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NLS2015017
February 11, 2015

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: Revision to Nebraska Public Power District's Response to Nuclear Regulatory Commission Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Seismic Aspects of Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident
Cooper Nuclear Station, Docket No. 50-298, DPR-46

References:
1. NPPD Letter to Nuclear Regulatory Commission, "Nebraska Public Power District's Seismic Hazard and Screening Report (CEUS Sites) - Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 31, 2014

2. NEI Letter to NRC, "Proposed Path Forward for NTTF Recommendation 2.1: Seismic Reevaluations," dated April 9, 2013

3. NRC Letter to NPPD, "Cooper Nuclear Station - Screening and Prioritization Results of Information Provided Pursuant to Title 10 of the Code of Federal Regulations Part 50, Section 50.54(f), Seismic Hazard Reevaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident (TAC No. MF3734)," dated October 28, 2014

Dear Sir or Madam:

In Reference 1, Nebraska Public Power District (NPPD) submitted the Seismic Hazard Evaluation and Screening Report for Cooper Nuclear Station (CNS). Reference 1 concluded that CNS screened out of the expedited seismic evaluation process. During subsequent review of Reference 1, an error was discovered in the computer models that developed the ground motion response spectrum (GMRS) for CNS; specifically, incorrect kappa values were utilized.

The purpose of this letter is to submit a revised Seismic Hazard Evaluation and Screening Report for CNS (Enclosure). Revision 2 of the Seismic Hazard Evaluation and Screening Report incorporates the following changes:

A010
NRR

- Updated GMRS information
- Added Appendix B, IPEEE Adequacy Review
- Added Appendix C, Soil Failure and Liquefaction Evaluation for IPEEE Adequacy Review

Based on the results of the enclosed screening evaluation, NPPD will perform an expedited seismic evaluation in accordance with the Nuclear Energy Institute's seismic proposed path forward letter (Reference 2) and will submit the expedited seismic evaluation by May 1, 2015, as discussed in the Nuclear Regulatory Commission's screening and prioritization results for CNS (Reference 3).

There are no new regulatory commitments contained in this letter.

I declare under penalty of perjury that the foregoing is true and correct.

Executed On 2/11/15
(Date)

Sincerely,

Oscar A. Limpias
Vice President - Nuclear and
Chief Nuclear Officer

/bk

Enclosure: Seismic Hazard Evaluation and Screening Report for Cooper Nuclear Station, Revision 2

cc: Regional Administrator, w/ enclosure
USNRC - Region IV

Director, w/ enclosure
USNRC - Office of Nuclear Reactor Regulation

Cooper Project Manager, w/ enclosure
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector, w/ enclosure
USNRC - CNS

CNS Records, w/ enclosure

NPG Distribution, w/o enclosure

NLS2015017

ENCLOSURE

SEISMIC HAZARD EVALUATION AND SCREENING REPORT FOR COOPER NUCLEAR STATION, REVISION 2

Engineering Report No. 14-003 Rev 1

Page 1 of 4

Engineering Report Cover Sheet

Engineering Report Title:

50.54(f) Section 2.1 Seismic –

Black & Veatch *Seismic Hazard and Screening Report Cooper Nuclear Station* Acceptance

Engineering Report Type:

New ☐ Revision ☒ Cancelled ☐ Superseded ☐

Superseded by: ______

EC No. N/A - Exempt (Admin)

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(5) **Quality-Related:** ☐ Yes ☒ No

Prepared by: Derek Helmick [signature] Date: 1/2/2015

Responsible Engineer (Print Name/Sign)

Design Verified: N/A Date: ______

Design Verifier (if required) Print Name/Sign)

Reviewed by: Michael Pettini [signature] Date: 1/2/2015

Reviewer (Print Name/Sign)

Approved by: Marshall Van Winkle [signature] Date: 1-7-15

Supervisor / Manager (Print Name/Sign)

1. Scope and Objective

In responding to the Fukushima Near-Term Task Force Recommendation 2.1 Seismic; Cooper Nuclear Station (CNS) contracted Black & Veatch Corporation as a subject matter expert to develop the Seismic Hazard and Screening Report in accordance with EPRI Report *Screening, Prioritization and Implementation Details* (SPID) [Reference 2].

The information within this report is intended for use in responding to the Fukushima Near-Term Task Force Recommendation 2.1: Seismic.

This Engineering Report accepts the revised Seismic Hazard and Screening Report which was developed by Black & Veatch for CNS.

2. Design Inputs

The design inputs are as listed:

1. CNS Letter NLS2013085, "*Nebraska Public Power District's Response to Nuclear Regulatory Commission Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Seismic Aspects of Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident- 1.5 Year Response for CEUS Sites Cooper Nuclear Station*", NRC Docket No. 50-298, License No. DPR-46.

3. Assumptions

No assumptions were made by CNS in the development of this Engineering Report.

4. Detailed Discussion

Following the accident at the Fukushima Daiichi nuclear power plant resulting from the March 11, 2011, Great Tohoku Earthquake and subsequent tsunami, the Nuclear Regulatory Commission (NRC) established a Near Term Task Force (NTTF) to conduct a systematic review of NRC processes and regulations and to determine if the agency should make additional improvements to its regulatory system. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. NTTF Recommendation 2.1 for seismic hazards, as amended by the SRMs associated with SECY-11-0124 and SECY-11-0137, instructed the NRC staff to issue requests for information to licensees pursuant to Sections 161.c, 103.b, and 182.a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). This information request was for licensees under 10 CFR 50 to reevaluate the seismic hazards at their sites against present-day NRC requirements and guidance. Based upon this information, the NRC staff will determine whether additional regulatory actions are necessary (e.g., update the design basis and SSCs important to safety) to protect against the updated hazards. In developing Recommendation 2.1, the NTTF recognized that the state of knowledge of seismic hazard within the United States (U.S.) has evolved and the level of conservatism in the determination of the original seismic design bases should be reexamined.

The Electric Power Research Institute (EPRI) took the responsibility of developing new Ground Motion Response Spectra (GMRS) for each site in the industry. The new GMRS that was generated utilizes newly developed methodology.

EPRI, in conjunction with the Nuclear Energy Institute (NEI), developed the Seismic Evaluation Guidance (SPID) [Reference 2] for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic and the Template for the Seismic Hazard and Screening Reports for Central and Eastern United States (CEUS) Plants (Attachment 2).

Black & Veatch *Seismic Hazard and Screening* Report Revision 2, is accepted at CNS, is included as Attachment A to this Report. All comments have been resolved and no further changes are necessary.

5. Summary of Results

The results presented by Black and Veatch in *CNS Seismic Hazard and Screening Report* can be found in Attachment A. Discussion of the methodology used in the development of the Seismic Hazard and Screening Report is specifically addressed within EPRI Report 1025287 and will not be discussed in this report.

Review of Seismic Hazard and Screening Report resulted in comments that were resolved accordingly. No further review is necessary.

6. Conclusions and Recommendations

1. Black & Veatch Seismic Hazard and Screening Report Revision 2 Cooper Nuclear Station Report (Attachment A) is acceptable for adoption at CNS.

7. References

1. CNS Engineering Report 14-002 Revision 1, "LCI GMRS Report Acceptance"
2. EPRI Report 1025287 "Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic" Dated February 2013

8. Attachments

A. Black & Veatch "Seismic Hazard and Screening Report Cooper Nuclear Station – Final" Revision 2; December 19, 2014

B. EPRI Final Template for "Seismic Hazard and Screening Report (Example Submittal for CEUS Site)" February 25, 2014

ENGINEERING REPORT ER 2014-003

Attachment A

Black & Veatch

"Seismic Hazard and Screening Report Cooper Nuclear Station – Revision 2"
December 19, 2014

Nebraska Public Power District

Cooper Nuclear Station

Contract No. TA4700001660

Seismic Hazard and Screening Report

CLIENT APP.: NA

BLACK & VEATCH

Overland Park, KS

2	12/19/2014	Issued for Use (RAR-0003)	JPK	SES	DVR-0006, DVR-0007	ADB
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THIS DOCUMENT CONTAINS SAFETY-RELATED ITEMS
☐ YES ☒ NO

THIS DOCUMENT CONTAINS SEISMIC CATEGORY I ITEMS
☐ YES ☒ NO

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Seismic Hazard and Screening Report
Cooper Nuclear Station

December 19, 2014

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Acronym List

ADS	Automatic Depressurization System
BDB	Beyond Design Basis
BE	Best Estimate
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
CDFM	Conservative Deterministic Failure Margin
CEUS	Central and Eastern United States
CEUS SSC	Central and Eastern United States Seismic Source Characterization
CFR	Code of Federal Regulations
CNS	Cooper Nuclear Station
CPT	Cone Penetration Tests
CR	Condition Report
ECC_GC	Extended Continental Crust-Gulf Coast
EPRI	Electric Power Research Institute
ERM-N	Eastern Rift Margin Fault Northern Segment
ERM-S	Eastern Rift Margin Fault Southern Segment
FRS	Floor Response Spectra
FS	Factor of Safety
GMM	Ground Motion Model
GMRS	Ground Motion Response Spectrum
HCLPF	High Confidence of Low Probability of Failure
HPCI	High-Pressure Coolant Injection
HVAC	Heating, Ventilating, and Air Conditioning
IBEB	Illinois Basis Extended Basement
IHS	IPEEE HCLPF Spectrum
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
ISFSI	Independent Fuel Storage Installation
ISRS	In-Structure Response Spectra
LB	Lower Bound
MESE-N	Mesozoic and Younger Extended Prior – Narrow
MESE-W	Mesozoic and Younger Extended Prior - Wide
MIDC_A	Midcontinent-Craton Alternative A
MIDC_B	Midcontinent-Craton Alternative B
MIDC_C	Midcontinent-Craton Alternative C
MIDC_D	Midcontinent-Craton Alternative D
MM	Modified Mercalli
MSL	Mean Sea Level
NMESE-N	Non-Mesozoic and Younger Extended Prior - Narrow
NMESE-W	Non-Mesozoic and Younger Extended Prior - Wide
NMFS	New Madrid Fault System
NPPD	Nebraska Public Power District
NRC	Nuclear Regulatory Commission
NTTF	Near Term Task Force

OBE	Operating Basis Earthquake
OKA	Oklahoma Aulacogen
PGA	Peak Ground Acceleration
PSA	Probabilistic Safety Assessment
PSHA	Probabilistic Seismic Hazard Analysis
RAI	Request for Additional Information
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RLE	Review Level Earthquake
RLME	Repeated Large Magnitude Earthquake
RR	Reelfoot Rift
RR-RCG	Reelfoot Rift Including the Rough Creek Graben
RVT	Random Vibration Theory
SBLOCA	Small Break Loss of Coolant Accident
SEL	Seismic Equipment List
SER	Staff Evaluation Report
SEWS	Screening Evaluation Worksheets
SMA	Seismic Margin Assessment
SME	Seismic Margin Earthquake
SMM	Seismic Margin Methodology
SPID	Screening, Prioritization, and Implementation Details
SPLD	Success Path Logic Diagrams
SPRA	Seismic Probabilistic Risk Assessment
SPT	Standard Penetration Test
SQUG GIP	Seismic Qualification Utility Group Generic Implementation Procedure
SRT	Seismic Review Team
SRV	Safety Relief Valve
SSC	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
SSEL	Safe Shutdown Equipment List
SSI	Soil-Structure Interaction
STUDY_R	Study Region
SWEL	Seismic Walkdown Equipment List
SWC	Seismic Walkdown Checklist
TER	Technical Evaluation Report
UB	Upper Bound
UHRS	Uniform Hazard Response Spectra
USAR	Updated Safety Analysis Report
USC&GS	United States Coast & Geodetic Survey
USGS	United States Geological Survey
USI	Unresolved Safety Issue

1.0 Introduction

Following the accident at the Fukushima Daiichi nuclear power plant resulting from the March 11, 2011, Great Tohoku Earthquake and subsequent tsunami, the Nuclear Regulatory Commission (NRC) established a Near Term Task Force (NTTF) to conduct a systematic review of NRC processes and regulations. The NTTF was also tasked with determining whether the NRC should make additional improvements to its regulatory system. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. Subsequently, the NRC issued a 50.54(f) letter[1] that requests information to assure these recommendations are addressed by all U.S. nuclear plants. The 50.54(f) letter[1] requests that licensees and holders of construction permits under Title 10 Code of Federal Regulations (CFR) Part 50 reevaluate the seismic hazards at their sites against present-day NRC requirements and guidance. Depending on the comparison between the reevaluated seismic hazard and the current design basis, the result is either no further risk evaluation or the performance of a seismic risk assessment. Risk assessment approaches that are acceptable to the staff include a seismic probabilistic risk assessment (SPRA), or a seismic margin assessment (SMA). Based upon the risk assessment results, the NRC staff will determine whether additional regulatory actions are necessary.

This report provides the information requested in items (1) through (7) of the "Requested Information" section and Attachment 1 of the 50.54(f) letter[1] pertaining to NTTF Recommendation 2.1: Seismic for the Cooper Nuclear Station (CNS), located in Nemaha County, Nebraska. In providing this information, Nebraska Public Power District (NPPD) followed the guidance provided in the *Seismic Evaluation Guidance: Screening, Prioritization, and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic*[3]. The Augmented Approach, *Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic*[17], has been developed as the process for evaluating critical plant equipment as an interim action to demonstrate additional plant safety margin, prior to performing the complete plant seismic risk evaluations.

The original geologic and seismic siting investigations for CNS were performed in accordance with Appendix A to 10 CFR Part 100 as it existed prior to the construction permits. To the extent discussed in the Updated Safety Analysis Report (USAR), CNS meets the General Design Criterion 2 in Appendix A to 10 CFR Part 50 which was not part of the original licensing basis. The Safe Shutdown Earthquake (SSE) Ground Motion was subsequently evaluated against criteria in Appendix A to 10 CFR Part 100 and found to be acceptable. This SSE was used for the design of seismic Category I structures, systems and components (SSC).

In response to the 50.54(f) letter[1] and following the guidance provided in the SPID[3], a seismic hazard reevaluation was performed. For screening purposes a Ground Motion Response Spectrum (GMRS) was developed. The GMRS is documented in CNS Engineering Report ER-2014-002[15].

Based on the results of the screening evaluation, the plant screens out of a risk evaluation, but screens in for a spent fuel pool evaluation and high frequency confirmation.

Revision 2 incorporates the following changes. Individual tracking of revisions is not provided due to the extensive revisions incorporated.

- Updated GMRS Information from Reference 15
- Added Appendix B, IPEEE Adequacy Review
- Added Appendix C, Soil Failure and Liquefaction Evaluation for IPEEE Adequacy Review

2.0 Seismic Hazard Reevaluation

Section II-5 of the CNS USAR[4] contains the following description of the site:

> "Cooper Nuclear Station (CNS) is located in Nemaha County, Southeastern Nebraska, on the west bank of the Missouri River. It is situated on the first bottomland of the broad, nearly level, flood plain which is approximately six miles wide at the site. The natural relief is about ten feet."

Section II-5 of the CNS USAR[4] contains the following description of seismicity:

> "The earthquakes most significant for the evaluation of the seismicity of the site are the New Madrid earthquakes of 1811 and 1812; the Lincoln, Nebraska, earthquake of 1877; the Tecumseh, Nebraska, earthquake of 1935; and the El Reno, Oklahoma, earthquake of 1952. On the basis of the historical earthquake records, it is concluded that:
>
> - There is a reasonable chance that during the life of the nuclear power station, earthquakes would affect the site with an intensity Modified Mercalli (MM) VII.
> - The hypothetical maximum possible intensity of ground motion at the site would result from a local earthquake smaller than the New Madrid earthquakes of 1811 and 1812."
>
> "Small slips appear to occur along the Humboldt Fault and many of the regional earthquakes had epicenters in the vicinity of the Nemaha Anticline and Humboldt Fault. However, important displacements of the Humboldt Fault have not occurred for 200 million years and it is improbable that future earthquakes with epicenters located in the vicinity of the Humboldt Fault will have epicentral intensities greater than MM VII."
>
> "There is no evidence at the site of either a fault or other bedrock discontinuity which would tend to increase the seismicity of the site as compared to nearby sites."

2.1 Regional and Local Geology

2.1.1 Regional Geology

Section II-5 of the CNS USAR[4] contains the following description of the regional geology:

> "The principal geologic strata in the region in order of increasing depth are soil deposits, sedimentary rocks, and deep basement igneous rocks. The soil deposits consist of loess and till in the uplands, and either stratified or heterogeneous alluvium in the flood plains. Thickness of deposits varies from a few feet to about 100 feet for loess, none to several feet for till, and less than 10 feet to more than 100 feet for alluvium. The rock strata are gently dipping sedimentary rocks mainly Paleozoic in age. Alternating beds of shale, limestone, sandstone, and occasional thin beds of coal are present. The total thickness varies from over 3,500 feet near the site to about 500 feet, 30 miles west. The deep basement igneous rocks are Precambrian in origin, chiefly primary granite or granitoid rocks."
>
> "The major geologic structures in the region are the Nemaha Anticline, Forest City Basin, Humboldt Fault, and Thurman-Wilson Fault. Except for the Forest City Basin, none of these structures is in the immediate vicinity of the site. The closest one, 20 miles to the west, is the Nemaha Anticline and its associated Humboldt Fault."

"The Nemaha Anticline is a major structural feature of the midcontinent which separates two depositional basins, the Forest City Basin on its east flank and the northern extension of the Salina Basin on the west. It is a sharp uplift of Precambrian granite. The anticline is believed to have first come into existence by folding and faulting at the close of the Proterozoic. Its development of near orogenic proportions occurred near the end of the Mississippian and continued through Pennsylvanian into early Permian. By early Permian, major tectonic movements appear to have ceased. The anticline trends southward from Omaha, through Nebraska, across Kansas, and into northern Oklahoma. The crest of the buried mountain range is irregular; its depth below ground surface varies from 400 feet at the Nebraska-Kansas line to 3,000 feet at the Kansas-Oklahoma line. The anticline has a very steep eastern front which is faulted in several areas. The most notable fault is the Humboldt Fault, principally a normal fault striking in a general north-south direction. Vertical displacement of 1,000 to 1,500 feet in Nebraska and in the vicinity of Nebraska City, Nebraska, are reported."

"The Forest City Basin underlies the site. Its basinal axis in Nebraska lies close to and roughly parallels the Nemaha Anticline on the east. Its west flank shares a common front with the steep eastern flank of the Nemaha Anticline."

"The Thurman-Wilson Fault is associated with the Redfield Anticline which strikes southwest from approximately Des Moines, Iowa, toward Lincoln, Nebraska. The fault is about 40 miles north of the site and is located south of the crest of the anticlinal axis. The fault has a southward displacement of about ten feet."

2.1.2 Local Geology

Section II-5 of the CNS USAR[4] contains the following description of the local geology:

"Locally, the stratigraphy is best represented by a section through the bluffs along the western boundary of the site. It shows Peorian loess, Kansas till, limestone and shale of the Permian system, and limestone, shale, sandstone, and occasional thin beds of coal of the Pennsylvanian system. The contact between the two systems is unconformable and occurs in the bluff at approximately elevation 930 feet mean sea level (MSL)."

"Detailed classification of rock cores obtained in borings at the site show excellent correlation with published regional stratigraphic columns in both sequence and thickness."

"The geologic structures occurring within the rocks at the site are minor. Field observations suggest the possibility of minor plains-type folding resulting from differential compaction of underlying sediments. No faults have been found at the site or in the local area, nor are any known of or suspected."

"Locally, three principal types of soils are found, each of different geologic origin; loess and till in the bluffs and alluvial and glacial deposits in the flood plains."

"The loess are wind-blown silts. The topography of the loess reflects the surface configuration of the underlying till or rock. Its ability to maintain steep faces is responsible for the near vertical slopes in the upper portion of the bluffs. The Kansan till underlies the loess. It is a heterogeneous mixture of clay, silt, sand, gravel, cobble, and boulder, and is five to ten feet thick. In an unleached and unoxidized condition, it is commonly a dark gray silty clay which contains erratics and locally derived cobbles and boulders. Sand lenses are distributed throughout the deposit. Complete removal of calcareous minerals in the upper limits of the till produces the highly tenacious gumbotil."

"The alluvial deposits in the flood plain at the site vary in thickness from 62 to 71 feet. Two major subtypes of different geologic origin are present; the surficial fine-grained soils and the underlying sands."

"The surficial fine-grained soils are recent alluvial deposits derived from the meandering Missouri River. Evidences of the meander were analyzed by a stereoscopic study of aerial photographs. The surficial soils consist of meander-belt and back-swamp deposits, ranging in thickness from 10 to 25 feet. For the most part, these deposits are silty sand, sandy silt, silty clay, and clay, and may be encountered in localized pockets or in complex combinations."

"The underlying sands appear to be either fluvial or glacial outwash deposits or both. The amount of silt and clay size particles is generally small. They grade from fine to coarse with increasing depth. Lenses of clay, coarse sand, and fine gravel are distributed irregularly throughout the deposit."

2.2 Probabilistic Seismic Hazard Analysis

2.2.1 Probabilistic Seismic Hazard Analysis Results

In accordance with the 50.54(f) letter[1] and following the guidance in the SPID[3], a probabilistic seismic hazard analysis (PSHA) was completed using the recently developed Central and Eastern United States Seismic Source Characterization (CEUS SSC) for Nuclear Facilities[5] together with the updated Electric Power Research Institute (EPRI) Ground Motion Model (GMM) for the CEUS[6]. For the PSHA, a lower-bound moment magnitude of 5.0 was used, as specified in the 50.54(f) letter[1].

For the PSHA, the CEUS SSC background seismic sources out to a distance of 400 miles (640 km) around CNS were included. This distance exceeds the 200 mile (320 km) recommendation contained in Regulatory Guide 1.208[7] and was chosen for completeness. Background sources included in this site analysis are the following:

1. Extended Continental Crust-Gulf Coast (ECC_GC)
2. Illinois Basin Extended Basement (IBEB)
3. Mesozoic and younger extended prior – narrow (MESE-N)
4. Mesozoic and younger extended prior – wide (MESE-W)
5. Midcontinent-Craton alternative A (MIDC_A)
6. Midcontinent-Craton alternative B (MIDC_B)
7. Midcontinent-Craton alternative C (MIDC_C)
8. Midcontinent-Craton alternative D (MIDC_D)
9. Non-Mesozoic and younger extended prior – narrow (NMESE-N)
10. Non-Mesozoic and younger extended prior – wide (NMESE-W)
11. Oklahoma Aulacogen (OKA)
12. Reelfoot Rift (RR)
13. Reelfoot Rift including the Rough Creek Graben (RR-RCG)
14. Study region (STUDY_R)

For sources of large magnitude earthquakes, designated repeated large magnitude earthquake (RLME) sources in CEUS SSC[5], the following sources lie within 621 miles (1,000 km) of the site and were included in the analysis:

1. Cheraw
2. Commerce
3. Eastern Rift Margin Fault northern segment (ERM-N)
4. Eastern Rift Margin Fault southern segment (ERM-S)
5. Marianna
6. Meers
7. New Madrid Fault System (NMFS)
8. Wabash Valley

For each of the above background and RLME sources, the mid-continent version of the updated CEUS EPRI GMM[6] was used.

2.2.2 Base Rock Seismic Hazard Curves

Consistent with the SPID[3] Subsection 2.5.3, base rock seismic hazard curves are not provided because the site amplification approach referred to as Method 3 has been used. Seismic hazard curves are shown in Section 2.3.7 at the SSE control point elevation (869.5 feet), which is the base of the Control Building.

2.3 Site Response Evaluation

Following the guidance contained in Seismic Enclosure 1 of the March 12, 2012, 50.54(f) letter[1] and in the SPID[3] for nuclear power plant sites that are not sited on hard rock (defined as shear wave velocity of 9,300 feet per second [2.83 km/sec]), a site response analysis was performed for CNS and is documented in CNS Engineering Report ER-2014-002[15].

2.3.1 Description of Subsurface Material

CNS is located in Nemaha County, Southeastern Nebraska on the west bank of the Missouri River. It is situated on the first bottomland of the broad, nearly level, flood plain which is about 6 miles (20 km) wide at the site. The basic information used to create the site geologic profile at CNS is shown in Table 2.3.1-1a (for shallow stratigraphy) and Table 2.3.1-1b (for deep stratigraphy). This profile was developed using information documented in EPRI Data Request Report[8] and CNS Engineering Report ER-2014-002[15]. As indicated in EPRI Data Request Report[8], the SSE control point is defined at elevation 869.5 feet, and the profile was modeled up to this elevation. The profile consists of about 49.5 feet (15 m) of fill and compacted alluvium overlying about 3,450 feet (1,052 m) of firm sedimentary rock. Precambrian basement rock is estimated to be at a depth of about 3,500 feet (1,067 m).

Table 2.3.1-1b provides elevations for the four deepest bedrock stratigraphic units – Silurian, Ordovician, Cambrian, and Precambrian.

The Precambrian basement rock in Table 2.3.1-1b is estimated to be approximately 700 feet (213.4 m) higher than the elevation provided previously in NPPD letter[12] to the NRC. This difference is due to variations in the interpretation of the regional geology near the site. However, a

Precambrian basement rock depth of about 3,500 feet (1,067 m) is consistent with the thickness of Paleozoic sedimentary rocks reported in USAR Section II-5.1[4] and by EPRI Data Request Report[8] and CNS Engineering Report ER-2014-002[15]. There are no differences between Tables 2.3.1-1a or the three shear wave velocity profiles (Table 2.3.2-2 and Figure 2.3.2-1) and the corresponding tables and figure presented previously in NPPD letter[12] to the NRC.

