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H.B. Robinson Steam Electric Plant, Unit No. 2
Docket No. 50-261
Renewed License No. DPR-23

10 CFR 50.54

Serial: RNP-RA/12-0128

NOV 2 7 2012

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

Subject: H. B. Robinson Steam Electric Plant, Unit No. 2 Response to Recommendation 2.3 "Seismic Walkdown" of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident

Reference: 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

Dear Sir or Madam:

By letter dated March 12, 2012, the Nuclear Regulatory Commission (NRC) issued a Request for Additional Information (above Reference) requesting Licensees to provide information regarding recommendation 2.3 (Seismic) to support the evaluation of the NRC staff recommendations for the Near-Term Task Force (NTTF) review of the accident at the Fukushima Dai-Ichi nuclear facility.

By this letter, Carolina Power and Light Company (CP&L), submits the H. B. Robinson Steam Electric Plant, Unit No. 2 response regarding the performance of seismic walkdowns to identify and address degraded, non-conforming or unanalyzed conditions and to verify the current plant configuration with the current seismic licensing basis.

The information provided herein and the activities described in this report are consistent with the guidance provided by the Electric Power Research Institute's (EPRI) 2012 Technical Report 1025286 titled "Seismic Walkdown Guidance For Resolution of Fukushima Near-Term Task Force Recommendation 2.3: Seismic."

Enclosure I to this letter provides the Seismic Walkdown Report.

Enclosure II to this letter provides the Report Review by Site Management.

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CP&L requests that Attachments 6 and 7 of Enclosure I to this letter, which contain security-related information, be withheld from public disclosure in accordance with 10 CFR 2.390.

This document contains the following Regulatory Commitment:

An updated report with the results of restricted access inspections for the entries in the Table below will be submitted by February 28, 2014.

Feature	Inspection Date
AFW MDPS TO SG-B SQUARE	February 28, 2014
480 V EMERGENCY BUS E1	February 28, 2014
HAGAN RACK 30	February 28, 2014
HAGAN RACKS 1-13,26	February 28, 2014
INVERTER-A	February 28, 2014

Any other actions discussed in this document should be considered intended or planned actions. They are included for informational purposes but are not considered Regulatory Commitments.

If you have any questions regarding this submittal, please contact Mr. Richard Hightower, Supervisor – Licensing/Regulatory Programs at (843) 857-1329.

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

27 NOVEMBER 2012

Sincerely,

William R. Gideon
Site Vice President
H. B. Robinson Steam Electric Plant, Unit No. 2

WRG/am

Enclosures: I. Seismic Walkdown Report
II. Report Review by Site Management

cc: Ms. Araceli Billoch-Colón, NRC Project Manager, NRR
Mr. V. M. McCree, NRC, Region II
NRC Resident Inspector

Attachments 6 and 7 to Enclosure I of this letter contain
SECURITY-RELATED INFORMATION - WITHHOLD UNDER 10 CFR 2.390
Upon removal of Attachments 6 and 7 from Enclosure I, this letter is decontrolled.

ENCLOSURE I

SEISMIC WALKDOWN REPORT

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
RENEWED LICENSE NO. DPR-23

1.0 Introduction

The Nuclear Regulatory Commission (NRC) has issued a Request for Information pursuant to Title 10 of the Code of Federal Regulations 50.54(f) (hereafter 50.54(f) letter) regarding "Recommendations 2.1, 2.3, and 9.3 of the Near-Term Task Force review of insights from the Fukushima Dai-Ichi Accident" resulting from the Great Tohoku Earthquake and subsequent tsunami. This submittal report, pursuant to the NRC's request for information, is offered to address the scope associated only with the 50.54(f) letter Enclosure 3, NTTF Recommendation 2.3 Seismic. Specifically, this report provides information for the H.B. Robinson Steam Electric Plant Unit 2 (RNP) regarding the performance of seismic walkdowns to identify and address degraded, non-conforming or unanalyzed conditions and to verify the current plant configuration with the current seismic licensing basis. The information provided herein and the activities described in this report are consistent with the guidance provided by the Electric Power Research Institute's (EPRI) 2012 Technical Report 1025286 "Seismic Walkdown Guidance: for Resolution of Fukushima Near-Term Task Force Recommendation 2.3: Seismic." The NRC, in its letter dated May 31, 2012, endorsed the EPRI guidance document.

Per EPRI 1025286, the 2.3 Seismic Walkdown inspections were to be non-intrusive visual inspections of primarily plant Seismic Category I SSCs per generic definition contained in EPRI guidance document. During the inspections, observed degraded, nonconforming, or unanalyzed conditions were to be addressed through the corrective action program (CAP). Based on the EPRI guidance document, the list of SSCs for inspection were to be obtained through a systematic selection process to establish a broad, diverse and representative Seismic Walkdown Equipment List (SWEL). The SWEL for RNP was made up of two separate lists: SWEL 1 included 138 SSCs from various locations throughout the plant and SWEL 2 included a shorter list of Spent Fuel Pool (SFP) SSCs.

This selection process for the SSCs combined with the inspection checklist attributes assessed the seismic capabilities of the plant. These attributes pertain to SSC anchorage, interaction and other considerations based on NRC and industry insights of the Fukushima Dai-Ichi Accident.

Similar past seismic efforts include the Individual Plant Examination for External Events (IPEEE) and the Unresolved Safety Issue A-46 "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors" programs. Many of the same SSCs inspected for the A-46 Program were re-inspected for the current 2.3 Seismic Walkdowns. Most of the SWEL items originated from the A-46 Program Safe Shutdown Equipment List (SSEL). These programs occurred in the 1990s. The A-46 program reviewed all of the seismic equipment in older nuclear plants and assessed their seismic capability related to experience based data and calculations. Where needed, equipment modifications were made to meet the required seismic capabilities. The IPEEE program used Seismic Margin programs to assess the plants' capabilities to perform when subjected to a larger Review Level Earthquake (RLE). Modifications were also performed as a result if necessary. The A-46 outliers for RNP were subsumed by the IPEEE program.

The 2.3 Seismic Walkdown inspections were performed to visually check the condition of the SSCs and its anchorage to meet its seismic design basis. Also inspected are the surrounding equipment and area for interactions with other SSCs, fire hazards, water spray, and housekeeping issues that may interact with the SSCs. Conditions found were recorded on the developed checklists and evaluated. Any condition that was a potential adverse seismic condition (PASC) was further evaluated for its ability to meet its seismic design basis requirements and put into the plant CAP, if necessary. In addition to checking the SSCs with respect to their design basis, this report discusses the general adequacy of licensee monitoring and maintenance procedure by reviewing walkdown observations.

2.0 Seismic Licensing Basis

The Seismic Licensing Basis for RNP is described in the Updated Final Safety Analysis Report (UFSAR). Compliance with the Seismic Licensing Basis assures that SSCs important to safety can perform their safety function both during and after a Design Basis Earthquake (DBE). SSCs important to safety are classified as Quality Class A, which consists of Class I and Class II SSCs. Class I SSCs are designed and built to withstand the maximum potential earthquake stresses for the particular region where a nuclear plant is sited. Those items vital to safe shutdown and isolation of the reactor or whose failure either singularly or in combination with the failure of another structure or piece of equipment could result in radiation doses with consequences potentially exceeding guidelines of 10CFR100 or whose failure might cause or increase the severity of an accident are given the classification of Class I. Those items that are important to reactor operation, but are not essential to safe shutdown and isolation of the reactor or are systems involving orders of magnitude lower radioactive material inventories where a hypothetical accident could result in the release of such inventory and the resulting dose rate at the site boundary would not approach the guideline limit of 10CFR100 are given the classification of Class II. Those items not related to reactor operation or safety were designated Class III.

All systems and components designated Class I were designed so that there would be no loss of function in the event of the maximum hypothetical ground acceleration acting in the horizontal and vertical directions simultaneously. The working stress for both Class I and Class II items is kept within code allowable values for the design earthquake. The following paragraphs describe the SSCs that comply with site characteristics, earthquake characteristics, the seismic design requirements for SSCs and the various codes and standards used for seismic designs at the RNP.

The plant is located in northwest Darlington County, South Carolina, approximately three miles west-northwest of Hartsville, SC, 25 miles northwest of Florence, SC, 35 miles north-northeast of Sumter, SC, and 56 miles east-northeast of Columbia, SC. The North Carolina border is 28 miles north of the site and the Atlantic Ocean is approximately 88 miles southeast. The plant is located atop more than 400 ft. of unconsolidated Coastal Plain sediments composed largely of sands with some clay and indurated layers overlie the crystalline basement. The surface of the basement rock slopes from the outcrop zone approximately 15 miles northwest of the site toward the Atlantic Coast. The basement surface is estimated at more than 3000 ft. below sea level. The upper soil (30 ft. depth) consists of surface alluvium over about 430 ft. of the Middendorf formation. The Middendorf is made up of sands, silty and sandy clay, sandstone, and siltstone. Compressional wave velocities are 17,500 fps in the basement rock, 7,200 fps in the Middendorf, and 1,500 fps in the top 30 ft. of alluvium. The alluvium and portions of the Middendorf formation occurring near the surface exhibit lenses of compressible material and for this reason piles have been selected for the support of all major structures. The pile foundations are supported by the stiff silty, clayey, and sandy soils encountered at about 50 ft. below existing grade and below the underlying dense sands.

The largest earthquake in this region occurred at Charleston in August, 1886. Charleston is approximately 120 miles south of the site. This shock had an intensity of about Modified Mercalli IX at the epicenter and it is estimated that this shock had a Magnitude of 6 1/2 to 7 with epicentral acceleration of 0.25g to 0.30g. However, damage was confined to a relatively small area and no permanent scars remain to give testimony to the shock. Aftershocks of the main earthquake had intensities ranging up to Modified Mercalli VII. It is unlikely that intensity at the RNP site exceeded VI for the largest Charleston shock.

Only one earthquake of intensity V or greater has ever been recorded within 50 miles of the RNP site. This shock occurred on October 26, 1959, near McBee, Chesterfield County, South Carolina, with an Intensity of Modified Mercalli VI. The epicenter was located about 15 miles from the RNP site. The estimated intensity at the RNP site was about V.

On the basis of historical data, it is expected that the RNP site area could experience a shock in the order of the 1959 McBee shock once during the life of the plant. This shock could be as far distant as in 1959, or perhaps closer. On a conservative basis, Magnitude 4.5 earthquake was selected with an epicentral distance of less than ten miles. This earthquake is the Operating Basis Earthquake (OBE) and although the probable ground acceleration would be .07 to .09g, a value of 0.1g is used. The vertical component of the OBE is 2/3 of the horizontal acceleration.

To provide an adequate margin of safety, a maximum earthquake ground acceleration of 0.2g was selected for the hypothetical earthquake (Design Basis Earthquake – DBE). It is important to note that even if an earthquake comparable to the Charleston shock were to occur 35 miles from the site, the ground acceleration would not exceed 0.2g. The vertical component of the DBE is 2/3 of the horizontal acceleration.

Structures, equipment and safety related piping were designed in accordance with the following criteria:

> Stress and deformation behavior of structures, piping, and equipment were maintained within the allowable limits when subjected to loads such as dead, live, pressure, and thermal, under normal operating conditions combined with the seismic effects resulting from the response to the OBE.

