NLS2015043

ENCLOSURE

EXPEDITED SEISMIC EVALUATION PROCESS REPORT FOR COOPER NUCLEAR STATION

ATTACHMENT 9.1 **ENGINEERING REPORT COVER SHEET & INSTRUCTIONS**

SHEET 1 OF 2

Engineering Report No. 15-007 Rev 0
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Engineering Report Cover Sheet

Engineering Report Title:

Stevenson & Associates -
Expedited Seismic Evaluation Process (ESEP) Report in Response to the 50.54(f) Information Request Regarding Fukushima Near-Term Task Force Recommendation 2.1: Seismic for the Cooper Nuclear Station Acceptance

Engineering Report Type:

New ☒ Revision ☐ Cancelled ☐ Superseded ☐
Superseded by: ______

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Prepared by: Stevenson & Associates Date: 4/15/2015
Responsible Engineer (Print Name/Sign)

Design Verified: N/A Date: N/A
Design Verifier (if required) (Print Name/Sign)

Reviewed by: Michael Pettine Derek Helmick Date: 4/15/2015
Reviewer (Print Name/Sign)

Approved by: Marshall Van Winkle Date: 4-16-15
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 Stevenson & Associates as a subject matter expert to develop the Expedited Seismic Evaluation Process (ESEP) in accordance with Electrical Power Research Institute (EPRI) "Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic" [Reference 6]. According to Reference 6; "*The ESEP was developed to focus initial resources on the review of a subset of the plant equipment that can be relied upon to protect the reactor core following beyond design basis seismic events*".

This Engineering Report will review and accept the ESEP Report prepared by Stevenson & Associates.

2. Design Inputs

The design inputs are as listed:

1. NPPD Letter NLS2015017 to NRC, "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 dated February 11, 2015.

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.

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 for CNS [Reference 13] utilizes up-to-date models representing seismic sources for Central and Eastern United States (CEUS) Plants, ground motion equations, and site amplification.

EPRI, in conjunction with the Nuclear Energy Institute (NEI), developed the Seismic Evaluation Guidance (SPID) [Reference 4] 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.

In Reference 5, Nebraska Public Power District (NPPD) submitted the Seismic Hazard Evaluation and Screening Report for Cooper Nuclear Station (CNS); which concluded that the expedited seismic evaluation process (ESEP) was required.

The Expedited Seismic Equipment List (ESEL) [Reference 1] is a subset of permanent plant equipment required for successful implementation of the mitigation strategies for Extended Loss of AC Power (ELAP) and Loss of Ultimate Heat Sink (LUHS) due to a beyond design-basis seismic event.

The ESEP addresses the requested information part of the 50.54(f) Letter [Reference 2] that requests "interim evaluations and actions taken or planned to address the higher seismic hazard relative to the design basis, as appropriate, prior to completion of the risk evaluation."

Stevenson & Associates Report 13C4215-RPT-004 *Expedited Seismic Evaluation Process (ESEP) Report in Response to the 50.54(f) Information Request Regarding Fukushima Near-Term Task Force Recommendation 2.1: Seismic for the Cooper Nuclear Station* is accepted at CNS and is included as Attachment A to this Engineering Report. All comments have been resolved and no further changes are necessary.

5. Summary of Results

The results presented by Stevenson & Associates Report 13C4215-RPT-004 *Expedited Seismic Evaluation Process (ESEP) Report in Response to the 50.54(f) Information Request Regarding Fukushima Near-Term Task Force Recommendation 2.1: Seismic for the Cooper Nuclear Station* can be found in Attachment A. Discussion of the methodology used in the development of the ESEP Report is specifically addressed within EPRI Report 3002000704 [Reference 6] and will not be discussed in this Engineering Report. Review of ESEP Report resulted in comments that were resolved accordingly. No further review is necessary.

6. Conclusions and Recommendations

1) Stevenson & Associates Report 13C4215-RPT-004 *Expedited Seismic Evaluation Process (ESEP) Report in Response to the 50.54(f) Information Request Regarding Fukushima Near-Term Task Force Recommendation 2.1: Seismic for the Cooper Nuclear Station* [Attachment A] is acceptable for adoption at CNS.

7. References

1. CNS Engineering Report 15-006 Revision 0, "Stevenson & Associates - Expedited Seismic Equipment List Walk Down Report Cooper Nuclear Station Acceptance"
2. Nuclear Regulatory Commission (E Leeds and M 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," dated March 12, 2012
3. Nuclear Regulatory Commission Order Number EA-12-049, Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events, dated March 12, 2012
4. 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
5. NPPD Letter NLS2015017 to NRC, "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," dated February 11, 2015
6. EPRI Report 3002000704 "Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic" Final Report Dated May 2013
7. NEI 12-06, Rev. 0, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide", August 2012
8. NRC Letter NLS2013024 from Nebraska Public Power District (ML13070A009), "Overall Integrated Plan in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design Basis External Events (Order Number EA-12-049)", February 28, 2013
9. NRC Letter NLS2012109 from Nebraska Public Power District (ML12310A200), "Cooper Nuclear Station's First Six-Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design-Basis External Events (Order Number EA-12-049)", August 27, 2013
10. NRC Letter NLS2014019 from Nebraska Public Power District (ML14064A201), "Nebraska Public Power District's Second Six-Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design-Basis External Events (Order Number EA-12-049)", February 26, 2014

11. NRC Letter NLS2014082 from Nebraska Public Power District, "Nebraska Public Power District's Third Six-Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design-Basis External Events (Order Number EA-12-049)", August 26, 2014
12. NRC Letter NLS2015019 from Nebraska Public Power District (ML15062A040), "Nebraska Public Power District's Fourth Six-Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design-Basis External Events (Order Number EA-12- 049)", February 23, 2015
13. CNS Engineering Report 14-002 Revision 1, "LCI GMRS Report Acceptance"

8. Attachments

A. Stevenson & Associates *Expedited Seismic Evaluation Process (ESEP) Report in Response to the 50.54(f) Information Request Regarding Fukushima Near-Term Task Force Recommendation 2.1: Seismic for the Cooper Nuclear Station* Report 13C4215-RPT-004 Revision 2; April 15, 2015

ENGINEERING REPORT ER 2015-007

Attachment A

Stevenson & Associates

"Expedited Seismic Evaluation Process (ESEP) Report in Response to the 50.54(f) Information Request Regarding Fukushima Near-Term Task Force Recommendation 2.1: Seismic for the Cooper Nuclear Station"

Report 13C4215-RPT-004 Revision 2

April 15, 2015

Stevenson & Associates
Engineering Solutions for Nuclear Energy

Document No: 13C4215-RPT-004
Revision 2

April 2015

Expedited Seismic Evaluation Process (ESEP) Report in Response to the 50.54(f) Information Request Regarding Fukushima Near-Term Task Force Recommendation 2.1: Seismic for the Cooper Nuclear Station

Prepared for:

Nebraska Public Power District
Cooper Nuclear Station
Brownville, Nebraska

Stevenson & Associates
1646 North Litchfield Road, Suite 250
Goodyear, AZ 85395

www.vecsa.com Phone: 781-932-9580 Fax: 781-933-4428

REVISION RECORD

Initial Issue (Rev. 0)		Date
Prepared by:	Samer El-Bahey, Ph.D., P.E. Senior Engineer	4/13/2015
Reviewed by:	Apostolos Karavoussianis Senior Consultant	4/13/2015
Approved by:	Apostolos Karavoussianis Senior Consultant	4/13/2015

Revisions

No.	Prepared/Date	Reviewed/Date	Approved/Date
1	Samer EL-Bahey 4/14/15	Apostolos Karavoussianis 4/14/15	Apostolos Karavoussianis 4/14/15
2	Samer EL-Bahey 4/15/15	Apostolos Karavoussianis 4/15/15	Apostolos Karavoussianis 4/15/15

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1 PURPOSE AND OBJECTIVE

Following the accident at the Fukushima Dai-ichi 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. Subsequently, the NRC issued a 50.54(f) letter on March 12, 2012 (Ref. 1) requesting information to assure that these recommendations are addressed by all U.S. nuclear power plants. The 50.54(f) letter requests that licensees and holders of construction permits under 10 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, further risk assessment may be required. Assessment approaches acceptable to the staff include a seismic probabilistic risk assessment (SPRA), or a seismic margin assessment (SMA). Based upon the assessment results, the NRC staff will determine whether additional regulatory actions are necessary.

This report describes the Expedited Seismic Evaluation Process (ESEP) undertaken for Cooper Nuclear Station (CNS). The intent of the ESEP is to perform an interim action in response to the NRC's 50.54(f) letter (Ref. 1) to demonstrate seismic margin through a review of a subset of the plant equipment that can be relied upon to protect the reactor core and containment following beyond design basis seismic events.

The ESEP is implemented using the methodologies in the NRC endorsed guidance in EPRI Report 3002000704, Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic (Ref. 2).

The objective of this report is to provide summary information describing the ESEP evaluations and results for Cooper Nuclear Station. The level of detail provided in the report is intended to enable NRC to understand the inputs used, the evaluations performed, and the decisions made as a result of the interim evaluations.

2 BRIEF SUMMARY OF THE FLEX SEISMIC IMPLEMENTATION STRATEGIES*

A simplified description of the CNS Overall Integrated Plan (Ref 3.) and subsequent 6 month updates through February 2015 (Ref. 4a, 4b, 4c, and 4d) to mitigate the postulated extended loss of ac power event is that the licensee will initially remove the core decay heat by using both the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems concurrently. The steam-driven HPCI and RCIC pumps will initially supply water to the reactor vessel from the Emergency Condensate Storage Tanks (ECSTs) or the suppression pool (torus), depending on availability. HPCI equipment will be secured after one cycle or 10 minutes to maintain battery life and RCIC will be used as the primary make-up equipment to maintain reactor level. Steam from the reactor will be vented through the safety relief valves to the torus. The SAMG portable diesel driven generator will be made available to power 250 volt dc and 125 volt dc battery chargers. Once RCIC operation is no longer possible, a FLEX portable diesel driven pump will be placed in service providing makeup water to the reactor via the residual heat removal (RHR) low pressure coolant injection (LPCI) lines. In the long-term, additional equipment, such as 4160 volt ac diesel generators and diesel driven pumps, will be delivered from one of two Regional Response Centers established by the nuclear power industry to provide supplemental accident mitigation equipment to power an RHR pump and enter shutdown cooling.

CNS plans to use containment venting via the hardened containment vent system (HCVS) to maintain containment (torus and drywell) pressure and temperature within acceptable values. The exact timing and strategy for venting is still under evaluation. The SAMG or a FLEX portable diesel driven generator will extend battery life to support the HCVS and associated instrumentation beyond the 24 hour time required by order EA-13-109.

The SFP will initially heat up due to the unavailability of the normal cooling system. A portable FLEX pump will be aligned and used to add water to the SFP via installed piping or hoses to maintain level as the pool boils. This will maintain a sufficient amount of water above the top of the fuel assemblies for cooling and shielding purposes. Additional equipment provided by the Regional Response Center will provide backup portable pumps and generators for SFP level instrumentation.

* This section is based upon input received from Cooper Nuclear Station in Reference 21.

3 EQUIPMENT SELECTION PROCESS AND ESEL

The selection of equipment for the ESEL followed the guidelines of EPRI Report 3002000704 (Ref. 2). The ESEL is presented in Attachment A.

