Entergy

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Michael J. Colomb
Site Vice President

JAFP-12-0134
November 27, 2012

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

SUBJECT: Seismic Walkdown Report – Entergy's Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Seismic Aspects of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident

James A. FitzPatrick Nuclear Plant Power
Docket No. 50-333
License No. DPR-59

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, March 12, 2012 (ML12053A340)
2. Entergy's 120-Day Response to NRC Request for Information (RFI) Pursuant to 10 CFR 50.54(f) Regarding the Seismic Aspects of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident James A. FitzPatrick Nuclear Power Plant, July 9, 2012, JAFP-12-0075 (ML12192A515)

Dear Sir or Madam:

On March 12, 2012, the NRC issued Reference 1 to all power reactor licensees. Enclosure 3 of the Reference 1 contains specific requested actions, requested information, and required responses associated with Recommendation 2.3 for seismic walkdowns. Entergy Nuclear Operations, Inc. (Entergy) confirmed in Reference 2 that it would use the seismic walkdown procedure (EPRI Report 1025286, Seismic Walkdown Guidance For Resolution of Fukushima Near-Term Task Force Recommendation 2.3: Seismic) as endorsed by the NRC as the basis to conduct the walkdowns and develop the needed information at the James A. FitzPatrick Nuclear Power Plant (JAF).

Pursuant to Required Response 2 of Enclosure 3 of Reference 1, Entergy Nuclear Operations, Inc. (Entergy) is providing the Seismic Walkdown Report for the JAF in Attachment 1. All potentially adverse seismic conditions identified in the report have been evaluated in the Corrective Action Program for operability concerns and no immediate operability concerns have been identified.

Should you have any questions regarding this submittal, please contact Mr. Chris Adner, Licensing Manager, at (315) 349-6766.

A001
NRR

This letter contains new regulatory commitments, which are identified in Attachment 2.

I declare under penalty of perjury that the foregoing is true and correct; executed on November 27, 2012.

Sincerely,

Michael J. Colomb
Site Vice President

MJC/CA/kp

Attachments: 1. James A. FitzPatrick Nuclear Power Plant Seismic Walkdown Report, JAF-RPT-12-00014
2. List of Regulatory Commitments

cc: Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
2100 Renaissance Boulevard, Suite 100
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Resident Inspector's Office
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Director, Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
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ATTACHMENT 1

JAFP-12-0134

James A. FitzPatrick Nuclear Power Plant
Seismic Walkdown Report

Entergy

ENTERGY NUCLEAR
Engineering Report Cover Sheet

Engineering Report Title:

James A. FitzPatrick (JAF) Nuclear Power Plant Seismic Walkdown Report for Resolution of Fukushima Near-Term Task Force Recommendation 2.3: Seismic

Engineering Report Type:

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

Applicable Site(s)

IP1	☐	IP2	☐	IP3	☐	JAF	☒	PNPS	☐	VY	☐	WPO	☐
ANO1	☐	ANO2	☐	ECH	☐	GGNS	☐	RBS	☐	WF3	☐	PLP	☐

EC No. EC-40677

Report Origin: ☒ Entergy ☐ Vendor
Vendor Document No.: ______

Quality-Related: ☐ Yes ☒ No

Prepared by:	Yaroslav Losev / [signature] (Print Name/Sign)	Date: 11/21/12
Prepared by:	Pouria Pourghobadi / [signature] (Print Name/Sign)	Date: 11/21/2012
Reviewed by:	Laura Maclay / T. [signature] for Laura Maclay Per email Authorization Other (Print Name/Sign)	Date: 11/21/12
Reviewed by:	Richard Casella / Richard Casella Other (Print Name/Sign)	Date: 11/21/12
Reviewed by:	Tom T. Panayotidi / [signature] Peer Review Team Leader (Print Name/Sign)	Date: 11/21/12
Approved by:	Vincent Bacanskas / [signature] Manager (Print Name/Sign)	Date: 11/21/12
Approved by:	N/A Other (Print Name/Sign)	Date: —

James A. FitzPatrick (JAF) Nuclear Power Plant Seismic Walkdown Report

for Resolution of Fukushima Near-Term Task Force Recommendation 2.3: Seismic

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1.0 SCOPE AND OBJECTIVE

The Great Tohoku Earthquake of March 11, 2011 and the resulting tsunami caused an accident at the Fukushima Dai-ichi nuclear power plant in Japan. In response to this accident, the Nuclear Regulatory Commission (NRC) established the Near-Term Task Force (NTTF). The NTTF was tasked with conducting a systematic and methodical review of NRC processes and regulations and determining if the agency should make additional improvements to its regulatory system. On March 12, 2012 the NRC issued a 10CFR50.54(f) Letter, Pursuant to Title 10 of the Code of Federal Regulations Part 50, Subsection 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 [Ref. 10.1], requesting information from all licensees to support the NRC staff's evaluation of several of the NTTF recommendations. To support NTTF Recommendation 2.3, Enclosure 3 to the NRC requested that all licensees perform seismic walkdowns to gather and report information from the plant related to degraded, non-conforming, or unanalyzed conditions with respect to its current seismic licensing basis.

The Electric Power Research Institute (EPRI), with support and direction from the Nuclear Energy Institute (NEI), published industry guidance for conducting and documenting the seismic walkdowns. The guidance represented the results of extensive interaction between NRC, NEI, and other stakeholders. This industry guidance document, EPRI Report 1025286 [Ref. 10.2], hereafter referred to as "the Guidance," was formally endorsed by the NRC on May 31, 2012. Entergy James A. FitzPatrick (JAF) Nuclear Power Plant has committed [Ref. 10.12] to using this NRC-endorsed guidance as the basis for conducting and documenting seismic walkdowns for resolution of NTTF Recommendation 2.3: Seismic.

Entergy fleet procedure EN-DC-168, "Fukushima Near Term Task Force Recommendation 2.3 Seismic Walk-Down Procedure" [Ref. 10.11], outlines the steps required to gather information as needed to respond to the March 12,2012, 10CFR50.54(f) Letter as it pertains to the USNRC Near-Term Task Force (NTTF) Recommendation 2.3, Seismic.

The objective of this report is to document the results of the seismic walkdown effort undertaken for resolution of NTTF Recommendation 2.3: Seismic in accordance with the Guidance, and provide the information necessary for responding to Enclosure 3 to the 50.54(f) Letter.

2.0 SEISMIC LICENSING BASIS SUMMARY

JAF is a single unit BWR-4 (Boiling Water Reactor) with a Mark I containment, located in Oswego County, New York. General Electric (GE) designed the nuclear steam supply system and the turbine-generator. Stone & Webster was the Architect/Engineer for the plant. JAF began commercial operation in July of 1975, and is currently rated at 2,536 MWt power [Ref. 10.3, Section 1.2] and has a rated gross electrical output of approximately 881 MWe when operating at full power. This section summarizes the seismic licensing basis of structures, systems and components (SSCs) at JAF, which bound the context of the NTTF 2.3 Seismic Walkdown program.

2.1 SAFE SHUTDOWN EARTHQUAKE (SSE)

The seismic design for Class I structures (including the reactor building and all engineered safeguards) is based on dynamic analysis using acceleration response spectrum curves which are normalized to a ground motion of 0.08 g, for the Operating Basis Earthquake, and 0.15 g, for the Design Basis Earthquake. The basis of this design criterion is presented in Reference 10.3, Section 2.6. Class I seismically designed structures may be referred to as "Seismic Class I" structures [Ref. 10.3, Section 12.4.6.1].

The horizontal seismic forces were determined using a lumped mass frequency response analysis considering flexural, translational and rocking (in some cases) response. These analyses take into account rock-structure interaction.

The vertical response spectrum is assumed to be two-thirds the horizontal response spectrum of each earthquake and is considered to act simultaneously. Where applicable, the stresses are added directly.

The damping value of 2 percent of critical for concrete structures under Operating Basis Earthquake is less than the range of 3 to 5 percent for design within code allowable stresses recommended by Newmark and Hall in their paper "Design Criteria for Nuclear Reactors Subject to Earthquake Hazards." Under the Operating Basis Earthquake, the stresses are within the allowable code stresses; therefore, little cracking will occur in the concrete.

Newmark suggests a value of 7 to 10 percent of critical damping for stress levels at or just below yield point. To be conservative and minimize cracking in the concrete under the Design Basis Earthquake, 5 percent of critical damping is used.

Horizontal and vertical displacements due to Operating Basis and Design Basis Earthquakes are determined for all Class I structures. Based on calculated displacements, adequate space is provided between adjacent structures to ensure that basic structural elements do not strike each other when subjected to the worst combination of rocking, bending and shear deflections and translation movements that might be induced by an earthquake. All Class I systems passing between adjacent structures are designed to withstand the maximum combination of movement between the adjacent structures without loss of function. The effect of the relative movement between buildings is considered in the piping stress analysis and in the design and location of supports.

2.2 DESIGN CODES, STANDARDS, AND METHODS

Class I structures and equipment are those that are necessary to ensure: a) the integrity of the reactor coolant pressure boundary, b) The capability to shut down the reactor and maintain it in a safe, shutdown condition, or c) the capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the guidelines of 10 CFR 100. Class II structures and equipment are those which may be essential to operation of the plant, but are not included in Class I. Class III structures and equipment are those that are not included in Class I or Class II. A "Component Quality Assurance Category List" further defines "Class I", "Class II", or "Class III" structures, systems, and components as "Quality Assurance" (QA) SR, QP, or NSR. QA Categories SR, QP and NSR are synonymous with the previously used I, M and II/III (or II or III separately) categories, respectively. [Ref. 10.3, Section 12.2.1].

The Reference 10.8 document provides the basic criteria for the safety related Balance of Plant (BOP) pipe stress analysis and pipe support qualification and/or design for the JAFNPP. BOP piping systems are those systems which are not part of the General Electric (GE) Nuclear Steam Supply System (NSSS). A listing of the seismically qualified BOP pipe lines is provided in Reference 10.8, Section 7.0.

Class IE - The safety classification of the electric equipment and systems, including their supporting systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or otherwise are essential in mitigating the consequences of an accident [Ref. 10.10].

<u>Mechanical Equipment and Piping Code Applicability</u>

- USAS (ANSI) B31.1, 1967 Edition through 1969 Addenda, Power Piping Code.
- ASME Boiler and Pressure Vessel Code for Nuclear Vessels, Section III, Subsection B, 1968 Edition including the 1968 Summer Addendum.

<u>Pipe Support Code Applicability</u>

- USAS (ANSI) B31.1, 1967 Edition through 1969 Addenda, Power Piping Code.
- AISC Specification for Design, Fabrication, and Erection of Structural Steel for Buildings.