These allowable limits are defined in appropriate design standards such as:

> ASME Boiler and Pressure Vessel Code Section VIII, 1968 Edition with Summer 1968 addenda
>
> American National Standards Institute (ANSI) Code for Pressure Piping ANSI B31.1.0, Power Piping, 1967
>
> ACI 318-63 Building Code Requirements for Reinforced Concrete
>
> American Institute of Steel Construction (AISC) Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, 1963 edition and 1978 edition for current work

The seismic analysis is described in section 3.7.2 of the UFSAR and includes the following design information:

a. For the design earthquake ground acceleration of 0.1 g horizontally coincident with a vertical acceleration of 0.067 g, allowable stress limits were taken at 1.33 times allowable code stresses. Primary steady state stresses are maintained within the allowable stress limits accepted as good practice and, where applicable, set forth in the appropriate design standards, e.g., ASME Boiler and Pressure Vessel Code, USAS B31.1 Code for Pressure Piping, and AISC Specifications for the Design and Erection of Structural Steel for Buildings.

b. Primary steady state stresses when combined with the seismic stress resulting from the response to a ground acceleration of 0.133g acting in the vertical and 0.2g acting in the horizontal planes simultaneously, are limited so that the function of the component, system, or structure shall not be impaired as to prevent a safe and orderly shutdown of the plant.

Seismic Class I Instrumentation and Electrical Equipment design is found in UFSAR Section 3.10. The maximum hypothetical ground motion horizontal acceleration for the plant is 0.2 g. Plant instrumentation and electrical equipment was qualified as outlined in UFSAR Section 3.10 and in accordance with the following:

Electrical and control equipment which initiates reactor trips and/or actuates safeguards systems must be capable of performing its functions during and after an earthquake that has occurred at the plant site. To demonstrate the ability of this equipment to perform under earthquake conditions, selected types of this essential equipment representative of all protection and safeguards circuits and equipment were subjected to vibration tests which simulated the seismic conditions.

Other Class I equipment (flexible and rigid) were evaluated to assure functional adequacy when considering potential equipment resonance with the building during earthquake conditions. Details of these analyses are described in UFSAR Section 3.7.3. Tanks were originally qualified under TID 7024 Chapter VI. However, they were reanalyzed under resolution of USI A-40 via the USI A-46 program.

For Regulatory Guide 1.97 equipment use of IEEE 344-'71 and some enhancements from IEEE 344-'75 is required.

Unresolved Safety Issue (USI) A-46, "Seismic Qualification of Equipment in Operating Plants," was addressed in accordance with the Seismic Qualification Utility Group (SQUG) developed Generic Implementation Procedure, Revision 2 as corrected on February 14, 1992 (GIP-2). Verification of the seismic adequacy of mechanical and electrical equipment included:

- Training of Seismic Evaluation Personnel
- Identification of Safe Shutdown Equipment
- Screening Verification and Walkdown
- Outlier Identification and Resolution

Evaluations were also performed for relay functionality review, tanks and heat exchangers review, and cable and conduit raceway review.

Revision 3 of the SQUG Generic Implementation Procedure (GIP-03), as modified and supplemented by the Nuclear Regulatory Commission Supplemental Safety Evaluation Report No. 2 (SSER No. 2) and SSER No. 3, can be used as an alternative to existing methods for the seismic design and verification of modified, new and replacement equipment. However, this alternative method is not used for NRC Regulatory Guide 1.97 equipment. The commitment for Regulatory Guide 1.97 equipment is described above.

3.0 Personnel Qualifications

The personnel involved in the Fukushima NTTF Recommendation 2.3 Seismic walkdown activity at RNP came from a variety of backgrounds in the nuclear industry. The following personnel supported the walkdown activities with their qualifications as detailed below.

3.1 Equipment Selection Personnel

3.1.1 Billy Alumbaugh

Billy R. Alumbaugh is a Registered Professional Engineer and has over 30 years engineering experience including 16 years nuclear experience with site experience working for a utility and as a consultant. Progressive experience in civil engineering ranges from individual contribution to supervisory and project management. Supervised multiple engineers at operating nuclear facility and was involved in several projects including: Control Room expansion, Equipment obsolescence, Dry Fuel Storage, and Containment redesign/design pressure uprate. Training received includes Auxiliary Operator, Waste Control Operator, Systems Training, 10CFR50.59 certification, and modification/change control. As a consultant, he served as the Civil/Structural Engineering Design Lead for the new plant Design Certification and Combined Operating License projects providing a technical review of civil based licensing responses to clients or the NRC and project management. More recently served as the Civil-Structural-Architect Discipline Manager for the detailed design phase of the US-APWR including all aspects of the design including the site specific and Design Control Document seismic evaluations. Billy Alumbaugh has an MS and a BS degree in Civil Engineering.

3.1.2 Harold Bamberger

Harold Bamberger has over 40 years of experience in both field and office functions required for designing, analyzing, and installing piping and pipe supports for metallic and non-metallic systems in major power, chemical, and pharmaceutical facilities. Mr. Bamberger has worked for various nuclear power plants in design and review of piping, piping supports and other nuclear structure using ASME Section III, ASME/ANSI B31.1 and B31.3, and applicable nuclear plant procedures. Mr. Bamberger is a Registered Professional Engineer and holds an AD in Mechanical Engineering Technology with additional classes in Mechanical Engineering and Technology.

3.2 Seismic Walkdown Engineers

3.2.1 Jose Olmeda

Mr. Olmeda has over 30 years of experience in the Analysis and Design of nuclear related facilities, components structures and systems. This includes the use of Industry Standards and codes as: ACI-318, ACI-349, AISC-ASD/LRFD, SEI/ASCE-7, ASCE-4-98, ASCE-43-05, ANSI/AISC N690, NFPA-17, and the Life Safety Code. He has a high degree of knowledge of the theory and applications of finite elements using structural analysis and design software such as GT-Strudl, STAAD.Pro2007 and other common design software in the structural engineering field. He is a longstanding member with the American Concrete Institute, American Institute of Steel Construction, and a Charter Member of the Structural Engineers Institute of the American Society of Civil Engineers. Mr. Olmeda maintains an Associated Membership in the ACI-118 Committee, Use of Computers for Concrete Applications.

3.2.2 Harold Bamberger – see above

3.2.3 Les Galazka

Mr. Galazka has over 30 years of leadership/project management experience, including engineering. He has 26 years of nuclear experience including Structural, Mechanical, Piping and Pipe Support, Start–up, Nuclear Waste Process and Management, and System Engineering. He has six years of international experience on construction, equipment installation and testing, QC, engineering and management. He holds a Bachelors and a Masters in Mechanical/Structural Engineering.

3.2.4 Primo Novero

Primo Novero has over 46 years of engineering experience, most of which in structural design, construction, and environmental. Mr. Novero has 34 years of nuclear structural design experience in various structures, systems and components involving different materials, and in diverse topical matters including seismic. Primo Novero has a BS in Civil Engineering and Environmental Engineering, and is a Registered Professional Engineer in the field of Civil, Structural and Environmental.

3.3 Licensing Basis Reviewers

3.3.1 Timothy Rouns

Timothy Rouns has over thirty years of engineering experience, including over 20 years as a Civil/Structural Engineer and 8 years as a Fire Protection Engineer. He has detailed knowledge of AISC, ACI and NFPA codes and experience using ANSI and ASME piping codes in design. He has a BS in Civil Engineering.

3.4 IPEEE Reviewers

3.4.1 Harold Bamberger – see above

3.5 Peer Review Team Members

3.5.1 Jerome Panfil

Jerome Panfil has over 30 years experience in the nuclear engineering profession as a civil/structural engineer on a variety of projects. He has extensive experience in the structural analysis, design, and installation of plant modifications in support of nuclear power station construction and operations. He has been a member of an IPEEE program team at a major nuclear plant. He holds a BS and MS in Architecture/Structural Engineering and is a licensed (i.e., Professional Engineer) Structural Engineer.

3.5.2 Louis Wade

Louis Wade has over 30 years experience in Quality Assurance/Quality Control (QA/QC), Project Management, and QA/QC consulting. Over 15 years in management positions associated with construction, maintenance, modifications, including work package control, and operation of DOE and NRC regulated facilities, Vitrification Facilities, Radioactive Waste Facilities, Gaseous Diffusion Facilities, and TRU Waste Characterization and Disposal. Mr. Wade is an ASQ Certified Quality Auditor 10600, Lead Auditor per ANSI N45.2.23, and Lead Auditor per ASME-NQA 1.

4.0 Selection of SSCs

4.1 SWEL 1 Development

The selection of SSCs included in SWEL 1 for RNP was based on the guidance provided in EPRI Guidelines, Section 3. Plant staff participated in the SSC selection process and concurred with the SSCs selected for SWEL 1. The inspection of items on this list addresses safe shutdown and containment integrity at the plant. This selection process was conducted by experienced personnel and plant operations staff members selecting SSCs based on the EPRI Guidance using screening selection criteria. These screens are listed as follows:

- Screen #1: Seismic Category I
- Screen #2: Equipment or systems NOT regularly inspected
- Screen #3: Supports five safety functions
    - Reactor reactivity control
    - Reactor coolant pressure control
    - Reactor coolant inventory control
    - Decay Heat Removal
    - Containment function
- Screen #4: Sample considerations (systems, major new/replacement, equipment types, environments, IPEEE enhancements)

The list of equipment resulting from Screen #3 is Base List 1. At RNP, the Base List 1 was created as suggested by the EPRI guidance document, through the use of a previous equipment list from implementation of the combined USI A-46 and IPEEE Seismic program. Per EPRI 1025286, the first screen narrows the list to SSCs classified as Seismic Category I items because only those have a defined seismic licensing basis against which to evaluate the as-installed configuration. For RNP, these items are not classified as Seismic Category I. They are typically classified as seismic Class I or seismic Class II. The second screen further narrows the list by selecting only those remaining items that do not have regular inspections to confirm their configuration is consistent with the licensing basis. The third screen ensures that those remaining items are associated with at least one of the five safety functions. The A-46/IPEEE SSEL met the criteria for Screens #1, #2, and #3 (although the fifth safety function was not explicitly defined in the A-46/IPEEE SSEL, personnel performing the selection verified that the SSEL contained equipment supporting all safety functions including the containment function), and thus using the A-46/IPEEE SSEL was appropriate.

Some items from the A-46/IPEEE SSEL were dropped during application of Screen 3 because they were not associated with any of the 5 safety functions.

Once Base List 1 was established, Screen #4 was applied to ensure the inspections encompassed a broad and varying array of equipment. Screen #4 included selection considerations compiled from the EPRI guidance document and from the 50.54(f) letter Enclosure 3. This resulted in the creation of SWEL 1. Considerations made for the creation of SWEL 1 are detailed in the sections below.

4.1.1 Equipment types/classes

One of the sampling objectives was to select items from all equipment classes where possible. A breakdown of the inspected items into the various equipment classes is provided in the following table.

