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December 16, 2014

U.S. Nuclear Regulatory Commission
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SUBJECT: Pilgrim's Expedited Seismic Evaluation Process 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

Pilgrim Nuclear Power Station
Docket No. 50-293
License No. DPR-35

LETTER NUMBER 2.14.082

REFERENCES:

1. NRC Letter "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 (ML12053A340)
2. NEI Letter to NRC, "Proposed Path Forward for NTTF Recommendation 2.1: Seismic Reevaluations", dated April 9, 2013 (ML13101A345)
3. NRC Letter, "Electric Power Research Institute Final Draft Report XXXXXX, Seismic Evaluation Guidance: Augmented Approach for the Resolution of Near-Term Task Force Recommendation 2.1: Seismic, as an Acceptable Alternative to the March 12, 2012, Information Request for Seismic Reevaluations", dated May 7, 2013 (ML13106A331)

Dear Sir or Madam:

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued Reference 1 to all power reactor licensees and holders of construction permits in active or deferred status. Enclosure 1 of Reference 1 requested each addressee located in the Central and Eastern United States (CEUS) to submit a Seismic Hazard Evaluation within 1.5 years from the date of Reference 1.

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In Reference 2, the Nuclear Energy Institute (NEI) requested NRC agreement to delay submittal of the final CEUS Seismic Hazard and Screening Reports so that an update to the Electric Power Research Institute (EPRI) ground motion attenuation model could be completed and used to develop that information. NEI proposed that descriptions of subsurface materials and properties and base case velocity profiles be submitted to the NRC by September 12, 2013, with the remaining seismic hazard and screening information submitted by March 31, 2014. NRC agreed with that proposed path forward in Reference 3.

Reference 1 requested that licensees provide 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. In accordance with the NRC endorsed guidance in Reference 3, the attached Expedited Seismic Evaluation Process Report for Pilgrim Nuclear Power Station (Attachment 1) provides the information described in Section 7 of Reference 3 in accordance with the schedule identified in Reference 2.

This letter contains new regulatory commitments as shown in Attachment 2.

Should you have any questions concerning the content of this letter, please contact Mr. Everett (Chip) Perkins Jr. at (508) 830-8323.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 16, 2014.

Sincerely,

John A. Dent Jr.
Site Vice President

JAD/rmb

Attachment: 1] Expedited Seismic Evaluation Process Report for Pilgrim Nuclear Power Station

2] List of Regulatory Commitments for Pilgrim Nuclear Power Station

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Pilgrim Nuclear Power Station

ATTACHMENT 1 to

PNPS Letter 2.14.082

EXPEDITED SEISMIC EVALUATION PROCESS REPORT

FOR

PILGRIM NUCLEAR POWER STATION

EXPEDITED SEISMIC EVALUATION PROCESS (ESEP) REPORT FOR PILGRIM NUCLEAR POWER STATION (PNPS)

Table of Contents

Table of Contents
(continued)

List of Tables

List of Figures

1.0 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 [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 Pilgrim Nuclear Power Station (PNPS). The intent of the ESEP is to perform an interim action in response to the NRC's 50.54(f) letter to demonstrate seismic margin through a review of a subset of the plant equipment that can be relied upon to protect the reactor core following beyond design basis seismic events.

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

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

2.0 BRIEF SUMMARY OF THE FLEX SEISMIC IMPLEMENTATION STRATEGIES

The PNPS FLEX strategies for Reactor Core Cooling and Heat Removal, Reactor Inventory Control, and Containment Function are summarized below. This summary is derived from the PNPS Overall Integrated Plan (OIP) in Response to the March 12, 2012, Commission Order EA-12-049 submitted in February 2013 [3] and is consistent with the third six month status report issued to the NRC in August 2014 [4].

For Phase 1 Core cooling and inventory control are achieved during the first six (6) hours using the Reactor Core Isolation Cooling (RCIC) system aligned to take suction from the torus. Pressure control and heat removal are accomplished by Safety Relief Valves (SRV) venting to the torus. At six (6) hours, a controlled depressurization is commenced based on the Emergency Operating Procedure (EOP) heat capacity temperature limit curve. The depressurization is carried out over three (3) hours using RCIC and cycling the SRVs.

At nine (9) hours (beginning of Phase 2), the operators will transition from the installed RCIC system to diesel powered FLEX low pressure injection pumps, taking suction from the Ultimate Heat Sink (UHS) and connecting through the Condensate Storage Tank (CST) suction line for injection via either the High Pressure Coolant Injection (HPCI) or RCIC idle pump and normal pump discharge path to the Reactor Pressure Vessel (RPV) feedwater lines. An alternate FLEX injection point is to the Residual Heat

Removal (RHR) system via the readily accessible Firewater to Service Water Cross-tie to RHR, which provides a path into the RPV, Drywell Spray, or Torus via the RHR system.

At 16 hours, the torus will be vented via the hardened containment vent to provide containment heat removal, and to begin a long term strategy of reactor feedwater makeup and boiling to protect the core and containment.

At 72 hours (beginning of Phase 3), the water source for the FLEX injection pumps will be transitioned to the mobile water tank fed from the on-site ground water wells. The ground water wells will be powered by a portable 100 kVA generator. The flow from the mobile storage tank will be passed through a FLEX Demineralizer vessel and injected into the RPV via the CST storage tank suction line (same flow path as Phase 2).

Initially containment integrity is maintained by normal design features of the containment (e.g., containment isolation valves). At 16 hours, when torus water temperature reaches 280°F, torus venting will commence through the Hardened Containment Venting System to provide containment heat removal and protect containment integrity. Containment venting will not begin until after reactor depressurization to ensure sufficient containment pressure and net positive suction head for RCIC or HPCI operation.

Necessary electrical components are outlined in the PNPS FLEX OIP submittal, and primarily entail a 125 volt motor control center, vital batteries, battery chargers, and 250 volt DC batteries and battery chargers. Other supporting components include monitoring instrumentation for core cooling, reactor coolant inventory, and containment integrity.

Figure 1 through Figure 5 of [3] provide the FLEX flow paths for PNPS Phases 1 through 3.

3.0 EQUIPMENT SELECTION PROCESS AND ESEL

The selection of equipment for the Expedited Seismic Equipment List (ESEL) followed the guidelines of EPRI 3002000704 [2]. The ESEL for PNPS is presented in Attachment A. Information presented in Attachment A is drawn from the following references [3], [4], [5], [6], [7], [8], [9], [10], [11], [12], [13], [14], [15], [16], [17], [18], [19], [20], [21], [22], [23], [24], [25], [26], [27], [28], [29], [30], [31], [32], [33], [34], [35], and [36].

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 PNPS OIP in Response to the March 12, 2012, Commission Order EA-12-049 [3]. The OIP provides the PNPS 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 PNPS OIP. FLEX recovery actions are excluded from the ESEP scope per EPRI 3002000704 [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 subcriticality, and containment integrity functions. Portable and pre-staged FLEX equipment (not permanently installed) are excluded from the ESEL per EPRI 3002000704.

The ESEL component selection followed the EPRI guidance outlined in Section 3.2 of EPRI 3002000704.

1. The scope of components is limited to that required to accomplish the core cooling and containment safety functions identified in Table 3-1 of EPRI 3002000704. The instrumentation monitoring requirements for core cooling/containment safety functions are limited to those outlined in the EPRI 3002000704 guidance, and are a subset of those outlined in the PNPS OIP.
2. The scope of components is limited to installed plant equipment, and FLEX connections necessary to implement the PNPS OIP 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 3002000704 [2] guidance are:
   - Structures (e.g. containment, reactor building, control building, auxiliary building, etc.).
   - Piping, cabling, conduit, HVAC, and their supports.
   - Manual valves, check 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. RPV 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 PNPS OIP [3] to determine the major equipment involved in the FLEX strategies. Further reviews of plant drawings (e.g., Piping and Instrumentation Diagrams (P&IDs) and Electrical One Line Diagrams) were performed to identify the boundaries of the flowpaths to be used in the FLEX strategies and to identify specific components in the flowpaths 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 strategy electrical or fluid flowpath. 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 drawings, system descriptions, design basis documents, etc., as necessary.

Cabinets and equipment controls containing relays, contactors, switches, potentiometers, circuit breakers and other electrical and instrumentation that could be affected by high-frequency earthquake motions and that impact the operation of equipment in the ESEL are required to be on the ESEL. These cabinets and components were identified in the ESEL. For the ESEL, the relays identified were in the

RCIC and Automatic Depressurization System (ADS), and malfunction of these relays during a seismic event could lead to the failure of the reactor core cooling safety function.

For Phase 1, RCIC is the primary path for inventory control and core cooling. Therefore, the RCIC system was used as the basis for the Phase 1 ESEL. For Phase 2 and Phase 3, the RCIC system was also used to provide the pathway for RPV injection utilizing portable injection pumps. Relays that could malfunction during a seismic event and prevent successful RCIC or ADS operation were included in the ESEL.

For each parameter monitored during the FLEX implementation, a single indication was selected for inclusion in the ESEL. For each parameter indication, the components along the flow path from measurement to indication were included, since any failure along the path would lead to failure of that indication. Components such as flow elements were considered as part of the piping and were not included in the ESEL.

3.1.2 Power Operated Valves

Page 3-3 of EPRI 3002000704 [2] notes that power operated valves not required to change state as part of the FLEX mitigation strategies 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 PNPS 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 be added to the ESEL as these components provide completely passive locations for pulling or installing cables. No breaks or connections in the cabling were included in pull boxes. Pull boxes were considered part of conduit and cabling, which were excluded in accordance with EPRI 3002000704 [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; however, the cabinets are included in the ESEL to ensure industry knowledge on panel/anchorage failure vulnerabilities is addressed.

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 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 PNPS OIP [3] 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 3002000704 [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

RCIC is the primary system for Phase 1 and was presented as the single success path in the PNPS ESEL. Therefore, no additional justification is required.

4.0 GROUND MOTION RESPONSE SPECTRUM (GMRS)

4.1 Plot of GMRS Submitted by the Licensee

The Safe Shutdown Earthquake (SSE) control point elevation is defined at the bottom of the Reactor Building foundation at elevation -26 ft MSL which is 48 ft below grade based on Section 2.5.3.3.2, Section 2.5.2.4.3, and Figure 12.2-6 of the Final Safety Analysis Report (FSAR) [37]. Table 4-1 shows the GMRS acceleration for a range of frequencies [38]. The GMRS at the control point is shown in Figure 4-1.