Applicable USNRC Regulatory Guides

Because JAFNPP was designed before the establishment of Regulatory Guides, no Regulatory Guide is directly applicable. However, the Regulatory Guide 1.61, Revision O, Damping Values for Seismic Design of Nuclear Power Plants, October 1973, was used for fluid transient analyses [10.8].

The USNRC Regulatory Guide 1.92, Section 1.2.1 "The Grouping Method." [Ref. 10.3, Section 12.5.4], was used for combining modal responses in the seismic reanalysis for the Wide Range and Narrow Range reactor water level piping systems.

Applicable IE Bulletins

- IE Bulletin 79-02, Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts.
- IE Bulletin 79-07, Seismic Stress Analysis of Safety-Related Piping.
- IE Bulletin 79-14, Seismic Analysis for As-Built Safety-Related Piping Systems including all supplements.
- IE Bulletin 80-11, Masonry Wall Design.

Mechanical Equipment and Equipment Support Analysis

- Overall equipment structural response includes loads from nozzles, equipment deadweight, design pressure, operating temperature, equipment earthquake loads and other dynamic loads. Reference 10.8, Section 5.0, Tables 5.0-1 and 5.0-2A through 2E show the accepted equipment nozzle loads that were established during the plant design stage and later during the as-built stress analysis effort.
- The criteria, method of analysis, and summary of critical stresses for various equipment are included in UFSAR Table 16.2-7. [Ref. 10.3]

Structures and Seismic Input to Structures and Equipment

- The seismic motion induced at the pipe supports in the structure is likely to be different from the ground motions. Since the various parts of the structures oscillate in different magnitudes and directions, the piping systems are essentially subjected to different excitations at each pipe anchor and restraint location. Therefore, amplified response spectra (ARS) for the maximum acceleration at various elevations throughout the structures are determined and the spectrum which is closest to and higher in elevation than the center of mass of piping is used. The JAF ARS curves are provided in

Attachment 3-3 (Vol. II), the design criteria for BOP piping stress and supports. [Ref. 10.8]

- The amplified response spectra were developed using the "Frequency Response Method", a modified Biggs method. A response spectrum is an envelope of the maximum accelerations of a damped single-degree-freedom system with variable frequencies due to the building motion at a specific elevation. The building motion at a specific elevation is approximated by a series of sinusoidal motions with the calculated building frequencies and their corresponding acceleration amplitudes at that elevation. Specifically, the amplified response spectra were developed for several selected elevations of each building for Operating Basis and Design Basis Earthquakes for an equipment (piping) damping value of 0.5% and 1.0%, respectively. [Ref. 10.8]

<u>Electrical Raceways</u>

- Reference 10.9 provides a standard for the routing of conduit and the selection of conduit supports. This standard applies to Nuclear Generation personnel, and to any organization which performs design of seismic electrical conduit and conduit supports. This applies to Safety Related, Augmented Quality, and Non-Safety Related structures, systems and components [Ref. 10.9].

<u>Seismic Interaction (spatial, fire, and flood)</u>

- The separation distance criteria between redundant electrical raceways at the time of JAF construction required separation of 3ft horizontally and 7ft vertically. Using this criteria will result in conservative separation distances for redundant circuits. JAF-RPT-ELEC-02075, Table 1 provides minimum allowable separation distances for redundant cables in General Plant Areas that may be used as alternate reduced separation distance criteria only for installations where the 3 feet horizontal / 7 foot vertical criteria stated in the UFSAR, Section 7.1.9 cannot be met [Ref. 10.10].

- A safety design basis for the Primary Containment and Reactor Vessel Isolation Control System is to ensure closure of Group A (communicate with the Reactor Vessel) and Group B (communicate with Primary Containment Free Space) automatic isolation valves is initiated, when required, with sufficient reliability. UFSAR, Section 7.3.2, states there is sufficient electrical and physical separation between trip channels monitoring the same essential variable to prevent environmental factors, electrical faults, and physical events, such as a fire, from impairing the ability of the system to respond correctly.

- The use of Generic Implementation Procedure (GIP) For Seismic Adequacy of Equipment and Parts, as modified and supplemented by the U.S. Nuclear Regulatory Commission Supplemental Safety Evaluation Report (SSER) No. 2 and SSER No. 3, may be used as an alternative method to existing methods for the seismic design and verification of existing, modified, new and replacement equipment and parts classified as Class 1. Only those portions of the GIP listed in "Use of Generic Implementation Procedure (GIP) for New and Replacement Equipment and Parts (NARE)" shall be used. The other portions of the GIP are not applicable since they contain administrative, licensing, and documentation information which is applicable only to the Unresolved Safety Issue (USI) A-46 program [Ref. 10.3, Section 12.5.6].

- Automatic water sprays, are provided in the Reactor Building at various area boundaries to isolate fire areas from each other. The water spray piping is seismically supported. [Ref. 10.3, Section 9.8.3.1.4]

  All fire protection water piping and mechanical equipment up to and including flow control valves in the Fire Protection Systems protecting Class I systems and components listed below are designed to QA Category QP criteria.

  1. High Pressure Coolant Injection Pump.
  2. Reactor Core Isolation Cooling Pump.
  3. Emergency Diesel-Generator Rooms.
  4. Diesel Driven Fire Pump.
  5. Standby Gas Treatment System Charcoal Filters.

  The fire protection piping for the Battery Room Corridor is seismically supported from the alarm check valve on manifold No. 7 through the sprinkler discharge piping. The system is QA class QP, seismically supported to prevent it from interfering with any safety related system or components in the Battery Room Corridor or Cable Tunnels.

3.0 SEISMIC WALKDOWN PROGRAM IMPLEMENTATION APPROACH

JAF has committed, per JAFP-12-0075 [Ref. 10.12], to conduct and document seismic walkdowns for resolution of NTTF Recommendation 2.3: Seismic in accordance with the EPRI Seismic Walkdown Guidance [Ref. 10.2]. The approach provided in the Guidance for addressing the actions and information requested in Enclosure 3 to the 50.54(f) Letter includes the following activities, the results of which are presented in the sections shown in parenthesis:

- Assignment of appropriately qualified personnel (Section 4.0)
- Reporting of actions taken to reduce or eliminate the seismic vulnerabilities identified by the Individual Plant Examination of External Events (IPEEE) program (Section 5.0)
- Selection of structures, systems and components (SSCs) to be evaluated (Section 6.0)
- Performance of the seismic walkdowns and area walk-bys (Section 7.0)
- Evaluation and treatment of potentially adverse seismic conditions with respect to the seismic licensing basis of the plant (Section 8.0)
- Performance of peer reviews (Section 9.0)

The coordination and conduct of these activities was initiated and tracked by Entergy corporate leadership, which provided guidance to each Entergy site throughout the seismic walkdown program, including JAF. Entergy contracted with an outside nuclear services company to provide engineering and project management resources to supplement and assist each individual site. JAF had dedicated engineering contractors, supported by their own project management and technical oversight, who worked closely with plant personnel.

4.0 PERSONNEL QUALIFICATIONS

The NTTF 2.3 Seismic Walkdown program involved the participation of numerous personnel with various different responsibilities. This section identifies the project team members and their project responsibilities and provides brief experience summaries for each. Training certificates of those qualified as Seismic Walkdown Engineers (SWEs) are included in Attachment H.

Table 4-1 summarizes the names and responsibilities of personnel used to conduct the seismic walkdowns. Experience summaries of each person follow.

Table 4-1

Name	Equipment Selection Personnel	Seismic Walkdown Engineer	Licensing Basis Reviewer	IPEEE Reviewer
Richard Casella (Entergy)	X	X	X	
Alan Porch (Entergy)		X		X
Jeffrey Cooney (Entergy)	X			
Yaroslav Losev (ENERCON)		X[2]	X	X
Pouria Pourghobadi (ENERCON)		X		
Donald Koberg (ARES)		X		
Harpreet Ghuman (ARES)		X		
Chris Sawatzke (Entergy)		X		
Bob Kester (Entergy)		X		
Roger Locy (Entergy)	X[1]			

Notes:

1. Plant operations representative
2. Designated lead SWE

<u>Richard Casella</u>

Mr. Casella graduated from the Pennsylvania State University with a Bachelor of Science degree in Civil Engineering in May 1976. Mr. Casella is a Registered Professional Engineer in the Commonwealth of Pennsylvania. Mr. Casella successfully completed the SQUG Walkdown Training course in June 2007. Mr. Casella's related experience is summarized below.

Mr. Casella has 36 years of experience in the Engineering Design of nuclear power plants. He spent 19 years at Stone and Webster Engineering Corporation (SWEC) associated with the design, startup, and operation of the Nine Mile Point 2 (NMP2) station. The first 7 years of his career were spent in the SWEC design office as a Pipe Stress analyst for ASME III Class 2, 3 and USAS B31.1 Class 4 piping systems. The remaining 12 years of his SWEC service were spent at NMP2 in Lycoming, NY. During this time, Mr. Casella was a Pipe Stress supervisor, then Lead Engineer for Pipe Stress and Supports, and then supported the transition of design responsibilities from SWEC to the Niagara Mohawk Power Corporation (NMPC). After plant startup, Rick worked as a Civil/Structural Design Engineer under NMPC authority until 1995.

Mr. Casella joined the New York Power Authority in October 1995 as a Civil/Structural Design Engineer at the James A. FitzPatrick (JAF) plant. His primary role in his 17 year tenure at JAF has been pipe stress. He has worked with ISI Class 1, 2, 3 and non-ISI piping. He has been involved with and has an understanding of the Mark I Containment work performed for JAF by the Teledyne Corporation. He has dealt with numerous seismic piping issues at JAF including many times assisting the Shift Manager with Operability determinations related to seismic piping and support issues. Mr. Casella has worked 9 Refuel Outages and several LCOs at JAF which have given him valuable "hands and eyes on" experience and knowledge of the plant and how it operates. Rick has also been associated with many plant modifications with seismic evaluations and calculations including Responsible Engineer for the replacement of 2-Stage Main Steam Safety Relief Valves with 3-Stage models. He is also experienced in Seismic Qualification of plant equipment and the Boiling Water Reactor Vessel Internals Program (BWR-VIP).

Alan Porch

Mr. Porch graduated from the Drexel University with a Bachelor of Science degree in Civil Engineering in June 1974. He is a Registered Professional Engineer in the State of New York. Al successfully completed the EPRI Training on Near Term Task Force Recommendation 2.3 – Plant Seismic Walkdowns in July 2012.