3.1 Equipment Selection Process and ESEL

The selection of equipment to be included on the ESEL was based on installed plant equipment credited in the FLEX strategies during Phase 1, 2 and 3 mitigation of a Beyond Design Basis External Event (BDBEE), as outlined in the Cooper Nuclear Station Overall Integrated Plan (OIP) (Ref. 3), and August 2013 (Ref. 4a), February 2014 (Ref. 4b), August 2014 (Ref. 4c), and February 2015 (Ref. 4d) six month updates, in Response to the March 12, 2012, Commission Order EA-12-049 (Ref. 1). The OIP provides the Cooper Nuclear Station FLEX mitigation strategy and serves as the basis for equipment selected for the ESEP.

The scope of "installed plant equipment" includes equipment relied upon for the FLEX strategies to sustain the critical functions of core cooling and containment integrity consistent with the Cooper Nuclear Station OIP (Ref. 3) including subsequent 6 month updates through August 2014 (Ref. 4a, 4b and 4c). FLEX recovery actions are excluded from the ESEP scope per EPRI Report 3002000704 (Ref. 2). The overall list of planned FLEX modifications and the scope for consideration herein is limited to those required to support core cooling, reactor coolant inventory and sub-criticality, and containment integrity functions. Portable and pre-staged FLEX equipment (not permanently installed) are excluded from the ESEL per EPRI Report 3002000704 (Ref. 2).

The ESEL component selection followed the EPRI guidance outlined in Section 3.2 of EPRI Report 3002000704 (Ref. 2).

1. The scope of components is limited to that required to accomplish the core cooling and containment safety functions identified in Table 3-2 of EPRI Report 3002000704. The instrumentation monitoring requirements for core cooling/containment safety functions are limited to those outlined in the EPRI Report 3002000704 guidance, and are a subset of those outlined in the Cooper Nuclear Station OIP (Ref. 3) including subsequent 6 month updates through August 2014 (Ref. 4a, 4b and 4c).
2. The scope of components is limited to installed plant equipment, and FLEX connections necessary to implement the Cooper Nuclear Station OIP (Ref. 3) including subsequent 6 month updates through August 2014 (Ref. 4a, 4b and 4c) as described in Section 2.
3. The scope of components assumes the credited FLEX connection modifications are implemented, and are limited to those required to support a single FLEX success path (i.e., either "Primary" or "Back-up/Alternate").
4. The "Primary" FLEX success path is to be specified. Selection of the "Back-up/Alternate" FLEX success path must be justified.
5. Phase 3 coping strategies are included in the ESEP scope, whereas recovery strategies are excluded.
6. Structures, systems, and components excluded per the EPRI Report 3002000704 (Ref. 2) guidance are:
   - Structures (e.g. containment, reactor building, control building, auxiliary building, etc.)
   - Piping, cabling, conduit, HVAC, and their supports.
   - Manual valves and rupture disks.
   - Power-operated valves not required to change state as part of the FLEX mitigation strategies.

- Nuclear steam supply system components (e.g. reactor pressure vessel and internals, reactor coolant pumps and seals, etc.)

7. For cases in which neither train was specified as a primary or back-up strategy, then only one train component (generally 'A' train) is included in the ESEL.

3.1.1 ESEL Development

The ESEL was developed by reviewing the Cooper Nuclear Station OIP (Ref. 3), including 6 month updates through August 2014 (Ref. 4a, 4b and 4c), to determine the major equipment involved in the FLEX Strategies. Further reviews of plant drawings (e.g., Process and Instrumentation Diagrams (P&IDs) and Electrical One Line Diagrams) were performed to identify the boundaries of the flow paths to be used in the FLEX strategies and to identify specific components in the flow paths needed to support implementation of the FLEX strategies.

Boundaries were established at an electrical or mechanical isolation device (e.g., isolation amplifier, valve, etc.) in branch circuits / branch lines off the defined electrical or fluid flow path. P&IDs were the primary reference documents used to identify mechanical components and instrumentation. The flow paths used for FLEX strategies were selected and specific components were identified using detailed equipment and instrument drawings, piping isometrics, electrical schematics and one-line diagrams, system descriptions, design basis documents, etc., as necessary.

The flow paths credited for the Cooper Nuclear Station are shown in Table 3.1 below.

Table 3.1 – Flow Paths Credited for ESEP

Flow Path	FLEX Drawing	P&IDs
Core Heat Removal using the Reactor Core Isolation Cooling (RCIC) system: Coolant from the Emergency Condensate Storage Tanks (ECSTs) to the Reactor Pressure Vessel (RPV) via the RCIC pump. Main Steam providing motive force to the RCIC pump turbine and exhausted to the Suppression Pool. Extended core cooling strategy is to place one loop of RHR into the Shutdown Cooling (SDC) mode, using a flex pump supplying the RHR Heat Exchanger with river water via the RHR piping.	Second Six-Month Status Report Attachment 3 (Ref. 4b)	2043 (Ref. 5a) 2040 SH1 (Ref. 5b)
Reactor Pressure Vessel (RPV) Pressure Control using the Automatic Depressurization System (ADS): Main Steam relieved through the ADS Safety/Relief Valves to the Suppression Pool.	Second Six-Month Status Report Attachment 3 (Ref. 4b)	2028 (Ref. 5c) 2010 SH2 (Ref. 5d)
RPV Make Up: Coolant from the yet to be installed on-site well to the ECSTs via the FLEX pump and water treatment skid.	Second Six-Month Status Report Attachment 3 (Ref. 4b)	2049 SH 2 (Ref. 5e)
Hardened Containment Vent: Torus vented to atmosphere.	N/A	2022 SH 1 (Ref. 5f)

3.1.2 Power Operated Valves

Page 3-3 of EPRI Report 3002000704 (Ref. 2) notes that power operated valves not required to change state are excluded from the ESEL. Page 3-2 also notes that "functional failure modes of electrical and mechanical portions of the installed Phase 1 equipment should be considered (e.g. RCIC/AFW trips)." To address this concern, the following guidance is applied in the Cooper Nuclear Station ESEL for functional failure modes associated with power operated valves:

- Power operated valves that remain energized during the Extended Loss of all AC Power (ELAP) events (such as DC powered valves), were included on the ESEL.
- Power operated valves not required to change state as part of the FLEX mitigation strategies were not included on the ESEL. The seismic event also causes the ELAP event; therefore, the valves are incapable of spurious operation as they would be de-energized.
- Power operated valves not required to change state as part of the FLEX mitigation strategies during Phase 1, and are re-energized and operated during subsequent Phase 2 and 3 strategies, were not evaluated for spurious valve operation as the seismic event that caused the ELAP has passed before the valves are re-powered.

3.1.3 Pull Boxes

Pull boxes were deemed unnecessary to add to the ESELs as these components provide completely passive locations for pulling or installing cables. No breaks or connections in the cabling are included in pull boxes. Pull boxes were considered part of conduit and cabling, which are excluded in accordance with EPRI Report 3002000704 (Ref. 2).

3.1.4 Termination Cabinets

Termination cabinets, including cabinets necessary for FLEX Phase 2 and Phase 3 connections, provide consolidated locations for permanently connecting multiple cables. The termination cabinets and the internal connections provide a completely passive function; and the cabinets are not included in the ESEL.

3.1.5 Critical Instrumentation Indicators

Critical indicators and recorders are typically physically located on panels/cabinets and are included as separate components; however, seismic evaluation of the instrument indication may be included in the panel/cabinet seismic evaluation (rule-of-the-box).

3.1.6 Phase 2 and Phase 3 Piping Connections

Item 2 in Section 3.1 above notes that the scope of equipment in the ESEL includes "...FLEX connections necessary to implement the Cooper Nuclear Station OIP (Ref. 3), including 6 month updates through August 2014 (Ref. 4a, 4b and 4c), as described in Section 2." Item 3 in Section 3.1 also notes that "The scope of components assumes the credited FLEX connection modifications are implemented, and are limited to those required to support a single FLEX success path (i.e., either "Primary" or "Back-up/Alternate")."

Item 6 in Section 3.1 above goes on to explain that "Piping, cabling, conduit, HVAC, and their supports" are excluded from the ESEL scope in accordance with EPRI Report 3002000704 (Ref. 2).

Therefore, piping and pipe supports associated with FLEX Phase 2 and Phase 3 connections are excluded from the scope of the ESEP evaluation. However, any active valves in FLEX Phase 2 and Phase 3 connection flow path are included in the ESEL.

3.2 Justification for use of Equipment that is not the primary means for FLEX implementation

No alternate equipment is used to support the "Primary Means" for FLEX implementation.

4 GROUND MOTION RESPONSE SPECTRUM (GMRS)

4.1 Plot of GMRS Submitted by the Licensee

In accordance with Section 2.4.2 of the SPID (Ref. 14), the licensing design basis definition of the SSE control point for CNS is used for comparison to the GMRS. Seismic Hazard and Screening Report (Ref. 6) lists the CNS SSE PGA to be 0.2g. Horizontal SSE spectral values are taken from Table 3.1-1 of Reference 6 and shown in Tables 4.1-1 (below).

Table 4.1-1 – CNS SSE (5% Damping)

SSE for CNS										
Freq.	0.5	1	1.8	2.5	3	5	9	25	33	100
SA (g)	0.13	0.19	0.41	0.5	0.53	0.42	0.34	0.26	0.2	0.2

The GMRS per the Seismic Hazard and Screening Report (Table 2.4-1 of Ref. 6,) is tabulated in Table 4.1-2 and shown in Figure 4.1 below:

Table 4.1-2 – CNS GMRS at Control Point (5% Damping)

GMRS for CNS			
Freq. (Hz)	**GMRS (g)**	**Freq. (Hz)**	**GMRS (g)**
100	0.241	3.5	0.364
90	0.242	3	0.294
80	0.245	2.5	0.209
70	0.249	2	0.162
60	0.258	1.5	0.116
50	0.282	1.25	0.096
40	0.321	1	0.082
35	0.342	0.9	0.076
30	0.359	0.8	0.069
25	0.386	0.7	0.063
20	0.417	0.6	0.060
15	0.463	0.5	0.055
12.5	0.486	0.4	0.044
10	0.465	0.35	0.039
9	0.449	0.3	0.033
8	0.430	0.25	0.028
7	0.417	0.2	0.022
6	0.422	0.15	0.017
5	0.454	0.125	0.014
4	0.415	0.1	0.011

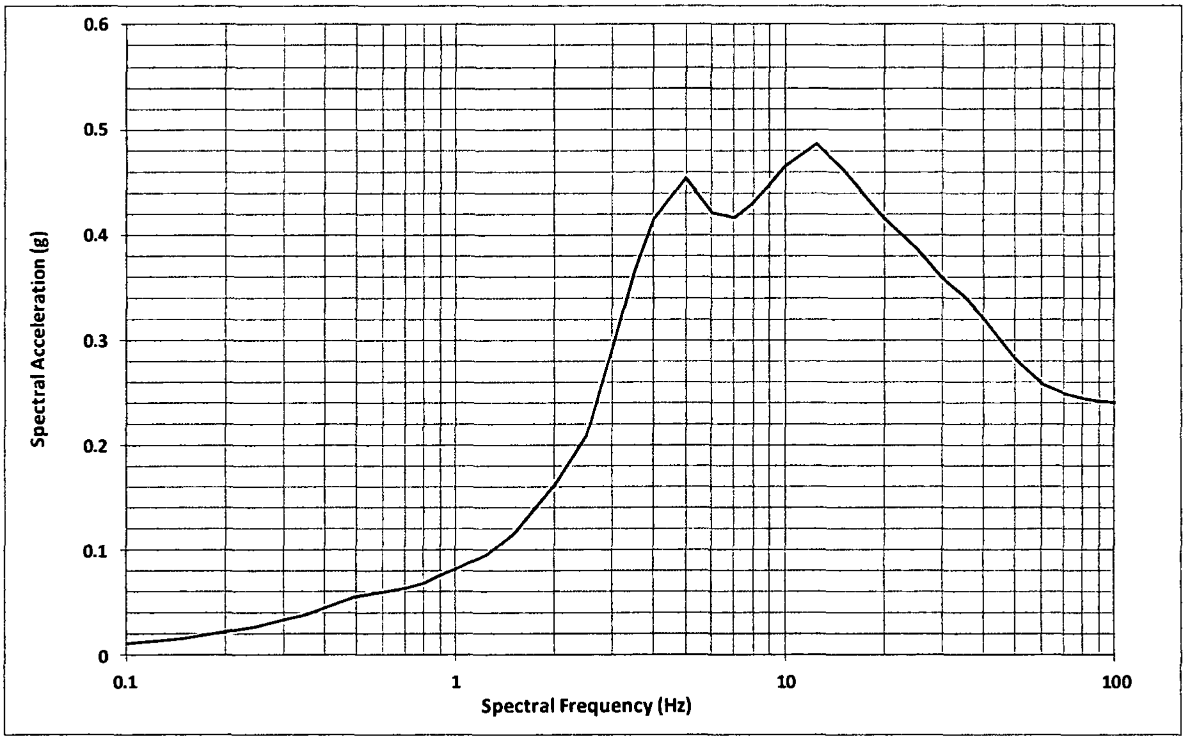

Figure 4.1 – CNS GMRS (5% Damping)