Class No.	Equipment Included	Base List 1 Total	Selected
0	Other	20	4
1	Motor Control Centers and Wall-Mounted Contactors	8	2
2	Low Voltage Switchgear and Breaker Panels	2	1
3	Medium Voltage Metal-Clad Switchgear	0	0
4	Transformers	2	1
5	Horizontal Pumps	15	7
6	Vertical Pumps	7	2
7	Pneumatic-Operated Valves	37	16
8	Motor-Operated and Solenoid Operated Valves	69	18
9	Fans	10	3
10	Air Handlers	6	2
11	Chillers	0	0
12	Air Compressors	0	0
13	Motor Generators	0	0
14	Distribution Panels and Automatic Transfer Switches	10	2
15	Battery Racks	2	1
16	Battery Chargers and Inverters	6	2
17	Engine Generators	2	1
18	Instrument Racks	108	40
19	Temperature Sensors	18	1
20	Instrument and Control Panels	60	11
21	Tanks and Heat Exchangers	31	12
	Total	413	126

The above stated equipment classes were determined by the industry and provided in the EPRI guidance. Equipment classes were not included in the SWEL if they were not represented on Base List 1. Since the A-46/IPEEE SSEL was the starting point for Base List 1, additional verification was performed to ensure that no other safety-related equipment existed at the plant for Equipment Classes 3, 11, 12 and 13. None were identified. The equipment class for each SSC is included in Base List 1 of Attachment 1.

4.1.2 Five safety functions

The appropriate proportion of SSCs serving each of the five safety functions on Base List 1 was maintained in the selection of SSCs for the SWEL 1 as follows:

Safety Function	Total SSCs	Selected SSCs
Reactor reactivity control	137	40
Reactor coolant pressure control	166	43
Reactor coolant inventory control	130	42
Decay heat removal	271	89
Containment function	173	51

This table demonstrates full coverage of the five safety functions for the selected SSCs. Base List 1 in Attachment 1 includes the safety function category of each SSC.

4.1.3 Locations

Although not required by the guidance, SSCs in a variety of plant locations were considered for inclusion on SWEL 1 including the Reactor Containment, Reactor Auxiliary Building, Control Building, Diesel Generator building, Service Water Intake Structure, Condensate Storage Tank, and the Diesel Fuel Storage. SWEL 1 in Attachment 2 includes the building location of each item.

4.1.4 Environments

SSCs from a variety of environments including dry and hot, wet and cold, mild and harsh, and inside and outside buildings were included for inspection in the SWEL 1. SWEL 1 in Attachment 2 includes the environment of each item.

4.1.5 Systems

During the SWEL 1 selection process, consideration was given to equipment of varying systems including the Chemical Volume Control, Auxiliary Feedwater, Main Steam, and Residual Heat Removal Systems. Table B-1 of Appendix E of the EPRI guidance was consulted to ensure systems to support safety functions were included. Additionally, equipment in the Service Water System that support access to the Ultimate Heat Sink was included in SWEL 1. SWEL 1 in Attachment 2 includes the system of each SSC.

4.1.6 Risk

The selection team was able to readily identify items that posed a higher risk ranking due to their knowledge and experience of nuclear plant operations and those SSCs that contribute to nuclear plant risk profiles. An element of the team's experience included knowledge of seismic PRA and other risk lists that comprise SSCs and conditions that combine probability and consequences of an event. Such items as emergency diesels, station batteries, core cooling systems, emergency cooling water systems, and 1E electrical switchgear are identified as critical equipment that have a higher risk profile. These items were included while maintaining a balance with the other requirements of SWEL equipment selection.

4.1.7 IPEEE vulnerabilities

No IPEEE seismic vulnerabilities were reported for RNP.

4.1.8 Modified, replacement, and new equipment

A review of plant modifications from 1995 (the initiation time of IPEEE) to 2011 was performed and key plant personnel were consulted to determine significant modifications to the plant. Twenty-nine of 41 identified replacement items were inspected. No SSCs considered to be new equipment were identified.

4.1.9 Accessibility

Before the walkdowns, some SSCs were determined to be inaccessible due to a variety of reasons, such as the item was in a high radiation area, blocked by sensitive instruments or were overhead and required scaffolding to access. When an item was removed from SWEL 1, a review of Base List 1 was completed to determine if similar equipment was accessible and a substitution was made. Items that did not have an acceptable substitute are to be inspected at a later date and are discussed in Section 5.6.

4.2 SWEL 2 Development

Equipment Selection Personnel along with plant operations and systems personnel developed the RNP Base List 2, SWEL 2 and Rapid Drain-Down List based on the EPRI guidance document which presents screening criteria to identify specific equipment that is unique to the SFP SSCs. Screen #1 and #2 limit SFP SSCs to those which have a Seismic Category I licensing basis (or in the case of RNP, Quality Class A Class I SSCs) and are capable of being visually reviewed in the plant. This list is determined to be Base List 2 and is included as Attachment 3.

The Rapid Drain-Down List identifies items that have the possibility of providing a hydraulic pathway for a rapid drain-down of the SFP within 72 hours after an earthquake to a level approximately 10 ft. above the spent fuel stored in the pool. It was determined that the return piping from the SFP cooling heat exchanger has a 1 in. branch line near the fuel pool normal water level with a 0.5 in. diameter hole to act as a vacuum breaker to prevent a siphoning effect on the pool. The other item was an SFP drain. This line is within 3 in. of the bottom of the pool. It is prevented from draining by two closed and locked valves, one inside the pool and one outside the pool, and a blind spectacle flange upstream of the outside valve. Only the spectacle flange was added to the SWEL 2 list since the return piping in question is seismic Class 1 and is covered under the ISI program. There are no other penetrations in the SFP within 10 ft. above the top of the fuel assemblies. The evaluation found no other drain down paths that would meet the Rapid Drain-down criteria and, therefore, the only item listed in Attachment 4 as meeting the rapid drain-down criteria was the blind spectacle flange. SWEL 2 is included as Attachment 5.

5.0 Seismic Walkdowns and Area Walk-Bys

The methodology used to complete the walkdowns and area walk-bys complies with the EPRI guidance. The walkdowns and area walk-bys were performed by the SWEs listed in Section 3.2 in groups of at least two. The SWEs used engineering judgment, based on their experience and training, to identify PASCs. After active discussion of all observations and judgments, all issues that were not resolved by consensus of the SWEs were further evaluated as described in Section 5.0 of the EPRI guidance document. Walkdown results were documented on the Seismic Walkdown Checklists and area walk-bys on Area Walk-By Checklists. These checklists are provided as Attachments 6 and 7, respectively.

5.1 Seismic Walkdown Methodology

The SWEL 1 and SWEL 2 lists were combined into one to develop the individual walkdown packages. Working with the site personnel, the walkdown packages were grouped based on elevation, location and the expected number of SSCs that could be walked down during the scheduled time and date. Two separate inspection teams were utilized; each team consisted of two Seismic Walkdown Engineers (SWEs), a seismic support engineer and a plant representative. A pre-job brief was performed prior to each day's walkdown activities to ensure team members could perform the task safely and effectively. When SWEL items were inaccessible at the time of inspection and an appropriate substitute was not available, the item was documented to be inspected at a future date as detailed in Section 5.6 below.

Seismic walkdowns were performed on each SWEL 1 and 2 and were evaluated for adverse anchorage conditions, adverse seismic spatial interactions, or other adverse seismic conditions as detailed below.

5.1.1 Adverse Anchorage Conditions

Lack of anchorage or inadequate anchorage has been the primary cause for malfunction and failure of equipment during an earthquake. During the walkdown inspection, the anchorage was inspected against specific design details for approximately 50% of the SWEL items that include anchorage.

For all SWEL items with anchorage, a general visual inspection of anchorage was performed to determine if the SSC had indications of the following:

- Bent, broken, missing, or loose hardware
- Corrosion that is more than mild surface oxidation
- Visible cracks within 10D of an anchor
- Gaps that may exist at the visible parts of the equipment foundation
- Other potential adverse concerns

In cases where the anchorage was inaccessible and a substitution was not possible, an alternate method was used to assess potential degraded, non-conforming, or unanalyzed conditions which included:

- An review of previous walkdown packages to validate prior inspection attributes for adequacy
- A determination whether the local environment could cause the degradation of anchorage or its installation, (e.g. adverse environment conditions):
  - Evidence of moisture or relatively high humidity,
  - Evidence of corrosion on other nearby components and
  - Anchorage, and/or indication of vibration that could loosen the fasteners.
- A check whether the equipment and its anchorage have been subjected to maintenance or modified since it was last walked down

The SWEs used engineering judgment to assess whether the anchorage is potentially vulnerable to seismic failure or malfunction. The basis for any judgment used in the assessment was documented in the seismic walkdown checklists.

5.1.2 Adverse Seismic Spatial Interactions

Seismic spatial interaction is the physical interaction between the SWEL item and a nearby SSC caused by relative motion between the two during an earthquake. The walkdown included an inspection of the adjacent and surrounding areas to each SWEL item for adverse seismic interaction conditions that could affect the capability of the item to perform its intended safety-related functions. The three types of seismic spatial interaction effects considered were: proximity to an item, failure of an SSC and falling on an item, and flexibility of attached lines impacting an item.

5.1.3 Other Adverse Seismic Conditions

In addition to adverse anchorage and spatial interaction conditions, other potentially adverse seismic conditions that could challenge the adequacy of SWEL items were also identified when present, such as:

- Degraded conditions
- Loose or missing fasteners that secure internal or external components to equipment
- Large, heavy components mounted on a cabinet that are not typically included by the original manufacturer
- Cabinet doors or panels that are not latched or fastened

5.2 Area Walk-By Methodology

The focus of the area walk-bys was to identify potentially adverse seismic conditions associated with other SSCs located in the vicinity of the SWEL item (either within the room or for large rooms within approximately 35 ft. from the item). The key examination factors that were considered included: anchorage conditions, significantly degraded equipment in the area, a visual assessment of cable/conduit raceways, HVAC ducting, housekeeping items that could cause adverse seismic interaction, and seismically induced fire and flooding/spray interactions as described below.

5.2.1 Seismically Induced Fire Interactions

During the seismic walkdown, the engineers visually assessed hazardous/flammable material (e.g., compressed flammable gas bottles, fuel tanks, other combustible material, etc.) and high voltage equipment located in the vicinity of the SWEL item to ensure adequate support and the absence of seismic interaction. The SWEs assessed the likelihood of seismically induced fire during the walkdowns. This was primarily accomplished by assessing seismic event impacts to flammable or combustible materials and high energy electrical equipment. Results of the findings are included in AWC.

5.2.2 Seismically Induced Flooding/Spray Interactions

Two examples of potential flooding sources are rupture of piping and vessels. Flooding is most likely to originate from threaded fire protection piping, sprinkler head impact, flexible headers and stiff branch pipes, non-ductile mechanical couplings, seismic anchor motion and failed supports.

As the SWEs performed the walkdowns, they visually assessed the potential sources of water (e.g. fire suppression piping, tanks, etc.) located in the vicinity of the subject SSC. The SWEs verified that the water sources had adequate support so that they are not likely to be a source of flooding or spray that could adversely affect the nearby SSCs. The items that were identified as potential flooding/spray conditions were documented on the AWC along with any assessment of the effects.

5.3 Results

When conditions were identified during the inspection that were not readily determined as acceptable, they were documented along with an evaluation of the condition using available design information and based on the SWEs' experience. SSCs may have been determined to be a PASC at the time of the inspection and noted as such on the checklist, or the condition may have been documented and further discussion completed before determining if it was a PASC. Non-PASC conditions found during the inspections are those evaluated and determined to not affect the ability of the item to perform its intended safety function during or after design basis ground motion as noted in the Current Licensing Basis. For those Items not readily evaluated to meet the current licensing basis, the item was entered into the Corrective Action Program (CAP) for resolution. Of the 138 SWEL items inspections and 56 area walk-bys, 9 PASCs were identified. For all PASCs identified, a licensing basis review was completed as detailed in Section 6.0 below. Additionally, an operability review determined that all the SSCs identified as a PASC were operable.