Mr. Porch has 34 years of experience in the Engineering Design of nuclear power plants. He spent 13 years at Stone and Webster Engineering Corp. (SWEC) associated with the design, startup support, and operation of the Nine Mile Point 2 (NMP2) Station. The first year of his career was spent in the SWEC design office as a structural design engineer performing steel and concrete design activities associated with the design of the Nine Mile Point NPP. The next 5 years he worked at the Fermi II NPP as a pipe support design engineer and the remaining 7 years of his SWEC service were spent at NMP2 in Lycoming, NY. During this time, Mr. Porch was a Pipe Support engineer and Modification engineer, and then supported the transition of design responsibilities from SWEC to the Niagara Mohawk Power Corporation (NMPC). After plant startup, Al worked as a Civil/Structural Design Engineer under NMPC authority until 1995.

Mr. Porch joined the New York Power Authority in September 1995 as a Civil/Structural Design Engineer at the James A. FitzPatrick (JAF) plant. His primary role in his 17 year tenure at JAF has been structural design support with a special attention given to pipe support design and acceptance. He has worked with ISI Class 1, 2, 3 and non-ISI piping systems. He has dealt with numerous seismic piping issues at JAF including assisting the Shift Manager with Operability determinations related to seismic piping and support issues. Mr. Porch has worked 9 Refuel Outages as well as forced outages and down powers and at JAF which have given him valuable "hands and eyes on" experience and knowledge of the plant and how it operates. Al has also been associated with many plant modifications with have included seismic evaluations and calculations including performing as Responsible Engineer for JAF's Pipe Support Program.

Jeffrey Cooney

Mr. Cooney is employed as a PSA Engineer for Entergy Nuclear Operations. He has been employed with the company over 4 years. His expertise is in Probabilistic Safety Assessment (PSA) which includes maintaining/updating the active site PSA model and ensuring that current industry standards, experience, and technology are incorporated appropriately into the model.

Yaroslav Losev

Mr. Losev graduated from the New Jersey Institute of Technology with a Bachelor of Science degree in Mechanical Engineering in June 2008. He has worked as an ENERCON Mechanical Engineering for three years. For the past 2 years, he has been working in the Structural Engineering department. Mr. Losev has successfully completed Training on Near Term Task Force Recommendation 2.3 Plant Walkdowns in 09/13/2012. Some of Mr. Losev's related experience has been summarized below.

As part of Exelon's ongoing commitments to comply with the Nuclear Regulatory Commission (NRC) requirements for post-fire safe shutdown promulgated in 10CFR50 certain scenarios have been identified by the Exelon Expert Panel that are related to the safe shutdown of Limerick Generating Station (LGS). Plant design changes are required to address issues related to Multiple Spurious Operations (MSOs) as outlined in Nuclear Energy Institute (NEI) 00-01, Rev. 2 (Guidance for Post-Fire Safe Shutdown Circuit Analysis) and ensure compliance with NRC Regulatory Guide 1.189, Rev 2 (Fire Protection for Nuclear Power Plants). Mr. Losev prepared technical evaluations and revised existing plant calculations on the capacity of existing raceway supports to support the additional dead weight load of the fire barrier systems including seismic requirements and considerations. He also developed technical evaluations for seismic temporary supports in number of other locations at LGS.

Engineering change package implementation in Indian Point (IPEC), Units 2 and 3, coating of the walls of the transformer moats for Main Transformers. During the stone removal to install coating, various existing non-safety related conduits and pipes had to be temporarily supported as part of this work activity. The supports were temporarily attached to existing structural steel and the steel columns, beams and foundations which were evaluated for the additional load. Mr. Losev designed and evaluated 27 uniquely different temporary supports to be installed while the stone was removed. Mr. Losev demonstrated extreme flexibility and imagination in designing these supports. He also showed technical rigor, fast adaptation to new designs into evaluation, clear client communications, and timely deliverable of the calculation despite fast track schedule of this project.

Mr. Losev provided support in walkdowns and design inputs for Honeywell at Metropolis Works (MTW) UF6 Processing Facility in determination and suggestions of seismic supports for their piping systems and equipment. He reviewed seismic calculations on equipment and provided conceptual designs for supports on the equipment and piping runs to meet NRC's requirements.

Mr. Losev provided mechanical/structural engineering support for American Electrical Power (AEP) D.C. Cook Generating Station on "Pipe Stress Analysis" re-evaluation. As part of D.C. Cook Generating Station's Large Bore Pipe Reconciliation Project (LBPRP), numerous safety related pipe stress calculations had to be re-evaluated. The objective of these calculations was to structurally qualify the piping, pipe supports, including integral welded attachments, penetrations/nozzles, and valve accelerations in accordance with the design limits for dead weight, thermal, flow transients, and seismic conditions, and to provide the technical basis for any recommended modifications to the system that would be required to meet the D.C. Cook Generating Station's acceptance criteria.

Pouria Pourghobadi

Mr. Pourghobadi has worked as an ENERCON Civil/Structural Engineer for the past year. Mr. Pourghobadi has successfully completed Training on Near Term Task Force Recommendation 2.3 Plant Walkdowns in 09/13/2012. Some of Mr. Pourghobadi's related experience has been summarized below.

As part of commitment to NRC, the Zion Solutions contracted ENERCON to provide Architectural and Engineering (A/E) Services for the design of an Independent Spent Fuel Storage Installation (ISFSI), various Fuel Handling Building modifications and operations. Mr. Pourghobadi reviewed calculations and drawings for a new set-down pad with a loaded MAGNASTOR Transfer CASK (MTC) in the lower level of the Reactor Building cavity floor and evaluation of the capacity of existing floor slab to support the additional dead weight load with seismic requirements and considerations of the loaded MTC.

Mr. Pourghobadi provided civil/structural engineering support to Exelon Peach Bottom Atomic Power Station (PBAPS) to design a 75 ft high lighting arrestor as part of an Engineering Change Package. Site unique topography dictated adoption of a more creative approach rather than the conventional methods of mast foundation design. Mr. Pourghobadi provided design inputs for development of the foundation and evaluated the dead load of the lighting arrestor onto the foundation design.

Donald Koberg

Mr. Koberg earned a Bachelor's degree from Washington State University in Mechanical Engineering in 2010. He has been working as a Mechanical Engineer at ARES Corporation for over 2 years. Throughout his two plus years at ARES Corporation he has performed many technical calculations relating to anchorage and support design of piping systems for seismic activities. Mr. Koberg has successfully completed Training on Near Term Task Force Recommendation 2.3 Plant Walkdowns on 07/26/2012. Some of Mr. Koberg's related experience is summarized below.

Mr. Koberg's experience with anchorage design consists of designing and analyzing anchorage of piping support system to ASME B31.3 requirements. Tasks included selection of material, support configuration, and general layout design of the pipe supports for stainless steel piping for use in waste retrieval activities at the Hanford Nuclear Reservation, Hanford, WA. Mr. Koberg has also analyzed various systems, structures and components for adherence to ASME B31.3, "Process Piping".

Mr. Koberg's activities include design and analysis of waste transfer piping systems including assisting on equipment design and system analysis. As part of the ASME B31.3 analyses, Mr. Koberg as analyzed multiple sections of piping systems and their supports for structural adequacy during seismic events. Analyzed equipment includes pump assemblies, waste distribution assemblies, and stainless steel piping assemblies.

Harpreet Ghuman

Mr. Ghuman earned a Bachelor's degree from Washington State University in Civil Engineering in 2008. He has been working as a Structural Engineer at ARES Corporation for the past 4 years. Throughout his four years at ARES Corporation he has performed many technical calculations relating to anchorage design, footing/slab design, and pipe support design for seismic activities. Mr. Ghuman has successfully completed Training on Near Term Task Force Recommendation 2.3 Plant Walkdowns on 07/26/2012. Some of Mr. Ghuman's related experience is summarized below.

Mr. Ghuman's experience in anchorage design consists of designing expansion or cast in place threaded rod/headed anchors for placement within concrete of various thicknesses and edge distance constraints in accordance with ACI-318, Appendix D. Also included within anchorage design is the design of at grade or embedded base plates. Mr. Ghuman also has experience in designing welds for mechanical supports. Mr. Ghuman's footing/slab design experience consists of designing the appropriate size concrete foundation including rebar for various mechanical supports.

Mr. Ghuman has experience in Near Term Task Force Recommendation 2.3 Plant Walkdowns. He was part of a team that performed these walkdowns for Duke Energy at the McGuire Nuclear Station Units A and B near Huntersville, NC from August 27, 2012 thru September 13, 2012. Mr. Ghuman's responsibilities during these walkdowns consisted of aiding in filling out the Area Walk-by (AWC) and Seismic Walkdown Checklists (SWC) for various areas and equipment within the plant. He also assisted in the preparation of packages, such as finding drawings/calculations that pertained to equipment on the Seismic Walkdown Equipment List (SWEL), and determined which components should be considered as part of the required 50% design verification components in accordance with EPRI Report 1025286.

Mr. Ghuman has worked in Hanford, WA on the contaminated groundwater in the 100 areas that reactor sites were required to be treated then pumped back into the river basin. Mr. Ghuman's task was for these projects was to design pipe supports for the piping line to allow for safe distribution of contaminated water during seismic or wind events. The design included, fabricating members from structural steel which includes weld and bolt design or constructing pipes supports out of UNISTRUT members. Mr. Ghuman also designed concrete foundations and sized the appropriate expansion or cast in place anchors for the pipe supports or various other mechanical equipment.

Chris Sawatzke

Mr. Sawatzke graduated from Michigan State University with a Bachelor of Science degree in Civil Engineering in September 1981. He has an Engineer-In-Training (EIT) in the State of Michigan. Chris successfully completed the EPRI Training on Near Term Task Force Recommendation 2.3 – Plant Seismic Walkdowns in July 2012. He has also successfully completed EPRI Training on Visual Examination Level II – Containment Inspection Program in September 2005 and the EPRI Comprehensive Coating Training in April 2002.

Mr. Sawatzke has 31 years of experience in the Engineering Design of nuclear power plants. He spent 13 years with Niagara Mohawk associated with the design and operation of the Nine Mile Point – Unit 2 Nuclear Plant. The first seven years of his career were spent working for Gilbert/Commonwealth at Washington Power Unit 2, Perry Nuclear Plant Unit 1, Browns Ferry Nuclear Power Plant and Sequoyah Nuclear Power Plant; and Nuclear Power Services (NPS) at South Texas Project Nuclear Plant. During this time, Mr. Sawatzke was a Design Engineer supporting the Civil/Structural Engineering Department performing steel and concrete design activities associated with the design of each of the specific nuclear power plants.