Table 4-1: GMRS for PNPS

Frequency (Hz)	GMRS (g)
100	5.05E-01
90	5.09E-01
80	5.18E-01
70	5.40E-01
60	5.93E-01
50	7.31E-01
40	8.57E-01
35	9.21E-01

Table 4-1: GMRS for PNPS (continued)

Frequency (Hz)	GMRS (g)
30	9.51E-01
25	9.22E-01
20	9.09E-01
15	9.61E-01
12.5	1.06E+00
10	1.18E+00
9	1.16E+00
8	1.10E+00
7	9.75E-01
6	8.07E-01
5	6.09E-01
4	3.83E-01
3.5	2.93E-01
3	2.19E-01
2.5	1.61E-01
2	1.29E-01
1.5	1.00E-01
1.25	7.82E-02
1	6.22E-02
0.9	5.87E-02
0.8	5.55E-02
0.7	5.18E-02
0.6	4.67E-02
0.5	3.92E-02
0.4	3.14E-02
0.35	2.74E-02
0.3	2.35E-02
0.25	1.96E-02
0.2	1.57E-02
0.15	1.18E-02
0.125	9.80E-03
0.1	7.84E-03

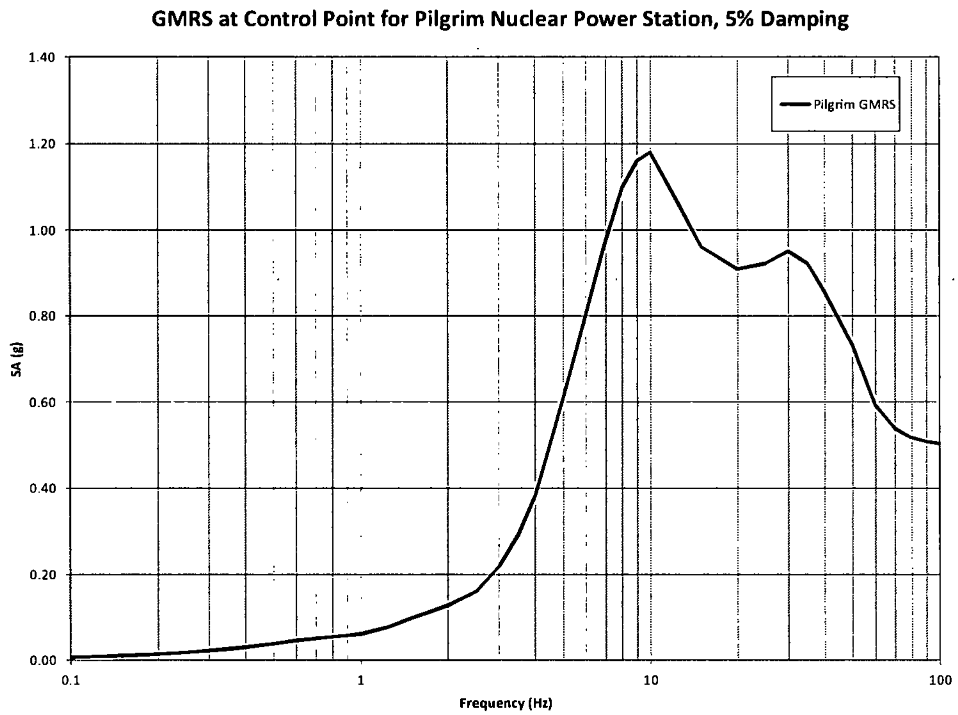

Figure 4-1: GMRS for PNPS

4.2 Comparison to SSE

The SSE is defined in the FSAR in terms of a Peak Ground Acceleration (PGA) and a design response spectrum. These spectra have been digitized and tabulated [39]. Table 4-2 shows the spectral acceleration values at selected frequencies for the 5% damped horizontal SSE.

Table 4-2: SSE for PNPS

Frequency (Hz)	Spectral Acceleration (g)
100	0.15
33	0.15
25	0.15
10	0.184
9	0.194
5	0.238
2.5	0.225
1	0.126
0.5	0.071

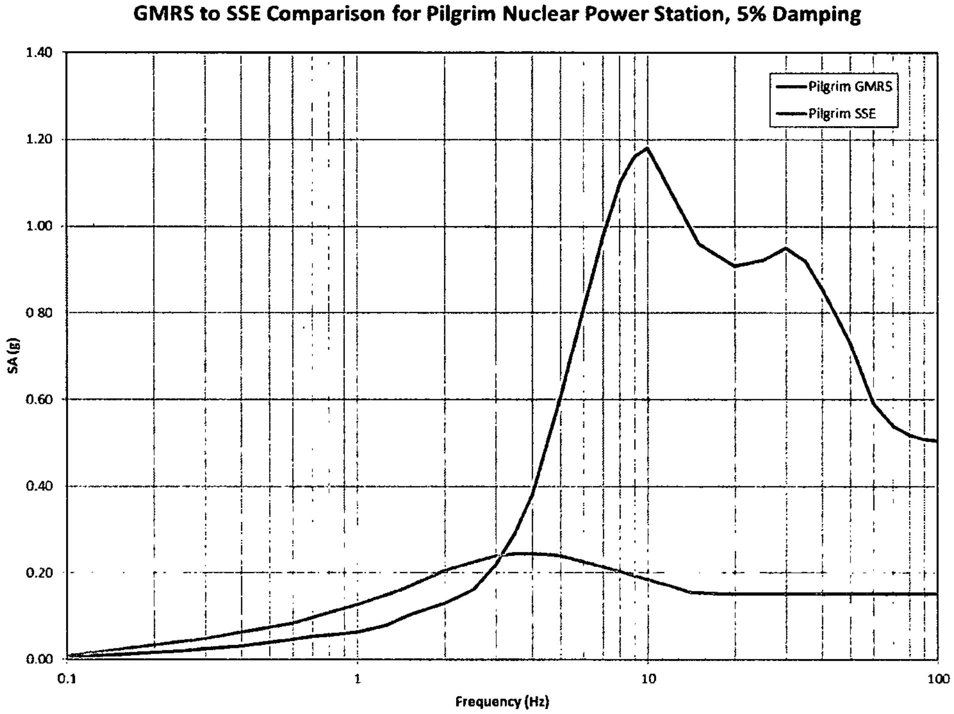

Figure 4-2: GMRS to SSE Comparison for PNPS

The SSE envelops the GMRS in the low frequency range up to approximately 3 Hz. The GMRS exceeds the SSE beyond that point. As the GMRS exceeds the SSE in the 1 to 10 Hz range, the plant does not screen out of the ESEP according to Section 2.2 of EPRI 3002000704 [2]. The two special screening considerations as described in Section 2.2.1 of EPRI 3002000704, namely a) Low-frequency GMRS exceedances at Low Seismic Hazard Sites and b) Narrow Band Exceedances in the 1 to 10 Hz range, provide criteria for accepting specific GMRS exceedances. However, the GMRS exceedances are not limited to the low frequency range and there are no narrow-banded exceedances. Therefore, these special screening considerations do not apply for PNPS and hence High Confidence of a Low Probability of Failure (HCLPF) evaluations were performed.

5.0 REVIEW LEVEL GROUND MOTION (RLGM)

5.1 Description of RLGM Selected

The RLGM is selected based on Approach 1 in Section 4 of EPRI 3002000704 [2]. The RLGM is developed based on the SSE [39].

The maximum GMRS/SSE ratio between 1 and 10 Hz range occurs at 10 Hz where the ratio is 1.18/0.184 = 6.41. As the maximum ratio of the GMRS to the SSE over the 1 to 10 Hz range exceeds a value of 2, the GMRS/SSE ratio is set to the maximum scaling factor value of 2.0 for PNPS in accordance with Section 4 of EPRI 3002000704. Table 5-1 lists the horizontal ground RLGM acceleration at 5% damping at selected frequencies and the plot is shown in Figure 5-1. The RLGM are generated by plotting the digitized data on a linear/linear graph paper, and connecting the points with straight lines.

Table 5-1: RLGM for PNPS

Frequency (Hz)	RLGM at 5% Damping (g)
0.10	0.016
0.30	0.094
0.60	0.166
1.00	0.252
1.40	0.318
2.00	0.412
2.50	0.450
3.00	0.476
3.50	0.488
4.00	0.488
4.50	0.484
5.00	0.476
5.50	0.466
6.00	0.450
7.00	0.428
8.00	0.406
10.00	0.368
12.00	0.338
14.00	0.308
18.00	0.300
24.00	0.300
33.00	0.300

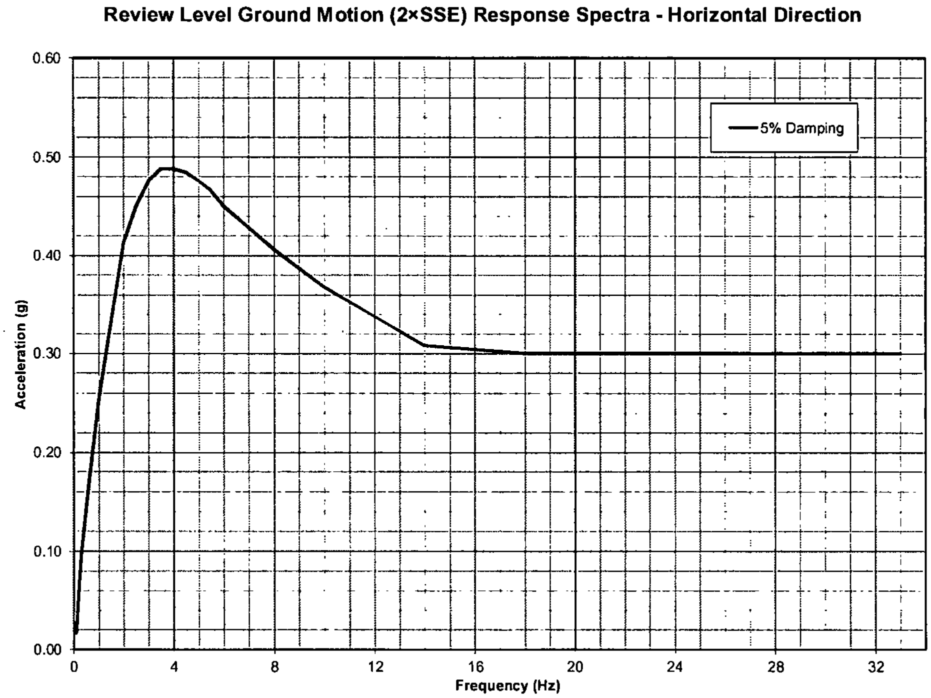

Figure 5-1: RLGM for PNPS

5.2 Method to Estimate In-Structure Response Spectra (ISRS)

The RLGM ISRS for PNPS are generated by scaling the SSE ISRS [39]. The following steps are used to generate the RLGM ISRS.

