbine Building is a Class II structure. As such, floor response spectra are not available. An estimate of the realistic, median-centered floor response spectra is made based on the guidance provided in the GIP [7.2] and realistic, median-centered floor response spectra that have been calculated for the Control Building and the Reactor Building in Reference [7.12].

During construction, the site was excavated to rock (elevation 820') and a controlled, compacted fill was placed up to the elevation of the building foundations. Once the buildings were constructed, a compacted fill was placed up to plant grade (903'). The

Turbine Building is founded on a 8' thick reinforced concrete mat. The top of the mat is at 877'-6". The basement floor slab, which is a 6" thick reinforced concrete slab supported on concrete walls and a sand fill, is at 882'-6". The lower part of the building is a reinforced concrete shear wall structure from the base mat up to the operating deck at 932'-6". Above the operating deck, a steel superstructure supports the crane (980') and the roof (1010').

The GIP specifies 1.5× SSE ground response spectra as an estimate of the realistic, median-centered floor response spectrum for elevations within about 40' above grade. The grade elevation has been a point of discussion for the GIP plants. For the CNS Turbine Building, the most liberal interpretation is yard grade (903'), and the most conservative interpretation is the top of the base mat (877'-6"). If the former is used, the highest elevation of interest (932'-6") is 29'-6" above grade; if the latter is used, it is 60'-0" above grade.

Reference [7.12] contains realistic, median-centered floor response spectra for the Control Building and the Reactor Building. These spectra were calculated for use in the IPEEE program (Individual Plant Examination for External Events). As specified in NUREG-1407 [7.8], a NUREG/CR-0098 [7.9] median soil spectrum anchored at 0.3g, rather than the CNS SSE ground response spectrum, was used for to develop these floor spectra. The IPEEE ground spectrum is referred to as the IPEEE Review Level Earthquake (RLE). Section 4.2.4 of the GIP states that realistic, median-centered floor spectra based on the IPEEE RLE can be scaled to obtain realistic, median-centered floor spectra based on the site SSE. The scale factor is the maximum (most conservative) ratio of the Reg. Guide 1.60 ground spectrum anchored to the site SSE peak ground acceleration to the IPEEE RLE. Figure 4.4 of Attachments 9.2 and 10.4 shows the IPEEE RLE, CNS SSE, and Reg. Guide 1.60 spectra. The peak of the IPEEE RLE is essentially equal to the peak of the Reg. Guide 1.60 spectrum anchored to the site SSE peak ground acceleration of 0.2g. Therefore the IPEEE RLE floor response spectra can be used directly (scale factor = 1) as the realistic, median-centered floor response spectra for the site SSE.

Figure 4.5 of Attachments 9.2 and 10.4 shows the IPEEE RLE floor response spectrum for elevation 932'-6" in the Control Building. For comparison, the IPEEE RLE ground response spectrum is also shown. Superimposed on the plot are 1.0×, 1.5×, and 2.0× the SSE ground response spectrum. Figure 4.6 of Attachments 9.2 and 10.4 shows the same spectra for the Reactor Building: the IPEEE RLE floor spectra for elevation 932'-6", the IPEEE RLE ground response spectrum, and 1.0×, 1.5×, and 2.0× the SSE ground response spectrum. Note that the IPEEE RLE floor response spectrum for 932'-6" is higher in the Control Building than in the Reactor Building. Reference [7.12] employed modern soil-structure interaction (SSI) analysis techniques, so the depth of a structure's embedment in the surrounding soil is an important factor. The Reactor Building basemat is at 859'-9", while the Control Building basemat is at 877'-6", the same elevation as the Turbine Building basemat.

Figures 4.5 and 4.6 of Attachments 9.2 and 10.4 show that the 2.0× SSE ground response spectrum envelops the floor response spectra at elevation 932'-6" in both the Control Building and the Reactor Building. Based on this result, 2.0× SSE is used as the seismic demand for all elevations in the Turbine Building up to and including 932'-6".

4.4.4 Reactor Building Seismic Demand

Using the same reasoning presented above for the Turbine Building, 2.0× SSE ground response spectrum is used as the seismic demand for all elevations in the Reactor Building up to and including 932'-6". Based on Figure 4.6 of Attachments 9.2 and 10.4, an argument could be made for using a lower value such as 1.5× SSE, but 2.0× SSE is used to maintain uniformity with the Turbine Building.

4.4.5 Piping and Pipe Support Capacity Criteria

In addition to the ground motion comparisons discussed in Section 4.3, References [7.1], [7.5], [7.13], [7.14], and [7.23] document damage surveys of piping at a number of industrial facilities subjected to strong motion earthquakes.

These references provide the results of an extensive survey of piping systems subjected to strong motion earthquakes. This survey includes data on the performance of over 1,000,000 feet of piping at 34 power facilities. All the reference studies indicate a very low piping failure rate (<1%), and conclude that the failures which did occur were due to isolated local weaknesses in piping systems which could be screened by an in-plant walk-down. The piping investigated in this survey experienced seismic ground accelerations, and exhibited response to those inputs, consistent with the use of the GIP Bounding Spectrum as a capacity. Therefore, Reference [7.13] then established the "GIP Bounding Spectrum" as representing the seismic capacity of welded steel piping supported at elevations up to 40' above grade. In addition, for welded steel piping supported at elevations higher than 40' above grade, Reference [7.13] established the following capacity spectra:

1. If the seismic demand is based on realistic, median-centered amplified floor response spectra, the capacity spectrum is 1.5× the Bounding Spectrum.

2. If the seismic demand is based on conservative amplified floor response spectra, the capacity spectrum is 2.25× the Bounding Spectrum.

Reference [7.13] also established that the capacity spectrum for piping with non-welded joints (e.g., threaded pipe) the capacity spectrum is 0.67× the applicable capacity spectrum for welded steel pipe.

Table 4.1 Piping Demand/Capacity Summary

Capacity	Demand
For welded steel piping supported less than 40' above grade for which no amplified floor response spectra are available:	
Bounding Spectrum	SSE Ground Response Spectrum
For welded steel piping supported higher than 40' above grade, or supported at elevations less than 40' above grade for which amplified floor spectra are available:	
1.5× Bounding Spectrum	Realistic, Median-Centered Floor Response Spectrum
2.25× Bounding Spectrum	Conservative, Design floor Response Spectrum
For steel piping with non-welded joints:	
0.67× applicable welded steel pipe capacity	Same as welded steel pipe

Based on the previous discussions, the seismic demand, as a realistic, median-centered floor response spectrum, can be represented by 2.0× CNS SSE ground response spectrum for all piping in the scope of this effort. The corresponding capacity is 1.5× Bounding Spectrum. Figure 4.7 of Attachments 9.2 and 10.4 shows that the capacity exceeds the demand. Thus 2.0× the CNS SSE ground response spectrum will be used as the seismic demand for the evaluation of all piping systems in the scope of this program.

For the walkdown screening and evaluation, the vertical demand used was 2/3 of the horizontal demand for all piping in the scope of this evaluation. This approach for definition of the vertical seismic demand (response spectra) in the Turbine Building (when response spectra is not available) has been accepted by the USNRC for the seismic ruggedness evaluation of MSIV leakage pathway piping systems on other nuclear power plants. This results in a vertical spectrum of 1-1/3 times the CNS SSE ground spectrum. This definition of the vertical seismic demand when used in conjunction with the 2 times the ground spectrum in the horizontal direction results in a slightly conservative estimate of a realistic, median-centered seismic demand. The licensing basis vertical seismic demand (response spectra) as defined in the CNS USAR is 2/3 times the ground response spectrum for all locations and elevations in all Class I structures (including the Reactor and Control buildings). This is based on the assessment that there is no significant amplification of the vertical seismic input ground motion by the CNS Class I building structures (buildings are rigid in the vertical direction). The Turbine building from the foundation to the turbine deck is a concrete shear wall structure similar in construction to the Reactor and Control Buildings. It is also reasonable to assume there is no significant amplification of the vertical seismic input ground motion by the Turbine Building. Therefore, the definition of the vertical seismic demand used in the MSIV Leakage pathway walkdown and evaluation is conservative with respect to the CNS licensing basis for Class I structures. For the resolution of non-conforming issues (outliers) resulting from the walkdown and screening evaluation a vertical seismic demand of 2/3 the ground spectrum will be used.

The seismic demand for outlier resolution will be 2 times the ground spectrum in the horizontal direction and 2/3 the ground spectrum in the vertical direction all piping systems in the scope of this program.

The use of this spectrum in conjunction with the discussion provided in previous sections insures the seismic capacity of the piping and piping support systems at the CNS exceeds the demand of the earthquake experience data.

4.4.6 Equipment Capacity/Criteria

Equipment is evaluated per the requirements of the GIP [7.2]. From GIP Section II.4.2, the demand/capacity criteria is summarized below:

Table 4.2 Equipment Demand/Capacity Summary

Capacity	**Demand**
For equipment which: (1) is supported at elevations less than 40' above grade for which no amplified floor response spectra are available and (2) has a fundamental frequency greater than 8 Hz (the frequency requirement does not apply to in-line equipment such as valves):	
Bounding Spectrum	SSE Ground Response Spectrum
GERS (Generic Equipment Response Spectrum)	2.25× SSE Ground Response Spectrum
For equipment which is supported at elevations more than 40' above grade or at lower elevations for which amplified floor response spectra are available:	
1.5× Bounding Spectrum	Realistic, Median-Centered Floor Response Spectrum
1.5× Bounding Spectrum	Conservative, Design floor Response Spectrum
GERS	1.5× Realistic, Median-Centered Floor Response Spectrum
GERS	Conservative, Design Floor Response Spectrum

Based on the discussion in previous sections, the seismic demand, as a realistic, median-centered floor response spectrum, can be represented by 2.0× CNS SSE ground response spectrum for all equipment in the scope of this effort. The corresponding capacity is 1.5× Bounding Spectrum. Figure 4.7 of Attachments 9.2 and 10.4 shows that the capacity exceeds the demand.

4.4.7 Condenser Demand Spectra

The CNS condenser is located below grade at the lowest level of the Turbine Building (Elevation 877.5 ft.). The applied seismic demand was the SSE ground spectrum.

4.5 Seismic Capacity and Seismic Interactions

4.5.1 Seismic Verification Walkdowns

Detailed seismic verification walkdowns were performed by qualified "seismic capability engineers" as defined by the "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, Rev. 3". Data gathered during these walkdowns was used as input for seismic evaluations and analyses, and for identifying potential seismic interaction concerns.

4.5.2 Turbine Building

4.5.2.1 Turbine Building Description

The Turbine Building (TB) houses the main turbine condenser and a majority of the primary MSIV leakage pathway (piping) from the MSIVs to the main turbine condenser (some of the piping is located in the Class I Reactor Building steam tunnel). The TB base mat is reinforced concrete. The TB is a reinforced concrete structure up to the operating floor. Structural steel framing (superstructure) rises above the operating floor. The interior walls of the TB are reinforced concrete, with concrete block enclosing smaller areas. The TB was designed to the requirements for Class II structures, systems, and components (SSCs), including 100-mph wind loading and 0.1g Uniform Building Code (UBC) seismic loading.

4.5.2.2 Turbine Building Structure Evaluation

NPPD has performed an evaluation [Reference 6.8] of the TB concrete structure to confirm that it is capable of remaining structurally intact without gross structural failure following a postulated SSE. Samples of key TB substructures (e.g., walls, floor slabs, and columns) were evaluated for increased seismic loading resulting from a postulated SSE. The horizontal seismic acceleration input to the operating floor of the TB at elevation 932 ft-6 in. due to the TB response was assumed to be 0.3g based on a comparison with Class I structures (Reactor Building and Control Building). The evaluations show that the increase in design loadings from the original seismic Class II criteria to the postulated SSE condition do not result in stresses that exceed the allowable limits applicable to the SSE load case. Therefore, NPPD concludes that there is sufficient margin in the original design to ensure that the concrete portion of the TB structure will remain intact during and following an SSE. These results are based primarily on the fact that allowable stresses are increased for the SSE load case and, consequently, the increase in seismic loading is offset by the increase in allowable stresses.

4.5.3 Main Turbine Condenser

4.5.3.1 Main Turbine Condenser Description

The main turbine condenser is a twin-shell, horizontal tube unit, cooled by river water. There are two shell units of the condenser. The condenser shell units are massive structures, with 7/8-inch thick steel shell walls, that contain substantial internal bracing and are seismically rugged. The main turbine condenser is located beneath the low pressure cylinders of the main turbine. To accommodate thermal expansion, a rubber belt expansion joint is provided for each condenser neck.

Each of the two shell units of the main turbine condenser is approximately 40 ft × 30 ft × 48 ft high. The base of each condenser shell unit is rigidly mounted to the reinforced concrete TB base mat which is 26 feet below grade. The top of each unit is located approximately 22 feet above grade elevation. These units are self-supporting structures that do not require any external support from the TB structure at any point other than the base anchorage. The base anchorage includes bolts for tension restraint, a centrally located seismic shear key, and a thrust anchor for resisting operating loads. The two shell units are interconnected by a large, rounded edge, rectangular-shaped steel passageway approximately 8 ft long with cross-sectional dimensions of 14 ft-6 in. × 9 ft-6 in. This interconnection was originally field welded to the condenser shells.

4.5.3.2 Main Turbine Condenser Structure Evaluation

An evaluation of the seismic ruggedness of condensers and condenser anchorage for GE BWR plants is reported in reference 6.5. The configurations of the GE BWR condensers were compared to condensers in the earthquake experience data. Condensers in the earthquake experience data exhibited substantial seismic ruggedness even when they were not designed to resist earthquakes. Comparisons of condenser designs in GE BWR plants with those in the earthquake experience data revealed the GE plant designs are similar to those that exhibited good earthquake performance. The study concluded that a failure and significant breach of pressure boundary in the event of a design basis earthquake is highly unlikely and contrary to a large body of historical experience data. The conclusions of that study were verified by detailed comparison of the CNS condenser configuration to the earthquake experience data. In particular, detailed comparisons to the Moss Landing and Ormond Beach condensers were performed.

The seismic adequacy of the CNS condenser was verified by reference to experience data contained in reference 6.5 with specific comparison to the Moss Landing and Ormond Beach condensers. Per reference 6.5, these condensers are of similar configuration to CNS and experienced strong motion in excess of the CNS design basis earthquake without failure. In addition, the adequacy of the CNS specific condenser configuration was verified by an evaluation of the CNS condenser anchorage capacity.

The CNS condenser is located below grade at the lowest level of the Turbine Building (Elevation 877.5 ft.). The applied seismic demand was the CNS SSE ground spectrum.