Table 2.3.1-1a Geologic Profile for Estimated Layer Thicknesses for CNS - Shallow Profile

Depth Range (feet)	Elevation (feet above MSL)	Soil/Rock Description	Density (pcf)	Shear Wave Velocity (fps)	Compressional Wave Velocity (fps)	Poisson's Ratio
0-5	902/903-898	Type I or Type II Fill	134	600	1600	0.27
5-8	898-895	Type I or Type II Fill	134	750	1600	0.27
8-13	895-890	Type I or Type II Fill	134	750	1600	0.27
13-23	890-880	Type I Fill/In-Situ Compacted Alluvium	134	850	1600	0.27
23-30	880-873	Type I Fill/In-Situ Compacted Alluvium	134	920	3295	0.42
30-33	873-870	Type I Fill/In-Situ Compacted Alluvium	133	920	5505	0.48
33-48	870-855	Type I Fill/In-Situ Compacted Alluvium	133	1020	5505	0.48
48-58	855-845	Type I Fill/In-Situ Compacted Alluvium	133	1030	5505	0.48
58-68	845-835	Type I Fill/In-Situ Compacted Alluvium	133	1040	5505	0.48
68-74	835-829	Type I Fill/In-Situ Compacted Alluvium	132	1040	2535	0.38
74-83	829-820	Type I Fill/In-Situ Compacted Alluvium	132	1120	6100	0.48
83-93	820-810	Soft Bedrock	140	1620	6420	0.47
93-118	810-785	Soft Bedrock	140	1760	6600	0.47
118-128	785-775	Harder Bedrock	160	2750	9970	0.45
>128	<775	Per Table B.5 NEDC 13-019	---	---	---	---

*From Table B.1 of NEDC 13-019[9]

Table 2.3.1-1b Geologic Profile for Estimated Layer Thicknesses for CNS – Deep Bedrock Stratigraphy

Elevation of Bottom of Unit (feet, MSL)	System	Series	Group(s)	Rock Types
600	Pennsylvanian	Virgil	Wabaunsee	Shale, Limestone, Sandstone, Coal
300	Pennsylvanian	Virgil	Shawnee	Limestone, Shale
150	Pennsylvanian	Virgil	Douglas	Shale, Sandstone, Limestone
100	Pennsylvanian	Missouri	Lansing	Limestone, Shale
-100	Pennsylvanian	Missouri	Kansas City	Shale, Limestone
-150	Pennsylvanian	Missouri	Pleasanton	Limestone, Shale
-350	Pennsylvanian	Missouri, Des Moines	Marmaton	Shale, Limestone, Coal
-1050	Pennsylvanian, Mississippian	Des Moines	Cherokee	Shale, Coal, Sandstone
-1350	Mississippian	-	Meramec, Osage, Kinderhook	Limestone, Chert, Shale
-1750	Devonian	-	-	Shale, Limestone
-2150	Silurian	-	-	Dolomite
unknown	Ordovician	-	Maquoketa, Galena (Viola), Decorah-Platteville, St. Peter, Oneoto (Up. Arbuckle)	Shale, Dolomite, Limestone, Sandstone
-2600 *(3500 ft deep)*	Cambrian	-	Bonneterre (Lr. Arbuckle), La Motte	Sandstone, Shale, Glauconite, Granite
Unknown	Precambrian	-	-	Metamorphic, Granite

Notes:
1. Elevations, systems, series, and groups were interpreted from USAR Figure II-5-3[4].
2. Elevations are in feet and were rounded to the nearest 50 feet.
3. Rock types are from Nebraska Geologic Survey Paper[11].

2.3.2 Development of Base Case Profiles and Nonlinear Material Properties

Table 2.3.1-1a shows the recommended shear wave velocities and unit weights along with elevations and corresponding stratigraphy. As indicated in EPRI Data Request Report[8] and CNS Engineering Report ER-2014-002[15], the SSE control point is at elevation 869.5 feet (2655 m) within Type I fill/in-situ compacted alluvium.

The source of shear-wave velocity measurements shown in Table 2.3.1-1a is unclear and is likely based on measured compressional–wave velocities and assumed Poisson's ratios. For the firm rock below a depth of 128 feet (39 m), 97 feet (30 m) below the SSE control point, a previously recommended estimate of shear-wave velocity is 7,292 feet per second (2,222 m/s) as defined in the EPRI Data Request Report[8].

The mean base case profile (P1) was based on the recommended densities and shear wave velocities listed in Table 2.3.1-1a along with a shear wave velocity of 7,292 ft/s (2,222 m/s) for the underlying firm rock. Lower-range (P2) and upper-range (P3) profiles were developed with scale factors of 1.25 for the top 49.5 feet (15 m) and 1.57 below to reflect increased epistemic uncertainty for assumed shear wave velocities. The scale factors of 1.25 and 1.57 reflect a $\sigma_{\mu ln}$ of about 0.2 and 0.35, respectively, based on the SPID[3] 10th and 90th fractiles, which implies a scale factor of 1.28 on $\sigma_{\mu ln}$. Depth to Precambrian basement was taken at 3,500 feet (1,067 m) randomized ± 1,050 feet (320 m). Profile P3, the stiffest profile, encountered hard rock shear wave velocities (9,285 ft/s, 2,890 m/s) at a depth below the SSE control point of about 97 feet (30 m). The three shear wave velocity profiles are shown on Figure 2.3.2-1 and listed in Table 2.3.2-2.

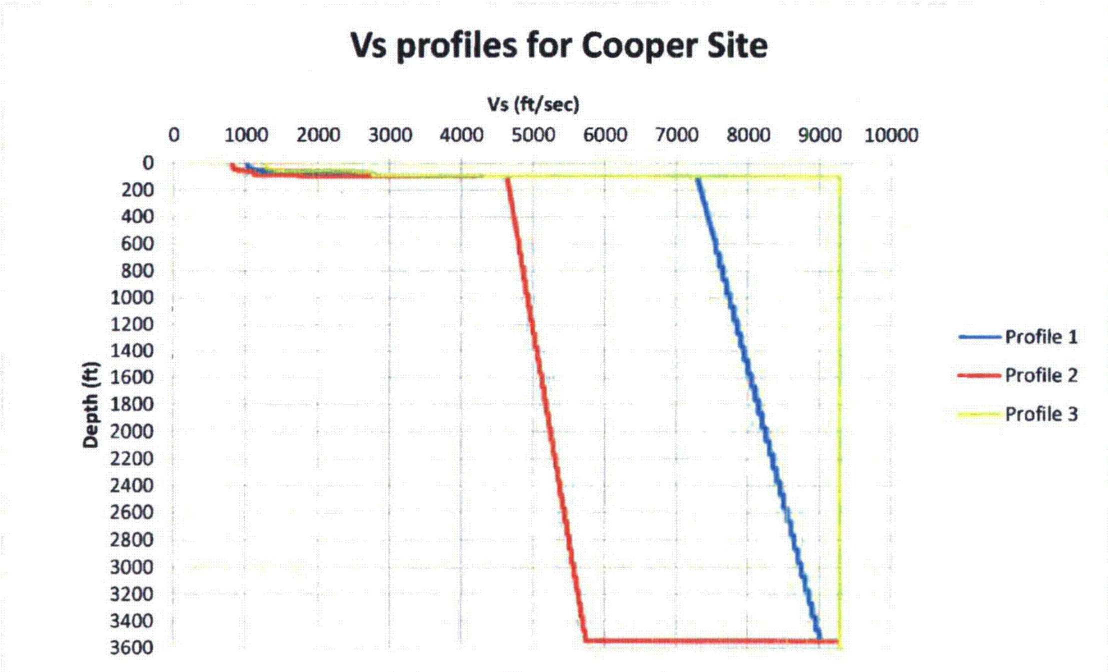

Figure 2.3.2-1 Shear Wave Velocity Profile Used in Site Response Calculations for CNS

Table 2.3.2-2 Geologic Profile and Estimated Layer Thickness for CNS

Profile 1			Profile 2			Profile 3		
Thickness (ft)	Depth (ft)	Vs (ft/s)	Thickness (ft)	Depth (ft)	Vs (ft/s)	Thickness (ft)	Depth (ft)	Vs (ft/s)
	0	1020		0	816		0	1275
10.0	10.0	1020	10.0	10.0	816	10.0	10.0	1275
4.5	14.5	1020	4.5	14.5	816	4.5	14.5	1275
10.0	24.5	1030	10.0	24.5	824	10.0	24.5	1288
10.0	34.5	1040	10.0	34.5	832	10.0	34.5	1300
6.0	40.5	1040	6.0	40.5	832	6.0	40.5	1300
9.0	49.5	1120	9.0	49.5	896	9.0	49.5	1400
10.0	59.5	1620	10.0	59.5	1032	10.0	59.5	2543
10.0	69.5	1760	10.0	69.5	1121	10.0	69.5	2763
10.0	79.5	1760	10.0	79.5	1121	10.0	79.5	2763
5.0	84.5	1760	5.0	84.5	1121	5.0	84.5	2763
10.0	94.5	2750	10.0	94.5	1752	10.0	94.5	4318
2.5	97.0	7292	2.5	97.0	4645	2.5	97.0	9285
10.0	107.0	7294	10.0	107.0	4647	10.0	107.0	9285
10.0	117.0	7299	10.0	117.0	4650	10.0	117.0	9285
10.0	127.0	7304	10.0	127.0	4653	10.0	127.0	9285
10.0	137.0	7309	10.0	137.0	4656	10.0	137.0	9285
10.0	147.0	7314	10.0	147.0	4659	10.0	147.0	9285
10.0	157.0	7319	10.0	157.0	4662	10.0	157.0	9285
10.0	167.0	7324	10.0	167.0	4666	10.0	167.0	9285
10.0	177.0	7329	10.0	177.0	4669	10.0	177.0	9285
10.0	187.0	7334	10.0	187.0	4672	10.0	187.0	9285
10.0	197.0	7339	10.0	197.0	4675	10.0	197.0	9285
10.0	207.0	7344	10.0	207.0	4678	10.0	207.0	9285
10.0	217.0	7349	10.0	217.0	4682	10.0	217.0	9285
10.0	227.0	7354	10.0	227.0	4685	10.0	227.0	9285
10.0	237.0	7359	10.0	237.0	4688	10.0	237.0	9285
10.0	247.0	7364	10.0	247.0	4691	10.0	247.0	9285
10.0	257.0	7369	10.0	257.0	4694	10.0	257.0	9285
10.0	267.0	7374	10.0	267.0	4698	10.0	267.0	9285
10.0	277.0	7379	10.0	277.0	4701	10.0	277.0	9285
10.0	287.0	7384	10.0	287.0	4704	10.0	287.0	9285
10.0	297.0	7389	10.0	297.0	4707	10.0	297.0	9285
10.0	307.0	7394	10.0	307.0	4710	10.0	307.0	9285
10.0	317.0	7399	10.0	317.0	4713	10.0	317.0	9285
10.0	327.0	7404	10.0	327.0	4717	10.0	327.0	9285
10.0	337.0	7409	10.0	337.0	4720	10.0	337.0	9285
10.0	347.0	7414	10.0	347.0	4723	10.0	347.0	9285

Table 2.3.2-2 (continued)

Profile 1			Profile 2			Profile 3		
Thickness (ft)	Depth (ft)	Vs (ft/s)	Thickness (ft)	Depth (ft)	Vs (ft/s)	Thickness (ft)	Depth (ft)	Vs (ft/s)
10.0	357.0	7419	10.0	357.0	4726	10.0	357.0	9285
10.0	367.0	7424	10.0	367.0	4729	10.0	367.0	9285
10.0	377.0	7429	10.0	377.0	4733	10.0	377.0	9285
10.0	387.0	7434	10.0	387.0	4736	10.0	387.0	9285
10.0	397.0	7439	10.0	397.0	4739	10.0	397.0	9285
10.0	407.0	7444	10.0	407.0	4742	10.0	407.0	9285
10.0	417.0	7449	10.0	417.0	4745	10.0	417.0	9285
10.0	427.0	7454	10.0	427.0	4748	10.0	427.0	9285
10.0	437.0	7459	10.0	437.0	4752	10.0	437.0	9285
10.0	447.0	7464	10.0	447.0	4755	10.0	447.0	9285
10.0	457.0	7469	10.0	457.0	4758	10.0	457.0	9285
10.0	467.0	7474	10.0	467.0	4761	10.0	467.0	9285
10.0	477.0	7479	10.0	477.0	4764	10.0	477.0	9285
10.0	487.0	7484	10.0	487.0	4768	10.0	487.0	9285
10.0	497.0	7489	10.0	497.0	4771	10.0	497.0	9285
10.0	507.0	7494	10.0	507.0	4774	10.0	507.0	9285
10.0	517.0	7499	10.0	517.0	4777	10.0	517.0	9285
10.0	527.0	7504	10.0	527.0	4780	10.0	527.0	9285
10.0	537.0	7509	10.0	537.0	4784	10.0	537.0	9285
10.0	547.0	7514	10.0	547.0	4787	10.0	547.0	9285
10.0	557.0	7519	10.0	557.0	4790	10.0	557.0	9285
10.0	567.0	7524	10.0	567.0	4793	10.0	567.0	9285
100.0	667.0	7549	100.0	667.0	4809	100.0	667.0	9285
100.0	767.0	7599	100.0	767.0	4841	100.0	767.0	9285
100.0	867.0	7649	100.0	867.0	4873	100.0	867.0	9285
100.0	967.0	7699	100.0	967.0	4905	100.0	967.0	9285
100.0	1067.0	7749	100.0	1067.0	4936	100.0	1067.0	9285
100.0	1167.0	7799	100.0	1167.0	4968	100.0	1167.0	9285
100.0	1266.9	7849	100.0	1266.9	5000	100.0	1266.9	9285
100.0	1366.9	7899	100.0	1366.9	5032	100.0	1366.9	9285
100.0	1466.9	7949	100.0	1466.9	5064	100.0	1466.9	9285
100.0	1566.9	7999	100.0	1566.9	5096	100.0	1566.9	9285
100.0	1666.9	8049	100.0	1666.9	5127	100.0	1666.9	9285
100.0	1766.9	8099	100.0	1766.9	5159	100.0	1766.9	9285
100.0	1866.9	8149	100.0	1866.9	5191	100.0	1866.9	9285
100.0	1966.9	8199	100.0	1966.9	5223	100.0	1966.9	9285
100.0	2066.9	8249	100.0	2066.9	5255	100.0	2066.9	9285
100.0	2166.9	8299	100.0	2166.9	5287	100.0	2166.9	9285

Table 2.3.2-2 (continued)

Profile 1			Profile 2			Profile 3		
Thickness (ft)	Depth (ft)	Vs (ft/s)	Thickness (ft)	Depth (ft)	Vs (ft/s)	Thickness (ft)	Depth (ft)	Vs (ft/s)
100.0	2266.9	8349	100.0	2266.9	5319	100.0	2266.9	9285
100.0	2366.9	8399	100.0	2366.9	5350	100.0	2366.9	9285
100.0	2466.9	8449	100.0	2466.9	5382	100.0	2466.9	9285
100.0	2566.9	8499	100.0	2566.9	5414	100.0	2566.9	9285
100.0	2666.9	8549	100.0	2666.9	5446	100.0	2666.9	9285
100.0	2766.9	8599	100.0	2766.9	5478	100.0	2766.9	9285
100.0	2866.9	8649	100.0	2866.9	5510	100.0	2866.9	9285
100.0	2966.9	8699	100.0	2966.9	5541	100.0	2966.9	9285
100.0	3066.9	8749	100.0	3066.9	5573	100.0	3066.9	9285
100.0	3166.9	8799	100.0	3166.9	5605	100.0	3166.9	9285
100.0	3266.8	8849	100.0	3266.8	5637	100.0	3266.8	9285
100.0	3366.8	8899	100.0	3366.8	5669	100.0	3366.8	9285
100.0	3466.8	8949	100.0	3466.8	5701	100.0	3466.8	9285
80.2	3547.1	8999	80.2	3547.1	5733	80.2	3547.1	9285
3280.8	6827.9	9285	3280.8	6827.9	9285	3280.8	6827.9	9285

2.3.2.1 Shear Modulus and Damping Curves

Recent nonlinear dynamic material properties were not available for the CNS soils and sedimentary rocks. To accommodate epistemic uncertainty in nonlinear dynamic material properties for the soils, two sets of shear modulus reduction and hysteretic damping curves were used. The rock material over the upper 500 feet (150 m) was assumed to have behavior that could be modeled as either linear or nonlinear. To represent this potential for either case in the upper 500 feet (150 m) of sedimentary rock at the CNS site, two sets of shear modulus reduction and hysteretic damping curves were used. Consistent with the SPID[3], the EPRI soil and rock curves (model M1) were considered to be appropriate to represent the upper range nonlinearity likely in the materials at this site and linear analyses for firm rock along with Peninsular Range curves for soils (model M2) was assumed to represent an equally plausible alternative soil and firm rock response across loading level. For the linear analyses, the low strain damping from the EPRI rock curves were used as the constant damping values in the upper 500 feet (150 m).

2.3.2.2 Kappa

Base case kappa estimates were determined using Section B-5.1.3.1 of the SPID[3] for a firm CEUS rock site. Kappa for a firm rock site with at least 3,000 feet (1 km) of sedimentary rock may be estimated from the average S-wave velocity over the upper 100 feet (V_{s100}) of the subsurface profile while for a site with less than 3,000 feet (1 km) of firm rock, kappa may be estimated with a Q_s of 40 below 500 feet (150 m) combined with the low strain damping from the EPRI rock curves and an additional kappa of 0.006s for the underlying hard rock.

For the CNS site, with about 50 feet (15 m) of soils overlying about 3,450 feet (1,052 m) of firm rock, kappa was estimated with the low strain damping over the top 500 feet (150 m) combined with a Q_S of 40 below and 0.006s for the underlying hard rock. The resulting kappa values were 0.021s, 0.030s, and 0.008s for base case profiles P1, P2, and P3, respectively. The kappa values are shown in Table 2.3.2-3.

Table 2.3.2-3 Kappa Values and Weights Used for Site Response Analyses

Velocity Profile	Kappa(s)
P1	0.021
P2	0.030
P3	0.008
	Weights
P1	0.4
P2	0.3
P3	0.3
G/G_{max} and Hysteretic Damping Curves	
M1	0.5
M2	0.5

2.3.3 Randomization of Base Case Profiles

To account for the aleatory variability in dynamic material properties that is expected to occur across a site at the scale of a typical nuclear facility, variability in the assumed shear wave velocity profiles has been incorporated in the site response calculations.

For the CNS site, randomized shear wave velocity profiles were developed from the base case profiles shown on Figure 2.3.2-1. Consistent with the discussion in Appendix B of the SPID[3], the velocity randomization procedure made use of random field models that describe the statistical correlation between layering and shear wave velocity. The default randomization parameters developed in a report[10] submitted to Brookhaven National Laboratory (BNL) for USGS "A" site conditions were used for this site. Thirty random velocity profiles were generated for each base case profile. These random velocity profiles were generated using a natural log standard deviation of 0.25 over the upper 50 feet (15 m) and 0.15 below that depth. As specified in the SPID[3], correlation of shear wave velocity between layers was modeled using the footprint correlation model. In the correlation model, a limit of ±2 standard deviations about the median value in each layer was assumed for the limits on random velocity fluctuations.

2.3.4 Input Spectra

Consistent with the guidance in Appendix B of the SPID[3], input Fourier amplitude spectra were defined for a single representative earthquake magnitude (**M** 6.5) using two different assumptions regarding the shape of the seismic source spectrum (single-corner and double-corner). A range of 11 different input amplitudes (median peak ground acceleration [PGA] ranging from 0.01g to 1.5g) were used in the site response analyses. The characteristics of the seismic source and upper crustal attenuation properties assumed for the analysis of the CNS site were the same as those identified in Tables B-4, B-5, B-6 and B-7 of the SPID[3] as appropriate for typical CEUS sites.

2.3.5 Methodology

To perform the site response analyses for the CNS site, a random vibration theory (RVT) approach was employed. This process utilizes a simple, efficient approach for computing site-specific amplification functions and is consistent with existing NRC guidance and the SPID[3]. The guidance contained in Appendix B of the SPID[3] on incorporating epistemic uncertainty in shear wave velocities, kappa, nonlinear dynamic properties and source spectra for plants with limited at-site information was followed for the CNS site.

2.3.6 Amplification Functions

The results of the site response analysis consist of amplification factors (5 percent damped pseudo absolute response spectra) that describe the amplification (or de-amplification) of hard reference rock motion as a function of frequency and input reference rock amplitude. The amplification factors are represented in terms of a median amplification value and an associated standard deviation (sigma) for each oscillator frequency and input rock amplitude. Consistent with the SPID[3], a minimum median amplification value of 0.5 was employed in the present analysis. Figure 2.3.6-1 illustrates the median and ±1 standard deviation in the predicted amplification factors developed for the eleven loading levels parameterized by the median reference (hard rock) peak acceleration (0.01g to 1.50g) for profile P1 and EPRI rock G/G_{max} and hysteretic damping curves. The variability in the amplification factors results from variability in shear wave velocity, depth to hard rock, and modulus reduction and hysteretic damping curves. To illustrate the effects of nonlinearity at the CNS firm rock site, Figure 2.3.6-2 shows the corresponding amplification factors developed with linear site response analyses (model M2). Tabulated values of the amplification factors are provided in Appendix A. Between the linear and nonlinear (equivalent-linear) analyses, Figures 2.3.6-1 and Figure 2.3.6-2 respectively show only a minor difference for frequencies below about 20 Hz and the 0.5g loading level and below. Above about the 0.5g loading level, the differences increase significantly.

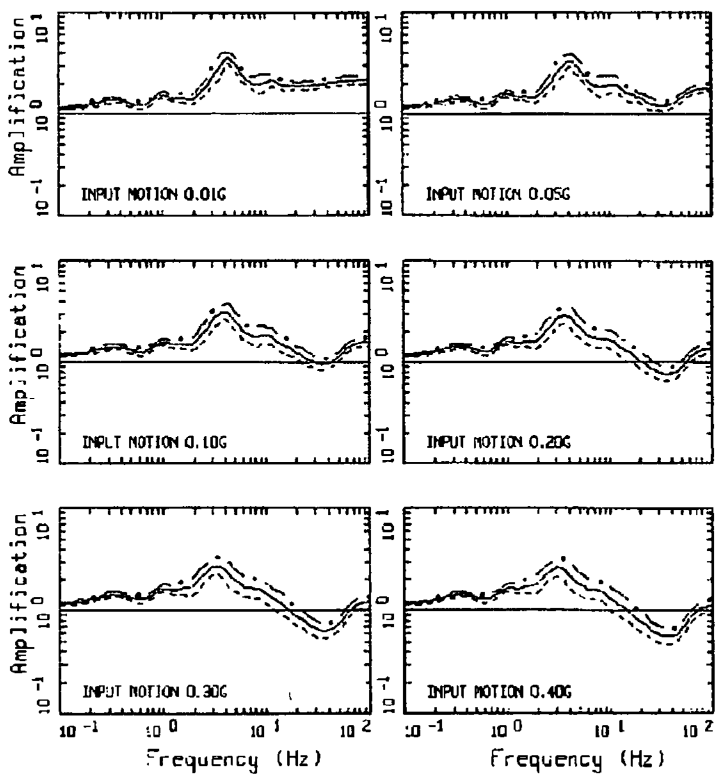

Figure 2.3.6-1 **Example Suite of Amplification Factors (5 percent damping pseudo absolute acceleration spectra) developed for the mean base case profile (P1), EPRI rock modulus reduction and hysteretic damping curves (model M1), and base-case kappa (K1) at 11 loading levels of hard rock median peak acceleration values from 0.01g to 1.50g. M 6.5 and single-corner source model per SPID[3].**

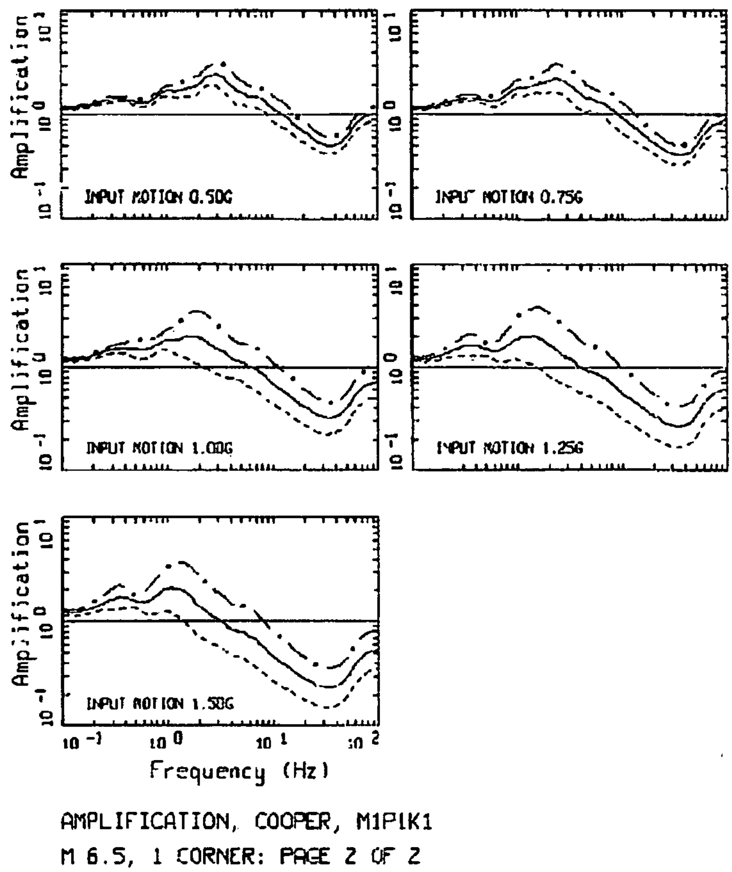

Figure 2.3.6-1 (continued)

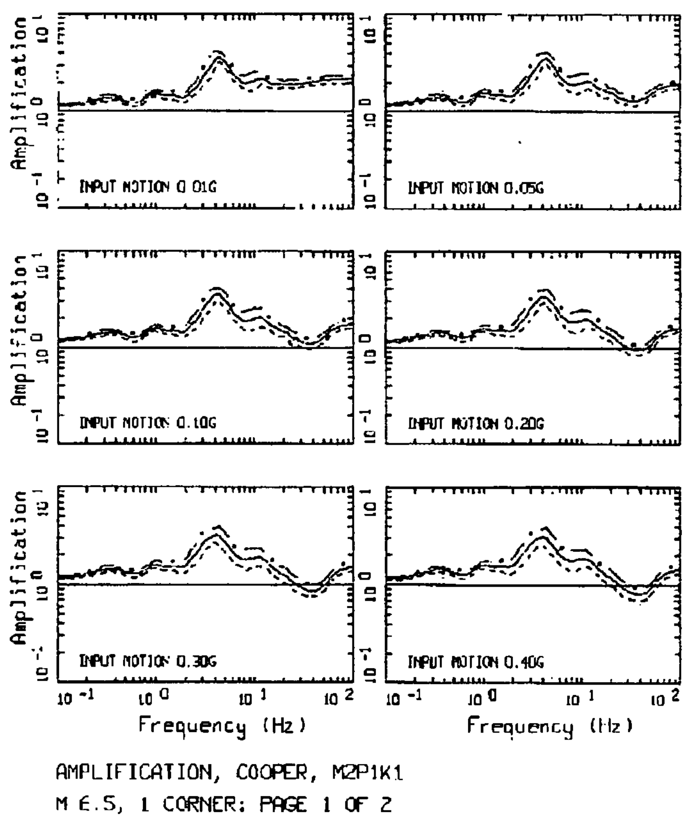

Figure 2.3.6-2 Example Suite of Amplification Factors (5 percent damping pseudo absolute acceleration spectra) developed for the mean base case profile (P1), linear site response (model M2), and base case kappa (K1) at 11 loading levels of hard rock median peak acceleration values from 0.01g to 1.50g. M 6.5 and single-corner source model per SPID[3].