1. Obtain the horizontal direction SSE ISRS for a particular damping value.
2. Calculate the horizontal RLGM ISRS by scaling the horizontal direction SSE ISRS by a factor of 2.0.
3. Repeat steps 1 and 2 to obtain RLGM ISRS for multiple damping values.

The vertical direction RLGM ISRS is obtained by scaling the vertical amplified ground response spectrum.

6.0 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 PGA for which there is a 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 3002000704 [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-SL, A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1) [40].

2. Probabilistic approach using the fragility analysis methodology of EPRI TR-103959, Methodology for Developing Seismic Fragilities [41].

6.1 Summary of Methodologies Used

PNPS performed a SPRA in 1994 as part of Individual Plant Examination for External Events (IPEEE) program. The SPRA is documented in [23] and consisted of screening walkdowns, fragility analysis and three-dimensional soil structure interaction (SSI) analysis. The SPRA was a hybrid of the conventional PRA and seismic margin assessment approaches. The seismic walkdowns for IPEEE were performed simultaneously with USI A-46 evaluations. Section 3.3 of [38] established that the results of the PNPS SPRA performed as part of IPEEE are not sufficient to serve as the basis for PNPS to screen-out of further risk assessment.

For ESEP, the SMA consisted of screening walkdowns and HCLPF calculations. The screening walkdowns used the screening tables from Chapter 2 of EPRI NP-6041-SL [40]. The walkdowns were conducted by engineers trained in EPRI NP-6041-SL and were documented on Screening Evaluation Work Sheets (SEWS) from EPRI NP-6041-SL. Anchorage capacity calculations used the CDFM criteria from EPRI NP-6041-SL. Seismic demand was based on EPRI 3002000704 [2] using an RLGM of 2xSSE with a PGA of 0.3g, Figure 5-1.

6.2 HCLPF Screening Process

For ESEP, the components are screened considering the RLGM (2×SSE) with a 0.3g PGA. The screening tables in EPRI NP-6041-SL [40] are based on ground peak spectral accelerations of 0.8g and 1.2g. These both exceed the RLGM peak spectral acceleration.

The ESEL components were prescreened based on Table 2-4 of EPRI NP-6041-SL. Additional pre-screening, specifically for anchorage, considered walkdown results and documentation from NTTF 2.3 and SEWS from IPEEE and USI A-46. Equipment anchorage was screened out in cases where previous evaluations showed large available margin against SSE. The remaining components (i.e., components that do not screen out), were identified as requiring HCLPF calculations. ESEL components were walked down and based on the equipment and anchorage conditions, prescreening decisions were confirmed and a final list of required HCLPF calculations was generated. Equipment for which the screening caveats were met and for which the anchorage capacity exceeded the RLGM seismic demand are screened out from ESEP seismic capacity determination because the HCLPF capacity exceeds the RLGM.

The PNPS ESEL contains 187 items. Of these, 16 are valves. In accordance with Table 2-4 of EPRI NP-6041-SL, active valves may be assigned a functional capacity of 0.8g peak spectral acceleration without any review other than looking for valves with large extended operators on small diameter piping, and anchorage is not a failure mode. Therefore, valves on the ESEL are screened out from ESEP seismic capacity determination, subject to the caveat regarding large extended operators on small diameter piping.

The non-valve components in the ESEL are screened based on the SMA results. If the SMA showed that the component met the EPRI NP-6041-SL screening caveats and the CDFM capacity exceeded the RLGM demand, the components are screened out from the ESEP capacity determination.

Six (6) block walls were identified in the proximity of ESEL equipment. These block walls were assessed for potential seismic interaction impact resulting from the RLGM by reviewing the existing plant documents and or by generating new analysis and found to be acceptable.

6.3 Seismic Walkdown Approach

6.3.1 Walkdown Approach

Walkdowns were performed in accordance with the criteria provided in Section 5 of EPRI 3002000704 [2], which refers to EPRI NP-6041-SL [40] for the Seismic Margin Assessment process. Pages 2-26 through 2-30 of EPRI NP-6041-SL 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] 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 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."

6.3.2 Application of Previous Walkdown Information

Several ESEL items were previously walked down during the PNPS seismic IPEEE program, for seismic IPEEE outlier resolutions in accordance with USI A-46 evaluation program and NTTF Recommendation 2.3. Those walkdown results were reviewed and the following steps were taken to confirm that the previous walkdown conclusions remained valid.

- A walk by was performed to confirm that the equipment material condition and configuration is consistent with the walkdown conclusions and that no new significant interactions related to block walls or piping attached to tanks exist.
- If the ESEL item was screened out based on the previous walkdown, that screening evaluation was reviewed and reconfirmed for the ESEP.

6.3.3 Significant Walkdown Findings

Consistent with the guidance from EPRI NP-6041-SL [40], no significant outliers or anchorage concerns were identified during the PNPS seismic walkdowns. Based on walkdown results, HCLPF capacity evaluations were recommended for the following eight (8) components:

- C7, Containment Isolation and Ventilation Vertical Board
- C2205A, Reactor Protection and NSS Inst. Rack
- C2251A, Jet Pump Instrument Rack A
- D16, 125 VDC Bus A
- D1, 125 VDC Battery Rack A
- D2, 125 VDC Battery Rack B
- D3, 125 VDC Battery Rack B
- EG-23, Vital MG Set

6.4 HCLPF Calculation Process

ESEL items identified for ESEP at PNPS were evaluated using the criteria in EPRI NP-6041-SL [40] and Section 5 of EPRI 3002000704 [2]. Those evaluations included the following steps:

- Performing seismic capability walkdowns for equipment not included in previous seismic walkdowns (SQUG, IPEEE, or NTTF 2.3) to evaluate the equipment installed plant conditions
- Performing screening evaluations using the screening tables in EPRI NP-6041-SL as described in Section 6.2
- 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. A total of seven (7) HCLPF calculations were performed to address the eight (8) components.

- C7, "Containment Isolation and Ventilation Vertical Board"
- C2205A, "Reactor Protection and NSS Inst. Rack"
- C2251A, "Jet Pump Instrument Rack A"
- D16, "125 VDC Bus A"
- D2 and D3, "125 VDC Battery Rack B"
- D1, "125 VDC Battery Rack A"
- EG-23, "Vital MG Set"

6.5 Functional Evaluations of Relays

Five (5) relays 13A-K3, -K5, -K7, K10, and -K22 associated with "RCIC Relay Vertical Board" cabinet C930, and three (3) relays 13A-K31, -K32, and -K33 associated with "Channel B Vertical Board" cabinet C933 were identified as seal in/lockout type needing HCLPF calculation. The relays were of three types General Electric 12HFA151A2F, General Electric 12HGA11A52F and Agastat 7014PB.

The relays were evaluated using the guidance provided in EPRI NP-6041-SL [40] for equipment qualified by testing. Subject relays were determined to have higher HCLPF values than the plant RLGM.

6.6 Tabulated ESEL HCLPF Values (Including Key Failure Modes)

Tabulated ESEL HCLPF values are provided in Attachment B. The following notes apply to the information in the tables.

- For items screened out using EPRI NP-6041-SL [40] screening tables, the HCLPF capacity is provided as >RLGM and the failure mode is listed as "Screened", (unless the controlling HCLPF value is governed by anchorage).
- For items where anchorage controls the HCLPF value, the HCLPF value is listed in the table and the failure mode is noted as "anchorage." For the items where the component function controls the HCLPF value, the HCLPF value is listed in the table and the failure mode is noted as "functional."

After performing the HCLPF calculations, the anchorage was determined to have adequate capacity for the design basis loads and HCLPF greater than RLGM for all components except EG-23. A modification is planned for EG-23 and the HCLPF capacity presented in Attachment B includes the proposed modifications.

7.0 INACCESSIBLE ITEMS

7.1 Identification of ESEL Item Inaccessible for Walkdowns

There are total of 13 Torus Water Temperature Elements (TEs) on the ESEL. Six (6) of the TEs were not walked down since they are located in a high dose area. The evaluation of subject TEs were done by comparison and similarity to the other seven (7) TEs that were walked down. The following is the list of the TEs that were not walked down:

- TE5021-01A
- TE5021-06A

- TE5021-07A
- TE5021-08A
- TE5021-10A
- TE5021-12A

Also, the two (2) valves and two (2) accumulator tanks listed below were not walked down, since they are located in the Dry well (inaccessible area). Subject components were evaluated based on the available photos, drawings, existing SEWS, and vendor information.

- RV-203-3B, Safety Relief Valve
- RV-203-3C, Safety Relief Valve
- T-221B, Accumulator Tank for SRV B
- T-221C, Accumulator Tank for SRV C

In addition two (2) junction boxes J599 and J600 were not walked down since they were not accessible for visual inspection due to their location. These items were assessed and found to be acceptable by comparison and similarity to J601 and J602 respectively, and by reviewing their A-46 SEWS, and existing analysis information.

7.2 Planned Walkdown / Evaluation Schedule / Close Out

There are no components that require follow up seismic walkdowns.

8.0 ESEP CONCLUSIONS AND RESULTS

8.1 Supporting Information

PNPS has performed the ESEP as an interim action in response to the NRC's 50.54(f) letter [1]. It was performed using the methodologies in the NRC endorsed guidance in EPRI 3002000704 [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 PNPS response to the NRC's 50.54(f) letter. On March 12, 2014, NEI submitted to the NRC results of a study [43] 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 [38]. 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 [42] concluded that the "fleet wide 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 PNPS was included in the fleet risk evaluation submitted in the March 12, 2014 NEI letter [43]; therefore, the conclusions in the NRC's May 9 letter also apply to PNPS.