The following table summarizes the condition and status of each (initially identified) potentially adverse seismic condition. All conditions were found to be in compliance with their seismic licensing basis.

Feature	Condition	Status of Resolution
North Service Water Header Strainer	Spalled grout in mounting pedestals and anchor bolts had lack of proper thread engagement	The condition was found to be bounded by the seismic licensing basis.
Reactor Coolant Pump "A" Seal Leak-off high range flow transmitters	An HVAC duct on threaded rod hangers does not appear to be seismically supported and is directly above flow transmitters	The condition was found to be bounded by the seismic licensing basis.
Refueling Water Storage Tank level transmitter	The grating over the Safety Injection pipe is not adequately anchored.	The condition was found to be bounded by the seismic licensing basis.
Reactor Protection Instrumentation Rack Area	Filing cabinet and rack of instruments secured with thin cables. Potential deficiencies are size of cable, slack in cable, and use of only one attachment point.	The condition was found to be bounded by the seismic licensing basis.
Emergency Diesel Generator Jacket Water Expansion Tank	The overflow pipe from the diesel engine to the bottom of the tank may not be sufficiently restrained in the north-south direction	The condition was found to be bounded by the seismic licensing basis.
Instrument Racks PT-125, 154, PI-154B, & LT-494	Two HVAC ducts near the instrument rack do not appear to be properly restrained	The condition was found to be bounded by the seismic licensing basis.
Emergency Diesel Generator A Air Receiver Tank	Service Water anchor bolt nut does not have full thread engagement	The condition was found to be bounded by the seismic licensing basis.
Spent Fuel Pit Cooling Pump "A"	Pump baseplate configuration not in conformance with drawing	The condition was found to be bounded by the seismic licensing basis.
SFP Spent Fuel Pit Hi/Lo Level Alarm	Support for LA-651 was found with two anchor bolts in lieu of four as shown in drawing	The condition was found to be bounded by the seismic licensing basis.

5.4 Maintenance Assessment

The maintenance assessment, as required as part of the 10CFR50.54(f) response, was completed by analyzing the number of housekeeping and maintenance issues identified during the walkdowns and area walk-bys and the determined causes during CAP evaluation. During the walkdowns, relatively few housekeeping problems and other minor issues were noted such as areas of general corrosion, slightly chipped grout, and minor spalling. Most mobile equipment, tables, and tools were either secured properly or located in safe locations away from plant equipment. Few issues were noted with cleaning equipment. These indicators suggest that monitoring and maintenance processes and procedures are adequate.

5.5 Planned or Newly Installed Protection or Mitigation Features

There were no planned or newly installed protection or mitigation features.

5.6 Inaccessible Items

There are five items that were not completely inspected since internal inspection posed a personnel and/or plant safety concern at the time of inspection. Work Requests have been issued to complete these inspections by December 31, 2013.

The expected inspection date is based on the next planned outage occurring approximately in the fourth quarter of 2013.

Feature	Inspection Date
AFW MDPS TO SG-B SQUARE	December 2013
480 V EMERGENCY BUS E1	December 2013
HAGAN RACK 30	December 2013
HAGAN RACKS 1-13,26	December 2013
INVERTER-A	December 2013

All cabinets were inspected/evaluated for anchorage and found to be acceptable. The cabinets will be opened in the future for further examination of internal components. The inspections will be completed and the updated report will be submitted by February 28, 2014.

6.0 Licensing Basis Evaluations

All of the potentially adverse seismic conditions that were identified during the seismic walkdowns and area walk-bys were found to meet the plant seismic licensing basis through evaluation in the plant CAP system. In all cases, all of the Class 1 SSC sampled would have been capable of fulfilling their intended safety function.

7.0 IPEEE Vulnerabilities Resolution

The IPEEE program identified no seismic vulnerabilities for RNP.

8.0 Peer Review

The Peer Review Report is included in Attachment 8.

Enclosure 1 Attachments

H.B. Robinson Steam Electric Plant Unit 2 Seismic Walkdown Report
(14 pages, including cover page)

Attachment 1: Base List 1

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
VENTILATION DAMPER					X
DAMPER FOR OUTSIDE AIR					X
SAFETY RELIEF VALVE (SRV-1)		X			
SAFETY RELIEF VALVE (SRV-2)		X			
SAFETY RELIEF VALVE (SRV-3)		X			
SW STRAINER A				X	
SW STRAINER B				X	
RELIEF VALVE		X			
RELIEF VALVE FOR SG A		X			
RELIEF VALVE FOR SG B		X			
RELIEF VALVE FOR SG C		X			
RELIEF VALVE FOR SG A		X			
RELIEF VALVE FOR SG B		X			
RELIEF VALVE FOR SG C		X			
RELIEF VALVE FOR SG A		X			
RELIEF VALVE FOR SG B		X			
RELIEF VALVE FOR SG C		X			
RELIEF VALVE FOR SG A		X			
RELIEF VALVE FOR SG B		X			
RELIEF VALVE FOR SG C		X			
125 VDC MCC-A	X	X	X	X	X
125 VDC MCC-B	X	X	X	X	X
MOTOR CONTROL CENTER	X	X	X	X	X
MOTOR CONTROL CENTER	X	X	X	X	X
MOTOR CONTROL CENTER	X	X	X	X	X
MOTOR CONTROL CENTER	X	X	X	X	X
MOTOR CONTROL CENTER	X	X	X	X	X
MOTOR CONTROL CENTER	X	X	X	X	X
480 V EMERGENCY BUS E1	X	X	X	X	X
480 V EMERGENCY BUS E2	X	X	X	X	X
CONSTANT VOLTAGE	X	X	X	X	X

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
CONSTANT VOLTAGE	X	X	X	X	X
AUX FEEDWATER MOTOR					X
AUX FEEDWATER MOTOR					X
AUX FEED WATER STEAM					X
BORIC ACID TANK TRANSFER	X				
BORIC ACID TANK TRANSFER	X				
CCW PUMP A	X	X	X	X	X
CCW PUMP B	X	X	X	X	X
CCW PUMP C	X	X	X	X	X
CHARGING PUMP B AND COOLER			X		
CHARGING PUMP C AND COOLER			X		
FUEL OIL TRANSFER PUMP A	X	X	X	X	X
SI PUMP A	X	X	X	X	X
SI PUMP B	X	X	X	X	X
WASTE GAS COMPRESSOR A					
WASTE GAS COMPRESSOR B					
FUEL OIL TRANSFER PUMP B	X	X	X	X	X
RHR PUMP A UNIT				X	
RHR PUMP B UNIT				X	
SW PUMP A				X	
SW PUMP B				X	
SW PUMP C				X	
SW PUMP D				X	
AO DAMPER FROM OUTSIDE AIR					X
AO DAMPER FROM OUTSIDE AIR					X
RCP A SEAL DISCH AOV					X
RCP B SEAL DISCH AOV					X
RCP-C SEAL DISCH AOV					X
LOOP 1 HOT LEG INJECTION AOV					X
LOOP 2 COLD LEG INJECTION					X
FL CUT VALVE MDP-A					X
PISTON OP VALVE MDP-B					X
SW-A FLOW CONTROL VALVE					

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
SW-B FLOW CONTROL VALVE					
FLOW CONTROL VALVE SCR A					
FLOW CONTROL VALVE SCR B					
FLOW CONTROL VALVE SCR C					
FL CONTROL VALVE					
CHARGING ISOLATION LINE AOV				X	
STM DMP N2 ACC INLT PCV AT N2				X	
STEAM DUMP N2 DISCH PRES				X	
STEAM DUMP N2 DISCH PRES				X	
STEAM DUMP N2 DISCH PRES				X	
PORV-1		X			
PORV-2		X			
PRESSURE REGULATOR INLET				X	
PRESSURE REGULATOR INLET				X	
PRESSURE REGULATOR INLET				X	
SG PORV FOR SG A		X		X	
SG PORV FOR SG B		X		X	
SG PORV FOR SG C		X		X	
AOV RWST/SI RETURN	X				
AOV RWST/SI RETURN	X				
TEMP CONTROL VALVE (AOV) DG-A		X	X	X	X
TEMP CONTROL VALVE (AOV) DG-B		X	X	X	X
TEMP CONT VALVE (AOV)		X			
MDP-A TEMP CNT VALVE (AOV)		X			
MDP-B TEMP CNT VALVE (AOV)		X		X	
AOV TEMPERATURE CONTROL		X		X	
AOV TEMPERATURE CONTROL		X		X	
MOV COMMON		X		X	X
MOV COMMON		X		X	X
MOV - COMMON		X		X	X
MOV COMMON HEADER MDP		X		X	X
MOV MDP-A		X		X	X
MOV MDP-B		X		X	X