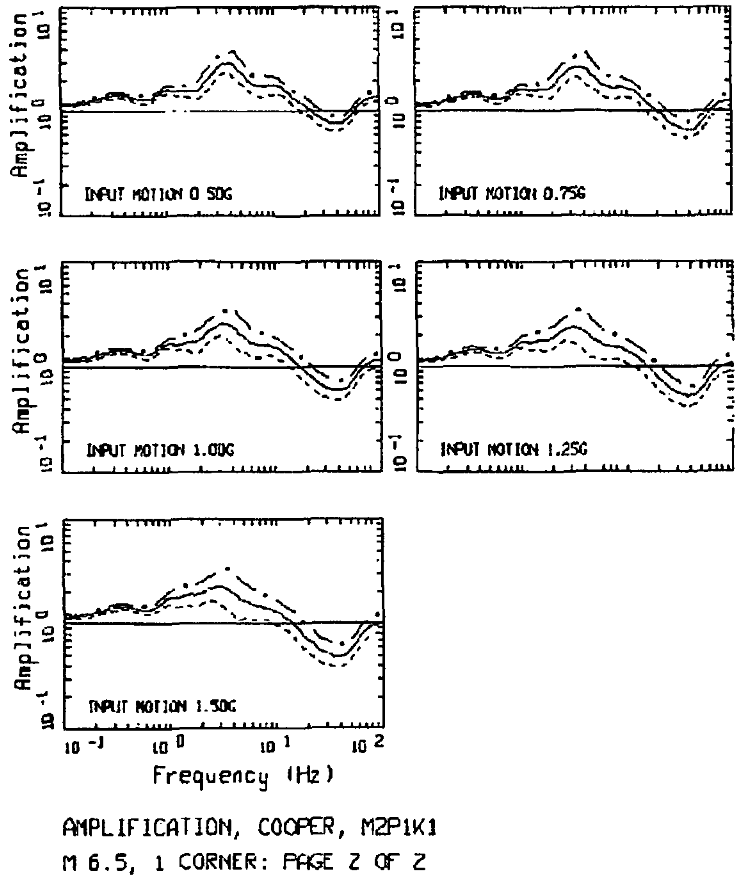

Figure 2.3.6-2 (continued)

2.3.7 Control Point Seismic Hazard Curves

The procedure to develop probabilistic site-specific control point hazard curves used in the present analysis follows the methodology described in Section B-6.0 of the SPID[3]. This procedure (referred to as Method 3) computes a site-specific control point hazard curve for a broad range of spectral accelerations given the site-specific bedrock hazard curve and site-specific estimates of soil or soft-rock response and associated uncertainties. This process is repeated for each of the seven spectral frequencies for which ground motion equations are available. The dynamic response of the materials below the control point was represented by the frequency- and amplitude-dependent amplification functions (median values and standard deviations) developed and described in the previous section. The resulting control point mean hazard curves for CNS are shown on Figure 2.3.7-1 for the seven spectral frequencies for which ground motion equations are defined. Tabulated values of the control point hazard curves are provided in Appendix A.

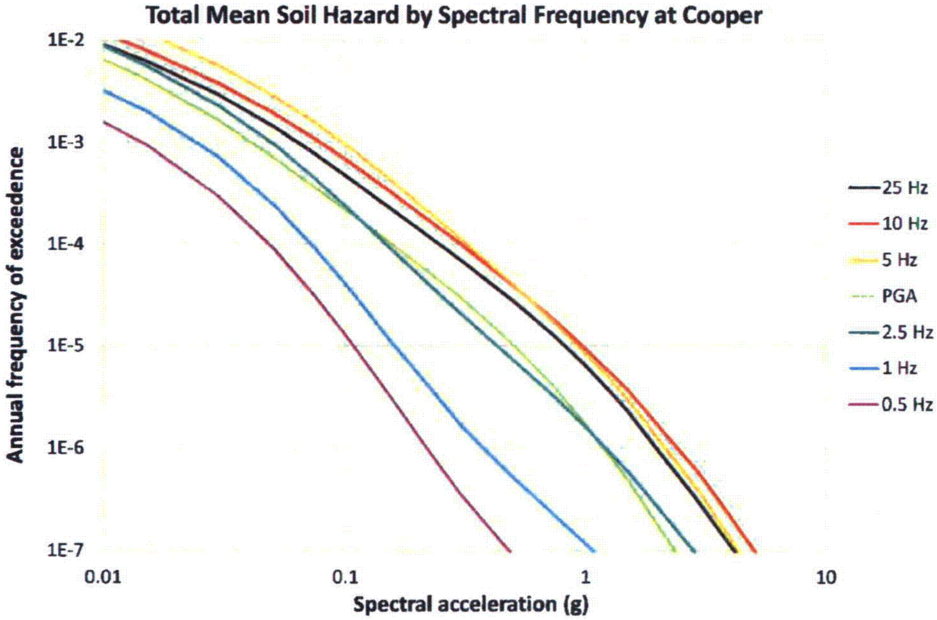

Figure 2.3.7-1 Control Point Mean Hazard Curves for Spectral Frequencies of 0.5, 1, 2.5, 5, 10, 25 and 100 Hz at CNS

2.4 Control Point Response Spectra

The control point hazard curves described above have been used to develop uniform hazard response spectra (UHRS) and the ground motion response spectrum (GMRS). The UHRS were obtained through linear interpolation in log-log space to estimate the spectral acceleration at each spectral frequency for the 1E-4 and 1E-5 per year hazard levels.

Table 2.4-1 shows the UHRS and GMRS accelerations for a range of spectral frequencies.

The 1E-4 and 1E-5 UHRS are used to compute the GMRS at the control point and are shown in Figure 2.4-1.

Table 2.4-1 UHRS for 10^{-4} and 10^{-5} and GMRS at Control Point for CNS

Freq. (Hz)	10^{-4} UHRS (g)	10^{-5} UHRS (g)	GMRS (g)
100	1.58E-01	5.07E-01	2.41E-01
90	1.58E-01	5.11E-01	2.42E-01
80	1.59E-01	5.16E-01	2.45E-01
70	1.61E-01	5.25E-01	2.49E-01
60	1.65E-01	5.47E-01	2.58E-01
50	1.79E-01	5.98E-01	2.82E-01
40	2.02E-01	6.84E-01	3.21E-01
35	2.15E-01	7.27E-01	3.42E-01
30	2.24E-01	7.64E-01	3.59E-01
25	2.46E-01	8.19E-01	3.86E-01
20	2.65E-01	8.84E-01	4.17E-01
15	3.04E-01	9.75E-01	4.63E-01
12.5	3.17E-01	1.02E+00	4.86E-01
10	3.05E-01	9.80E-01	4.65E-01
9	2.92E-01	9.46E-01	4.49E-01
8	2.80E-01	9.06E-01	4.30E-01
7	2.75E-01	8.76E-01	4.17E-01
6	2.89E-01	8.79E-01	4.22E-01
5	3.24E-01	9.37E-01	4.54E-01
4	3.04E-01	8.50E-01	4.15E-01
3.5	2.64E-01	7.47E-01	3.64E-01
3	2.12E-01	6.03E-01	2.94E-01
2.5	1.48E-01	4.30E-01	2.09E-01
2	1.16E-01	3.34E-01	1.62E-01
1.5	8.83E-02	2.35E-01	1.16E-01
1.25	7.78E-02	1.91E-01	9.57E-02
1	7.24E-02	1.61E-01	8.23E-02
0.9	6.73E-02	1.49E-01	7.64E-02
0.8	6.08E-02	1.35E-01	6.90E-02
0.7	5.58E-02	1.24E-01	6.34E-02
0.6	5.26E-02	1.17E-01	5.97E-02
0.5	4.86E-02	1.08E-01	5.54E-02
0.4	3.89E-02	8.66E-02	4.43E-02
0.35	3.40E-02	7.58E-02	3.88E-02
0.3	2.92E-02	6.50E-02	3.32E-02
0.25	2.43E-02	5.42E-02	2.77E-02
0.2	1.94E-02	4.33E-02	2.21E-02
0.15	1.46E-02	3.25E-02	1.66E-02
0.125	1.22E-02	2.71E-02	1.38E-02
0.1	9.72E-03	2.17E-02	1.11E-02

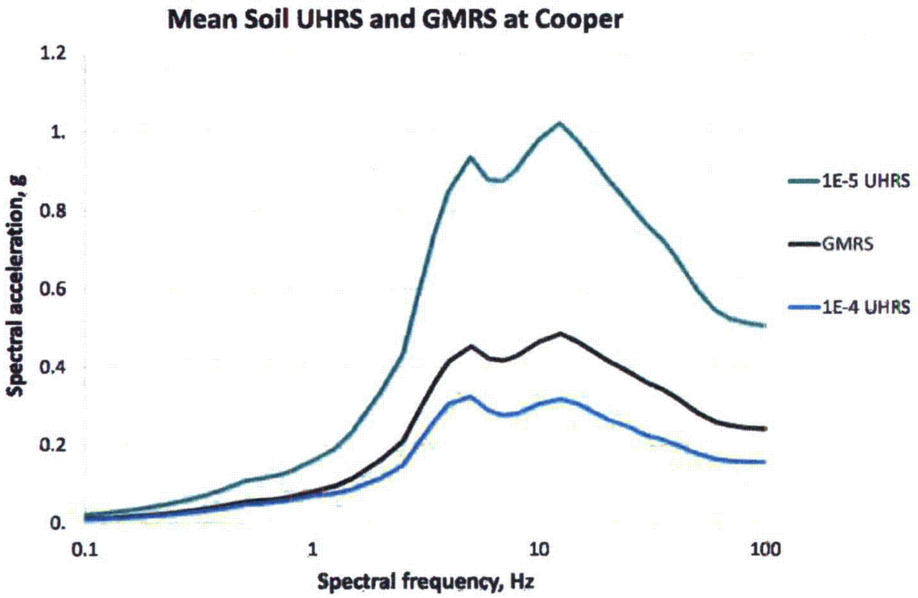

Figure 2.4-1 UHRS for 1E-4 and 1E-5 and GMRS at Control Point for CNS (5 Percent Damping)

3.0 Plant Design Basis and Beyond Design Basis Evaluation Ground Motion

The design basis for CNS is identified in the USAR[4].

An evaluation for beyond design basis (BDB) ground motion was performed in the Individual Plant Examination of External Events (IPEEE). The IPEEE plant level HCLPF response spectrum is provided in Section 3.3 for screening purposes.

3.1 SSE Description of Spectral Shape

The SSE was developed in accordance with 10 CFR Part 100, Appendix A that existed at the time of the construction permit through an evaluation of the maximum earthquake potential for the region surrounding the site. The SSE for CNS was developed based on the U.S. Coast & Geodetic Survey (USC&GS) Seismic-Probability Map, the records of historical earthquakes, and the regional and local geologic structural features according to CNS USAR, Section II-5.2[4]. Considering the historical seismicity of the site region, CNS determined that an earthquake with an intensity of VII on the Modified Mercalli Scale (MM) would affect the site during the life of the nuclear power station. The hypothetical maximum possible intensity of ground motion at the site would likely result from a local earthquake smaller than the New Madrid earthquakes of 1811 and 1812. CNS USAR, Section II-5.2[4], considered it improbable that future local earthquakes (e.g., the Humboldt Fault) would have epicentral intensities greater than MM VII.

Considering the regional and local geology and seismology at CNS as stated in the CNS USAR, Chapter II[4], a hypothetical maximum possible design earthquake (i.e., SSE) with a PGA of 0.2g was selected for structural analysis. The 0.2g value was chosen for the horizontal component of the acceleration at both the rock surface, approximate elevation of 820 feet MSL, and the base of the structures. The application of the SSE at the base of each Class I structure is based on the assumption that the structures are founded on a dense structural fill.

Also from the USAR[4], the SSE response spectrum was developed using the accelerogram of the N69W component of the July 21, 1952, Kern County earthquake recorded at Taft, California. This accelerogram was selected for reasons of geology, geometry, seismology, and comparison with other spectra. The SSE response spectrum developed for CNS is shown in Table 3.1-1 and is similar to the average spectrum recommended by the U.S. Atomic Energy Commission TID-7024[13].

Table 3.1-1 SSE for CNS (5 Percent Damping)

Frequency (Hz)	100/PGA	33	25	9	5	3	2.5	1.8	1	0.5
Spectral Acc. (g)	0.20	0.20	0.26	0.34	0.42	0.53	0.50	0.41	0.19	0.13

3.2 Control Point Elevation(s)

A single SSE control point is defined at elevation 869.5 feet of the Control Building according to the EPRI Data Request Report[8].

The CNS SSE has multiple control points described in the CNS USAR[4]. As part of the CNS IPEEE report[28], which produced the IHS, A soil-structure interaction (SSI) analysis was performed for both the Reactor Building and Control Building. For the comparison of the GMRS, IHS and SSE, the

Control Building control point elevation (869.5 feet) is used because it is at a higher elevation than the Reactor Building control point elevation. This is consistent with the SPID[3] guidance.

3.3 IPEEE Description and Capacity Response Spectrum

A focused scope Seismic Margins Assessment (SMA) was performed to support the IPEEE for CNS. The results of the IPEEE were submitted to the NRC in the CNS IPEEE report[28]. Results of the NRC review are documented in the NRC letter[25] to NPPD dated April 27, 2001.

The CNS Seismic IPEEE was performed using the SMA option per the methodology of NP-6041-SL[26]. With this method, a seismic margins earthquake (SME) was postulated and the SSCs needed for safe shutdown were then evaluated for the SME demand. SSCs that were determined to have sufficient capacity to survive the SME without loss of function were screened out. SSCs that did not screen were subjected to a more detailed evaluation, including calculation of a IPEEE high-confidence-low-probability of failure (HCLPF) Spectrum (IHS). A 0.30g PGA earthquake level and the NUREG/CR-0098[27] median response spectra shape were used to develop the IHS.

The IPEEE was reviewed for adequacy utilizing the guidance provided in Section 3.3 of the SPID[3]. The IPEEE adequacy determination according to SPID[3] Section 3.3.1 is included in Appendix B.

The results of the review have shown, in accordance with the criteria established in SPID[3] Section 3.3, that the IPEEE is adequate to support screening of the updated seismic hazard for CNS. The review also concluded that the risk insights obtained from the IPEEE are still valid under the current plant configuration.

The full scope detailed review of relay chatter required in SPID[3] Section 3.3.1 has not been completed. As identified in the NEI letter[22] to NRC dated October 3, 2013, the relay chatter review is intended to be on the same schedule as the high frequency confirmation as proposed in the NEI letter[23] to NRC dated April 9, 2013 and accepted in the NRC letter[24] dated May 7, 2013.

The full evaluation of soil failures required in SPID[3] Section 3.3.1 has been completed. The results of the evaluation are provided in Appendix C.

The 5% damped horizontal IHS spectral accelerations are provided in Table 3.3-1. The IHS, GMRS and SSE are shown in Figure 3.3-1.

Table 3.3-1 IHS for CNS (5 Percent Damping)

Freq (Hz)	Spectral Acc. (g)
100.00	0.300
50.00	0.300
33.00	0.300
30.00	0.316
25.00	0.347
20.00	0.391
15.00	0.455
10.00	0.564
8.00	0.635
1.64	0.635
1.50	0.580
1.25	0.483
1.00	0.387
0.75	0.290
0.50	0.193
0.25	0.098
0.20	0.061
0.15	0.034
0.10	0.015

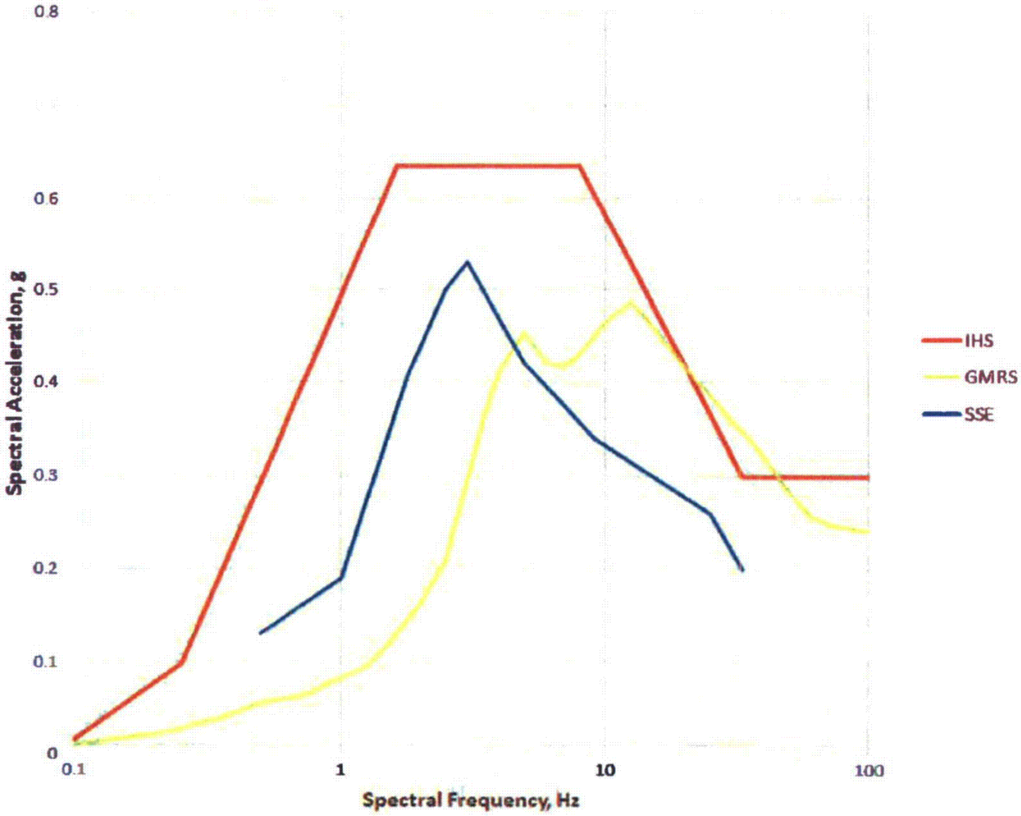

Figure 3.3-1 IHS, GMRS and SSE for CNS (5 Percent Damping)

4.0 Screening Evaluation

In accordance with SPID[3] Section 3, a screening evaluation was performed. The IHS was used for the risk evaluation screening and high frequency screening. However, the SSE was used for the spent fuel pool evaluation screening because the spent fuel pool was not evaluated under IPEEE.

4.1 Risk Evaluation Screening (1 to 10 Hz)

In the 1 to 10 Hz part of the response spectrum, the IHS exceeds the GMRS. Based on this comparison, a risk evaluation will not be performed.

4.2 High Frequency Screening (> 10 Hz)

Above 10 Hz, the GMRS exceeds the IHS. Therefore, the plant screens in for a high frequency confirmation.

4.3 Spent Fuel Pool Evaluation Screening (1 to 10 Hz)

In the 1 to 10 Hz part of the response spectrum, the GMRS exceeds the SSE. Therefore, the plant screens in for a spent fuel pool evaluation.

5.0 Interim Actions

Based on the screening evaluation, the expedited seismic evaluation described in EPRI 3002000704[17] will be performed as proposed in an NEI letter[23] to NRC dated April 9, 2013, and agreed to by the NRC letter[24] dated May 7, 2013.

Consistent with NRC letter[18] dated February 20, 2014, the seismic hazard reevaluations presented herein are distinct from the current design and licensing bases of CNS. Therefore, the results do not call into question the operability or functionality of SSCs and are not reportable pursuant to 10 CFR 50.72, "Immediate notification requirements for operating nuclear power reactors," and 10 CFR 50.73, "Licensee event report system".

The NRC letter[18] also requests that licensees provide an interim evaluation or actions to demonstrate that the plant can cope with the reevaluated hazard while the expedited approach and risk evaluations are conducted. In response to that request, NEI letter[19] to the NRC dated March 12, 2014, provides seismic core damage risk estimates using the updated seismic hazards for the operating nuclear plants in the Central and Eastern United States. These risk estimates continue to support the following conclusions of the NRC GI-199 Safety/Risk Assessment:

> Overall seismic core damage risk estimates are consistent with the Commission's Safety Goal Policy Statement because they are within the subsidiary objective of 10^{-4}/year for core damage frequency. The GI-199 Safety/Risk Assessment, based in part on information from the U.S. Nuclear Regulatory Commission's (NRC's) Individual Plant Examination of External Events (IPEEE) program, indicates that no concern exists regarding adequate protection and that the current seismic design of operating reactors provides a safety margin to withstand potential earthquakes exceeding the original design basis.

CNS is included in the March 12, 2014 risk estimates per NEI letter[19]. Using the methodology described in the NEI letter[19], all plants were shown to be below 10^{-4}/year; thus, the above conclusions apply.

5.1 NTTF 2.3 – Seismic Walkdowns

CNS performed seismic walkdowns to meet Near-Term Task Force Recommendation 2.3: Seismic. As part of this program a total of 104 seismic walkdowns and 60 area walk-bys were conducted resulting in 53 Condition Reports (CR). A summary of these CR's is available in the Seismic Walkdown Report[16]. These CR's are documented in a table in the Seismic Walkdown Report[16] along with the categorization of the action to have them resolved.

Also as part of the 2.3 walkdown, IPEEE vulnerabilities were reviewed and evaluated. This evaluation concluded that there are no IPEEE vulnerabilities at CNS. Details of these evaluations are available for review in Seismic Walkdown Report[16]. Final walkdowns have been completed during plant outages and are available in CNS Engineering Report ER-2014-020[2].

6.0 Conclusions

In accordance with the 50.54(f) letter[1], a seismic hazard and screening evaluation was performed for CNS. A GMRS was developed solely for the purpose of screening for additional evaluations in accordance with the SPID[3].

Based on the results of the screening evaluation, the plant screens out of a risk evaluation, but screens in for a spent fuel pool evaluation and high frequency confirmation.

7.0 References

1. U.S. Nuclear Regulatory Commission (Eric J. Leeds and Michael R. Johnson), Letter to All Power Reactor Licensees et al., "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendations 2.1, 2.3 and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," March 12, 2012.

2. Nebraska Public Power District, *Cooper Nuclear Station 50.54(f) Section 2.3 Seismic Walkdowns Supplemental Report*, CNS Engineering Report ER-2014-020, Rev. 0.

3. Electric Power Research Institute, *Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic*, 1025287, EPRI, Palo Alto, California, February 2013.

4. Nebraska Public Power District, CNS USAR Chapter II, "Station Site and Environs."

5. U.S. Nuclear Regulatory Commission, *Central and Eastern United States Seismic Source Characterization for Nuclear Facilities*, NUREG-2115, 6 Volumes, NRC (Published Jointly with EPRI and DOE), Washington, D.C., January 2012.

6. Electric Power Research Institute, *EPRI (2004, 2006) Ground-Motion Model (GMM) Review Project*, 3002000717, EPRI, Palo Alto, California, June 2013.

7. U.S. Nuclear Regulatory Commission, *A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion*, Regulatory Guide 1.208, NRC, Washington, D.C., March 2007.

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9. Nebraska Public Power District, *Review of ZNE Calculation 12-366, Determination of Dynamic Soil and Rock Properties for Class I Structures at the CNS Site*, NEDC 13-019, Rev. 0.

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11. G. E. Condra, *Geologic Cross-Section, Forest City, Missouri to Du Bois, Nebraska*, Nebraska Geologic Survey, 1935.

12. Nebraska Public Power District (Patrick L. Pope), Letter NLS2013085 to U.S. Nuclear Regulatory Commission (Document Control Desk), "Nebraska Public Power District's Response to Nuclear Regulatory Commission Request for Information Pursuant to 10CFR 50.54(f) Regarding the Seismic Aspects of Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident – 1.5 Year Response for CEUS Sites," September 12, 2013.

13. U.S. Atomic Energy Commission, *Nuclear Reactors and Earthquakes*, TID-7024, U.S. Atomic Energy Commission, Washington, D.C., August 1963.

14. Not Used.

15. Nebraska Public Power District, *50.54(f) Section 2.1 Seismic – LCI GMRS Report Acceptance*, CNS Engineering Report ER-2014-002, Rev. 1.

16. Nebraska Public Power District (Brian J. O'Grady), Letter NLS2012125 to U.S. Nuclear Regulatory Commission (Document Control Desk), "Seismic Walkdown Report – Nebraska Public Power District's Response to Nuclear Regulatory Commission Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Seismic Aspects of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," November 27, 2012.

17. Electric Power Research Institute, *Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1 - Seismic*, 3002000704, EPRI, Palo Alto, California, May 2013.

18. U.S. Nuclear Regulatory Commission (Eric J. Leeds), Letter to All Power Reactor Licensees et al., "Supplemental Information Related to Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Seismic Hazard Reevaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," February 20, 2014.

19. Nuclear Energy Institute (Anthony R. Pietrangelo), Letter to U.S. Nuclear Regulatory Commission (Eric J. Leeds) "Seismic Risk Evaluations for Plants in the Central and Eastern United States," March 12, 2014.

20. Not Used.

21. Not Used.

22. Nuclear Energy Institute (Kimberly A. Keithline), Letter to U.S. Nuclear Regulatory Commission (David L. Skeen), "Relay Chatter Reviews for Seismic Hazard Screening," October 3, 2013.

23. Nuclear Energy Institute (Anthony R. Pietrangelo), Letter to U.S. Nuclear Regulatory Commission (David L. Skeen), "Proposed Path Forward for NTTF Recommendation 2.1: Seismic Reevaluations," April 9, 2013.