Mr. Sawatzke joined Entergy Nuclear Operations in October 2001 as a Senior Civil/Structural Design Engineer at the James A. FitzPatrick (JAF) Nuclear Power Plant. His role in his 11 year tenure at JAF has been structural steel and concrete design for Systems, Structures and Component's (SSC's). He has worked with ISI Class 1, 2, 3 and Non-ISI piping systems. He has dealt with numerous seismic piping and structural issues at JAF including assisting the plant Shift Manager with Operability Determinations related to seismic piping and pipe support issues. Mr. Sawatzke has worked 6 Refuel Outages as well as forced outages and down powers at JAF, 5 refuel outages at other Entergy Fleet plants, and 7 refuel outages as well as forced outages at Nine Mile Nuclear Plant Unit 2; which has provided him valuable "hands on" experience and knowledge of the various plants and systems and how they operate. Mr. Sawatzke has also been associated with many plant modifications as a Responsible Engineer which included seismic evaluations and formal calculations for the JAF Pipe Support Program.

<u>Bob Kester</u>

Mr. Kester graduated from Lafayette College with a Bachelor of Science degree in Civil Engineering in May 1980. Mr. Kester successfully completed the SQUG Walkdown Training course in August 1993, and performed SQUG USI A-46 walkdowns for James A. Fitzpatrick Nuclear Plant in 1995.

Mr. Kester has 32 years of experience in the Engineering Design of nuclear power plants. He spent 10 years at Stone and Webster Engineering Corporation (SWEC) associated with the design of the River Bend and Nine Mile Point 2 (NMP2) stations. During this period, Bob's experience was primarily associated with pipe stress and support design, and included over 8 years as a field engineering, which provided valuable experience that integrated aspects of design criteria, design changes, construction and inspection requirements.

Mr. Kester's career shifted to an operating nuclear power station, working for the utility company at the James A. FitzPatrick (JAF) plant since December 1989. In addition to being involved in Plant Modification designs and the JAF SQUG program, Bob has had diverse experiences in civil, structural, and mechanical engineering disciplines as a Plant Engineer. This role has often required a practical approach to seismic evaluations in support of Operability determinations related to plant condition reports. Mr. Kester has worked numerous plant Refuel Outages and system LCOs at JAF which have given him valuable "hands and eyes on" experience and knowledge of the physical plant, and how it operated and maintained. Bob's involvement with numerous plant modifications has included seismic evaluations for structures, piping, tubing, raceways, and miscellaneous equipment, which has entailed formal design calculations, simplified qualitative evaluations, and also the use of EPRI's GIP & STERI methodology. For over 15 years in a Plant Support Engineering group, Bob has been the primary responsible engineer at JAF for structural evaluations of temporary conditions in the plant including scaffolding, shielding, leak repairs, freeze seals, as well as staging & storage of transient equipment.

<u>Roger Locy</u>

Mr. Locy's education and training is summarized as follows: Machinist Mate "A" School, U.S. Navy, completed May 1966. Basic Nuclear Power School, U.S. Navy, completed April 1967.

Naval Reactor Prototype, U.S. Navy, completed October 1967. BWR Technology, General Electric, completed December 1972. Site training programs, Duane Arnold Energy Center. Numerous courses for RO cold license and requalification. License number OP-3424. Site training programs, Duane Arnold Energy Center. Numerous courses for SRO license and requalification. License Number SOP-2849. Site training programs, James A. FitzPatrick Nuclear Power Plant. Numerous courses for SRO license and requalification. License number SOP-3218. Regents College, The University of the State of New York. Presently have earned 110 credits toward a Baccalaureate Degree in Nuclear Technology.

From November 1967 to April 1972, U.S.S. Enterprise-Nuclear Powered Aircraft Carrier. From May 1972 to July 1977, Operations Department, Duane Arnold Energy Center. From August 1977 to February 1982, Shift Supervisor, James A. FitzPatrick Nuclear Power Plant. From March 1982 to March 1985, Waste Management General Supervisor, James A. FitzPatrick Nuclear Power Plant. Established Decontamination and Shipping section of Radiological Waste Group in the Operations Department. Responsible for operation of the Radwaste Facility, all Radwaste shipments for disposal and area/equipment decontamination. From April 1985 to March 1989, Assistant Operations Manager, James A. FitzPatrick Nuclear Power Plant. Assisted the Operations Manager with the day to day operations of the plant. From April 1989 to March 1997, Operations Manager, James A. FitzPatrick Nuclear Power Plant. Responsible for the safe and efficient operation of the plant. Provide management over view of operating shifts, operation support and Radwaste Facility operation. Held a SRO license. From March 1997 to November 2000, Training Manager, James A. FitzPatrick Nuclear Power Plant. Responsible for the design, development, implementation and evaluation of training programs ensuring regulatory compliance, cost effectiveness and plant staff qualification. From November 2000 to June 2006, Project Manager Operations Support, FitzPatrick Nuclear Power Plant, responsible for operations input to outage planning, maintenance rule operations representative, BWROG Scram Frequency Reduction Committee representative, perform root cause analysis for department events and operations department training coordinator. From September 2006 to April 2007, Project Manager Operations Training Improvement Program, Ginna Nuclear Power Station. Responsible for the completion of the Operations Training Excellence Plan completion. Monitored both quality and timeliness of action close out. Oversight of a Operation Lesson Plan Upgrade Program. Supervised 5 contract lesson plan developers. From October 2007 to August 2009, Operations Procedure Group Lead, Nine Mile Point Nuclear Station. Responsible for the development and maintenance of all Operations Procedures for both units. Oversight of the WordPerfect to Word conversion of all site procedures. Revised procedures to support outage activities and modifications. From November 2009 to Present, License Renewal Project Senior Project Manager, James A. FitzPatrick Nuclear Power Plant. Responsible for identification, performance and documentation of One-Time Inspections. Assisted with monitoring progress of completion of NRC License Renewal Commitments.

4.1 EQUIPMENT SELECTION PERSONNEL

A total of 3 individuals served as Equipment Selection Personnel – see Table 4-1.

4.2 SEISMIC WALKDOWN ENGINEERS

A total of 8 individuals served as Seismic Walkdown Engineers – see Table 4-1.

4.3 LICENSING BASIS REVIEWERS

A total of 2 individuals served as Licensing Basis Reviewers – see Table 4-1.

4.4 IPEEE REVIEWERS

A total of 2 individuals served as IPEEE Reviewers – see Table 4-1.

4.5 PEER REVIEW TEAM

Table 4-2 summarizes the names and responsibilities of personnel used to conduct peer reviews of the seismic walkdown program. Experience summaries of each person follow.

Table 4-2

Name	SWEL Peer Reviewer	Walkdown Peer Reviewer	Licensing Basis Peer Reviewer	Submittal Report Peer Reviewer
Tom Panayotidi (ENERCON)		X[2]		X[1,2]
Alan Porch (Entergy)	X		X	
Richard Sullivan (Entergy)	X[2]			
Laura Maclay (ENERCON)		X		X
Jeffrey Horton (ENERCON)			X[2]	
Richard Casella (Entergy)				X

Notes:

1. Peer Review Team Leader
2. Lead peer reviewer of particular activity

Tom Panayotidi

Dr. Panayotidi has graduated with a Doctorate of Engineering Science in Civil Engineering/Engineering Mechanics, with emphasis in finite element analysis, particularly for seismic and other dynamic loads. He has worked as an ENERCON Civil/Structural Consulting Engineer for the past year, and has successfully completed Training on Near Term Task Force Recommendation 2.3 Plant Walkdowns on 09/13/2012. Dr. Panayotidi has over 30 years' experience as a Structural/Seismic Engineer in the nuclear field. Some of his related experience is summarized below.

Dr. Panayotidi prepared submittal report for OPPD Fort Calhoun Station per NTTF Recommendation 2.3: Seismic. He reviewed calculations and drawings for the OPPD Fort Calhoun Station Flood Recovery and Geotechnical/Seismic Evaluation.

Dr. Panayotidi also has experience in Standard Plant Design of Nuclear Island for Mitsubishi Heavy Industries – US Advanced Pressurized Water Reactor: Seismic Soil-Structure Interaction Analysis of Reactor Building Complex, Foundation Stability for sliding, overturning, bearing pressure (uplift condition), shear key design, nonlinear transient displacement calculation to predict foundation sliding, and Slope stability under seismic loading.

Dr. Panayotidi also has experience in Standard Plant Design of new generation compact 125 MW nuclear station for B&W mPower Project: Seismic Soil-Structure-Interaction analysis of underground (buried) nuclear island, he development of ground motion synthetic time history from high frequency CEUS design spectrum, as well as NRC 1.60 spectrum, and generation of in-structure-response-spectra (ISRS).

Dr. Panayotidi performed evaluation for NPPD Cooper Nuclear Station. He provided analytical review of the Reactor Building Crane Upgrade: Re-rate analysis of Cooper Nuclear Station Reactor Building, due to an increase in refueling crane capacity. He also evaluated Reactor building integrity for all applicable loads, including earthquake, tornado, seismic, and crane lifted loads.

Dr. Panayotidi developed worked on Accelerator Production of Tritium, DOE, for Savannah River Site: Seismic analysis of reinforced concrete building, including 3-D soil-structure interaction effects due to 60ft embedment, using SASSI. He also performed calculation of strained (iterated) soil properties, convolution and de-convolution of input motion using SHAKE91. Dr. Panayotidi also performed seismic anchor motion and soil-structure-interaction analysis of 1-mile long underground accelerator tunnel.

Dr. Panayotidi worked on design of 250 MW single-shaft, in-line gas turbine/steam turbine/generator concrete pedestal for River Road Generating Project (WA), including design of batter and vertical foundation piles, steel framing to support hot/cold piping in generation building.

<u>Richard Sullivan</u>

Mr. Sullivan graduated from University of Tennessee with a Bachelor of Science degree in Electrical Engineering. Some of Mr. Sullivan's related experience and awards are summarized below.

Navy Achievement Medal for superior management of the ship's Quality Assurance Program during two heavy maintenance periods.

Extensive experience in the operation and maintenance of a variety mechanical and electrical systems including: steam systems, cooling water systems, hydraulic systems, atmospheric controls, ventilation systems, electrical distribution, digital and logic systems, and electrical generation.