The CNS condenser design data is similar to or bounded by data for the two experience data sites. The CNS SSE ground spectrum is enveloped by the Moss Landing spectrum. The Ormond Beach estimated PGA demand due to the February 21, 1973 Point Mugu earthquake was 0.20g, which is equivalent to the CNS PGA of 0.2g. The CNS condenser design data is also well represented by the data presented in reference 6.5, Appendix D, Table 4-3. The comparison verifies that the results of the reference 6.5 evaluation for structural integrity are applicable to the CNS condenser.

The comparison of condenser data and the anchorage capacity evaluations demonstrates that the conclusions presented in reference 6.5, Appendix D can be applied to the CNS condenser. That is, a failure and significant breach of the condenser pressure boundary in the event of a design basis earthquake is highly unlikely and contrary to the experience data.

4.5.3.3 Main Turbine Condenser Anchorage Evaluation

NPPD has performed a calculation to evaluate the seismic capability of the main turbine condenser anchorage for postulated SSE loading. The calculation determined that seismic loading up to approximately 0.6g horizontal acceleration can be postulated before any tension in the four perimeter anchorages of a condenser shell unit would be developed from a postulated seismic event. In addition, the calculation determined that the seismic anchorage in the center of each condenser shell unit is capable of resisting a horizontal acceleration up to approximately 1g when using stress allowables for the loading condition that includes the SSE load. The maximum expected horizontal acceleration for the postulated SSE would be less than 0.6g; therefore, the calculation concludes that the existing tension and shear anchorage details for the condenser shell units are adequate to ensure that the condenser units will remain intact for postulated SSE loading.

The main turbine condenser is a seismic Class II structure/component that was originally designed for lateral seismic forces resulting from a horizontal base shear of 0.1g (UBC provisions) in combination with design operating loads (e.g., shell design pressures of 20 psig and 30" Hg vacuum). Vertical seismic loading was not included in the original design; however, the previously mentioned calculation has concluded that the vertical seismic acceleration for a postulated SSE would not have a substantial effect on the condenser shell unit anchorage.

4.5.4 MSIV Leakage Pathway (Piping) to the Condenser

The evaluation of piping included the following steps:

(a) Walkdowns of the piping systems and associated supports which included identification of items judged to have inadequate seismic capacity, worst case pipe supports, and items requiring limited analytical reviews.
(b) A comparison of piping system demand versus experience-based capacity.
(c) Limited analytical reviews and pipe support evaluations for piping systems identified during the walkdowns.
(d) Generation of Piping System Seismic Screening Work Sheet (PSSSWS), a formal method of documenting the walkdown, the limited analytical reviews, the worst case support evaluations, and the final seismic capacity evaluation.

4.5.4.1 Comparison to the Experience Data

4.5.4.1.1 Piping Considerations

The first step was formal documentation of the demand capacity review. The leakage path piping was then compared to the piping in the experience data to insure the piping systems at the Cooper Nuclear Station are within the experience data and within the ANSI B31.1 Power Piping Code. Key parameters in the comparison include the following:

(a) Piping is fabricated and designed to B31.1, B31.3 or ASME BPVC Section III.
(b) Piping sizes and materials fabrication fall within experience data.
(c) Piping support vertical and lateral spans (lengths between vertical and lateral supports) fall within the experience data. This was accomplished by verifying that the normalized span ratio criteria given below were met. These normalized span ratio criteria were developed based on a review of the data in reference [7.1]. This normalized span ratio criteria was used as a method to conduct the screening of the piping systems during the walkdowns. It is consistent with the support span data provided in reference [7.1] and reference [7.23]. It was correlated in this manner to provide a mechanism to more concisely present a comparison of the CNS support span data to that in the experience data as given in reference [7.1] and [7.23].

   For Welded Steel Pipe:
   (1) Vertical Spans are less than (1.5) times the suggested B31.1 Deadweight Spans.
   (2) Horizontal Spans are less than six (6) times the suggested B31.1 Deadweight Spans.

   For Threaded Steel Pipe:
   (1) Vertical Spans are less than (1.5) times the suggested B31.1 Deadweight Spans.
   (2) Horizontal Spans are less than four (4) times the suggested B31.1 Deadweight Spans.

(d) Piping operating pressures and temperatures fall within the experience data.

(e) Piping does not exhibit known failure modes or areas of potential weakness.
(f) Piping support system is adequate, consistent with the piping systems in the experience data, and would be expected to exhibit a ductile failure mode.

A comparison to the experience data was performed for the CNS leakage path piping. For that comparison, materials, sizes, and span ratio data were compared to piping in the experience data to verify that the CNS piping is adequately represented in the experience data.

4.5.4.1.2 Equipment Considerations

In many instances, piping systems terminate at mechanical equipment such as pumps and tanks. There are three items of concern at these equipment piping interface locations.

(a) Anchorage of the equipment
(b) Nozzle loads applied to the equipment by the piping
(c) Equipment displacements applied to the piping system.

The walkdown procedure requires that a Seismic Review Team (SRT) address these concerns using the SQUG-GIP methodology to verify the adequacy of the attached and inline equipment.

4.5.4.2 Analysis and Local Evaluation of Piping and Pipe Supports

This section defines the capacity criteria that was used in the limited analytical reviews and analysis of piping systems and in the evaluation of worse case supports. The capacity criteria was a stress-based criteria, and the demand criteria is in terms of an applicable input seismic excitation level. For specific analytical reviews such as Rod Hanger Fatigue reviews a different Demand/Capacity criteria is used and if so was defined in the applicable analytical review package.

4.5.4.2.1 Piping

The majority of piping systems under review were originally designed to the 1967 B31.1, "Power Piping Code". The original design only considered loadings due to pressure, dead load, design mechanical loads, 0.1g seismic loads, and thermal loads. The original design capacity criteria for the piping can be summarized as follows:

(a) "The sum of the longitudinal stresses due to pressure, weight, and other sustained loads shall not exceed the allowable stress in the hot condition, S_h." [B31.1-1967 Section 103.3.2(d)]

(b) "The longitudinal pressure stress, S_{LP}, shall be calculated as $Pd_i^2/(D_o^2-d_i^2)$, where:

P = Design Pressure (psig)
D_o = Outside Diameter (in)
d_i = Inside Diameter (in)
[B31.1-1967 Section 103.3.2(d)]

(c) "The thermal expansion stress, S_E, (calculated per Section 119.6.4), shall be less than S_A, where S_A is calculated per section 102.3.2(c). Furthermore, when the sum of the longitudinal stresses, as calculated in (a) above, is less than S_h, the difference between S_h and this sum may be added to the S_A value for determining the allowable value of S_E [B31.1-1967 Section 102.3.2(c)].

(d) "The sum of the longitudinal stresses produced by internal pressure, live and dead loads and those produced by occasional loads such as the temporary supporting of extra weight may exceed the allowable stress values given in the allowable stress tables the amounts and duration of time given in Paragraph 102.2.4 [B31.1-1967, Section 102.3.3(a)] {This permits a 20% increase in the S_h value}.

Since the consideration of a Design Basis SSE event was not in the original design basis for the piping systems under review, the capacity criteria given below was established for use in piping system limited analytical reviews and detailed analyses:

$$P + .75i[(M_A/Z)] < 1.0\ S \qquad (5.1a)$$

$$i(M_C/Z) < S_A + \{S-(P+.75i[M_A/Z])\} \qquad (5.1b)$$

$$P + .75i[(M_A/Z) + (M_{BI}/Z)] < 2.4\ S \qquad (5.2)$$

$$i[(M_C/Z) + (M_{Bsam}/Z)] < 2.5\ S_A \qquad (5.3)$$

P = Pressure Loadings
M_A = Applied Moments Due to Deadweight Loadings
M_{BI} = Applied Moments due to SSE seismic Inertial Loadings

M_{Bsam}	=	½ the range of Applied SSE Moments due to Seismic Anchor Motion Loadings
M_C	=	Range of Applied Moments due to Thermal Expansion and Thermal Anchor Motions
Z	=	Piping Section Modulus
S	=	Allowable Primary Stress limit per the B31.1 Code
S_A	=	Allowable Expansion Stress range per B31.1 Code
i	=	Stress intensification factor as defined in the B31.1 Code

Equations 5.1a and 5.1b are the standard deadweight and thermal stress evaluation equations per the current B31.1 Power Piping Code. It is consistent with the original design basis for applied deadweight and thermal loadings.

The basis for the establishment of equation 5.2 is as follows:

(a) It is slightly more conservative than the capacity criteria used by the ASME Boiler and Pressure Vessel Code, Section III, Division 1, for Class 3 piping systems subjected to Level D or faulted loading conditions (1992 and earlier Editions). The SSE event is classified as a Level D or Faulted condition for the CNS Plant. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Class 3, capacity criteria is the criteria specified for Quality Class C piping in Regulatory Guideline 1.26. The systems under review were judged to fall within the classification criteria of Quality Class C piping systems, as put forth in Section C.2.d of Regulatory Guideline 1.26.

(b) S is the basic allowable material stress per the B31.1 Power piping Code, which is the lesser of 5/8 S_y (2/3 S_y in later code editions) or $S_u/4$. The majority of the piping under review is A-106B Carbon steel pipe, which has S=15000 psi, S_y=35000 psi and S_u=60000 psi. Therefore, Equation 5.2 limits the Pressure + Deadweight + Seismic Inertial Stresses to less than 1.03S_y which insures elastic behavior, i.e., no significant yield or inelastic behavior would be permitted to occur during or after a Design Basis SSE event. Furthermore, the 2.4 S limit is significantly below the 1.5 S_y to 2.0 S_y limit put forth by the USNRC in NUREG-1367 [7.10] for insuring piping system functional capability.

The basis for the establishment of equation 5.3 is as follows:

(a) The ASME Boiler and Pressure Vessel, Section III, Division 1, Class 3 capacity criteria requires that the sum of the longitudinal bending stresses due to thermal expansion, thermal anchor motions, and OBE seismic anchor motions be limited to S_A. The ASME Boiler and Pressure Vessel, Section III, Division 1, Class 3 capacity criteria for Level C (Emergency) and Level D (Faulted) Conditions explicitly excludes consideration of stresses resulting from SSE seismic anchor motions. The basis for this exclusion is, for Level C and D loadings, the Code only requires the consideration of primary stresses and does not require explicit consideration of secondary stresses. The code classifies stress from seismic anchor motions as secondary stresses (displacement limited stresses). For this program it was decided to explicitly consider SSE seismic anchor motions in conjunction with thermal anchor motions and thermal expansion stresses. This was done (1) to insure that the secondary stresses which could occur during a design basis SSE event would be bounded and (2) from the experience data, piping failures due to large seismic anchor motions is a known failure mode. The capacity criteria selected for this review was 2.5 S_A which is slightly more than twice the capacity criteria used for the Level B load case of OBE seismic anchor motions, thermal expansion stresses and thermal anchor motions.

(b) S_A for carbon steel pipe is approximately 1.5 S which is approximately 5/8 S_y (2/3 S_y in later editions). The majority of the piping is A-106B GR. B CS with S=15000 psi and S_y = 36000 psi 2.5 S_A = (2.5 × 1.5 × 15000) = 56250 and therefore, 2.5 S_A is approximately 1.6 S_y. The applied stresses are secondary; limiting the range of applied stress to less than 2 S_y insures that elastic shakedown will occur, no significant membrane stress rupture will occur, and the accumulated cyclic damage will be elastic. Therefore, given the limited number of cycles of strong motion in a Design Basis SSE (10 to 20 cycles) and that elastic cycling below the 2.0 S_y will occur, a fatigue failure due to the SAM's from one SSE would not occur. Therefore, the 1.6 S_y secondary stress range limit used is significantly less than the upper bound limit of 2S_y and with this limit no fatigue failures due to one SSE event would be anticipated.

For dynamic analysis of piping in both the Turbine Building and the Reactor Building, the horizontal seismic demand is the 5% damped 2.0x CNS SSE ground response spectrum shown in Figure 4.7 of Attachments 9.2 and 10.4. The vertical demand of 2/3 of the horizontal demand is conservatively used instead of 2/3 of the horizontal ground spectra as specified in the CNS USAR. CNS may revise this conservative vertical demand if necessary to eliminate modifications or additional analyses. The discussion establishing this spectrum as representing the realistic, median-centered floor response spectrum for all elevations of interest is provided in Section 4.4. The basis for using 5% damping is discussed in the following paragraph.

Recent criteria and studies including Regulatory Guideline 1.61 [7.18], the ASME Boiler and Pressure Vessel Code Section III, Division 1, Appendix N [7.19], and NUREG/CR-0098 [7.9] specify levels of damping for the SSE analysis of piping systems. In all the aforementioned documents, the basis of the determination of damping values is primarily the stress level in the component, not the basis or methodology used for response spectrum generation. That is, once a response spectrum is selected, the specified damping is based on the response of the structure under analysis in terms of fabrication methods and member stress levels. Newmark and Hall (accepted industry experts in seismic design and analysis) in NUREG/CR-0098, specify damping values of 2% to 3% for piping stressed to no more than ½ S_y and 5% to 7% for piping stressed to approximately the yield point, (S_y or F_y). They also state that "..... the lower levels of the pair of values given for each item are considered to be nearly lower bounds and are, therefore, highly conservative ..... the upper bound values are considered to be average or slightly above average values...". The ASME Boiler and Pressure Vessel Code, Section III, Division 1 , Appendix N, currently specifies 5% damping for the evaluation of the piping systems at both the Level B and Level D conditions. The Level D condition corresponds to the SSE event under evaluation here.

Based on these considerations 2 times the 5% damped ground spectra will be used for the analysis and evaluation of the seismic ruggedness of piping systems in the MSIV Leakage Pathway.

For static analysis, the methodology in Reference [7.17] can be used with the demand response spectrum described above. The methodology put forth in reference [7.17] and [7.22] is now incorporated in Appendix N of Section III of the ASME Boiler and Pressure Vessel Code (Article N-1225).

For localized evaluation of piping systems containing multiple changes of directions between lateral supports which provide the same direction of restraint, the piping can be projected into single beams in each of two orthogonal horizontal directions and the vertical direction. This would result in three separate collapsed beams. This approach called the "collapsed beam" approach is used in conjunction with the methodology of reference [7.11] and [7.18]. In creating these collapsed beams, segments of piping parallel to direction of interest shall be considered as concentrated loads of magnitude equal to the (weight/length) times length of the pipe segments. Segments of piping perpendicular to direction of interest shall be assembled into a continuous beam. The approach is shown in Figure 5.1 of Attachment 10.4. The maximum piping moment in each of the three projected simple beams is then determined by superposition of the concentrated loads upon a continuous beam.