In addition, the March 12, 2014 NEI letter 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.

The fleet of currently operating nuclear power plants was designed using conservative practices, such that the plants have 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 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, etc.)

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.

The intent of the ESEP is to perform an interim action in response to the NRC's 50.54(f) letter to demonstrate seismic margin through a review of a subset of the plant equipment that can be relied upon to protect the reactor core following beyond design basis seismic events. The RLGM used for the ESEP evaluation is a scaled version of the plant's SSE rather than the actual GMRS. To more fully characterize the risk impacts of the seismic ground motion represented by the GMRS on a plant specific basis, a more detailed seismic risk assessment (SPRA or risk-based SMA) is to be performed in accordance with EPRI 1025287 [44] . As identified in the PNPS Seismic Hazard and GMRS submittal [38], PNPS screens in for a risk evaluation. The complete risk evaluation will more completely characterize the probabilistic seismic ground motion input into the plant, the plant response to that probabilistic seismic ground motion input, and the resulting plant risk characterization. PNPS will complete that evaluation in accordance with the schedule identified in NEI's letter dated April 9, 2013 [45] and endorsed by the NRC in their May 7, 2013 letter [46].

8.2 Identification of Planned Modifications

Insights from the ESEP identified the following item where the HCLPF is below the RLGM and plant modifications will be made in accordance with EPRI 3002000704 [2] to enhance the seismic capacity of the plant.

- Vital MG Set EG-23 anchorage had a HCLPF capacity below RLGM. A modification is planned to provide additional seismic margin such that the HCLPF will exceed the RLGM.

8.3 Modification Implementation Schedule

Plant modifications described in Section 8.2 will be performed in accordance with the schedule identified in NEI letter dated April 9, 2013 [45], which states that plant modifications not requiring a planned refueling outage will be completed by December 2016 and modifications requiring a refueling outage will be completed within two planned refueling outages after December 31, 2014.

8.4 Summary of Regulatory Commitments

The following actions will be performed as a result of the ESEP.

Action #	Equipment ID	Equipment Description	Action Description	Completion Date
1	EG-23	Vital MG Set	Modify anchorage such that HCLPF > RLGM	As described in Section 8.3
2	N/A	N/A	Submit a letter to NRC summarizing the HCLPF results of Item 1 confirming implementation of the plant modification associated with item 1	Within 60 days following completion of ESEP activities, including item 1

9.0 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, NRC ADAMS Accession No. ML12053A340.
2. EPRI 3002000704, "Seismic Evaluation Guidance, Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," May 2013.
3. Entergy Letter to U.S. NRC, letter number 2.13.012 "Overall Integrated Plan in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design Basis Events (Order Number EA-12-049)," February 28, 2013, NRC ADAM Accession No. ML13063A063.
4. Entergy Letter to U.S. NRC, letter number 2.14.061 "Pilgrim Nuclear Power Station'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 28, 2014, NRC ADAMS Accession No. ML14253A189.

5. Entergy Drawing M245, Revision 39, "P&ID RCIC System."
6. Entergy Drawing E13, Revision E83, "Single Line Relay & Meter Diagram 125V and 250V DC Systems."
7. Entergy Drawing M246SH1, Revision 32, "P&ID RCIC System."
8. Entergy Drawing M1G13-11, Revision E17, "Elementary Diagram RCIC System, Sheet 3 of 9."
9. Entergy Drawing M1G20-9, Revision E9 "Elementary Diagram RCIC System, Sheet 3 of 8."
10. Entergy Drawing M1G12-12, Revision E14, "Elementary Diagram RCIC System, Sheet 2 of 9."
11. Entergy Drawing M252SH1, Revision 69, "P & ID Nuclear Boiler."
12. Entergy Drawing M1R4-10, Revision 25, "Elementary Diagram Automatic Blowdown System, Sheet 1 of 2."
13. Entergy Drawing M253SH1, Revision 45, "Nuclear Boiler Vessel Instrumentation."
14. Entergy Drawing M253SH2, Revision 29, "Nuclear Boiler Vessel Instrumentation."
15. Entergy Drawing E91, Revision E8, "Wiring Block Diagram RCIC System, Sheet 1 of 2."
16. Entergy Drawing M241SH1, Revision 87, "P&ID Residual Heat Removal System."
17. Entergy Drawing M227SH1, Revision 60, "P&ID Containment Atmospheric Control System."
18. Entergy Drawing E14SH1, Revision 39, "Single Line Diagram 120 V Instrument AC Vital and Reactor Protection AC System & ±24 VDC Power System."
19. Entergy Drawing M1R8-2, Revision 10, "Elementary Diagram Automatic Blowdown System."
20. Entergy Document ELNRC1.2.96.085, BECo Letter 96-085, "Summary Report, Generic Letter 87-02, Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46."
21. Entergy Drawing E727, Revision E7, "Elementary Diagram Emergency Core Cooling System Analog Trip Cabinet C2233A Section A."
22. Entergy Drawing M1P361-2, Revision E3, "Arrangement Diagram Reactor Core Isolation Cooling System Instrument Rack C2258, Sheet 3 of 4."
23. Entergy Correspondence IPEEE, "Pilgrim Nuclear Power Station Individual Plant Examination for External Events (GL 88-20)," dated July 1994.
24. Entergy Drawing M1P302-15, Revision E11, "Arrangement Drawing Control Room Panel C903, Sheet 2 of 2."
25. Entergy Drawing E692, Revision E6, "Elementary Diagram Torus Water Temperature Monitoring System Channel A."
26. Entergy Drawing M206B, Revision E0, "Control Room and Local Panel Instruments."
27. Entergy Drawing E401SH2, Revision E0, "Schematic Diagram Containment Atmospheric Control System."

28. Entergy Drawing M227A32-10, Revision E0, "C170/C171 Post Accident Monitoring System."

29. Entergy Document EQDFDPT1001-604A, Revision 7, "Equipment Qualification Data Sheet DFPT1001-604A."

30. Entergy Drawing M227-154, Revision E8, "Front View Layout for Containment Ventilation Isolation & Gas Treatment – Vertical Board C-7."

31. Entergy Drawing E401SH3, Revision E1, "Schematic Diagram Containment Atmospheric Control System."

32. Entergy Drawing M227C1, Revision E0, "Arrangement Drawing Torus Water Temperature Monitoring Sys. Panel C179."

33. Entergy Drawing M1G17-7 Revision E6, "Elementary Diagram RCIC System, Sheet 7 of 9."

34. Entergy Drawing M1P355-5 Revision 4, "Arrangement Drawing Leak Detection System Instrument Rack C2257."

35. Entergy Document EC45555 Revision 1, "FLEX Alternate Power to 125VDC and 250VDC Battery Chargers (Base EC)."

36. Entergy Document EC42259 Revision 0, "PNPS FLEX Strategy Master EC for Beyond-Design-Basis External Events (BDBEEs) Diverse & Flexible Coping Strategy (FLEX) Implementation."

37. "Pilgrim Nuclear Power Station - Final Safety Analysis Report," Revision 29, Docket No. 50-293, October 2013.

38. Entergy Letter Number 2.14.026, John A. Dent Jr. to NRC, "Entergy's Seismic Hazard and Screening Report (CEUS Sites), Response to NRC 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 March 31, 2014." NRC ADAMS Accession No. ML14092A023.

39. Entergy Document C114ERQE1, Revision E1, "Seismic Response Spectra," October 2005. (Stored in Merlin as C114ERQE0)

40. EPRI-NP-6041-SL, "Methodology for Assessment of Nuclear Power Plant Seismic Margin," Revision 1, August 1991.

41. EPRI TR-103959, "Methodology for Developing Seismic Fragilities," July 1994.

42. 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, NRC ADAMS Accession No. ML14111A147.

43. 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.

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

45. 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, NRC ADAMS Accession No. ML13101A379.

46. NRC (E Leeds) Letter to NEI (J Pollock), "Electric Power Research Institute Final Draft Report xxxxx, "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, NRC ADAMS Accession No. ML13106A331.

47. Entergy Document EC53987, "Expedited Seismic Evaluation Process (ESEP) Report (Submittal to NRC – Fukushima 50.54F Response", the following AREVA documents are captured in the plant document management system:

    a. AREVA Document 51-9212954-002, "ESEP Expedited Seismic Equipment List (ESEL) – Pilgrim Nuclear Power Station."

    b. AREVA Calculation 32-9224839-001, "Pilgrim ESEP HCLPF Calculation - Reactor Protection and NSS Instrument Rack, C2205A."

    c. AREVA Calculation 32-9225449-001, "Pilgrim ESEP HCLPF Calculation - Containment Isolation and Ventilation Vertical Board, C7."

    d. AREVA Calculation 32-9225625-002, "Pilgrim ESEP HCLPF Calculation - 125 VDC Bus A (D16)."

    e. AREVA Calculation 32-9225626-001, "Pilgrim ESEP HCLPF Calculation - Jet Pump A Instrument Rack C2251A."

    f. AREVA Calculation 32-9226829-000, "Pilgrim ESEP HCLPF Calculation – Relays for C930 and C933."

    g. AREVA Calculation 32-9226906-001, "Pilgrim ESEP HCLPF Calculation – Motor-Generator Set, EG-23."

    h. AREVA Calculation 32-9229135-000, "Pilgrim ESEP HCLPF Calculation - Battery Racks D2 and D3."

    i. AREVA Calculation 32-9229524-000, "Pilgrim ESEP HCLPF Calculation – 125 VDC Battery Rack A, D1."

48. Entergy Calculation C15.0.3623, "FLEX Transfer Switch Mounting Evaluation for Battery Chargers D11, D12, D13, D14, D15," Revision 0A.

49. Entergy Calculation C15.0.3624, "FLEX Transfer Switch Mounting Evaluation for 120 VAC Panels Y3, Y31, Y4, Y41, Y13, and Y14," Revision 0.