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25. U.S. Nuclear Regulatory Commission (Mohan C. Thadani), Letter to Nebraska Public Power District (J. H. Swailes), "Cooper Nuclear Station – Review of Individual Plant Examination of External Events (TAC NO. 83611)," April 27, 2001.

26. Electric Power Research Institute, *A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1)*, NP-6041-SL, EPRI, Palo Alto, California, August 1991.

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28. Nebraska Public Power District (G. R. Horn), Letter NLS960143 to U.S. Nuclear Regulatory Commission (Document Control Desk), "Individual Plant Examination for External Events (IPEEE) Report - 10 CFR 50.54(f)," October 30, 1996.

Appendix A – Seismic Hazard Data and Tables

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Table A-1a Mean and Fractile Seismic Hazard Curves for 0.5 Hz at Cooper

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	1.82E-02	8.23E-03	1.20E-02	1.77E-02	2.42E-02	2.96E-02
0.001	1.09E-02	4.50E-03	6.64E-03	1.04E-02	1.51E-02	1.92E-02
0.005	3.08E-03	5.20E-04	1.11E-03	2.72E-03	5.05E-03	6.93E-03
0.01	1.58E-03	1.13E-04	3.09E-04	1.16E-03	2.92E-03	4.43E-03
0.015	9.57E-04	3.95E-05	1.20E-04	5.66E-04	1.87E-03	3.14E-03
0.03	2.99E-04	5.27E-06	1.77E-05	1.10E-04	5.66E-04	1.21E-03
0.05	9.38E-05	1.05E-06	3.57E-06	2.42E-05	1.53E-04	4.19E-04
0.075	3.11E-05	2.72E-07	9.51E-07	6.54E-06	4.43E-05	1.42E-04
0.1	1.30E-05	9.79E-08	3.63E-07	2.46E-06	1.69E-05	5.75E-05
0.15	3.45E-06	1.98E-08	8.85E-08	6.26E-07	4.31E-06	1.51E-05
0.3	3.65E-07	1.05E-09	6.36E-09	6.09E-08	4.56E-07	1.69E-06
0.5	9.11E-08	2.07E-10	8.00E-10	1.02E-08	1.01E-07	4.37E-07
0.75	3.46E-08	1.62E-10	2.32E-10	2.39E-09	3.09E-08	1.62E-07
1.	1.77E-08	1.62E-10	1.64E-10	8.72E-10	1.32E-08	7.89E-08
1.5	6.68E-09	1.32E-10	1.62E-10	2.68E-10	3.73E-09	2.76E-08
3.	1.06E-09	1.21E-10	1.32E-10	1.62E-10	4.31E-10	3.57E-09
5.	2.25E-10	1.21E-10	1.32E-10	1.62E-10	1.72E-10	6.83E-10
7.5	5.73E-11	1.21E-10	1.32E-10	1.62E-10	1.62E-10	2.42E-10
10.	2.01E-11	1.21E-10	1.32E-10	1.62E-10	1.62E-10	1.69E-10

Table A-1b Mean and Fractile Seismic Hazard Curves for 1 Hz at Cooper

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	3.70E-02	1.74E-02	2.46E-02	3.68E-02	4.90E-02	5.66E-02
0.001	2.45E-02	1.01E-02	1.51E-02	2.39E-02	3.37E-02	4.07E-02
0.005	6.39E-03	1.98E-03	3.33E-03	6.00E-03	9.51E-03	1.21E-02
0.01	3.23E-03	6.36E-04	1.25E-03	2.88E-03	5.20E-03	7.13E-03
0.015	2.03E-03	2.72E-04	5.91E-04	1.64E-03	3.52E-03	5.12E-03
0.03	7.22E-04	5.05E-05	1.20E-04	4.31E-04	1.36E-03	2.32E-03
0.05	2.53E-04	1.25E-05	3.05E-05	1.18E-04	4.50E-04	9.51E-04
0.075	9.15E-05	3.90E-06	9.51E-06	3.79E-05	1.49E-04	3.57E-04
0.1	4.08E-05	1.64E-06	4.13E-06	1.62E-05	6.36E-05	1.57E-04
0.15	1.22E-05	4.56E-07	1.25E-06	4.90E-06	1.84E-05	4.63E-05
0.3	1.71E-06	4.07E-08	1.49E-07	7.03E-07	2.80E-06	6.83E-06
0.5	5.16E-07	5.27E-09	2.60E-08	1.82E-07	8.35E-07	2.16E-06
0.75	2.18E-07	9.93E-10	5.91E-09	6.09E-08	3.42E-07	9.51E-07
1.	1.18E-07	3.57E-10	1.95E-09	2.72E-08	1.79E-07	5.35E-07
1.5	4.80E-08	1.74E-10	4.70E-10	8.00E-09	6.73E-08	2.22E-07
3.	8.65E-09	1.62E-10	1.62E-10	8.47E-10	1.01E-08	4.01E-08
5.	2.02E-09	1.32E-10	1.62E-10	2.29E-10	2.01E-09	8.98E-09
7.5	5.59E-10	1.27E-10	1.36E-10	1.62E-10	5.50E-10	2.39E-09
10.	2.08E-10	1.21E-10	1.32E-10	1.62E-10	2.68E-10	9.24E-10

Table A-1c Mean and Fractile Seismic Hazard Curves for 2.5 Hz at Cooper

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	5.57E-02	3.90E-02	4.56E-02	5.50E-02	6.64E-02	7.23E-02
0.001	4.63E-02	2.68E-02	3.37E-02	4.63E-02	5.83E-02	6.64E-02
0.005	1.72E-02	6.64E-03	9.51E-03	1.57E-02	2.53E-02	3.23E-02
0.01	8.86E-03	2.84E-03	4.43E-03	8.12E-03	1.34E-02	1.74E-02
0.015	5.67E-03	1.51E-03	2.49E-03	5.12E-03	8.85E-03	1.18E-02
0.03	2.29E-03	3.95E-04	7.23E-04	1.84E-03	3.90E-03	5.66E-03
0.05	9.87E-04	1.29E-04	2.46E-04	6.83E-04	1.72E-03	2.88E-03
0.075	4.47E-04	5.12E-05	9.93E-05	2.84E-04	7.55E-04	1.42E-03
0.1	2.41E-04	2.64E-05	5.20E-05	1.49E-04	3.95E-04	7.66E-04
0.15	9.71E-05	1.04E-05	2.10E-05	6.00E-05	1.57E-04	3.05E-04
0.3	2.12E-05	2.10E-06	4.77E-06	1.38E-05	3.57E-05	6.54E-05
0.5	7.31E-06	6.09E-07	1.57E-06	4.77E-06	1.25E-05	2.25E-05
0.75	3.09E-06	2.07E-07	5.91E-07	1.95E-06	5.27E-06	9.79E-06
1.	1.61E-06	8.85E-08	2.76E-07	9.79E-07	2.80E-06	5.27E-06
1.5	5.96E-07	2.35E-08	8.23E-08	3.33E-07	1.04E-06	2.04E-06
3.	8.35E-08	1.77E-09	6.73E-09	3.47E-08	1.44E-07	3.28E-07
5.	1.58E-08	2.92E-10	8.23E-10	4.56E-09	2.49E-08	6.83E-08
7.5	3.76E-09	1.62E-10	2.22E-10	7.77E-10	5.20E-09	1.72E-08
10.	1.29E-09	1.44E-10	1.62E-10	2.80E-10	1.57E-09	6.00E-09

Table A-1d Mean and Fractile Seismic Hazard Curves for 5 Hz at Cooper

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	6.10E-02	4.70E-02	5.12E-02	6.09E-02	7.03E-02	7.55E-02
0.001	5.65E-02	4.01E-02	4.63E-02	5.66E-02	6.73E-02	7.23E-02
0.005	2.98E-02	1.40E-02	1.92E-02	2.92E-02	4.07E-02	4.77E-02
0.01	1.75E-02	7.23E-03	1.04E-02	1.64E-02	2.46E-02	3.05E-02
0.015	1.20E-02	4.56E-03	6.83E-03	1.13E-02	1.74E-02	2.19E-02
0.03	5.72E-03	1.77E-03	2.80E-03	5.20E-03	8.72E-03	1.13E-02
0.05	2.95E-03	7.66E-04	1.23E-03	2.53E-03	4.70E-03	6.64E-03
0.075	1.59E-03	3.63E-04	5.91E-04	1.25E-03	2.57E-03	3.95E-03
0.1	9.73E-04	2.04E-04	3.42E-04	7.34E-04	1.55E-03	2.57E-03
0.15	4.61E-04	8.60E-05	1.49E-04	3.33E-04	7.13E-04	1.25E-03
0.3	1.17E-04	1.74E-05	3.33E-05	8.23E-05	1.87E-04	3.23E-04
0.5	4.06E-05	4.90E-06	1.04E-05	2.84E-05	6.73E-05	1.16E-04
0.75	1.68E-05	1.60E-06	3.68E-06	1.15E-05	2.88E-05	4.98E-05
1.	8.59E-06	6.45E-07	1.62E-06	5.58E-06	1.53E-05	2.64E-05
1.5	3.04E-06	1.46E-07	4.31E-07	1.82E-06	5.58E-06	9.93E-06
3.	3.71E-07	6.26E-09	2.72E-08	1.69E-07	7.03E-07	1.38E-06
5.	6.00E-08	4.83E-10	2.49E-09	1.98E-08	1.08E-07	2.49E-07
7.5	1.26E-08	1.69E-10	3.90E-10	3.01E-09	2.04E-08	5.58E-08
10.	4.07E-09	1.62E-10	1.84E-10	8.00E-10	5.91E-09	1.84E-08

Table A-1e Mean and Fractile Seismic Hazard Curves for 10 Hz at Cooper

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	5.85E-02	4.37E-02	4.90E-02	5.83E-02	6.83E-02	7.34E-02
0.001	5.08E-02	3.37E-02	4.13E-02	5.12E-02	6.09E-02	6.64E-02
0.005	2.11E-02	1.05E-02	1.46E-02	2.04E-02	2.72E-02	3.42E-02
0.01	1.18E-02	5.35E-03	7.55E-03	1.11E-02	1.57E-02	2.07E-02
0.015	8.07E-03	3.33E-03	4.83E-03	7.55E-03	1.11E-02	1.51E-02
0.03	3.84E-03	1.34E-03	1.98E-03	3.37E-03	5.58E-03	8.12E-03
0.05	1.99E-03	6.45E-04	9.37E-04	1.64E-03	2.92E-03	4.63E-03
0.075	1.09E-03	3.37E-04	4.98E-04	8.72E-04	1.55E-03	2.76E-03
0.1	6.93E-04	2.01E-04	3.05E-04	5.42E-04	9.79E-04	1.79E-03
0.15	3.50E-04	9.11E-05	1.44E-04	2.68E-04	4.98E-04	9.24E-04
0.3	1.03E-04	2.10E-05	3.84E-05	7.89E-05	1.60E-04	2.68E-04
0.5	3.99E-05	6.73E-06	1.36E-05	3.05E-05	6.36E-05	1.05E-04
0.75	1.78E-05	2.39E-06	5.27E-06	1.32E-05	2.92E-05	4.77E-05
1.	9.57E-06	9.79E-07	2.39E-06	6.73E-06	1.62E-05	2.72E-05
1.5	3.70E-06	2.19E-07	6.26E-07	2.39E-06	6.64E-06	1.15E-05
3.	5.55E-07	8.47E-09	4.25E-08	2.64E-07	1.07E-06	2.04E-06
5.	1.04E-07	6.17E-10	4.83E-09	3.52E-08	1.98E-07	4.25E-07
7.5	2.33E-08	1.84E-10	7.34E-10	5.66E-09	4.19E-08	1.01E-07
10.	7.42E-09	1.62E-10	2.49E-10	1.53E-09	1.25E-08	3.28E-08

Table A-1f Mean and Fractile Seismic Hazard Curves for 25 Hz at Cooper

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	5.38E-02	2.92E-02	4.43E-02	5.42E-02	6.45E-02	7.03E-02
0.001	4.35E-02	1.98E-02	3.37E-02	4.37E-02	5.42E-02	6.17E-02
0.005	1.59E-02	6.26E-03	1.02E-02	1.49E-02	2.10E-02	3.05E-02
0.01	9.00E-03	3.19E-03	5.12E-03	8.12E-03	1.23E-02	1.90E-02
0.015	6.25E-03	2.07E-03	3.33E-03	5.50E-03	8.85E-03	1.38E-02
0.03	2.94E-03	8.98E-04	1.34E-03	2.39E-03	4.31E-03	7.34E-03
0.05	1.45E-03	3.63E-04	5.75E-04	1.10E-03	2.13E-03	4.01E-03
0.075	7.65E-04	1.49E-04	2.53E-04	5.58E-04	1.13E-03	2.16E-03
0.1	4.74E-04	7.66E-05	1.36E-04	3.33E-04	7.23E-04	1.38E-03
0.15	2.36E-04	3.01E-05	5.75E-05	1.60E-04	3.73E-04	7.03E-04
0.3	7.06E-05	7.03E-06	1.49E-05	4.63E-05	1.18E-04	2.19E-04
0.5	2.77E-05	2.46E-06	5.66E-06	1.79E-05	4.70E-05	8.47E-05
0.75	1.22E-05	1.02E-06	2.53E-06	7.66E-06	2.10E-05	3.79E-05
1.	6.37E-06	5.35E-07	1.34E-06	3.84E-06	1.13E-05	2.01E-05
1.5	2.27E-06	1.77E-07	4.63E-07	1.32E-06	4.07E-06	7.45E-06
3.	2.82E-07	9.11E-09	3.47E-08	1.57E-07	5.12E-07	9.51E-07
5.	5.50E-08	5.05E-10	2.84E-09	2.32E-08	1.01E-07	2.19E-07
7.5	1.65E-08	1.62E-10	3.84E-10	3.95E-09	3.05E-08	7.34E-08
10.	7.40E-09	1.62E-10	1.79E-10	1.07E-09	1.32E-08	3.52E-08

Table A-1g Mean and Fractile Seismic Hazard Curves for PGA at Cooper

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	5.10E-02	2.29E-02	4.01E-02	5.12E-02	6.36E-02	6.93E-02
0.001	3.94E-02	1.49E-02	2.80E-02	3.95E-02	5.20E-02	5.91E-02
0.005	1.23E-02	4.01E-03	7.55E-03	1.15E-02	1.69E-02	2.35E-02
0.01	6.44E-03	1.90E-03	3.47E-03	5.75E-03	9.11E-03	1.38E-02
0.015	4.17E-03	1.20E-03	2.01E-03	3.57E-03	6.09E-03	9.79E-03
0.03	1.65E-03	4.43E-04	6.73E-04	1.25E-03	2.32E-03	4.77E-03
0.05	7.24E-04	1.64E-04	2.60E-04	5.05E-04	1.01E-03	2.29E-03
0.075	3.62E-04	6.83E-05	1.16E-04	2.39E-04	5.27E-04	1.18E-03
0.1	2.21E-04	3.68E-05	6.64E-05	1.40E-04	3.33E-04	7.23E-04
0.15	1.10E-04	1.57E-05	3.09E-05	6.83E-05	1.69E-04	3.52E-04
0.3	3.08E-05	3.33E-06	7.55E-06	1.92E-05	4.83E-05	9.51E-05
0.5	1.04E-05	6.93E-07	1.90E-06	6.26E-06	1.67E-05	3.33E-05
0.75	3.80E-06	1.29E-07	4.63E-07	2.10E-06	6.45E-06	1.29E-05
1.	1.71E-06	3.09E-08	1.51E-07	8.60E-07	3.01E-06	6.09E-06
1.5	4.86E-07	3.28E-09	2.64E-08	1.95E-07	8.72E-07	1.87E-06
3.	4.29E-08	1.72E-10	7.23E-10	9.65E-09	7.13E-08	1.87E-07
5.	6.14E-09	1.44E-10	1.62E-10	8.35E-10	8.35E-09	2.80E-08
7.5	1.18E-09	1.32E-10	1.62E-10	2.07E-10	1.34E-09	5.50E-09
10.	3.38E-10	1.21E-10	1.32E-10	1.62E-10	4.07E-10	1.64E-09

Table A-2a Medians and Logarithmic Sigmas of Amplification Factors for CNS

PGA	Median AF	Sigma ln(AF)	25 Hz	Median AF	Sigma ln(AF)	10 Hz	Median AF	Sigma ln(AF)	5 Hz	Median AF	Sigma ln(AF)
1.00E-02	2.30E+00	1.00E-01	1.30E-02	2.01E+00	9.69E-02	1.90E-02	2.05E+00	1.90E-01	2.09E-02	3.15E+00	1.56E-01
4.95E-02	1.93E+00	1.01E-01	1.02E-01	1.41E+00	1.53E-01	9.99E-02	1.89E+00	2.14E-01	8.24E-02	2.96E+00	1.64E-01
9.64E-02	1.73E+00	1.05E-01	2.13E-01	1.25E+00	1.72E-01	1.85E-01	1.82E+00	2.16E-01	1.44E-01	2.80E+00	1.79E-01
1.94E-01	1.53E+00	1.12E-01	4.43E-01	1.07E+00	1.96E-01	3.56E-01	1.70E+00	2.17E-01	2.65E-01	2.57E+00	2.11E-01
2.92E-01	1.40E+00	1.21E-01	6.76E-01	9.58E-01	2.13E-01	5.23E-01	1.60E+00	2.24E-01	3.84E-01	2.38E+00	2.38E-01
3.91E-01	1.31E+00	1.30E-01	9.09E-01	8.72E-01	2.28E-01	6.90E-01	1.52E+00	2.32E-01	5.02E-01	2.22E+00	2.60E-01
4.93E-01	1.23E+00	1.40E-01	1.15E+00	8.03E-01	2.43E-01	8.61E-01	1.45E+00	2.40E-01	6.22E-01	2.09E+00	2.81E-01
7.41E-01	1.08E+00	1.71E-01	1.73E+00	6.79E-01	2.77E-01	1.27E+00	1.30E+00	2.64E-01	9.13E-01	1.83E+00	3.21E-01
1.01E+00	9.64E-01	2.04E-01	2.36E+00	5.83E-01	3.12E-01	1.72E+00	1.16E+00	2.91E-01	1.22E+00	1.62E+00	3.61E-01
1.28E+00	8.68E-01	2.47E-01	3.01E+00	5.07E-01	3.55E-01	2.17E+00	1.03E+00	3.26E-01	1.54E+00	1.43E+00	4.08E-01
1.55E+00	7.98E-01	2.59E-01	3.63E+00	5.00E-01	3.77E-01	2.61E+00	9.36E-01	3.24E-01	1.85E+00	1.32E+00	4.24E-01
2.5 Hz	**Median AF**	**Sigma ln(AF)**	**1 Hz**	**Median AF**	**Sigma ln(AF)**	**0.5 Hz**	**Median AF**	**Sigma ln(AF)**			
2.18E-02	1.94E+00	1.98E-01	1.27E-02	1.54E+00	1.02E-01	8.25E-03	1.39E+00	1.01E-01			
7.05E-02	2.06E+00	2.07E-01	3.43E-02	1.58E+00	1.02E-01	1.96E-02	1.42E+00	9.99E-02			
1.18E-01	2.12E+00	2.01E-01	5.51E-02	1.60E+00	1.03E-01	3.02E-02	1.42E+00	9.98E-02			
2.12E-01	2.18E+00	1.97E-01	9.63E-02	1.64E+00	1.09E-01	5.11E-02	1.44E+00	1.01E-01			
3.04E-01	2.19E+00	1.99E-01	1.36E-01	1.68E+00	1.20E-01	7.10E-02	1.45E+00	1.04E-01			
3.94E-01	2.19E+00	2.08E-01	1.75E-01	1.73E+00	1.36E-01	9.06E-02	1.47E+00	1.08E-01			
4.86E-01	2.17E+00	2.23E-01	2.14E-01	1.78E+00	1.55E-01	1.10E-01	1.47E+00	1.09E-01			
7.09E-01	2.08E+00	2.72E-01	3.10E-01	1.85E+00	2.44E-01	1.58E-01	1.49E+00	1.30E-01			
9.47E-01	1.93E+00	3.33E-01	4.12E-01	1.85E+00	3.08E-01	2.09E-01	1.52E+00	1.70E-01			
1.19E+00	1.80E+00	3.74E-01	5.18E-01	1.85E+00	3.42E-01	2.62E-01	1.53E+00	2.06E-01			
1.43E+00	1.75E+00	3.66E-01	6.19E-01	1.87E+00	3.58E-01	3.12E-01	1.54E+00	2.41E-01			

Appendix A (continued)

Tables and figures showing median amplification factors and uncertainties.

Note that per discussion with the NRC on February 5, 2014, these tables and figures concentrate on the frequency range of 0.5 Hz to 25 Hz, with values up to 100 Hz included, and a single value of 0.1 Hz included for completeness.

Table A-3a Median AFs and Sigmas for Model 1, Profile 1, for 2 PGA Levels

M1P1K1	Rock PGA=0.0964			M1P1K1	PGA=0.493		
Freq. (Hz)	Soil_SA	Med. AF	Sigma ln(AF)	Freq. (Hz)	Soil_SA	Med. AF	Sigma ln(AF)
100.0	0.154	1.602	0.100	100.0	0.510	1.036	0.154
87.1	0.155	1.577	0.100	87.1	0.512	1.007	0.154
75.9	0.156	1.531	0.101	75.9	0.514	0.957	0.155
66.1	0.158	1.445	0.102	66.1	0.516	0.867	0.156
57.5	0.161	1.295	0.105	57.5	0.521	0.731	0.159
50.1	0.166	1.137	0.110	50.1	0.530	0.611	0.165
43.7	0.175	1.019	0.120	43.7	0.545	0.531	0.173
38.0	0.184	0.965	0.123	38.0	0.565	0.506	0.183
33.1	0.197	0.962	0.141	33.1	0.590	0.505	0.192
28.8	0.210	1.008	0.139	28.8	0.617	0.535	0.202
25.1	0.226	1.063	0.161	25.1	0.653	0.568	0.212
21.9	0.241	1.171	0.162	21.9	0.699	0.647	0.221
19.1	0.264	1.280	0.126	19.1	0.748	0.710	0.239
16.6	0.282	1.403	0.197	16.6	0.788	0.787	0.242
14.5	0.288	1.481	0.190	14.5	0.849	0.896	0.258
12.6	0.314	1.644	0.175	12.6	0.911	0.996	0.280
11.0	0.345	1.834	0.198	11.0	0.962	1.087	0.283
9.5	0.336	1.849	0.209	9.5	1.017	1.211	0.285
8.3	0.306	1.807	0.233	8.3	1.061	1.380	0.251
7.2	0.293	1.831	0.206	7.2	1.057	1.478	0.243
6.3	0.295	1.954	0.183	6.3	1.010	1.512	0.235
5.5	0.333	2.285	0.205	5.5	1.003	1.580	0.261
4.8	0.386	2.691	0.227	4.8	1.040	1.683	0.287
4.2	0.436	3.115	0.192	4.2	1.114	1.868	0.328
3.6	0.420	3.069	0.176	3.6	1.253	2.169	0.319
3.2	0.357	2.756	0.212	3.2	1.317	2.430	0.282
2.8	0.279	2.261	0.228	2.8	1.267	2.473	0.234
2.4	0.217	1.896	0.200	2.4	1.099	2.334	0.204
2.1	0.169	1.617	0.155	2.1	0.882	2.068	0.194
1.8	0.140	1.491	0.135	1.8	0.711	1.871	0.215
1.6	0.124	1.520	0.117	1.6	0.600	1.827	0.203
1.4	0.107	1.510	0.098	1.4	0.491	1.744	0.167
1.2	0.095	1.517	0.083	1.2	0.417	1.687	0.136
1.0	0.090	1.579	0.087	1.0	0.379	1.709	0.120
0.91	0.080	1.540	0.063	0.91	0.329	1.639	0.089
0.79	0.066	1.398	0.069	0.79	0.266	1.472	0.083
0.69	0.055	1.291	0.088	0.69	0.215	1.348	0.094
0.60	0.047	1.263	0.095	0.60	0.181	1.309	0.098
0.52	0.041	1.293	0.084	0.52	0.156	1.332	0.086
0.46	0.036	1.348	0.062	0.46	0.134	1.380	0.065
0.10	0.001	1.177	0.024	0.10	0.005	1.181	0.026

Table A-3b Median AFs and Sigmas for Model 2, Profile 1, for 2 PGA Levels

M2P1K1		PGA=0.0964		M2P1K1		PGA=0.493	
Freq. (Hz)	Soil_SA	Med. AF	Sigma ln(AF)	Freq. (Hz)	Soil_SA	Med. AF	Sigma ln(AF)
100.0	0.170	1.760	0.094	100.0	0.684	1.389	0.112
87.1	0.171	1.733	0.094	87.1	0.688	1.354	0.113
75.9	0.172	1.686	0.095	75.9	0.693	1.292	0.113
66.1	0.174	1.595	0.096	66.1	0.702	1.178	0.115
57.5	0.178	1.438	0.097	57.5	0.718	1.008	0.120
50.1	0.186	1.276	0.103	50.1	0.749	0.864	0.127
43.7	0.198	1.152	0.108	43.7	0.796	0.776	0.140
38.0	0.211	1.105	0.106	38.0	0.854	0.765	0.150
33.1	0.227	1.110	0.128	33.1	0.913	0.783	0.160
28.8	0.245	1.179	0.120	28.8	0.973	0.844	0.173
25.1	0.262	1.233	0.160	25.1	1.036	0.901	0.183
21.9	0.284	1.380	0.144	21.9	1.122	1.038	0.174
19.1	0.306	1.483	0.130	19.1	1.222	1.159	0.198
16.6	0.323	1.608	0.189	16.6	1.261	1.259	0.167
14.5	0.334	1.721	0.145	14.5	1.357	1.432	0.169
12.6	0.371	1.942	0.191	12.6	1.492	1.631	0.161
11.0	0.380	2.016	0.206	11.0	1.567	1.771	0.192
9.5	0.351	1.931	0.242	9.5	1.502	1.790	0.221
8.3	0.315	1.865	0.238	8.3	1.346	1.750	0.253
7.2	0.309	1.934	0.175	7.2	1.260	1.760	0.232
6.3	0.327	2.160	0.155	6.3	1.248	1.867	0.219
5.5	0.386	2.650	0.174	5.5	1.375	2.167	0.249
4.8	0.456	3.185	0.177	4.8	1.566	2.535	0.282
4.2	0.489	3.501	0.136	4.2	1.737	2.914	0.275
3.6	0.429	3.138	0.225	3.6	1.703	2.948	0.203
3.2	0.344	2.657	0.257	3.2	1.513	2.792	0.225
2.8	0.261	2.109	0.210	2.8	1.225	2.392	0.261
2.4	0.203	1.773	0.165	2.4	0.948	2.014	0.246
2.1	0.161	1.535	0.129	2.1	0.727	1.705	0.206
1.8	0.135	1.437	0.124	1.8	0.590	1.553	0.179
1.6	0.121	1.481	0.108	1.6	0.515	1.570	0.144
1.4	0.105	1.482	0.090	1.4	0.436	1.550	0.105
1.2	0.094	1.497	0.079	1.2	0.382	1.548	0.092
1.0	0.089	1.565	0.085	1.0	0.356	1.607	0.095
0.91	0.080	1.530	0.061	0.91	0.314	1.565	0.067
0.79	0.066	1.391	0.067	0.79	0.257	1.420	0.068
0.69	0.055	1.287	0.088	0.69	0.210	1.313	0.088
0.60	0.047	1.261	0.096	0.60	0.178	1.284	0.096
0.52	0.041	1.292	0.085	0.52	0.154	1.313	0.086
0.46	0.036	1.348	0.064	0.46	0.133	1.367	0.066
0.10	0.001	1.178	0.025	0.10	0.005	1.178	0.027

Appendix B – IPEEE Adequacy Review

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B.1.0 Background

The Nuclear Regulatory Commission (NRC) staff issued Generic Letter 88-20[15], Supplement 4 on June 28, 1991, requesting that each licensee conduct an individual plant examination of external events (IPEEE) for severe accident vulnerabilities. Concurrently, NUREG-1407[1], "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" was issued to provide utilities with detailed guidance for performance of the IPEEE.