4.5.4.2.2 Pipe Supports

Thermal, deadweight, and seismic pipe support loads for the pipe support evaluation of a specific walkdown package shall be taken from the computer analysis if a computer analysis was conducted for the system. Alternatively, if the simplified method described above is used for piping seismic stress evaluation, the span on either side of the support would be evaluated and the reactions from each of the individual span evaluations would then be absolutely summed.

For the analytical review of pipe supports, the following capacity criteria is used:

(a) Structural Steel

DWT + TH ≤ 1.0 AISC Normal Allowable (S) (5.4)

DWT + TH + SSE (Inertia and SAM) ≤ AISC Part 2 Allowable
{DWT + TH + SSE (Inertia and SAM) ≤ 1.7 S} (5.5)

(b) Component Supports

DWT + TH ≤ 1.0 MSS-SP-58 Allowable (5.6)

DWT + TH + SSE (Inertia and SAM) ≤ 2.0 MSS-SP-58 Allow. (5.7)

(c) Anchor Bolts

1) Anchor Bolt Equation for small bore piping (≤ 2-1/2" in diameter), supports when the applied loads are manually calculated and when the review team considers prying to be insignificant.

$$(VL/AVL)^{5/3} + (TL/ATL)^{5/3} \leq 1.0 \tag{5.8}$$

2) Anchor Bolt Interaction Equation for large bore analyzed piping and when Review Team considers Prying to be significant.:

$$(VL/AVL)^{5/3} + (1.25 \times TL/ATL)^{5/3} \leq 1.0 \tag{5.9}$$

Where:

VL	=	Applied Shear Load
AVL	=	Allowable shear load for the anchor bolt under review
TL	=	Applied Tensile Load
ATL	=	Allowable Tensile Load for the Anchor bolt under Review

The AISC – Steel Construction Manual (SCM) that will be used is the 6th Edition [7.20]. This is the edition that was used in the original design and construction of CNS. The 1.7 factor is based on the use of AISC-SCM Part 2 "Plastic Design Criteria" for the load combination involving the SSE. MSS-SP-58 [7.21] is the Industry Commercial Standard for the design and manufacture of Pipe Support Components. It is the basis of all manufacturer rated capacities given in manufacturer's catalogs. The ASME Boiler and Pressure Vessel Code Case N-500-1[7.19] permits an increase of 2.0 of the MSS-SP-58 rated loads for the Level D load case. The level D load case is equivalent to the SSE event under evaluation here. The anchor bolt evaluation uses a parabolic interaction equation. This equation was based on work conducted by Teledyne Engineering Services as part of the resolution of IE Bulletin 79-02 and is summarized and discussed in reference [7.26]. Based on data presented in reference [7.26] the parabolic relationship more accurately represents the tension-shear interaction relationship for concrete anchorage (then the linear interaction equation typically used) and provides a realistic definition of the anchorage capacity. Furthermore, it is also recommended for concrete anchorage design by the Portland Cement Association [7.27].

The effects of prying are addressed as follows:

a) For small bore pipe supports the applied piping loads and member sizes are small and, therefore, for existing supports prying effects are not considered to be significant and will not be considered. This is true for piping evaluated using manual calculations and/or computer analysis.

b) For large bore supports if the applied piping loads are determined using manual calculations then they are considered to be conservative estimates and additional consideration of prying effects is not warranted.
c) If large bore piping systems are computer analyzed then the effects of prying are considered by increasing the applied tensile bolt loads by a factor of 1.25.

Based on investigations conducted by CNS during the USI A-46 resolution program and field reviews during this program it was concluded the concrete anchor bolts used for pipe supports were "Phillips - Redheads" of the self drilling type, unless noted otherwise. The anchor bolt capacities of Appendix C of the SQUG-GIP were used, as they provide well established capacities for anchorage consistent with the use of experience data. Furthermore, since the SQUG-GIP was used to evaluate the equipment, the use of the SQUG-GIP anchorage capacities provided a consistent factor of safety between the piping support anchorage and the equipment anchorage. If anchor bolts exist that are not given in the SQUG-GIP, then the manufacturer's capacities will be used with a factor of safety 3.0. This will make the factor of safety compatible with the factor of Safety used in the SQUG-GIP. It will still be higher than the factor of safety of 2.0 permitted by the USNRC for operability evaluations.

Seismic pipe support loads are either obtained from detailed or simplified piping analyses or estimated as follows:

a) Determine the span length of piping expected to be restrained by the support.
b) Determine the total weight of the span, including pipe material, fluid, insulation, valves or other in-line components, and any other weights in the span length.
c) For horizontal loads, multiply the total weight by the peak of the horizontal demand spectrum (2.0 times the 5% damped Horizontal Ground Response Spectrum)). For vertical loads use 2/3 of the horizontal value (conservative as previously noted).

This is typically called the tributary mass approach.

4.5.4.3 Related Equipment

The seismic adequacy of related equipment was verified using the GIP methodology as detailed in reference [7.2]. Seismic capacity, caveat compliance, anchorage, and seismic spatial interaction concerns were addressed. The GIP Bounding Spectrum that was obtained from earthquake experience data was used to establish seismic capacity of all related equipment.

The majority of the related equipment are valves located at the lower elevations. Valve operability is not a concern for Cooper Nuclear Station because the applicable valves are either not required to reposition to establish the leakage path, or fail safe with respect to the leakage path. Since there is no reliance on standby power, none of the motor-operated valves were credited for operation.

4.5.4.4 Summary of Results for Piping and Pipe Supports

4.5.4.4.1 Results Overview

The piping material data, size, and schedules were obtained from piping and instrument diagrams (P&IDs) and line specifications. The line specifications also provide the design pressure and temperature data. The walkdowns and associated analysis evaluated the seismic capacity of the subject piping system. If the piping and/or piping supports could not be demonstrated to be seismically rugged, then suggested physical plant modifications were identified, which could make the piping systems seismically rugged.

Worst case supports were identified, and detailed evaluations are conducted for these supports. Short rod hangers susceptible to fatigue failure and U-bolts subjected to significant lateral loads are identified. Detailed evaluations are conducted to evaluate both the fatigue capacity of the rod hangers and the lateral load capacity of the U-bolts. For initial field screening purposes short rod hangers are rod hangers whose length was less than the smaller of 9" or 5 pipe diameters.

Three Piping systems were selected for detailed computer analysis:

a) Main Steam System (including the bypass piping)
b) Primary Leakage Pathway
c) Alternative Leakage Pathway

These systems were selected because they are the primary mechanisms for the delivery of the MSIV leakage to the condenser. In addition, these systems represent a wide range of piping sizes from 30" to 1" and multiple sizes in between. The analysis was conducted for Deadweight, Normal Operating Thermal, and SSE Seismic Loads. The loads from these analysis were used to conduct detailed evaluations of the associated pipe supports.

The seismic analysis was a detailed response spectra modal analysis using the methods of ASME BPVC, Appendix N. In addition, limited analytical reviews were conducted for portions of other piping systems which could be considered outside the screening criteria, which involved complex spatial interactions, were subject to high seismic anchor motions or for which a highly accurate prediction of piping support loads was required.

4.5.4.4.2 Correlation with the Piping Experience Data

After completion of the piping system walkdowns, evaluations were conducted to insure that the CNS piping systems fall within the range of the piping systems which constitute the experience data.

4.5.4.4.2.1 Piping Sizes

Table 6-1 of Attachments 9.2 and 10.4 presents a summary of the various piping, sizes, schedules and D/t (same as OD/t) ratios for each of the walkdown packages. Table 6-2 of Attachments 9.2 and 10.4 presents a general summary of the same data for the piping systems which constitute the experience data. More detailed summaries of the piping and the associated experience data are contained in reference [7.1]. Table 6-3 of Attachments 9.2 and 10.4 presents a comparison of the D/t ranges of the CNS piping to the experience data piping. The CNS piping systems in the leakage path are enveloped by the experience data with the following exceptions:

1. The experience data does not specifically identify the existence of 5" diameter piping.
2. The CNS 3/4" Piping has a lower bound D/t of 3.4 verses 5 in the experience data.

3. The CNS 1" piping has lower bound D/t of 4 versus 5 in the experience data.
4. The CNS 1" Piping has a lower bound D/t of 5 verses 7 in the experience data.
5. The CNS 24" piping has lower bound D/t ratio 20 versus 23 in the experience data.

For items (2) through (4), these lower D/t ratios are due to the use of thicker wall piping which would be stronger and have higher capacity than the experience data piping and therefore are not a concern. For (5), the exceedance is only 12 percent which is less than typical piping system fabrication tolerances. Therefore, this piping is adequately represented in the experience data. The 5" diameter, although not explicitly in the database, are enveloped by larger and smaller sizes. Therefore, this piping is adequately enveloped by the experience data and the supporting analysis.

4.5.4.4.2.2 Materials

Table 6.4 (a) of Attachments 9.2 and 10.4 provides a summary of the allowable stress capacity of the predominant piping materials of the experience data piping. Table 6.4 (b) of Attachments 9.2 and 10.4 provides a similar summary for the CNS piping. These tables demonstrate that the CNS piping in leakage path is adequately represented in the experience data piping.

4.5.4.4.2.3 Support Spans

Table 6-5 of Attachments 9.2 and 10.4 provides a summary of minimum and maximum ratios of the actual vertical support spans to the suggested ANSI B31.1 deadweight spans and the actual lateral support spans to the suggested ANSI B31.1 spans. Table 6-6 of Attachments 9.2 and 10.4 provides the suggested B31.1 deadweight support spans. Figures 6-1 through 6-4 of Attachments 9.2 and 10.4 compare the Cooper Nuclear Station piping maximum span ratios, Vertical Support Ratio (VSR) and Lateral to Vertical Support Span Ratio (LVSSR) to the experience piping span ratio data. These figures demonstrate that the CNS piping support spans are well represented and adequately enveloped by the piping experience data. The normalized span ratios are based on the actual span data from reference [7.1] and [7.23] and were used to expedite the walkdown process.

4.5.4.4.3 Summary of the In-depth Piping Analyses

This section provides a summary of the simplified and detailed piping analysis which were conducted for selected systems in the MSIV leakage path. Detailed response spectra modal analysis response spectra modal analysis were conducted for several piping systems. The analysis were conducted using the methods of the ASME BPVC, Section III, Division 1, Appendix N. The analysis was a realistic seismic analysis with intermodal and inter spatial combinations by the SRSS method. The input spectra used in all analysis was 2.0× the ground spectra and the missing mass correction was applied to simulate the high frequency response (>33hz). In addition to the Seismic Analysis, Deadweight and Normal Operating Thermal analysis were conducted.

Three Piping systems were selected for detailed computer analysis:

a. Main Steam System (including by pass piping)
b. Primary Leakage Pathway
c. Alternative Leakage Pathway

These systems were selected because they are the primary mechanisms for the delivery of the MSIV leakage to the condenser. In addition, these systems represent a wide range of piping sizes from 30" down to 1" Piping and multiple sizes in between. The analysis was conducted for Deadweight, Normal Operating Thermal, and SSE Seismic Loads. The loads from these analysis were used to conduct detailed evaluations of the associated pipe supports.

In addition to detailed dynamic piping analyses described in localized equivalent static analyses were used to (1) evaluate SAMs, (2) evaluate spatial interaction concerns (3) evaluate localized areas of seismic vulnerability and (4) to determine loads used in the detailed support evaluations. Table 6-9 of Attachments 9.2 and 10.4 provides a summary of the equivalent static analyses conducted.

4.5.4.4.4 Summary of Detailed Support Qualifications

Detailed support qualifications were conducted for all supports associated with the piping analysis described in the previous sections. In addition, support qualifications were conducted for worse case supports identified during the walkdowns. The basis for the determination of these worst case supports included the following concerns.

a) Short, fixed, or "hard spot" rod hangers that were judged to be susceptible to fatigue failure during a design basis SSE event.
b) U-bolts susceptible to significant lateral loads. In many cases a system may contain multiple U-Bolts that could experience significant lateral loads. In such cases one or two enveloping evaluations for such a system were conducted.
c) Supports that were judged to be the most susceptible to failure during a design basis seismic event based on field review.
d) Supports judged to have no ductile anchorages.

Table 6-10 of Attachments 9.2 and 10.4 provides a summary of the number of supports subjected to detailed analytical reviews and the basis of these reviews. These supports represent over 30% of the support population in the MSIV leak path. In addition, these supports are most susceptible to failure during a design basis seismic event. By demonstrating the acceptability of these supports, it is reasonable to assume that the supports for the MSIV leak path piping have adequate seismic capacity.

4.5.4.5 Summary of Results for Related Equipment

A Screening Evaluation Work Sheet (SEWS) was completed for each item. Each SEWS contains a capacity versus demand comparison, a checklist of bounding spectrum caveats, an anchorage review checklist, a spatial interaction checklist, notes, and attached pictures (if available). The SEWS identify the determination of whether the item is acceptable or is an outlier, and are then signed by the SRT (Seismic Review Team). A detailed list of the related equipment is specified in Section 2 of Attachment 10.4. Appendix C of Attachment 10.4 contains detailed Outlier Summary Sheets for these concerns. The outliers are summarized in section 4.6 of this EE.

The majority of the related equipment are valves. All valves were found to meet GIP screening criteria. Valve operability is not a concern because all of the valves are passive in the case of motor-operated valves, or fail safe as in the case of air- and solenoid-operated valves.

4.6 Proposed Modifications and Outlier Identification & Resolution

4.6.1 Pathway Modifications

In order to limit the scope of evaluations, limit modifications to existing piping, and to reduce the scope of future required maintenance/surveillance activities, and yet still fully meet all the applicable guidelines of the SE for the BWROG Report, NPPD has elected to modify the existing leakage flow path boundaries and/or change positions of specific flow path valves following a LOCA.

The proposed modifications under consideration are: 1) the addition of new isolation valves on various existing leakage path branches; and, 2) the addition of a new Main Steam bypass piping system or changes in the motive power source of the existing Main Steam bypass valves. These changes are proposed to establish a more direct leakage pathway from the MSIVs to the condenser than that which is currently provided. The proposed valves and additional piping will be installed in the future. Post-accident operator action will be required for some of the system isolations. The simplified system flow diagram showing the proposed piping and valves is included as Attachment 9.1.

4.6.2 Outlier Identification

a) NPPD has identified 44 existing pipe supports that potentially require some modification or additional justification to meet the new acceptance criteria.
b) Instances where an existing concrete block wall poses a seismic interaction concern with instrument tubing and four pressure indicators.
c) Two instances where Fire Protection piping includes cast iron fittings that potentially pose a seismic interaction concern with pathway equipment.
d) Two valve supports that require additional anchorage to meet the new acceptance criteria.
e) An overhead light fixture poses a potential seismic interaction concern with existing instrument air tubing.
f) An instrument rack that supports several instruments requires additional anchorage to meet the new acceptance criteria.