4.2 Comparison to SSE

As identified in the Seismic Hazard and Screening Report (Ref. 6), the GMRS exceeds the SSE in portions of the 1-10 Hz range as shown in Figure 4.2 below:

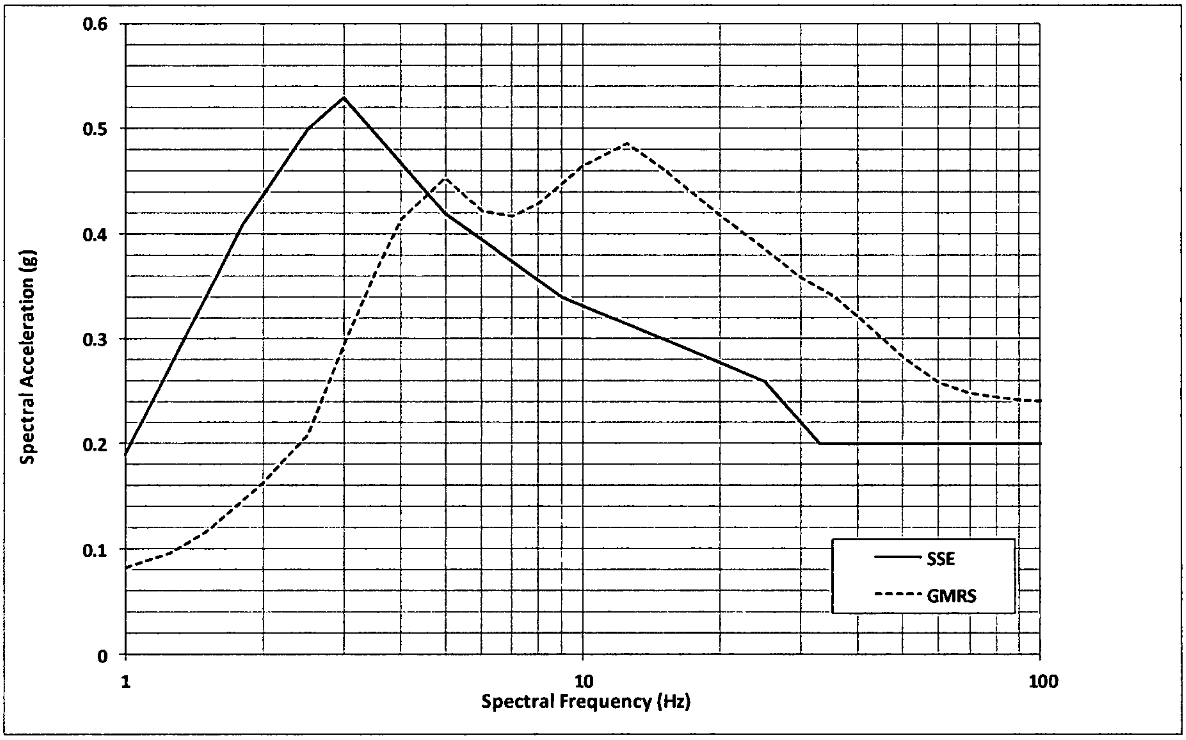

Figure 4.2 – CNS GMRS vs. SSE (5% Damping)

5 REVIEW LEVEL GROUND MOTION (RLGM)

5.1 Description of RLGM Selected

The RLGM for CNS was determined in accordance with Section 4 of EPRI 30020000704 (Ref. 2) by linearly scaling the CNS SSE by the maximum GMRS/SSE ratio between the 1 and 10 hertz range.

From review of Figure 4.2-1, the maximum GMRS/SSE ration occurs at 10 Hz where the GMRS spectral acceleration is 0.465g. The SSE shape requires logarithmic interpolation between control points at 9 and 25 Hz. The SSE spectral acceleration at 10 Hz is determined as follows:

$$S_{a,10Hz} = 10^{\left((\log(10Hz)-\log(9Hz)) * \frac{\log(0.26g)-\log(0.34g)}{\log(25Hz)-\log(9Hz)} + \log(0.34g) \right)} = 0.334g$$

The maximum GMRS/SSE ratio between 1 - 10Hz is then calculated to be 1.39 at 10 Hz (witness 0.465 g / 0.334 g = 1.39).

The resulting 5% damped RLGM based on scaling the horizontal SSE by the maximum GMRS/SSE ratio of 1.39 is shown in Table 5.1 and Figure 5.1 below.

Table 5.1 – CNS RLGM (5% Damping)

RLGM for CNS										
Freq.	0.5	1	1.8	2.5	3	5	9	25	33	100
SA (g)	0.18	0.26	0.57	0.70	0.74	0.58	0.47	0.36	0.28	0.28

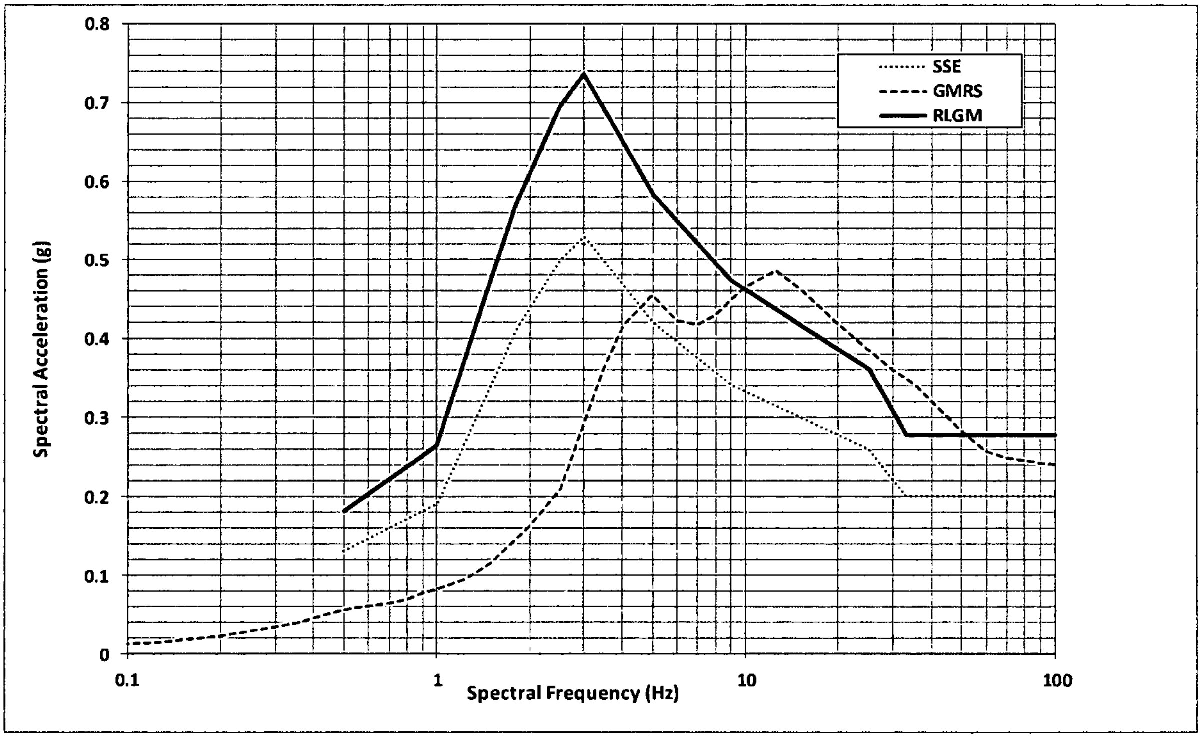

Figure 5.1 – Comparison of CNS RLGM, GMRS & SSE (5% Damping)

5.2 Method to Estimate ISRS

The method used to derive the ESEP in-structure response spectra (ISRS) was to uniformly scale the existing SSE-based ISRS from Reference 18 by the maximum GMRS/SSE ratio of 1.39. The scaled ISRS was determined for all buildings and elevations where ESEL items are located at CNS.

6 SEISMIC MARGIN EVALUATION APPROACH

It is necessary to demonstrate that ESEL items have sufficient seismic capacity to meet or exceed the demand characterized by the RLGM. The seismic capacity is characterized as the peak ground acceleration (PGA) for which there is a high confidence of a low probability of failure (HCLPF). The PGA is associated with a specific spectral shape, in this case the 5%-damped RLGM spectral shape. The HCLPF capacity must be equal to or greater than the RLGM PGA. The criteria for seismic capacity determination are given in Section 5 of EPRI Report 3002000704 (Ref. 2).

There are two basic approaches for developing HCLPF capacities:

1. Deterministic approach using the conservative deterministic failure margin (CDFM) methodology of EPRI NP-6041 (Ref. 7).
2. Probabilistic approach using the fragility analysis methodology of EPRI TR-103959 (Ref. 8).

For CNS, the deterministic approach using the CDFM methodology of EPRI NP-6041 (Ref. 7) was used to determine HCLPFs capacities.

6.1 Summary of Methodologies Used

CNS conservatively applied the methodology of EPRI NP-6041 (Ref. 7) to all items on the ESEL. The screening walkdowns used the screening tables from Chapter 2 of EPRI NP-6041 (Ref. 7). The walkdowns were conducted by engineers who as a minimum attended the SQUG Walkdown Screening and Seismic Evaluation Training Course. The walkdowns were documented on Screening Evaluation Work Sheets from EPRI NP-6041 (Ref. 7). Anchorage capacity calculations use the CDFM criteria established within EPRI NP-6041 (Ref. 7) with CNS specific allowables and material strengths used as applicable. The input seismic demand used was the RLGM provided in Table 5.1 and Figure 5.1.

6.2 HCLPF Screening Process

From Table 5.1, the spectral peak of the RLGM (amplified PGA) for CNS equals 0.74g and occurs at 3 Hz. The screening tables in EPRI NP-6041 (Ref. 7) are based on ground peak spectral accelerations of 0.8g (1st screening column) and 1.2g (2nd screening column). Accordingly, the ESEL component can be screened against the 1st screening column (< 0.8g) criteria of NP-6041-SL Table 2-4; however the 2nd screening column (0.8g - 1.2g) will be used to gain additional functional seismic margin.

One ESEL component was located 40 feet above grade. For components located 40 feet above grade or more, screening based on ground peak spectral acceleration is not applicable and additional consideration is required. In accordance with Appendix B of EPRI 1019200 (Ref. 19), components that are above 40 feet from grade and have corresponding ISRS at the base of component not in exceedance of 1.8g in the component frequency range of interest may be screened using the caveats of the 2nd screening column.