50. Entergy Calculation C15.0.3631, "Nitrogen Cylinder and Pressure Regulator Supports – Backup Nitrogen Supply to Main Steam SRVs RV-203-3B and RV-203-3C," Revision 0B

ATTACHMENT A – PNPS ESEL

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
1	MO-1301-22	RCIC Pump CST Suction Valve	Open	Closed	Must close to swap RCIC suction to the suppression pool	[5]
2	D754	DC Bus D7 Supply Breaker to MO-1301-22	Energized	Energized	Supply power to MO-1301-22	[6]
3	TBD	FLEX Primary External Water Source Injection Point	Closed	Open	The FLEX Primary External Water Source Injection Point is to the HPCI & RCIC System 18-inch CST common suction line and shall be established at the CST T-105A/B piping vault on the plant-side of the 26-HO-78 & 79 CST 18-inch manual isolation butterfly valves.	[3]
4	MO-1301-25	RCIC Suction from Torus	Closed	Open	Normally closed, must open to supply RCIC suction from torus	[5]
5	D761	DC Bus D7 supply breaker to MO-1301-25	Energized	Energized	Supply power to MO-1301-25	[6]
6	MO-1301-26	RCIC Suction from Torus	Closed	Open	Normally closed, must open to supply RCIC suction from torus	[5]
7	D764	DC Bus D7 supply breaker to MO-1301-26	Energized	Energized	Supply power to MO-1301-26	[6]
8	P-206	RCIC Pump	Idle	Operating	Provides RPV makeup in Phase 1	[7]
9	MO-1301-49	RCIC Discharge Isolation Valve	Closed	Open	Normally closed, must open to supply RCIC injection flow	[5]
10	D774	DC Bus D7 Supply Breaker to MO-1301-49	Energized	Energized	Supply power to MO-1301-49	[6]
11	PCV 1301-43	RCIC Lube Oil Cooling Water Pressure Control Valve	Open	Open/Close as needed	Controls cooling water flow to RCIC lube oil cooler	[7]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
12	MO-1301-62	RCIC Turbine Lube Oil Inlet Valve	Closed	Open	Normally closed, opens to allow cooling water flow to RCIC turbine lube oil cooler	[7]
13	D794	DC Bus D7 Supply Breaker to MO-1301-62	Energized	Energized	Supply power to MO-1301-62	[6]
14	E-201	RCIC Barometric Condenser and Vacuum Tank	Idle	Operating	Collects and condenses RCIC gland seal leakage	[7]
15	E-204	RCIC Lube Oil Cooler	Idle	Operating	Cools RCIC lube oil	[7]
16	P-221	RCIC Vacuum Tank Condensate Pump	Idle	Operating	Pumps condensate from vacuum tank to pump suction	[7]
17	D712	DC Bus D7 Supply Breaker to P-221	Energized	Energized	Supply power to P-221	[6]
18	P-222	RCIC Vacuum Pump	Idle	Operating	Maintains vacuum on barometric condenser	[7]
19	D714	DC Bus D7 Supply Breaker to P-222	Energized	Energized	Supply power to P-222	[6]
20	MO-1301-61	RCIC Turbine Steam Inlet Valve	Closed	Open	Opens on RCIC start to admit steam to RCIC turbine	[7]
21	D751	DC Bus D7 Supply Breaker to MO-1301-61	Energized	Energized	Supply power to MO-1301-61	[6]
22	SV-1301-1	RCIC Turbine Trip Throttle Valve	Open	Open	Closes on protective signals that will be bypassed by procedure	[7]
23	HO-1301-159	RCIC Turbine Governor Valve	Open	Open/Close as needed	Modulates steam flow to RCIC turbine	[7]
24	X-202	RCIC Turbine	Idle	Operating	Supplies motive force to RCIC pump	[7]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
25	FT-1360-4	RCIC Pump Flow Transmitter for Turbine Control	On	On	Transmits RCIC flow to flow control circuit	[5][22]
26	SQRT1340-10	RCIC Flow Square Root Converter	Energized	Energized	RCIC flow integrator for RCIC flow control	[5]
27	FIC-1340-1	RCIC Flow Indicating Controller	Energized	Energized	Controls RCIC flow	[5]
28	DC/AC 1340-16	DC/AC Inverter for RCIC Flow Controller	Energized	Energized	Supplies AC Power to RCIC flow control circuitry	[5][8]
29	C1303	RCIC Local Controls	Energized	Energized	Local control of RCIC turbine	[15]
30	D4-3	Supply Breaker for RCIC in Panels C904 and C939	Closed	Closed	Power to the logic and indications in C904 and C939	[6]
31	13A-K1	RCIC Auto Initiation Logic	Deenergized	Energized	Start RCIC on low-low water level (C930)	[10]
32	13A-K10	RCIC Steam Supply Low Pressure	Deenergized	Deenergized	Could prevent RCIC operation	[10]
33	13A-K11	Turbine Trip Auxiliary Relay	Deenergized	Deenergized	Could prevent RCIC operation	[10]
34	13A-K13	Pump Discharge Low Flow	Deenergized	Deenergized	Could prevent RCIC operation	[10]
35	13A-K14	Pump Suction Low Pressure	Deenergized	Deenergized	Could prevent RCIC operation	[10]
36	13A-K17	Turbine Exhaust High Pressure	Deenergized	Deenergized	Could prevent RCIC operation	[10]
37	13A-K18	MO-1301-25 Position Monitor	Deenergized	Energized	Monitors valve position	[10]
38	13A-K2	RCIC Auto Initiation Logic	Deenergized	Energized	Start RCIC on low-low water level (C930)	[10]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
39	13A-K22	RCIC Auto Isolation Relay	Deenergized	Deenergized	Could prevent RCIC operation	[10]
40	13A-K3	Pump/Turbine Room High Temperature	Deenergized	Deenergized	Could prevent RCIC operation	[10]
41	13A-K31	Pump/Turbine Room High Temperature	Deenergized	Deenergized	Could prevent RCIC operation	[9]
42	13A-K32	RCIC Valve Station High Temperature	Deenergized	Deenergized	Could prevent RCIC operation	[9]
43	13A-K33	RCIC Steam Line High Differential Pressure	Deenergized	Deenergized	Could prevent RCIC operation	[9]
44	13A-K34	RCIC Auto Isolation Relay	Deenergized	Deenergized	Could prevent RCIC operation	[9]
45	13A-K5	RCIC Valve Station High Temperature	Deenergized	Deenergized	Could prevent RCIC operation	[10]
46	13A-K7	RCIC Steam Line High Differential Pressure	Deenergized	Deenergized	Could prevent RCIC operation	[10]
47	RV-203-3B	Safety Relief Valve B	Shut	Open	Used for pressure control and cooldown	[11]
48	RV-203-3C	Safety Relief Valve C	Shut	Open	Used for pressure control and cooldown	[11]
49	SV-203-3B	Solenoid Pilot Valve for SRV B	Deenergized	Energized	Pilot valve for associated SRV	[11][19]
50	SV-203-3C	Solenoid Pilot Valve for SRV C	Deenergized	Energized	Pilot valve for associated SRV	[11][19]
51	T-221B	Accumulator Tank for SRV B	Operable	Operable	Holds operating nitrogen for SRV	[11]
52	T-221C	Accumulator Tank for SRV C	Operable	Operable	Holds operating nitrogen for SRV	[11]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
53	PCV-203-11	Pressure Control Valve for Backup Nitrogen Supply	Idle	In service	Controls the pressure supplied to the SRV accumulators	[11]
54	2E-S10A	ADS Inhibit Switch	Normal	Inhibit	Used to inhibit automatic initiation of Automatic Depressurization System (ADS)	[12]
55	2E-S1B	SRV B Valve Actuation	Auto	Open	Open SRV B for cooldown	[12]
56	2E-S1C	SRV C Valve Actuation	Auto	Open	Open SRV C for cooldown	[12][19]
57	2E-K12A	CSCS Pump Running Interlock	Deenergized	Deenergized	Included for relay chatter impacts	[12][20]
58	2E-K12B	CSCS Pump Running Interlock	Deenergized	Deenergized	Included for relay chatter impacts	[12][20]
59	2E-K13B	SRV Power Supply Control Logic	Energized	Energized	Included for relay chatter impacts	[12][20]
60	2E-K13C	SRV Power Supply Control Logic	Energized	Energized	Included for relay chatter impacts	[12][16][19][20]
61	2E-K7A	Reactor Water Low Low Level Interlock	Deenergized	Energized	Included for relay chatter impacts	[12][20]
62	2E-K7B	Reactor Water Low Low Level Interlock	Deenergized	Energized	Included for relay chatter impacts	[12][20]
63	LI-263-100A	Reactor Water Level Narrow Range	Operable	Operable	Reactor level indication	[13][20]
64	E/I-263-72A	Voltage Current Converter	Operable	Operable	Reactor level indication	[13][20]
65	LI-263-106A	Reactor Water Level Indicator	Operable	Operable	Reactor level indication	[13][20]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
66	E/I-263-73A	Voltage Current Converter	Operable	Operable	Reactor level indication	[13][20]
67	LIS-263-73A	Level Indicating Switch	Operable	Operable	Reactor level indication	[13]
68	LT-263-73A	Level Transmitter	Operable	Operable	Reactor level indication	[13][20]
69	LIS-263-72A	Reactor Water Level Narrow Range	Operable	Operable	Reactor level indication	[13]
70	LT-263-72A	Level Transmitter	Operable	Operable	Reactor level indication	[13][20]
71	PI-640-25A	Reactor Pressure	Operable	Operable	Reactor pressure indication	[13]
72	PT-647A	Reactor Pressure Transmitter	Operable	Operable	Reactor pressure indication	[13][20]
73	PI-263-49A	Reactor Pressure	Operable	Operable	Reactor pressure indication	[13][20]
74	E/I-263-49A	Voltage Current Converter	Operable	Operable	Reactor pressure indication	[13][20]
75	PIS-263-49A	Reactor Pressure	Operable	Operable	Reactor pressure indication	[13]
76	PT-263-49A	Reactor Pressure Transmitter	Operable	Operable	Reactor pressure indication	[13][20]
77	C2205A	Reactor Protection and NSS Instrument Rack	Operable	Operable	Pressure and level transmitter support	[13]
78	C2233A	Emergency Core Cooling System (ECCS) Analog Trip Cabinet	Operable	Operable	Pressure and level transmitter support	[13][15]
79	C2251A	Jet Pump Instrument Rack A	Operable	Operable	Pressure and level transmitter support	[13][14]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
80	C904	RWCU and Recirc. Bench Board	Operable	Operable	Powered from D4, contains RCIC controls	[5][6][15][27]
81	C903	Reactor and Containment Cooling Bench Board	Operable	Operable	Indications provided here	[5][6][12][13][15][16][24][25]
82	C905	Reactor Control Bench Board	Operable	Operable	Indications provided here, powered from D6	[6][13]
83	C129B	Containment Pressure Switch Instrument Rack	Operable	Operable	Indications provided here	[16][21]
84	C2258	RCIC Instrument Rack	Operable	Operable	Instrument rack for RCIC flow transmitter	[15][22]
85	C930	RCIC Relay Vertical Board	Operable	Operable	Contains RCIC logic. IPEEE correlated failure of C930, C932, C933.	[15][23]
86	C932	Channel A Vertical Board	Operable	Operable	Contains RCIC logic. IPEEE correlated failure of C930, C932, C933.	[12][15][23]
87	C933	Channel B Vertical Board	Operable	Operable	Contains RCIC logic. IPEEE correlated failure of C930, C932, C933.	[15][23]
88	C941	Primary Containment Isolation Relay Cabinet	Operable	Operable	Contains isolation logic that could disable RCIC or HPCI operation	[6]
