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JAFP-14-0143
December 30, 2014

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: Entergy's Expedited Seismic Evaluation Process Report (CEUS Sites), Response 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

James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
License No. DPR-059

Reference:

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, ML12053A340, March 12, 2012
2. NEI letter, Proposed Path Forward for NTTF Recommendation 2.1: Seismic Reevaluations, ML13101A345, dated April 9, 2013
3. NRC Letter to NEI, Electric Power Research Institute Final Draft Report XXXXXX, "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, ML13106A331, dated May 7, 2013
4. Electric Power Research Institute Final Report 3002000704, "Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," dated May 2013
5. Entergy letter, Entergy's 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, JAFP-13-0056, dated April 29, 2013

Dear Sir or Madam:

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued a 50.54(f) letter 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 and Screening Report within 1.5 years from the date of Reference 1.

In Reference 2, the Nuclear Energy Institute (NEI) requested NRC agreement to delay submittal of the final CEUS Seismic Hazard Evaluation 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 Enclosure provides the Expedited Seismic Evaluation Process Report for JAF information described in Section 7 of Reference 4 in accordance with the schedule identified in Reference 2.

This letter contains new regulatory commitments in the Attachment. If you have any questions regarding this report, please contact Chris M. Adner, Regulatory Assurance Manager, at 315-349-6766.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 30th day of December, 2014.

Sincerely,

Brian R. Sullivan
Site Vice President

BRS/CMA/mh

Attachment: Regulatory Commitments
Enclosure: Expedited Seismic Evaluation Process (ESEP) Report for James A. FitzPatrick Nuclear Power Plant (JAF)

cc: NRC Regional Administrator
NRC Resident Inspector
Mr. Douglas Pickett, Senior Project Manager
Ms. Bridget Frymire, NYSPSC
Mr. John B. Rhodes, President NYSERDA

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Attachment

Regulatory Commitments

(1 Page)

Regulatory Commitments

This table identifies actions discussed in this letter for which Entergy commits to perform. Any other actions discussed in this submittal are described for the NRC's information and are **not** commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Entergy will perform seismic walkdowns at JAF for inaccessible items listed in Section 7.1	X		No later than the end of the first planned JAF refueling outage after December 31, 2014.
Entergy will generate HCLPF calculations for inaccessible items listed in Section 7.1	X		No later than 90 days following the end of the first planned JAF refueling outage after December 31, 2014.
Entergy will implement any necessary JAF modifications for inaccessible items listed in Section 7.1 based on the schedule commitment to complete this activity in JAFP-13-0056 dated April 29, 2013	X		No later than the end of the second planned JAF refueling outage after December 31, 2014.per JAFP-13-0056

JAFP-14-0143

Enclosure

Expedited Seismic Evaluation Process (ESEP) Report for James A. FitzPatrick Nuclear Power Plant (JAF)

(46 Page)

EXPEDITED SEISMIC EVALUATION PROCESS (ESEP) REPORT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (JAF)

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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 James A. FitzPatrick. 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 James A. FitzPatrick FLEX strategies for Reactor Core Cooling and Heat Removal, Reactor Inventory Control/Long Term Subcriticality, and Containment Function are summarized below. This summary is derived from the James A. FitzPatrick Overall Integrated Plan (OIP) in Response to the March 12, 2012, Commission Order EA-12-049 [3] and is consistent with the third six-month Status Report [4].

Core cooling and inventory control are achieved during the first five (5) hours using the Reactor Core Isolation Cooling (RCIC) system initially aligned to take suction from the Condensate Storage Tank (CST), with the suction swapped to the torus when the operators determine the event is a Beyond Design Basis External Event (BDBEE). Pressure control and heat removal are accomplished by Safety Relief Valves (SRVs) venting to the torus. At approximately five (5) hours, suction will be swapped back to the CST for torus temperature control and a controlled depressurization is commenced using RCIC and cycling the SRVs.

At about ten (10) hours (beginning of Phase 2), the operators will need to connect and run a portable 200 kW FLEX diesel generator to the Class 1E 600 VAC electrical buses to re-power the battery chargers to maintain DC control power.

At 23 hours, the torus will be vented via the hardened containment vent to maintain containment parameters within acceptable limits and within the limits that support continued use of the RCIC system.

The torus and CST will enable the RCIC to provide make-up for at least 35 hours without replacement. Prior to depletion of the CST, James A. FitzPatrick will establish the flow path from the seismically-qualified diesel-driven fire pump 76P-1 to provide makeup directly to the reactor pressure vessel.

For Phase 3, the reactor core cooling strategy is to place one loop of RHR into the shutdown cooling mode. This will be accomplished by powering an RHR pump from either Class 1E emergency bus 10500 or Class 1E emergency bus 10600 utilizing the 4160 VAC FLEX portable diesel generator. A modification will be implemented to provide a cross-connection between the fire protection system and one train of the RHR service water system. The seismically qualified diesel-driven fire pump (76P-1) will be used to provide lake water to RHR service water side of the appropriate RHR heat exchanger.

Necessary electrical components are outlined in the James A. FitzPatrick FLEX OIP submittal, and primarily entail 125 VDC power buses, motor control centers, vital batteries, battery chargers, 600 VAC buses, and 4160 VAC buses.

The FLEX strategy credits the monitoring of plant parameters, either from the control room, using available electric power supplied from the batteries or taken locally. If instrumentation is to be monitored from the control room, it will be powered from 125 VDC either directly or through (future) inverters for 120 VAC to some instruments.

Figures 1 and 2 in the James A. FitzPatrick FLEX OIP submittal provide the FLEX flow paths for James A. FitzPatrick Phases 1, 2 and 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 James A. FitzPatrick 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], [36], [37], and [38].

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 BDBEE, as outlined in the James A. FitzPatrick OIP in Response to the March 12, 2012, Commission Order EA-12-049 [3] and is consistent with the third six-month Status Report [4]. The OIP provides the James A. FitzPatrick 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 James A. FitzPatrick 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 3-2 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 James A. FitzPatrick OIP.
2. The scope of components is limited to installed plant equipment, and FLEX connections necessary to implement the James A. FitzPatrick 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 James A. FitzPatrick 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 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 strategy 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 drawings, system descriptions, design basis documents, as necessary.

Cabinets containing electrical and instrumentation that could be affected by 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 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 and 2 ESEL. For Phase 2 and Phase 3, the RHR is used to provide the pathway for reactor pressure vessel injection utilizing the seismic grade fire pump or portable injection pumps.

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)." To address this concern, the following guidance is applied in the James A. FitzPatrick ESEL for functional failure modes associated with power operated valves:

- Power operated valves that remain energized during the Extended Loss of 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 James A. FitzPatrick 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 Phases 1 and 2 and is presented as the single success path in the James A. FitzPatrick ESEL. RHR, Train A, is the primary system for Phase 3. 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 depth 12 ft, which is the top of the Oswego sandstone where all plant structures are founded [39]. Table 4-1 shows the GMRS acceleration for a range of spectral frequencies [40]. The GMRS at the control point is shown in Figure 4-1.

Table 4-1: GMRS for James A. FitzPatrick

Frequency (Hz)	GMRS (g)
100	1.20E-01
90	1.20E-01
80	1.22E-01
70	1.25E-01
60	1.33E-01
50	1.55E-01
40	1.87E-01
35	1.99E-01
30	2.09E-01
25	2.16E-01
20	2.31E-01
15	2.41E-01

Table 4-1: GMRS for James A. FitzPatrick (continued)

Frequency (Hz)	GMRS (g)
12.5	2.39E-01
10	2.33E-01
9	2.26E-01
8	2.15E-01
7	2.04E-01
6	1.88E-01
5	1.70E-01
4	1.44E-01
3.5	1.27E-01
3	1.14E-01
2.5	9.44E-02
2	8.47E-02
1.5	7.50E-02
1.25	7.13E-02
1	6.38E-02
0.9	6.05E-02
0.8	5.66E-02
0.7	5.11E-02
0.6	4.47E-02
0.5	3.76E-02
0.4	3.00E-02
0.35	2.63E-02
0.3	2.25E-02
0.25	1.88E-02
0.2	1.50E-02
0.15	1.13E-02
0.125	9.39E-03
0.1	7.51E-03