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
MOV COMMON HEADER		X		X	
MOV COMMON HEADER MDP		X		X	
MOV FOR CCW/RCP INLET		X		X	
MOV FOR CCW/RCP INLET		X		X	
MOV CCW/THERMAL BARRIER		X		X	
MOV RHR HX A DISCHARGE TO		X		X	
MOV RHR HX B OUTLET		X		X	
BORIC ACID ISOLATION VALVE	X				
MOV SEAL WATER RETURN LINE					X
SOLENOID VALVE TO RVI-1				X	
SOLENOID VALVE TO RVI-2				X	
SOLENOID VALVE TO RVI-3				X	
SOLENOID VALVE				X	
SOLENOID VALVE				X	
CCW RETURN HEADER MOV		X		X	
SOLENOID INLET VALVE TO RVI-1				X	
SOLENOID INLET VALVE TO RVI-3				X	
SOLENOID VALVE TO RVI-2				X	
SOLENOID INLET VALVE TO RVI-1				X	
SOLENOID INLET VALVE TO RVI-3				X	X
SOLENOID VALVE TO RVI-2				X	
VCT ISOLATION MOV					
(MOV) 2A MS SUPPLY		X			
(MOV) TR B MS SUPPLY (SDP)		X			
(MOV) TR C MS SUPPLY (SDP)		X			
PRESS CONTROL VALVE ACC A		X			
PRESS CONTROL VALVE ACC B		X			
MOV TO PORV 456 (BLOCK		X		X	
MOV TO PORV 455C (BLOCK		X		X	
RHR/SI COLD LEG JUNCTION MOV				X	
RHR/SI COLD LEG JUNCTION MOV				X	
RHR LOOP 2 HOT LEG ISOLATION				X	
RHR SUCTION LINE MOV				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
RHR PUMP INLET - PUMP A				X	
RHR PUMP INLET - PUMP B				X	
RHR HX OUTLET MOV - PUMP A				X	
RHR HX OUTLET MOV - PUMP B				X	
RHR/CONTAINMENT SUMP				X	
RHR/CONTAINMENT SUMP				X	
RHR/SI CONT SUMP ISOL MOV				X	
RHR/SI CONT SUMP ISOL MOV				X	
RWST/RHR ISOL MOV				X	
RWST/RHR ISOLATION MOV				X	
MOV SI/RHR BOUNDARY				X	
MOV SI/RHR BOUNDARY				X	
MOV RWST DSCH LINE			X	X	
MOV RWST DSCH LINE			X	X	
MOV SI/BIT INLET VALVE	X				X
MOV SI/BIT INLET VALVE	X				X
MOV BIT OUTLET VALVES	X				X
MOV BIT OUTLET VALVES	X				X
MOV SI DSCH PATH FOR PUMPS	X				X
PORV 456 SOLENOID VALVES					X
SOLENOID VALVES TO PORV 455C					X
PORV 456 SOLENOID VALVES					X
SOLENOID VALVES TO PORV 455C					X
ISOL VALVE TO SOUTH SUPPLY			X	X	
SW DSCH HDR X-CONNECTION			X	X	
SW DSCH HDR X-CONNECTION			X	X	
ISOL VALVE TO NORTH SUPPLY			X	X	
MOV BUTTERFLY TB ISOL			X	X	X
SW/TB ISOL MOVS			X	X	X
SW/TB ISOL MOVS			X	X	X
EXHAUST FAN FOR EDG-B	X	X	X	X	X
EXHAUST FAN EDG-A HVAC	X	X	X	X	X
SI/CS PUMP RM HVAC				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
SI/CS PUMP RM HVAC				X	
AFW PUMP RM HVAC				X	
AFW PUMP RM HVAC				X	
RHR PUMP RM HVAC				X	
RHR PUMP RM HVAC				X	
SUPPLY FAN	X	X	X	X	X
SUPPLY FAN	X	X	X	X	X
EDG A AIR DRYER SW SIDE	X	X	X	X	X
EDG B AIR DRYER SW SIDE	X	X	X	X	X
CNTRL RM AIR HANDLNG UNIT &					X
AIR CLEANING UNIT					X
H&V EQUIPMENT ROOM COOLER				X	
H&V EQUIPMENT ROOM COOLER				X	
118V INSTRUMENT BUS 1	X	X	X	X	X
118V INSTRUMENT BUS 2	X	X	X	X	X
118V INSTRUMENT BUS 3	X	X	X	X	X
118V INSTRUMENT BUS 4	X	X	X	X	X
118V INSTRUMENT BUS 6	X	X	X	X	X
118V INSTRUMENT BUS 7A	X	X	X	X	X
118V INSTRUMENT BUS 7B	X	X	X	X	X
118V INSTRUMENT BUS 8	X	X	X	X	X
118V INSTRUMENT BUS 9	X	X	X	X	X
118V INSTRUMENT BUS 9B	X	X	X	X	X
STATION BATTERY "A"	X	X	X	X	X
STATION BATTERY "B"	X	X	X	X	X
BATTERY CHARGER "A"	X	X	X	X	X
BATTERY CHARGER "B"	X	X	X	X	X
BATTER CHARGER "A1"	X	X	X	X	X
BATTER CHARGER "B1"	X	X	X	X	X
INVERTER-A	X	X	X	X	X
INVERTER-B	X	X	X	X	X
EMERGENCY DIESEL GENERATOR	X	X	X	X	X
EMERGENCY DIESEL GENERATOR	X	X	X	X	X

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
PRESSURE SWITCH FOR FP/A					X
PRESSURE SWITCH FOR FP/A					X
PRESSURE SWITCH FOR FP/B	X	X	X	X	X
PRESSURE SWITCH FOR FP/A					X
PRESSURE SWITCH FOR FP/B	X	X	X	X	X
PRESSURE SWITCH FOR FP/B					X
PRESSURE SWITCH FOR FP/B	X	X	X	X	X
PRESSURE SWITCH FOR FP/B					X
DIFF PRESS SWITCH SW-A				X	
DIFF PRESS SWITCH SW-B				X	
ERFIS MULTIPLEXER 3		X			X
FIRE DAMPER POWER SUPPLY					X
FLOW INDICATOR CONROL -				X	
FLOW INDICATOR CONTROLLER -				X	
FLOW INDICATOR CONTROLLER -				X	
FLOW INDICATING CONTROL-CSP DISCHARGE				X	
FLOW INDICATING CONTROLLER-				X	
FLOW TRANSMITTER FOR BAT TO CHARGING PUMP	X				
CHARGING FLOW TRANSMITTER	X	X	X	X	X
FL TRANS MDP-A				X	
FL TRANS MDPB				X	
AFW MDPS TO SG-A FLOW TRANS				X	
AFW MDPS TO SG-B FLOW TRANS				X	
AFW MDPS TO SG-C FLOW TRANS				X	
AFW SDP TO SG-A FLOW TRANS				X	
AFW SDP TO SG-B FLOW TRANS				X	
AFW SDP TO SG-C FLOW TRANS				X	
RCP-C SEAL LEAK-OFF HI RANGE FLOW TRANS			X		
RCP-C SEAL LEAK-OFF LO RANGE FLOW TRANS			X		
RCP-B SEAL LEAK-OFF HI RANGE FLOW TRANS			X		
RCP-B SEAL LEAK-OFF LO RANGE FLOW TRANS			X		
RCP-A SEAL LEAK-OFF HI RANGE FLOW TANS			X		
RCP-A SEAL LEAK-OFF LO RANGE FLOW TRANS			X		

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
FLOW TRANSMITTER CCW				X	
COMMON FL TRANS				X	X
SI FLOW TRANSMITTER	X				X
AFW MDPS TO SG-A SQUARE				X	
AFW MDPS TO SG-B SQUARE				X	
AFW MDPS TO SG-C SQUARE				X	
AFW SDP TO SG-A SQUARE ROOT EXTRACTOR				X	
AFW SDP TO SG-B SQUARE ROOT EXTRACTOR				X	
AFW SDP TO SG-C SQUARE ROOT EXTRACTOR				X	
INSTRUMENT RACK FOR PT-950,					X
INSTRUMENT RACK FOR PT-951,					X
CCW SURGE TANK LOCAL LEVEL INDICATOR		X		X	X
DOST LEVEL INDICATOR SWITCH	X	X	X	X	X
CST LEVEL SIGNAL ISOLATOR					X
CST LEVEL SIGNAL ISOLATOR					X
POWER SUPPLY			X		
LEVEL TRANSMITTER BAT-A	X			X	X
LEVEL TRANSMITTER BAT-B	X			X	X
VOLUME CONTROL TANK LEVEL TRANSMITTER				X	
CST LEVEL TRANS				X	
CST LEVEL TRANS				X	
CCW SURGE TANK LEVEL				X	
RWST LEVEL TRANS			X		
RWST LEVEL TRANS			X		
PRESSURE CONTROL - COMMON		X			
REMOTE RCP-C TB DIFF PRESS			X		
REMOTE RCP B TB DIFF. PRESS.			X		
REMOTE RCP-A T.B. DIFF. PRESS. INDICATOR			X		
REMOTE RCP-C SEAL DISCH			X		
REMOTE RCP-B DISCH PRESS			X		
REMOTE RCP-A SEAL DISCH			X		
PIC FOR SDP TURBINE (GOVERNS)			X		
SEAL INJECT FILTER PRES IND			X		

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
PRESS INDICAT CONTROLLER				X	
PRESS INDIC CONTLR FOR SG B				X	
PRESS INDIC CONTR FOR SG C				X	
DG-A PRESSURE SWITCH				X	
DG-B PRESSURE SWITCH				X	
OUTLET PRESSURE SWITCH	X	X	X	X	X
OUTLET PRESSURE SWITCH	X	X	X	X	X
PRESSURE SWITCH FOR FP/A					X
PRESSURE SWITCH FOR FP/B					X
PRESSURE SWITCH FOR FP/A					X
PRESSURE SWITCH FOR FP/B					X
PRESSURE SWITCH (H/L)					
MDP-A TRIPS MDP-A				X	
MDP-A TRIPS MDP-A				X	
PR SW TRIP MDP-B				X	
LOW PRESS				X	
SDP LINE TRIPS				X	
SDP LINE TRIPS				X	
VCT PRESSURE TRANSMITTER			X		
CHARGING PRESSURE			X		
PRESSURE TRANS RCP-C	X	X			X
PRESSURE TRANS RCP-B	X	X			X
PRESSURE TRANS RCP-A	X	X			X
MDP COMMON HEADER PRESS TR				X	
COMMON PR TRANS				X	
RCP-C SEAL DISCH PRESS TRANS	X	X			X
RCP B SEAL DISCH PRESS TRANS	X	X			X
RCP-A SEAL DISCH PRESS TRANS	X	X			X
NORTH SW SUPPLY HDR PRESS				X	
SOUTH SW SUPPLY HDR PRESS				X	
SG A MAIN STEAM LINE PRES				X	
SG B MAIN STEAM LINE PRESS				X	
SG C MAIN STEAM LINE				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
PRESSURE TRANSMITTER FOR		X			X
PRESS TRANS BIT		X			X
SI/BIT PRESS TRANS		X			X
TEMPERATURE INDICATOR	X				
TEMPERATURE INDICATOR	X				
TEMP IND CNTLLR - THERMAL			X	X	X
LOW LEVEL AMPLIFIER		X		X	
LOW LEVEL AMPLIFIER		X		X	
SIGNAL ISOLATOR		X		X	
VOLUME CONTROL TANK TEMP				X	X
HX DISCHARGE TEMPERATURE	X	X		X	X
TEMP ELEMENT RCP-C	X	X		X	X
TEMP ELEMENT RCP-B	X	X		X	X
TEMP ELEMENT RCP-A	X	X		X	X
TEMP ELEMENT CVC-SEAL	X	X		X	X
LOOP 1 TEMPERATURE ELEMENT	X	X		X	X
LOOP 1 TEMPERATURE ELEMENT	X	X		X	X
LOOP 1 TEMPERATURE ELEMENT	X	X	X	X	X
LOOP 2 TEMPERATURE ELEMENT	X	X	X	X	X
LOOP 2 TEMPERATURE ELEMENT	X	X	X	X	X
LOOP 3 TEMPERATURE ELEMENT	X	X	X	X	X
LOOP 3 TEMPERATURE ELEMENT	X	X	X	X	X
PORV DISCH TEMP ELEMENT	X	X	X	X	X
TEMPERATURE ELEMENT SRV-3	X	X	X	X	X
TEMPERATURE ELEMENT SRV-2	X	X	X	X	X
TEMPERATURE ELEMENT SRV-1	X	X	X	X	X
TEMPERATURE ELEMENT CCW				X	
PANEL A-65V: PSL-1616 & PSL-1684				X	
AUX RELAY RKS: A-F (RELAYS	X	X	X	X	X
AUX RELAY RKS: G-M (NO I)	X	X	X	X	X
RELAY BOX	X	X	X	X	X
RELAY BOX	X	X	X	X	X
CET PANEL INCLUDES TM-577 &	X	X	X	X	X