The screening of anchorage for non-valve components was evaluated either by the Seismic Review Team (SRT) judgment or simple analysis. For non-valve components whose anchorage could not readily be screened by SRT judgment or simple analysis, CDFM HCLPF calculations (Ref. 9) were performed. This is documented in Attachment B.

6.3 Seismic Walkdown Approach

6.3.1 Walkdown Approach

Walkdowns for CNS were performed in accordance with the criteria provided in Section 5 of EPRI Report 3002000704 (Ref. 2), which refers to EPRI NP-6041 (Ref. 7) for the Seismic Margin Assessment process. Pages 2-26 through 2-30 of EPRI NP-6041 (Ref. 7) describe the seismic walkdown criteria, including the following key criteria:

> *"The SRT [Seismic Review Team] should "walk by" 100% of all components which are reasonably accessible and in non-radioactive or low radioactive environments. Seismic capability assessment of components which are inaccessible, in high-radioactive environments, or possibly within contaminated containment, will have to rely more on alternate means such as photographic inspection, more reliance on seismic reanalysis, and possibly, smaller inspection teams and more hurried inspections. A 100% "walk by" does not mean complete inspection of each component, nor does it mean requiring an electrician or other technician to de-energize and open cabinets or panels for detailed inspection of all components. This walkdown is not intended to be a QA or QC review or a review of the adequacy of the component at the SSE level.*
>
> *If the SRT has a reasonable basis for assuming that the group of components are similar and are similarly anchored, then it is only necessary to inspect one component out of this group. The "similarity-basis" should be developed before the walkdown during the seismic capability preparatory work (Step 3) by reference to drawings, calculations or specifications. The one component or each type which is selected should be thoroughly inspected which probably does mean de-energizing and opening cabinets or panels for this very limited sample. Generally, a spare representative component can be found so as to enable the inspection to be performed while the plant is in operation. At least for the one component of each type which is selected, anchorage should be thoroughly inspected.*
>
> *The walkdown procedure should be performed in an ad hoc manner. For each class of components the SRT should look closely at the first items and compare the field configurations with the construction drawings and/or specifications. If a one-to-one correspondence is found, then subsequent items do not have to be inspected in as great a detail. Ultimately the walkdown becomes a "walk by" of the component class as the SRT becomes confident that the construction pattern is typical. This procedure for inspection should be repeated for each component class; although, during the actual walkdown the SRT may be inspecting several classes of components in parallel. If serious exceptions to the drawings or questionable construction practices are found then the system or component class must be inspected in closer detail until the systematic deficiency is defined.*
>
> *The 100% "walk by" is to look for outliers, lack of similarity, anchorage which is different from that shown on drawings or prescribed in criteria for that component, potential SI [Seismic Interaction][2] (Ref. 2, page 5-4) problems, situations that are at odds with the team members' past experience, and any other areas of serious seismic concern. If any such concerns surface, then the limited sample size of one component of each type for thorough inspection will have to be increased. The increase in sample size which should be inspected will depend upon the number of outliers and different anchorages, etc., which are observed. It is up to the SRT to ultimately*

[2] EPRI Report 3002000704 (Ref 2) page 5-4 limits the ESEP seismic interaction reviews to "nearby block walls" and "piping attached to tanks" which are reviewed "to address the possibility of failures due to differential displacements." Other potential seismic interaction evaluations are "deferred to the full seismic risk evaluations performed in accordance with EPRI 1025287 (Ref. 14)."

select the sample size since they are the ones who are responsible for the seismic adequacy of all elements which they screen from the margin review. Appendix D gives guidance for sampling selection"

The CNS walkdowns included as a minimum a 100% walk-by of all items on the ESEL except as noted in Section 7.0. Any previous walkdown information that was relied upon for SRT judgment is documented in Section 6.3.2. ESEP Walkdown and Screening Report (Ref. 20) documents the walkdown results.

6.3.2 Application of Previous Walkdown Information

Previous seismic walkdowns were used to support the ESEP seismic walkdowns and evaluations. Several ESEL items were previously walked down during the CNS Seismic IPEEE (Ref. 16a) and USI A-46 (Ref. 16b) program. Those previous program performed extensive walkdowns including the opening of electrical components, such as, MCCs, Switchgears, Control Cabinet, etc…, hence eliminating the need to open potentially energized equipment.

The previous walkdown observations and photographs were reviewed and steps were taken to confirm that the previous walkdowns remain valid. A walk by was performed to confirm that the equipment material condition and configuration is consistent with the previous walkdown observations and that no new significant interactions related to block walls or piping attached to tanks exist.

In general, detailed inspections were performed for ESEP and included, as a minimum, a walk-by of all the components on the ESEL by the SRT with exception of items inside the Drywell, the Steam Tunnel as listed below, and items added to the ESEL later after the walkdowns. A detailed discussion and resolution for each of the items listed below is provided in Section 7.0.

- Safety Relief Valves (SRV), Located inside the Drywell.
    - MS-RV-71A
    - MS-RV-71B
    - MS-RV-71C
    - MS-RV-71D
    - MS-RV-71E
    - MS-RV-71F
    - MS-RV-71G
    - MS-RV-71H
- SRV Accumulators, Located inside the Drywell
    - IA-ACC-256A
    - IA-ACC-256B
    - IA-ACC-256C
    - IA-ACC-256D
    - IA-ACC-256E
    - IA-ACC-256F
    - IA-ACC-256G
    - IA-ACC-256H
- Drywell Temperature Elements, Located inside the Drywell
    - PC-TE-505A
    - PC-TE-505B

  - PC-TE-505C
  - PC-TE-505D
  - PC-TE-505E
- RCIC Outboard Steam Supply Isolation Valve, RCIC-MO-16, Located inside the steam tunnel
- RCIC Pump Discharge To RX Valve, RCIC-MO-21, Located inside the steam tunnel
- Critical Motor Control Center MCC RA added to the ESEL after the walkdown was concluded
- Analog Process Cabinet LRP-PNL-PL1 added to the ESEL after the walkdown was concluded

6.3.3 Significant Walkdown Findings

Consistent with the guidance from NP-6041 (Ref. 7), no significant outliers or anchorage concerns were identified during the CNS seismic walkdowns.

6.4 HCLPF Calculation Process

ESEL items were evaluated using the criteria in EPRI NP-6041 (Ref. 7). Those evaluations included the following steps:

- Performing seismic capability walkdowns for equipment to evaluate the equipment installed plant conditions
- Performing screening evaluations using the screening tables in EPRI NP-6041 (Ref. 7) as described in Section 6.2 and
- Performing HCLPF calculations considering various failure modes that include both structural failure modes (e.g. anchorage, load path etc.) and functional failure modes.

All HCLPF calculations were performed using the CDFM methodology and are documented in the HCLPF calculations (Ref. 9).

Anchorage configurations for non-valve components were evaluated either by SRT judgment, large margins in existing design basis calculations, or CDFM based HCLPF calculations (Ref. 9a, 9b, and 9c). The results of these analysis methods are documented in Attachment B. For components beyond 40 feet above grade, Table 2-4 of NP-6041 (Ref. 7) is not directly applicable.

EPRI Report 3002000704 (Ref. 2) Section 5 references to EPRI 1019200 (Ref. 19) with respect to screening criteria beyond 40 feet above grade. This guide update allows multiplying the screening lane spectral acceleration value ranges by a factor of 1.5 in order to account for spectral accelerations at the base of the component[3]. This screening level at the base of a component is compared to the ISRS demand corresponding to the RLGM. For example, by factoring the acceleration ranges for screening lane 2 of NP-6041-SL Table 2-4, the capacity at the base of a component is bounded by 1.2g*1.5 = 1.8g. This is compared with the seismic demand presented by the ISRS (as opposed to the RLGM).

As described in the begin of Section 6, for HCLPF calculations the Conservative, Deterministic Failure Margin (CDFM) analysis criteria established in Section 6 of EPRI NP-6041 (Ref. 7) are used for a detailed analysis of components. The relevant CDFM criteria from EPRI NP-6041 (Ref. 7) are summarized in Table 6.4.

[3] Page A-22 of NP-6041 (Ref. 7) also references the use of 1.5 times the bounding spectra for comparison against the floor spectra.

Table 6.4 – HCLPF Calculation Summary

Load combination:	Normal + Seismic Margin Earthquake (SME)[4]
Ground response spectrum:	Conservatively specified (84% non-exceedance probability)
Damping:	Conservative estimate of median damping.
Structural model:	Best estimate (median) + uncertainty variation in frequency.
Soil-structure interaction	Best estimate (median) + parameter variation
Material strength:	Code specified minimum strength or 95% exceedance of actual strength if test data is available.
Static capacity equations:	Code ultimate strength (ACI), maximum strength (AISC), Service Level D (ASME) or functional limits. If test data is available to demonstrate excessive conservatism of code equations then use 84% exceedance of test data for capacity equations.
Inelastic energy absorption:	For non-brittle failure modes and linear analysis, use 80% of computed seismic stress in capacity evaluation to account for ductility benefits or perform nonlinear analysis and use 95% exceedance ductility levels.
In-structure (floor) spectra generation:	Use frequency shifting rather than peak broadening to account for uncertainty and use median damping.

The HCLPF capacity is equal to the PGA at which the strength limit is reached. The HCLPF earthquake load is calculated as follows:

U = Normal + Ec

Where:

- U = Ultimate strength per Section 6 of EPRI NP-6041 (Ref. 7)
- Ec = HCLPF earthquake load
- Normal = Normal operating loads (dead and live load expected to be present, etc.)

For this calculation, the HCLPF earthquake load is related to a fixed reference earthquake:

Ec = SFc*Eref

Where:

- Eref = reference earthquake from the relevant in-structure response spectrum (ISRS)
- SFc = component-specific scale factor that satisfies U = Normal +Ec

The HCLPF will be defined as the PGA produced by Ec. Because the CNS RLGM PGA is 0.28g:
HCLPF = 0.28g*SFc

[4] The SME pertaining to HCLPF calculations for CNS is equivalent to the RLGM.

6.5 Functional Evaluation of Relays

The CNS ESEL does not contain any relays or switches associated with the FLEX Phase 1 response, therefore, no evaluations were performed for relay functionality.

6.6 Tabulated ESEL HCLPF Values (including Key failure modes)

Tabulated ESEL HCLPF values including the key failure modes are included in Attachment B.

- For items screened out using NP-6041 (Ref. 7) screening tables, or based on SMA analysis in the checklists and no HCLPFs were calculated, the HCLPF is listed as "> RLGM" and the failure mode is listed as "Screened."
- For items where anchorage controls the HCLPF value, the anchorage HCLPF value is listed in the table and the failure mode is noted as "Anchorage".
- For items where an equipment capacity based upon the screening lane values of Table 2-4 of EPRI NP-6041 (Ref. 7) controls the HCLPF value (e.g. anchorage HCLPF capacity exceeds the equipment capacity derived from screening lanes), the screening lane HCLPF value is listed in the table and the failure mode is listed as "Equipment Capacity." Based on NP-6041 Table 2-4 lane 2, this limit is equal to 0.45g for items below 40 feet above grade.

The "Equipment Capacity" limits from above are calculated as follows:

The upper-bound spectral peak to NP-6041 Table 2-4 lane 2 is 1.2g. From Table 5.1, the RLGM spectral peak is 0.74g and the PGA is 0.28g. Thus, for equipment less than 40 feet above grade, the "Equipment Capacity" HCLPF is limited to 1.2g / 0.74g * 0.28g PGA = 0.45g PGA. For equipment located greater than 40 feet above grade, if the associated ISRS spectral accelerations in the component frequency range of interest do not exceed 1.5 times the NP-6041 Table 2-4 lane 2 bounding spectrum (e.g. 1.8g peak spectral acceleration), the "Equipment Capacity" HCLPF is conservatively limited to the RLGM PGA of 0.28g.