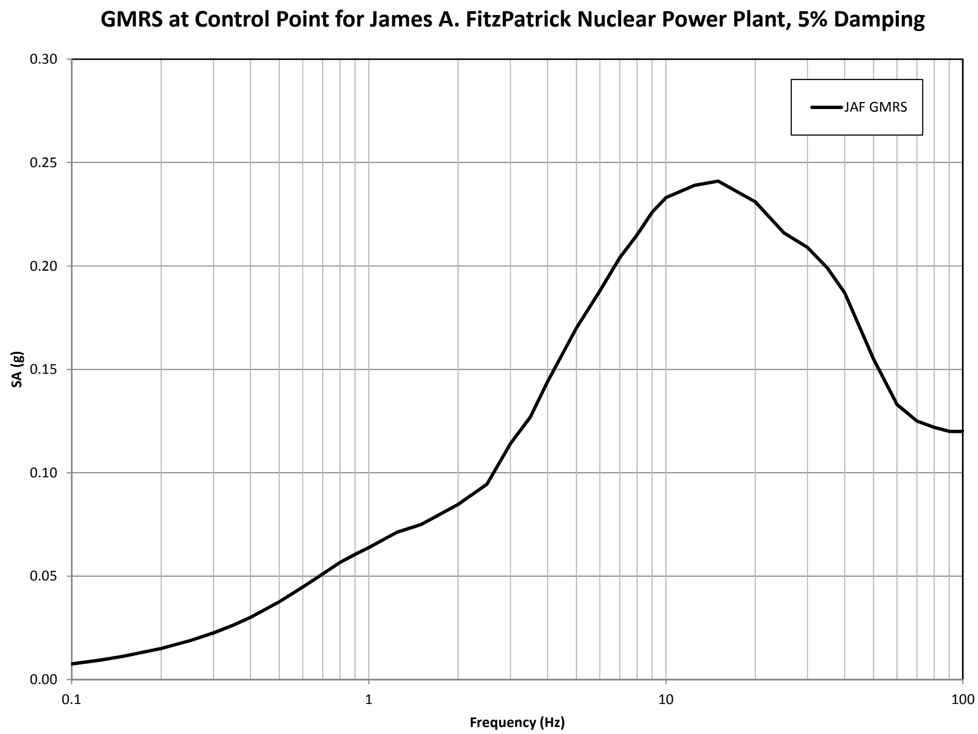

Figure 4-1: GMRS for James A. FitzPatrick

4.2 Comparison to SSE

The SSE corresponds to a horizontal acceleration of 0.15g. 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 [40] [41]. Table 4-2 shows the spectral acceleration values at selected frequencies for the 5% damped horizontal SSE.

Table 4-2: SSE for James A. FitzPatrick

Frequency (Hz)	Spectral Acceleration (g)
100	0.15
25	0.15
10	0.15
5	0.21
2.5	0.22
1	0.13
0.5	0.064

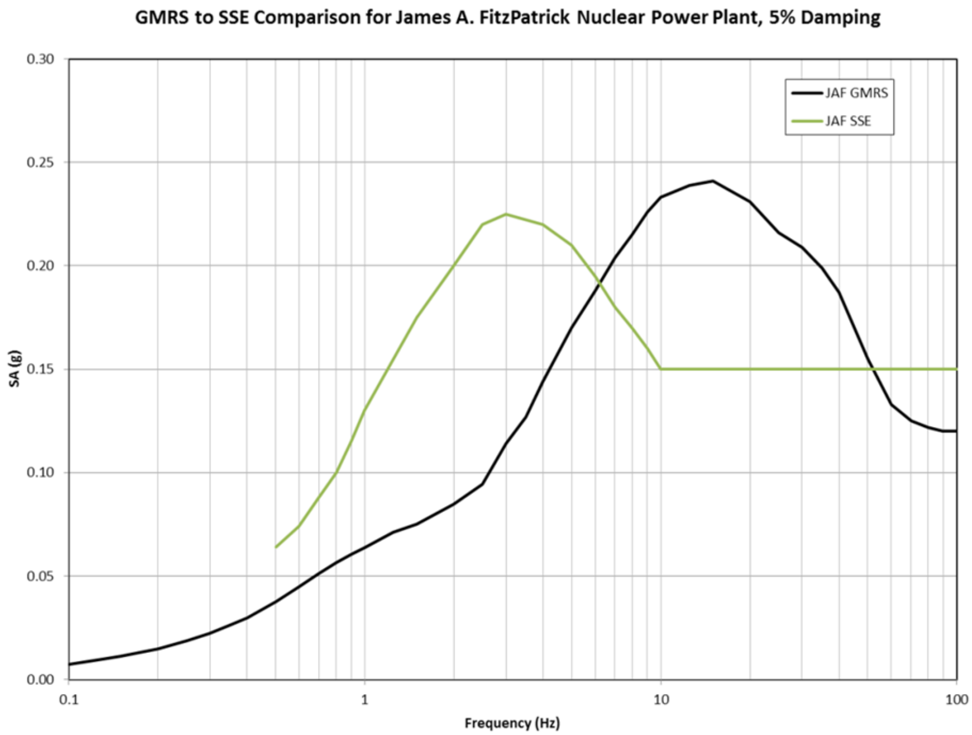

Figure 4-2: GMRS to SSE Comparison for James A. FitzPatrick

The SSE envelops the GMRS for lower frequencies up to nearly 6 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 James A. FitzPatrick and hence High Confidence of a Low Probability of Failure (HCLPF) evaluations are to be 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.

The maximum GMRS/SSE ratio between 1 and 10 Hz range occurs at 10 Hz where the ratio is 0.233/0.15 = 1.55. The GMRS/SSE ratio is set to the scaling factor value of 1.55 for James A. FitzPatrick 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 is generated by plotting the digitized data on a log/linear graph paper, and connecting the points with straight lines.

Table 5-1: RLGM for James A. FitzPatrick

Frequency (Hz)	RLGM at 5% Damping (g)
0.50	0.099
1.00	0.202
2.50	0.342
5.00	0.326
10.00	0.233
25.00	0.233
100.00	0.233

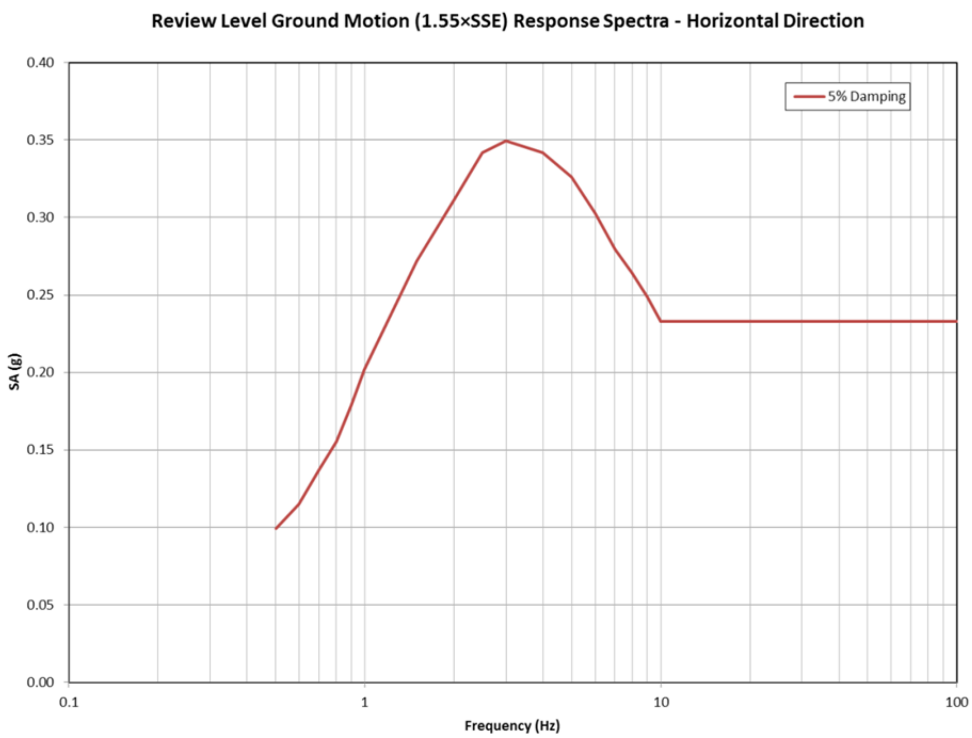

Figure 5-1: RLGM for James A. FitzPatrick

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

The RLGM ISRS for James A. FitzPatrick 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 1.55.
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) [42].
2. Probabilistic approach using the fragility analysis methodology of EPRI TR-103959, Methodology for Developing Seismic Fragilities [43].