From January, 2001 to September, 2007 and from October, 2007 to present, in James A. FitzPatrick Nuclear Power Plant, Mr. Sullivan coordinated and developed Operations schedule activities and tagout preparation. Contributed to INPO 1 rating in 2004 by outstanding simulator scenarios and professional shift operations. Aided in the development, implementation, and enforcement of high operational standards. Contributed to the record breaking capacity factor year at JAF in 2001. Developed plant start up procedures in a flow chart format on own initiative. From May, 1997 to December, 2000, Mr. Sullivan developed and implemented three site-wide Emergency Plan Drills which included scenario design and simulator interface. Coordinated with other departments in the development of plant Emergency Operating Procedures and Severe Accident Procedure. Developed JAF licensed operator annual requalification examination. From November, 1994 to April, 1997, Mr. Sullivan assisted in two error-free refueling outages as Refuel Floor SRO. In October, 1994 Mr. Sullivan earned Senior Reactor Operator License from the Nuclear Regulatory Commission.

From 1988 to February, 1992, in the United States Navy, Mr. Sullivan served on board USS Nevada (SSBN-733) as Tactical Systems Department Head, Strategic Missiles Officer, and Damage Controls Assistant. Quality Assurance Officer: Coordinated ship's force and shipyard maintenance on various systems on a Trident submarine ensuring all specifications were met. Also responsible for training 30 personnel ship-wide.

<u>Laura Maclay</u>

Ms. Maclay has over five years of experience as a structural engineer, three years with ENERCON Services. Ms. Maclay holds a Bachelor's degree in Structural Engineering from Drexel University and is a qualified Seismic Walkdown Engineer as stated on her EPRI training certificate dated July 26, 2012. Her tasks have ranged from assisting with the development and preparation of design change packages to performing design calculations and markups, comment resolutions, and drawing revisions. Ms. Maclay spent a year on site at Turkey Point Nuclear Plant preparing structural evaluations of SSC's for an Extended Power Uprate (EPU). Her work included designing safety related supports for computer and electrical equipment for the Turbine Digital Controls Upgrade package and other similar packages. Ms. Maclay's responsibilities also included the review of calculations, drawings and vendor documentation for the seismic evaluation of the Unit 3 Palfinger Crane inside containment and new platforms in the High Pressure Turbine enclosure.

Recent work includes Fukushima flooding walkdowns at Limerick Generating Station and seismic walkdowns at Plant Farley. As a member of a two person team, Ms. Maclay was responsible for evaluating equipment anchorage, spatial interactions and potentially adverse conditions.

Jeffrey Horton

Mr. Horton is a Licensed Professional Engineer with 39 years of experience in the structural design of nuclear power components, pipe systems and building structures. Mr. Horton is currently employed as a Lead Civil/Structural Engineer. He holds a B.S. in Aerospace Engineer from Park's College of St Louis University in Missouri and a M.S. in Material Science specializing in Solid Mechanics from Rutgers University in New Jersey. He is a qualified SWE with extensive experience in the seismic design of components and pipe systems in Nuclear power plants.

5.0 IPEEE VULNERABILITIES REPORTING

During the IPEEE program in response to NRC Generic Letter 88-20 [Ref. 10.4], plant-specific seismic vulnerabilities were identified at many plants. In this context, "vulnerabilities" refers to conditions found during the IPEEE program related to seismic anomalies, outliers, or other findings.

IPEEE Reviewers (see Section 4.4) reviewed the IPEEE final report [Ref. 10.5] and supporting documentation to identify items determined to present a seismic vulnerability by the IPEEE program. IPEEE Reviewers then reviewed additional plant documentation to identify the eventual resolutions to those seismic vulnerabilities not resolved by the completion of the IPEEE program.

The seismic vulnerabilities identified for JAF during the IPEEE program are reported in Attachment A. A total of 1 seismic vulnerability was identified by the JAF IPEEE program. For the identified seismic vulnerability, the table in Attachment A includes three pieces of information requested by Enclosure 3 of the 50.54(f) Letter:

- a description of the action taken to eliminate or reduce the seismic vulnerability
- whether the configuration management program has maintained the IPEEE action (including procedural changes) such that the vulnerability continues to be addressed
- date when the resolution actions were completed.

The list of IPEEE vulnerabilities provided in Attachment A was used to ensure that some equipment enhanced as a result of the IPEEE program were included in SWEL1 (see Section 6.1.2). Documents describing these equipment enhancements and other modifications initiated by identification of IPEEE vulnerabilities were available and provided to the SWEs during the NTTF 2.3 Seismic Walkdowns.

6.0 SEISMIC WALKDOWN EQUIPMENT LIST (SWEL) DEVELOPMENT

This section summarizes the process used to select the SSCs that were included in the SWEL in accordance with Section 3 of the Guidance. A team of equipment selection personnel with extensive knowledge of plant systems and components was selected to develop the SWEL. The SWEL is comprised of two groups of items:

- SWEL 1 consists of a sample of equipment related to safe shutdown of the reactor and maintain containment integrity (five safety functions)
- SWEL 2 consists of items related to the spent fuel pool

The final SWEL is the combination of SWEL1 and SWEL2. The development of these two groups is described in the following sections.

6.1 SAMPLE OF REQUIRED ITEMS FOR THE FIVE SAFETY FUNCTIONS

Safe shutdown of the reactor involves four safety functions:

- Reactor reactivity control (RRC)
- Reactor coolant pressure control (RCPC)
- Reactor coolant inventory control (RCIC)
- Decay heat removal (DHR)

Maintaining containment integrity is a fifth safety function

- Containment function (CF)

The overall process for developing a sample of equipment to support these five safety functions is summarized in Figure 1-1 of the Guidance. The equipment coming out of Screen #3 and entering Screen #4 is defined as Base List 1. The equipment coming out of Screen #4 is the first Seismic Walkdown Equipment List, or SWEL 1. Development of these lists is described separately in the following sections.

6.1.1 Base List 1

Based on Figure 1-1 and Section 3 of the Guidance, Base List 1 represents a set of Seismic Category (SC) I equipment or systems that support the five safety functions. The IPEEE program was intended to address the seismic margin of SSCs associated with each of the five safety functions. At JAF the EPRI Seismic Margin Assessment (EPRI SMA) method was used to complete the seismic IPEEE program, based on EPRI Report NP-6041 titled "A Methodology for assessment of Nuclear Power Plant Seismic Margin." Base List 1 was developed using both IPEEE report [Ref. 10.5] and A-46 Safe Shutdown Equipment List (SSEL) [Ref. 10.13]. This equipment list of SSCs is consistent with the requirements of Screens #1 through #3 of the Guidance. Therefore, the components listed on both the USI-46 composite SSEL and the IPEEE Shutdown Equipment List are initially used as the NTTF 2.3 Seismic Walkdown Base List 1. Base List 1 is presented as Table 9.4.1 in Attachment B, and has 699 total items. The following components were added to both Base List 1 and SWEL 1.

- o Core Spray Pump 14P-1A and RHR Service Water Strainer 10S-5A (both on the IPEEE, but not on A-46 Safe Shutdown Equipment List (SSEL))
- o Standby Gas Treatment Filter Train A Inlet Isolation valve 01-125MOV-14A (listed on the IPEEE, but not on A-46 SSEL)
- o Administrative Building Ventilation Control Panel, 72HV-7A (although this is Safety Related component, it is not listed on either the IPEEE or A-46 SSEL)
- o Emergency Diesel Generator A Air Start Compressor A1, 93AC-A1 (this is an Augmented Quality component and is the only component in compressor equipment class. It is not listed on either the IPEEE or A-46 SSEL)
- o Reactor Core Isolation Cooling Pump, 13P-1 (this is an Augmented Quality component, which is included on the IPEEE report)
- o Components 71ACUPS and 71PT-71ACUPS are currently classified as Non-Safety Related (per Equipment Database), these two components were on the original A-46 SSEL and support at least one of the 5 Safety functions.

The following components where replaced in SWEL 1 and are not currently shown on the Base List 1.

- o SGT Filter Train A Inlet Isolation Valve 01-125MOV-14A replaced with SGT Filter Train B Inlet Isolation Valve 01-125MOV-14B due to component restrictions.

6.1.2 SWEL 1

Based on Figure 1-1 and Section 3 of the Guidance, SWEL 1 is a broad population of items on Base List 1 including representative items from some of the variations within each of five sample selection attributes. The selection of SWEL 1 items includes consideration of the importance of the contribution to risk for the SSCs. Equipment Selection Personnel (see Section 4.1) developed SWEL 1 using an iterative process. The following paragraphs describe how the equipment selected for inclusion on the final SWEL 1 are representative with respect to each of the five sample selection attributes while considering risk significance. In general, preference for inclusion on

SWEL 1 was given to items that are accessible and have visible anchorage. SWEL 1 is presented as Table 9.4.2 as in Attachment B, and has 117 total items.

Variety of Types of Systems
Items were selected from Base List 1 ensuring that each of the five safety functions was well represented. Additionally, components from a variety of frontline and support systems, as listed in Appendix E of the Guidance, were selected. The system type of each item on SWEL 1 is listed on Table 9.4.2 of Attachment B.

Major New and Replacement Equipment
With assistance from plant operations, equipment selection personnel identified items on Base List 1 which are either major new or replacement equipment installed within the past 15 years, or have been modified or upgraded recently. These items are designated as such on Base List 1 on Table 9.4.1 of Attachment B. A robust sampling of these items is represented on SWEL 1. The following components were chosen as items that have been replaced since completion of the original SSEL.

- 02RV-71E - Main-Steam Safety Relief Valve
- 71INV-3A - "A" LPCI Inverter
- 70RWC-2A (CND) - "A" Control Room Chiller Condenser
- 23MOV-14 - HPCI Turbine Steam Supply Isolation Valve
- 23AOV-53 - HPCI Turbine Steam Supply Drain Trap T-3 Bypass Valve
- 10S-5A - RHRSW Strainer
- 71SB-2 - "B" Station Battery

Variety of Equipment Types
Items were selected from Base List 1 ensuring that each of the equipment classes represented there was also represented on SWEL 1, in the same approximate ratios. The different equipment classes considered are listed in Appendix B of the Guidance. The equipment class of each item on SWEL 1 is listed on Table 9.4.2 of Attachment B. Note that SWEL 1 does not include Class 13 components, because these are not represented on Base List 1. A single Class 12 component (93AC-A1) is included on the SWEL. Although it is only classified as Augmented Quality, there are no compressors designated as Safety Related at JAF.

Variety of Environments
Items were selected from Base List 1 located in a variety of buildings, rooms, and elevations. These item locations included environments that were both inside and

outside, as well as having high temperature and/or elevated humidity. The location and environment of each item on SWEL 1 is listed on Table 9.4.2 of Attachment B.