7 INACCESSIBLE ITEMS

7.1 Identification of ESEL Items Inaccessible For Walkdowns

Twenty four (24) ESEL items were not accessible to the SRT during the ESEP walkdowns at CNS due to plant operation, and two (2) ESEL items that were added late after the walkdowns are evaluated based on photographs provided by CNS. A description of circumstances and disposition for these items is provided below.

SRVs and their accumulators (see Section 6.3.2 for component IDs):

The SRVs were not walked down by the SRT due to Radiation Protection concerns given that the components are located within the Drywell and the station was not in outage during the available SRT walkdown window. The SRVs and their accumulators were walked down as part of the A-46 and IPEEE programs. In addition to the A-46 SEWS observations and photographs, the station provide the SRT with additional photograph and design documents. The SRT reviewed design documents, A-46 SEWS and photographs and determined to be acceptable for evaluation of the SRVs and their accumulators (including the consideration of potential seismic interaction), with no further walkdowns, in accordance with the methodology of NP-6041.

Drywell Temperature Elements (see Section 6.3.2 for component IDs):

The Drywell temperature elements were not walked down by the SRT due to Radiation Protection concerns given that the components are located within the Drywell and the station was not in outage during the available SRT walkdown window. The temperature elements were walked down as part of the A-46 and IPEEE programs. The SRT reviewed the A-46 SEWS observation and photographs and determined to be acceptable for evaluation of the temperature elements (including the consideration of potential seismic interaction), with no further walkdowns, in accordance with the methodology of NP-6041.

RCIC-MO-16 and RCIC-MO-21:

These valve were not walked down by the SRT due to the components being located in a contaminated and high radiation area, i.e. Steam Tunnel. Station photos and or design documentation were reviewed by the SRT and determined to be acceptable for evaluation of these RCIC MOVs (including the consideration of potential seismic interaction), with no further walkdowns, in accordance with the methodology of NP-6041.

MCC RA and LRP-PNL-PL1

These items were added late after the walkdowns were performed and are screened based on photographs provided by CNS, design documentation, and previous walkdowns and reviewed by the SRT and determined to be acceptable.

7.2 Planned Walkdown / Evaluation Schedule / Close Out

Since all items that were inaccessible during the ESEP were resolved by alternative means (i.e. confirmatory photos, A-46 SEWS and design documentation) to the satisfaction of the SRT, no additional walkdowns are required.

8 ESEP CONCLUSIONS AND RESULTS

8.1 Supporting Information

CNS has performed the ESEP as an interim action in response to the NRC's 50.54(f) letter (Ref. 1). It was performed using the methodologies in the NRC endorsed guidance in EPRI Report 3002000704 (Ref. 2).

The ESEP provides an important demonstration of seismic margin and expedites plant safety enhancements through evaluations and potential near-term modifications of plant equipment that can be relied upon to protect the reactor core following beyond design basis seismic events.

The ESEP is part of the overall CNS response to the NRC's 50.54(f) letter (Ref. 1). On March 12, 2014, NEI submitted to the NRC results of a study (Ref. 11) of seismic core damage risk estimates based on updated seismic hazard information as it applies to operating nuclear reactors in the Central and Eastern United States (CEUS). The study concluded that "site-specific seismic hazards show that there [...] has not been an overall increase in seismic risk for the fleet of U.S. plants" based on the re-evaluated seismic hazards. As such, the "current seismic design of operating reactors continues to provide a safety margin to withstand potential earthquakes exceeding the seismic design basis."

The NRC's May 9, 2014 NTTF 2.1 Screening and Prioritization letter (Ref. 13) concluded that the "fleetwide seismic risk estimates are consistent with the approach and results used in the GI-199 safety/risk assessment." The letter also stated that "As a result, the staff has confirmed that the conclusions reached in GI-199 safety/risk assessment remain valid and that the plants can continue to operate while additional evaluations are conducted."

An assessment of the change in seismic risk for CNS was included in the risk evaluation submitted in the March 12, 2014 NEI Letter (Ref 11). However, due to discrepancies in Letter NLS2014027 (Ref. 23) regarding Nebraska Public Power District's seismic hazard and screening report (CEUS Sites) for CNS, dated March 31, 2014, and the NRC's May 9, 2014, letter to all power reactor licensees regarding screening and prioritization results for seismic hazard re-evaluations (Ref. 13), CNS took action to evaluate and resolve differences in GMRS models between the NRC and NEI/CNS. As a result of the evaluation CNS performed an IPEEE Adequacy Review and submitted Letter NLS2015017 (Ref. 6), a revised seismic hazard evaluation and screening report for CNS, dated February 11, 2015. Therefore, the conclusions in the pending NRC Response to NLS2015017 will govern CNS Response and Commitment.

In addition, the March 12, 2014 NEI letter (Ref. 11) provided an attached "Perspectives on the Seismic Capacity of Operating Plants," which (1) assessed a number of qualitative reasons why the design of Structures, Systems, and Components (SSCs) inherently contain margin beyond their design level, (2) discussed industrial seismic experience databases of performance of industry facility components similar to nuclear SSCs, and (3) discussed earthquake experience at operating plants.

CNS was designed using conservative practices, such that the plant has significant margin to withstand large ground motions safely. This has been borne out for those plants that have actually experienced significant earthquakes. The seismic design process has inherent (and intentional) conservatisms which result in significant seismic margins within structures, systems and components (SSCs). These conservatisms are reflected in several key aspects of the seismic design process, including:

- Safety factors applied in design calculations
- Damping values used in dynamic analysis of SSCs
- Bounding synthetic time histories for in-structure response spectra calculations
- Broadening criteria for in-structure response spectra

- Response spectra enveloping criteria typically used in SSC analysis and testing applications
- Response spectra based frequency domain analysis rather than explicit time history based time domain analysis
- Bounding requirements in codes and standards
- Use of minimum strength requirements of structural components (concrete and steel)
- Bounding testing requirements
- Ductile behavior of the primary materials (that is, not crediting the additional capacity of materials such as steel and reinforced concrete beyond the essentially elastic range

These design practices combine to result in margins such that the SSCs will continue to fulfill their functions at ground motions well above the SSE.

8.2 Summary of ESEP Identified and Planned Modifications

The results of the CNS ESEP performed as an interim action in response to the NRC's 50.54(f) letter (Ref. 1) using the methodologies in the NRC endorsed guidance in EPRI Report 3002000704 (Ref. 2) show that all equipment evaluated are adequate in resisting the seismic loads expected to result from the site RLGM. Therefore, no plant modifications are required as a result of the CNS ESEP.

8.3 Modification Implementation Schedule

No modification implementation schedule is required because no modifications are required.

8.4 Summary of Regulatory Commitments

No regulatory commitments are required.

9. REFERENCES

1. NRC (E Leeds and M 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. Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1 – Seismic. EPRI, Palo Alto, CA: May 2013. 3002000704

3. NRC Letter NLS2013024 from Nebraska Public Power District (ML13070A009), "Overall Integrated Plan in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design Basis External Events (Order Number EA-12-049)", February 28, 2013

4. (a) NRC Letter NLS2012109 from Nebraska Public Power District (ML12310A200), "Cooper Nuclear Station's First Six-Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design-Basis External Events (Order Number EA-12-049)", August 27, 2013

   (b) NRC Letter NLS2014019 from Nebraska Public Power District (ML14064A201), "Nebraska Public Power District's Second Six-Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design-Basis External Events (Order Number EA-12-049)", February 26, 2014

   (c) NRC Letter NLS2014082 from Nebraska Public Power District, "Nebraska Public Power District's Third Six-Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design-Basis External Events (Order Number EA-12-049)", August 26, 2014

   (d) NRC Letter NLS2015019 from Nebraska Public Power District, "Nebraska Public Power District's Fourth Six-Month Status Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design-Basis External Events (Order Number EA-12-049)", February 23, 2015

5. (a) Burns & Roe Drawing 2043, Rev. N55, (RET. 454003636) Cooper Nuclear Station Flow Diagram Reactor Core Isolation Coolant And Reactor Feed Systems

   (b) Burns & Roe Drawing 2040, SH 1, Rev. N82, (RET. 454003633) Cooper Nuclear Station Flow Diagram Residual Heat Removal System

   (c) Burns & Roe Drawing 2028, Rev. N52, (RET. 454003618) Cooper Nuclear Station Flow Diagram Reactor Building & Drywell Equipment Drain System

   (d) Burns & Roe Drawing 2010, SH 2, Rev. N95, (RET. 454003594) Cooper Nuclear Station Flow Diagram Instrument Air Reactor Building

   (e) Burns & Roe Drawing 2049, SH 2, Rev N37, (RET. 454003676) Cooper Nuclear Station Flow Diagram Condensate Supply System

(f) Burns & Roe Drawing 2022, SH 1, Rev. N78, (RET. 454003610) Cooper Nuclear Station Flow Diagram Primary Containment Cooling & Nitrogen Inerting System

6. NPPD Letter NLS2015017 to NRC, "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," dated February 11, 2015
7. A Methodology for Assessment of Nuclear Power Plant Seismic Margin, Rev. 1, August 1991, Electric Power Research Institute, Palo Alto, CA. EPRI NP 6041
8. Methodology for Developing Seismic Fragilities, August 1991, EPRI, Palo Alto, CA. 1994, TR-103959
9. S&A calculations (CNS owner acceptance calculation NEDC 15-048):
    a. 14C4215-CAL-001 Rev. 0, "Seismic HCLPF Capacity for Condensate Storage Tanks ECST 1A and ECST 1B"
    b. 14C4215-CAL-002 Rev. 0, "Seismic HCLPF Capacity for Mechanical Equipment for ESEP"
    c. 14C4215-CAL-003 Rev. 0, "Seismic HCLPF Capacity for Electrical Equipment for ESEP"
10. Nuclear Regulatory Commission, NUREG/CR-0098, Development of Criteria for Seismic Review of Selected Nuclear Power Plants, published May 1978
11. Nuclear Energy Institute (NEI), A. Pietrangelo, Letter to D. Skeen of the USNRC, "Seismic Core Damage Risk Estimates Using the Updated Seismic Hazards for the Operating Nuclear Plants in the Central and Eastern United States", March 12, 2014
12. Nuclear Energy Institute (NEI), A. Pietrangelo, Letter to D. Skeen of the USNRC, "Proposed Path Forward for NTTF Recommendation 2.1: Seismic Reevaluations", April 9, 2013
13. NRC (E Leeds) Letter to All Power Reactor Licensees et al., "Screening and Prioritization Results Regarding Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(F) Regarding Seismic Hazard Re-Evaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights From the Fukushima Dai-Ichi Accident," May 9, 2014
14. Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic. EPRI, Palo Alto, CA: February 2013 (EPRI 1025287)
15. NRC (E Leeds) Letter to NEI (J Pollock), "Electric Power Research Institute Final Draft Report, "Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," as an Acceptable Alternative to the March 12, 2012, Information Request for Seismic Reevaluations," May 7, 2013
16. (a) Report NLS960143, "Individual Plant Examination for External Events (IPEEE) Report - 10 CFR 50.54(f) Cooper Nuclear Station, NRC Docket No. 50-298, License No. DPR-46"

    (b) Report NLS960076, "Submittal of the Unresolved Safety Issue (USI) A-46 Summary Report Cooper Nuclear Station, NRC Docket No. 50-298, License No. DPR-46"