6.1 Summary of Methodologies Used

James A. FitzPatrick performed a 0.3g focused-scope SMA in accordance with the methodology of NUREG-1407[44] in 1996 as part of Individual Plant Examination of External Events (IPEEE) program. The SMA is documented in [45] and consisted of screening evaluations, seismic walkdowns, and a review of the plant seismic design basis. The SMA was performed in accordance with EPRI NP-6041-SL [42]. The evaluation of mechanical and electrical equipment relied heavily on the walkdowns conducted for the USI A-46 seismic evaluation. Section 3.3 and Appendix B of [40] established that the results of the James A. FitzPatrick IPEEE are adequate to support screening of the updated seismic hazard for James A. FitzPatrick. Consequently, for ESEP, the results of HCLPF evaluations performed for IPEEE are used to screen out components with capacity that exceeds RLGM.

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. 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 1.55xSSE with a PGA of 0.233g, Figure 5-1.

6.2 HCLPF Screening Process

For ESEP, the components are screened considering the RLGM (1.55×SSE) with a 0.233g PGA. The screening tables in EPRI NP-6041-SL [42] 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 James A. FitzPatrick ESEL contains 145 items. Of these, 45 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. Additionally, items with HCLPF capacities greater than RLGM that were calculated in [45] were also screened out.

Block walls located in the proximity of ESEL equipment were assessed for potential seismic interaction impact resulting from the RLGM by reviewing the existing plant documents 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 [42] 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 James A. FitzPatrick seismic IPEEE program, for the 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 previous walkdowns, 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 [42], no significant outliers or anchorage concerns were identified during the James A. FitzPatrick seismic walkdowns. Based on walkdown results, no HCLPF capacity evaluations were required.

6.4 HCLPF Calculation Process

ESEL items identified for ESEP at James A. FitzPatrick were evaluated using the criteria in EPRI NP-6041-SL [42] 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 (USI A-46, 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

6.5 Functional Evaluations of Relays

No seal in /lockout type relays were identified on James A. FitzPatrick ESEL. Therefore, no relay evaluations were performed.

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 [42] 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."

ESEL components were determined to have adequate capacity for the design basis loads and HCLPF greater than RLGM.

7.0 INACCESSIBLE ITEMS

7.1 Identification of ESEL Item Inaccessible for Walkdowns

Forty-four (44) components on the ESEL were inaccessible and not walked down since they are either located in the Primary Containment Building, or were otherwise determined to be inaccessible due to radiological, safety, or plant operational constraints present at the time. The following is the list of subject components that require follow up seismic walkdowns:

- 02RV-71A, ADS Main Steam Line A Safety/Relief Valve
- 02RV-71B, ADS Main Steam Line A Safety/Relief Valve
- 02RV-71C, ADS Main Steam Line B Safety/Relief Valve
- 02RV-71D, ADS Main Steam Line B Safety/Relief Valve
- 02RV-71E, ADS Main Steam Line C Safety/Relief Valve
- 02RV-71F, Main Steam Line C Manual Safety Relief Valve
- 02RV-71G, ADS Main Steam Line C Safety/Relief Valve
- 02RV-71H, ADS Main Steam Line D Safety/Relief Valve
- 02RV-71J, Main Steam Line D Manual Safety Relief Valve
- 02RV-71K, Main Steam Line A Manual Safety Relief Valve
- 02RV-71L, Main Steam Line D Manual Safety Relief Valve
- 02SOV-71A1, ADS/MST A 02TV-71A Auto/CR Manual Pilot Solenoid Valve
- 02SOV-71B1, ADS/MST A 02RV-71B Auto/CR Manual Pilot Solenoid Valve
- 02SOV-71C1, ADS/MST B 02RV-71C Auto/CR Manual Pilot Solenoid Valve

- 02SOV-71D1, ADS/MST B 02RV-71D Auto/CR Manual Pilot Solenoid Valve
- 02SOV-71E1, ADS/MST C 02RV-71E Auto/CR Manual Pilot Solenoid Valve
- 02SOV-71F1, MST C 02RV-71F Control Room Manual Pilot Solenoid Valve
- 02SOV-71G1, ADS/MST C 02RV-71G Auto/CR Manual Pilot Solenoid Valve
- 02SOV-71H1, ADS/MST D 02RV-71H Auto/CR Manual Pilot Solenoid Valve
- 02SOV-71J1, MST D 02RV-71J Control Room Manual Pilot Solenoid Valve
- 02SOV-71K1, MST A 02RV-71K Control Room Manual Pilot Solenoid Valve
- 02SOV-71L1, MST D 02RV-71L Control Room Manual Pilot Solenoid Valve
- 10E-2A, Residual Heat Removal System Heat Exchanger A
- 10MOV-17, RHR SDC Outbd Isol. Valve
- 10MOV-18, RHR SDC Inbd Isol. Valve
- 10MOV-89A, RHR Heat Exch A Service Water Outlet Isol Valve
- 13MOV-41, RCIC Pump Suct From Suppr Pool INBD Isol Valve
- 13MOV-15, RCIC Steam Supply INBD Isol Valve
- 13MOV-16, RCIC Turbine Steam Supply Outbd Isol Valve
- 16-1RTD-108, LRT Drywell Area 4 Resist Temp Detector
- 16-1RTD-131A, Torus Bulk Temp Monitor 0 Azimuth Bay L X-232 Resist Temp Detector
- 23LT-203A2, Wide Range Containment Level HPCI Logic Level Xmitter (LO Tap)
- 27PT-101A, Torus Wide Range Press Xmitter
- 39ACC-256A, IAS 02RV-71A Air Accumulator
- 39ACC-256B, IAS 02RV-71B Air Accumulator
- 39ACC-256C, IAS 02RV-71C Air Accumulator
- 39ACC-256D, IAS 02RV-71D Air Accumulator
- 39ACC-256E, IAS 02RV-71E/F Air Accumulator
- 39ACC-256G, IAS 02RV-71G Air Accumulator
- 39ACC-256H, IAS 02RV-71H Air Accumulator
- 39ACC-256J, IAS 02RV-71J Air Accumulator
- 39ACC-256K, IAS 02RV-71K Air Accumulator
- 39ACC-256L, IAS 02RV-71L Air Accumulator
- Torus, Suppression Pool

7.2 Planned Walkdown / Evaluation Schedule / Close Out

The walkdowns of the inaccessible items identified in Section 7.1 are scheduled to be performed on no later than the second planned refueling outage after December 31, 2014.

8.0 ESEP CONCLUSIONS AND RESULTS

8.1 Supporting Information

James A. FitzPatrick 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 James A. FitzPatrick response to the NRC's 50.54(f) letter. On March 12, 2014, NEI submitted to the NRC results of a study [47] 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 [46] concluded that the "fleet wide seismic risk estimates are consistent with the approach and results used in the Gl-199 safety/risk assessment." The letter also stated that "As a result, the staff has confirmed that the conclusions reached in Gl-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 James A. FitzPatrick was included in the fleet risk evaluation submitted in the March 12, 2014 NEI letter [47] therefore, the conclusions in the NRC's May 9 letter also apply to James A. FitzPatrick.

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.

Based on the results of the screening evaluation performed in [40], James A. FitzPatrick screens-out of a risk evaluation. The NRC Screening and Prioritization Results letter concluded James A. FitzPatrick conditionally screens-in for the seismic risk evaluation [46] for the purpose of prioritizing and conducting additional evaluations. Consistent with [40] and detailed in this submittal, the IHS bound the GMRS in the 1 Hz to 10 Hz range [48]. Entergy is available to continue interactions with the NRC in support of the determination that a seismic risk evaluation is not required for James A. FitzPatrick.

8.2 Identification of Planned Modifications

Insights from the ESEP identified that there is no plant modification required.

8.3 Modification Implementation Schedule

There is no plant modification required.