IPEEE Enhancements

The IPEEE does not include any specific vulnerabilities for components which could be considered for the SWEL 1 (see Section 5.0). However, the following components were chosen based on their "lower" seismic capabilities. Note that the bottom 3 listed components are associated with seismic induced anchorage failure at ground accelerations between 0.31g and 0.41g:

10E-2A	"A" RHR Heat Exchanger
10S-5A	"A" RHRSW Strainer
09-32	Channel "A" RHR/RCIC Relay Panel
71MCC-161	600V Motor Control Center (Bus 116100)
71DSC-11561	L15 Unit Substation Transformer T-13 High Side Disch SW

Risk Significance

Information from the plant Probabilistic Risk Assessment (PRA) model and the Maintenance Rule implementation documentation were used to determine whether items were risk significant. Risk significance was determined by using Risk Importance Measures, Risk Achievement Worth (RAW), and Fussell-Vesely (FV). This risk was considered using a threshold value of RAW ≥ 2 and FV ≥ 0.0001. Higher risk components were given added consideration for selection as a SWEL 1 item.

6.2 SPENT FUEL POOL ITEMS

The overall process for developing a sample of SSCs associated with the spent fuel pool (SFP) is summarized in Figure 1-2 of the Guidance. The equipment coming out of Screen #2 and entering Screen #3 is defined as Base List 2. The equipment coming out of Screen #4 is the equipment that could potentially cause the SFP to drain rapidly. The equipment coming out of Screen #3 and Screen #4 is the second Seismic Walkdown Equipment List, or SWEL 2. Development of these lists is described separately in the following sections.

6.2.1 Base List 2

Based on Figure 1-2 and Section 3 of the Guidance, Base List 2 represents the Seismic Category (SC) I equipment or systems associated with the SFP. To develop Base List 2, Equipment Selection Personnel (see Section 4.1) reviewed plant design and licensing basis documentation and plant drawings for the SFP and its associated cooling system. Base List 2 is presented as Table 9.4.3 in Attachment B, and has 13 total items. The following components were replaced in SWEL 2 and are not currently shown on the Base List.

- o Decay Heat Removal Strainer Bypass Valve 32DHR-5 replaced with Decay Heat removal Cooling Water Return Isolation Valve 32DHR-18 because of radiological considerations.
- o Decay Heat Removal SFP Water Primary Pump A 32P-1A was removed from the list due to radiological reasons.

6.2.2 Rapid Drain-Down

Rapid drain-down is defined in EN-DC-168, Attachment 9.4 [Ref. 10.11], as lowering the water level to the top of the fuel assemblies within 72 hours after an earthquake. Consistent with the Guidance, the Equipment Selection Personnel (see Section 4.1) identified SSCs that could cause the SFP to drain rapidly by first reviewing the SFP documentation to identify penetrations below about 10 ft above the top of the fuel assemblies.

Because this review found no such SFP penetrations, there is no potential for rapid drain-down and no items were included on the rapid drain-down list to include on SWEL 2.

6.2.3 SWEL 2

Based on Figure 1-2 and Section 3 of the Guidance, SWEL 2 is a broad population of items on Base List 2 including representative items from some of the variations within each of four sample selection attributes (using sample process similar to SWEL 1), plus each item that could potentially cause rapid-drain down of the SFP. Due to the population of items on Base List 2 being much smaller than Base List 1, the sampling attributes are satisfied differently for SWEL 2 than for SWEL 1. The following paragraphs describe how the equipment selected from Base List 2 for inclusion on SWEL 2 are representative with respect to each of the four sample selection attributes. SWEL 2 is presented as Table 9.4.5 in Attachment B, and has 11 total items. The SFP at JAF has no qualified rapid drain-down (RDD) components, as described in EN-DC-168, Attachment 9.4 [Ref. 10.11]; therefore no RDDs were included in SWEL 2 list.

<u>Variety of Types of Systems</u>
There are 2 systems, Spent Fuel Cooling and Decay Heat Removal, associated with SFP. Each of these systems is represented on SWEL 2.

Major New and Replacement Equipment

New and Replaced components are identified in Table 9.4.5, Column "N/R". Out of 11 SWEL 2 components, 9 were not replaced within 15 years; therefore considered to be new, and 2 components are newly installed.

Variety of Equipment Types

There are 5 different equipment classes represented on Base List 2: 01, 05, 07, 20 and 21. Each of these equipment classes is represented on SWEL 2.

Variety of Environments

10 out of 11 SFP components noted on SWEL 2 are inside the Reactor Building and are thus located in similar environments. The remaining component is located outside.

6.3 DEFERRED, INACCESSIBLE ITEMS on SWEL

The intent of adding each item on the SWEL is for it to be walked down as part of the NTTF 2.3 Seismic Walkdown program. To be able to perform the seismic walkdowns of these items, it is necessary to have access to them and to be able to view their anchorage. In some cases, it was not feasible to gain access to the equipment or view its anchorage during the entire 180-day response period of Enclosure 3 to the 50.54(f) Letter. For these cases, walkdowns of these items have been deferred until the next refueling outage (RFO) in September of 2014, or the items were deleted from the list. An updated submittal report incorporating these deferred walkdowns will be provided 90 days after the end of RFO21.

Deferred and deleted items are summarized in the table below. The reason is identified as either ACC (indicating that the item was deleted because of ALARA reasons) or CAB (indicating that the item requires opening cabinet/panel doors which was not permitted by plant Operations personnel during the walkdown period, due to being energized or otherwise). A total of 26 items from which 23 items are deferred, 3 are in high dose areas and will not be deferred. The 23 deferred items are cabinets/panels required to be opened.

SWEL#	Equipment ID	Description	Location	Reason
SWEL 1-163	12MOV-18	RWCU supply outbound isolation valve, is located in a Locked High Radiation Area (LHRA). Dose levels are high, both in outage and normal plant operation. This item is deleted from the SWEL list because of dose concern and will not be deferred.	RB. EL 300', Column 3, Line R	ACC

SWEL#	Equipment ID	Description	Location	Reason
SWEL 2-2	19FPC-32	The skimmer surge tank A condensate make-up check valve, is located in a LHRA behind the "A" Skimmer Surge Tank. This item is deleted from the SWEL list because of dose concern and will not be deferred.	RB. EL 369.6', Column 3.5, Line Y	ACC
SWEL 2-11	32P-1A	The decay heat removal SFP water primary pump "A", is located in a high radiological area. This item is deleted from the SWEL list because of dose concern and will not be deferred.	RB. 326', Column 3, Line T	ACC
SWEL1-52	09-3	The nuclear station main control board, is deferred. WO # 52389703 was initiated to track the walkdown of this component on 12/2012.	AD. 300', Column 10, Line F	CAB
SWEL1-430	71-10502	The 4160V switchgear distribution (BUS 10500) is deferred. WR # 309411 was initiated to track the walkdown of this component on 09/2013.	EG. 272', Column 24, Line A1	CAB
SWEL1-433	71-10560	The 4160V switchgear distribution (BUS 10500) is deferred. WO # 52448178 was initiated to track the walkdown of this component on 06/2014.	EG. 272', Column 26, Line A1	CAB
SWEL1-438	71-11502	The 600V switchgear distribution (bus 11500) breaker 02 is deferred. WO # 52450763 was initiated to track the walkdown of this component on 06/2014.	RB. 300', Column 26, Line A1	CAB
SWEL1-439	71-11602	The 4160V switchgear distribution (BUS 10600) is deferred. WR # 290278 was initiated to track the walkdown of this component.	EG. 272', Column 2, Line R	CAB
SWEL1-446	71BAT-3A	The LPCI inverter battery is deferred. WO # 52437751 was initiated to track the walkdown of this component on 08/2013.	RB. 344.6', Column 5.5	CAB
SWEL1-448	71BC-1A	The 125 VDC station battery charger is deferred. WO # 52440826 was initiated to track the walkdown of this component on 08/2014.	BR. 272, Column 12.5, Line E	CAB
SWEL1-450	71BCB-2A	The battery A control board is deferred. WO # 52421057 was initiated to track the walkdown of this component on 05/2013.	BR. 272, Column 13, Line C	CAB

SWEL#	Equipment ID	Description	Location	Reason
SWEL1-462	71DSC-11561	The L15 unit substation transformer T-13 high side discharge SW is deferred. WR # 290280 was initiated to track the walkdown of this component.	RB. 300', Column 2, Line R	CAB
SWEL1-470	71INV-3A	The LPCI MOV independent power supply A inverter is deferred. WO # 52391223 was initiated to track the walkdown of this component on 01/2013.	RB. 344.6', Column 5.5	CAB
SWEL1-474	71L25	The 600V switchgear distribution (BUS 12500) is deferred. WR # 290281 was initiated to track the walkdown of this component on 10/2014.	EB. 272', Column 18.5, Line A1	CAB
SWEL1-481	71MCC-161	The 600V motor control center (BUS 116100) is deferred. WR # 290282 was initiated to track the walkdown of this component on 09/2016.	EB. 272', Column 1.5, Line W	CAB
SWEL1-487	71MCC-252	The 600V motor control center (BUS 125200) is deferred. WO # 52404915 was initiated to track the walkdown of this component on 03/2013.	EB. 272', Column 18, Line A	CAB
SWEL1-489	71MCC-254	The 600V motor control center (BUS 125400) is deferred. WO # 52380939 was initiated to track the walkdown of this component on 11/2012.	EB. 272', Column 23, Line A1	CAB
SWEL1-624	93ECP-A	The EDG A engine control panel is deferred. WO # 52419775 was initiated to track the walkdown of this component on 05/2013.	EG. 272', Column 24	CAB
SWEL1-628	93ECSP-A	The EDG A engine control sub panel is deferred. WR # 290283 was initiated to track the walkdown of this component.	EG. 272', Column 24	CAB
SWEL1-629	93ECSP-B	The EDG B engine control sub panel is deferred. WR # 290284 was initiated to track the walkdown of this component.	EG. 272', Column 26, Line A3	CAB
SWEL1-636	93EGP-A	The EDG C generator control panel is deferred. WO # 52419771 was initiated to track the walkdown of this component on 05/2013.	EG. 272', Column 24, Line A1	CAB

SWEL#	Equipment ID	Description	Location	Reason
SWEL1-637	93EGP-B	The EDG B generator control panel is deferred. WO # 52427217 was initiated to track the walkdown of this component on 07/2013.	EG. 272', Column 26, Line A1	CAB
SWEL1-640	93FPAC	The EDG A & C forced paralleling panel is deferred. WO # 52286384 was initiated to track the walkdown of this component on 09/2014.	EG. 272', Column 24.5, Line A	CAB
SWEL2-013	71MCC-120-OE1	The 32P-1A(M) decay heat removal SFP water primary pump A motor is deferred. WR # 290285 was initiated to track the walkdown of this component on 03/2013.	YD. 293'	CAB
SWEL1-493	71MCC-264	The 600V Motor Control Center (BUS 126400) is deferred. WO # 52449942 was initiated to track the walkdown of this component on 10/2013.	EB. 272 Column 25.5, Line A1	CAB
SWEL1-456	71BMCC-6	The Reactor Building DC Motor Control Center is deferred. WR # 290578 was initiated to track the walkdown of this component.	EB. 272 Column 8, Line Y	CAB

7.0 SEISMIC WALKDOWNS AND AREA WALK-BYS

The NTTF 2.3 Seismic Walkdown program conducted in accordance with the Guidance involves two primary walkdown activities, Seismic Walkdowns and Area Walk-Bys. These activities were conducted at JAF, by teams of two trained and qualified SWEs (see Section 4.2). JAF in house Civil Engineers performed a portion of the walkdowns. Both SWEs on these teams have several years of seismic experience. For the balance of the walkdowns (Performed by contractors), each team (2 teams total) included one engineer with at least several years of experience in seismic design and qualification of nuclear power plant SSCs, whereas the second engineer had somewhat less (though sufficient) experience. In certain instances, the teams (both JAF and contractors) periodically "shuffled" personnel to cross-check consistency between the two teams.