17. CNS Updated Final Safety Analysis Report (UFSAR)

18. NED C87-162 Rev. 2, "CNS Frequency vs. Acceleration Response Spectra Curves"

19. EPRI Technical Report (TR) 1019200, "Seismic Fragility Applications Guide Update" December 2009.

20. S&A Report 14C4215-RPT-003 Rev. 2, (CNS Engineering Report Number ER 2015-006) "Seismic Evaluation of Equipment at Cooper Nuclear Station for the Expedited Seismic Evaluation Process"

21. S&A Letter Received from Client, 13C4215-LRC-001, " Emailing: FLEX audit intro.docx", 2/25/2015

22. S&A Report 14C4215-RPT-001 Rev. 3, (CNS Engineering Report Number ER 2015-006) "Development of Expedited Seismic Equipment List"

23. NPPD Letter NLS2014027 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

24. NRC Letter NLS2014101 from Nebraska Public Power District, "Nebraska Public Power District's First Six-Month Status Report in Response to June 6, 2013 Commission Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions (Order Number EA-13-109)", December 19, 2014

APPENDIX A: Cooper Nuclear Station (CNS) ESEL (Ref. 22)

Item #	Equipment ID	Description	Equipment Normal State	Equipment Desired State	Notes
83	13-1	RCIC PUMP	Off	On	
84	13-2	RCIC TURBINE	Off	On	
121	ASD-ADS/REC PANEL	ADS ALTERNATE SHUTDOWN PANEL	In Service	In Service	Temperature Indicators: PC-TI-2A, PC-TI-2C, PC-TI-2E, PC-TI-2G
46	ECST 1A	EMERG COND. STOR. TK. 1A	N/A	N/A	
47	ECST 1B	EMERG COND. STOR. TK. 1B	N/A	N/A	
29	EE-BAT-125 1A	DIV. I 125 VDC STATION BATTERY	Energized	Energized	
35	EE-BAT-125 1B	DIV. II 125 VDC STATION BATTERY	Energized	Energized	
30	EE-BAT-250 1A	DIV. I 250 VDC STATION BATTERY	Energized	Energized	
44	EE-CHR-125C	125VDC BATTERY CHARGER 1C	Energized	Energized	
45	EE-CHR-250C	250V BATTERY CHARGER 1C	Energized	Energized	
42	EE-DSC-250C	FEED TO 250VDC STATION SERVICE BATTERY CHARGER 1C	Energized	Energized	
41	EE-IVTR-1A	DIV I INVERTER	In Service	In Service	
88	EE-PNL-125ASD	125 VDC ASD HPCI DISTRBUTION PANEL	Energized	Energized	PNL inside HPCI ASD PANEL
32	EE-PNL-A	125VDC DISTRIBUTION PANEL A	Energized	Energized	
17	EE-PNL-AA2	125VDC POWER PANEL	In Service	In Service	
99	EE-PNL-AA3	125VDC POWER PANEL	Energized	Energized	Phase 3
37	EE-PNL-B	125VDC DISTRIBUTION PANEL B	Energized	Energized	

Item #	Equipment ID	Description	Equipment Normal State	Equipment Desired State	Notes
39	EE-PNL-BB2	125VDC POWER PANEL	In Service	In Service	
125	EE-PNL-CCP1A	CRITICAL CONTROL PANEL CCP1A	Energized	Energized	Phase 2
127	EE-PNL-CDP1A	CRITICAL DISTRIBUTION PANEL CDP1A	Energized	Energized	Phase 2
40	EE-PNL-NBPP	115/230 POWER PANEL	In Service	In Service	
89	EE-STRR-125B	125 VDC STARTER RACK B	Energized	Energized	
33	EE-STRR-125RCIC	125VDC RCIC STARTER RACK	Energized	Energized	
38	EE-STRR-250A	250VDC DIV I STARTER RACK	In Service	In Service	
31	EE-SWGR-125A	125VDC SWITCHGEAR BUS 1A	Energized	Energized	
36	EE-SWGR-125B	125VDC SWITCHGEAR BUS 1B	Energized	Energized	
34	EE-SWGR-250A	250VDC SWITCHGEAR BUS 1A	Energized	Energized	
98	EE-SWGR-4160F	4160V DIV I BUS	Energized	Energized	Phase 3
101	EE-SWGR-480F	480V SWGR CRITICAL BUS 1F	Energized	Energized	Phase 3
100	EE-XFMR-480F	STATION SERVICE TRANSFORMER 1F	Energized	Energized	Phase 3
85	EGM	RCIC GOVERNOR	Energized	Energized	
120	HPCI ASD PANEL	HPCI ALTERNATE SHUTDOWN PANEL	In Service	In Service	Level Indicators: NBI-LI-185B, NBI-LI-191B, PC-LI-110 , CM-LI-1681B
82	HPCI-PI-117A	ECST LEVEL	In Service	In Service	
9	IA-ACC-256A	SRV A ACCUMULATOR	Closed	Open/Closed	
10	IA-ACC-256B	SRV B ACCUMULATOR	Closed	Open/Closed	

Item #	Equipment ID	Description	Equipment Normal State	Equipment Desired State	Notes
11	IA-ACC-256C	SRV C ACCUMULATOR	Closed	Open/Closed	
12	IA-ACC-256D	SRV D ACCUMULATOR	Closed	Open/Closed	
13	IA-ACC-256E	SRV E ACCUMULATOR	Closed	Open/Closed	
14	IA-ACC-256F	SRV F ACCUMULATOR	Closed	Open/Closed	
15	IA-ACC-256G	SRV G ACCUMULATOR	Closed	Open/Closed	
16	IA-ACC-256H	SRV H ACCUMULATOR	Closed	Open/Closed	
130	LR-104	RACK-LR-104	In Service	In Service	Contains Indicator PC-PI-513
129	LR-139	RACK-LR-139	In Service	In Service	Contains Transmitter PC-PT-4A1
119	LRP-PNL-H	CONTROL ROOM VERTICAL PANEL H	In Service	In Service	Temperature Indicators: PC-TI-505A, PC-TI-505B, PC-TI-505C, PC-TI-505D, PC-TI-505E
132	LRP-PNL-PL1	Analog Process Cabinet	In Service	In Service	Signal conditioning for pressure transmitters PC-PT-4A1, PC-PT-30A, PC-PT-512A
102	MCC-CA	CRITICAL MCC CA	Energized	Energized	Phase 3
103	MCC-K	CRITICAL MCC K	Energized	Energized	Phase 3
43	MCC-LX	FEED TO 125VDC and 250VDC STATION SERVICE BATTERY CHARGERS 1C	N/A	N/A	
104	MCC-Q	CRITICAL MCC Q	Energized	Energized	Phase 3
131	MCC-RA	CRITICAL MCC RA	In Service	In Service	Powered via future manual transfer switch from future MOV-UPS for Phase 1, and MCC-K for Phase 2

Item #	Equipment ID	Description	Equipment Normal State	Equipment Desired State	Notes
1	MS-RV-71A	SAFETY RELIEF VALVE A	Closed	Open/Closed	SRV is normally closed but must be electrically opened to control reactor pressure.
2	MS-RV-71B	SAFETY RELIEF VALVE B	Closed	Open/Closed	SRV is normally closed but must be electrically opened to control reactor pressure.
3	MS-RV-71C	SAFETY RELIEF VALVE C	Closed	Open/Closed	SRV is normally closed but must be electrically opened to control reactor pressure.
4	MS-RV-71D	SAFETY RELIEF VALVE D	Closed	Open/Closed	SRV is normally closed but must be electrically opened to control reactor pressure.
5	MS-RV-71E	SAFETY RELIEF VALVE E	Closed	Open/Closed	SRV is normally closed but must be electrically opened to control reactor pressure.
6	MS-RV-71F	SAFETY RELIEF VALVE F	Closed	Open/Closed	SRV is normally closed but must be electrically opened to control reactor pressure.
7	MS-RV-71G	SAFETY RELIEF VALVE G	Closed	Open/Closed	SRV is normally closed but must be electrically opened to control reactor pressure.
8	MS-RV-71H	SAFETY RELIEF VALVE H	Closed	Open/Closed	SRV is normally closed but must be electrically opened to control reactor pressure.
54	NBI-PI-60A	RPV PRESSURE	In Service	In Service	Identified as NBI-PIS-60A in FLEX Strategy (Refs. 3 & 4), located on Rack 25-5
55	NBI-PI-60B	RPV PRESSURE	In Service	In Service	Identified as NBI-PIS-60B in FLEX Strategy (Refs. 3 & 4), located on Rack 25-6
56	NBI-PI-61	RPV PRESSURE	In Service	In Service	Located on Rack 25-51
109	Panel 9-15	Relay Panel 9-15	In Service	In Service	RPS Trip System A Cabinet

Item #	Equipment ID	Description	Equipment Normal State	Equipment Desired State	Notes
110	Panel 9-17	Relay Panel 9-17	In Service	In Service	RPS Trip System B Cabinet
111	Panel 9-18	Relay Panel 9-18	In Service	In Service	Reactor Vessel Level Control Cabinet
86	PANEL 9-3	Control Panel 9-3	In Service	In Service	
112	Panel 9-32	Relay Panel 9-32	In Service	In Service	Engineered Safeguard Relay Cabinet 1
113	Panel 9-33	Relay Panel 9-33	In Service	In Service	Engineered Safeguard Relay Cabinet 2
87	PANEL 9-4	Control Panel 9-4	In Service	In Service	
114	Panel 9-45	Relay Panel 9-45	In Service	In Service	Auto Blowdown Relay Cabinet
128	PANEL 9-5	Control Panel 9-5	In Service	In Service	Contains Indicators: RFC-LI-94A, RFC-LI-94B, RFC-LI-94C, RFC-PI-90A, RFC-PI-90B, RFC-PI-90C
108	Panel P2	CONTROL ROOM PANEL P2	In Service	In Service	NLS2014101 Hardened Containment Vents Capable of Operation Under Severe Accident Conditions (Order Number EA-13-109) (Ref. 24)
106	PC-AO-237	TORUS INLET OUTBOARD ISOLATION VALVE	Closed	Open	NLS2014101 Hardened Containment Vents Capable of Operation Under Severe Accident Conditions (Order Number EA-13-109) (Ref. 24)
107	PC-AO-32	TORUS INLET OUTBOARD ISOLATION VALVE	Closed	Open	NLS2014101 Hardened Containment Vents Capable of Operation Under Severe Accident Conditions (Order Number EA-13-109) (Ref. 24)
122	PC-LRPR-1A	CONTAINMENT/TORUS WIDE RANGE LEVEL RECORDER	In Service	In Service	Transmitters: PC-PT-4A1, PC-PT-30A, PC-PT-512A
105	PC-MO-233	TORUS INLET INBOARD ISOLATION VALVE	Closed	Open	NLS2014101 Hardened Containment Vents Capable of Operation Under Severe Accident Conditions (Order Number EA-13-109) (Ref. 24)

Item #	Equipment ID	Description	Equipment Normal State	Equipment Desired State	Notes
59	PC-PI-20	TORUS PRESSURE	In Service	In Service	
58	PC-PI-2104AG	DRYWELL PRESSURE	In Service	In Service	
60	PC-PI-2104BG	TORUS PRESSURE	In Service	In Service	
57	PC-PI-513	DRYWELL PRESSURE	In Service	In Service	Located on Rack-LR-104
124	PC-PT-30A	WIDE RANGE TORUS PRESSURE TRANSMITTER	In Service	In Service	Located near MCC-K
123	PC-PT-4A1	DRYWELL PRESSURE TRANSMITTER	In Service	In Service	Located on Rack-LR-139
66	PC-TE-1A	TORUS TEMPERATURE	In Service	In Service	TE is inside the Torus.
67	PC-TE-1B	TORUS TEMPERATURE	In Service	In Service	TE is inside the Torus.
68	PC-TE-1C	TORUS TEMPERATURE	In Service	In Service	TE is inside the Torus.
69	PC-TE-1D	TORUS TEMPERATURE	In Service	In Service	TE is inside the Torus.
70	PC-TE-1E	TORUS TEMPERATURE	In Service	In Service	TE is inside the Torus.
71	PC-TE-1F	TORUS TEMPERATURE	In Service	In Service	TE is inside the Torus.
72	PC-TE-1G	TORUS TEMPERATURE	In Service	In Service	TE is inside the Torus.
73	PC-TE-1H	TORUS TEMPERATURE	In Service	In Service	TE is inside the Torus.
74	PC-TE-2A	TORUS TEMPERATURE	In Service	In Service	TE is inside the Torus.
75	PC-TE-2B	TORUS TEMPERATURE	In Service	In Service	TE is inside the Torus.
76	PC-TE-2C	TORUS TEMPERATURE	In Service	In Service	TE is inside the Torus.
77	PC-TE-2D	TORUS TEMPERATURE	In Service	In Service	TE is inside the Torus.