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	N/A	N/A	Perform seismic walkdowns, generate HCLPF calculations and design and implement any necessary modifications for inaccessible items listed in Section 7.1	No later than the end of the second planned refueling outage after December 31, 2014.
2	N/A	N/A	Submit a letter to NRC summarizing the HCLPF results of Action 1 and confirming implementation of the plant modifications associated with Action 1	Within 60 days following completion of ESEP activities, including Action 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 JAFP-13-0025 "Overall Integrated Plan in Response to March 12, 2012, Commission Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," February 28, 2013, NRC ADAMS Accession No. ML13063A287.
4. Entergy Letter to U.S. NRC, letter number JAFP-14-0105, "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. ML14241A261.
5. Entergy Drawing FM-22A, Rev. 56, "Flow Diagram, Reactor Core Isolation Cooling, System 13."
6. Entergy Drawing FM-29A, Rev. 58, "Flow Diagram, Main Steam, System 29."
7. Entergy Drawing FM-47A, Rev. 52, Flow Diagram, Nuclear Boiler Vessel Instruments, System 02-3."
8. Entergy Drawing FM-18A, Rev. 57, "Flow Diagram, Drywell Inserting C.A.D. and Purge, System 27."
9. Entergy Drawing FM-20A, Rev. 72, "Flow Diagram, Residual Heat Removal, System 10."
10. Entergy Drawing FM-20B, Rev. 72, "Flow Diagram, Residual Heat Removal, System 10."
11. Entergy Drawing FE-1AH, Rev. 32, "125V DC One Line Diagram, Sheet 1."
12. Entergy Drawing FE-1AJ, Rev. 21, "125V DC One Line Diagram, Sheet 2."
13. Entergy Drawing FE-1AL, Rev. 28, "125V DC One Line Diagram, Sheet 4."
14. Entergy Drawing FE-1AX, Rev. 20, "125V DC One Line Diagram, Sheet 7."
15. Entergy Drawing FE-1H, Rev. 14, "4160V One Line Diagram, Sh. 4, Emergency Bus 10500."
16. Entergy Drawing FE-1BH, Rev. 11, "600V One Line Diagram, Sh.17, 71MCC-156 & 71MCC-166."
17. Entergy Drawing FE-1R, Rev. 29, "600V One Line Diagram, Sh.7, 71MCC-131, 141, 252, & 262."
18. Entergy Drawing FE-1Z, Rev. 26, "600V One Line Diagram, Sh.15, 71MCC-253, 263, 254, & 264."
19. Entergy Drawing FE-3DD, Rev. 16, "External Connections, Residual Heat Removal Panel 09-32, Sh. 2, System 10."
20. Entergy Drawing SE-9NM, Rev. 25, "Distribution Panel 71ACA2 Emergency Control & Instrument Bus A2."
21. Entergy Drawing SE-9PL, Rev. 10, "71UPP Uninterruptable Power Supply UPS Static Inverter."
22. Entergy Drawing SE-11A, Rev. 17, "Distribution Panel 71ACAUPS Uninterruptable Power."

23. Entergy Drawing SE-11D, Rev. 11, “Distribution Panel 71ESSA1 Safeguard Control & Instrument Bus A1.”

24. Entergy Drawing 1.61-154, Rev. 13, “Elem Diag RCIC Sys.”

25. Entergy Drawing 1.61-156, Rev. 6, “Elem Diag RCIC Sys.”

26. Entergy Drawing 1.49-164, Rev. 1, “DC/AC Inverter 71-INV-1A & 1B Schematic Diagram.”

27. Entergy Drawing LP-02-3AD, Rev. 2, “LOOP Diagram, NBI Reactor Wide Range Level Transmitter.”

28. Entergy Drawing LP-02-3AA, Rev. 1, “LOOP Diagram, Reactor Vessel Shroud Level, NBI/RHR Interlock Level.”

29. Entergy Drawing LP-33-209, Rev. 3, “Condensate Storage Tanks 12A & 12B Level.”

30. Entergy Drawing LP-06A, Rev. 2, “LOOP Diagram FWC ECCS Monitor, Reactor Pressure.”

31. Entergy Drawing LP-27-115A1, Rev. 2, “I&C LOOP Diagram Drywell Pressure (NR) (Div. I - RED).”

32. Entergy Drawing LP-27-115A2, Rev. 2, “I&C LOOP Diagram Drywell Pressure (WR) (Div. I - RED).”

33. Entergy Drawing LP-27-118, Rev. 4, “LOOP Diagram Reactor Building Suppression Chamber Pressure.”

34. Entergy Drawing LP-16-1-60, Rev. 3, “Reactor Building Drywell Temperature A.”

35. Entergy Drawing LP-16-1-50, Rev. 4, “Reactor Building Suppression Pool Temperature A.”

36. Entergy Drawing LP-23AJ, Rev. 2, “LOOP Diagram HPCI, Containment Wide Range Level.”

37. Entergy Drawing LP-23AG, Rev. 4, “LOOP Diagram HPCI, Suppression Pool Water.”

38. Entergy Drawing FB-48A, Rev. 34, “Flow Diagram, Fire Protection Water Piping, System 76.”

39. “James A. FitzPatrick Nuclear Power Plant FSAR Update,” Docket No. 50-333, 2013.

40. Entergy Letter to NRC, letter number JAFP-14-0039, “Entergy'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,” March 31, 2014, NRC ADAMS Accession No. ML14090A243.

41. EPRI Document, “Fitzpatrick Seismic Hazard and Screening Report,” Revision 1, February 27, 2014.

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

43. EPRI TR-103959, “Methodology for Developing Seismic Fragilities,” July 1994.

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

45. Entergy Document JAF-RPT-MISC-02211, “James A. FitzPatrick Nuclear Power Plant Individual Plant Examination of External Events,” Revision 0, June 1996.

46. 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.

47. 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.

48. Memorandum to David Skeens, Director, Japan Lessons Learned Project Directorate, Office of Nuclear Reactor Regulation from Scott Flanders, Director, Division of Site Safety and Environmental Analysis, Office of New Reactors, Subject: "Support Document for Screening and Prioritization Results Regarding Seismic Hazard Re-Evaluations for Operating Reactors in the Central and Eastern United States," May 21, 2014, NRC ADAMS Accession No. ML14136A126.

49. Entergy Document EC52427, "Fukushima – Acceptance of Expedited Seismic Evaluation Program (ESEP) Documentation," the following AREVA documents are captured in the plant document management system:

    a. AREVA Document 51-9219585-003, "ESEP Expedited Seismic Equipment List (ESEL) – James A. FitzPatrick Nuclear Power Plant."