89	PIS-1001-89A	Drywell Pressure Indicating Switch	Operable	Operable	Containment pressure indication	[16][21]
90	PT-1001-89A	Drywell Pressure Transmitter	Operable	Operable	Containment pressure indication	[16][21]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
91	LI-1001-604A	Torus Water level Indicator	Energized	Energized	Wide range torus water level in Phase 2	[16][28]
92	DPT1001 - 604A	Torus Water level Transmitter	Energized	Energized	Wide range torus water level in Phase 2	[16][29]
93	TI-5021-02A	Torus Water Bulk Temperature	Energized	Energized	Torus bulk water temperature for Phase 2, powered from Y31	[16][24][25]
94	TRU-5021-01A	Torus Water Temperature Recorder	Energized	Energized	Processes bulk and local torus water temperatures for Phase 2, powered from Y31	[16][25][26]
95	TE5021-01A	Torus Water Temperature Element	Energized	Energized	Torus water temperature element for Phase 2, powered by Y31	[16][25]
96	TE5021-02A	Torus Water Temperature Element	Energized	Energized	Torus water temperature element for Phase 2, powered by Y31	[16][25]
97	TE5021-03A	Torus Water Temperature Element	Energized	Energized	Torus water temperature element for Phase 2, powered by Y31	[16][25]
98	TE5021-04A	Torus Water Temperature Element	Energized	Energized	Torus water temperature element for Phase 2, powered by Y31	[16][25]
99	TE5021-05A	Torus Water Temperature Element	Energized	Energized	Torus water temperature element for Phase 2, powered by Y31	[16][25]
100	TE5021-06A	Torus Water Temperature Element	Energized	Energized	Torus water temperature element for Phase 2, powered by Y31	[16][25]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
101	TE5021-07A	Torus Water Temperature Element	Energized	Energized	Torus water temperature element for Phase 2, powered by Y31	[16][25]
102	TE5021-08A	Torus Water Temperature Element	Energized	Energized	Torus water temperature element for Phase 2, powered by Y31	[16][25]
103	TE5021-09A	Torus Water Temperature Element	Energized	Energized	Torus water temperature element for Phase 2, powered by Y31	[16][25]
104	TE5021-10A	Torus Water Temperature Element	Energized	Energized	Torus water temperature element for Phase 2, powered by Y31	[16][25]
105	TE5021-11A	Torus Water Temperature Element	Energized	Energized	Torus water temperature element for Phase 2, powered by Y31	[16][25]
106	TE5021-12A	Torus Water Temperature Element	Energized	Energized	Torus water temperature element for Phase 2, powered by Y31	[16][25]
107	TE5021-13A	Torus Water Temperature Element	Energized	Energized	Torus water temperature element for Phase 2, powered by Y31	[16][25]
108	C7	Containment Isolation and Ventilation Vertical Board	Operable	Operable	Included because the controls for the direct torus vent path are located on this panel	[27][30]
109	C170	PAM panel	Operable	Operable	Indications provided here	[14][16][26]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
110	C174	PASS Isolation Valve Control Panel	Operable	Operable	Indications provided here	[6]
111	C179	Torus Water Temperature Signal Processing Cabinet	Operable	Operable	Torus temperature signals processed here	[25][32]
112	AO-5042B	Torus Purge Exhaust Isolation Valve (inboard)	Closed	Open	Inboard valve for hardened containment venting system. Powered from 125 VDC bus D5.	[17]
113	SV-5042B	Torus Purge Exhaust Isolation Solenoid Valve (inboard)	Deenergized	Energized	Solenoid valve for AO-5042B for DTV	[17][31]
114	D5-15	DC Breaker Supply to C7	Energized	Energized	Power supply for solenoid valve for AO-5042B for DTV	[6][31]
115	AO-5025	Direct Torus Vent	Locked Closed	Open	Open to provide flow path for containment heat removal during Phase 2. Powered from 125VDC Bus D4.	[17][27]
116	SV-5025	Direct Torus Vent Solenoid Valve	Deenergized	Energized	Solenoid valve for AO-5025 for DTV	[17][27]
117	D4-15	DC Breaker Supply to C7	Energized	Energized	Power supply for solenoid valve for AO-5025 for DTV	[6][27]
118	N2-401A N2-402A N2-401B N2-402B	Backup Nitrogen Bottles for operation of AO-5042B and AO-5025	TBD	TBD	Local backup nitrogen for direct torus vent valves	[3]
119	D7	125VDC Motor Control Center for RCIC	Energized	Energized	Power for RCIC valves	[6][15]
120	72-166	Supply for D7	Closed	Closed	Power for RCIC valves	[6]
121	D36	Extension of 125 VDC Panel A	Energized	Energized	Power for analog trip system	[6]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
122	D36-8	ECCS Analog Trip Cabinet C2233A Power Breaker	Closed	Closed	Power for analog trip system	[6]
123	D4	125 VDC Distribution Panel A	Energized	Energized	125 DC Bus A for CSCS logic	[6]
124	72-165	Supply for D4	Closed	Closed	Power for D4	[6]
125	72-16A	D16 Internal Breaker	Closed	Closed	Bus continuity	[6]
126	D16	125 VDC Bus A	Energized	Energized	125 DC Power Bus A	[6]
127	72-161	Battery A Output Breaker	Closed	Closed	Supply power to D16	[6]
128	D29	Battery A Current Limiter	Intact	Intact	Protective device	[6]
129	D1	125 VDC Battery Rack A	Float Charge	Discharge	125 VDC A power source	[6]
130	72-162	Battery Charger D11 Supply to D16	Closed	Closed	Connection for portable generator	[3][6]
131	D37	Extension of 125 VDC Panel B	Energized	Energized	Power for analog trip system	[6]
132	D5	125 VDC Distribution Panel B	Energized	Energized	125 DC Bus B for CSCS logic	[6]
133	72-175	Supply Breaker for Bus D5	Closed	Closed	Power for D5	[6]
134	D17	125 VDC Bus B	Energized	Energized	125 VDC power Bus B	[6]
135	72-17A	D17 Internal Breaker	Closed	Closed	Bus continuity	[6]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
136	D2	125 VDC Battery B	Float Charge	Discharge	125 VDC B power source	[6]
137	72-171	Battery B Output Breaker	Closed	Closed	Supply power to D17	[6]
138	D30	Battery B Current Limiter	Intact	Intact	Protective device	[6]
139	72-172	Battery Charger D12 Supply to D17	Closed	Closed	Connection for portable generator	[3][6]
140	Y2	Vital Services Power Supply	Energized	Energized	Vital bus for Indication and control	[18]
141	Y12	Auto Transfer Switch for Y2	MG Set Supply	MG Set Supply	Power from vital motor-generator set to Y2	[18]
142	EG-23	Vital Motor-Generator Set	Operating	Operating	Supply power to Y2 and requires D6	[18]
143	72-1022	D10 Breaker to Vital Motor-Generator	Closed	Closed	Supply power to DC motor - Vital motor-generator set	[6][18]
144	D10	250 VDC Power Bus	Energized	Energized	250 VDC power bus	[6][18]
145	72-1013	250 VDC Battery D3 Output Breaker	Closed	Closed	Supply power to D10	[6]
146	D31	250 VDC Battery D3 Current Limiter	Intact	Intact	Protective Device	[6]
147	D3	250 VDC Battery	Float Charge	Discharge	250 VDC power Source	[6]
148	72-1014	Battery Charger D13 Supply to D10	Closed	Closed	Connection for portable generator	[6]
149	D13	250 VDC Battery Charger	Energized	Energized from mobile generator	Connection for portable generator	[6]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
150	D6	125VDC Distribution Panel C	Energized	Energized	Vital instruments and controls	[6]
151	Y10	125VDC Control Power Transfer	Operable	Operable	D6 normally supplied from D16	[6]
152	D32	D16 Control Logic Y10 Switching	Closed	Closed	D6 normally supplied from D16	[6]
153	D33	D17 Control Logic Y10 Switching	Open	Open	D6 normally supplied from D16	[6]
154	Y3	Safeguard 120VAC "A" Control Power Supply Panel	Energized	Energized	Power supply for I&C	[18]
155	Y31	Safeguard 120VAC "A" H_2O_2 Control Power Supply Panel	Energized	Energized	Power supply for I&C	[18]
156	D11	125 VDC Battery Charger A	Energized	Energized from mobile generator	Connection for portable generator	[6]
157	D12	125 VDC Battery Charger B	Energized	Energized from mobile generator	Connection for portable generator	[6]
158	C2257B	Instrument Rack 2257B	Operable	Operable		[5][34]
159	dPIS-1360-1A	dP Switch for RCIC Steam Isolation	Operable	Operable		[5][34]
160	dPIS-1360-1B	dP Switch for RCIC Steam Isolation	Operable	Operable		[5][34]
161	PS-1360-9A	Pressure Switch for RCIC Steam Isolation	Operable	Operable		[5][34]
162	PS-1360-9B	Pressure Switch for RCIC Steam Isolation	Operable	Operable		[5][34]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
163	PS-1360-9C	Pressure Switch for RCIC Steam Isolation	Operable	Operable		[5][34]
164	PS-1360-9D	Pressure Switch for RCIC Steam Isolation	Operable	Operable		[5][34]
165	J315	Junction Box	Operable	Operable		[10]
166	J317	Junction Box	Operable	Operable		[9]
167	J602	Junction Box	Operable	Operable		[9]
168	J599	Junction Box	Operable	Operable		[10]
169	J600	Junction Box	Operable	Operable		[10]
170	J601	Junction Box	Operable	Operable		[9]
171	TS-1360-14C	Temperature Switch for RCIC Isolation	Operable	Operable		[10]
172	TS-1360-15A	Temperature Switch for RCIC Isolation	Operable	Operable		[10]
173	TS-1360-15C	Temperature Switch for RCIC Isolation	Operable	Operable		[10]
174	TS-1360-16C	Temperature Switch for RCIC Isolation	Operable	Operable		[10]
175	TS-1360-16D	Temperature Switch for RCIC Isolation	Operable	Operable		[9]
176	TS-1360-17A	Temperature Switch for RCIC Isolation	Operable	Operable		[10]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
177	TS-1360-17B	Temperature Switch for RCIC Isolation	Operable	Operable		[9]
178	TS-1360-17C	Temperature Switch for RCIC Isolation	Operable	Operable		[10]
179	TS-1360-17D	Temperature Switch for RCIC Isolation	Operable	Operable		[9]
180	TS-1360-15B	Temperature Switch for RCIC Isolation	Operable	Operable		[9]
181	TS-1360-14D	Temperature Switch for RCIC Isolation	Operable	Operable		[9]
182	TS-1360-15D	Temperature Switch for RCIC Isolation	Operable	Operable		[9]
183	D1513	FLEX AC Power Transfer Switch to Repower D11	Operable	Operable		[3][6][35][36]
184	D1413	FLEX AC Power Transfer Switch to Repower D12	Operable	Operable		[3][35][36]
185	D1414B	FLEX AC Power Transfer Switch to Repower D13	Operable	Operable		[3][6][35][36]
186	N17115*	FLEX AC Power Transfer Switch to Repower Y3	Operable	Operable		[3][36]
187	N17115*	FLEX AC Power Transfer Switch to Repower Y31 *Note that a single switch, N17115, repowers Y3 and Y31	Operable	Operable		[3][36]