4.6.3 Outlier Resolution

In order to comply with the guidelines of the NRC's SE on the BWROG Report, NPPD will implement modifications and/or perform additional detailed analyses to resolve the piping/pipe support and equipment outliers identified above.

4.7 Maintenance of Pathway and Condenser

4.7.1 Structural Inspections/Maintenance Rule

The TB structure and piping and equipment supports within the TB are already subject to periodic structural inspections in support of NPPD's Maintenance Rule activities. In addition, the main steam piping system is inspected each cycle to potentially identify any deficiencies with pipe supports.

NPPD has expanded the scope of the Maintenance Rule structural inspections to include the subject SSCs as determined to be appropriate by the CNS Maintenance Rule program.

4.7.2 Inspections, Surveillance Testing, and Preventive Maintenance

SSCs that form a part of the MSIV leakage pathway to the condenser will be appropriately included in CNS's Maintenance Rule, Preventive Maintenance, and Surveillance Testing programs to the extent necessary for ensuring continued compliance with the SE for the BWROG Report. These program updates will be completed as part of the modifications to the pathway.

4.8 Updated Safety Analysis Report (USAR) Revision

Upon final review and approval by the NRC, the CNS USAR will be revised to reflect the design and licensing basis for CNS reliance on the MSIV leakage pathway to the condenser for radioactive iodine plateout and the seismic qualification of the pathway.

4.9 Technical Specification Changes

No Technical Specification changes are proposed for implementing this EE; however, this EE will be used to support meeting the license condition specified in Section 1 above.

5 CONCLUSIONS

5.1 Conclusion on Structural Integrity

Following the resolution of the identified "outliers", including the implementation of specified modifications, the applicable SSCs will meet full qualification to the applicable portions of the NRC's SE for the BWROG Report for crediting iodine removal in the main turbine condenser.

In the interim, NPPD will continue to credit iodine plateout in the main turbine condenser on an operability basis as previously accepted by the NRC. The NRC previously acknowledged that CNS would require modifications to meet the new acceptance criteria of the BWROG report by stating that "*This seismic evaluation will employ an analytical methodology acceptable to the staff and will identify any modifications necessary to support the evaluation*".

6 REFERENCES FOR EE 01-147

6.1 Letter from NRC to J. H. Swailes, NPPD, dated April 7, 2000, Cooper Nuclear Station-Issuance of Amendment on Design-Basis Accident Radiological Assessment Calculational Methodology Revision (TAC No. MA7758)

6.2 NPPD Letter to the NRC dated March 24, 2000 (NLS2000035)

6.3 NPPD Letter to the NRC dated March 29, 2000 (NLS2000036)

6.4 NRC staff's safety evaluation dated March 3, 1999, "Safety Evaluation of GE Topical Report, NEDC-31858P, Revision 2, 'BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems,' September 1993

6.5 GE Topical Report, NEDC-31858P-A, Revision 2, 'BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems,' August 1999

6.6 NEDC 92-143, Rev. 1, "Turbine Building Main Steam Pipe Analysis"

6.7 NEDC 2000-026, Rev. 2, "Evaluation of the Seismic Capability of the Condenser Anchorage for Safe Shutdown Earthquake Loading"

6.8 NEDC 2000-027, Rev. 0, "Evaluation of the Seismic Capability of the Turbine Building for Safe Shutdown Earthquake Loading"

6.9 NEDC 00-028, Rev. 0, "Structural Integrity Walkdowns of the MSIV Leakage Paths to the Condenser"

6.10 NEDC 00-029, Rev. 1, "Post-LOCA MSIV Leakage Path to Main Condenser"

6.11 NEDC 00-029A, Rev. 0, "Post-LOCA MSIV Leakage Evaluation"

6.12 NEDC 00-078, Rev. 0, "Review of Stevenson and Associates Report No. 00C4152-R01, Assessment of the Seismic Capacity of the MSIV Leakage Pathway for the Cooper Nuclear Station"

6.13 NEDC 01-006, Rev. 0, "MSIV Leakage Pathway Piping System Walkdown and Evaluation Packages"

6.14 NEDC 01-007, Rev. 0, "Seismic Qualification of Main Steam Drain Lines"

6.15 NEDC 01-008, Rev. 0, "Evaluation of the Seismic Capability of the MSIV Equipment for Safe Shutdown Earthquake Loading"

6.16 Seismic Qualification Utility Group (SQUG), Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, Rev. 3

6.17 Stevenson and Associates General Services Agreement 97A-MS16, Task Authorization 8

6.18 Stevenson and Associates General Services Agreement 97A-MS16, Task Authorization 4700000205

6.19 USAR Section XII, loepxviii11

6.20 USAR Appendix C, loepxviii11

6.21 CNS Technical Specifications

6.22 Cooper Nuclear Station Plant Unique Analysis Report Mark I Containment Program Revision 0, April, 1982

6.23 FSAR Amendment No. 9, FSAR Question No. 12.42

6.24 Original Operating License SER (OL-SER) section 6.2.2

6.25 NPPD Letter to the NRC dated December 22, 1999 (NLS990122)

6.26 NPPD's response to NRC RAI dated March 20, 2000 (NLS2000029)

6.27 NEDC 92-151, Rev. 2, "Code Qualification of Pipe Supports for the Turbine Building Main Steam System"

6.28 NRC letter dated 10-23-01 from M. Thadani (NRC) to D. Wilson (NPPD), "Cooper Nuclear Station-Issuance of Amendment Regarding Revised Radiological Dose Assessment and Technical Specification Changes (TAC No. MB1419)"

7 REFERENCES FROM ATTACHMENT 10.4

7.1 NEDC-31858P, Revision 2, General Electric, "BWROG Report for Increasing MSIV Leakage Ratio Limits and Elimination of Leakage Control Systems," September 1993, (principally Appendix D thereof).

7.2 EPRI/SQUG, "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment," Revision 2, February 1992.

7.3 "Supplemental Safety Evaluation Report No. 2 (SSER #2) on GIP-2, "USNRC, Washington, DC, May 22, 1992.

7.4 "Cooper Nuclear Power Station Verification of Seismic Adequacy of Mechanical and Electrical Equipment, Unresolved Safety Issue A-46 (SQUG)".

7.5 EPRI Report NP-5617 Volume 1 and 2, "Recommended Piping Seismic Adequacy Criteria Based on Performance during and after Earthquakes," January 1988.

7.6 Cooper, Updated Safety Analysis Report (USAR).

7.7 SSRAP, "Use of Experience and Test Data to Show the Ruggedness of Equipment in Nuclear Power Plants, Rev. 4.0, February 1991.

7.8 NUREG-1407 "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities", June 1991.

7.9 NUREG/CR-0098 "Development of Criteria for Seismic Review of Selected Nuclear Power Plants, May 1978.

7.10 NUREG-1367, "Functional Capability of Piping Systems", D. Terao and E.C. Rodabaugh, November 1992.

7.11 Letter from Frank A Akstulewicz, Acting Chief Generic Issues and Environmental Projects Branch, Division of Regulatory Improvements, Office of Nuclear Regulation, USNRC to Mr. T.A. Green, Sr., Project Manager, BWR Owner's Group Projects, dated, March 3, 1999, entitled," Safety Evaluation of GE Topical Report NEDC-31858P, Revision 2, 'BWROG Report for increasing MSIV Leakage Limits and Elimination of Leakage Control Systems', September 1993".

7.12 EQE International, "Generation of Conservative Design and Median-Centered In-Structure Response Spectra for the Cooper Nuclear Plant Control and Reactor Buildings", July, 1994.

7.13 NUREG/CR-6240, Stevenson & Associates, "Application of Bounding Spectra to Seismic Design of Piping Based on the Performance of Above Ground Piping in Power Plants Subject to Strong Motion Earthquakes", February 1995.

7.14 NUREG/CR-6239, Stevenson & Associates, "Survey of Strong Motion Effects on Thermal Power Plants in California with Emphasis on Piping Systems", December 1995.

7.15 Regulatory Guide 1.60, Revision 1, "Design Response spectra for Seismic Design of Nuclear Power Plants", December 1973.

7.16 B31.1, "Power Piping Code", 1973 Edition, Including the 1973 Addenda.

7.17 Adams, Timothy, M., and Stevenson, John, D., "Further Development of a Static Analysis Method for Piping Systems", 1995 PVP Conference, PVP Volume 313-2, pp 33-57.

7.18 USNRC Regulatory Guideline 1.61, Entitled "Damping Values for Seismic Design of Nuclear Plants".

7.19 ASME Boiler and Pressure Vessel Code, Section III, Division 1, 2001 Edition, including Sections III Codes Cases.

7.20 AISC - Steel Construction Manual, 6th Edition.

7.21 MSS - SP - 58, "Pipe Hangers and Supports - Materials, Design, and Manufacture".

7.22 WRC Bulletin 441, T. Adams and J. Stevenson, "Development of a Comprehensive Static Analysis Method Piping Systems", May 1999.

7.23 BWROG - 96011, K.P. Donovan to Mr. Dennis Crutchfield "BWR Owner's Group Response to Request Additional Information Regarding Topical Report NEDC - 31858 Rev 2; BWROG Report for increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems, February 19, 1996.

7.24 NEDC 00-029, "Post-LOCA MSIV Leakage Path to Main Condenser", Rev. 1

7.25 SE - Duane Arnold Energy Center - Amendment No. 207 to Facility Operating License No DRP-49, February 25, 1995.

7.26 EPRI Report TR-101968, Tier 2, Volume 3, "Guidelines and Criteria for Nuclear Piping and Support Evaluation and Design", Volume 3: Anchor Bolt Capacity and Installation Tolerances

7.27 Cook, Ronald A., Portland Cement Association Bulletin, "Strength Design of Anchorage to Concrete"

7.28 NPPD Procedure DEDP-09, "Seismic Walkdown Qualification of the MSIV Leakage Pathway to the Condenser", Rev. 0, 3/8/00

8 RECOMMENDATIONS

8.1 This EE should be used to develop the request to the NRC for reviewing the *"seismic evaluation to ensure the structural integrity of the main steam line piping from the main steam isolation valves (MSIV) to the main turbine condenser, the main turbine condenser, and the turbine building"* as required by the License Condition identified herein.

8.2 Revise this EE to reflect NRC review and approval documentation.

9 ATTACHMENTS FOR NRC SUBMITTAL

9.1 Drawing CNS-MS-43, Rev. 1, "Leakage Paths from Outboard MSIVs Cooper Nuclear Station"

9.2 Selected Figures and Tables from Attachment 10.4

10 OTHER ATTACHMENTS

10.1 Procedure 0.8, 10CFR50.59 Forms

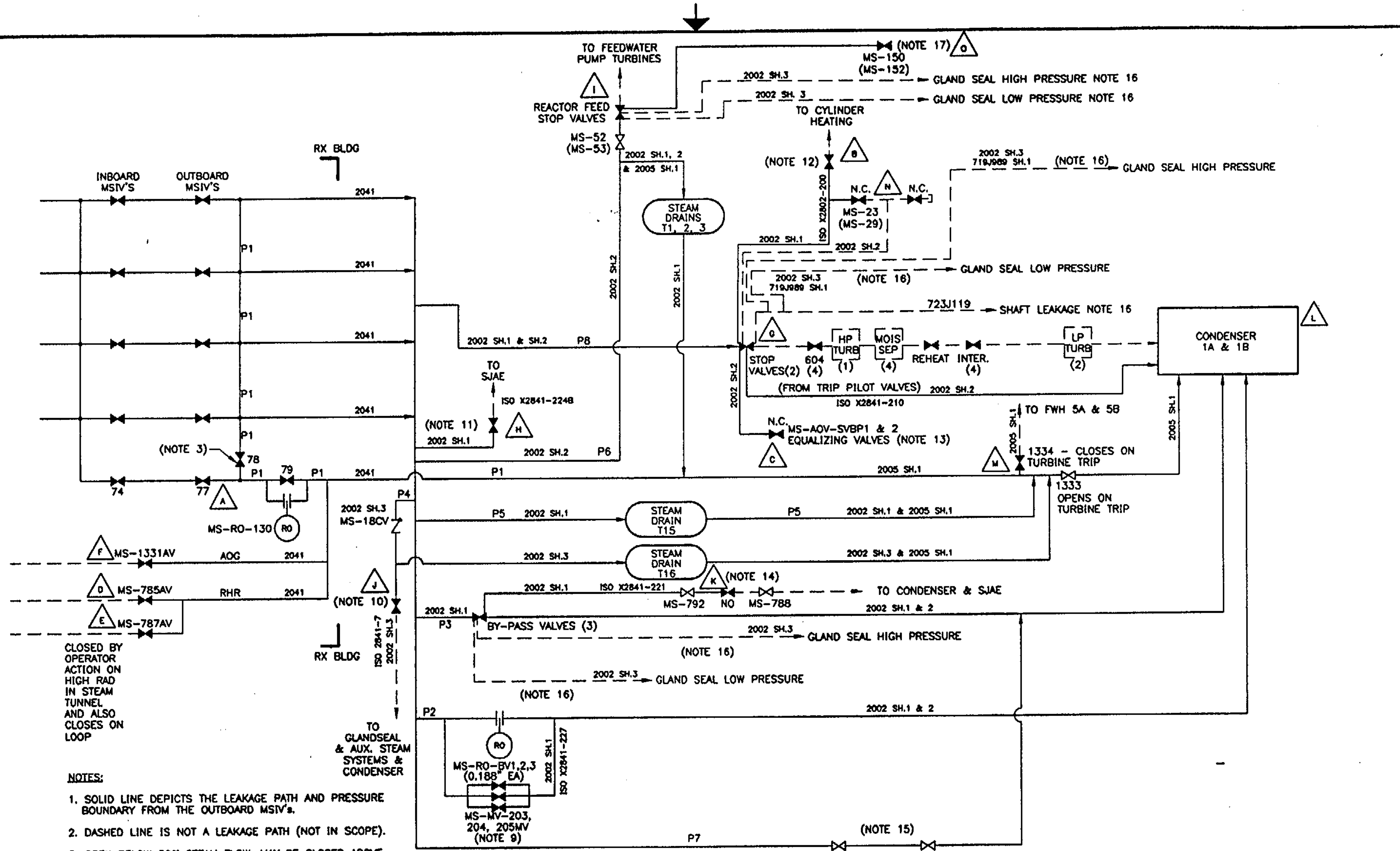
10.2 Procedure 3.3SAFE Forms

10.3 Procedure 3.4.8 IDV Forms

10.4 Stevenson and Associates Report AR-001, Rev. 0, "Seismic Evaluation of MSIV Leakage Pathway at Cooper Nuclear Power Station"

Attachment 9.1

Drawing CNS-MS-43



NOTES:

1. SOLID LINE DEPICTS THE LEAKAGE PATH AND PRESSURE BOUNDARY FROM THE OUTBOARD MSIV's.
2. DASHED LINE IS NOT A LEAKAGE PATH (NOT IN SCOPE).
3. OPEN BELOW 50% STEAM FLOW. MAY BE CLOSED ABOVE 50% STEAM FLOW. BOTH POSITIONS ARE CONSIDERED.
4. THIS NOTE NOT USED.
5. NOT ALL LINES SHOWN (e.g. INSTRUMENT AND SENSING LINES).
6. FLOW DRAWING NUMBERS SHOWN ON LINE e.g.: 2041
7. POTENTIAL LEAKAGE PATHS SHOWN WITH A "P" DESIGNATION EXAMPLE: P1.
8. BOUNDARY CONDITIONS SHOWN WITH A LETTER IN A TRIANGLE EXAMPLE: A
9. NORMALLY CLOSED.
10. ADD NEW VALVE TO CLOSE POST-LOCA.
11. ADD NEW VALVE TO CLOSE POST-LOCA.
12. ADD 2 NEW VALVES TO CLOSE POST-LOCA.
13. EQUALIZING VALVES NORMALLY CLOSED.
14. ADD NEW VALVE CONVIENTLY LOCATED TO CLOSE POST-LOCA.
15. ADD 2 NEW VALVES TO OPEN POST LOCA.
16. INSIGNIFICANT LEAKAGE.
17. CLOSE MS-150, 152 POST LOCA. FOR SEAT DRAINAGE.
18. VALVES SHOWN IN POST LOCA POSITION.

