Letter Enclosures Contain Security-Related Information
Withhold in Accordance with 10 CFR 2.390

Duke Energy®

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10 CFR 50.54

November 27, 2012
Serial: BSEP 12-0127

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

Subject: Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Docket Nos. 50-325, 50-324
Recommendation 2.3 Seismic Walkdown of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident

References:

1. Response to 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, ADAMS Accession Number ML12053A340
2. Letter from David L. Skeen (USNRC) to Adrian P. Heymer (NEI), *Endorsement of Electric Power Research Institute (EPRI) Draft Report 1025286, "Seismic Walkdown Guidance," dated May 31, 2012, ADAMS Accession Number ML12145A529*

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued a Request for Additional Information (i.e., Reference 1) requesting licensees to provide information regarding Recommendation 2.3, Seismic, to support the evaluation of the NRC staff recommendation for the Near-Term Task Force (NTTF) review of the accident at the Fukushima Dai-ichi nuclear facility. By this letter, Carolina Power & Light Company (CP&L) submits the Brunswick Steam Electric Plant (BSEP) 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.

Enclosure 1 provides a signature sheet documenting site review of the BSEP, Unit 1 seismic walkdown report. The BSEP, Unit 1 seismic walkdown report is provided in Enclosure 2. Enclosure 3 provides a signature sheet documenting site review of the BSEP, Unit 2 seismic walkdown report. The BSEP, Unit 2 seismic walkdown report is provided in Enclosure 4.

The activities described and information provided in these reports are consistent with the guidance provided in the EPRI Report 1025286, *Seismic Walkdown Guidance: For Resolution of Fukushima Near-Term Task Force Recommendation 2.3: Seismic*, dated June 2012. The NRC endorsed the seismic walkdown guidance on May 31, 2012 (i.e., Reference 2)

Attachments 6 and 7 to Enclosures 2 and 4 Contain Security-Related Information
Withhold in Accordance with 10 CFR 2.390
Upon removal of Attachments 6 and 7 from Enclosures 2 and 4, this letter is decontrolled.

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Attachments 6 and 7 of the Unit 1 and Unit 2 seismic walkdown reports (i.e., Enclosures 2 and 4, respectively) have been determined to contain security-sensitive information. As such, those attachments should be withheld from public disclosure in accordance with 10 CFR 2.390.

This letter contains new regulatory commitments. Inspections for certain inaccessible equipment, as identified in the Unit 1 and Unit 2 seismic walkdown reports, will be completed and updated seismic walkdown reports will be submitted by September 30, 2016, and September 30, 2017, for BSEP, Units 1 and 2, respectively.

Please refer any questions regarding this submittal to Mr. Lee Grzeck, Manager – Regulatory Affairs, at (910) 457-2487.

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

Sincerely,

Michael J. Annacone

WRM/wrm

Enclosures:
1. Brunswick Steam Electric Plant, Unit 1 Seismic Walkdown Report Review by Site Management
2. Brunswick Steam Electric Plant, Unit 1 Seismic Walkdown Report **(Attachments 6 and 7 of this enclosure contain Security-Related Information – Withhold in Accordance with 10 CFR 2.390)**
3. Brunswick Steam Electric Plant, Unit 2 Seismic Walkdown Report Review by Site Management
4. Brunswick Steam Electric Plant, Unit 2 Seismic Walkdown Report **(Attachments 6 and 7 of this enclosure contain Security-Related Information – Withhold in Accordance with 10 CFR 2.390)**
5. List of Regulatory Commitments

cc (with enclosures):

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cc (without Attachments 6 and 7 of Enclosures 2 and 4):

Chair - North Carolina Utilities Commission
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Brunswick Steam Electric Plant, Unit 1
Seismic Walkdown Report Review by Site Management

The report titled *Brunswick Steam Electric Plant Unit 1 Seismic Walkdown Report* (i.e., Enclosure 2) is provided to the Nuclear Regulatory Commission in response to its request for information. Specifically, by letter dated March 12, 2012, the NRC 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 Brunswick Steam Electric Plant, Unit 1 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 therein 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 signatures below document site management review of this document:

Signatures	Date
Site Fukushima Program Manager Lenny Beller / L Beller	11/27/12
Site Seismic Engineer Dan Zebroski / DW	11/27/12
Site Design Engineering Manager Joe Price / Hal Sanders for J. Price	11/27/2012

Brunswick Steam Electric Plant, Unit 1
Seismic Walkdown Report

Attachments 6 and 7 Contain Security-Related Information
Withhold in Accordance with 10 CFR 2.390
Upon removal of Attachments 6 and 7, this document is decontrolled.

Brunswick Steam Electric Plant Unit 1 Seismic Walkdown Report

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Attachment 6: Seismic Walkdown Checklists **(Attachment Contains Security-Related Information)**
Attachment 7: Area Walk-By Checklists **(Attachment Contains Security-Related Information)**
Attachment 8: Peer Review Report

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 (NTTF) 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 Brunswick Steam Electric Plant (BSEP) Unit 1 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 NRC, in its letter dated May 31, 2012, endorsed the EPRI guidance document.