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
HAGAN RACKS 1-13,26	X	X	X	X	X
DG-A MOTOR 480V POWER BOX A	X	X	X	X	X
DGA-CONTROL SWITCHBOARD A	X	X	X	X	X
DG-A VOLTAGE REGULATOR	X	X	X	X	X
DG-A MOTOR CONTROL PANEL	X	X	X	X	X
DG-A EXPANSION TANK	X	X	X	X	X
DG-B MOTOR 480V POWER BOX B	X	X	X	X	X
DGB-CONTROL SWITCHBOARD B	X	X	X	X	X
DG-B VOLTAGE REGULATOR	X	X	X	X	X
DG-B MOTOR CONTROL PANEL	X	X	X	X	X
DG-B EXPANSION TANK	X	X	X	X	X
ERFIS MULTIPLEXER 2 Q-LIST				X	
FIRE DETECTOR ACTUATION				X	
FIRE DETECTOR ACTUATION				X	
FIRE DETECTOR ACTUATION				X	
FIRE DETECTOR ACTUATION				X	
FIRE DAMPER POWER SUPPLY				X	
INADEQUATE CORE COOLING				X	
INADEQUATE CORE COOLING				X	
INSTRUMENT RACK: PT-131,156, PI-156B & LT-474				X	
INSTRUMENT RACK: PT-128,155, PI-155B & LT-484				X	
INSTRUMENT RACK: PT-125,154, PI-154B & LT-494				X	
SG-A LEVEL INDICATOR				X	
SG-A LEVEL INDI AFWP RM PNL				X	
SG-B LEVEL INDICATOR				X	
SG-B LEVEL INDIC AFWP ROOM				X	
SG-C LEVEL INDI MEZZANINE				X	
SG-C LEVEL INDICATOR AFW				X	
NUCLEAR INSTRUMENTATION	X				
POST ACCIDENT MONITOR PANEL					
POST ACCIDENT MONITOR PANEL					
PRESS CAB: LT-459, PT-455		X			
PRESS CAB: LT-460, PT-456		X			

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
PRESS CAB: LT-461, PT-457		X			
PRESS CAB: LT-462, PT-444,445,500		X			
HAGAN RACK 29			X		
HAGAN RACK 30			X		
MISC. RELAY RACK 50 - RELAYS	X				
SAFEGUARDS RACK 51 - RELAYS	X				
RPS RACK 53 - 57 RELAYS	X				
RPS RACK 58 - 62 REALYS	X				
SAFEGUARDS RACK 63 & 64	X				
CABINET FOR REACTOR	X				
RADIATION MONITORING SYSTEM					
ROD POSITION TRANSLATING	X				
ROD POSITION TRANSLATING	X				
ROD POSITION TRANSLATING	X				
ROD POSITION TRANSLATING	X				
RTGB	X	X	X	X	X
RVLIS INSTRUMENTATION RACK	X	X	X	X	X
SIGNAL ISOLATOR		X	X	X	X
SIGNAL PROCESSOR CABINET CH	X			X	
SIGNAL PROCESSOR CABINET CH	X			X	
HAGAN RACKS 14-25,27,28	X	X	X	X	X
AIR RECEIVER A TANK	X	X	X	X	X
AIR RECEIVER B TANK	X	X	X	X	X
BORIC ACID TANK A	X				
BORIC ACID TANK B	X				
CCW HX A	X				
CCW HX B	X				
COMPONENT COOLING WATER				X	
CONDESATE STORAGE TANK				X	
REGENERATIVE HX	X				
FUEL OIL DAY TANK A	X	X	X	X	X
FUEL OIL DAY TANK B	X	X	X	X	X
DIESEL OIL STORAGE TANK	X	X	X	X	X

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
EXCESS LETDOWN HEAT EXCH-	X	X	X	X	X
JACKET WATER EXPANSION	X	X	X	X	X
JACKET WATER EXPANSION	X	X	X	X	X
N2 ACCUMULATOR A	X			X	X
N2 ACCUMULATOR B	X			X	X
NON-REGEN. HX CCW SHELL SIDE				X	
PRESSURIZER STEAM SAMPLE HX		X	X	X	X
PRESSURIZER LIQUID SAMPLE HX		X	X	X	X
RHR HEAT EXCHANGER A				X	
RHR HEAT EXCHANGER B				X	
RWST TANK W/VENT				X	
REACTOR COOLANT SAMPLE HX			X	X	X
SG BLOWDOWN SAMPLE HX A			X	X	X
SG BLOWDOWN SAMPLE HX B			X	X	X
SG BLOWDOWN SAMPLE HX C			X	X	X
SPENT FUEL PIT HEAT					
STEAM DUMP N2 ACCUMULATOR				X	
SEAL WATER HX				X	
VOLUME CONTROL TANK				X	

Attachment 2: SWEL 1

Feature M=Modification or Replacement, A= A-46 Outlier	**Class**	**Reactor Reactivity Control**	**Reactor Coolant Pressure Control**	**Reactor Coolant Inventory Control**	**Decay Heat Removal**	**Containment Function**	**Location**	**Dry Env.**	**Hot Env.**	**Wet Env.**	**Cold Env.**	**Mild Env.**	**Harsh Env.**	**Inside**	**Outside**	**System**	**Modification***
VENTILATION DAMPER	0					X	TB	X	X			X		X		HVAC	
DAMPER FOR OUTSIDE AIR	0					X	TB	X	X			X		X		HVAC	
SW STRAINER A	0				X		SW		X	X	X	X			X	SW	
SW STRAINER B	0				X		SW		X	X	X	X		X		SW	
MOTOR CONTROL CENTER	1	X	X	X	X	X	RAB	X	X			X		X		AC	
MOTOR CONTROL CENTER	1	X	X	X	X	X	RAB	X	X			X		X		AC	M
480 V EMERGENCY BUS E1	2	X	X	X	X	X	RAB	X	X			X		X		AC	A
CONSTANT VOLTAGE	4	X	X	X	X	X	RAB	X	X			X		X		AC	A
AUX FEEDWATER MOTOR	5					X	RAB	X	X		X	X			X	AFW	M
AUX FEED WATER STEAM	5					X	TB	X	X			X		X		AFW	A
CCW PUMP A	5	X	X	X	X	X	RAB	X	X			X		X		CCW	M
CCW PUMP C	5	X	X	X	X	X	RAB	X	X			X		X		CCW	M
CHARGING PUMP B AND COOLER	5			X			RAB	X	X			X		X		CVCS	A
FUEL OIL TRANSFER PUMP A	5	X	X	X	X	X	YARD		X	X	X	X			X	AC	M
WASTE GAS COMPRESSOR A	5						RAB	X	X			X		X		CCW	
FUEL OIL TRANSFER PUMP B	6	X	X	X	X	X	YARD		X	X	X	X			X	AC	M
SW PUMP A	6				X		SW		X	X	X	X			X	SW	
AO DAMPER FROM OUTSIDE AIR	7					X	TB	X	X			X		X		HVAC	
AO DAMPER FROM OUTSIDE AIR	7					X	TB	X	X			X		X		HVAC	
FL CUT VALVE MDP-A	7					X	RAB	X	X			X		X		AFW	
PISTON OP VALVE MDP-B	7					X	RAB	X	X			X		X		AFW	
FL CONTROL VALVE	7						TB	X	X		X	X			X	AFW	
STM DMP N2 ACC INLT PCV AT N2	7				X		N2 SHED		X	X	X	X			X	SG	

Feature M=Modification or Replacement, A= A-46 Outlier	Class	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function	Location	Dry Env.	Hot Env.	Wet Env.	Cold Env.	Mild Env.	Harsh Env.	Inside	Outside	System	Modification*
STEAM DUMP N2 DISCH PRES	7				X		TB		X	X	X	X			X	SG	
STEAM DUMP N2 DISCH PRES	7				X		TB		X	X	X	X			X	SG	
STEAM DUMP N2 DISCH PRES	7				X		TB		X	X	X	X			X	SG	
PRESSURE REGULATOR INLET	7				X		TB	X	X			X			X	SG	
PRESSURE REGULATOR INLET	7				X		TB	X	X			X			X	SG	
PRESSURE REGULATOR INLET	7				X		TB	X	X			X			X	SG	
SG PORV FOR SG A	7		X		X		TB	X	X		X	X			X	SG	
SG PORV FOR SG C	7		X		X		TB	X	X		X	X			X	SG	
TEMP CONTROL VALVE (AOV) DG-B	7		X	X	X	X	RAB	X	X			X			X	SW	
TEMP CONT VALVE (AOV)	7		X				TB	X	X			X			X	SW	A
MOV - COMMON	8		X		X	X	TB	X	X			X			X	AFW	
MOV MDP-B	8		X		X	X	RAB	X	X			X		X		AFW	
MOV COMMON HEADER MDP	8		X		X		RAB	X	X			X		X		AFW	
BORIC ACID ISOLATION VALVE	8	X					RAB	X	X			X		X		CVCS	
SOLENOID VALVE TO RVI-1	8				X		TB	X	X			X			X	SG	
SOLENOID VALVE TO RVI-2	8				X		TB	X	X			X			X	SG	
SOLENOID VALVE TO RVI-3	8				X		TB	X	X			X			X	SG	
SOLENOID INLET VALVE TO RVI-1	8				X		TB	X	X			X		X		SG	
SOLENOID INLET VALVE TO RVI-3	8				X		TB	X	X			X		X		SG	
SOLENOID VALVE TO RVI-2	8				X		TB	X	X			X			X	SG	
SOLENOID INLET VALVE TO RVI-1	8				X		TB	X	X			X			X	SG	
SOLENOID INLET VALVE TO RVI-3	8				X	X	TB	X	X			X			X	SG	
SOLENOID VALVE TO RVI-2	8				X		TB	X	X			X			X	SG	

Feature M=Modification or Replacement, A= A-46 Outlier	Class	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function	Location	Dry Env.	Hot Env.	Wet Env.	Cold Env.	Mild Env.	Harsh Env.	Inside	Outside	System	Modification*
(MOV) 2A MS SUPPLY	8		X				TB	X	X			X			X	AFW	
(MOV) TR C MS SUPPLY (SDP)	8		X				TB	X	X			X			X	AFW	
PRESS CONTROL VALVE ACC A	8		X				RC	X	X			X		X		PORVs	
PRESS CONTROL VALVE ACC B	8		X				RC	X	X			X		X		PORVs	
MOV BUTTERFLY TB ISOL	8			X	X	X	RAB	X	X			X		X		SW	
EXHAUST FAN EDG-A HVAC	9	X	X	X	X	X	RAB	X	X			X		X		HVAC	M
RHR PUMP RM HVAC	9				X		NW OF	X	X			X		X		HVAC	
SUPPLY FAN	9	X	X	X	X	X	RAB	X	X			X		X		HVAC	
EDG A AIR DRYER SW SIDE	10	X	X	X	X	X	RAB	X	X			X		X		SW	A,M
CNTRL RM AIR HANDLNG UNIT &	10					X	RAB	X	X			X		X		HVAC	M
118V INSTRUMENT BUS 2	14	X	X	X	X	X	RAB	X	X			X		X		AC	
118V INSTRUMENT BUS 7B	14	X	X	X	X	X	RAB	X	X			X		X		AC	
STATION BATTERY "A"	15	X	X	X	X	X	RAB	X	X			X		X		DC	
BATTERY CHARGER "B"	16	X	X	X	X	X	RAB	X	X			X		X		DC	M
INVERTER-A	16	X	X	X	X	X	RAB	X	X			X		X		AC	M
EMERGENCY DIESEL GENERATOR	17	X	X	X	X	X	RAB	X	X			X		X		AC	M
PRESSURE SWITCH FOR FP/A	18					X	RAB	X	X			X		X		SIESMIC FIRE INTERACTI ON	
PRESSURE SWITCH FOR FP/B	18	X	X	X	X	X	TGB	X	X			X			X	SIESMIC FIRE INTERACTI ON	
DIFF PRESS SWITCH SW-A	18				X		SW		X	X	X	X			X	SW	
FLOW INDICATOR CONTROLLER -	18				X		NW OF	X	X			X		X		CCW	