NO.	REVISIONS			
1	NEDC 00-029	RGA 12/4/01	CHK BY:	APPD BY:

INFORMATION ONLY

CADD DRAWING
DO NOT REVISE MANUALLY

LEAKAGE PATHS FROM OUTBOARD MSIV's
COOPER NUCLEAR STATION

DRAWN	DATE
RGA	8/24/01
CHECKED	1/17/02
APPROVED	2/4/02

Nebraska Public Power District

FILMED: CNS-MS-43 | REVISION: 1 | CADD FILE: C0043212

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Attachment 9.2

Selected Figures and Tables from Attachment 10.4

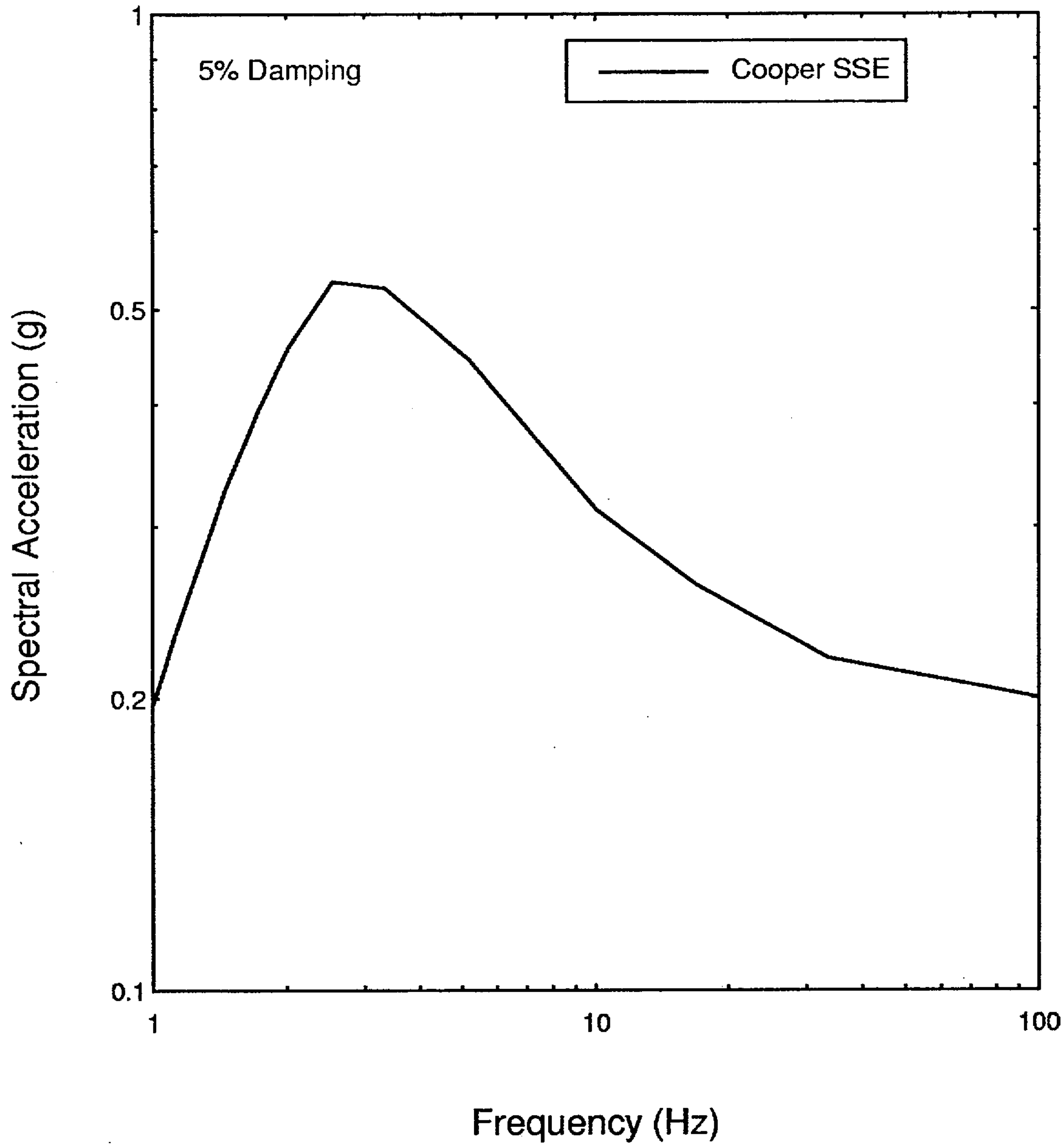

Figure 4.1 Cooper Nuclear Station Safe Shutdown Earthquake Ground Response Spectrum.

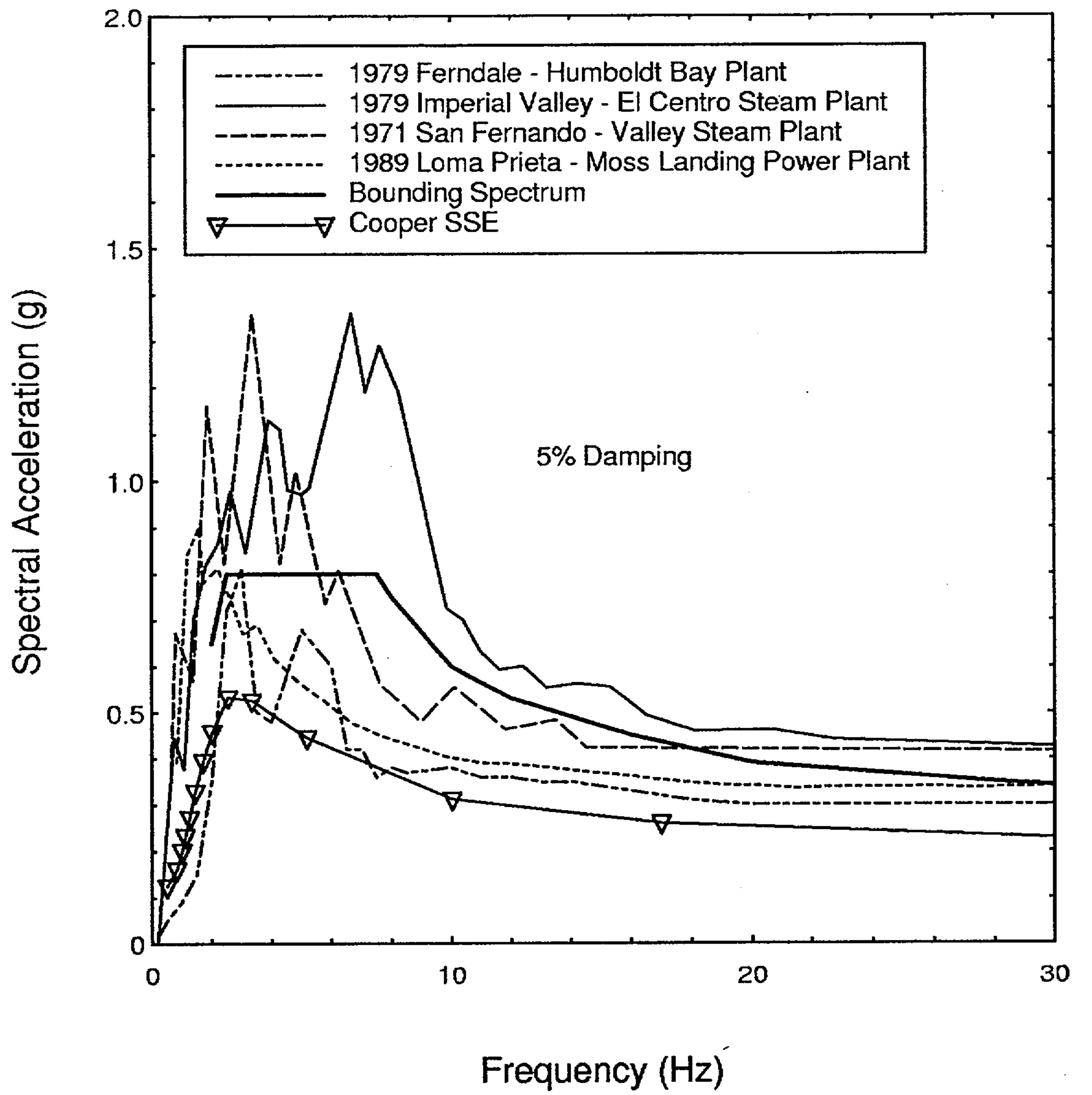

Figure 4.2 **Comparison of the CNS SSE Ground Response Spectrum, Bounding Spectrum and the Ground Spectra from the BWROG Survey**

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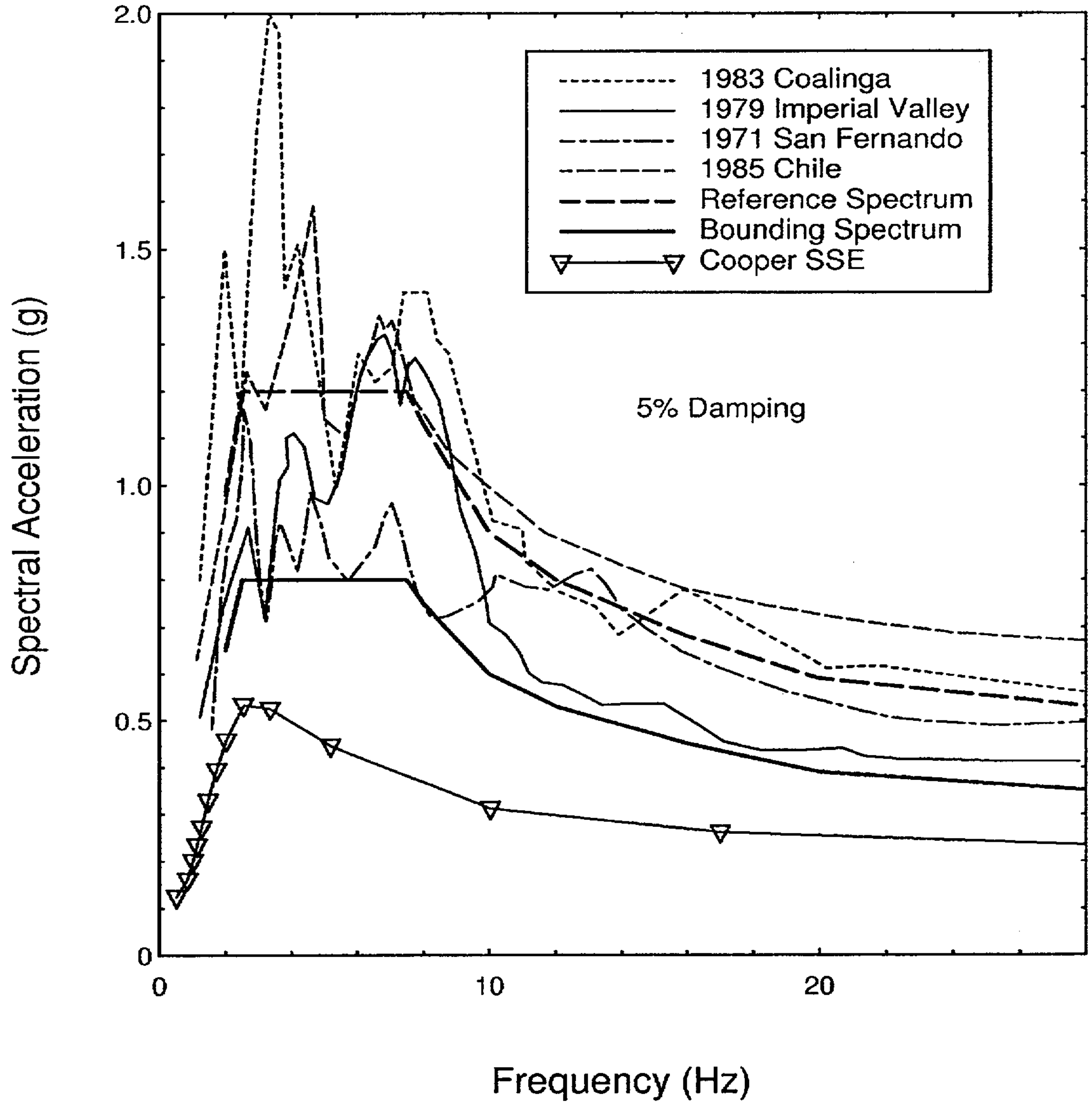

Figure 4.3 Reference Spectrum and Bounding Spectrum as developed in Reference [13].