Item #	Equipment ID	Description	Equipment Normal State	Equipment Desired State	Notes
78	PC-TE-2E	TORUS TEMPERATURE	In Service	In Service	TE is inside the Torus.
79	PC-TE-2F	TORUS TEMPERATURE	In Service	In Service	TE is inside the Torus.
80	PC-TE-2G	TORUS TEMPERATURE	In Service	In Service	TE is inside the Torus.
81	PC-TE-2H	TORUS TEMPERATURE	In Service	In Service	TE is inside the Torus.
61	PC-TE-505A	DRYWELL TEMPERATURE	In Service	In Service	
62	PC-TE-505B	DRYWELL TEMPERATURE	In Service	In Service	
63	PC-TE-505C	DRYWELL TEMPERATURE	In Service	In Service	
64	PC-TE-505D	DRYWELL TEMPERATURE	In Service	In Service	
65	PC-TE-505E	DRYWELL TEMPERATURE	In Service	In Service	
115	Rack 25-5	INSTRUMENT RACK 25-5	In Service	In Service	Transmitters: NBI-LT-52A, NBI-LT-52C, NBI-PT-53A, NBI-PT-53C, PC-PT-512A & Switches: NBI-LIS-83A, NBI-LIS-101A, NBI-LIS-101B, NBI-LIS-57A, NBI-LIS-57B, NBI-LIS-72A, NBI-LIS-72C.
117	Rack 25-51	INSTRUMENT RACK 25-51	In Service	In Service	Transmitters: NBI-LITS-73A
118	Rack 25-52	INSTRUMENT RACK 25-52	In Service	In Service	Transmitters: NBI-LITS-73B, NBI-PIS-52D, NBI-LT-91B
116	Rack 25-6	INSTRUMENT RACK 25-6	In Service	In Service	Transmitters: NBI-LT-52B, NBI-LT-59B & Switches: NBI-LIS-83B, NBI-LIS-101C, NBI-LIS-101D, NBI-LIS-58A, NBI-LIS-58B, NBI-LIS-72B, NBI-LIS-72D, NBI-PIS-52B
27	RCIC-MO-131	STM SUPP TO TURB VALVE	Closed	Open	
28	RCIC-MO-132	TURB OIL COOLING WTR SUPP VALVE	Closed	Open	

Item #	Equipment ID	Description	Equipment Normal State	Equipment Desired State	Notes
18	RCIC-MO-14	STEAM TRIP AND THROTTLE VALVE	Open	Open/Closed	
19	RCIC-MO-16	OUTBOARD STM SUPP ISOL VALVE	Closed	Open/Closed	
20	RCIC-MO-18	ECST PUMP SUCT VALVE	Open	Open/Closed	
21	RCIC-MO-20	RCIC DISCHARGE VALVE	Open	Open/Closed	
22	RCIC-MO-21	PUMP DISCH TO RX	Closed	Open/Closed	
23	RCIC-MO-27	MIN FLOW BYP	Closed	Closed	
24	RCIC-MO-30	TEST BYP TO ECST VALVE	Closed	Closed/Open	
25	RCIC-MO-33	ECST TEST LINE SHUTOFF VALVE	Closed	Closed	
26	RCIC-MO-41	SUCTION FROM THE SUPPRESSION CHAMBER	Closed	Open	
48	RFC-LI-94A	RPV LEVEL NARROW RANGE	In Service	In Service	Transmitter NBI-LT-52A located on Rack 25-5. LI located on Panel 9-5
49	RFC-LI-94B	RPV LEVEL NARROW RANGE	In Service	In Service	Transmitter NBI-LT-52B located on Rack 25-6. LI located on Panel 9-5
50	RFC-LI-94C	RPV LEVEL NARROW RANGE	In Service	In Service	Transmitter NBI-LT-52C located on Rack 25-5. LI located on Panel 9-5
51	RFC-PI-90A	RPV PRESSURE	In Service	In Service	Transmitter NBI-PT-53A located on Rack 25-5. PI located on Panel 9-5
52	RFC-PI-90B	RPV PRESSURE	In Service	In Service	Transmitter NBI-PT-53B located on Rack 25-6. PI located on Panel 9-5
53	RFC-PI-90C	RPV PRESSURE	In Service	In Service	Transmitter NBI-PT-53C located on Rack 25-5. PI located on Panel 9-5
96	RHR-1A	RHR PUMP 1-A	In Service	In Service	Phase 3
97	RHR-HX-1A	RHR HEAT EXCHANGER 1A	In Service	In Service	Phase 3
90	RHR-MO-12A	RHR HX-A OUTLET VALVE	In Service	In Service	Phase 3

Item #	Equipment ID	Description	Equipment Normal State	Equipment Desired State	Notes
91	RHR-MO-13A	RHR PUMP A TORUS SUCTION VALVE	In Service	In Service	Phase 3
92	RHR-MO-15A	RHR PUMP A SDC SUCTION VALVE	In Service	In Service	Phase 3
93	RHR-MO-25A	RHR INBD INJECTION VLV	In Service	In Service	Phase 3
94	RHR-MO-27A	RHR LOOP A OUTBD INJECTION VLV	In Service	In Service	Phase 3
95	RHR-MO-65A	RHR HX-A INLET VALVE	In Service	In Service	Phase 3

APPENDIX B: CNS ESEP HCLPF Values and Failure Modes Tabulation

Equipment ID	Description	Bldg.	Elev.	HCLPF (g, PGA)	Failure Mode	Basis
13-1	RCIC PUMP	RB	860	0.31	Anchorage	Evaluated per 13C4215-CAL-002
13-2	RCIC TURBINE	RB	860	0.31	Anchorage	Evaluated per 13C4215-CAL-002
ASD-ADS/REC PANEL	ADS ALTERNATE SHUTDOWN PANEL	RB	903	>RLGM	Screened	SRT Disposition
ECST 1A	EMERG COND. STOR. TK. 1A	CB	877	0.45	Anchorage	Evaluated per 13C4215-CAL-001
ECST 1B	EMERG COND. STOR. TK. 1B	CB	877	0.45	Anchorage	Evaluated per 13C4215-CAL-001
EE-BAT-125 1A	DIV. I 125 VDC STATION BATTERY	CB	903	0.45	Functional	Evaluated per 13C4215-CAL-003
EE-BAT-125 1B	DIV. II 125 VDC STATION BATTERY	CB	903	0.45	Anchorage	Evaluated per 13C4215-CAL-003
EE-BAT-250 1A	DIV. I 250 VDC STATION BATTERY	CB	903	0.45	Functional	Evaluated per 13C4215-CAL-003
EE-CHR-125C	125VDC BATTERY CHARGER 1C	CB	903	0.31	Anchorage	Evaluated per 13C4215-CAL-003
EE-CHR-250C	250V BATTERY CHARGER 1C	CB	903	0.31	Anchorage	Evaluated per 13C4215-CAL-003
EE-DSC-250C	FEED TO 250VDC STATION SERVICE BATTERY CHARGER 1C	CB	903	0.45	Functional	Evaluated per 13C4215-CAL-003
EE-IVTR-1A	DIV I INVERTER	CB	903	0.45	Functional	Evaluated per 13C4215-CAL-003
EE-PNL-125ASD	125 VDC ASD HPCI DISTRBUTION PANEL	RB	903	0.45	Functional	Evaluated per 13C4215-CAL-003
EE-PNL-A	125VDC DISTRIBUTION PANEL A	CB	903	0.45	Functional	Evaluated per 13C4215-CAL-003
EE-PNL-AA2	125VDC POWER PANEL	CB	903	0.45	Functional	Evaluated per 13C4215-CAL-003
EE-PNL-AA3	125VDC POWER PANEL	RB	903	0.45	Functional	Evaluated per 13C4215-CAL-003
EE-PNL-B	125VDC DISTRIBUTION PANEL B	CB	903	0.45	Functional	Evaluated per 13C4215-CAL-003
EE-PNL-BB2	125VDC POWER PANEL	CB	903	0.45	Functional	Evaluated per 13C4215-CAL-003
EE-PNL-CCP1A	CRITICAL CONTROL PANEL CCP1A	CB	918	0.45	Functional	Evaluated per 13C4215-CAL-003
EE-PNL-CDP1A	CRITICAL DISTRIBUTION PANEL CDP1A	CB	903	0.45	Functional	Evaluated per 13C4215-CAL-003
EE-PNL-NBPP	115/230 POWER PANEL	CB	918	0.45	Functional	Evaluated per 13C4215-CAL-003
EE-STRR-125B	125 VDC STARTER RACK B	RB	859	>RLGM	Screened	SRT disposition
EE-STRR-125RCIC	125VDC RCIC STARTER RACK	RB	903	0.38	Anchorage	Evaluated per 13C4215-CAL-003