A seismic margin assessment (SMA) was performed for the seismic portion of the Cooper Nuclear Station (CNS) IPEEE report[5] using the EPRI SMA methodology, NP-6041-SL[3] with enhancements identified in NUREG-1407[1]. CNS performed a 0.3g PGA focused scope SMA utilizing a NUREG/CR-0098[11] spectral shape for a soil site. The EPRI SMA method was used because it was compatible with the Unresolved Safety Issue (USI) A-46 assessment that was performed in parallel with the IPEEE evaluation. The Safe Shutdown Equipment List (SSEL) identified for the IPEEE effort included components that were also on the USI A-46 Seismic Equipment List (SEL) and efficiencies were gained in the evaluation as a result. The calculated plant-level high confidence of low probability of failure (HCLPF) for CNS resulting from performance of the IPEEE was 0.3g PGA. The results of the CNS USI A-46 report[8] were provided to the NRC in a letter dated June 13, 1996. The results of the CNS IPEEE report[5] were provided to NRC in a letter dated October 30, 1996.

The NRC staff submitted a Request for Additional Information[24] (RAI) related to the IPEEE in a letter dated June 3, 1998, and an RAI[22] related to USI A-46 in a letter dated October 7, 1998. Identifying that commitments made in the IPEEE submittal were fulfilled, CNS submitted a closure letter in an RAI response[4] to the NRC on January 25, 1999 for USI A-46 commitments and an RAI response[12] on January 28, 1999 for remaining items for the IPEEE indicating that actions associated with IPEEE were completed.

The NRC issued its Staff Evaluation Report (SER) in a letter[7] dated April 27, 2001 for the CNS IPEEE report[5]. The SER included in NRC letter[7] concluded that the CNS IPEEE process was capable of identifying the most likely severe accidents and severe accident vulnerabilities, meeting the intent of Generic Letter 88-20[15], Supplement 4.

B.2.0 General Considerations

The CNS IPEEE report[5] is a focused scope EPRI SMA. The IPEEE HCLPF Spectrum (IHS) is developed in accordance with NUREG/CR-0098[11] anchored at 0.3g PGA. This spectrum is used for comparison of the new Ground Motion Response Spectrum (GMRS) and is based on the plant-level HCLPF reported to the NRC in the CNS IPEEE report[5]. The EPRI SMA was selected as the method for the IPEEE evaluation because it was compatible with the USI A-46 assessment being conducted in parallel with the IPEEE work. The Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure (GIP)[13] used for USI A-46 allowed for coordination of activities to support both projects. The conservative deterministic failure margin (CDFM) approach was used to calculate the HCLPF capacities of components that did not screen out of evaluation by the SQUG GIP[13] methodology. The results of this assessment determined that the plant-level HCLPF was greater than the Review Level Earthquake (RLE) of 0.3g PGA. Therefore, the IHS is anchored at the NUREG/CR-0098[11] RLE of 0.3g PGA.

Figure B.2-1 provides a comparison of the IHS, GMRS and SSE. The IHS is a NUREG/CR-0098[11] spectral shape for a soil site anchored at a 0.3g PGA. The IHS completely bounds the GMRS between

0.1 Hz and 20 Hz. The plot shows that the GMRS slightly exceeds IHS in the 20 to 45 Hz spectral range. The IHS demonstrates that the plant has margin for the new GMRS. Between 1 Hz and 10 Hz, the GMRS has a peak acceleration of 0.45g at 5 Hz. The GMRS has a second peak of 0.48g at 13 Hz. Both peaks are bounded by the IHS, which has a peak acceleration between 1 and 10 Hz of 0.64g.

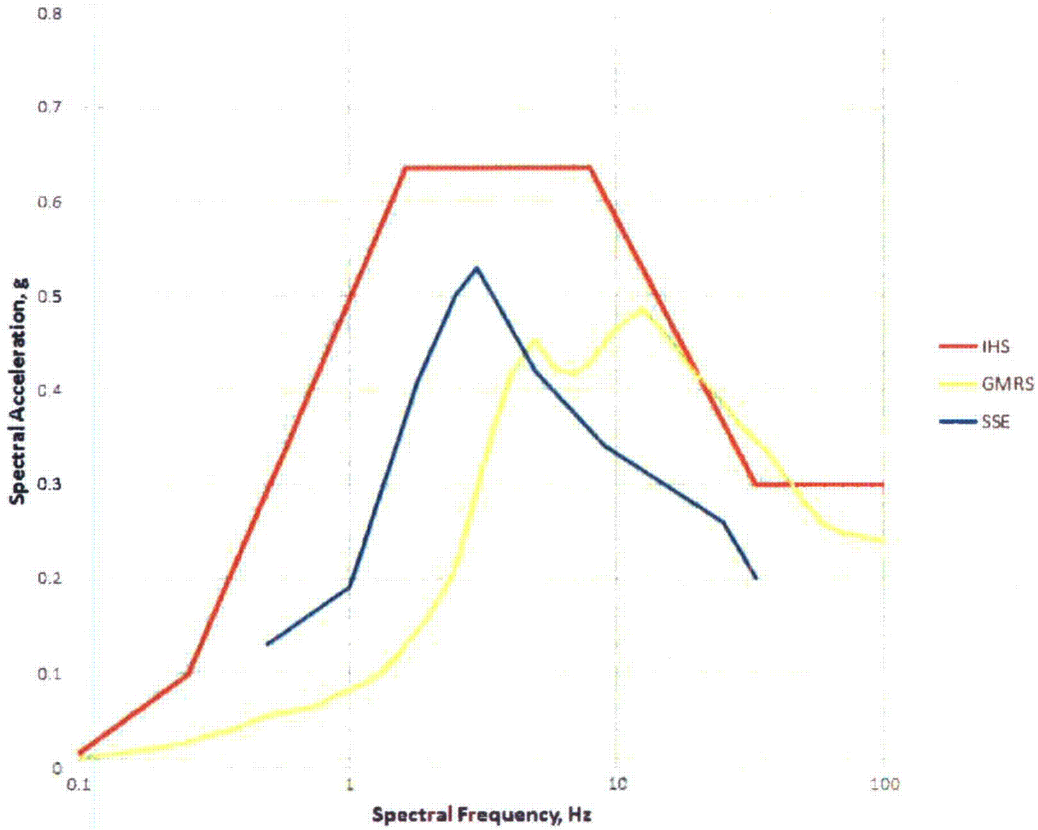

Figure B.2-1 IHS, GMRS and SSE for CNS (5 Percent Damping)

Modifications that were required to achieve the plant-level HCLPF and the confirmation that these modifications are still in place are described in Section B.3.0.

B.2.1 Relay Chatter

The CNS relay evaluation for IPEEE was consistent with the requirements of a focused scope evaluation, as described in NUREG-1407[1]. A full scope detailed review of relay chatter will be performed as an enhancement of the relay evaluation consistent with the requirements of SPID[2], Section 3.3.1. The relay chatter evaluation will be performed on the schedule provided in NEI letter[23] to NRC dated October 3, 2013.

B.2.2 Soil Failure Evaluation

Based on the updated assessment of liquefaction, soil failure from liquefaction will not occur in the structural fill for the CNS IHS with a PGA of 0.3g. Additionally, liquefaction in the native soils will not impact equipment required for safe shutdown. Other soil failure mechanisms such as failure of dams, levees, and dikes; building interaction; and buried structures on the IPEEE SMA equipment list were previously evaluated in the IPEEE. Details of the soil failure evaluation are provided in Appendix C. This evaluation was performed to NUREG-1407[1], Section 3.2.4.3 full scope criteria.

The following sections summarize the results of the IPEEE adequacy evaluation according to the guidance of the SPID[2].

B.3.0 Prerequisites

The following items have been addressed in order to use the IPEEE analysis for screening purposes and to demonstrate that the IPEEE results can be used for comparison with the GMRS:

1. Confirmation that commitments made under the IPEEE have been met.
2. Confirmation that all of the modifications and other changes credited in the IPEEE analysis are in place.
3. Confirmation that any identified deficiencies or weaknesses to NUREG-1407[1] in the CNS IPEEE SER included in NRC letter[7] are properly justified to ensure that the IPEEE conclusions remain valid.
4. Confirmation that major plant modifications since the completion of the IPEEE have not degraded/impacted the conclusion reached in the IPEEE.

Confirmation of IPEEE Commitments

In accordance with the CNS IPEEE report[5], six items were found to have a seismic capacity of less than 0.3g PGA. The table from Chapter 3 of the CNS IPEEE report[5] is presented below as Table B.3-1. All six of these items were identified as plant-specific vulnerabilities (seismically weak components) in the IPEEE program. A description of the actions taken to eliminate or reduce these vulnerabilities is provided in Attachment B2 as part of Table 2 from NPPD calculation NEDC 98-045[6].

The NRC engaged Brookhaven National Laboratory (BNL) to perform a Technical Evaluation Report (TER) on the CNS IPEEE report[5]. In Section 1.3 of the BNL TER report included in the SER in NRC letter[7] it is stated, in regards to IPEEE and USI A-46 outliers, that "five of these items were resolved under the USI A-46 programs and were determined to have a seismic capacity at least equal to 0.3g RLE specified for IPEEE. The last item consists of two vibration-isolated air handling systems, for which a high confidence of low probability of failure (HCLPF) capacity of 0.21g was calculated. In response to an RAI [RAI response[12]], the licensee stated that subsequent system analysis removed the air handling systems from the Safe Shutdown Equipment List (SSEL)."

"The licensee concluded that the CNS is seismically rugged and generally capable of withstanding the 0.3g RLE. In response to an RAI [RAI response[12]], the licensee stated that all exceptions have been resolved, and the plant's HCLPF is at least 0.3g."

With the resolution of identified vulnerabilities and outliers in the IPEEE analysis, the commitments made under the IPEEE have been met.

Table B.3-1 USI A-46 and IPEEE Outliers

CAPACITY (PGA)	DESCRIPTION
(not calculated)	*480V Critical Switchgear 1G (EE-SWGR-4160G)*: This switchgear did not screen because it contains essential (chatter-sensitive) relays and is subject to impact due to an adjacent concrete beam No criteria exists that defines the level of impact at which relay chatter occurs, so a seismic capacity cannot be calculated. This equipment is on the A-46 SSEL - this issue will be dispositioned as part of the A-46 outlier resolution process.
(not calculated)	*Aux Relay Room Panels (LRP-PNL-(9-32, 33, 41, 42 ,45))*: These cabinets were not screened because they contain essential (chatter sensitive) relays, and there are gaps of up to 1/2" between the base of the cabinets and the floor. Because of the gaps, the base of the cabinet could pound against the floor and cause relay chatter. This equipment is on the A-46 SSEL - this issue will be dispositioned as part of the A-46 outlier resolution process.
(not calculated)	*Jet Pump Instrument Rack A (LRP-PNL-(25-51))* This welded steel instrument rack is adjacent - but not attached - to a smaller rack. There is a gap of about 1/4" between the two racks, and the subject rack contains essential (chatter sensitive) relays, so it was not screened. No criteria exists that defines the level of impact at which relay chatter occurs, so a seismic capacity cannot be calculated. This equipment is on the A-46 SSEL - this issue will be dispositioned as part of the A-46 outlier resolution process.
(not calculated)	*Solatron/Accuvolt Line Conditioners (EE-XFMR-RPS1A, B)*: These transformers anchored to the outside of the control building wall, which is a Class I structure, but project into the adjacent Multi-Purpose Facility, which is a Class II structure. The equipment did not screen because there are building components nearby that could fall on the equipment. This equipment is on the A-46 SSEL - this issue will be dispositioned as part of the A-46 outlier resolution process.
0.14g	*Raceway Support, Reactor Building 903'*: From the A-46 raceway walkdowns and analytical reviews, the worst-case support is a braced Unistrut trapeze frame in the NE corner of elevation 903' of the reactor building (suspended from elevation 931') The hanger has a dead weight of approximately 5000 lbs and is anchored with two 1/2" shell anchors This equipment is on the A-46 SSEL - this issue will be dispositioned as part of the A-46 outlier resolution process.
0.21g	*SE and NE Quad Recirculation Fans (HV-FAN-(FC-R-1E and F))*: These Air Handling Units did not screen because they are mounted on vibration isolators A seismic capacity was calculated by requiring no net uplift on the isolators. This equipment is not on the A-46 SSEL and its resolution will be handled in the IPEEE Issue Resolution Plan described in Section 7 0.

Confirmation of Modifications and Changes Credited in IPEEE Analysis

All modifications and changes credited in IPEEE analysis, including outlier resolutions, have been addressed and resolved per the BNL TER included in the SER in NRC letter[7]. The CNS IPEEE report[5] states that all of the outliers in the IPEEE analysis that are also on the USI A-46 SSEL "will be dispositioned as part of the USI A-46 outlier resolution process." The USI A-46 report[8] provides documentation of outlier resolution.

Table 3.2 of the USI A-46 report[8] gives a list of USI A-46 outliers as well as the recommended resolutions from the seismic review team (SRT). In the SER in NRC letter[7], it was concluded that five of the six outliers in the USI A-46 program relative to IPEEE were resolved and were determined to have a capacity of at least equal to the 0.3g PGA. In an RAI response[12], CNS stated that subsequent system analysis removed the sixth outlier from the SSEL. Therefore, the SER in NRC letter[7] concluded that the CNS IPEEE is complete with regards to the information that was requested by Supplement 4 to Generic Letter 88-20[15].

Confirmation that Any Deficiencies/Weaknesses to NUREG-1407 in the Plant-Specific SER Are Justified to Keep IPEEE Valid

The SER in NRC letter[7] indicates that the IPEEE conclusions submitted by CNS are consistent with the guidance provided in NUREG-1407[1]. There are no identified deficiencies or weaknesses to NUREG-1407[1]. Therefore, the CNS IPEEE report[5] is complete with regards to the information requested by Supplement 4 to Generic Letter 88-20[15] and the guidance specified in NUREG-1407[1].

Confirmation that Major Plant Modifications Have Not Impacted Conclusions from IPEEE

As part of the resolution for NTTF Recommendation 2.3: Seismic, walkdowns were performed of new or replacement equipment installed since the completion of the seismic IPEEE evaluations. The walkdowns covered a sample of new or replacement equipment noted on the seismic walkdown equipment list (SWEL). The Seismic Walkdown Report[9] included in NPPD letter to NRC dated November 27, 2012 identifies seven new or replaced pieces of equipment included in the walkdown. Only one of the seven items did not meet the seismic walkdown checklist (SWC) criteria; this outlying item required a condition report. Condition report CR-CNS-2012-06657 for the outlier confirmed that the item of concern does not affect the safe shutdown of the plant under adverse seismic conditions.

As indicated in Section 9.1 of the Seismic Walkdown Report[9], the SWEL adequately demonstrates "a diversity in system types, both major new and replacement equipment, and diversity in types of equipment and environments." Upon review of the Seismic Walkdown Report[9] and the SWC presented in Attachment C therein, it was concluded that the major new and modified equipment at CNS have not degraded/impacted the conclusions reached in the CNS IPEEE report[5].

B.4.0 Adequacy Demonstration

The following sections summarize the adequacy demonstration according to the guidance of the SPID[2].

B.4.1 Structural Models and Structural Response Analysis

In response to Supplement 4 to Generic Letter 88-20[15], new soil-structure interaction (SSI) analyses were performed for the Control Building and the Reactor Building to generate mediancentered floor response spectra. NEDC 87-162[10] contains three levels of reviews: operating basis earthquake (OBE), SSE, and RLE. Because the OBE and SSE are not applicable to NTTF Recommendation 2.1: Seismic screening, only the RLE SSI analyses performed in support of IPEEE are evaluated in this report. The information presented in this section is adapted from the NEDC 87-162[10] unless noted otherwise.

Methodology

The SSI analyses were performed with the substructure method using the software program CLASSI. CLASSI models the structure as surface-founded on a rigid basemat. To account for embedment of CNS structures, a pre-processor, SUPELM, was used to model embedment effects relative to impedance and scattering. The SSI analyses were separated into the following segments:

- Specify free-field ground motion.
- Define soil profile and perform site response analysis.
- Calculate foundation impedance functions.
- Calculate foundation input motion (scattering).
- Determine fixed-base dynamic characteristics of the structure.
- Combine previous steps to calculate response of the coupled SSI system.

<u>Specify Free-Field Ground Motion</u>

Because the buildings at CNS are founded on structural fill, the ground motion used was NUREG/CR-0098[11] median soil spectrum anchored to 0.3g PGA. The spectrum was assumed to represent the motion on the free surface of the soil at elevation 902 feet according to the CNS IPEEE report[5].

<u>Define Soil Profile and Perform Site Response Analysis</u>

The best estimate (BE) low-strain soil properties are shown in Table B.4-1 and are documented in Table 1 of NEDC 87-162[10]. The IPEEE BE low-strain profile is similar to the profile shown in Table 2.3.1-1a in the CNS Seismic Hazard and Screening Report.

Table B.4-1 IPEEE Best Estimate Low-Strain Soil Properties

Layer No.	Elevation, MSL (ft)	Unit Weight (kcf)	Poisson Ratio	Shear Modulus (ksf)	Shear Wave Velocity (ft/s)	P-Wave[1] Velocity (ft/s)	Thickness (ft)
1	902.0-895.0	0.130	0.390	1266.0	560.0	1319.0	7.0
2	895.0-890.0	0.130	0.390	1796.0	667.0	1571.0	5.0
3	890.0-885.0	0.130	0.390	2235.0	744.0	1752.0	5.0
4	885.0-880.0	0.130	0.390	2597.0	802.0	1889.0	5.0
5	880.0-875.0	0.130	0.489	2849.0	840.0	5800.0	5.0

Table B.4-1 (continued)

Layer No.	Elevation, MSL (ft)	Unit Weight (kcf)	Poisson Ratio	Shear Modulus (ksf)	Shear Wave Velocity (ft/s)	P-Wave(1) Velocity (ft/s)	Thickness (ft)
6	875.0-870.0	0.130	0.489	3000.0	862.0	5800.0	5.0
7	870.0-865.0	0.130	0.488	3148.0	883.0	5800.0	5.0
8	865.0-860.0	0.130	0.488	3292.0	903.0	5800.0	5.0
9	860.0-855.0	0.130	0.487	3425.0	921.0	5800.0	5.0
10	855.0-850.0	0.130	0.486	3667.0	953.0	5800.0	5.0
11	850.0-845.0	0.130	0.485	3885.0	981.0	5800.0	5.0
12	845.0-840.0	0.130	0.485	3973.0	992.0	5800.0	5.0
13	840.0-835.0	0.130	0.485	4062.0	1003.0	5800.0	5.0
14	835.0-830.0	0.130	0.484	4143.0	1013.0	5800.0	5.0
15	830.0-822.0	0.130	0.484	4258.0	1027.0	5800.0	8.0
16	Below 822.0	0.130	0.390	72923.0	4250.0	10000.0	Inf.

(1)Groundwater elevation was defined at elevation 880.0 feet.

A site response analysis was performed with the computer program SHAKE to develop strain-compatible soil properties. The shear modulus and damping curves for sand used in the analysis were generated by EERC-84/14[16]. The mean curves were used for the analysis. The control point of the RLE motion applied in SHAKE was ground surface (elevation 902 feet).

The SHAKE analysis was performed only for the BE soil profile. The strain-compatible shear modulus values for lower bound (LB) and upper bound (UB) were determined by multiplying the BE values by 0.67 and 1.5, respectively. The strain-compatible damping values for LB and UB were set equal to BE. The BE strain-compatible soil properties are presented in Table B.4-2 and are documented in EQE calculations 50130-C-004[19] and 50130-C-005[20].

Table B.4-2 IPEEE Best Estimate Strain-Compatible Soil Properties

Layer No.	Elevation, MSL (ft)	Unit Weight (kcf)	Poisson Ratio	Shear Modulus (ksf)	Shear Wave Velocity (ft/s)	Damping Coeff.	Thickness (ft)
1	902.0-895.0	0.130	0.390	1037.2	506.9	0.043	7.0
2	895.0-890.0	0.130	0.390	1247.9	556.0	0.066	5.0
3	890.0-885.0	0.130	0.390	1456.6	600.7	0.074	5.0
4	885.0-881.0	0.130	0.390	1617.5	633.0	0.080	4.0
5	881.0-879.0	0.130	0.440	1648.8	639.1	0.083	2.0
6	879.0-875.0	0.130	0.489	1680.0	645.1	0.086	4.0
7	875.0-869.5	0.130	0.489	1672.8	643.7	0.092	5.5
8	869.5-865.0	0.130	0.488	1672.8	643.7	0.097	4.5
9	865.0-861.0	0.130	0.488	1676.3	644.4	0.102	4.0
10	861.0-859.0	0.130	0.488	1684.5	646.0	0.104	2.0
11	859.0-855.0	0.130	0.487	1692.7	647.5	0.105	4.0
12	855.0-850.0	0.130	0.486	1825.2	672.4	0.105	5.0
13	850.0-845.0	0.130	0.485	1962.3	697.2	0.103	5.0
14	845.0-840.0	0.130	0.485	2010.9	705.8	0.103	5.0
15	840.0-835.0	0.130	0.485	2070.9	716.2	0.102	5.0
16	835.0-830.0	0.130	0.484	2142.3	728.4	0.100	5.0
17	830.0-822.0	0.130	0.484	2244.3	745.6	0.098	8.0
18	822.0-807.0	0.130	0.390	71808.7	4217.4	0.008	15.0
19	807.0-792.0	0.130	0.390	71254.0	4201.1	0.009	15.0
20	792.0-777.0	0.130	0.390	70782.8	4187.2	0.010	15.0

Calculate Foundation Impedance Functions

The Control Building foundation impedance was determined using CLASSI by modeling the basemat dimensions and soil layers. The bottom of the basemat is at elevation 869 feet 6 inches and was treated as a surface founded structure. Embedment effects were neglected.

The Reactor Building foundation impedance was determined using SUPELM by modeling the basemat dimensions and soil layers. Partial embedment was considered by modeling the basemat at elevation 849 feet 9 inches and ground surface at Elevation 887 feet 4 inches. The Reactor Building foundation was treated as partially embedded due to excavation of adjacent structures.

Each building's frequency-dependent foundation impedance was determined for LB, BE, and UB soil profiles. However, impedance analyses were not performed directly for RLE strain-compatible LB, BE, and UB soil profiles. Instead, the factors in Table B.4-3 were applied to the results of impedance analyses performed for each building with BE SSE strain-compatible soil profiles. The factors in Table B.4-3 are documented in the EQE calculations 50130-C-004[19] and 50130-C-005[20].

Table B.4-3 IPEEE Impedance and Scattering Functions Scaling Factors

Soil Profile	Shear Modulus	Damping Coefficient
Best Estimate (BE)	0.83	1.22
Lower Bound (LB)	0.55	1.22
Upper Bound (UB)	1.25	1.22

Calculate Foundation Input Motion (Scattering)

The scattering analysis for the Control Building foundation was performed using SUPELM by modeling the basemat dimensions and soil layers. Embedment was considered by modeling the bottom of foundation at elevation 869 feet 6 inches and ground surface at elevation 902 feet.

The scattering analysis for the Reactor Building foundation was performed using SUPELM by modeling the basemat dimensions and soil layers. Partial embedment was considered by modeling the bottom of foundation at elevation 849 feet 9 inches and ground surface at elevation 887 feet 4 inches. The Reactor Building foundation was treated as partially embedded due to excavation of adjacent structures.

Each building's frequency-dependent scattering analysis was determined for LB, BE, and UB soil profiles. However, scattering analyses were not performed directly for RLE strain-compatible LB, BE, and UB soil profiles. Instead, the factors in Table B.4-3 were applied to the results of scattering analyses performed for each building with BE SSE strain-compatible soil profiles.

Determine Fixed-Base Dynamic Characteristics of the Structure

A 3-D lumped-mass-stick model was developed, and subsequent response analyses were performed for each of the Control Building and Reactor Building.

The Control Building was modeled with three "sets of sticks." The first set of sticks was for the concrete portion of the structure. It consisted of four vertical beam elements representing walls between floors and eight horizontal rigid links. The vertical sticks were located at the centers of rigidity between every two adjacent floors, and the mass of each floor was located at the gravity center. The eccentricities between the centers of mass and rigidity at each floor were connected with horizontal rigid links to include torsional effects. The second and third sets of sticks were for the two steel emergency condensate storage tanks located at the foundation level, and each stick included one vertical element and one mass point only. A uniform damping level of 7 percent was used for RLE throughout the structure. The tributary mass of each floor was lumped and modeled at the mid-thickness of each floor slab. The floor-to-floor stiffness calculations of each stick included all significant walls. Uncracked concrete properties were used in the analysis.