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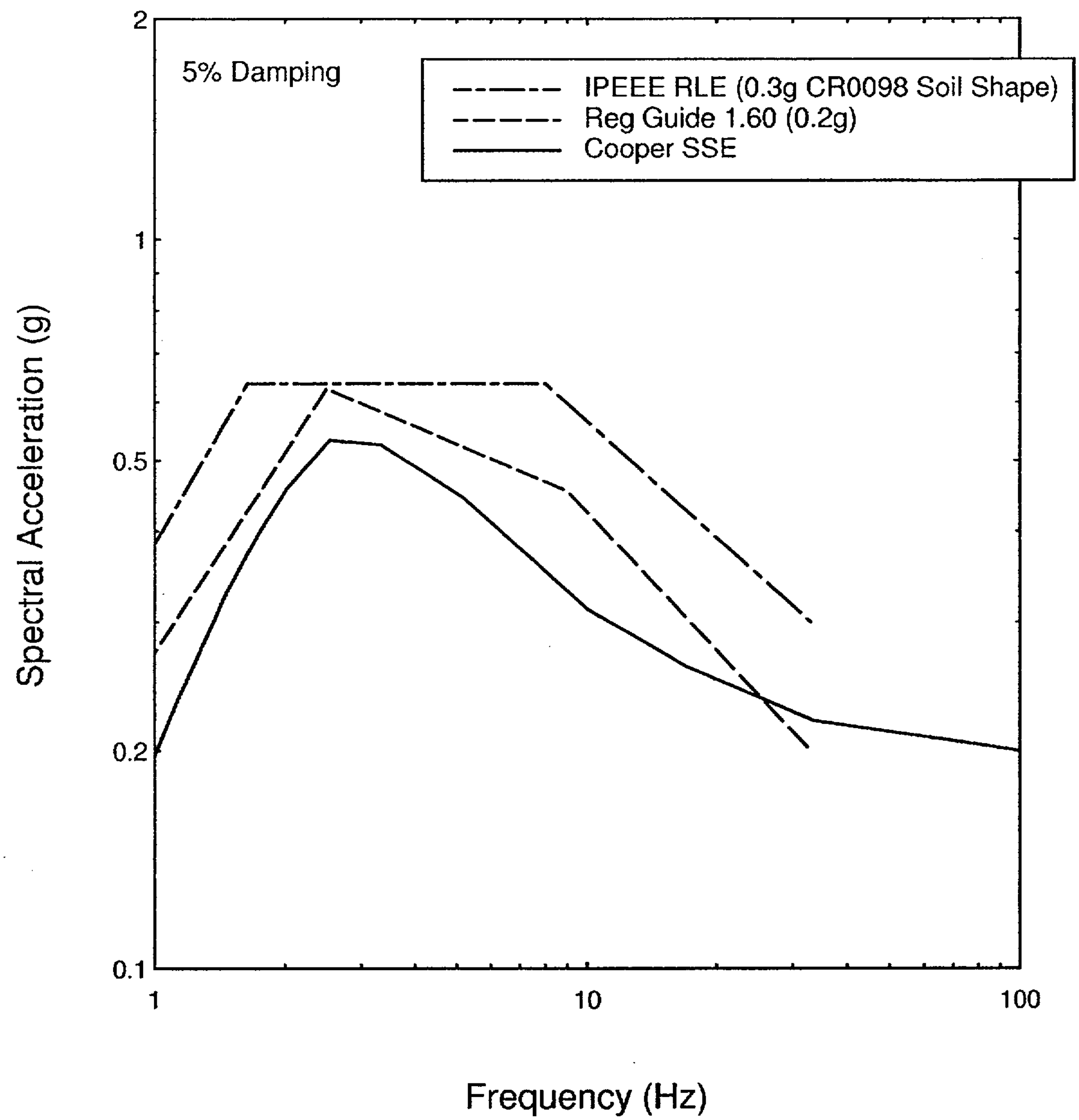

Figure 4.4 IPEEE RLE (Review Level Earthquake) v. Cooper SSE (Safe Shutdown Earthquake)

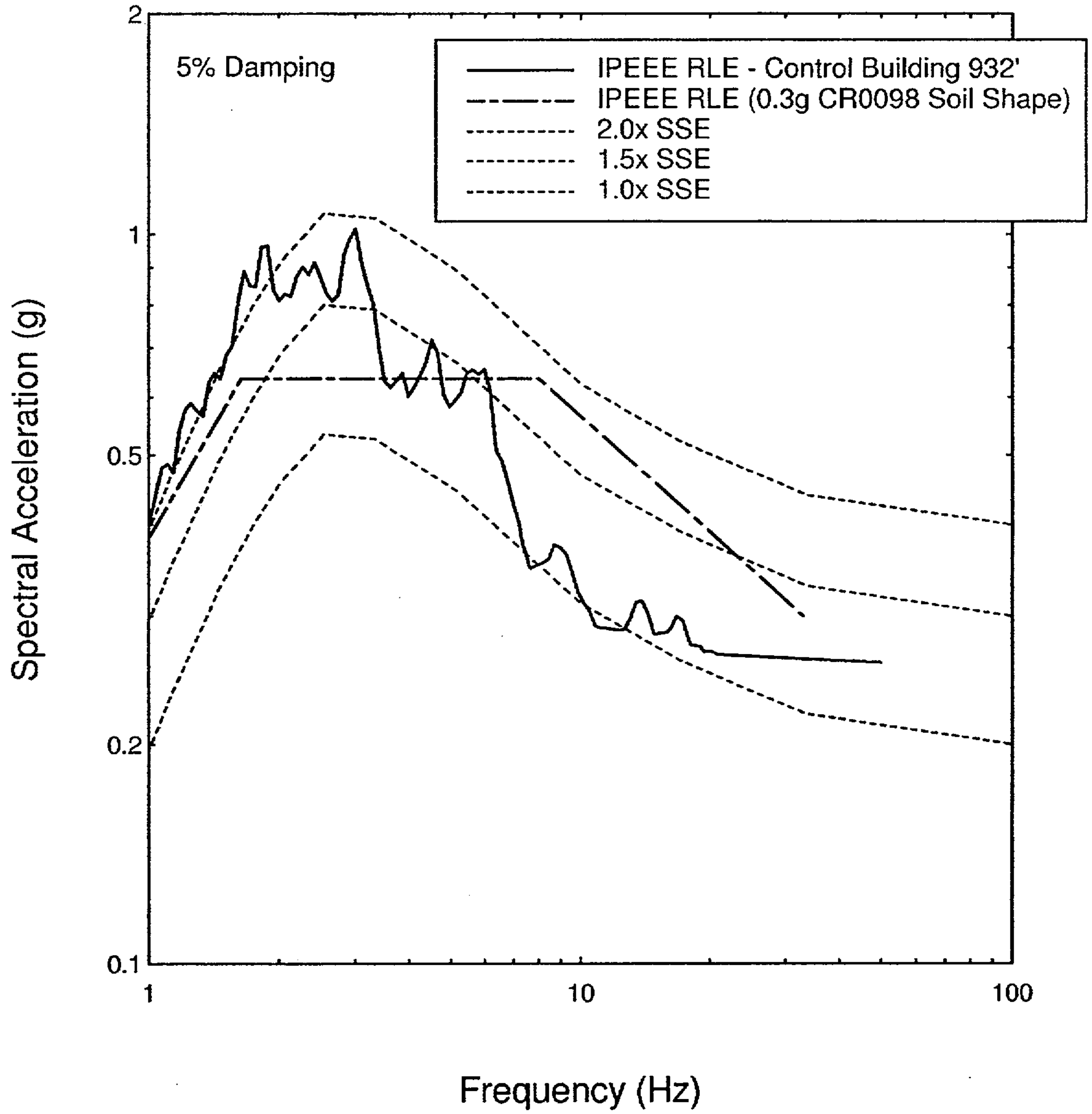

Figure 4.5 IPEEE RLE Floor Response Spectra for the Control Building

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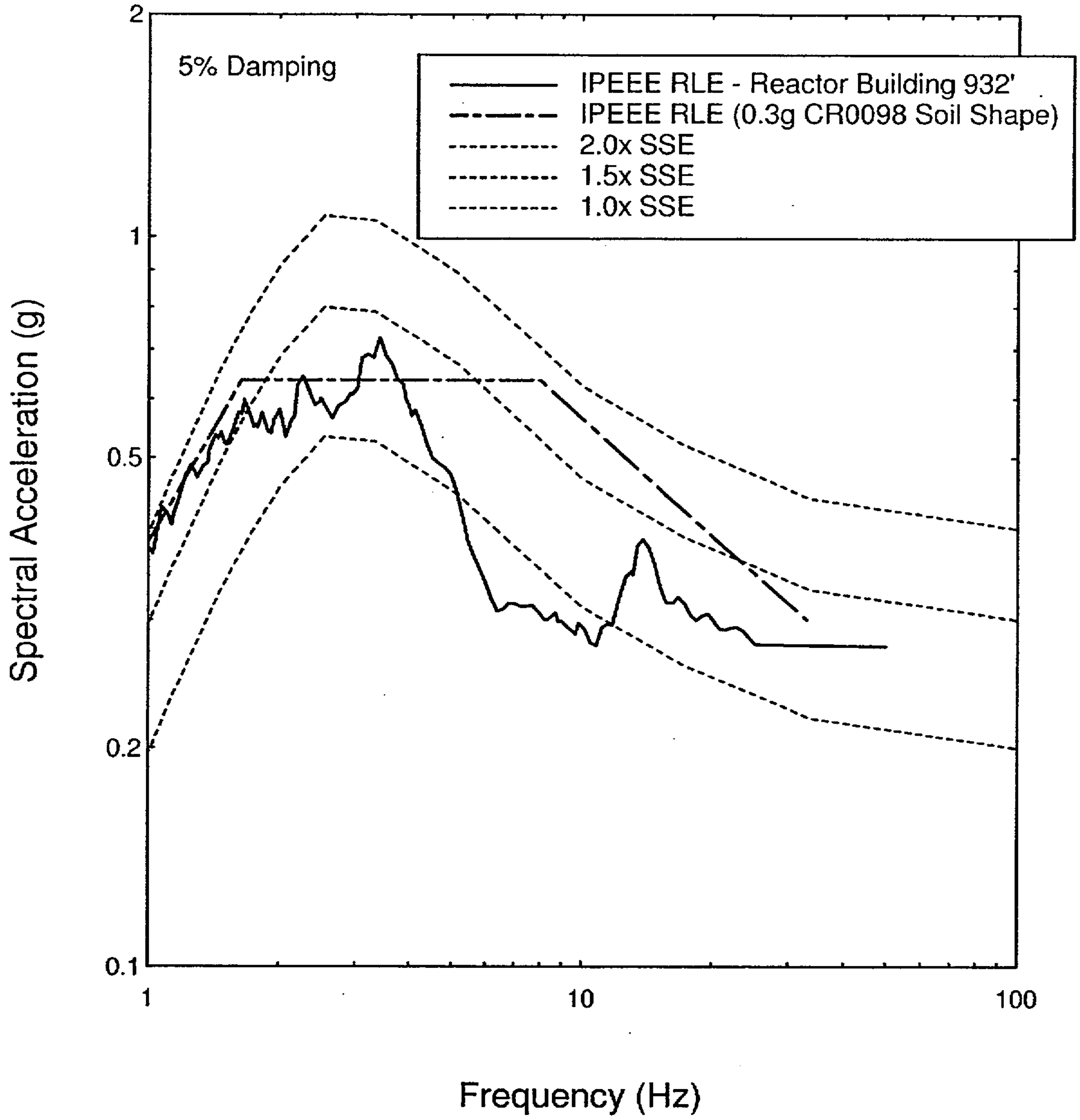

Figure 4.6 IPEEE RLE Floor Response Spectra for the Reactor Building

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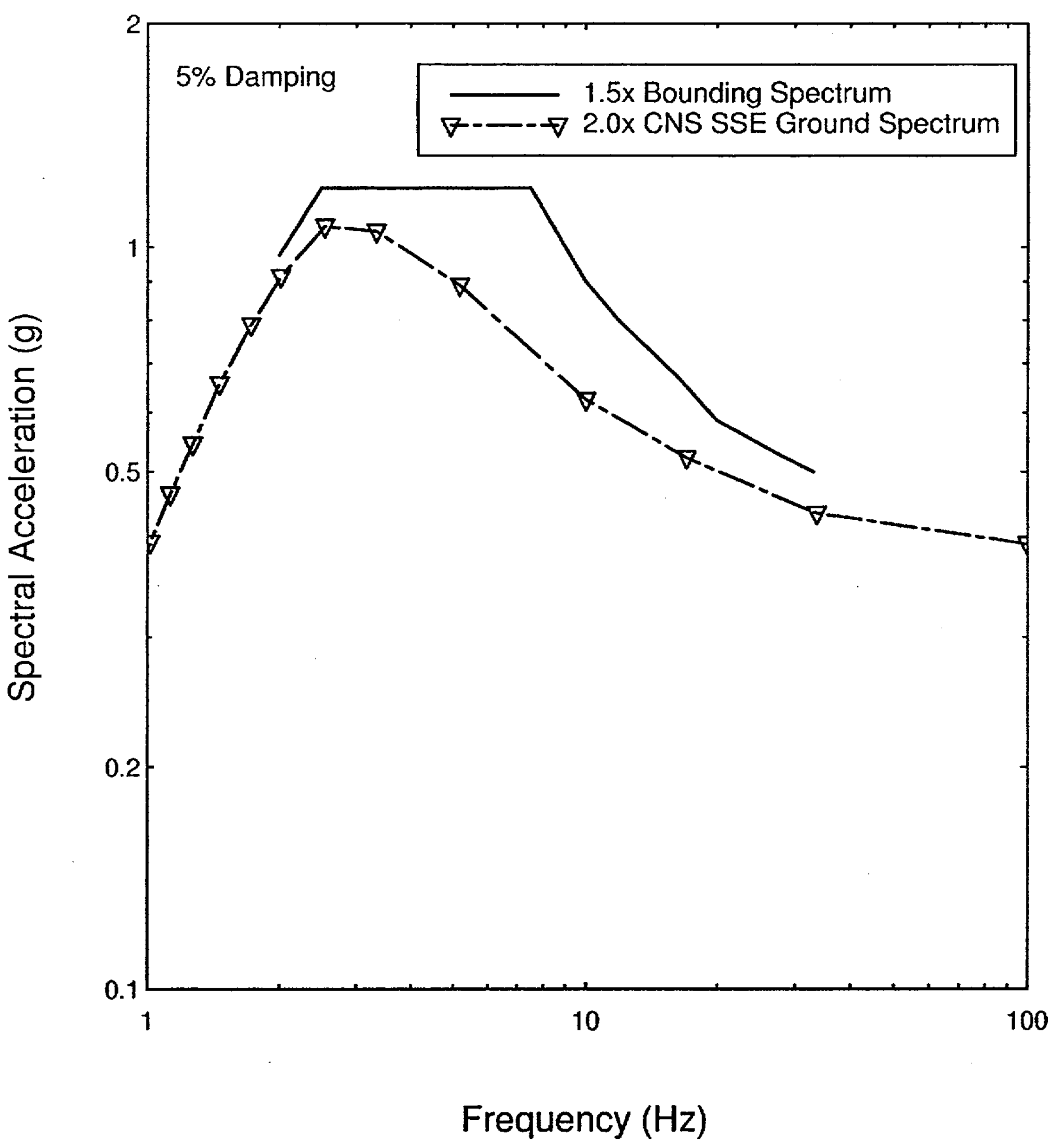