The seismic walkdowns and area walk-bys were conducted over a span of approximately 6 weeks, starting in mid-September of 2012. Pre-job briefs were performed prior to walkdowns. This pre-job brief was used to outline the components and areas that would be walked down, to ensure consistency between the teams, to reinforce expectations and process for identifying potentially adverse seismic conditions (and other non-seismic plant conditions), and to allow team members to ask questions and share feedback.

7.1 SEISMIC WALKDOWNS

Seismic walkdowns were performed in accordance with Section 4 of the Guidance for all items on the SWEL (SWEL 1 plus SWEL 2), except for those determined to be inaccessible and deferred (see Section 6.3). To document the results of the walkdown, a Seismic Walkdown Checklist (SWC) with the same content as that included in Appendix C of the Guidance was created for each item. Additionally, photographs were taken of each item and included on the corresponding SWC.

In some cases, the SWE teams conducted preliminary "scouting" walkdowns to get a general understanding of plant layout and to identify items on the draft SWEL that were inaccessible. Items that were identified to be inaccessible on these "scouting" walkdowns were discussed with the Equipment Selection Personnel and were either deleted or deferred while ensuring that the overall integrity of the final SWEL was not compromised.

Prior to performance of the walkdowns, documentation packages were developed that contained the SWC with preliminary data entered and other pertinent information including the location drawings, response spectra information, previous IPEEE seismic walkdown documentation, and anchorage drawings where applicable. These documentation packages accompanied the SWE teams into the plant during the seismic walkdowns.

Walkdown inspections focused on anchorages and seismic spatial interactions, but also included inspections for other potentially adverse seismic conditions. Anchorage, in all cases, was considered to specifically mean anchorage of the component to the structure where applicable. This included anchor bolts to concrete walls or floors, structural bolts to structural steel and welds to structural steel or embedded plates. For welds, the walkdown team looked for cracks and corrosion in the weld and base metal. Other bolts or connections, such as flange bolts on in-line components were not considered as equipment anchorage. These bolts and connections were evaluated by the SWEs and any potential adverse seismic concerns were documented under "other adverse seismic conditions" rather than under "anchorage". Thus, components with no attachments to the structure are considered as not having anchorage. Nevertheless, the attachment of these components to other equipment was evaluated and inspected for potentially adverse seismic conditions.

Cabinets/panels on the SWEL that could be reasonably opened without undue safety or operational hazard were opened during the walkdown. This allowed visual observation of internal anchorage to the structure (where present), as well as inspection for "other adverse seismic conditions" related to internal components that could be observed without breaking the plane of the door. Where opening the cabinet/panel was considered to exhibit undue safety or operational hazards, it was considered inaccessible and the completion of the walkdown of that item was deferred to a later time (see Section 6.3). Where opening the cabinet/panel required extensive disassembly (e.g., doors or panels were secured by more than latches, thumbscrews, or similar), justification for how the inspection met the program goal without opening the cabinet/panel was included on the SWC and the walkdown of that item is considered complete.

In addition to the general inspection requirements, at least 50% of the SWEL items having anchorage required confirmation that the anchorage configuration was consistent with plant documentation. Of the 128 SWEL items, 81 were considered to have anchorage (i.e., removing in-line/line-mounted components). Of these 81 anchored components, the walkdowns of 54, not counting deferred components (See Section 6.3), included anchorage configuration verification, which is greater than 50%. When anchorage configuration verification was conducted, the specific plant documentation used for comparison to the as-found conditions was referenced on the SWC.

The SWC for each SWEL item, where a seismic walkdown has been initiated is included in Attachment C. A total of 128 SWCs are attached, 105 with completion status marked "Y" ("Y"- Yes, Walkdown is completed), 0 with completion status marked "N" ("N"- No, Walkdown has not been performed), and 23 with completion status marked "U" ("U"- Uncertain, More information on the component is required). SWCs considered and marked "Uncertain", are those where a walkdown was initiated, but whose completion was ultimately deferred because the cabinet/panel could not be opened during the walkdown period. Therefore, the

105 completed SWCs represent the completed walkdowns of the 128 SWEL items accessible during the walkdown period.

7.2 AREA WALK-BYS

Seismic area walk-bys were performed in accordance with Section 4 of the Guidance for all plant areas containing items on the SWEL (SWEL 1 plus SWEL 2). Area walk-bys were not deferred where components were deferred simply to open cabinets/panels. A separate Area Walk-By Checklist (AWC) with the same content as that included in Appendix C of the Guidance was used to document the results of each area walk-by performed. Photographs were taken of each area, and included on the corresponding SWC.

Area walk-bys were conducted once for plant areas containing multiple SWEL items in close proximity to each other. In cases where the room or area containing a component was very large, the extent of the area encompassed by the area walk-by was limited to a radius of approximately 35ft around the subject equipment. In some cases, the extent of the areas included in the area walk-bys is described on the AWC for that area. Because certain areas contained more than one SWEL item, there are fewer total area walk-bys conducted than seismic walkdowns. A total of 61 area walk-bys was necessary to cover all plant areas containing at least one SWEL item.

The AWC for each area walk-by completed is included in Attachment D. A total of 61 AWCs are attached, which represent all of the areas containing a SWEL item that were accessible during the walkdown period. No additional area walk-bys of areas need to be performed, since walk-bys for the deferred items have been completed (see Section 6.3).

8.0 LICENSING BASIS EVALUATIONS

During the course of the seismic walkdowns and area walk-bys, SWE teams sought to identify existing degraded, non-conforming, or unanalyzed plant conditions with respect to its current seismic licensing basis identify. This section summarizes the process used to handle conditions identified, what conditions were found, and how they were treated for eventual resolution.

<u>CONDITON IDENTIFICATION</u>

When a potentially adverse condition was observed by a SWE team in the field, the condition was noted on the SWC or AWC form and briefly discussed between the two SWEs to determine whether it was a potentially adverse seismic condition. These initial conclusions were based on conservative engineering judgment and the training required for SWE qualification.

For conditions that were reasonably judged as insignificant to seismic response, the disposition was included on the SWC or AWC checklist and the appropriate question was marked "Y", indicating that no associated potentially adverse seismic condition was observed. Unusual or uncertain conditions were reported to site personnel for further resolution (see Section 8.2). A total of 17 seismically insignificant conditions were identified. These conditions were generally related to either housekeeping or mild degradation.

For conditions that were judged as potentially significant to seismic response, then the condition was photographed and the appropriate question on the SWC or AWC was marked "N" indicating that a potentially adverse seismic condition was observed. The condition was then immediately reported to site personnel for further resolution and was documented for reporting in Attachment E. A total of 17 potentially adverse seismic conditions were identified. These conditions were generally related to non-conforming anchorage, spatial interaction, non-conforming support spacing, or inadequate line flexibility.

<u>CONDITION RESOLUTION</u>

Conditions observed during the seismic walkdowns and area walk-bys determined to be potentially adverse seismic conditions are summarized in Attachment E, including how each condition has been addressed and its current status. Each potentially adverse seismic condition is addressed either with a Licensing Basis Evaluation (LBE) to determine whether it requires entry into the Corrective Action Program (CAP), or by entering it into the CAP directly. The decision to conduct a LBE or enter the condition directly into the CAP was made on a case-by-case basis, based on the perceived efficiency of each process for eventual resolution of each specific condition.

Unusual conditions that were not seismically significant were immediately brought to attention to plant personnel. Further resolution of these conditions is not tracked or reported as part of the NTTF 2.3 Seismic Walkdown program, except by noting that the condition was observed by SWE and it was immediately brought to attention to plant personnel on the applicable SWCs and AWCs.

8.1 LICENSING BASIS EVALUATIONS

Potentially adverse seismic conditions identified as part of the NTTF 2.3 Seismic Walkdown program may be evaluated by comparison to the current licensing basis (CLB) of the plant as it relates to the seismic adequacy of the equipment in question, as is described in Section 5 of the Guidance. If the identified condition is consistent with existing seismic documentation associated with that item, then no further action is required. If the identified condition cannot easily be shown to be consistent with existing seismic documentation, or no seismic documentation exists, then the condition is entered into the CAP.

Of the 17 identified potentially adverse seismic conditions, 14 did not require a LBE. All items summarized in Attachment E, are entered into the CAP or justification of their acceptability is provided in Attachment F.

8.2 CORRECTIVE ACTION PROGRAM ENTRIES

Conditions identified during the seismic walkdowns and area walk-bys that required further resolution were entered into the plant's Corrective Action Program (CAP) for further review and disposition in accordance with the plant's existing processes and procedures. Conditions entered into the CAP included three types of unusual conditions identified:

- Seismically insignificant unusual conditions
- Potentially adverse seismic condition that does not pass a LBE
- Potentially adverse seismic condition that bypasses a LBE

A total of 15 Condition Reports (CRs) were generated from the CAP as a result of the NTTF 2.3 Seismic Walkdown program. A total of 15 CRs were written relative to potentially adverse seismic conditions identified. The CR numbers, current status, and resolution (where applicable and available) are summarized for these potentially adverse seismic conditions in Attachment E.