ATTACHMENT B – ESEP HCLPF VALUES AND FAILURE MODES TABULATION

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
1	MO-1301-22	RCIC Pump CST Suction Valve	>RLGM	Screened	
2	D754	DC Bus D7 Supply Breaker to MO-1301-22	>RLGM	Screened	
3	TBD	FLEX Primary External Water Source Injection Point	>RLGM	Screened	Note 5
4	MO-1301-25	RCIC Suction from Torus	>RLGM	Screened	
5	D761	DC Bus D7 supply breaker to MO-1301-25	>RLGM	Screened	
6	MO-1301-26	RCIC Suction from Torus	>RLGM	Screened	
7	D764	DC Bus D7 supply breaker to MO-1301-26	>RLGM	Screened	
8	P-206	RCIC Pump	>RLGM	Screened	Note 1
9	MO-1301-49	RCIC Discharge Isolation Valve	>RLGM	Screened	
10	D774	DC Bus D7 Supply Breaker to MO-1301-49	>RLGM	Screened	
11	PCV 1301-43	RCIC Lube Oil Cooling Water Pressure Control Valve	>RLGM	Screened	
12	MO-1301-62	RCIC Turbine Lube Oil Inlet Valve	>RLGM	Screened	
13	D794	DC Bus D7 Supply Breaker to MO-1301-62	>RLGM	Screened	
14	E-201	RCIC Barometric Condenser and Vacuum Tank	>RLGM	Screened	Note 1
15	E-204	RCIC Lube Oil Cooler	>RLGM	Screened	Note 1
16	P-221	RCIC Vacuum Tank Condensate Pump	>RLGM	Screened	Note 1
17	D712	DC Bus D7 Supply Breaker to P-221	>RLGM	Screened	
18	P-222	RCIC Vacuum Pump	>RLGM	Screened	Note 1
19	D714	DC Bus D7 Supply Breaker to P-222	>RLGM	Screened	
20	MO-1301-61	RCIC Turbine Steam Inlet Valve	>RLGM	Screened	

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
21	D751	DC Bus D7 Supply Breaker to MO-1301-61	>RLGM	Screened	
22	SV-1301-1	RCIC Turbine Trip Throttle Valve	>RLGM	Screened	
23	HO-1301-159	RCIC Turbine Governor Valve	>RLGM	Screened	
24	X-202	RCIC Turbine	>RLGM	Screened	Note 1
25	FT-1360-4	RCIC Pump Flow Transmitter for Turbine Control	>RLGM	Screened	
26	SQRT1340-10	RCIC Flow Square Root Converter	>RLGM	Screened	
27	FIC-1340-1	RCIC Flow Indicating Controller	>RLGM	Screened	
28	DC/AC 1340-16	DC/AC Inverter for RCIC Flow Controller	>RLGM	Screened	
29	C1303	RCIC Local Controls	>RLGM	Screened	Note 2
30	D4-3	Supply Breaker for RCIC in Panels C904 and C939	>RLGM	Screened	
31	13A-K1	RCIC Auto Initiation Logic	>RLGM	Screened	
32	13A-K10	RCIC Steam Supply Low Pressure	0.33	Relay Function	
33	13A-K11	Turbine Trip Auxiliary Relay	>RLGM	Screened	
34	13A-K13	Pump Discharge Low Flow	>RLGM	Screened	
35	13A-K14	Pump Suction Low Pressure	>RLGM	Screened	
36	13A-K17	Turbine Exhaust High Pressure	>RLGM	Screened	
37	13A-K18	MO-1301-25 Position Monitor	>RLGM	Screened	
38	13A-K2	RCIC Auto Initiation Logic	>RLGM	Screened	
39	13A-K22	RCIC Auto Isolation Relay	0.33	Relay Function	
40	13A-K3	Pump/Turbine Room High Temperature	0.39	Relay Function	

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
41	13A-K31	Pump/Turbine Room High Temperature	0.39	Relay Function	
42	13A-K32	RCIC Valve Station High Temperature	0.39	Relay Function	
43	13A-K33	RCIC Steam Line High Differential Pressure	0.44	Relay Function	
44	13A-K34	RCIC Auto Isolation Relay	>RLGM	Screened	
45	13A-K5	RCIC Valve Station High Temperature	0.39	Relay Function	
46	13A-K7	RCIC Steam Line High Differential Pressure	0.44	Relay Function	
47	RV-203-3B	Safety Relief Valve B	>RLGM	Screened	
48	RV-203-3C	Safety Relief Valve C	>RLGM	Screened	
49	SV-203-3B	Solenoid Pilot Valve for SRV B	>RLGM	Screened	
50	SV-203-3C	Solenoid Pilot Valve for SRV C	>RLGM	Screened	
51	T-221B	Accumulator Tank for SRV B	>RLGM	Screened	Note 2
52	T-221C	Accumulator Tank for SRV C	>RLGM	Screened	Note 2
53	PCV-203-11	Pressure Control Valve for Backup Nitrogen Supply	>RLGM	Screened	
54	2E-S10A	ADS Inhibit Switch	>RLGM	Screened	
55	2E-S1B	SRV B Valve Actuation	>RLGM	Screened	
56	2E-S1C	SRV C Valve Actuation	>RLGM	Screened	
57	2E-K12A	CSCS Pump Running Interlock	>RLGM	Screened	
58	2E-K12B	CSCS Pump Running Interlock	>RLGM	Screened	
59	2E-K13B	SRV Power Supply Control Logic	>RLGM	Screened	
60	2E-K13C	SRV Power Supply Control Logic	>RLGM	Screened	

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
61	2E-K7A	Reactor Water Low Low Level Interlock	>RLGM	Screened	
62	2E-K7B	Reactor Water Low Low Level Interlock	>RLGM	Screened	
63	LI-263-100A	Reactor Water Level Narrow Range	>RLGM	Screened	
64	E/I-263-72A	Voltage Current Converter	>RLGM	Screened	
65	LI-263-106A	Reactor Water Level Indicator	>RLGM	Screened	
66	E/I-263-73A	Voltage Current Converter	>RLGM	Screened	
67	LIS-263-73A	Level Indicating Switch	>RLGM	Screened	
68	LT-263-73A	Level Transmitter	>RLGM	Screened	
69	LIS-263-72A	Reactor Water Level Narrow Range	>RLGM	Screened	
70	LT-263-72A	Level Transmitter	>RLGM	Screened	
71	PI-640-25A	Reactor Pressure	>RLGM	Screened	
72	PT-647A	Reactor Pressure Transmitter	>RLGM	Screened	
73	PI-263-49A	Reactor Pressure	>RLGM	Screened	
74	E/I-263-49A	Voltage Current Converter	>RLGM	Screened	
75	PIS-263-49A	Reactor Pressure	>RLGM	Screened	
76	PT-263-49A	Reactor Pressure Transmitter	>RLGM	Screened	
77	C2205A	Reactor Protection and NSS Instrument Rack	0.38	Functional	
78	C2233A	Emergency Core Cooling System (ECCS) Analog Trip Cabinet	>RLGM	Screened	Note 1
79	C2251A	Jet Pump Instrument Rack A	0.44	Anchorage	
80	C904	RWCU and Recirc. Bench Board	>RLGM	Screened	Note 1

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
81	C903	Reactor and Containment Cooling Bench Board	>RLGM	Screened	Note 1
82	C905	Reactor Control Bench Board	>RLGM	Screened	Note 1
83	C129B	Containment Pressure Switch Instrument Rack	>RLGM	Screened	Note 1
84	C2258	RCIC Instrument Rack	>RLGM	Screened	Note 1
85	C930	RCIC Relay Vertical Board	0.33	Relay Function	Note 1
86	C932	Channel A Vertical Board	>RLGM	Screened	Note 1
87	C933	Channel B Vertical Board	0.39	Relay Function	Note 1
88	C941	Primary Containment Isolation Relay Cabinet	>RLGM	Screened	Note 1
89	PIS-1001-89A	Drywell Pressure Indicating Switch	>RLGM	Screened	
90	PT-1001-89A	Drywell Pressure Transmitter	>RLGM	Screened	
91	LI-1001-604A	Torus Water level Indicator	>RLGM	Screened	
92	DPT1001 -604A	Torus Water level Transmitter	>RLGM	Screened	Note 2
93	TI-5021-02A	Torus Water Bulk Temperature	>RLGM	Screened	
94	TRU-5021-01A	Torus Water Temperature Recorder	>RLGM	Screened	
95	TE5021-01A	Torus Water Temperature Element	>RLGM	Screened	
96	TE5021-02A	Torus Water Temperature Element	>RLGM	Screened	
97	TE5021-03A	Torus Water Temperature Element	>RLGM	Screened	
98	TE5021-04A	Torus Water Temperature Element	>RLGM	Screened	
99	TE5021-05A	Torus Water Temperature Element	>RLGM	Screened	
100	TE5021-06A	Torus Water Temperature Element	>RLGM	Screened	