Figure 4.7 1.5 x Bounding Spectrum (Capacity) v. 2.0x CNS SSE Spectrum (Demand)

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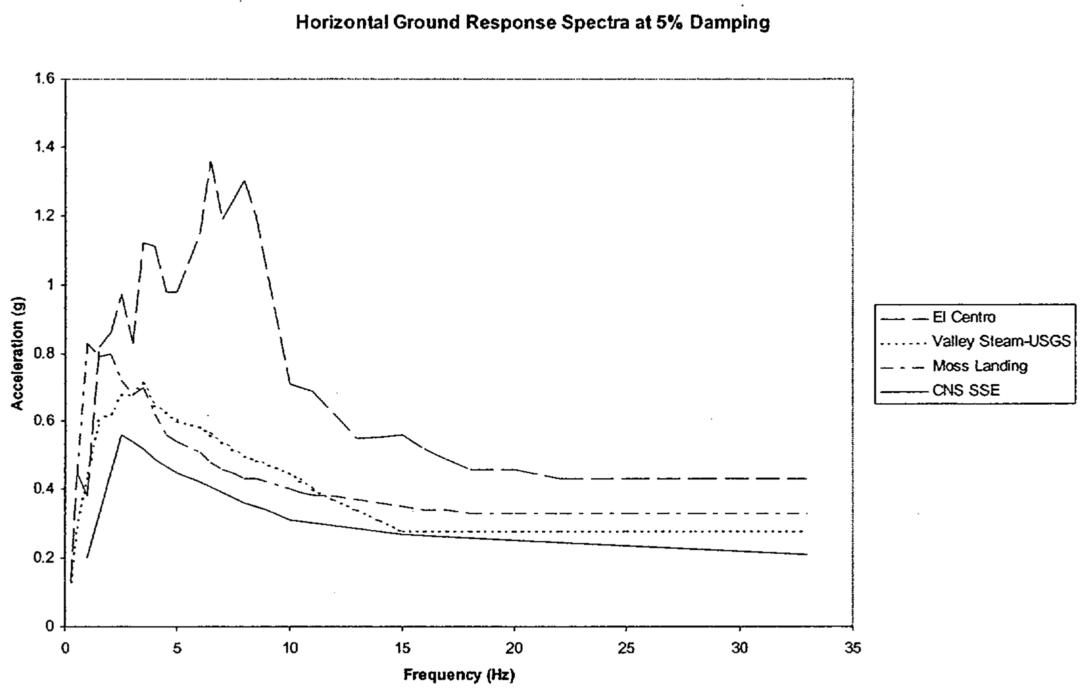

Figure 4.8 Selected Spectra from references [1], [8] vs. CNS SSE Ground Spectrum

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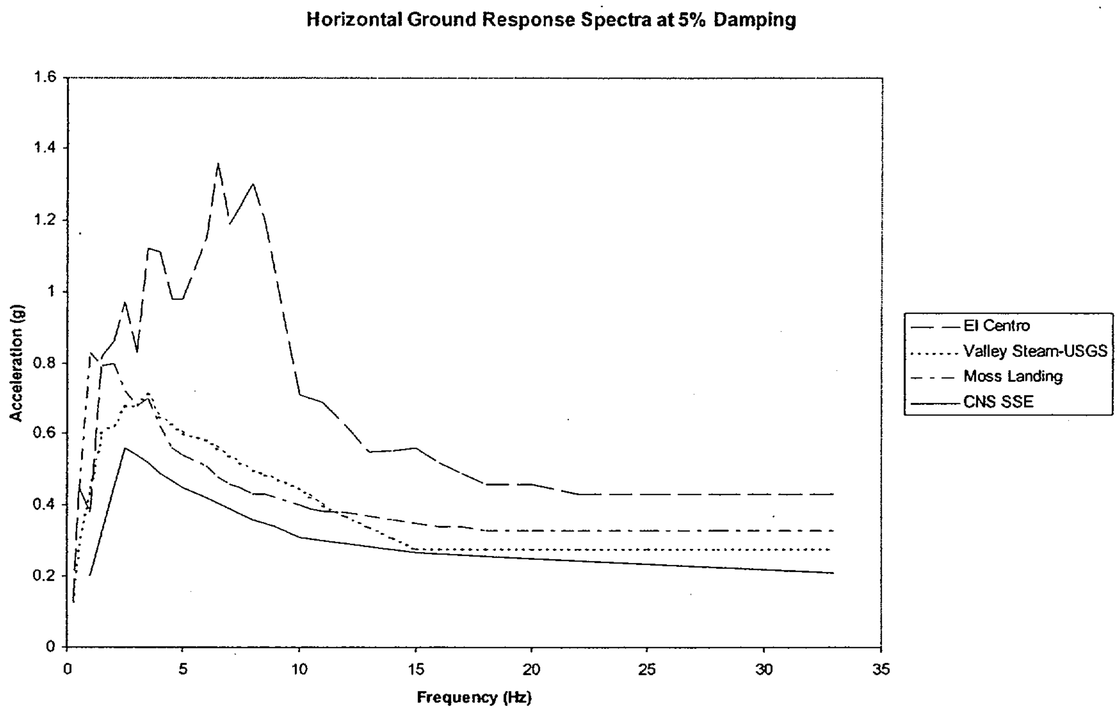

Figure 4.9 Selected Spectra from references [1], [11] vs. CNS SSE Ground Spectrum [5.9]

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Table 6-1: Summary of Piping Properties for the CNS Leakage Path Piping

Walkdown Package	Pipe Size NPS (in)	Pipe Schedule	Pipe OD (in)	Pipe Wall (in)	OD/t	Material ASTM/ASME Designation
NPPD-1	8	120	8.625	0.718	12	A106B
	18	80	18	0.937	19	A106B
	20	80	20	1.031	19	A106B
	24	80	24	1.218	20	A106B
	26	-	26.125	1.063	24.5	A106B
	30	-	30	1.18	25.4	A106B
	36	-	36	----	----	A106B
3223-01	2	160	2.375	0.394	7	A106B
	3	160	3.5	0.438	8	A106B
	1	XXS	1.315	0.358	3.6	A106B
	3/4	XXS	1.05	0.308	3.4	A106B
	8	120	8.625	0.718	12	A106B
3223-02	2	XXS	2.375	0.436	5	A106B
	1	XXS	1.315	0.358	3.6	A106B
	3/4	XXS	1.05	0.308	3.4	A106B
3223-03	6	80	6.625	0.432	1513	A106B
	4	80	4.5	0.337		A106B
3223-04A	2	XXS	2.375	0.436	53.6	A106B
	1	XXS	1.315	0.358		A106B
3223-04B	2	XXS	2.375	0.436	53.6	A106B
	1	XXS	1.315	0.358		A106B
3223-04C	2	XXS	2.375	0.436	53.6	A106B
	1	XXS	1.315	0.358		A106B
3223-05A	3/4	XXS	1.05	0.308	3.4	A106B
3223-05B	3/4	XXS	1.05	0.308	3.4	A106B
3223-05C	1 ½	XXS	1.9	0.4	4.7	A106B
3223-05D	1 ½	XXS	1.9	0.4	4.7	A106B
3223–05E	3/4	XXS	1.05	0.308	3.4	A106B

Table 6-1: Summary of Piping Properties for the CNS Leakage Path Piping

Walkdown Package	Pipe Size NPS (in)	Pipe Schedule	Pipe OD (in)	Pipe Wall (in)	OD/t	Material ASTM/ASME Designation
3223-06	3/4	XXS	1.05	0.308	3.4	A106B
	2	160	2.375	0.343	6.9	A106B
3223-07	½	XXS	0.84	0.294	3	A106B
	1	XXS	1.315	0.358	3.6	A106B
	2	160	2.375	0.343	6.9	A106B
3223-08	2	160	2.375	0.436	5.4	A106B
	3	160	3.5	0.438	8	A106B
	1	XXS	1.315	0.358	3.7	A106B
	1	XXS	1.315	0.358	3.7	A-312/316
	3	160	3.5	0.438	8	A-312/316
	6	80	6.625	0.432	15	A-312/316
3223-10	2	XXS	2.375	0.436	5	A106B
3223-16	3/4	XXS	1.05	0.308	3.4	A-312/316
	1	XXS	1.315	0.358	3.6	A-312/316
	16	STD	16	0.375	42	A-312/316
3223-24	1	XXS	1.315	0.358	3.6	A106B
	3/4	XXS	1.05	0.308	3.4	A106B
3223-26	3/4	160	1.05	0.218	4.8	A106B
	1	160	1.315	0.25	5.3	A106B
	1 ½	160	1.9	0.281	6.8	A106B
	2	160	2.375	0.343	6.9	A106B
3223-43	4	160	4.5	0.531	8.5	A106B
3223-44	4	160	4.5	0.531	8.5	A106B
3223-45	2	160	2.375	0.343	6.9	A106B
3223-46	2	160	2.375	0.343	6.9	A106B
3223-48A	1	XXS	1.315	0.358	3.6	A106B
3223-48B	1	XXS	1.315	0.358	3.6	A106B
3223-48C	1	XXS	1.315	0.358	3.6	A106B

Table 6-1: Summary of Piping Properties for the CNS Leakage Path Piping

Walkdown Package	Pipe Size NPS (in)	Pipe Schedule	Pipe OD (in)	Pipe Wall (in)	OD/t	Material ASTM/ASME Designation
3223-48D	1	XXS	1.315	0.358	3.6	A106B
3223-49A	3/4	80	1.05	.154	6.8	A106B
	1	80	1.315	.17	7.3	A106B
	1-1/2	80	1.9	.20	9.5	A106B
3223-49B	3/4	80	1.05	.154	6.8	A106B
	1	80	1.315	.179	7.3	A106B
	1-1/2	80	1.9	.20	9.5	A106B
3223-25-01	3	40	3.5	.216	16.2	A106B
	5	40	5.563	.258	21.	
3223-25-14	2	160	2.375	0.343	6.9	A106B
	1	XXS	1.315	0.358	3.7	A106B
3223-IC-07	-	-	½" Tubing	-	-	304L/304/316
3223-IC-08	-	-	½" Tubing	-	-	304L/304/316
3223-IC-09	-	-	½" Tubing	-	-	304L/304/316
3223-IC-10	-	-	½" Tubing	-	-	304L/304/316
3223-IC-11	-	-	½" Tubing	-	-	304L/304/316
3223-IC-12	-	-	½" Tubing	-	-	304L/304/316
3223-IC-13	-	-	½" Tubing	-	-	304L/304/316
3223-IC-14	-	-	3/4" Tubing	-	-	304L/304/316
3223-IC-15	-	-	3/4" Tubing	-	-	304L/304/316

Table 6-2: Seismic Experience Piping Data [1]

Plant	Pipe Size NPS (in)	Pipe Schedule	Pipe OD (in)	Pipe Wall (in)	OD/t
Valley Steam Plant Units 1 and 2	24	20	24.00	0.375	64
	20	20	20.00	0.375	53
	18	30	18.00	0.437	41
	16	30	16.00	0.375	43
	14	30	14.00	0.375	37
	12	40	12.75	0.406	31
	12	30	12.75	0.33	39
	10	160	10.75	1.125	10
	8	160	8.6250	0.906	10
	6	40	6.6250	0.28	24
	4	160	4.5000	0.531	8
	4	40	4.5000	0.237	19
	3	160	3.5000	0.437	8
	3	80	3.5000	0.3	12
	3	40	3.5000	0.216	16
	2	160	2.3750	0.343	7
	2	40	2.3750	0.154	15
	1 ½	160	1.9000	0.281	7
	1 ½	40	1.9000	0.145	13
	1	40	1.3150	0.133	10
	3/4	160	1.0500	0.218	5
	3/4	40	1.0500	0.113	9
Moss Landing Units 1, 2, & 3	16	N/A	16.00	1.394	11
	12	N/A	12.75	1.148	11
Moss Landing Units 4 & 5	24	40	24.00	0.687	35
	24	N/A	24.00	1.066	23
	-	N/A	18.30	2.287	8
	16	40	16.00	0.5	32
	16	N/A	16.00	0.902	18
	-	N/A	13.20	1.668	8
Moss Landing Units 6 & 7	30	N/A	30.00	0.632	47
	26	N/A	26.00	1.128	23
	18	N/A	18.00	3.444	5
	12	N/A	12.75	2.444	5
	12	N/A	12.75	0.601	21

Table 6-2 Seismic Experience Piping Data [1]

Plant	Pipe Size NPS (in)	Pipe Schedule	Pipe OD (in)	Pipe Wall (in)	OD/t
Humboldt Unit 3	12	80	12.75	0.687	19
	10	80	10.75	0.593	18
	6	80	6.625	0.432	15
El Centro Steam Plant	20	STD	20.00	0.375	53
	18	160	18.00	1.7810	10
	18	XS	18.00	0.5000	36
	18	STD	18.00	0.3750	48
	14	40	14.00	0.4370	32
	14	STD	14.00	0.3750	37
	12	160	12.75	1.3120	10
	12	STD	12.75	0.3750	34
	10	40	10.75	0.3650	29
	8	160	8.625	0.9060	10
	8	120	8.625	0.7180	12
	8	40	8.625	0.3220	27
	6	120	6.625	0.5620	12
	6	40	6.625	0.2800	24
	4	80	4.500	0.3370	13
	4	40	4.500	0.2370	19
	3	160	3.50	0.4370	8
	3	80	3.50	0.3000	12
	3	40	3.50	0.2160	16
	2	160	2.375	0.3430	7
	2	80	2.375	0.2180	11
	2	40	2.375	0.1540	15
	1 ½	160	1.90	0.2810	7
	1 ½	80	1.90	0.2000	10
	1 ½	40	1.90	0.1450	13
	1	80	1.315	0.1790	7
	1	40	1.315	0.1330	10
	3/4	80	1.050	0.1540	7
	3/4	40	1.050	0.1130	9

Table 6-3: D/t Range Comparison

Nominal Pipe Size (NPS) (ID)	Cooper Piping D/t Ranges	Experience Data Piping D/t Ranges
¾	3.4-7	5-9
1	4-5.5	5-20
-	5	-
1 ½	5-7	7-13
2	5-7	5-15
3	8-16	8-16
4	8-13	8-19
5	21	N/A
6	15	9-24
8	12	10-31
16	42	11-43
18	19	5-41
20	19	53
24	20	23-35
26	23	23
30	25	23 - 47
36	26	N/A

Table 6-4(a): Predominant Materials of the Experience Data

Material ASTM Designation	ANSI B31.1 Allowable Stress, psi
A53 B	15000
A106 B	15000
A335	14000
A120	(1)
A139	12000

(1) Stress allowables not provided by B31.1. B31.9 provides an allowable stress value of 10000.