ATTACHMENT A – JAMES A. FITZPATRICK ESEL

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
1	13P-1	RCIC Turbine Driven Pump	Off	On	Powered from 125VDC (71DC-A2)	[5]
2	13E-1	RCIC Barometric Condenser	Available	Available	-	[5]
3	13E-2	RCIC Turbine Lube Oil Cooler	Available	Available	-	[5]
4	13TK-1	RCIC Vacuum Tank	Available	Available	-	[5]
5	13MOV-18	RCIC Pump Suct from Cond Stor Isol Valve	Open	Cycled	Powered from 125VDC (BMCC-1).	[5]
6	13MOV-41	RCIC Pump Suct From Suppr Pool INBD Isol Valve	Closed	Cycled	Powered from 125VDC (BMCC-3)	[5]
7	13MOV-39	RCIC Pump Suct From Suppr Pool Outboard Isol Valve	Closed	Cycled	Powered from 125VDC (BMCC-3)	[5]
8	13MOV-15	RCIC Steam Supply INBD Isol Valve	Open	Open	Powered from AC. May be excluded since normally open, required open.	[5]
9	13MOV-16	RCIC Turbine Steam Supply Outbd Isol Valve	Open	Open	Powered from 125VDC (BMCC-1)	[5]
10	13MOV-131	RCIC Turbine Steam Inlet Isol Valve	Closed	Open	Powered from 125VDC (BMCC-3)	[5]
11	13HOV-1	RCIC Trip Vlv	Open	Open	Powered from 125VDC (71DC-A2)	[5]
12	13HOV-2	RCIC Turbine Governor Valve	Open	Throttled	Powered from 125VDC (71DC-A2)	[5]
13	13MOV-132	RCIC Turb Lube Oil Cooler Water Supply Isol Valve	Closed	Open	Powered from 125VDC (BMCC-3)	[5]
14	13PCV-23	RCIC Turb Lube Oil Cooler Water Supply Press Control Valve	Open	Throttled	Self-actuated	[5]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
15	13MOV-20	RCIC Pump Disch to Reactor Outbd Isol Valve	Open	Open	Powered from 125VDC (BMCC-1). May be excluded since normally open, required open.	[5]
16	13MOV-21	RCIC Pump Disch to Reactor Inbd Isol Valve	Closed	Open	Powered from 125VDC (BMCC-1)	[5]
17	13P-3	RCIC Barometric Cndsr Vacuum Pump	Off	On	Powered from 125VDC (BMCC-1)	[5]
18	13P-4	RCIC Condensate Pump	Off	On	Powered from 125VDC (BMCC-1)	[5]
19	33TK-12A	Condensate Storage Tank A	Available	Available	-	[5]
20	33TK-12B	Condensate Storage Tank B	Available	Available	-	[5]
21	Torus	Suppression Pool	Available	Available	-	[5]
22	02RV-71A	ADS Main Steam Line A Safety/Relief Valve	Closed	Cycle	-	[6]
23	39ACC-256A	IAS 02RV-71A Air Accumulator	Available	Available	-	[6]
24	02SOV-71A1	ADS/MST A 02TV-71A Auto/CR Manual Pilot Solenoid Valve	Closed	Cycle	Powered from 125VDC (71DC-A2)	[6]
25	02RV-71B	ADS Main Steam Line A Safety/Relief Valve	Closed	Cycle	-	[6]
26	39ACC-256B	IAS 02RV-71B Air Accumulator	Available	Available	The accumulator is shown on reference for 02RV-71A only – typical for all, except 71E and F share an accumulator.	[6]
27	02SOV-71B1	ADS/MST A 02RV-71B Auto/CR Manual Pilot Solenoid Valve	Closed	Cycle	Powered from 125VDC (71DC-A2) Shown on reference only for 02SOV71A1 – typical for all SOVs.	[6]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
28	02RV-71C	ADS Main Steam Line B Safety/Relief Valve	Closed	Cycle		[6]
29	39ACC-256C	IAS 02RV-71C Air Accumulator	Available	Available	The accumulator is shown on reference for 02RV-71A only – typical for all, except 71E and F share an accumulator.	[6]
30	02SOV-71C1	ADS/MST B 02RV-71C Auto/CR Manual Pilot Solenoid Valve	Closed	Cycle	Powered from 125VDC (71DC-A2) Shown on reference only for 02SOV71A1 – typical for all SOVs.	[6]
31	02RV-71D	ADS Main Steam Line B Safety/Relief Valve	Closed	Cycle	-	[6]
32	39ACC-256D	IAS 02RV-71D Air Accumulator	Available	Available	The accumulator is shown on reference for 02RV-71A only – typical for all, except 71E and F share an accumulator.	[6]
33	02SOV-71D1	ADS/MST B 02RV-71D Auto/CR Manual Pilot Solenoid Valve	Closed	Cycle	Powered from 125VDC (71DC-A2) Shown on reference only for 02SOV71A1 – typical for all SOVs.	[6]
34	02RV-71E	ADS Main Steam Line C Safety/Relief Valve	Closed	Cycle	-	[6]
35	39ACC-256E	IAS 02RV-71E/F Air Accumulator	Available	Available	The accumulator is shown on reference for 02RV-71A only – typical for all, except 71E and F share an accumulator.	[6]
36	02SOV-71E1	ADS/MST C 02RV-71E Auto/CR Manual Pilot Solenoid Valve	Closed	Cycle	Powered from 125VDC (71DC-A2) Shown on reference only for 02SOV71A1 – typical for all SOVs.	[6]
37	02RV-71F	Main Steam Line C Manual Safety Relief Valve	Closed	Cycle	-	[6]
38	02SOV-71F1	MST C 02RV-71F Control Room Manual Pilot Solenoid Valve	Available	Available	Powered from 125VDC (71DC-A2) Shown on reference only for 02SOV71A1 – typical for all SOVs.	[6]
39	02RV-71G	ADS Main Steam Line C Safety/Relief Valve	Closed	Cycle	-	[6]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
40	39ACC-256G	IAS 02RV-71G Air Accumulator	Closed	Cycle	The accumulator is shown on reference for 02RV-71A only – typical for all, except 71E and F share an accumulator.	[6]
41	02SOV-71G1	ADS/MST C 02RV-71G Auto/CR Manual Pilot Solenoid Valve	Available	Available	Powered from 125VDC (71DC-A2) Shown on reference only for 02SOV71A1 – typical for all SOVs.	[6]
42	02RV-71H	ADS Main Steam Line D Safety/Relief Valve	Closed	Cycle	-	[6]
43	39ACC-256H	IAS 02RV-71H Air Accumulator	Closed	Cycle	The accumulator is shown on reference for 02RV-71A only – typical for all, except 71E and F share an accumulator.	[6]
44	02SOV-71H1	ADS/MST D 02RV-71H Auto/CR Manual Pilot Solenoid Valve	Available	Available	Powered from 125VDC (71DC-A2) Shown on reference only for 02SOV71A1 – typical for all SOVs.	[6]
45	02RV-71J	Main Steam Line D Manual Safety Relief Valve	Closed	Cycle	-	[6]
46	39ACC-256J	IAS 02RV-71J Air Accumulator	Closed	Cycle	The accumulator is shown on reference for 02RV-71A only – typical for all, except 71E and F share an accumulator.	[6]
47	02SOV-71J1	MST D 02RV-71J Control Room Manual Pilot Solenoid Valve	Available	Available	Powered from 125VDC (71DC-A2) Shown on reference only for 02SOV71A1 – typical for all SOVs.	[6]
48	02RV-71K	Main Steam Line A Manual Safety Relief Valve	Closed	Cycle	-	[6]
49	39ACC-256K	IAS 02RV-71K Air Accumulator	Closed	Cycle	The accumulator is shown on reference for 02RV-71A only – typical for all, except 71E and F share an accumulator.	[6]
50	02SOV-71K1	MST A 02RV-71K Control Room Manual Pilot Solenoid Valve	Available	Available	Powered from 125VDC (71DC-A2) Shown on reference only for 02SOV71A1 – typical for all SOVs.	[6]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
51	02RV-71L	Main Steam Line D Manual Safety Relief Valve	Closed	Cycle	-	[6]
52	39ACC-256L	IAS 02RV-71L Air Accumulator	Closed	Cycle	The accumulator is shown on reference for 02RV-71A only – typical for all, except 71E and F share an accumulator.	[6]
53	02SOV-71L1	MST D 02RV-71L Control Room Manual Pilot Solenoid Valve	Available	Available	Powered from 125VDC (71DC-A2) Shown on reference only for 02SOV71A1 – typical for all SOVs.	[6]
54	27TK-7A	Safety-related Nitrogen Tank	Closed	Cycle	Provide backup for instrument air for SRVs	[8]
55	27TK-7B	Safety-related Nitrogen Tank	Available	Available	Provide backup for instrument air for SRVs	[8]
56	76P-1	West Diesel Fire Pump	Off	On	Seismic qualified	[38]
57	TBD	Reliable Hardened Vent	Closed	Cycled	Not yet installed	[3]
58	02-3LI-85A	RX Water Lvl	Available	Available	Powered from 13P/S-107	[27]
59	02-3LT-85A	Reactor Vessel Wide Range Level Xmitter	Available	Available	Powered from 13P/S-107	[27]
60	13INV-152	Inverter 13-152	Available	Available	Powered from 71DC-A2	[24]
61	13P/S-107	Single Nest Power Supply	Available	Available	Powered from 13INV-152	[24]
62	06-LI-094A	Reactor Water A Level Indicator	Available	Available	Power from DC A	[7]
63	06-LI-094C	Rx Water Lvl A	Available	Available	Power from DC A	[7]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
64	02-3LI-91	RX Wtr Lvl - Fuel Zone	Available	Available	Battery A	[28]
65	02-3LT-73	Reactor Vessel RHR Interlock Level Xmitter	Available	Available	Input to 02-LI-91	[28]
66	02-3MTU-273	Containment Spray Perm Master Trip Unit	Available	Available	Input to 02-LI-91	[28]
67	33LI-101A	CST Level	Available	Available	Power from 33E/S-G	[29]
68	33LT-101	Condensate Storage Tanks Level Xmitter	Available	Available	Power from 33E/S-G	[29]