The Reactor Building was modeled with four "sets of sticks" representing the following:

1. Reactor building structure.
2. Drywell.
3. Sacrificial shield wall and pedestal.
4. Reactor vessel and internals.

The Reactor Building stick model consisted of multiple non-coaxial vertical beam elements located at centers of rigidities between different floors. The mass was located at the center of gravity at each floor, and the eccentricities between centers of mass and rigidity were connected with horizontal rigid links to include torsional effect. The tributary mass of each floor was lumped and modeled at the mid-thickness of each floor slab. The floor-to-floor stiffness calculations of each stick included all significant walls.

The stick models of drywell, sacrificial shield wall and pedestal, and reactor vessel and internals were all composed of coaxial vertical beam elements representing floor stiffness. The sticks were located along the vertical center of the reactor vessel since all these structures were symmetric about the center. A sufficient number of mass points along the vertical body of these structures were modeled to capture dynamic behavior of the structures. The support was located at the mid-thickness of the basemat slab at elevation 854 feet 9 inches at the center of the reactor vessel.

Due to the different structural material used on various parts of the Reactor Building, different damping levels were used: 7 percent for reinforced concrete building, steel superstructure, and reinforced concrete pedestal; 4 percent for drywell, shield wall, and reactor vessel; and 3 percent for reactor piping. Therefore, composite modal damping was calculated to capture the contribution of these different materials on each mode of the structure. Uncracked concrete properties were used in the analysis.

<u>Combine Previous Steps to Calculate Response of the Coupled SSI System</u>

SSI analyses were performed separately using CLASSI for the Control Building and Reactor Building by combining free-field ground motions, foundation impedance, foundation input motion (scattering), and fixed-base structure models.

CLASSI solves the equation of motion in the frequency domain at selected frequencies. The Control Building analyses were solved at 31 evenly distributed frequencies ranging from 0.1 Hz to 30.1 Hz and at a frequency increment of 1.0 Hz as documented in EQE calculation 50130-C-006[17]. The Reactor Building analyses were solved at 66 evenly distributed frequencies ranging from 0.05 Hz to 32.55 Hz and at a frequency increment of 0.5 Hz as documented in EQE calculation 50130-C-008[18].

Compliance with NUREG-1407

The RLE was a NUREG/CR-0098[11] median soil spectrum anchored at 0.3g. The ground motion derived from this shape was applied at the ground surface. The effects of embedment and soil-structure interaction were considered. Therefore, the applicable guidance of NUREG-1407[1] was followed.

Adequacy for Screening

The IPEEE methodology and results relative to structural modeling and structural response analysis are adequate for screening purposes.

B.4.2 In-Structure Demands and ISRS

Methodology

The SSI analyses discussed in Subsection B.4.1 produced acceleration time histories at the center of mass for each floor slab elevation, as well as the drywell and shield wall nodes of the Reactor Building; responses at the floor slab mass centers were then transformed to corner locations for enveloping purposes.

Response spectra for the RLE were calculated with spectral damping ratios of 3 percent, 5 percent, 7 percent, and 10 percent. Response spectra for each direction were enveloped over all locations for each floor. The enveloped spectra were further enveloped to include the BE, LB, and UB soil cases resulting in the final response spectra for the RLE seismic input.

Compliance with NUREG-1407

In-structure demand and in-structure response spectra (ISRS) are the output of structural modeling and structural response analysis. Therefore, the applicable guidance of NUREG-1407[1] was followed.

Adequacy for Screening

The IPEEE methodology and results relative to in-structure demand and ISRS are adequate for screening purposes.

B.4.3 Selection of Seismic Equipment List (SEL)/Safe Shutdown Equipment List (SSEL)

Methodology

The selection of equipment for the SSEL in the CNS IPEEE report[5] was developed in accordance with the guidance of NUREG-1407[1] and NP-6041-SL[3]. Consistent with this guidance, the CNS IPEEE SSEL was developed using the safe shutdown paths and equipment identified for USI A-46. The CNS IPEEE SSEL includes the same systems and components credited by the USI A-46 SSEL with the addition of systems required for the alternate paths, plus components necessary to mitigate a small break loss of coolant accident (SBLOCA) and evaluate containment performance.

During the BNL review of the CNS IPEEE report[5], NRC developed an RAI[24] regarding the use of low-pressure injection systems only and requested the basis for not including the high-pressure systems High-Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) in the SSEL. As noted in Section 2.4 of the BNL TER included in the SER in NRC letter[7], the RAI response[12] clarifies the rationale for selecting the low pressure injection system for the success paths. The RAI response[12] states that based on conservative reasoning it may be operationally desirable to depressurize during a postulated RLE; therefore, the Automatic Depressurization System (ADS) was included in both success paths. The BNL TER included in the SER in NRC letter[7] indicates that the response to the RAI is reasonable and that the equipment in the SSEL can provide two success paths for safe shutdown under transient and small SBLOCA conditions.

As noted in the RAI response[12], there were six items that were indicated as having an HCLPF value that was below the 0.3g PGA level. Of these six items, five were on the USI A-46 SSEL, and the outlier resolution was completed. The sixth item was heating, ventilating, and air conditioning (HVAC) equipment in the southeast and northeast quadrants (HV-FAN-FC-R-1E and HV-FAN-FC-R-1F), and in the RAI response[12], CNS clarified that these components can be removed from the SSEL. The basis for removing them is that the plant shutdown can be achieved with only one Residual Heat Removal (RHR) System pump, and room cooling is not needed for running a single RHR pump; therefore, the room coolers are not required. The resolution of the outliers is also noted in Section 2.13 of the BNL TER included in the SER in NRC letter[7].

Compliance with NUREG-1407

As noted in Section 3.0 of the BNL TER included in the SER in NRC letter[7], the CNS IPEEE report[5] is consistent with the guidance of NUREG-1407[1], and the completeness of the documentation is a strength of the submittal.

Adequacy for Screening

The CNS IPEEE SSEL was developed in accordance with the guidance of NUREG-1407[1] and the selection of equipment on the SSEL was previously discussed with the NRC in an RAI response[12]; therefore, the CNS IPEEE SSEL is acceptable for screening purposes.

B.4.4 Screening of Components

Methodology

The seismic margin methodology (SMM) used by CNS to screen components followed the methodology described in NP-6041-SL[3]. This approach was selected to take advantage of the similarities between the NP-6041-SL[3] methodology for evaluating seismic margin and the methodology previously utilized by CNS to address USI A-46. The evaluation and screening of mechanical and electrical equipment relied heavily on the USI A-46 walkdowns; in general, equipment that meets USI A-46 requirements also meets the NP-6041-SL[3] screening criteria.

For the seismic IPEEE, NUREG-1407[1] specifies CNS to be a 0.3g PGA focused scope plant. As such, the objective of the SMA was to evaluate the capacity of plant components relative to a 0.3g PGA RLE. The evaluated components included the structures, equipment, and distribution systems that were considered necessary to achieve and maintain safe shutdown from a normal plant operating condition for at least 72 hours. The evaluation followed the NP-6041-SL[3] procedure and consisted of first determining which components can be screened for a 0.3g PGA RLE; components that did not screen were considered outliers, and a specific seismic capacity was calculated for such components.

Most of the components in the plant screened based on the criteria summarized in Tables 2-3 and 2-4 of NP-6041-SL[3] for the 0.3g PGA RLE. Components that satisfied the screening criteria were screened out from further evaluation. If those criteria were not met, then seismic capacity for a component was calculated.

RLE floor response spectra were developed for the Control Building and Reactor Building in calculation NEDC 87-162[10]. In general, the SSE spectra enveloped the RLE spectra. This confirms the expectation that the original seismic analysis was conservative and that the acceleration levels for the 0.2g PGA SSE are higher than the acceleration levels computed using SSI techniques for the RLE. The results from the SSI analysis were sufficient to state that the structures, equipment, and distribution systems at CNS designed for the 0.2g PGA SSE can be screened for the 0.3g PGA RLE.

Compliance with NUREG-1407

The floor response spectra for the RLE were enveloped at frequencies greater than 2 Hz by the design basis SSE floor response spectra used for the USI A-46 evaluations. Additionally, the evaluation considered that spectral values below 2 Hz do not affect anchorage. Thus, equipment that satisfied the USI A-46 anchorage requirements were also screened for the 0.3g PGA RLE.

The SQUG GIP[13] methodology used to address USI A-46 conforms to the screening criteria of NUREG-1407[1].

Adequacy for Screening

The screening of components performed for IPEEE is adequate for screening purposes.

B.4.5 Walkdowns

Methodology

Walkdown activities conducted by CNS to address USI A-46 utilized the walkdown procedures illustrated in the SQUG GIP[13]. Screening Evaluation Worksheets (SEWS) were used during the plant walkdown to document the results of the evaluation and screen components from further evaluations. The SEWS incorporated the following four screening guidelines to verify the seismic adequacy of an item of equipment: (1) seismic capacity compared to seismic demand; (2) caveats; (3) anchorage; and (4) seismic interaction. A combination of screening verifications and walkdowns were performed to populate the SEWS and verify the seismic adequacy of active mechanical and electrical equipment.

NUREG-1407[1] states in Subsection 3.2.4.1 that "a walkdown should be performed and documented in accordance with the recommendations contained in EPRI NP-6041-SL"[3]. However, in Subsection 6.3.3.3 of NUREG-1407[1], it is concluded that plants that use the SQUG GIP[13] methodology for walkdowns also meet the NUREG-1407[1].

The walkdown seismic review team for both the USI A-46 scope and the IPEEE consisted of two seismic capability engineers with experience in Civil/Structural and Mechanical engineering. The walkdown team included at least one licensed professional engineer. Plant staff also provided assistance during walkdowns. The walkdown team members were trained in SQUG GIP[13] methodologies for seismic evaluation and screening.

Compliance with NUREG-1407

The walkdown analysis conducted to address USI A-46 also satisfied the IPEEE criteria as specified in NUREG-1407[1].

Adequacy for Screening

The walkdowns performed for IPEEE are adequate for screening purposes.

B.4.6 Fragility Evaluations

Methodology

The CNS IPEEE report[5] was an EPRI SMA; therefore, seismic fragilities were not calculated. The CNS IPEEE report[5] used a CDFM approach to determine the plant-level HCLPF capacity.

The intent of the CNS IPEEE report[5] was to determine plant-level HCLPF seismic capacity at the RLE. The SQUG GIP[13] methodology utilized at CNS to address USI A-46 enveloped the seismic margin methodology described in NP-6041-SL[3]. Using SQUG GIP[13] methodology for walkdown and screening, a majority of the equipment was screened from further analysis. For components and equipment that did not screen, a HCLPF capacity was calculated for each component using the CDFM approach. The capacity of the weakest component determined the plant-level HCLPF capacity. The guidance in Subsection 3.2.4.6 of NUREG-1407[1] allows for a CDFM approach for computing component and plant HCLPF values.

A total of six items were identified in the CNS IPEEE report[5] as seismically weak components that did not meet the screening criteria of 0.3g PGA RLE. These components are listed in Table B.3-1. Five of the items were addressed under the USI A-46 program per an RAI response[4]. The sixth item, the southeast and northeast Quad Recirculation Fans (HV-FAN-FC-R-1E and HV-FAN-FC-R-1F), was removed from the SSEL as stated in an RAI response[12]. Resolution of these items is provided in Attachment B2. With the resolution of these six items, the plant-level IPEEE HCLPF spectrum (IHS) was determined to equal the RLE spectrum with a 0.3g PGA and a NUREG/CR-0098[11] shape. The IHS is shown on Figure B.2-1.

Compliance with NUREG-1407

The seismic capacity and component screening methodology was conducted in accordance with the guidance in NUREG-1407[1].

Adequacy for Screening

The methodology for seismic capacity and screening was conducted in accordance with the guidance in NUREG-1407[1] and thus, provides an HCLPF capacity and IHS that are adequate for screening purposes.

B.4.7 System Modeling

Methodology

The success paths used in the CNS IPEEE report[5] were developed using the methodology in Section 3 of NP-6041-SL[3]. This is indicated in the BNL TER included in the SER in NRC letter[7]. Plant-specific success path logic diagrams (SPLD) were not developed for the CNS IPEEE report[5], and this was also noted in the BNL TER included in the SER in NRC letter[7]. However, there was no indication that the lack of a plant-specific SPLD impacted the technical adequacy of the CNS IPEEE report[5]. Section 3 of NP-6041-SL[3] states that the path selection process should only consider paths which the plant operators would use based on procedures, training, and available instrumentation and indicators. The purpose of the plant-specific SPLD is to develop an SSEL that uses equipment and procedures for which operators are trained. Subsection 3.1.2.1.7 of CNS IPEEE report[5] described how the SSEL was reviewed using a procedure review and a simulator validation to conclude that the SSEL is comprehensive and appropriate. Therefore, although a plant-specific SPLD is not included in the CNS IPEEE report[5], the success paths available using the SSEL have been confirmed to be those that operators are trained to utilize and the intent of the SPLD is satisfied.

The success paths for CNS only use low-pressure injection systems, and the high-pressure systems are not credited. During the BNL review of the CNS IPEEE report[5], an RAI[24] was generated regarding the use of low-pressure injection systems only. As noted in Section 2.4 of the BNL TER included in the SER in NRC letter[7], the RAI response[12] clarifies the rationale for selecting the low-pressure injection system for the success paths. The RAI response[12] states that based on conservative reasoning it may be operationally desirable to depressurize during an RLE; the ADS was included in both success paths. The BNL TER included in the SER in NRC letter[7] indicates that the RAI response[12] is reasonable and that the equipment in the SSEL can provide two success paths for safe shutdown under transient and SBLOCA conditions.

In the CNS IPEEE report[5], the non-seismic failures and human actions were evaluated using the CNS Probabilistic Safety Assessment (PSA) model. The turbine trip combined with a loss of off-site power was modeled in the CNS PSA using only the SSEL systems to predict the plant post-seismic reliability. The non-seismic-related conditional plant response was estimated to be 7.17E-03/day using the modeled event sequence. As stated in the CNS IPEEE report[5], the post-event reliability provides assurance that the overall post-seismic reliability of CNS is high when also considering the low recurrence frequency of the RLE; therefore, the event-related core damage frequency (CDF) is low. Since the quantified plant response included some credited human actions, CNS performed an additional assessment where the conditional plant response was modified by overstating the human probability factors by two orders of magnitude. The results of the assessment showed that the conditional response was increased by only 2% over the 7.17E-03/day value previously indicated. Therefore, CNS concluded that the post-seismic event human actions are not significant.

Section 2.10 of the BNL TER included in the SER in NRC letter[7] indicates that manual depressurization of the reactor is required for both success paths, and the effects of operator failure to depressurize needed to be discussed, and an RAI[24] was generated requesting this information. In an RAI response[12], CNS clarified that the ADS would act automatically if the high-pressure systems were to malfunction during a seismic event. As noted in Section 2.10 of the BNL TER included within the SER in NRC letter[7], the issue was addressed through the RAI response[12], which also concluded that the non-seismic failures and human actions are adequately addressed in the CNS IPEEE report[5].

Compliance with NUREG-1407

The system modeling methodology was conducted in accordance with the guidance in NUREG-1407[1] and provides an SSEL that has at least two success paths, and adequately addresses non-seismic failures and human actions.

Adequacy for Screening

The methodology used is in compliance with NUREG-1407[1] and the IPEEE system modeling results are adequate for screening purposes.

B.4.8 Containment Performance

Methodology

CNS is a boiling water reactor (BWR) Type 4 Mark I containment, and both the steel drywell and the Torus (suppression pool) were screened for the 0.3g PGA RLE in accordance with the CNS IPEEE report[5]. The drywell is penetrated by two equipment hatches and a personnel hatch, and the hatches do not rely on inflatable seals or other active means for function. In addition to the hatches, there are multiple piping, instrumentation line, and electrical penetrations. The mechanical and electrical penetrations typically consist of a steel pipe welded to the drywell that passes through a sleeve cast in the bioshield wall. The drywell hatches and penetrations were walked down for potentially vulnerable conditions (e.g., spatial interactions, unique penetration configurations), and none were found according to the CNS IPEEE report[5].

Components required for containment isolation are included in the SSEL, and a relay review was also performed. The containment isolation systems reviewed were the Primary Containment Isolation System, Main Steam Isolation Valves, Reactor Building Heating and Ventilation Systems, and Standby Gas Treatment System. As noted in Subsection 3.1.5.1 of the CNS IPEEE report[5], the systems were analyzed for their containment isolation function and not for their accident mitigation functions.

Compliance with NUREG-1407

The containment performance evaluation was performed in accordance with the guidance in NUREG-1407[1].

Adequacy for Screening

The methodology used for containment performance in the IPEEE is adequate for screening purposes.

B.4.9 Peer Review

Methodology

The guidance in NUREG-1407[1] and Supplement 4 to Generic Letter 88-20[15] required that the CNS IPEEE report[5] receive a peer review by individuals not associated with the initial evaluation to evaluate and ensure the accuracy of the documentation and to validate the process and results of the CNS IPEEE report[5]. NUREG-1407[1] clarified that the review was not intended to be a detailed review or that is was to be performed in accordance with 10 CFR 50, Appendix B. The guidance of NUREG-1407[1] clarified that the review was a critical review that validates the process, the methodologies, and results.

The peer review of the CNS IPEEE report[5] included the following components:

1. Review by competent and responsible in-house CNS engineering and operations personnel.
2. Review by a competent person with broad overview experience.
3. Review by outside persons that included personnel from other nuclear utilities undergoing similar efforts and outside consultants.

The peer review team is shown in Table B.4-4.

Table B.4-4 IPEEE Peer Review Team

Name	Area of Expertise
Mr. Jim Moody	Senior BWR Level 1/2 Probabilistic Safety Assessment and IPEEE Consultant Fire and Seismic Probabilistic Safety Assessment Expert
Mr. Karl N. Fleming	Senior External Events Risk Analyst IPEEE Consultant Fire and Seismic Probabilistic Safety Assessment Expert
Mr. David A. Bidwell	External Events Probabilistic Safety Assessment Expert
Mr. David R. Buttemer	Seismic Margins Method Expert
Mr. Greg Kruger	Utility Level 1/2 and IPEEE Technical Consultant Boiling Water Reactors Owners Group (BWROG) Probabilistic Safety Assessment Lead

The peer review generated a set of comments and resolutions. The comments and resolutions related to the seismic evaluation are provided in Attachment B1, Table B1-1.

Compliance with NUREG-1407

The peer review for the IPEEE report followed the guidance provided in NUREG-1407[1].

Adequacy for Screening

The peer review for the IPEEE report is adequate for screening purposes.

B.5.0 Conclusion

The CNS IPEEE was a focused scope SMA submittal and requires the performance of a detailed review of relay chatter and full evaluation of soil failures to be considered a full scope SMA. A soil failure analysis has been completed with satisfactory results and is provided in Appendix C. A relay evaluation consistent with a full scope IPEEE, as described in NUREG-1407[1], will be performed on the schedule provided in NEI letter[23] to NRC dated October 3, 2013.

Based on the IPEEE adequacy review performed in accordance with the guidance contained in SPID[2] and documented herein, with the exception of the completion of the detailed relay chatter review, the results of the CNS IPEEE report[5] are considered adequate for screening and the risk insights gained from the IPEEE remain valid under the current plant configuration.

B.6.0 References

1. U.S. Nuclear Regulatory Commission, *Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities*, NUREG-1407, NRC, Washington, D.C., June 1991.

2. Electric Power Research Institute, *Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic*, 1025287, EPRI, Palo Alto, California, February 2013.

3. Electric Power Research Institute, *A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1)*, NP-6041-SL, EPRI, Palo Alto, California, August 1991.

4. Nebraska Public Power District (John H. Swailes), Letter NLS980192 to U.S. Nuclear Regulatory Commission (Document Control Desk), "Response to Request for Additional Information Related to USI A-46 and Notification of Outlier Resolution," January 25, 1999.

5. Nebraska Public Power District (G. R. Horn), Letter NLS960143 to U.S. Nuclear Regulatory Commission (Document Control Desk), "Individual Plant Examination for External Events (IPEEE) Report - 10 CFR 50.54(f)," October 30, 1996.

6. Nebraska Public Power District, *SQUG USI A-46 Outlier Resolutions*, NEDC 98-045, Rev. 0.

7. U.S. Nuclear Regulatory Commission (Mohan C. Thadani), Letter to Nebraska Public Power District (John H. Swailes), "Cooper Nuclear Station – Review of Individual Plant Examination of External Events (TAC NO. 83611)," April 27, 2001.

8. Nebraska Public Power District (G. R. Horn), Letter NLS960076 to U.S. Nuclear Regulatory Commission (Document Control Desk), "Submittal of the Unresolved Safety Issue (USI) A-46 Summary Report," June 13, 1996.

9. Nebraska Public Power District (Brian J. O'Grady), Letter NLS2012125 to U.S. Nuclear Regulatory Commission (Document Control Desk), "Seismic Walkdown Report - Nebraska Public Power District's Response to Nuclear Regulatory Commission Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Seismic Aspects of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," November 27, 2012.

10. Nebraska Public Power District, *CNS Frequency versus Acceleration Response Spectra Curves*, NEDC 87-162, Rev. 5.

11. U.S. Nuclear Regulatory Commission, *Development of Criteria for Seismic Review of Selected Nuclear Power Plants*, NUREG/CR-0098, NRC, Washington, D.C., May 1978.

12. Nebraska Public Power District (John H. Swailes), Letter NLS990008 to U.S. Nuclear Regulatory Commission (Document Control Desk), "Response to Request for Additional Information – Individual Plant Examination for External Events (IPEEE)," January 28, 1999.

13. Seismic Qualification Utility Group, *Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment*, GIP-2, February 14, 1992.

14. Not Used.

15. U.S. Nuclear Regulatory Commission (James G. Partlow), Letter to all Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Individual Plant Examination of External Events (IPEEE) For Severe Accident Vulnerabilities – 10 CFR 50.54(f) (Generic Letter No. 88-20, Supplement 4)," June 28, 1991.

16. Seed et al., *Moduli and Damping Factors for Dynamic Analysis of Cohesionless Soils*, Earthquake Engineering Research Center Report No. UCB/EERC-84/14, University of California, Berkeley, September 1984.

17. EQE International, *Control Building A-46 (SSE/OBE) and IPEEE Analysis*, 50130-C-006, Prepared for Nebraska Public Power District, Rev. 0.

18. EQE International, *Reactor Building A-46 (SSE/OBE) and IPEEE Analysis*, 50130-C-008, Prepared for Nebraska Public Power District, Rev. 0.

19. EQE International, *Control Building Impedances*, 50130-C-004, Prepared for Nebraska Public Power District, Rev. 0.

20. EQE International, *Reactor Building Impedances*, 50130-C-005, Prepared for Nebraska Public Power District, Rev. 0.

21. Burns & Roe, Inc., "Structural Control Building Foundation Plan & Sections," Drawing 4171, Prepared for Nebraska Public Power District, Rev. N01.

22. U.S. Nuclear Regulatory Commission (James R. Hall), Letter to Nebraska Public Power District (G. R. Horn), "Request for Additional Information Related to Unresolved Safety Issue A-46, Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Cooper Nuclear Station (TAC. No. M69439)," October 7, 1998.

23. Nuclear Energy Institute (Kimberly A. Keithline), Letter to U.S. Nuclear Regulatory Commission (David L. Skeen), "Relay Chatter Reviews for Seismic Hazard Screening," October 3, 2013.

24. U.S. Nuclear Regulatory Commission (James R. Hall), Letter to Nebraska Public Power District (G. R. Horn), "Request for Additional Information Related to the Individual Plant Examination of External Events (IPEEE) for the Cooper Nuclear Station (TAC No. M83611)," June 3, 1998.

Attachment B1 - IPEEE Peer Review Comments and Resolution

Table B1-1 IPEEE Peer Review Comments and Resolutions

No.	Comment	Resolution
S-1	Section 3.1.1.1 notes that: "the RLE floor response spectra (FRS) in Figure 3.1.2 are lower than the design basis SSE FRS used in the A-46 evaluations. Therefore, equipment that satisfied the A-46 anchorage requirements was also screened to the 0.3 pga RLE." It was noted that the RLE exceeds the SSE for frequencies <2 Hz. This comment is repeated for a similar assertion in Subsection 3.1.3.1. Please justify the acceptability of this difference.	Comment noted and clarification added to the section regarding the caveat to the assertion that the SSE FRS used in the A-46 evaluations bound the RLE FRS. The clarification indicates an exception below 2 Hz and indicates that the spectral values below 2 Hz do not affect the anchorage evaluations because the equipment generally has a fundamental frequency well above 2 Hz.
S-2	Subsection 3.1.1 indicates that the damping values used were 5% (OBE) and 7% (DBE) for concrete structures, and 2% (OBE and DBE) for steel frame structures. Please provide references for the values used.	The reference is the CNS USAR, and an appropriate revision was made to the report.
S-3	Subsection 3.1.1.4 - Hydrodynamic loads indicate that the configuration of the suppression chamber/drywell structure supports an assumption that hydrodynamic loads do not need to be considered. It is noted that if an RLE were to occur, the main turbine would likely trip immediately due to vibration signals, the steam bypass valves would likely close on loss of condenser vacuum on loss of off-site power to the circulating water pumps, and safety relief valve (SRV) air clearing/hydrodynamic loads would probably occur during the period of strong ground. In view of this reasonable scenario, is the assumption regarding exclusion of hydrodynamic loads from consideration reasonable?	It is acknowledged that SRV discharge will in all likelihood occur during the earthquake as a result of the turbine trip due to loss of off-site power. This will, as indicated in the comment, result in hydrodynamic loads. The subject paragraph is intended to state that these loads were considered in the SMA evaluation of the torus, but because the torus is founded directly on the reactor building foundation mat and has expansion joints in the vent pipes that connect it to the drywell, vibrations due to hydrodynamic effects were not considered for any other components in the plant.
S-4	The SSEL used for the SMA should be compared to the components in the Level I Individual Plant Examination (IPE) as a means to ensure that the scope was accurate.	A review of the SMA SSEL against the CNS PSA model was performed, and no significant differences were noted in the scope. The CNS PSA was used in the evaluation of non-seismic failures and human actions portion of the SMA.