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
101	TE5021-07A	Torus Water Temperature Element	>RLGM	Screened	
102	TE5021-08A	Torus Water Temperature Element	>RLGM	Screened	
103	TE5021-09A	Torus Water Temperature Element	>RLGM	Screened	
104	TE5021-10A	Torus Water Temperature Element	>RLGM	Screened	
105	TE5021-11A	Torus Water Temperature Element	>RLGM	Screened	
106	TE5021-12A	Torus Water Temperature Element	>RLGM	Screened	
107	TE5021-13A	Torus Water Temperature Element	>RLGM	Screened	
108	C7	Containment Isolation and Ventilation Vertical Board	0.31	Anchorage	
109	C170	PAM panel	>RLGM	Screened	Note 1
110	C174	PASS Isolation Valve Control Panel	>RLGM	Screened	Note 1
111	C179	Torus Water Temperature Signal Processing Cabinet	>RLGM	Screened	Note 1
112	AO-5042B	Torus Purge Exhaust Isolation Valve (inboard)	>RLGM	Screened	
113	SV-5042B	Torus Purge Exhaust Isolation Solenoid Valve (inboard)	>RLGM	Screened	
114	D5-15	DC Breaker Supply to C7	>RLGM	Screened	
115	AO-5025	Direct Torus Vent	>RLGM	Screened	
116	SV-5025	Direct Torus Vent Solenoid Valve	>RLGM	Screened	
117	D4-15	DC Breaker Supply to C7	>RLGM	Screened	
118	N2-401A N2-402A N2-401B N2-402B	Backup Nitrogen Bottles for operation of AO-5042B and AO-5025	>RLGM	Screened	Note 6
119	D7	125VDC Motor Control Center for RCIC	>RLGM	Screened	Note 1
120	72-166	Supply for D7	>RLGM	Screened	

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
121	D36	Extension of 125 VDC Panel A	>RLGM	Screened	Note 2
122	D36-8	ECCS Analog Trip Cabinet C2233A Power Breaker	>RLGM	Screened	
123	D4	125 VDC Distribution Panel A	>RLGM	Screened	Note 2
124	72-165	Supply for D4	>RLGM	Screened	
125	72-16A	D16 Internal Breaker	>RLGM	Screened	
126	D16	125 VDC Bus A	0.37	Anchorage	
127	72-161	Battery A Output Breaker	>RLGM	Screened	
128	D29	Battery A Current Limiter	>RLGM	Screened	Note 2
129	D1	125 VDC Battery Rack A	0.31	Anchorage	
130	72-162	Battery Charger D11 Supply to D16	>RLGM	Screened	
131	D37	Extension of 125 VDC Panel B	>RLGM	Screened	Note 2
132	D5	125 VDC Distribution Panel B	>RLGM	Screened	Note 2
133	72-175	Supply Breaker for Bus D5	>RLGM	Screened	
134	D17	125 VDC Bus B	>RLGM	Screened	Note 1
135	72-17A	D17 Internal Breaker	>RLGM	Screened	
136	D2	125 VDC Battery B	0.30	Anchorage	
137	72-171	Battery B Output Breaker	>RLGM	Screened	
138	D30	Battery B Current Limiter	>RLGM	Screened	Note 2
139	72-172	Battery Charger D12 Supply to D17	>RLGM	Screened	
140	Y2	Vital Services Power Supply	>RLGM	Screened	Note 2
141	Y12	Auto Transfer Switch for Y2	>RLGM	Screened	Note 1

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
142	EG-23	Vital Motor-Generator Set	0.32	Anchorage	HCLPF calculated with modifications to anchorage.
143	72-1022	D10 Breaker to Vital Motor-Generator	>RLGM	Screened	
144	D10	250 VDC Power Bus	>RLGM	Screened	Note 1
145	72-1013	250 VDC Battery D3 Output Breaker	>RLGM	Screened	
146	D31	250 VDC Battery D3 Current Limiter	>RLGM	Screened	Note 2
147	D3	250 VDC Battery	0.40	Anchorage	
148	72-1014	Battery Charger D13 Supply to D10	>RLGM	Screened	
149	D13	250 VDC Battery Charger	>RLGM	Screened	Note 1
150	D6	125VDC Distribution Panel C	>RLGM	Screened	Note 2
151	Y10	125VDC Control Power Transfer	>RLGM	Screened	Note 1
152	D32	D16 Control Logic Y10 Switching	>RLGM	Screened	Note 2
153	D33	D17 Control Logic Y10 Switching	>RLGM	Screened	Note 2
154	Y3	Safeguard 120VAC "A" Control Power Supply Panel	>RLGM	Screened	Note 1
155	Y31	Safeguard 120VAC "A" H2O2 Control Power Supply Panel	>RLGM	Screened	Note 2
156	D11	125 VDC Battery Charger A	>RLGM	Screened	Note 1
157	D12	125 VDC Battery Charger B	>RLGM	Screened	Note 1
158	C2257B	Instrument Rack 2257B	>RLGM	Screened	Note 1
159	dPIS-1360-1A	dP Switch for RCIC Steam Isolation	>RLGM	Screened	
160	dPIS-1360-1B	dP Switch for RCIC Steam Isolation	>RLGM	Screened	
161	PS-1360-9A	Pressure Switch for RCIC Steam Isolation	>RLGM	Screened	

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
162	PS-1360-9B	Pressure Switch for RCIC Steam Isolation	>RLGM	Screened	
163	PS-1360-9C	Pressure Switch for RCIC Steam Isolation	>RLGM	Screened	
164	PS-1360-9D	Pressure Switch for RCIC Steam Isolation	>RLGM	Screened	
165	J315	Junction Box	>RLGM	Screened	Note 2
166	J317	Junction Box	>RLGM	Screened	Note 2
167	J602	Junction Box	>RLGM	Screened	Note 1
168	J599	Junction Box	>RLGM	Screened	Note 2
169	J600	Junction Box	>RLGM	Screened	Note 1
170	J601	Junction Box	>RLGM	Screened	Note 2
171	TS-1360-14C	Temperature Switch for RCIC Isolation	>RLGM	Screened	
172	TS-1360-15A	Temperature Switch for RCIC Isolation	>RLGM	Screened	
173	TS-1360-15C	Temperature Switch for RCIC Isolation	>RLGM	Screened	
174	TS-1360-16C	Temperature Switch for RCIC Isolation	>RLGM	Screened	
175	TS-1360-16D	Temperature Switch for RCIC Isolation	>RLGM	Screened	
176	TS-1360-17A	Temperature Switch for RCIC Isolation	>RLGM	Screened	
177	TS-1360-17B	Temperature Switch for RCIC Isolation	>RLGM	Screened	
178	TS-1360-17C	Temperature Switch for RCIC Isolation	>RLGM	Screened	
179	TS-1360-17D	Temperature Switch for RCIC Isolation	>RLGM	Screened	
180	TS-1360-15B	Temperature Switch for RCIC Isolation	>RLGM	Screened	
181	TS-1360-14D	Temperature Switch for RCIC Isolation	>RLGM	Screened	

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
182	TS-1360-15D	Temperature Switch for RCIC Isolation	>RLGM	Screened	
183	D1513	FLEX AC Power Transfer Switch to Repower D11	>RLGM	Screened	Note 3
184	D1413	FLEX AC Power Transfer Switch to Repower D12	>RLGM	Screened	Note 3
185	D1414B	FLEX AC Power Transfer Switch to Repower D13	>RLGM	Screened	Note 3
186	N17115*	FLEX AC Power Transfer Switch to Repower Y3	>RLGM	Screened	Note 4
187	N17115*	FLEX AC Power Transfer Switch to Repower Y31 *Note that a single switch, N17115, repowers Y3 and Y31	>RLGM	Screened	Note 4

Notes:
1. Anchorage screened out based on available margin during walkdown by SRT.
2. Anchorage screened out during walkdown validation by SRT.
3. This component is evaluated in Entergy Calculation C15.0.3623 [48] for 2xSSE.
4. This component is evaluated in Entergy Calculation C15.0.3624 [49] for 2xSSE.
5. Entergy document no. EC-0000042259 [36] is reviewed. Design input 59 within this document states that seismic demand is based on 2xSSE. Thus, this component, which is yet to be installed, is screened for ESEP based on seismic design criteria.
6. This component is evaluated in Entergy Calculation C15.0.3631 [50] for 2xSSE.

ATTACHMENT 2 to

PNPS Letter 2.14.082

LIST OF REGULATORY COMMITMENTS

FOR

PILGRIM NUCLEAR POWER STATION

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check One)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Modify Vital MG Set EG-23 anchorage. Vital MG Set EG-23 anchorage had a High Confidence of a Low Probability of Failure (HCLPF) capacity below the Review Level Ground Motion (RLGM). A modification is planned to provide additional seismic margin such that the HCLPF will exceed RLGM. NRC Commitment No. A16866	[✓]		*On a schedule specified in Section 8.4 of the Expedited Seismic Evaluation Process Report May 31, 2017
Submit a letter to NRC summarizing the HCLPF results of item 1 above confirming implementation of the plant modification associated with item 1. NRC Commitment No. A16867	[✓]		Within 60 days following completion of ESEP activities, including item 1 above. July 30, 2017

*Plant modifications not requiring a planned refueling outage will be completed by December 2016 and modifications requiring a refueling outage will be completed within two planned refueling outages after December 31, 2014. The modification to the Vital MG-Set EG-23 will be done during a planned refueling outage.