Feature M=Modification or Replacement, A= A-46 Outlier	Class	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function	Location	Dry Env.	Hot Env.	Wet Env.	Cold Env.	Mild Env.	Harsh Env.	Inside	Outside	System	Modification*
FLOW INDICATING CONTROL-CSP DISCHARGE	18				X		RAB	X	X			X		X		CCW	
FLOW INDICATING CONTROLLER-	18				X		RAB	X	X			X		X		CCW	
FLOW TRANSMITTER FOR BAT TO CHARGING PUMP	18	X					RAB	X	X			X		X		CVCS	
CHARGING FLOW TRANSMITTER	18	X	X	X	X	X	RAB	X	X			X		X		CVCS	A
FL TRANS MDP-A	18				X		RAB	X	X			X		X		AFW	
FL TRANS MDPB	18				X		RAB	X	X			X		X		AFW	
AFW SDP TO SG-B FLOW TRANS	18				X		TB	X	X			X		X		AFW	
RCP-C SEAL LEAK-OFF HI RANGE FLOW TRANS	18			X			RC	X	X			X		X		CVCS	M
RCP-B SEAL LEAK-OFF HI RANGE FLOW TRANS	18			X			RC	X	X			X		X		CVCS	M
RCP-A SEAL LEAK-OFF HI RANGE FLOW TANS	18			X			RC	X	X			X		X		CVCS	M
FLOW TRANSMITTER CCW	18				X		RAB	X	X			X		X		CCW	M
COMMON FL TRANS	18				X	X	TB	X	X			X			X	AFW	
SI FLOW TRANSMITTER	18	X				X	RAB	X	X			X		X		SI	
AFW MDPS TO SG-B SQUARE	18				X		RAB	X	X			X			X	AFW	
INSTRUMENT RACK FOR PT-950,	18					X	RAB	X	X			X		X		PORVs	
DOST LEVEL INDICATOR SWITCH	18	X	X	X	X	X	YARD		X	X	X	X			X	AC	
POWER SUPPLY	18			X			NORTH	X	X	X	X	X			X	SI	
LEVEL TRANSMITTER BAT-A	18	X			X	X	RAB	X	X			X		X		CVCS	
CST LEVEL TRANS	18				X		TB	X	X		X	X			X	AFW	
CCW SURGE TANK LEVEL	18				X		RAB	X	X		X	X				CCW	
RWST LEVEL TRANS	18			X			NORTH	X	X	X	X	X			X	SI	

Feature M=Modification or Replacement, A= A-46 Outlier	Class	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function	Location	Dry Env.	Hot Env.	Wet Env.	Cold Env.	Mild Env.	Harsh Env.	Inside	Outside	System	Modification*
PRESSURE CONTROL - COMMON	18		X				RAB	X	X			X		X		CCW	
REMOTE RCP B TB DIFF. PRESS.	18			X			RAB	X	X			X		X		CVCS	
PRESS INDIC CONTLR FOR SG B	18				X		TB	X	X			X			X	SG	M
DG-A PRESSURE SWITCH	18				X		RAB	X	X			X		X		AC	
OUTLET PRESSURE SWITCH	18	X	X	X	X	X	RAB	X	X			X		X		AC	
PRESSURE SWITCH FOR FP/A	18					X	RAB	X	X			X		X		SIESMIC FIRE INTERACTI ON	
MDP-A TRIPS MDP-A	18				X		RAB	X	X			X		X		AFW	
VCT PRESSURE TRANSMITTER	18			X			RAB	X	X			X		X		CVCS	
MDP COMMON HEADER PRESS TR	18				X		RAB	X	X			X		X		AFW	
NORTH SW SUPPLY HDR PRESS	18				X		RAB	X	X			X		X		SW	
SG A MAIN STEAM LINE PRES	18				X		TB	X	X			X			X	SG	
SG C MAIN STEAM LINE	18				X		TB	X	X			X			X	SG	
TEMPERATURE INDICATOR	18	X					RAB	X	X			X	X			CVCS	
TEMPERATURE INDICATOR	18	X					RAB	X	X			X		X		CVCS	
LOW LEVEL AMPLIFIER	18		X		X		RAB	X	X			X		X		RPS/CRD/NI S (SCRAM)	
TEMPERATURE ELEMENT CCW	19				X		RAB	X	X			X		X		CCW	
PANEL A-65V: PSL-1616 & PSL-1684	20				X		RAB	X	X			X		X		CAB	
HAGAN RACKS 1-13,26	20	X	X	X	X	X	RAB	X	X			X		X		CAB	M
DGB-CONTROL SWITCHBOARD B	20	X	X	X	X	X	RAB	X	X			X		X		AC	M
INADEQUATE CORE COOLING	20				X		RAB	X	X			X		X		CAB	

Feature M=Modification or Replacement, A= A-46 Outlier	Class	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function	Location	Dry Env.	Hot Env.	Wet Env.	Cold Env.	Mild Env.	Harsh Env.	Inside	Outside	System	Modification*
INSTRUMENT RACK: PT-128,155, PI-155B & LT-484	20				X		RC	X	X			X		X		CVCS	
INSTRUMENT RACK: PT-125,154, PI-154B & LT-494	20				X		RC	X	X			X		X		CVCS	
SG-A LEVEL INDI AFWP RM PNL	20				X		RAB	X	X			X		X		AFW	
NUCLEAR INSTRUMENTATION	20	X					RAB	X	X			X		X		RPS/CRD/NIS (SCRAM)	A
HAGAN RACK 30	20			X			RAB	X	X			X		X		CAB	M
RTGB	20	X	X	X	X	X	RAB	X			X	X		X		CAB	A,M
SIGNAL PROCESSOR CABINET CH	20	X			X		RAB	X	X			X		X		RPS/CRD/NIS (SCRAM)	
AIR RECEIVER A TANK	21	X	X	X	X	X	RAB	X	X			X		X		AC	
AIR RECEIVER B TANK	21	X	X	X	X	X	RAB	X	X			X		X		AC	
BORIC ACID TANK B	21	X					RAB	X	X			X		X		CVCS	
CCW HX A	21	X					RAB	X	X			X		X		CCW	A
COMPONENT COOLING WATER	21				X		RAB	X	X			X		X		CCW	
FUEL OIL DAY TANK A	21	X	X	X	X	X	RAB	X	X			X		X		AC	
FUEL OIL DAY TANK B	21	X	X	X	X	X	RAB	X	X			X		X		AC	
JACKET WATER EXPANSION	21	X	X	X	X	X	RAB	X	X			X		X		AC	
JACKET WATER EXPANSION	21	X	X	X	X	X	RAB	X	X			X		X		AC	
RHR HEAT EXCHANGER A	21				X		RAB	X	X			X		X		RHR	M
SG BLOWDOWN SAMPLE HX B	21			X	X	X	RAB	X	X			X		X		CCW	
STEAM DUMP N2 ACCUMULATOR	21				X		TB	X	X			X			X	SG	

Attachment 3: Base List 2

B(STOP)/SFPSP	SPENT FUEL PIT SKIMMER PUMP PUSH BUTTON STOP SW
PB(STRT)/SFPSP	SPENT FUEL PIT SKIMMER PUMP PUSH BUTTON START SW
PB1-STOP/PMP-B	SFP COOLING PUMP B PUSH BUTTON STOP SWITCH 1
PB1-STRT/PMP-B	SFP COOLING PUMP B PUSH BUTTON START SWITCH 1
SS1/SFPC-PMP-A	SFP COOLING PUMP A START STOP SWITCH 1
SS2/SFPC-PMP-A	SFP COOLING PUMP A START STOP SWITCH 2
SS2/SFPC-PMP-B	SFP COOLING PUMP B START STOP SWITCH 2
SFPC-HTX	SPENT FUEL PIT HEAT EXCHANGER
SFPC-742	EMERGENCY COOLING CONNECTION
SFPC-793	SFPC PUMP LO LEVEL SUCTION
SFPC-796	SFPC PUMP "B" HI LEVEL SUCTION
SFPC-797	SFPC PUMP LO LEVEL SUCTION
SFPC-798A	PURIFICATION LOOP INLET
SFPC-798B	PURIFICATION LOOP OUTLET
SFPC-799A	PI-652 ISOLATION
SFPC-799D	SFPC PUMP DISCHARGE VENT
SFPC-802B	RWP PUMP DISCHARGE TO MIXED BED DEMIN
SFPC-802C	RC FILTER RETURN TO RCS VIA HIGH HEAD SI
SFPC-805A	RWP PUMP SUCTION FROM RWST
SFPC-819	SFP HX INLET
SFPC-820	SPENT FUEL PIT HEAT EXCHANGER OUTLET
SFPC-821A	SFP HX TUBE SIDE VENT
SFPC-821B	SFP HX TUBE SIDE DRAIN
SFPC-821C	SFPC PUMP "B" CASING DRAIN
SFPC-824J	SFPC PUMP "B" CASING VENT
SFPC-836A	SFPC PUMP "A" HI LEVEL SUCTION
SFPC-836B	SFPC PUMP "A" DISCHARGE
SFPC-837	SFPC PUMP "B" DISCHARGE
SFPC-838A	SFPC PUMP "A" CASING DRAIN
SFPC-838B	SFPC PUMP "A" CASING VENT
LA-651	SFP SPENT FUEL PIT HI/LO LEVEL ALARM
RO-2049	SPECTACLE FLANGE FOR LINE NO. 8-AC-151R-58
RO-2050	SPECTACLE FLANGE FOR LINE NO. 4-AC-151R-59
AC-204	SFP PUMP B DISCHARGE
AC-204A	SFP PUMP A DISCHARGE
AC-58	SFP PUMP B SUCTION
AC-58A	SFP PUMP A SUCTION
AC-59	SFP PUMP SUCTION
AC-60	SFP PUMPS DISCHARGE TO SFP HEAT EXCHANGER
AC-60A	SFP PUMP A DISCHARGE TO SFP HEAT EXCHANGER
AC-61	SFP HEAT EXCHANGER TO SPENT FUEL PIT
AC-63	SFP FILTER TO SPENT FUEL PIT
SFPC-PMP-A	SPENT FUEL PIT COOLING PUMP "A"
SFPC-PMP-B	SPENT FUEL PIT COOLING PUMP "B"
TE-651A	SPENT FUEL PIT TEMP MONITORING THERMOCOUPLE
TE-651B	SPENT FUEL PIT TEMP MONITORING THERMOCOUPLE
TIC-651A	SPENT FUEL PIT HI/LO TEMP IND SWITCH
TIC-651B	SPENT FUEL PIT HI/LO TEMP IND SWITCH
TW-653	THERMOWELL FOR SFP HEAT EXCHANGER RETURN LINE TEMP INDICATOR

Attachment 4: Rapid Drain Down List

Feature
SPECTACLE FLANGE

Attachment 5: SWEL 2

Feature	Building	Rapid Drain Down
SPENT FUEL PIT SKIMMER PUMP PUSH BUTTON STOP SW	FHB	
SFP COOLING PUMP B PUSH BUTTON STOP SWITCH 1	FHB	
SFP COOLING PUMP A START STOP SWITCH 1	FHB	
SFP COOLING PUMP A START STOP SWITCH 2	FHB	
SFP COOLING PUMP B START STOP SWITCH 2	FHB	
SFP SPENT FUEL PIT HI/LO LEVEL ALARM	FHB	
SPENT FUEL PIT COOLING PUMP "A"	FHB	
SPENT FUEL PIT TEMP MONITORING THERMOCOUPLE	FHB	
SPENT FUEL PIT TEMP MONITORING THERMOCOUPLE	FHB	
SPENT FUEL PIT HI/LO TEMP IND SWITCH	FHB	
SPENT FUEL PIT HI/LO TEMP IND SWITCH	FHB	
SPECTACLE FLANGE	FHB	X