69	33E/S-G	BOP Inst Pwr Supp	Available	Available	Power from 71ACUPS-1. In panel 09BOP-P/S-1	[29]
70	09BOP/PS-1	BOP Inst Pwr Supp Panel	Available	Available		[29]
71	06PI-61A	Reactor Vessel Press Indic	Available	Available	120VAC (71ACA2) –Backup 71INV-1A (71DC-A5)	[30]
72	06PT-61A	ECCS Loop A Feedwater Control Reactor Press Xmitter	Available	Available	120VAC (71ACA2) –Backup 71INV-1A (71DC-A5)	[30]
73	06SCM-61A	Reactor Press "A" Signal Conditioner	Available	Available	120VAC (71ACA2) –Backup 71INV-1A (71DC-A5)	[30]
74	06SDM-61A	Reactor Press "A" Sig Dist Module	Available	Available	120VAC (71ACA2) –Backup 71INV-1A (71DC-A5)	[30]
75	27PI-115A1	NR PC Press Indicator	Available	Available	120VAC (71ACA2) –Backup 71INV-1A (71DC-A5)	[31]
76	27PT-115A1	Drywell Div 1 Narrow Range Press Xmitter	Available	Available	120VAC (71ACA2) –Backup 71INV-1A (71DC-A5)	[31]
77	27SCM-115A	CAD Drywell Press Div 1 Input Module	Available	Available	120VAC (71ACA2) –Backup 71INV-1A (71DC-A5)	[31]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
78	27SDM-115A	CAD Drywell Press Div 1 Distribution Module	Available	Available	120VAC (71ACA2) –Backup 71INV-1A (71DC-A5)	[31]
79	27PI-115A2	WR PC Press Indicator	Available	Available	120VAC (71ACA2) –Backup 71INV-1A (71DC-A5)	[32]
80	27PT-115A2	Drywell Div I Wide Range Press Xmitter	Available	Available	120VAC (71ACA2) –Backup 71INV-1A (71DC-A5)	[32]
81	27PR-101A	Suppression Chamber Monitor Press Recorder	Available	Available	Powered from 10P/S-100A	[33]
82	27PT-101A	Torus Wide Range Press Xmitter	Available	Available	Powered from 10P/S-100A	[32]
83	27SDM-101A	Suppression Chamber Monitor Signal Dist Module	Available	Available	Powered from 10P/S-100A	[32]
84	16-1TR-108	LRT Drywell Temp Mon Temp Recorder	Available	Available	Powered from 10P/S-100A	[34]
85	16-1RTD-108	LRT Drywell Area 4 Resist Temp Detector	Available	Available	Powered from 10P/S-100A	[34]
86	16-1SDM-108	LRT Drywell Temp Mon Signal Dist Module	Available	Available	Powered from 10P/S-100A	[34]
87	10P/S-100A	Power Supply	Available	Available	Powered from 120VAC (71ESSA1)	[34]
88	16-1TR-131A	Torus Bulk Temp Mon Average Temp Recorder	Available	Available	Powered from 23E/S-200A	[35]
89	16-1RTD-131A	Torus Bulk Temp Monitor 0 Azimuth Bay L X-232 Resist Temp Detector	Available	Available	Powered from 23E/S-200A	[35]
90	16-1SDM-131A	Torus Temp Mon A Signal Dist Module	Available	Available	Powered from 23E/S-200A	[35]
91	23E/S-200A	Power Supply PS-1A	Available	Available	Powered from 120VAC (71ACA2), backup by 71INV-1A (71DC-A5)	[35]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
92	23LI-203A	PC Lvl Indicator	Available	Available	120VAC (71ACA2), backup by 71INV-1A (71DC-A5)	[36]
93	23LT-203A1	Wide Range Containment Level HPCI Logic Level Xmitter (HI Tap)	Available	Available	120VAC (71ACA2), backup by 71INV-1A (71DC-A5)	[36]
94	23LT-203A2	Wide Range Containment Level HPCI Logic Level Xmitter (LO Tap)	Available	Available	120VAC (71ACA2), backup by 71INV-1A (71DC-A5)	[36]
95	23SCM-203A	HPCI Drywell Sump Level Div I Input Signal Module	Available	Available	120VAC (71ACA2), backup by 71INV-1A (71DC-A5)	[36]
96	23SUM-203A	HPCI Drywell/Torus Diff Press Subtraction Module	Available	Available	120VAC (71ACA2), backup by 71INV-1A (71DC-A5)	[36]
97	23LI-202A	Suppression Chamber Water Level Indic	Available	Available	120VAC (71ACA2), backup by 71INV-1A (71DC-A5)	[37]
98	23LT-202A	Suppression Pool HPCI Logic Level Xmitter	Available	Available	120VAC (71ACA2), backup by 71INV-1A (71DC-A5)	[37]
99	23SCM-202A	HPCI Suppression Chamber Level Div I Input Signal Module	Available	Available	120VAC (71ACA2), backup by 71INV-1A (71DC-A5)	[37]
100	23SDM-203A	HPCI Drywell Sump Level Div I Signal Distrib Module	Available	Available	120VAC (71ACA2), backup by 71INV-1A (71DC-A5)	[37]
101	25-05	Reactor Protection and NSSS System Rack	Available	Available	-	[27][30]
102	25-51	Jet Pump Instrument Rack 25-51	Available	Available	-	[28]
103	27MAP	Monitoring Analysis Panel	Available	Available	-	[30]
104	09-3	Nuclear Station Main Control Board	Available	Available	-	[28][30]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
105	09-4	RWCU & Recirc Control Panel	Available	Available	-	[28]
106	09-5	Reactor Control Main Control Board	Available	Available	-	[27]
107	09-6	BOP Main Control Board Panel(MECH)	Available	Available	-	[29]
108	09-24	Process Instrumentation Panel (Div I)	Available	Available	-	[27]
109	09-30	Relay Cabinet Channel ‘A’ RCIC Panel	Available	Available	-	[25]
110	09-32	Channel ‘A’ RHR/RCIC Relay Panel	Available	Available	Relay panel (for various valve control circuits – reference is an example)	[19]
111	09-95	Emergency Core Cooling System DIV 1 A/C Trip Cabinet	Available	Available	-	[28]
112	TBD	RHV Instrumentation	Available	Available	Not yet installed	[3]
113	71SB-1	125 Volt Station Battery A	Operating	Operating	-	[11]
114	71BC-1A	125 VDC Station Battery Charger	Operating	Operating	Powered from 71MCC-252	[11]
115	71BCB-2A	Battery Control Board A	Operating	Operating	-	[11]
116	71DC-A2	Relay Room Distribution Cabinet	Operating	Operating	-	[13]
117	71DC-A5	Relay Room Distribution Cabinet	Operating	Operating	-	[14]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
118	71BMCC-1	Reactor Building West Crescent Motor Control Center	Operating	Operating	-	[12]
119	71BMCC-3	Reactor Building West Crescent Motor Control Center	Operating	Operating	-	[12]
120	71ESSA1	Relay Room Safeguard Power Distribution Panel	Operating	Operating	120VAC Instrument power. Powered from 71MCC-252	[23]
121	TBD	120V Instrument Power Inverter 1	Operating	Operating	To power required instruments from battery. Not yet installed	[3]
122	71ACA2	Relay Room Emergency Power Distribution Panel	Operating	Operating	Instrument Power	[20]
123	71ACUPS	Dist. Panel – Uninterruptible Bus	Operating	Operating	Powered from 71UPP	[22]
124	71UPP	UPS Static Inverter	Operating	Operating	Powered from 71MCC-262 (not selected train), backup from 71MCC-252 or 71BCB-2A	[21]
125	71INV-1A	Instrument power inverter	Operating	Operating	Instrument power. Powered from 71DC-A5	[26]
126	71H05	4160V Switchgear Distribution (Bus 10500)	Operating	Operating	4160V FLEX generator connection point – primarily for RHR	[15]
127	71MCC-156	600V Motor Control Center (Bus 115600)	Operating	Operating	May be required to open 10MOV-18	[16]
128	71MCC-252	600V Motor Control Center Bus 125200	Operating	Operating	Power for battery charger 71BC-1A	[17]
129	71MCC-254	600V Motor Control Center Bus 125400	Operating	Operating	Power for 71MCC-252	[18]
130	10P-3A	Residual Heat Removal Pump A	Off	On	Phase 3 installed equipment	[9]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
131	10P-2A	RHR Keep-Full Pump A	Off	On	Phase 3 installed equipment	[9]
132	10E-2A	Residual Heat Removal System Heat Exchanger A	Available	Available	Phase 3 installed equipment	[10]
133	10MOV-25A	RHR A LPCI Inbd Inj Valve	Closed	Open	Phase 2/3 installed equipment	[9]
134	10MOV-66A	RHR Heat Exch A Bypass Valve	Open	Closed	Phase 3 installed equipment	[9]
135	10MOV-27A	RHR A LPCI Outbd Inj Valve	Open	Open	Phase 2/3 installed equipment. May be excluded since normally open, required open.	[9]
136	10MOV-13A	RHR Pump A Suction Torus Isol. Valve	Open	Closed	71MCC-153	[9]
137	10MOV-15A	RHR Pump A SDC Suction Isol Valve	Closed	Open	71MCC-153	[9]
138	10MOV-17	RHR SDC Outbd Isol. Valve	Closed	Open	71BMCC-4	[9]
139	10MOV-18	RHR SDC Inbd Isol. Valve	Closed	Open		[9]
140	10MOV-148A	RHRSW A to RHR Cross Tie Upstr Isol Valve	Closed	Open	Phase 2 installed equipment	[10]
141	10MOV-149A	RHRSW A to RHR Cross Tie Dnstr Isol Valve	Closed	Open	Phase 2 installed equipment	[10]
142	10RHR-432	RHRSW - Fire Protection Cross-Tie Isol Valve	Closed	Open	Manually opened	[10]
143	10MOV-89A	RHR Heat Exch A Service Water Outlet Isol Valve	Closed	Closed/ Open	Closed (Ph 2) Open (Ph 3)	[10]