Table 6-4(b): Predominant Materials of CNS Piping

Material ASTM Designation	ANSI B31.1 Allowable Stress, psi
A106 B	15000
312-304	15900
312-316	17000
312-304L	13700

Table 6-5: CNS Span Ratios Compared to ANSI B31.1 Suggested Deadweight Spacing

Walkdown Package	*Pipe Type SB = Small Bore (<2.5") LB = Large Bore (>2.5") [Based on Predominant Pipe Size]*	*Actual Maximum Vertical Support Spacing Ratio to B31.1 Suggested Support Spacing (2)*	*Actual Minimum Vertical Support Spacing Ratio to B31.1 Suggested Support Spacing*	*Actual Maximum Lateral Support Spacing Ratio to B31.1 Suggested support Spacing (LVSSR-Max) (2)*	*Actual Minimum Lateral Support Spacing Ratio to B31.1 Suggested Support Spacing (LVSSR - Min)*
NPPD-1	LB	(3)	---	(3)	---
3223-01	SB	(3)	---	(3)	---
3223-02	SB	3.2(1)	<1.0	3.3	<1.0
3223-03	LB	2.8(1)	<1.0	4	<1.0
3223-04A	SB	1.9(1)	<1.0	3.1	<1.0
3223-04B	SB	2.8(1)	<1.0	3.2	<1.0
3223-04C	SB	2.4(1)	<1.0	5.2	<1.0
3223-05A	SB	<1.0	<1.0	<1.0	<1.0
3223-05B	SB	<1.0	<1.0	<1.0	<1.0
3223-05C	SB	<1.0	<1.0	<1.0	<1.0
3223-05D	SB	<1.0	<1.0	<1.0	<1.0
3223–05E	SB	<1.0	<1.0	<1.0	<1.0
3223-06	SB	1.5(1)	<1.0	5.0	3.6
3223-07	SB	1.4	<1.0	4.5	<1.0
3223-08	SB	(3)	—	(3)	---
3223-10	SB	4	<1.0	3.9	<1.0
3223-16	LB	2.3	<1.0	3.6	<1.0
3223-24	SB	3.0	≈1.0	6.0	≈1.0
3223-26	SB	4.0(1)	0.9	2.6	0.9
3223-43	LB	<1.0	<1.0	<1.0	<1.0
3223-44	LB	<1.0	<1.0	<1.0	<1.0
3223-45	SB	2.6(1)	<1.0	3.1	<3.0

Table 6-5: CNS Span Ratios Compared to ANSI B31.1 Suggested Deadweight Spacing

Walkdown Package	*Pipe Type SB = Small Bore (<2.5") LB = Large Bore (>2.5") [Based on Predominant Pipe Size]*	*Actual Maximum Vertical Support Spacing Ratio to B31.1 Suggested Support Spacing (2)*	*Actual Minimum Vertical Support Spacing Ratio to B31.1 Suggested Support Spacing*	*Actual Maximum Lateral Support Spacing Ratio to B31.1 Suggested support Spacing (LVSSR-Max) (2)*	*Actual Minimum Lateral Support Spacing Ratio to B31.1 Suggested Support Spacing (LVSSR - Min)*
3223-46	SB	2.6(1)	<1.0	3.1	<3.0
3223-48A	SB	<1.0	<1.0	<1.0	<3.0
3223-48B	SB	<1.0	<1.0	<1.0	<3.0
3223-48C	SB	<1.0	<1.0	<1.0	<1.0
3223-48D	SB	<1.0	<1.0	<1.0	<1.0
3223-49A	SB	<1.5	<1.0	<1.0-2.0	<1.0
3223-49B	SB	<1.5	<1.0	<1.0-2.0	<1.0
3223-25-01	LB	1.5	<1.0	2	<1.0
3223-25-14	SB	1.5	<1.5	3	<3.0
3223-IC-07	½" Tubing	(4)	(4)	(4)	(4)
3223-IC-08	½" Tubing	(4)	(4)	(4)	(4)
3223-IC-09	½" Tubing	(4)	(4)	(4)	(4)
3223-IC-10	½" Tubing	(4)	(4)	(4)	(4)
3223-IC-11	½" Tubing	(4)	(4)	(4)	(4)
3223-IC-12	½" Tubing	(4)	(4)	(4)	(4)
3223-IC-13	½" Tubing	(4)	(4)	`(4)	(4)
3223-IC-14	3/4" Tubing	(4)	(4)	(4)	(4)
3223-IC-15	3/4" Tubing	(4)	(4)	`(4)	(4)

Notes for Table 6.5:

(1) These spans exclude consideration of spring hangers.

(2) Spans include consideration of modified or added supports.

(3) This was a system qualified by detailed analysis and therefore, span comparisons are not applicable.

(4) Tubing lines were reviewed and evaluated based on the judgement and experience of the review team members. Tubing data is not specifically provided in Ref [1].

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Table 6-6: Nominal Suggested Vertical Deadweight Spans per ANSI B31.1

Cooper Nominal Pipe Size** (in)	Outside Pipe Diameter (in)	Suggested B31.1 Deadweight Spans (ft) Water Service	Steam. Gas or Air Service
3/4	1.050	6*	8*
1	1.315	7	9
1 ½	1.900	9*	11*
2	2.375	10	13
3	3.500	12	15
3 ½	4.000	11*	12*
4	4.500	14	17
5	5.563	16	19*
6	6.625	17	21
8	8.625	19	24
10	10.750	21*	26*
12	12.750	23	30
14	14.000	25*	33*
16	16.000	27	35
18	18.000	29*	37*
24	24.000	32	42

*** Interpolated values – not given directly in ANSI B31.1.**
**** There are small amounts of ½" piping and I/C tubing (1/8", 1/4", ½", 5/8" and 3/4") not presented in this table.**

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Table 6-7: Cooper Condenser Design Data Versus Experience Data [1]

Parameter	Cooper	Moss Landing 6 & 7	Ormond Beach 1 & 2
Manufacturer	Maryland Shipbuilding and Dry Dock Co.	Ingersoll Rand	Southwestern
Flow Type	Single Pass	Single Pass	Single Pass
Shell Dimensions (L x W x H)	62' x 31' x 47'	65' x 36' x 47'	52' x 27' x 20'
Tube Area per Shell	465,000 ft^2	435,000 ft^2	210,000 ft^2
Shell Material	ASTM A285C	ASTM A285C	ASTM A285C
Shell Thickness	7/8 inch	¾ inch	¾ inch
Operating Weight	3,139,000 lbs.	3,115,000 lbs.	1,767,000 lbs.
Tube Material	Type 304 S.S.	Al-brass	90-10 Cu-Ni
Tube Size	7/8 inch	1 inch	1 inch
Tube Length	60 feet	65 feet	53 feet
Tube Wall Thickness	22 Bwg	18 Bwg	20 Bwg
Number of Tubes	25550	25590	15,220 per shell
Tube Sheet Material	Aluminum Bronze ASTM B-169D	Munz Metal	Munz Metal
Tube Sheet Thickness	1½ inch	1½ inch	1¼ inch
No. of Tube Support Plates	13 per shell	15	14
Tube Support Plate Material	ASTM A285C	not identified	ASTM A285C
Tube Support Plate Thick.	5/8 inch	3/4 inch	5/8 inch
Tube Support Plate Spacing	39 inches	48 inches	36 to 36.5 inches
Waterbox Material	ASTM A285C	2% Ni cast iron ASTM A-48 CL 30	ASTM A285C
Waterbox Plate Thickness	½-¾ inch	N/A	5/8 to 1 inch
Expansion Joint	Rubber belt	Rubber belt	St. steel
Hot Well Capacity	68,700 gallons	20,000 gallons	34,338 gallons
Hot Well Hold Time	N/A	N/A	N/A

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Table 6.9 - Summary of Equivalent Static Analyses Conducted

Based for the Equivalent Static Analyses	Number of Equivalent Static Analyses Conducted for this Reason
Evaluate SAMs	2
Evaluated Spatial Interactions	2
Evaluate Local Vulnerabilities	10
Determined Support Loads for Evaluation	5
Non-Conforming Materials	0

Table 6.10 - Summary of the Detailed Support Qualifications

Based of the Qualification	Number of Supports Evaluated
Rod Fatigue Concerns	3
Lateral U-Bolt Concerns	0(1)
Worse Case Support Reviews	12
Supports on Systems Subjected to Detailed Analysis	137
Total	**149**

Notes for Table 6.10:
(1) Lateral U-Bolts classified and considered with worst case supports

Table 6.11 - Maximum Primary Stress Levels for the Main Steam System (P+D+SSE)

Location		Equation 5.2b	
Analysis Node Point	Pipe Size (in)	Developed Stress (ksi)	Allowable Stress (ksi)
15	24	14.6	36
3285	26	41.3	42
2275	36	24.3	42
4055	30	24.5	42
4230	20	21.7	36
4230	8	26.2	36
4540	18	21.4	36

Table 6.12 Maximum Primary Stress for Package WD-3223-01 (P+D+SSE)

Location		Equation 5.2b	
Analysis Node Point	Pipe Size (in)	Developed Stress (ksi)	Allowable Stress (ksi)
945	1"	8.8	36
196	2"	7.2	36
445	3"	20.1	41.6
730, 735	6"	8.9	41.6

Table 6.13 Maximum Stress for Package WD-3223-08 (P+D+SSE)

Location		Equation 5.2b	
Analysis Node Point	Pipe Size (in)	Developed Stress (ksi)	Allowable Stress (ksi)
445	1"	17.7	36
345	2"	15.1	36
52	3"	15.5	36
643	3/4"	1.9	36

Figure 6.1 - Large Bore Vertical Support Span Ratio of the Subject CNS Piping Compared to the Experience Data

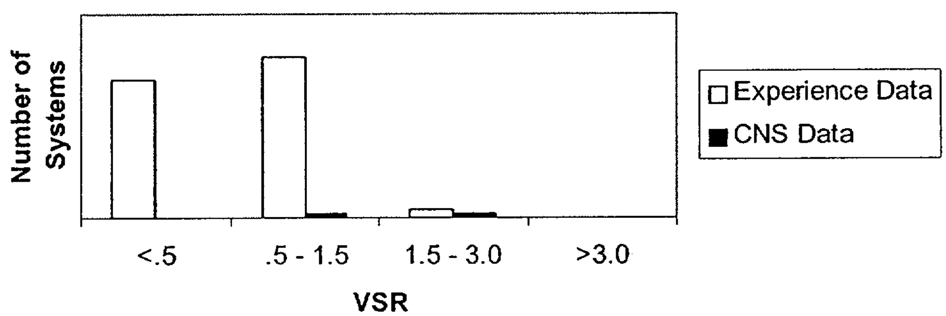


Figure 6.2 - Small Bore Vertical Support Span Ratio of the Subject CNS Piping Systems Compared to the Experience Data

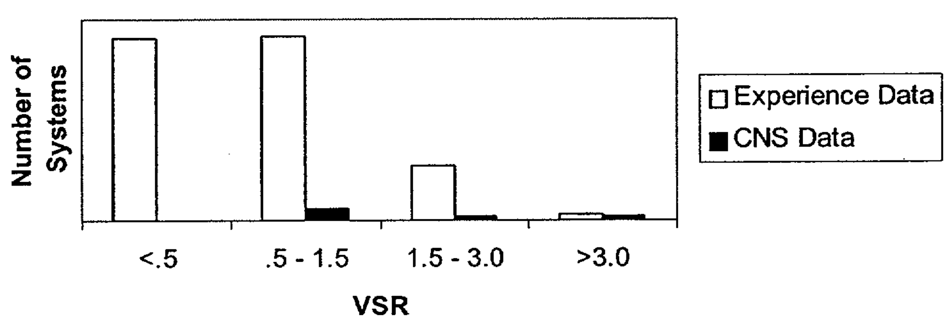


Figure 6.3 - Large Bore Piping Comparision of the LVSSR of the Subject CNS Piping to the Experience Data

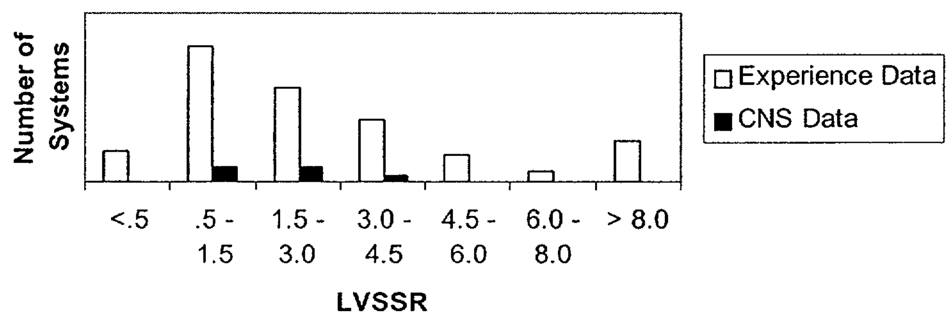
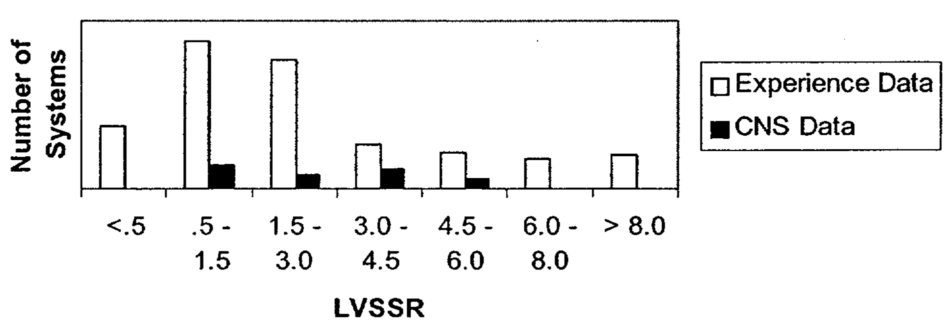


Figure 6.4 - Small Bore Piping Comparision of the LVVSR of the Subject CNS Piping to the Experience Data



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ATTACHMENT 3	LIST OF REGULATORY COMMITMENTS

Correspondence Number: NLS2002014

The following table identifies those actions committed to by the District in this document. Any other actions discussed in the submittal represent intended or planned actions by the District. They are described for information only and are not regulatory commitments. Please notify the NL&S Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
NPPD commits to submit the finalized description of modifications needed to configure the MSIV leakage pathway, which will take into account the resolution of identified NRC concerns, if any.	July 30, 2002