No.	Comment	Resolution
S-5	Seismic capacity calculation in Subsection 3.1.4.2 for the SE and NE Quad Recirculation Fans concludes that the seismic capacity is .21g. This infers an issue requiring resolution, but the text is not clear on this. Please clarify.	The intent of the seismic IPEEE is to determine the plant's seismic capacity, high confidence low probability of failure (HCLPF), which for the SMA approach is the seismic capacity of the weakest component. The procedure for an SMA is to pick a Review Level Earthquake (RLE), which has been specified by the NRC to be a 0.3g 0098 shape for CNS, "screen" most of the components in the plant for the RLE based on the guidelines in NP-6041, and then calculate specific capacities, HCLPFs, for the components that did not screen. The weakest component then becomes the plant HCLPF. The table in Subsection 3.1.1, lists the six items that did not screen for the 0.3g RLE. Five of the six are A-46 outliers and can be expected to screen once the outliers are resolved. The sixth item is the quad fans, which are not on the A-46 list, and have a calculated capacity of 0.21g. The conclusion from the SMA is that once the A-46 outliers are resolved, the CNS seismic capacity, HCLPF, is 0.21g. There is no regulatory requirement that all equipment screen for the RLE and, since the quad fans are not on the A-46 SSEL, there is no requirement to evaluate them as part of the A-46 outlier resolution process. Clarification regarding the above has been added to the submittal document.
S-6	Subsection 3.1.4.5 indicates under the evaluation of control room panel anchorage issues for Panels LRP-PNL-(9-27) and (9-28) that the RLE accelerations are 1.01g horizontally and 0.38g vertically; whereas, a review of Figure 3.1.2 indicates that the peak is about 0.8g. Please explain the difference.	The panels are in the control room, which is located on elevation 932' of the control building. The corresponding horizontal RLE floor spectrum is shown in the upper right hand plot of Figure 3.1.3. This spectrum has a peak of 1.01g at a frequency of approximately 3 Hz. This value is conservative, because the cabinets have a fundamental frequency greater than 3 Hz, but was used because the cabinets screened for the 0.3g RLE using this value. However, the reviewer is correct, there is an error in the calculation. Correcting this error using the current inputs for the calculation would result in a capacity of less than 0.3g, but the error can be corrected and the capacity kept above 0.3g using a less conservative input acceleration. This revision is reflected in the revised text for this section.

No.	Comment	Resolution
S-7	A review of the capacity calculation for the panels cited in Comment S-6 indicated the potential comments noted on the markup. Please resolve the noted issues.	See response to S-6.
S-8	Subsection 3.1.4.5 indicates that the hydrogen cooling system for the turbine generator is acceptable relative to the impact to SMA equipment. Notwithstanding this conclusion, it is reasonable to assume that H_2 will leak from the system in this event. Is it necessary to evaluate the SMA equipment for accumulation of H_2 at levels of greater than 4% and the potential for a consequential explosion.	This issue is evaluated under NPPD's evaluation of Generic Safety Issue 106 and is included in the IPEEE submittal Subsection 4.9.1.
S-9	The discussion in Subsection 3.1.4.5 indicates that the sprinklers at CNS are not subject to inadvertent actuation. The text applies to preaction sprinklers. In view of the recent California earthquake experience with fire sprinklers actuating over sensitive equipment (the Whittier earthquake), it may be prudent to expand the basis for this conclusion.	The SRT concluded that there are only two ways that the sprinklers could release water: (1) either the sprinkler heads break off; in the case of pre-action sprinklers, this still would not result in the discharge of water without some other coincident failure, or (2) the earthquake causes the heat detection elements to "fail" in a way that results in a false heat signal and a consequent release of water. As noted in Table 3-3, the SRT examined the sprinkler piping and judged it not vulnerable to seismically-induced failure. The SRT also examined the heat detection elements and judged them not to be seismically vulnerable.

Attachment B2 - Outlier Resolution Tables from NPPD Calculation NEDC 98-045

Table 1. Relay Outliers from Reference 6 pg 3

Relay Model	Relay IDs	Cabinets	Outlier Issue / Recommended Resolution
AGASTAT E7022PC	EE-REL-(27X3-1F) EE-REL-(27X3-1G)	EE-SWGR-4160F EE-SWGR-4160G	**Outlier Issue:** Demand (3.9g / 2.5g) > Capacity (4.0g / 1.6g) **Resolution:** The test report that is the source for the relay GERS capacity was reviewed. The relay GERS is a conservative lower bound of the actual test response spectrum. The actual test response spectrum envelops the seismic demand. See Section 2.1 for details.
BARTON 288A	NBI-PIS-52B NBI-PIS-52D	LRP-PNL-(25-52B) LRP-PNL-(25-6)	**Outlier Issue:** No seismic capacity data. **Resolution:** A seismic test report was located in CNS' equipment qualification files. The report shows that the relay's seismic capacity exceeds the demand. See Section 2.2 for details.
DYNALCO RT2347	DG-RT-3142 DG-RT-3143	DG-PNL-DG1(ECP) DG-PNL-DG2(ECP)	**Outlier Issue:** No seismic capacity data. **Resolution:** A seismic test report was located in CNS' equipment qualification files. The report shows that the relay's seismic capacity exceeds the demand. See Section 2.3 for details.
GE 12CEH51A1A	DG-REL-DG1(40) DG-REL-DG2(40)	DG-PNL-DG1(GCP) DG-PNL-DG2(GCP)	**Outlier Issue:** Low ruggedness relay. **Resolution:** A seismic qualification test report was obtained for a diesel generator control cabinet at another utility with similar control circuitry using the same relay. This report shows that the relay's seismic capacity exceeds the demand. See Section 2.4 for details.
GE 12CFD12B2A	DG-REL-DG1(87)A DG-REL-DG1(87)B DG-REL-DG1(87)C DG-REL-DG2(87)A DG-REL-DG2(87)B DG-REL-DG2(87)C	DG-PNL-DG1(GCP) DG-PNL-DG2(GCP)	**Outlier Issue:** Low ruggedness relay. **Resolution:** In-cabinet spectrum calculated both in an NPPD calculation and using GIP Screening Level 3 shows adequate margin with respect to seismic capacity from GE test data referenced in USNRC Information Notice 85-82. See Section 2.5 for details.
GE 12HFA151A	EE-REL-(27X-1F) EE-REL-(27X-1G) EE-REL-(27X8-1F) EE-REL-(27X8-1G) EE-REL-(27XX-1F) EE-REL-(27XX-1G)	EE-SWGR-4160F EE-SWGR-4160G	**Outlier Issue:** Demand (3.9g / 2.5g) > Capacity (3.0g / 1.5g) **Resolution:** The demand was originally calculated as 7x the floor response spectrum (GIP Screening Level 2). The demand was recalculated using a finite element model of the panels housing the relays (GIP Screening Level 4). The capacity is greater than the recalculated demand. See Section 2.6 for details.

Relay Model	Relay IDs	Cabinets	Outlier Issue / Recommended Resolution
GE 12IAV73A1A	DG-REL-DG1(27-59) DG-REL-DG2(27-59)	DG-PNL-DG1(GCP) DG-PNL-DG2(GCP)	**Outlier Issue:** No seismic capacity data. **Resolution:** A seismic qualification test report was obtained for a diesel generator control cabinet at another utility with similar control circuitry using the same relay. This report shows that the relay's seismic capacity exceeds the demand. See Section 2.7 for details.
Static-O-Ring 12TA-BB4-NX	PC-PS-101A PC-PS-101B PC-PS-101C PC-PS-101D	LRP-PNL-(25-5) LRP-PNL-(25-6)	**Outlier Issue:** Demand (2.5g / 1.6g) > Capacity (3.0g / 1.5g) **Resolution:** The original seismic capacity was based on the EPRI GERS for pressure switches, which is a lower bound envelope for a number of switches from different manufacturers. A seismic test report for this specific pressure switch was located in CNS' equipment qualification files. The report shows that the switch's seismic capacity exceeds the demand. See Section 2.8 for details.
Static-O-Ring 9TA-B4-NX	NBI-PS-52A2 NBI-PS-52C2	LRP-PNL-(25-5) LRP-PNL-(25-51)	**Outlier Issue:** Demand (2.9g / 1.8g) > Capacity (3.0g / 1.5g) **Resolution:** The original seismic capacity was based on the EPRI GERS for pressure switches, which is a lower bound envelope for a number of switches from different manufacturers. A seismic test report for this specific pressure switch was located in CNS' equipment qualification files. The report shows that the switch's seismic capacity exceeds the demand. See Section 2.9 for details.
YARWAY 4418C	NBI-LIS-72A NBI-LIS-72B NBI-LIS-72C NBI-LIS-72D	LRP-PNL-(25-5) LRP-PNL-(25-6)	**Outlier Issue:** No seismic capacity data. **Resolution:** This outlier is resolved because the Static-O-Ring outliers have been resolved. If the Static-O-Ring pressure switches do not "chatter", then chatter of the Yarway is not a concern.

Table 2. Unresolved Outliers – Seismic Evaluation of Equipment from Reference 6 pg 5

Equipment	Outlier Issue / Recommended Resolution
Control Room Cabinets LRP-PNL-(9-15, 17, 18) LRP-PNL-(9-19, 21) LRP-PNL-(9-3, 4, 5) LRP-PNL-C, G, H, J	**Outlier Issue:** LRP-PNL-(9-27) and LRP-PNL-(9-28) are not anchored along the front. LRP-PNL-(9-10) and LRP-PNL-(9-14) are not anchored in the front. LRP-PNL-K is not anchored and the anchorage of the adjacent LRP-PNL-R is not visible. LRP-PNL-(9-38) is not anchored in the front. These cabinets are not on the seismic SSEL, but are in the same row as cabinets, which are on the seismic SSEL. **Resolution:** The recommended resolution was to upgrade the anchorage of these cabinets. CNS Modification Package MP-083B has been issued to do so. Supporting calculations are contained in CNS Calculation NEDC 96-042.
Control Room Cabinets (see list above)	**Outlier Issue:** The control room ceiling consists of a two level suspended system. The first level is approximately 36" below the concrete roof slab and is constructed of acoustic tile with gypsum board backing. The second level of ceiling is approximately 24" below the first and consists of a typical commercial construction 2' x 4' suspended ceiling grid supporting 2' x 4' aluminum "egg-crate" diffusers. The diffusers sit on the grid, but are not positively attached. The SRT judged that the ceilings are adequately supported; however, the diffusers, which are not attached to the grid, may dislodge and fall onto control panels or personnel below. **Resolution:** The recommended resolution was to secure the diffusers to the grid using plastic ties. CNS Modification Package MP-083B has been issued to do so. To ensure that the ties are replaced when panels are removed, the modification package required the development of new Procedure Number 7.2.79, "Control Room Eggcrate Ceiling Tiles (Light Diffusers) Installation and Removal".
Control Room Cabinets (see list above)	**Outlier Issue:** There is an unsecured work table with a copier adjacent to LRP-PNL-(9-15). The table and copier could slide or fall and strike the cabinet. There are 2 large unsecured storage cabinets in the corner just west of LRP-PNL-(9-19). **Resolution:** These items have been removed from the control room.

Equipment	Outlier Issue / Recommended Resolution
Aux Relay Room Cabinets LRP-PNL-(9-32, 33) LRP-PNL-(9-41, 42) LRP-PNL-(9-45)	**Outlier Issue:** These cabinets contain essential (chatter sensitive) relays. These cabinets, along with others, form two cabinet rows. There are a number of anchorage issues, including inadequate anchorage of 9-32 and 9-33, a missing anchor bolt in 9-45, and gaps at the base of a number of the panels (the GIP requires that cabinets containing essential relays have no gaps at the base). **Resolution:** The recommended resolution was to upgrade the anchorage of these cabinets. CNS Modification Package MP-003B has been issued to do so. Supporting calculations are contained in CNS Calculation NEDC 96-042.
EE-SWGR-4160G	**Outlier Issue:** The switchgear contains essential (chatter sensitive) relays. At the south end, the upper part of the switchgear abuts - but is not attached to - a reinforced concrete beam. "Pounding" between the wall and switchgear may cause the essential relays to chatter. There are lights hanging on chains behind the switchgear. The lights can swing, strike the switchgear, and cause the essential relays to chatter. **Resolution:** This outlier was resolved by (1) attaching a brace between the switchgear and the concrete beam to prevent pounding, and (2) relocating the lights. See CNS Modification Package MP 97-080 and CNS Calculation NEDC 97-082.
EE-XFMR-RPS1A EE-XFMR-RPS1B	**Outlier Issue:** These two small transformers are mounted on the outside of the control building wall and project into the Multi-Purpose Facility, which is a Class II structure. There is a steel staircase and platform adjacent and overhead, which are mounted on a structure that appears to be supported by masonry walls. The staircase, platform, and supporting structure are not safety related, so their ability to withstand the design basis seismic event is not assured. If they fell, they could damage the transformers. **Resolution:** These items have been replaced on the SSEL with other equipment. See Section 3.1.

Equipment	Outlier Issue / Recommended Resolution
LRP-PNL-(25-1)	**Outlier Issue:** This rack contains essential (chatter sensitive) relays. There is an adjacent rack that is not bolted to the subject rack. There is a gap of about 1/4" between the two racks; "pounding" of the racks may cause the essential relays to chatter. **Resolution:** This outlier has been resolved analytically by showing that the ¼" gap is acceptable. See Section 3.2.
LRP-PNL-S192	**Outlier Issue:** A bolt that secures the upper left corner of an interior panel to the panel case is loose. **Resolution:** The bolt was tightened per MWR 96-02085.

Table 3. Unresolved Outliers - Raceway Evaluations from Reference 6 pg 8

Outlier Issue / Recommended Resolution
Outlier Issue: Hanger 144 in the cable spreading room (Control Building 918) was selected as one of twelve representative, worst-case raceway supports for limited analytical review. The hanger failed the dead load check due to local overstresses in several of the crosspieces that span between posts. The overall hanger and hanger anchorages satisfy all checks. See Reference 2, Appendix D, LAR #3 for details. **Resolution:** This outlier has been resolved based on the discussion in Section 4.1.
Outlier Issue: Hanger 89 on elevation 903 in the Reactor Building was selected as one of twelve representative, worst-case raceway supports for limited analytical review. The hanger failed the dead load, vertical capacity, and lateral load checks due to the loads developed in the anchor bolts. See Reference 2, Appendix D, LAR #8 for details. **Resolution:** This outlier has been resolved by upgrading its anchorage and evaluating the surrounding similar hangers. See Section 4.2.

Appendix C - Soil Failure and Liquefaction Evaluation for IPEEE Adequacy Review

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C.1.0 Background

The Nuclear Regulatory Commission (NRC) staff issued Generic Letter 88-20[15], Supplement 4 on June 28, 1991, requesting that each licensee conduct an individual plant examination of external events (IPEEE) for severe accident vulnerabilities. Concurrently, NUREG-1407[1] was issued to provide utilities with detailed guidance for performance of the IPEEE.

A seismic margin assessment (SMA) was performed for the seismic portion of the Cooper Nuclear Station (CNS) IPEEE report[8] using the Electric Power Research Institute (EPRI) SMA methodology, NP-6041-SL[5], with enhancements identified in NUREG-1407[1]. CNS performed a 0.3g peak ground acceleration (PGA) focused scope SMA utilizing a NUREG/CR-0098[9] spectral shape for a soil site. The calculated plant-level high confidence of low probability of failure (HCLPF) for CNS resulting from performance of the IPEEE was 0.3g PGA.

The SPID[2] indicates IPEEE focused scope margin submittals may be used for screening after enhancement to bring the focused scope assessment in line with full scope assessments. One of the enhancements is a soil failure evaluation. As presented in NUREG-1407[1], CNS is a 0.3g PGA focused scope SMA plant. Therefore, enhancement with a soil failure evaluation in accordance with NP-6041-SL[5] and present day NRC requirements is required to use the IPEEE HCLPF Spectrum (IHS) for screening in accordance with the SPID[2].

C.2.0 Soil Failures Evaluation

The information presented in Section C.2.0 is a summary of the soil failure evaluation documented in NPPD calculation NEDC 14-022[16].

C.2.1 Review of Existing Liquefaction Analyses

NP-6041-SL[5] provides guidance on soil failure evaluations and states that if there were soil failure issues at the design level earthquake, then these same issues should be investigated for the IHS. The primary soil failure evaluation in the CNS USAR[3] was liquefaction. Specifically, CNS USAR II-X[3] states that liquefaction of the native (in situ) materials was determined to be likely based on an initial analysis using the liquefaction Safe Shutdown Earthquake (SSE) (the liquefaction SSE is described in the USAR[3] and is similar to the SSE except it is scaled to a PGA of 0.25g instead of 0.2g and has a longer duration). To mitigate the liquefiable soils, the native soils were excavated and CNS was built above compacted structural fill with a thin layer (7 to 8 feet thick) of compacted alluvium above the bedrock near elevation 820 feet. The thin layer of in situ material was left in place to avoid exposing the shale bedrock to potential degradation or weathering but was compacted in place before placement of the structural fill. Based on the liquefaction analysis described in the CNS USAR[3], the structural fill at CNS was compacted to a minimum average relative density (D_r) to prevent liquefaction. These minimum average relative densities were D_r = 85 percent from elevation 903 feet to 855 feet; D_r = 80 percent from elevation 855 feet to 830 feet; and D_r = 75 percent from elevation 830 feet to the bedrock surface at approximately elevation 820 feet.

The CNS USAR[13] results indicate the structural fill will not liquefy at a PGA of 0.25g at bedrock (approximately elevation 820 feet). The CNS USAR[13] documents the results from 73 borings with Standard Penetration Test (SPTs) on 2.5 foot intervals completed to confirm that the average relative density of the compacted alluvium below elevation 830 feet is at least 75 percent in all but the upper portion of the compacted alluvium. The CNS USAR[13] states that the upper portion of the in situ soils was compacted to the required average relative density during compaction of the first lift of structural fill and that the relative density below elevation 830 feet was always greater than 75 percent. In the structural fill, the CNS USAR[13] states that between elevation 855 feet and 830 feet, and above elevation 855 feet, the standard plate load tests and Washington Dens-O-Meter tests indicated approximately 1 percent of the tests were less than the required value. Seven borings with SPTs were performed after completion of the structural fill to confirm the as-built relative density. The CNS USAR[13] states the results verified that the as-built structural fill and in situ compacted material meet the required relative densities.

C.2.2 Enhanced Assessment of Liquefaction

The enhanced assessment of liquefaction and soil strength loss is completed by reanalysis of the structural fill using the IHS PGA of 0.3g. Empirical procedures for liquefaction analysis follow Regulatory Guide 1.198[11], NUREG/CR-5741[12], and the NCEER paper[7]. Specifically, five soil borings (B-1, B-1A, B-2, B-3 and SF-1) with SPT data and shear wave velocity measurements were evaluated. All of these data were collected between 1998 and 2012 in the structural fill, and the shear wave velocities were measured with a downhole suspension logging tool as discussed in NPPD calculation NEDC 13-019[4]. No cone penetration tests (CPTs) have been completed within the structural fill for liquefaction evaluation. See Attachment C1 of this report for locations of soil borings.

The liquefaction evaluation included corrections for the SPT hammer energy when energy measurements were available. If hammer energy measurements were not available, a hammer efficiency of 60 percent was used consistent with ASTM D6066-11[14]. Based on the CNS Engineering Criteria Document[6] used for construction of the structural fill, total unit weights of 134 pcf, 133 pcf, and 132 pcf were assumed for soils with an average relative density of 85, 80, and 75 percent, respectively. The fines content of the soils was based on descriptions in the soil boring logs, laboratory analyses of grain size from soil boring samples, and laboratory analyses of grain size for the structural fill in the CNS USAR[3]. Generally, the amount of fine-grained material (amount passing through a No. 200 sieve) in the structural fill was between 5 and 10 percent, and higher values were used only when laboratory analyses or soil descriptions in boring logs indicated a greater percentage.

The earthquake magnitude for the empirical liquefaction evaluation was based on previous analyses and a review of NUREG-2115[10]. Previous analyses for liquefaction at CNS assumed an earthquake magnitude of 7.0; however, review of NUREG-2115[10] for the recent Central and Eastern United States Seismic Source Characterization (CEUS SSC) model indicated that CNS is in the Midcontinent-Craton seismotectonic zone, the generic "Study Region" distributed seismicity zone, and the Non-Mesozoic (and younger) distributed seismicity zone. A review of the weighted magnitude distributions for these distributed seismicity zones and seismotectonic zones indicated weighted average magnitudes from about 6.7 to 7.3 for the different zones. Therefore, a magnitude of 7.5 was assumed for the CNS empirical liquefaction evaluation.

The liquefaction evaluation for the IHS considered two depths to groundwater: (1) the observed groundwater depth during completion of the soil boring; and (2) flood conditions with groundwater at the ground surface (water at grade).

Figure C.2-1 shows the calculated factor of safety (FS) against liquefaction for the structural fill material at the observed groundwater depth. The FS against liquefaction exceeds 1.4 – the value in Regulatory Guide 1.198[11] that could indicate potential soil strength loss – at all but five sample locations. Two of the samples at depth in boring B-1A are described in the soil boring logs in NPPD calculation NEDC 13-019[4] as possible native material in a boring completed near the edge of the excavation for placement of structural fill, and one of the shallow SPT values in boring B-1A is in clay according to the soil boring log in NPPD calculation NEDC 13-019[4]. The shallow SPT values in borings B-1A and B-2 are above the bottom of all foundations and the observed groundwater level and do not impose a threat to the foundations. The two deeper SPT samples in the native material do not represent the structural fill or in situ compacted material and do not indicate potential soil failure in the structural fill. Additionally, because clay is not liquefiable, the isolated shallow SPT value in boring B-1A does not indicate potential soil failure in the structural fill. The SPT values indicate liquefaction or soil strength loss will not occur for the IHS in the structural fill and compacted alluvium at the observed groundwater depths.

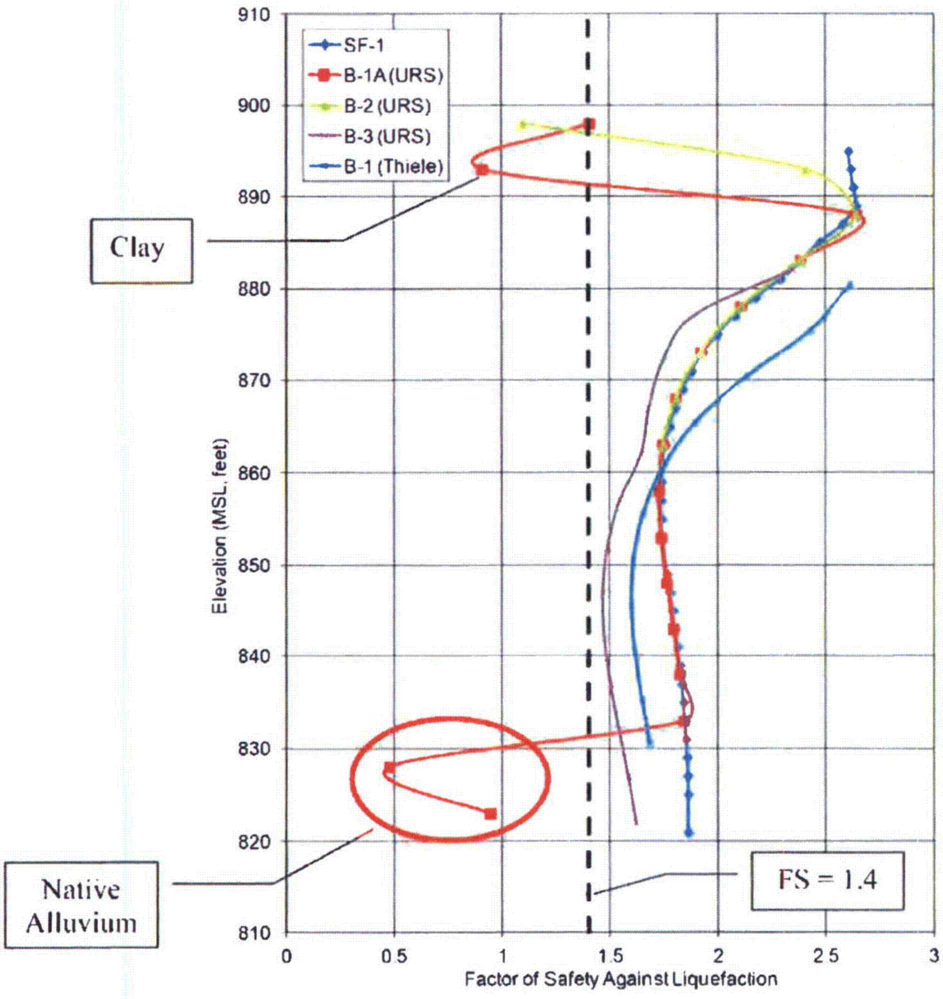

Figure C.2-1 **Factor of Safety Against Liquefaction for Structural Fill Material at Observed Groundwater Depth**

Figure C.2-2 shows the calculated FS against liquefaction for the flood groundwater depth (water at grade). The FS against liquefaction exceeds 1.4 at all but the same five sample locations identified above for the observed groundwater depths. The flood groundwater depth produces the lower FS against liquefaction but involves the less likely simultaneous occurrence of flood conditions with the IHS. Based on the empirical liquefaction evaluation, the SPT values indicate that liquefaction or soil strength loss will not occur for the IHS in the structural fill and compacted alluvium for the flood groundwater depth (water at grade).