ESEL Item Number	Equipment		Operating State		Notes/Comments	References
	ID	Description	Normal State	Desired State		
144	TBD	Fire Pump 76P-1 Temporary Connection Isolation Valve to RHRSW Piping	Closed	Open	To be manually Opened Not yet installed	[3]
145	TBD	Jib Crane for FLEX Pump	N/A	Available	To be permanently installed. Powered from FLEX DG. Not yet installed	[3]

ATTACHMENT B – ESEP HCLPF VALUES AND FAILURE MODES TABULATION

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
1	13P-1	RCIC Turbine Driven Pump	>RLGM	Screened	Note 1
2	13E-1	RCIC Barometric Condenser	>RLGM	Screened	Note 2
3	13E-2	RCIC Turbine Lube Oil Cooler	>RLGM	Screened	Note 2: In-Line Component
4	13TK-1	RCIC Vacuum Tank	>RLGM	Screened	Note 2
5	13MOV-18	RCIC Pump Suct from Cond Stor Isol Valve	>RLGM	Screened	
6	13MOV-41	RCIC Pump Suct From Suppr Pool INBD Isol Valve	TBD	TBD	Note 3
7	13MOV-39	RCIC Pump Suct From Suppr Pool Outboard Isol Valve	>RLGM	Screened	
8	13MOV-15	RCIC Steam Supply INBD Isol Valve	TBD	TBD	Note 3
9	13MOV-16	RCIC Turbine Steam Supply Outbd Isol Valve	TBD	TBD	Note 3
10	13MOV-131	RCIC Turbine Steam Inlet Isol Valve	>RLGM	Screened	
11	13HOV-1	RCIC Trip Vlv	>RLGM	Screened	
12	13HOV-2	RCIC Turbine Governor Valve	>RLGM	Screened	
13	13MOV-132	RCIC Turb Lube Oil Cooler Water Supply Isol Valve	>RLGM	Screened	
14	13PCV-23	RCIC Turb Lube Oil Cooler Water Supply Press Control Valve	>RLGM	Screened	
15	13MOV-20	RCIC Pump Disch to Reactor Outbd Isol Valve	>RLGM	Screened	
16	13MOV-21	RCIC Pump Disch to Reactor Inbd Isol Valve	>RLGM	Screened	
17	13P-3	RCIC Barometric Cndsr Vacuum Pump	>RLGM	Screened	Note 2
18	13P-4	RCIC Condensate Pump	>RLGM	Screened	Note 2
19	33TK-12A	Condensate Storage Tank A	>RLGM	Screened	Note 1
20	33TK-12B	Condensate Storage Tank B	>RLGM	Screened	Note 1

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
21	Torus	Suppression Pool	TBD	TBD	Note 3
22	02RV-71A	ADS Main Steam Line A Safety/Relief Valve	TBD	TBD	Note 3
23	39ACC-256A	IAS 02RV-71A Air Accumulator	TBD	TBD	Note 3
24	02SOV-71A1	ADS/MST A 02TV-71A Auto/CR Manual Pilot Solenoid Valve	TBD	TBD	Note 3
25	02RV-71B	ADS Main Steam Line A Safety/Relief Valve	TBD	TBD	Note 3
26	39ACC-256B	IAS 02RV-71B Air Accumulator	TBD	TBD	Note 3
27	02SOV-71B1	ADS/MST A 02RV-71B Auto/CR Manual Pilot Solenoid Valve	TBD	TBD	Note 3
28	02RV-71C	ADS Main Steam Line B Safety/Relief Valve	TBD	TBD	Note 3
29	39ACC-256C	IAS 02RV-71C Air Accumulator	TBD	TBD	Note 3
30	02SOV-71C1	ADS/MST B 02RV-71C Auto/CR Manual Pilot Solenoid Valve	TBD	TBD	Note 3
31	02RV-71D	ADS Main Steam Line B Safety/Relief Valve	TBD	TBD	Note 3
32	39ACC-256D	IAS 02RV-71D Air Accumulator	TBD	TBD	Note 3
33	02SOV-71D1	ADS/MST B 02RV-71D Auto/CR Manual Pilot Solenoid Valve	TBD	TBD	Note 3
34	02RV-71E	ADS Main Steam Line C Safety/Relief Valve	TBD	TBD	Note 3
35	39ACC-256E	IAS 02RV-71E/F Air Accumulator	TBD	TBD	Note 3
36	02SOV-71E1	ADS/MST C 02RV-71E Auto/CR Manual Pilot Solenoid Valve	TBD	TBD	Note 3
37	02RV-71F	Main Steam Line C Manual Safety Relief Valve	TBD	TBD	Note 3
38	02SOV-71F1	MST C 02RV-71F Control Room Manual Pilot Solenoid Valve	TBD	TBD	Note 3
39	02RV-71G	ADS Main Steam Line C Safety/Relief Valve	TBD	TBD	Note 3

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
40	39ACC-256G	IAS 02RV-71G Air Accumulator	TBD	TBD	Note 3
41	02SOV-71G1	ADS/MST C 02RV-71G Auto/CR Manual Pilot Solenoid Valve	TBD	TBD	Note 3
42	02RV-71H	ADS Main Steam Line D Safety/Relief Valve	TBD	TBD	Note 3
43	39ACC-256H	IAS 02RV-71H Air Accumulator	TBD	TBD	Note 3
44	02SOV-71H1	ADS/MST D 02RV-71H Auto/CR Manual Pilot Solenoid Valve	TBD	TBD	Note 3
45	02RV-71J	Main Steam Line D Manual Safety Relief Valve	TBD	TBD	Note 3
46	39ACC-256J	IAS 02RV-71J Air Accumulator	TBD	TBD	Note 3
47	02SOV-71J1	MST D 02RV-71J Control Room Manual Pilot Solenoid Valve	TBD	TBD	Note 3
48	02RV-71K	Main Steam Line A Manual Safety Relief Valve	TBD	TBD	Note 3
49	39ACC-256K	IAS 02RV-71K Air Accumulator	TBD	TBD	Note 3
50	02SOV-71K1	MST A 02RV-71K Control Room Manual Pilot Solenoid Valve	TBD	TBD	Note 3
51	02RV-71L	Main Steam Line D Manual Safety Relief Valve	TBD	TBD	Note 3
52	39ACC-256L	IAS 02RV-71L Air Accumulator	TBD	TBD	Note 3
53	02SOV-71L1	MST D 02RV-71L Control Room Manual Pilot Solenoid Valve	TBD	TBD	Note 3
54	27TK-7A	Safety-related Nitrogen Tank	>RLGM	Screened	Note 1
55	27TK-7B	Safety-related Nitrogen Tank	>RLGM	Screened	Note 1
56	76P-1	West Diesel Fire Pump	>RLGM	Screened	Note 2
57	TBD	Reliable Hardened Vent	Not Applicable	Not Applicable	New FLEX Component to be seismically designed.
58	02-3LI-85A	RX Water Lvl	>RLGM	Screened	Note 1