Equipment ID	Description	Bldg.	Elev.	HCLPF (g, PGA)	Failure Mode	Basis
EE-STRR-250A	250VDC DIV I STARTER RACK	RB	903	0.45	Functional	Evaluated per 13C4215-CAL-003
EE-SWGR-125A	125VDC SWITCHGEAR BUS 1A	CB	903	0.286	Anchorage	Evaluated per 13C4215-CAL-003
EE-SWGR-125B	125VDC SWITCHGEAR BUS 1B	CB	903	0.286	Anchorage	Evaluated per 13C4215-CAL-003
EE-SWGR-250A	250VDC SWITCHGEAR BUS 1A	CB	903	0.286	Anchorage	Evaluated per 13C4215-CAL-003
EE-SWGR-4160F	4160V DIV I BUS	RB	932	0.45	Functional	Evaluated per 13C4215-CAL-003
EE-SWGR-480F	480V SWGR CRITICAL BUS 1F	RB	932	0.45	Functional	Evaluated per 13C4215-CAL-003
EE-XFMR-480F	STATION SERVICE TRANSFORMER 1F	RB	932	0.45	Functional	Evaluated per 13C4215-CAL-003
EGM	RCIC GOVERNOR	RB	860	>RLGM	Screened	SRT Disposition
HPCI ASD PANEL	HPCI ALTERNATE SHUTDOWN PANEL	RB	903	0.45	Functional	Evaluated per 13C4215-CAL-003
HPCI-PI-117A	ECST LEVEL	CB	877	>RLGM	Screened	SRT disposition
IA-ACC-256A	SRV A ACCUMULATOR	DW	921	>RLGM	Screened	SRT disposition
IA-ACC-256B	SRV B ACCUMULATOR	DW	921	>RLGM	Screened	SRT disposition
IA-ACC-256C	SRV C ACCUMULATOR	DW	921	>RLGM	Screened	SRT disposition
IA-ACC-256D	SRV D ACCUMULATOR	DW	921	>RLGM	Screened	SRT disposition
IA-ACC-256E	SRV E ACCUMULATOR	DW	921	>RLGM	Screened	SRT disposition
IA-ACC-256F	SRV F ACCUMULATOR	DW	921	>RLGM	Screened	SRT disposition
IA-ACC-256G	SRV G ACCUMULATOR	DW	921	>RLGM	Screened	SRT disposition
IA-ACC-256H	SRV H ACCUMULATOR	DW	921	>RLGM	Screened	SRT disposition
LR-104	RACK-LR-104	RB	903	>RLGM	Screened	SRT disposition
LR-139	RACK-LR-139	RB	958	>RLGM	Screened	SRT disposition
LRP-PNL-H	CONTROL ROOM VERTICAL PANEL H	CB	932	0.45	Functional	Evaluated per 13C4215-CAL-003
LRP-PNL-PL1	ANALOG PROCESS CABINET	CB	918	>RLGM	Screened	SRT disposition
MCC-CA	CRITICAL MCC CA	RB	932	>RLGM	Screened	SRT disposition
MCC-K	CRITICAL MCC K	RB	903	>RLGM	Screened	SRT Disposition
MCC-LX	FEED TO 125VDC and 250VDC STATION SERVICE BATTERY CHARGERS 1C	CB	903	>RLGM	Screened	SRT Disposition
MCC-Q	CRITICAL MCC Q	RB	903	>RLGM	Screened	SRT Disposition
MCC-RA	CRITICAL MCC RA	RB	958	>RLGM	Screened	SRT Disposition
MS-RV-71A	SAFETY RELIEF VALVE A	DW	921	>RLGM	Screened	SRT disposition
MS-RV-71B	SAFETY RELIEF VALVE B	DW	921	>RLGM	Screened	SRT disposition
MS-RV-71C	SAFETY RELIEF VALVE C	DW	921	>RLGM	Screened	SRT disposition
MS-RV-71D	SAFETY RELIEF VALVE D	DW	921	>RLGM	Screened	SRT disposition
MS-RV-71E	SAFETY RELIEF VALVE E	DW	921	>RLGM	Screened	SRT disposition
MS-RV-71F	SAFETY RELIEF VALVE F	DW	921	>RLGM	Screened	SRT disposition

Equipment ID	Description	Bldg.	Elev.	HCLPF (g, PGA)	Failure Mode	Basis
MS-RV-71G	SAFETY RELIEF VALVE G	DW	921	>RLGM	Screened	SRT disposition
MS-RV-71H	SAFETY RELIEF VALVE H	DW	921	>RLGM	Screened	SRT disposition
NBI-PI-60A	RPV PRESSURE	RB	931	0.45	Functional	R.O.B to Rack 25-05
NBI-PI-60B	RPV PRESSURE	RB	931	0.45	Functional	R.O.B to Rack 25-06
NBI-PI-61	RPV PRESSURE	RB	903	0.45	Functional	R.O.B to Rack 25-51
Panel 9-15	Relay Panel 9-15	CB	932	0.45	Functional	Evaluated per 13C4215-CAL-003
Panel 9-17	Relay Panel 9-17	CB	932	0.45	Functional	Evaluated per 13C4215-CAL-003
Panel 9-18	Relay Panel 9-18	CB	932	0.45	Functional	Evaluated per 13C4215-CAL-003
PANEL 9-3	Control Panel 9-3	CB	932	0.45	Functional	Evaluated per 13C4215-CAL-003
Panel 9-32	Relay Panel 9-32	CB	903	0.38	Anchorage	Evaluated per 13C4215-CAL-003
Panel 9-33	Relay Panel 9-33	CB	903	0.38	Anchorage	Evaluated per 13C4215-CAL-003
PANEL 9-4	Control Panel 9-4	CB	932	0.45	Functional	Evaluated per 13C4215-CAL-003
Panel 9-45	Relay Panel 9-45	CB	903	0.38	Anchorage	Evaluated per 13C4215-CAL-003
PANEL 9-5	Control Panel 9-5	CB	932	0.45	Functional	Evaluated per 13C4215-CAL-003
Panel P2	CONTROL ROOM PANEL P2	CB	932	0.45	Functional	Evaluated per 13C4215-CAL-003
PC-AO-237	TORUS INLET OUTBOARD ISOLATION VALVE	RB	881	>RLGM	Screened	SRT Disposition
PC-AO-32	TORUS INLET OUTBOARD ISOLATION VALVE	RB	881	>RLGM	Screened	SRT Disposition
PC-LRPR-1A	CONTAINMENT/TORUS WIDE RANGE LEVEL RECORDER	CB	932	0.45	Functional	R.O.B of Panel 9-3
PC-MO-233	TORUS INLET INBOARD ISOLATION VALVE	RB	881	>RLGM	Screened	SRT disposition
PC-PI-20	TORUS PRESSURE	RB	903	0.381	Anchorage	Evaluated per 13C4215-CAL-003
PC-PI-2104AG	DRYWELL PRESSURE	RB	931	>RLGM	Screened	SRT disposition
PC-PI-2104BG	TORUS PRESSURE	RB	903	>RLGM	Screened	SRT disposition
PC-PI-513	DRYWELL PRESSURE	RB	903	>RLGM	Screened	R.O.B of rack LR-104
PC-PT-30A	WIDE RANGE TORUS PRESSURE TRANSMITTER	RB	903	>RLGM	Screened	SRT disposition
PC-PT-4A1	DRYWELL PRESSURE TRANSMITTER	RB	958	>RLGM	Screened	R.O.B of Rack LR-139
PC-TE-1A	TORUS TEMPERATURE	RB	859	>RLGM	Screened	SRT disposition
PC-TE-1B	TORUS TEMPERATURE	RB	859	>RLGM	Screened	SRT disposition
PC-TE-1C	TORUS TEMPERATURE	RB	859	>RLGM	Screened	SRT disposition

Equipment ID	Description	Bldg.	Elev.	HCLPF (g, PGA)	Failure Mode	Basis
PC-TE-1D	TORUS TEMPERATURE	RB	859	>RLGM	Screened	SRT disposition
PC-TE-1E	TORUS TEMPERATURE	RB	859	>RLGM	Screened	SRT disposition
PC-TE-1F	TORUS TEMPERATURE	RB	859	>RLGM	Screened	SRT disposition
PC-TE-1G	TORUS TEMPERATURE	RB	859	>RLGM	Screened	SRT disposition
PC-TE-1H	TORUS TEMPERATURE	RB	859	>RLGM	Screened	SRT disposition
PC-TE-2A	TORUS TEMPERATURE	RB	859	>RLGM	Screened	SRT disposition
PC-TE-2B	TORUS TEMPERATURE	RB	859	>RLGM	Screened	SRT disposition
PC-TE-2C	TORUS TEMPERATURE	RB	859	>RLGM	Screened	SRT disposition
PC-TE-2D	TORUS TEMPERATURE	RB	859	>RLGM	Screened	SRT disposition
PC-TE-2E	TORUS TEMPERATURE	RB	859	>RLGM	Screened	SRT disposition
PC-TE-2F	TORUS TEMPERATURE	RB	859	>RLGM	Screened	SRT disposition
PC-TE-2G	TORUS TEMPERATURE	RB	859	>RLGM	Screened	SRT disposition
PC-TE-2H	TORUS TEMPERATURE	RB	859	>RLGM	Screened	SRT disposition
PC-TE-505A	DRYWELL TEMPERATURE	DW	921	>RLGM	Screened	SRT disposition
PC-TE-505B	DRYWELL TEMPERATURE	DW	921	>RLGM	Screened	SRT disposition
PC-TE-505C	DRYWELL TEMPERATURE	DW	921	>RLGM	Screened	SRT disposition
PC-TE-505D	DRYWELL TEMPERATURE	DW	921	>RLGM	Screened	SRT disposition
PC-TE-505E	DRYWELL TEMPERATURE	DW	921	>RLGM	Screened	SRT disposition
Rack 25-5	INSTRUMENT RACK 25-5	RB	931	0.45	Functional	Evaluated per 13C4215-CAL-003
Rack 25-51	INSTRUMENT RACK 25-51	RB	903	0.45	Functional	Evaluated per 13C4215-CAL-003
Rack 25-52	INSTRUMENT RACK 25-52	RB	903	0.45	Functional	Evaluated per 13C4215-CAL-003
Rack 25-6	INSTRUMENT RACK 25-6	RB	931	0.45	Functional	Evaluated per 13C4215-CAL-003
RCIC-MO-131	STM SUPP TO TURB VALVE	RB	860	>RLGM	Screened	SRT Disposition
RCIC-MO-132	TURB OIL COOLING WTR SUPP VALVE	RB	860	>RLGM	Screened	SRT Disposition
RCIC-MO-14	STEAM TRIP AND THROTTLE VALVE	RB	860	>RLGM	Screened	SRT Disposition
RCIC-MO-16	OUTBOARD STM SUPP ISOL VALVE	RB	903	>RLGM	Screened	SRT Disposition
RCIC-MO-18	ECST PUMP SUCT VALVE	RB	860	>RLGM	Screened	SRT disposition
RCIC-MO-20	RCIC DISCHARGE VALVE	RB	881	>RLGM	Screened	SRT disposition
RCIC-MO-21	PUMP DISCH TO RX	RB	903	>RLGM	Screened	SRT disposition
RCIC-MO-27	MIN FLOW BYP	RB	860	>RLGM	Screened	SRT Disposition
RCIC-MO-30	TEST BYP TO ECST VALVE	RB	881	>RLGM	Screened	SRT Disposition
RCIC-MO-33	ECST TEST LINE SHUTOFF VALVE	RB	881	>RLGM	Screened	SRT disposition
RCIC-MO-41	SUCTION FROM THE SUPPRESSION CHAMBER	RB	860	>RLGM	Screened	SRT disposition
RFC-LI-94A	RPV LEVEL NARROW RANGE	CB	932	0.45	Functional	R.O.B of Panel 9-5
RFC-LI-94B	RPV LEVEL NARROW RANGE	CB	932	0.45	Functional	R.O.B of Panel 9-5

Equipment ID	Description	Bldg.	Elev.	HCLPF (g, PGA)	Failure Mode	Basis
RFC-LI-94C	RPV LEVEL NARROW RANGE	CB	932	0.45	Functional	R.O.B of Panel 9-5
RFC-PI-90A	RPV PRESSURE	CB	932	0.45	Functional	R.O.B of Panel 9-5
RFC-PI-90B	RPV PRESSURE	CB	932	0.45	Functional	R.O.B of Panel 9-5
RFC-PI-90C	RPV PRESSURE	CB	932	0.45	Functional	R.O.B of Panel 9-5
RHR-1A	RHR PUMP 1-A	RB	860	0.45	Functional	Evaluated per 13C4215-CAL-002
RHR-HX-1A	RHR HEAT EXCHANGER 1A	RB	931	0.43	Anchorage	Evaluated per 13C4215-CAL-002
RHR-MO-12A	RHR HX-A OUTLET VALVE	RB	903	>RLGM	Screened	SRT disposition
RHR-MO-13A	RHR PUMP A TORUS SUCTION VALVE	RB	859	>RLGM	Screened	SRT disposition
RHR-MO-15A	RHR PUMP A SDC SUCTION VALVE	RB	881	>RLGM	Screened	SRT disposition
RHR-MO-25A	RHR INBD INJECTION VLV	RB	903	>RLGM	Screened	SRT disposition
RHR-MO-27A	RHR LOOP A OUTBD INJECTION VLV	RB	903	>RLGM	Screened	SRT disposition
RHR-MO-65A	RHR HX-A INLET VALVE	RB	931	>RLGM	Screened	SRT disposition