8.3 PLANT CHANGES

The CAP entries (CRs) generated by the NTTF 2.3 Seismic Walkdown program are being resolved in accordance with the plant CAP process, including operability evaluations, extent of condition evaluations, and root cause analysis (where applicable). Initial evaluations indicate that no immediate plant changes are necessary. Final and complete resolutions of the CRs for seismically insignificant unusual conditions and potentially adverse seismic conditions will determine if future modifications to the plant are required. While no immediate plant modifications have been identified as a result of the seismic walkdowns and walk-bys, various cases were found where repairs are required or housekeeping issues are being addressed. Current status and resolutions (where applicable and available) for CRs related to potentially adverse seismic conditions are provided in Attachment E.

9.0 PEER REVIEW

9.1 PEER REVIEW PROCESS

The peer review for the Near Term Task Force (NTTF) Recommendation 2.3 Seismic Walkdowns was performed in accordance with and in conformance to Section 6 of the Guidance. The peer review included an evaluation of the following activities:

- review of the selection of the structures, systems, and components, (SSCs) that are included in the Seismic Walkdown Equipment List (SWEL);
- review of a sample of the checklists prepared for the Seismic Walkdowns and area walk-bys
- review of licensing basis evaluations and decisions for entering the potentially adverse conditions in to the plant's Corrective Action Plan (CAP); and
- review of the final submittal report.

At least two members of the peer review team (see Section 4.5) were involved in the peer review of each activity, the team member with the most relevant knowledge and experience taking the lead for that particular activity. A designated overall Peer Review Team Leader provided oversight related to the process and technical aspects of the peer review, paying special attention to the interface between peer review activities involving different members of the peer review team.

The peer review team was provided with an early draft of this submittal report for peer review. The peer review team verified that the submittal report met the objectives and requirements of Enclosure 3 to the 50.54(f) Letter, and documented the NTTF 2.3 Seismic Walkdown program performed in accordance with the EPRI Guidance. The peer review team provided the results of review activities to the SWE team for consideration. The SWE team satisfactorily addressed all peer review comments in the final version of the submittal report. The signature of the Peer Review Team Leader provides documentation that all elements of the peer review as described in Section 6 of the EPRI Guidance were completed.

9.2 PEER REVIEW RESULTS SUMMARY

The following sections summarize the process and results of each peer review activity.

9.2.1 Seismic Walkdown Equipment List Development

The selection of items for the SWEL was peer reviewed in accordance with Section 3 of the EPRI Guidance. Peer review comments were resolved and incorporated into the final SWEL, ensuring that all recommendations of the EPRI Guidance have been met.

The final SWEL contains a diverse sample of equipment required to perform the five safety functions specified in the EPRI Guidance, which are:

- Reactor reactivity control
- Reactor coolant pressure control
- Reactor coolant inventory control
- Decay heat removal
- Containment integrity

In addition, the peer review process verified that SWEL items included major new and replacement items, a variety of environments, equipment enhanced based on findings of the IPEEE (if any), and risk insight considerations.

The peer review checklist of the SWEL is provided in Attachment G.

9.2.2 Seismic Walkdowns and Area Walk-Bys

Peer review of the seismic walkdowns and area walk-bys was conducted by two peer reviewers, each of whom is a qualified SWE and has broad knowledge of seismic engineering applied to nuclear power plants. One of the peer reviewers participated in the seismic walkdown program for a different utility, and the other is engaged with the industry team which developed the Guidance (see Section 4.5). The peer reviews were conducted at JAF concurrent with the conduct of walkdowns, at approximately 50% completion. The peer review was performed as follows:

- The peer review team reviewed the walkdown packages (including checklists, photos, drawings, etc.) for SWEL items already completed to ensure that the checklists were completed in accordance with the Guidance. A total of 23 SWC and 16 AWC forms were reviewed, each representing approximately 18% and 26%, respectively, of their totals. In the context of the Guidance, the peer review team considered the number of walkdown packages reviewed to be appropriate. The packages reviewed represent a variety of equipment types in various plant areas. Specific SWC forms reviewed are SWEL1-001, 032, 052, 069, 137, 157, 213, 433, 448, 452, 457, 474, 494, 501, 519, 624, 646, 670, 683, 686, 690, SWEL2-007 and SWEL2-009. Specific AWC forms reviewed are AWC-003, 006, 009, 013, 015, 017, 018, 021, 022, 029, 033, 034, 045, 047, 049, and 057. During the selection of SWC's and AWC's to be peer reviewed, particular attention was given to obtaining a broad sample of items that encompass a variety of equipment and systems, equipment classes and environmental conditions.

- While reviewing the walkdown packages, the peer reviewers conducted informal interviews of the SWEs and asked clarifying questions to verify that they were conducting walkdowns and area walk-bys in accordance with the Guidance.
- The peer review team held a meeting with the SWE teams to provide feedback on the walkdown and walk-by packages reviewed and the informal interviews, and discuss potential modifications to the documentation packages in the context of the Guidance.
- The peer review team held a meeting with the SWE teams to provide feedback on the walkdown and walk-by observations, and discuss how lessons learned from review of the walkdown packages had been incorporated into the walkdown process.

As a result of the peer review activities, the SWE teams modified their documentation process to include additional clarifying details, particularly related to checklist questions marked "N/A" and where conditions were observed but judged as insignificant. The peer review team felt these modifications would be of benefit for future reviews of checklists incorporated into the final report. These modifications were recommended following review of the walkdown and area walk-by packages, and the observation walkdowns and area walk-bys demonstrated that the SWEs understood the recommendations and were incorporating them into the walkdown and area walk-by process. Previously completed checklists were revised to reflect lessons learned from the peer review process.

Based on completion of the walkdown and walk-by peer review activities described, the peer review team concludes that the SWE teams are familiar with and followed the process for conducting seismic walkdowns and area walk-bys in accordance with the Guidance. The SWE teams adequately demonstrated their ability to identify potentially adverse seismic conditions such as adverse anchorage, adverse spatial interaction, and other adverse conditions related to anchorage, and perform anchorage configuration verifications, where applicable. The SWEs also demonstrated the ability to identify seismically-induced flooding interactions and seismically-induced fire interactions such as the examples described in Section 4 of the Guidance. The SWEs demonstrated appropriate use of self checks and peer checks. They discussed their observations with a questioning attitude, and documented the results of the seismic walkdowns and area walk-bys on appropriate checklists.

9.2.3 Licensing Basis Evaluations

All potentially adverse seismic conditions were entered into the plant's CAP program for further review and disposition. See Attachment E for summary of CRs and LB evaluations. The review team verified the decisions for identifying such conditions as being sound, and the dispositions were conducted in accordance with the plant's CLB.

A peer review of the licensing basis evaluations was completed. Within these licensing basis evaluations, CRs were generated for maintenance issues to replace missing bolts, nuts or remove items for housekeeping issues, or to provide further, detailed resolution of the potentially adverse seismic condition when applicable. The remaining licensing basis evaluations were created to document potentially adverse seismic conditions that were immediately entered into the CAP for detailed evaluation and investigation. See Attachment F for detailed LB evaluations. The peer review of these LB evaluations ensured that all the information provided from the walkdown team to the licensing basis evaluation team member provided enough detail for accurate and timely resolution. See Attachment I for comments received on LB evaluations.

9.2.4 Submittal Report

The peer review team was provided with an early draft of this submittal report for peer review. The peer review team verified that the submittal report met the objectives and requirements of Enclosure 3 to the 50.54(f) Letter, and documented the NTTF 2.3 Seismic Walkdown program performed in accordance with the Guidance. The peer review team provided the results of review activities to the SWE team for consideration. The SWE team satisfactorily addressed all peer review comments in the final version of the submittal report. The signature of the Peer Review Team Leader provides documentation that all elements of the peer review as described in Section 6 of the Guidance were completed.

10.0 REFERENCES

10.1. 10CFR50.54(f) 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

10.2. EPRI 1025286, Seismic Walkdown Guidance for Resolution of Fukushima Near-Term Task Force Recommendation 2.3: Seismic, June 2012

10.3. James A. FitzPatrick Nuclear Power Plant Updated Final Safety Analysis Report (UFSAR)

10.4. Generic Letter No. 88-20, Supplement 4 and 5, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities

10.5. James A. Fitzpatrick Nuclear Power Plant Individual Plant Examination of External Events (IPEEE), JAF-RPT-MISC-02211, Revision 0, Submitted June 1996.

10.6. Generic Letter No. 87-03, Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46

10.7. Seismic Qualification Utility Group (SQUG) Procedure: Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Power Plant Equipment, Revision 3A, December 2001

10.8. JAF Document 18570.00, Rev. 1, "Design Criteria for Balance of Plant (BOP) Piping Stress and Supports – James A. FitzPatrick Nuclear Power Plant."

10.9. Engineering Standard Manual: CES-2B, Rev. 0, "Seismic Design of Electrical Conduit Supports for JAF."

10.10. JAF-RPT-ELEC-02075, Rev. 2, "Design Criteria for Independence of Redundant Electrical Circuits."

10.11. EN-DC-168, Rev. 0, "Fukushima Near-Term Task Force Recommendation 2.3 Seismic Walk-down Procedure."

10.12. JAFP-12-0075, "Entergy's 120-Day Response to the NRC Request for Information (RFI) Pursuant to 10CFR50.54(f) Regarding the Seismic Aspects of Recommendation 2.3 of the Near Term Task Force Review of Insights from the Fukushima Dai-ichi Accident."

10.13. USI A-46, "Seismic Evaluation Report Volume I/VI, Stevenson & Associates," September 1995.

10.14. James A. FitzPatrick Nuclear Power Plant, Safe Shutdown Equipment and Relay Evaluation For unresolved Safety Issue USI A-46," September 1995, Volume I of XII.

11.0 ATTACHMENTS

ATTACHMENT A – IPEEE VULNERABILTIES TABLE

ATTACHMENT B – SEISMIC WALKDOWN EQUIPMENT LISTS

ATTACHMENT C – SEISMIC WALKDOWN CHECKLISTS (SWCs)

ATTACHMENT D – AREA WALK-BY CHECKLISTS (AWCs)

ATTACHMENT E – POTENTIALLY ADVERSE SEISMIC CONDITIONS

ATTACHMENT F – LICENSING BASIS EVALUATION FORMS

ATTACHMENT G – PEER REVIEW CHECKLIST FOR SWEL

ATTACHMENT H – SEISMIC WALKDOWN ENGINEER TRAINING CERTIFICATES

ATTACHMENT I – PEER REVIEW COMMENTS

ATTACHMENT 2

JAFP-12-0134

List of Regulatory Commitments

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	
Entergy will perform walkdowns for equipment that could not be inspected as identified in Section 6.3 of the Seismic Walkdown Report. The walkdown will be performed during RFO 21.	✓		End of RFO21
Entergy will submit an updated Seismic Walkdown Report.	✓		90 days after the end of RFO21