Attachment 8: Peer Review Report

Robinson Nuclear Plant Seismic Walkdown Peer Review Report

Peer Review activities were performed on the Seismic Walkdown Program in addition to the Programmatic Controls / Oversight that were established for the project. A brief description of the Programmatic Controls / Oversight and Peer Review findings is provided below:

Programmatic Controls / Oversight

Programmatic Controls / Oversight were developed for the 2.3 Seismic Walkdowns and implanted at H.B. Robinson Nuclear Plant (RNP). A specification based on the EPRI guidance was established to control SWEL development and walkdown requirements. A specification was developed since EPRI 1025286 was written as guidance, whereas, the specification provided definitive criteria and control to avoid interpretation and promote consistency. The specification was inclusive of the EPRI guidance. A Quality Assurance (QA) person was present at the site during the inspection to assure form and specification compliance. Technical oversight was performed by the Project Manager (PM). The PM was onsite during the SWEL development and intermittently during the walkdowns and report generation phases of the project. An in-process review of work was performed during those intervals. Inspections at the four sites were being performed concurrently and lessons learned were relayed to the inspection teams at the other sites to determine if commonality was present within the fleet. These in-process reviews were performed through all phases of the project with the intent of meeting the intent of the EPRI guidance.

Peer Review

Separate from the programmatic controls implemented at the sites, Peer Review activities were performed on the seismic walkdown program that spanned from the development of the specification and Seismic Walkdown Equipment List (SWEL) through the physical walkdowns and ultimately to the report preparation and review. The Peer Review team concluded that the inspection program was performed in accordance with the guidance provided in EPRI 1025286, Seismic Walkdown Guidance for Resolution of Fukushima Near-Term Task Force Recommendation 2.3: Seismic, dated June, 2012. The Peer Review found the effort at RNP was performed in a competent manner and a very broad spectrum of components located throughout the power block were included in the program. The results were documented in a Duke (legacy Progress Energy) engineering change package.

Aspects of the program that were reviewed by the Peer Review justifying this statement are provided as follows:

Inspection Team

The Peer Review found Seismic Walkdown Engineers (SWE) performing the inspections were very experienced with a background in design engineering including seismic design at nuclear facilities dating back to design of the first generation nuclear power plants. SWEs had prior seismic walkdown experience at operating nuclear power plants, Department of Energy facilities, and other pertinent applications. Training consistent with the EPRI training was provided to all SWEs before any inspections were performed. The resumes of the SWEs were reviewed and it was determined that the SWEs were found to have qualifications that were consistent with the requirements of the regulatory guidance.

Selection of SWEL Items

The Peer Review concluded the process used to select SWEL items included both selected and diverse aspects. The list of equipment was obtained from the A-46 Safe Shutdown Equipment List (SSEL) and the appropriate screening filters identified in the EPRI guidance were applied. The number of items included in the SWEL represented an appropriate number of items in each equipment class when compared to the total number of items on the SSEL. The items that were individually selected typically were items that would have the most severe consequence in the event that the target item were to fail during the seismic event and resulted in components associated with the Emergency Diesel Generators, vital power, and heat removal systems, etc. being well represented. Other conditions given additional consideration included environmental and distribution into diverse structures; while items that are included in other programmatic inspections, (e.g. AOV, MOV, Appendix R, ASME Section XI Subsection IWE/IWL) were minimized. The process used to determine the SWEL items was determined to be in accordance with the EPRI guidance and adequately represents a diverse sample of the equipment required to perform the five safety functions.

The Peer Review confirmed site Operations experience was included in the review of the components to assure a representative distribution of equipment was included in the SWEL. Operations also performed preliminary walkdowns to determine if the components could be safely accessed. A selection/substitution criterion was established before the items were assessed and if items were judged inaccessible, then the substitution criteria was used. The Peer Review interviewed the personnel making the equipment selections and operations personnel to confirm an acceptable approach was used in selecting the equipment for sampling.

A sample of modifications performed at the site since the last IPEEE/A-46 inspection, previous A-46 outliers, and upgrades were reflected in the SWEL.

The SWEL contained 126 components in SWEL-1 and an additional 12 items in SWEL-2 totaling 138 total selected items for the combined SWEL. The number of items on SWEL-1 exceeds the recommended range of 90-120 items in the EPRI guidance and is considered conservative.

The process used to select the SWEL items, inclusion of Operations Personnel into the selection of the items, A-46 outliers and modifications were represented in the SWEL and the number and distribution of items was in accordance with the EPRI guidance and confirmed by the Peer Review utilizing the Peer Review Checklist for the SWEL.

Pre-Inspection Preparation

Peer Review was performed on the pre-inspection prepared walkdown packages which consisted of general configuration and structural drawings, anchorage detailing, and seismic demand on the anchorage and it was confirmed that these packages were available in the field during the inspection. The inspection packages were reviewed for thoroughness to the criteria and samples were selected to determine appropriateness of the information. At random intervals during the walkdown phase of the project, the SWEs were questioned to determine if they had been adequately prepared and specifically, they were questioned to determine if they knew the vertical and horizontal strong motion demand in the areas that they would be working. Additional instructions were provided during these intermediate assessments to affect subsequent inspections. The SWEs demonstrated that they had adequately prepared for the inspections prior to entering the field.

Conduct of Inspections

The Peer Review concluded the SWEs conducted field inspections with the walkdown packages "in-hand." The Seismic Walkdown Checklist (SWC) and Area Walk-By Checklist (AWC) were physically used in the field and place keeping practices were employed. The SWEL items were inspected; the forms were filled out in the field, and were reviewed by the SWEs before they left the area. As a result of conversations with the SWEs and Peer Review observations during the inspections, it was concluded that pertinent and thorough conversations occurred between the SWEs in the field to generally reach a consensus on a real time basis in the field. The inspections were performed in accordance with the EPRI guidance and within the confines of the controlling specification.

Review of Walkdown and Area Walk-By Checklist

The peer reviewers discussed the inspections with the SWEs prior to field implementation and sampled field reports during the inspections to determine adequacy of the inspection. The SWEs were asked to describe the encountered field conditions and the forms were reviewed to determine if the information was representative. The checklist was used predominately with hand written notes being used judiciously to reflect conditions. Intermediate guidance during the inspection process was provided and documentation was improved during the inspection phase of the project. Oversight provided during the walkdowns resulted in some of the first components being re-inspected to improve the field notes documentation.

The final documents (i.e., package including checklist, photographs, drawings, notes) were compared to the field notes with the QA representative reviewing 100% of the forms and the Peer Review reviewing over 30% of the forms. As a result of the Peer Review, there were some instances that required the SWE to obtain and/or delineate additional information in the walkdown packages. Once incorporated, the information presented on the forms was considered consistent with specification requirements and were judged representative of the field conditions.

Decisions for Entering Potential Adverse Seismic Conditions (PASCs) into CAP Process

The Peer Review concluded the identification of potential SSCs placed into the CAP process was in accordance with the controlling walkdown specification. The specification decision process delineated if items were to be initiated in CAP immediately or if they were to be evaluated in accordance with the NTTF 2.3 Seismic program. Site documentation, (e.g. original A-46/IPEEE inspection results, existing CAP Non-Conforming Record (NCRs), calculations, evaluations, etc.), was reviewed if the SWEs could not make an immediate acceptance determination. If the item was originally evaluated and marked as Unknown for PASC determination on the walkdown checklist and additional research did not yield a qualification of the existing condition, a NCR was initiated and the item was identified as a PASC. If additional information was located and the SWEs agreed on the status, the field notes were updated to reflect the acceptable condition. This was represented on the final walkdown and/or walk-by checklists, and no NCR would have been generated. The field notes were reviewed and evidence of documenting additional information was observed.

The Peer Review concluded that the process for evaluating identified issues in the field to determine if they were PASCs was in accordance with the EPRI guidance. The PASCs that were generated were reviewed and determined to meet the threshold for a NCR which was issued and documented in CAP.

Review of Licensing Basis

A Peer Review of the developed licensing basis evaluations, including the decisions for entering potentially adverse seismic conditions into RNP's CAP, was performed and found to be acceptable.

Review of Submittal Report

The Peer Review reviewed the submittal report and it was found to be consistent with the information provided in the inspection reports and the supporting documentation and met the objectives and requirements of the 50.54(f) letter.

Summary

The Peer Review concluded the program was controlled and performed in accordance with the guidance outlined in EPRI 1025286, Seismic Walkdown Guidance for Resolution of Fukushima Near-Term Task Force Recommendation 2.3: Seismic. The number of items in the SWEL met and exceeded the minimum requirements and was distributed appropriately among the various criteria. The types of issues encountered were appropriate for the seismic demand for the site.

Several significant modifications have been made at the site and these improvements were included in the component sampling.

Several housekeeping items were identified resulting in a number of work requests and CAP items. The site addressed most of the items during the inspections. A general impression of the SWEs was maintenance was being performed at the site and as a rule the site was conducting site work in accordance with the Station's Housekeeping procedures.

In conclusion, the Peer Review found the personnel involved in the inspections had sufficient knowledge of the site before the inspections and inspected the SWEL items in accordance with provided guidance. The conditions encountered and the degree of severity of the conditions indicates that RNP is conducting its maintenance and modification programs with consideration of seismic requirements. The performed inspections and assessments were conducted in accordance with the guidance provided in EPRI 1025286, Seismic Walkdown Guidance for Resolution of Fukushima Near-Term Task Force Recommendation 2.3: Seismic. The results were assessed to be reasonable and consistent and the number of PASCs is consistent with seismic demand for the region and age of the unit.

ENCLOSURE II

REPORT REVIEW BY SITE MANAGEMENT

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
RENEWED LICENSE NO. DPR-23

Report Review by Site Management

This submittal report is provided to the Nuclear Regulatory Commission in response to its request for information. Specifically, by letter dated March 12, 2012, the Staff requested licensees to provide information regarding recommendation 2.3 (Seismic) of the Near-Term Task Force Review of insights from the Fukushima Dai-Ichi Accident. The report provides information for the H. B. Robinson Steam Electric Plant, Unit No. 2 regarding the performance of seismic walkdowns to identify and address degraded, non-conforming or unanalyzed conditions and to verify the current plant configuration with the current seismic licensing basis. The information provided herein and the activities described in this report are consistent with the guidance provided by the Electric Power Research Institute's (EPRI) 2012 Technical Report 1025286 titled "Seismic Walkdown Guidance For Resolution of Fukushima Near-Term Task Force Recommendation 2.3: Seismic."

The signatures below document site management review of this document:

Signatures	Date
Site Fukushima Project Manager B. Steele	11\20\12
Site Responsible Seismic Engineer M. Ayoola	11\20\12
Site Engineering Management T. Rouns for Tim Rouns	11/26/12

Additionally, the Walkdown Report is submitted under cover letter signed by senior site management.