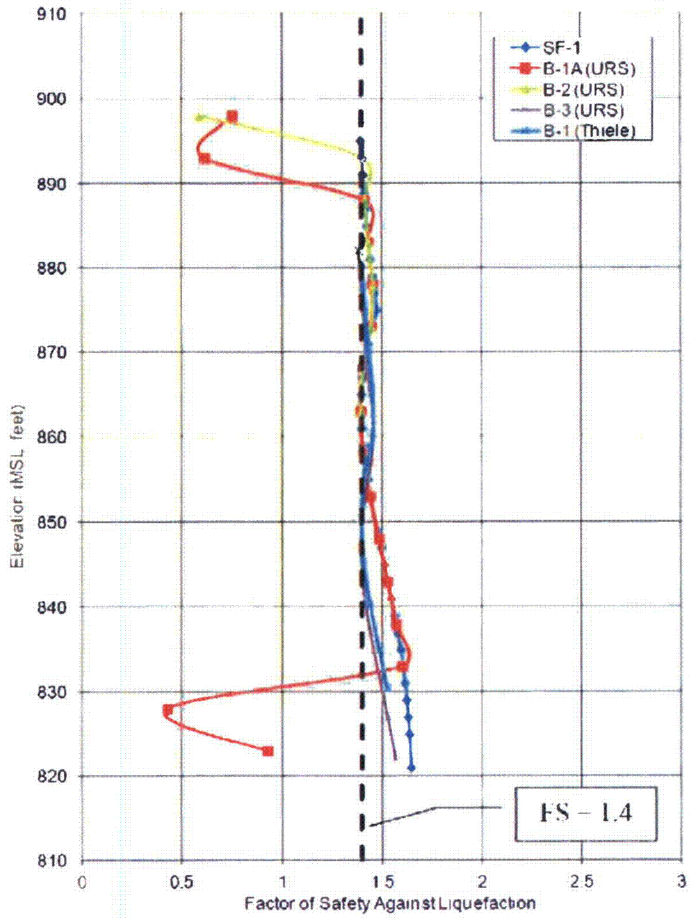

Figure C.2-2 Factor of Safety Against Liquefaction for Structural Fill Material at Flood Groundwater Depth (Water at Grade)

Evaluation of the measured shear wave velocities in the structural fill and compacted alluvium also indicate liquefaction will not occur at the observed or flood groundwater depths. In general, the overburden-corrected shear wave velocities (V_{S1}) based on the measured shear wave velocity data did not exceed the corresponding V_{S1}^* value. V_{S1}^* is a value that varies linearly from 656 ft/s to 705 ft/s for soils with fines contents of 35 to 5 percent, respectively, and represents the limiting upper value of V_{S1} for liquefaction occurrence as described in NCEER paper[7].

As a final check of the soil liquefaction results, the energy-corrected SPT N values (N_{60}) from the five soil borings completed between 1998 and 2012 in the structural fill were plotted against vertical effective stress in Figure C.2-3. These values were compared to Figure D-5-13 in the CNS USAR[13] to show that the structural fill and compacted alluvium met the average relative density requirements. The five soil borings completed between 1998 and 2012 are near the lower range of the verification soil borings presented in CNS USAR[13], Figure D-5-13, but indicate that the relative densities exceed the required values at the majority of the test locations.

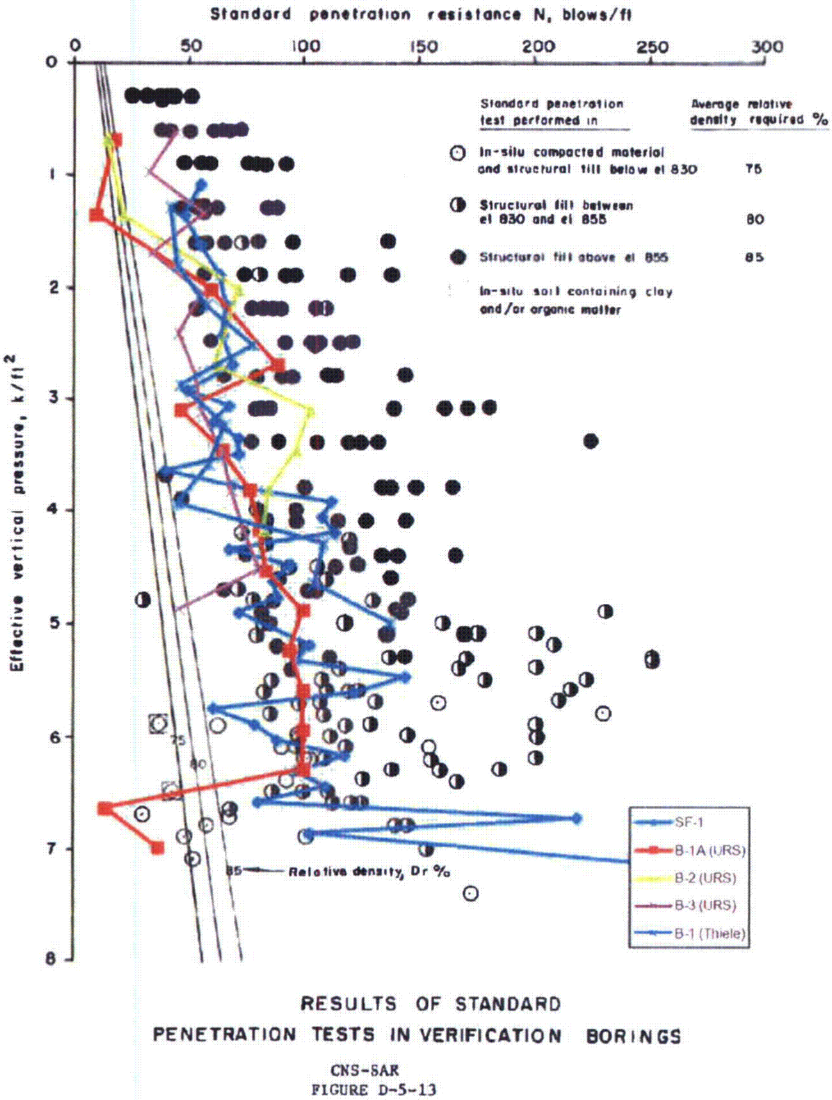

Figure C.2-3 **Comparison of Energy Corrected SPT Values (N_{60}) Versus Effective Vertical Pressure for Construction Verification Borings and 1998-2012 Soil Borings**

C.2.3 Native Alluvium Under Radwaste Building

As indicated in the CNS USAR[13], the below-grade portion of the Radwaste Building is a Class I structure not required for safe shutdown, and the above-grade portion of the Radwaste Building is a Class II structure. The eastern third of the Radwaste Building is founded partially on structural fill and compacted alluvium, and the western two-thirds are on structural fill over variable amounts of native alluvium. Structural fill exists above elevation 862 feet for the entire Radwaste Building area. The CNS USAR[13] indicates that borings completed in the Radwaste Building area indicated the relative density was 75 percent between elevations 862 feet and 845 feet for the native alluvium, and 65 percent below elevation 845 feet for the native alluvium. The liquefaction potential of the in situ material was evaluated for the liquefaction SSE in CNS USAR Appendix D[13] and indicated the resulting FS was above 1.0 at all elevations with a maximum FS of about 1.13 at elevation 845 feet. Based on a FS below 1.4, the native alluvium was also evaluated for the IHS.

Similar to the process for the structural fill and compacted alluvium, the assessment of liquefaction and soil strength loss was completed by reanalysis of the native alluvium using the IHS PGA of 0.3g and the empirical procedures for liquefaction analysis of Regulatory Guide 1.198[11], NUREG/CR-5741[12], and the NCEER paper[7]. Specifically, five soil borings (B-6, B-14, C-11, C-14 and RW-1) with SPT data and three shear wave velocity measurements from different locations across the site outside the structural fill were evaluated in NPPD calculation NEDC 13-019[4]. Three of the soil borings and all the shear wave velocity data – both downhole suspension logging and crosshole seismic data – were collected between 2006 and 2012. The two other soil borings were completed in 1967 during the original subsurface investigation for CNS. For boring locations see Figures C1-2 and C1-3 in Attachment C1 of this report. No CPTs have been completed in the native alluvium for liquefaction evaluation.

The liquefaction evaluation included corrections for the SPT hammer energy, when energy measurements were available. When structural fill was present above the native alluvium, the same total unit weights of 134 pcf, 133 pcf, and 132 pcf were used. For the native alluvium, a total unit weight of 125 pcf was assumed. The fines content of the soils was based on descriptions in the soil boring logs and laboratory analyses of grain size from soil boring samples. An earthquake magnitude of 7.5 was assumed based on the review of the CEUS SSC model in NUREG-2115[10] described previously. The depth to groundwater was either at the observed depth to groundwater during completion of the soil boring or at the ground surface to represent potential flood conditions.

Figure C.2-4 presents the calculated FS against liquefaction for the observed groundwater depths. The FS against liquefaction is typically below 1.0 when native alluvium was encountered. At isolated locations, more resistant and non-liquefiable soils are defined as FS of 1.0 or greater. However, soils with a FS between 1.0 and 1.4 have possible strength reduction (resistance) at some depths. Figure C.2-4 also presents the depth of the structural fill in soil boring RW-1, which was completed in 2012 adjacent to the northwest side of the Radwaste Building. Above approximately elevation 862 feet, structural fill is present around the Radwaste Building. The depth of the structural fill at elevation 869 feet in boring RW-1 is likely more shallow because of the slope of the excavation walls; however, structural fill that is not liquefiable is present above approximately elevation 862 feet in the Radwaste Building area.

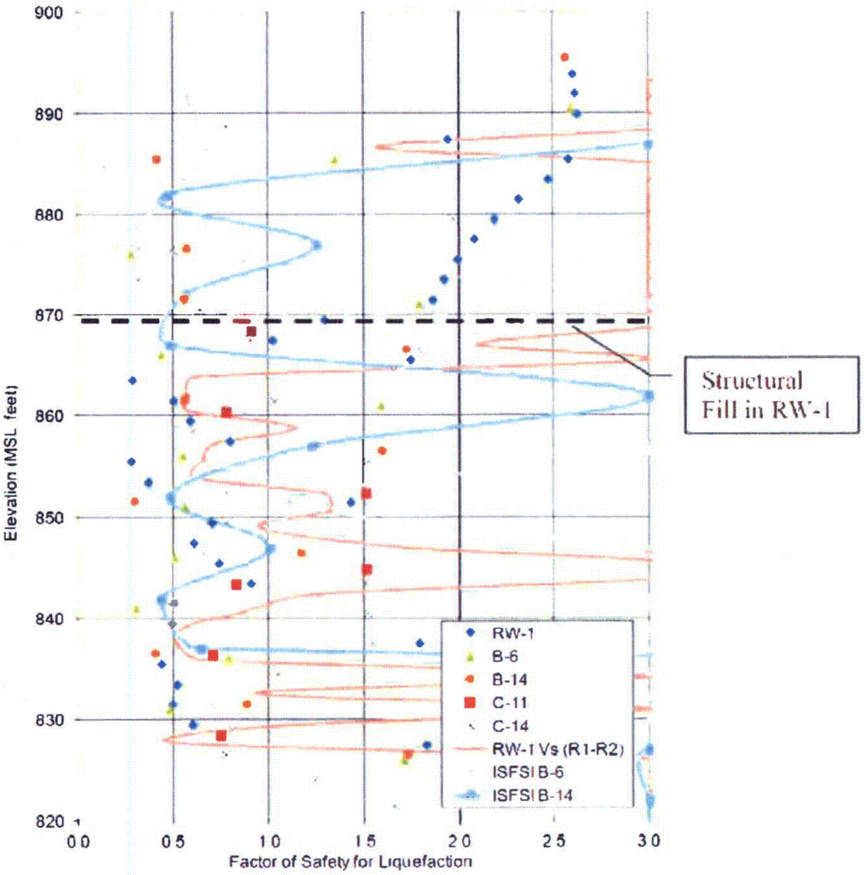

Figure C.2-4 Radwaste Building Factor of Safety Against Liquefaction for Structural Fill Material at Observed Groundwater Depth

Figure C.2-5 presents the calculated FS against liquefaction for the flood groundwater depth (water at grade). The FS against liquefaction is similar to the results for the observed groundwater depths and is typically less than 1.0 in the native alluvium. Based on the empirical liquefaction evaluation, the SPT values indicate liquefaction or soil strength loss will occur for the IHS in the native alluvium beneath the western portions of the Radwaste Building.

The FS against liquefaction based on the measured shear wave velocities is also presented on Figure C.2-4 and Figure C.2-5. The FS based on shear wave velocity measurements generally agree well with the SPT values. The crosshole seismic data from Independent Fuel Storage Installation (ISFSI) boring B-6 in NPPD calculation NEDC 13-019[4] is the only measurement that does not agree well with the SPT values. The FS against liquefaction for ISFSI boring B-6 are generally higher than those calculated with the other data and are not considered representative of the native alluvium.

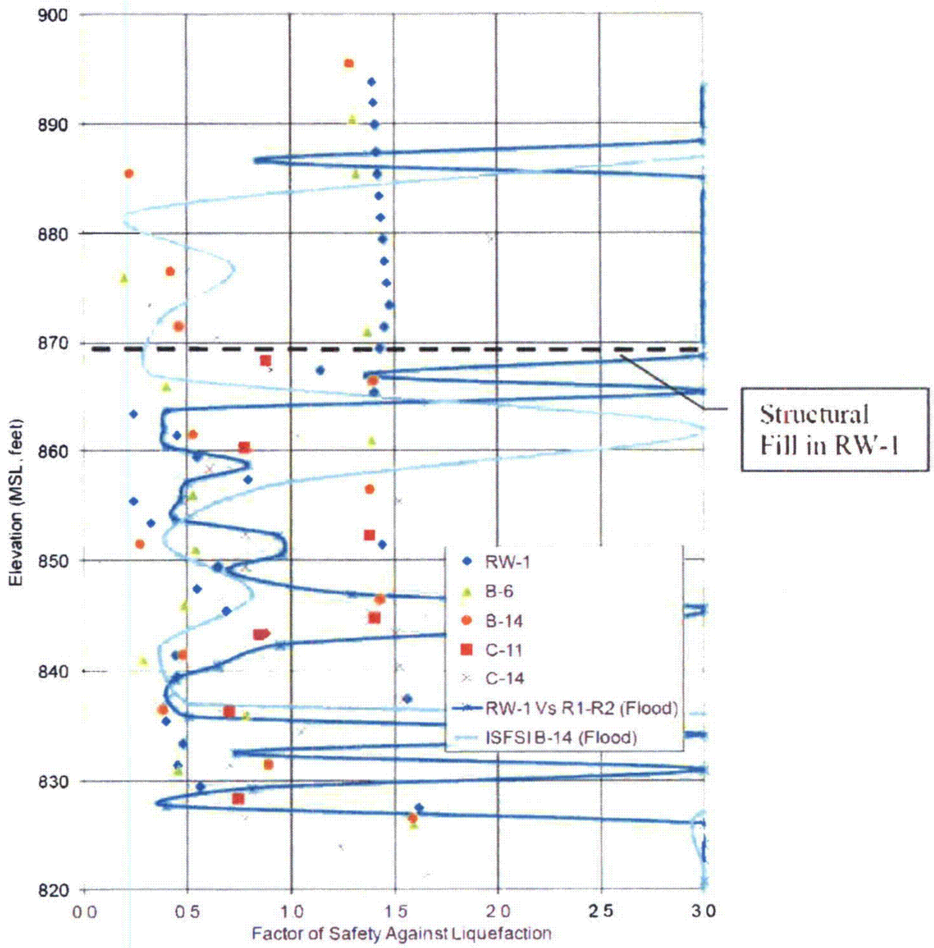

Figure C.2-5 **Radwaste Building Factor of Safety Against Liquefaction for Structural Fill Material at Flood Groundwater Depth (Water at Grade)**

As stated in the CNS USAR[3], the Radwaste Building is not required for safe shutdown of CNS. Additionally, none of the equipment on the IPEEE SMA equipment list is located in the Radwaste Building. Therefore, potentially liquefiable soils beneath the Radwaste Building do not prevent the safe shutdown of CNS. However, liquefaction of the soils beneath the Radwaste Building could create differential settlement that may result in rotation of the building and potentially create building interaction with structures that contain SMA equipment required for safe shutdown. However, the native alluvium is present only under the western portions of the Radwaste Building; therefore, the Radwaste Building would rotate away from the Control Building and Reactor Building, which are toward the east and southeast, and would not create building interaction.

C.3.0 Conclusion

Based on the updated assessment of liquefaction, soil failure from liquefaction will not occur in the structural fill for the IHS with a PGA of 0.3g. Additionally, liquefaction in the native soils will not impact equipment required for safe shutdown. Other soil failure mechanisms such as failure of dams, levees, and dikes; building interaction; and buried structures on the IPEEE SMA equipment list were previously evaluated in the IPEEE.

C.4.0 References

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6. Nebraska Public Power District, *CNS Engineering Criteria Document*, Rev. 2.

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15. U.S. Nuclear Regulatory Commission (James G. Partlow), Letter to all Licenses Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities – 10 CFR 50.54(f) (Generic Letter No. 88-20, Supplement 4)," June 28, 1991.

16. Nebraska Public Power District, *Owner Acceptance of Black and Veatch Calculation 51.1001 Soil Failure Evaluation*, NEDC 14-022, Rev. 1.

Attachment C1 - Soil Boring Locations from NPPD Calculation NEDC 13-019

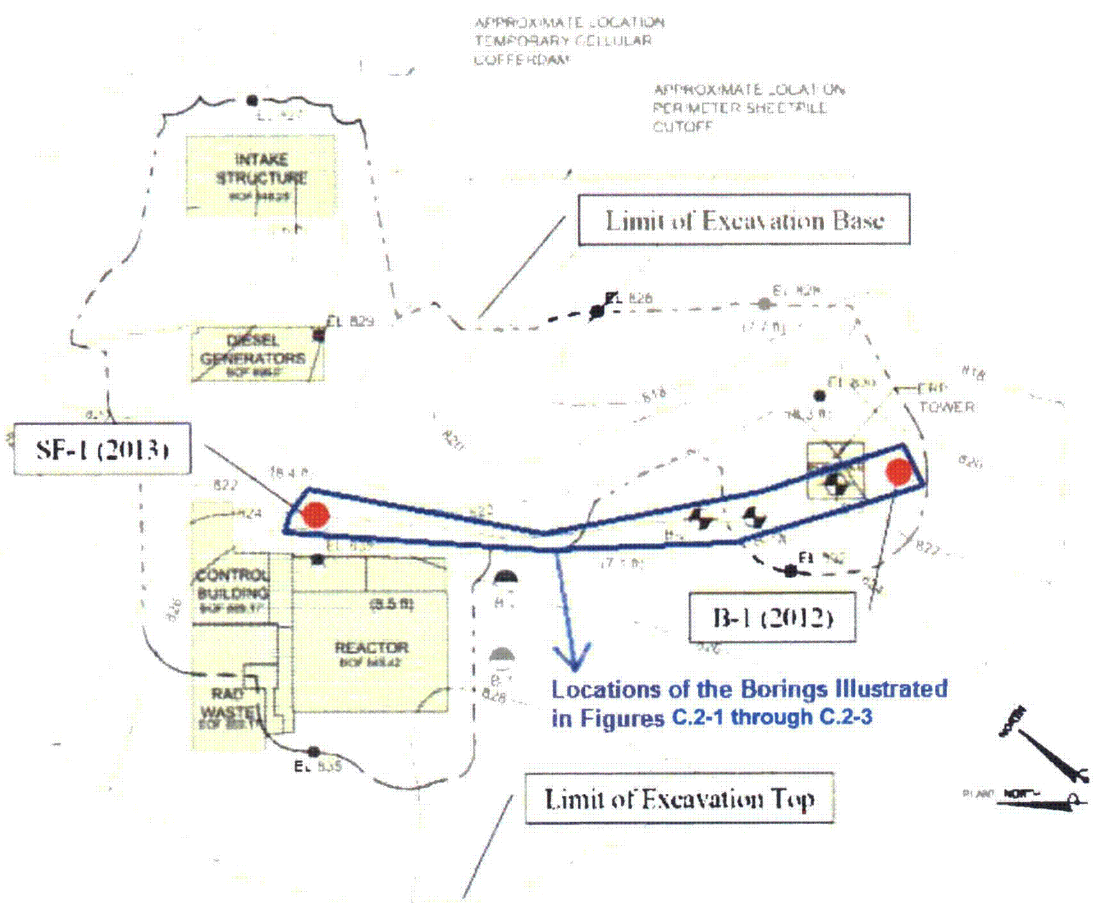

Figure C1-1 Soil Boring Locations B-1, B-1A, B-2, B-3, and SF-1

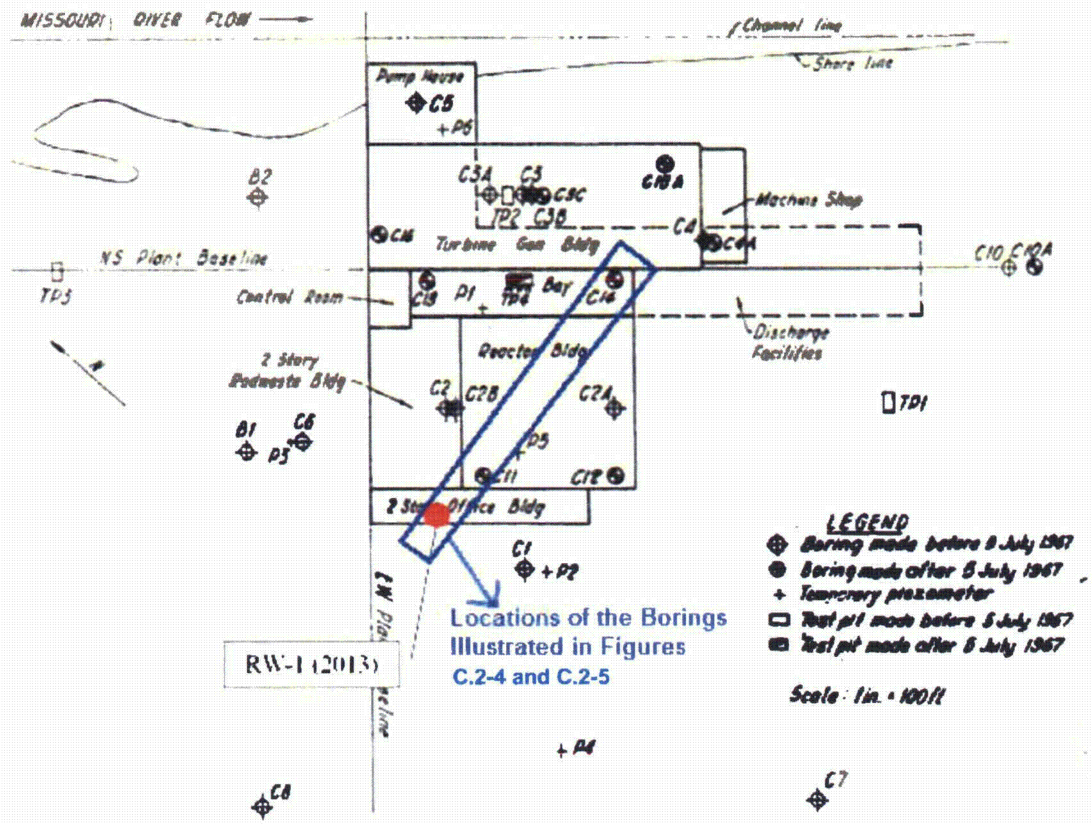

Figure C1-2 Soil Boring Locations C-11, C-14, and RW-1

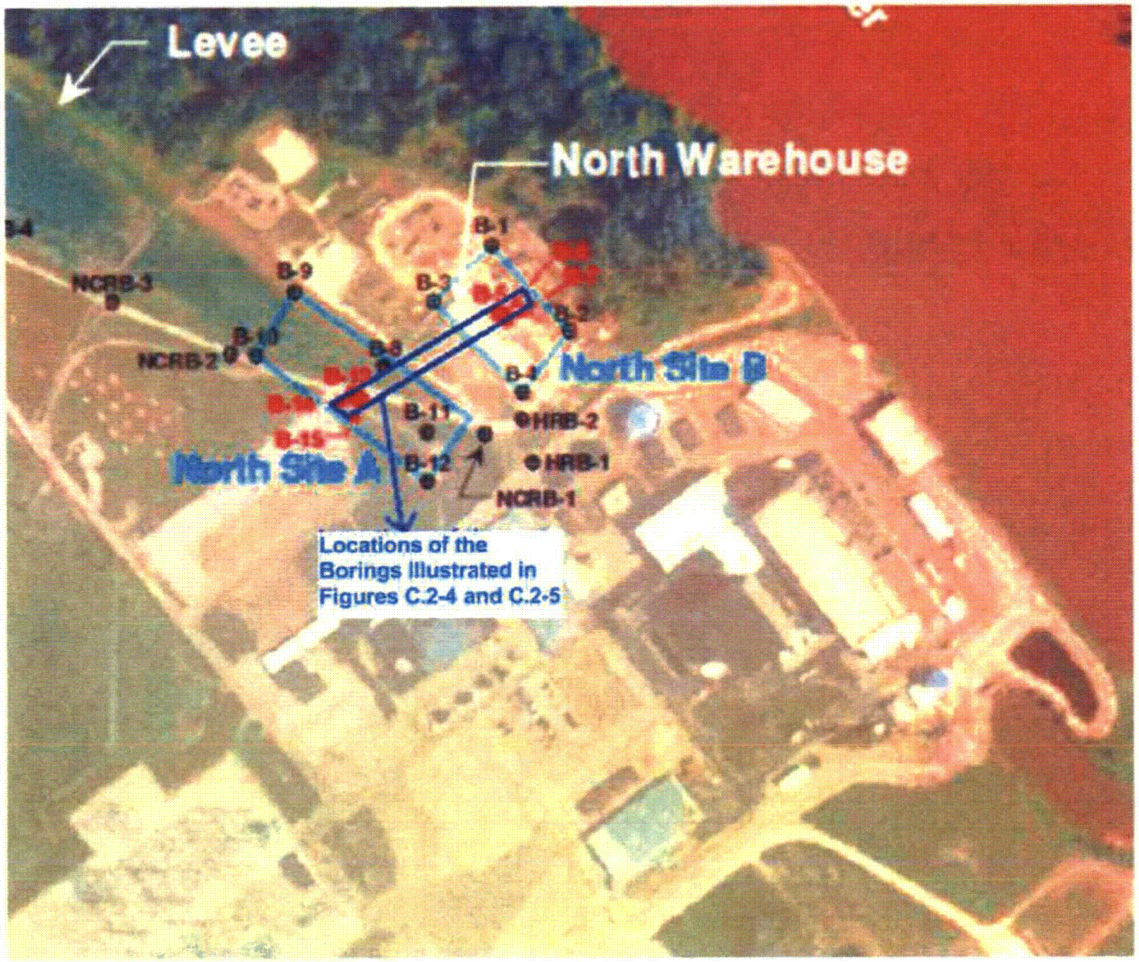

Figure C1-3 Soil Boring Location