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
59	02-3LT-85A	Reactor Vessel Wide Range Level Xmitter	>RLGM	Screened	
60	13INV-152	Inverter 13-152	>RLGM	Screened	Note 1
61	13P/S-107	Single Nest Power Supply	>RLGM	Screened	Note 1
62	06-LI-094A	Reactor Water A Level Indicator	>RLGM	Screened	Note 1
63	06-LI-094C	Rx Water Lvl A	>RLGM	Screened	Note 1
64	02-3LI-91	RX Wtr Lvl - Fuel Zone	>RLGM	Screened	Note 1
65	02-3LT-73	Reactor Vessel RHR Interlock Level Xmitter	>RLGM	Screened	
66	02-3MTU-273	Containment Spray Perm Master Trip Unit	>RLGM	Screened	Note 1
67	33LI-101A	CST Level	>RLGM	Screened	Note 1
68	33LT-101	Condensate Storage Tanks Level Xmitter	>RLGM	Screened	
69	33E/S-G	BOP Inst Pwr Supp	>RLGM	Screened	
70	09BOP/PS-1	BOP Inst Pwr Supp Panel	>RLGM	Screened	Note 1
71	06PI-61A	Reactor Vessel Press Indic	>RLGM	Screened	Note 1
72	06PT-61A	ECCS Loop A Feedwater Control Reactor Press Xmitter	>RLGM	Screened	
73	06SCM-61A	Reactor Press "A" Signal Conditioner	>RLGM	Screened	Note 1
74	06SDM-61A	Reactor Press "A" Sig Dist Module	>RLGM	Screened	Note 1
75	27PI-115A1	NR PC Press Indicator	>RLGM	Screened	Note 1
76	27PT-115A1	Drywell Div 1 Narrow Range Press Xmitter	>RLGM	Screened	
77	27SCM-115A	CAD Drywell Press Div 1 Input Module	>RLGM	Screened	Note 1
78	27SDM-115A	CAD Drywell Press Div 1 Distribution Module	>RLGM	Screened	Note 1

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
79	27PI-115A2	WR PC Press Indicator	>RLGM	Screened	Note 1
80	27PT-115A2	Drywell Div I Wide Range Press Xmitter	>RLGM	Screened	
81	27PR-101A	Suppression Chamber Monitor Press Recorder	>RLGM	Screened	Note 1
82	27PT-101A	Torus Wide Range Press Xmitter	TBD	TBD	Note 3
83	27SDM-101A	Suppression Chamber Monitor Signal Dist Module	>RLGM	Screened	
84	16-1TR-108	LRT Drywell Temp Mon Temp Recorder	>RLGM	Screened	Note 1
85	16-1RTD-108	LRT Drywell Area 4 Resist Temp Detector	TBD	TBD	Note 3
86	16-1SDM-108	LRT Drywell Temp Mon Signal Dist Module	>RLGM	Screened	
87	10P/S-100A	Power Supply	>RLGM	Screened	
88	16-1TR-131A	Torus Bulk Temp Mon Average Temp Recorder	>RLGM	Screened	Note 1
89	16-1RTD-131A	Torus Bulk Temp Monitor 0 Azimuth Bay L X-232 Resist Temp Detector	TBD	TBD	Note 3
90	16-1SDM-131A	Torus Temp Mon A Signal Dist Module	>RLGM	Screened	Note 1
91	23E/S-200A	Power Supply PS-1A	>RLGM	Screened	Note 1
92	23LI-203A	PC Lvl Indicator	>RLGM	Screened	Note 1
93	23LT-203A1	Wide Range Containment Level HPCI Logic Level Xmitter (HI Tap)	>RLGM	Screened	
94	23LT-203A2	Wide Range Containment Level HPCI Logic Level Xmitter (LO Tap)	TBD	TBD	Note 3
95	23SCM-203A	HPCI Drywell Sump Level Div I Input Signal Module	>RLGM	Screened	Note 1
96	23SUM-203A	HPCI Drywell/Torus Diff Press Subtraction Module	>RLGM	Screened	Note 1
97	23LI-202A	Suppression Chamber Water Level Indic	>RLGM	Screened	Note 1
98	23LT-202A	Suppression Pool HPCI Logic Level Xmitter	>RLGM	Screened	

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
99	23SCM-202A	HPCI Suppression Chamber Level Div I Input Signal Module	>RLGM	Screened	Note 1
100	23SDM-203A	HPCI Drywell Sump Level Div I Signal Distrib Module	>RLGM	Screened	Note 1
101	25-05	Reactor Protection and NSSS System Rack	>RLGM	Screened	Note 1
102	25-51	Jet Pump Instrument Rack 25-51	>RLGM	Screened	Note 1
103	27MAP	Monitoring Analysis Panel	>RLGM	Screened	Note 1
104	09-3	Nuclear Station Main Control Board	>RLGM	Screened	Note 1
105	09-4	RWCU & Recirc Control Panel	>RLGM	Screened	Note 1
106	09-5	Reactor Control Main Control Board	>RLGM	Screened	Note 1
107	09-6	BOP Main Control Board Panel(MECH)	>RLGM	Screened	Note 1
108	09-24	Process Instrumentation Panel (Div I)	>RLGM	Screened	Note 1
109	09-30	Relay Cabinet Channel ‘A’ RCIC Panel	>RLGM	Screened	Note 1
110	09-32	Channel ‘A’ RHR/RCIC Relay Panel	>RLGM	Screened	Note 1
111	09-95	Emergency Core Cooling System DIV 1 A/C Trip Cabinet	>RLGM	Screened	Note 1
112	TBD	RHV Instrumentation	Not Applicable	Not Applicable	New FLEX Component to be seismically designed.
113	71SB-1	125 Volt Station Battery A	>RLGM	Screened	Note 1
114	71BC-1A	125 VDC Station Battery Charger	>RLGM	Screened	Note 1
115	71BCB-2A	Battery Control Board A	>RLGM	Screened	Note 2
116	71DC-A2	Relay Room Distribution Cabinet	>RLGM	Screened	Note 2
117	71DC-A5	Relay Room Distribution Cabinet	>RLGM	Screened	Note 2

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
118	71BMCC-1	Reactor Building West Crescent Motor Control Center	>RLGM	Screened	Note 1
119	71BMCC-3	Reactor Building West Crescent Motor Control Center	>RLGM	Screened	Note 1
120	71ESSA1	Relay Room Safeguard Power Distribution Panel	>RLGM	Screened	Note 2
121	TBD	120V Instrument Power Inverter 1	Not Applicable	Not Applicable	New FLEX Component to be seismically designed.
122	71ACA2	Relay Room Emergency Power Distribution Panel	>RLGM	Screened	Note 1
123	71ACUPS	Dist. Panel – Uninterruptible Bus	>RLGM	Screened	Note 1
124	71UPP	UPS Static Inverter	>RLGM	Screened	Note 1
125	71INV-1A	Instrument power inverter	>RLGM	Screened	Note 2
126	71H05	4160V Switchgear Distribution (Bus 10500)	>RLGM	Screened	Note 1
127	71MCC-156	600V Motor Control Center (Bus 115600)	>RLGM	Screened	Note 1
128	71MCC-252	600V Motor Control Center Bus 125200	>RLGM	Screened	Note 1
129	71MCC-254	600V Motor Control Center Bus 125400	>RLGM	Screened	Note 1
130	10P-3A	Residual Heat Removal Pump A	>RLGM	Screened	Note 1
131	10P-2A	RHR Keep-Full Pump A	>RLGM	Screened	Note 2
132	10E-2A	Residual Heat Removal System Heat Exchanger A	TBD	TBD	Note 3
133	10MOV-25A	RHR A LPCI Inbd Inj Valve	>RLGM	Screened	
134	10MOV-66A	RHR Heat Exch A Bypass Valve	>RLGM	Screened	
135	10MOV-27A	RHR A LPCI Outbd Inj Valve	>RLGM	Screened	
136	10MOV-13A	RHR Pump A Suction Torus Isol. Valve	>RLGM	Screened	
137	10MOV-15A	RHR Pump A SDC Suction Isol Valve	>RLGM	Screened	

Item No.	Equipment ID	Equipment Description	HCLPF (g) / Screening Level	Failure Mode	Comments
138	10MOV-17	RHR SDC Outbd Isol. Valve	TBD	TBD	Note 3
139	10MOV-18	RHR SDC Inbd Isol. Valve	TBD	TBD	Note 3
140	10MOV-148A	RHRSW A to RHR Cross Tie Upstr Isol Valve	>RLGM	Screened	
141	10MOV-149A	RHRSW A to RHR Cross Tie Dnstr Isol Valve	>RLGM	Screened	
142	10RHR-432	RHRSW - Fire Protection Cross-Tie Isol Valve	>RLGM	Screened	
143	10MOV-89A	RHR Heat Exch A Service Water Outlet Isol Valve	TBD	TBD	Note 3
144	TBD	Fire Pump 76P-1 Temporary Connection Isolation Valve to RHRSW Piping	Not Applicable	Not Applicable	New FLEX Component to be seismically designed.
145	TBD	Jib Crane for FLEX Pump	Not Applicable	Not Applicable	New FLEX Component to be seismically designed.

Notes:

1. Anchorage screened out based on available margin during walkdown by SRT.
2. Anchorage screened out during walkdown validation by SRT.
3. Inaccessible. Per EPRI NP-6041-SLR1, Sec. 2, Seismic Capability Walkdown, Step 5 - This component was not walked down.