The 2.3 Seismic Walkdown inspections performed were non-intrusive visual inspections of primarily plant Seismic Class I structures, systems and components (SSCs). During the inspections, observed degraded, nonconforming, or unanalyzed conditions were identified and addressed through the corrective action program (CAP). Based on the EPRI guidance document, the list of SSCs for inspection were obtained through a systematic selection process to establish a broad, diverse and representative Seismic Walkdown Equipment List (SWEL). The SWEL was made up of two separate lists: SWEL 1 included 98 SSCs from various locations throughout the plant and SWEL 2 included a shorter specific list of six Spent Fuel Pool (SFP) SSCs.

The selection process for the SSCs combined with the inspection checklist attributes assessed design basis 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 (URI) A-46 "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors." Many of the same SSCs inspected for the IPEEE were re-inspected for the current 2.3 Seismic Walkdowns. Most of the SWEL items originated from the IPEEE Safe Shutdown Equipment List (SSEL). These programs occurred in the 1990s. The A-46 program reviewed equipment in the older nuclear plants with start-up dates prior to 1984 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 Assessment (SMA) programs to assess the plants capabilities to perform properly to a larger Review Level Earthquake (RLE). Modifications were also performed as a result, if necessary. For BSEP Unit 1, no IPEEE modifications were required.

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 found in the BSEP Updated Final Safety Analysis Report (UFSAR) and other design documents provides the description of those Systems, Structures, and Components (SSCs) that perform an important-to-safety function both during and after a Design Basis Earthquake (DBE). The following paragraphs describe the seismic design requirements of Seismic Classes I and II, geologic and seismic information, the site characteristics, earthquake characteristics, the seismic design requirements for SSCs and the various codes and standards used for seismic designs at BSEP Unit 1.

Certain plant structures must remain functional and/or protect vital equipment and systems, both during and following the most severe natural phenomenon postulated to occur at the site. In order to establish the loadings and loading combinations for which each individual structure was designed, buildings and their structural systems were separated into the following classes with respect to seismic design requirements

- Seismic Class I – structures and equipment are categorized as Class I if they are essential for safe shutdown or if failure could result in the release of radiation with dose consequences potentially exceeding the guidelines of 10 CFR 100.
- Seismic Class II – structures and equipment are categorized as Class II if their failure could not result in the release of radiation with dose consequences in excess of guidelines of 10 CFR 100.

Basic geologic and seismic information are presented in Section 2.5.1 of the BSEP UFSAR. The BSEP site is located approximately 2-1/2 miles north of Southport and 1-1/2 miles west of the Cape Fear River in southeastern North Carolina. Physiographically, the site is located on the Atlantic Coastal Plain about 90 miles southeast of the boundary between the flat lying deposits of the Coastal Plain and the folded formations of the Piedmont and Appalachian regions. This boundary is known as the Fall Line.

In the vicinity of the site, the Coastal Plain consists of approximately 1,500 feet of Cretaceous and younger deposits. In general, hard limestone exists from a depth of approximately 70 feet below existing ground surface and extends to a depth of 230 feet or more. The crystalline or metamorphic basement rock has been broadly warped into a tectonic feature known as the Cape Fear Arch.

From the standpoint of relief or physiography and structural geology, there are good reasons for the relative infrequency of earthquakes in the South Atlantic states. Earthquakes are most common in those areas characterized by narrow coastal plains, with recent mountains rising abruptly from near the coast and having narrow continental shelves extending seaward from the shore. North Carolina and adjoining states along the Atlantic Seaboard have a coastal plain 100 or more miles wide while mountains of ancient geologic origin occur another 100 miles inland. The continental shelf slopes gradually beneath the sea for another 50 to 100 miles before reaching its steeper margin. The world over, this combination of physiographic conditions is indicative of relative seismic stability.

The ultimate heat sink for BSEP Unit 1 is the Cape Fear River with the intake canal being the means by which the water is supplied to the intake structure. Cooling water flows to and from the plant through the canals discussed in Section 2.4.8 of the UFSAR. The circulating water system consists of an open intake canal, the circulating water and service water intake structures, the turbine condensers and piping, and a discharge canal terminating in the Atlantic Ocean. The intake canal is capable of supplying sufficient cooling water for all normal conditions or accident conditions such as a DBE. The only pumps which are required for support of safety-related equipment are the service water header pumps (i.e., intake structure) and the residual heat removal (RHR) service water pumps (i.e., reactor building). The service water intake structure is designed to functionally survive the DBE.

Site structures are founded on Class I backfill. The natural grade around the plant is approximately 20 feet above mean sea level. The area of the plant was excavated to an approximate elevation of -25 feet to dense sand and backfilled to grade to approximately +19 feet above mean sea level. The reactor building is founded on the dense sand and the other buildings are founded on the structural backfill that is founded on the dense sand. Beneath the dense sand is limestone bedrock.

The closest location of large earthquakes is around Charleston, SC located approximately 120 to 150 miles from the BSEP site. Based on possible earthquake effects from the Charleston area and from local faults within 20 to 25 mile radius, a ground acceleration of 0.08g was selected for the Operating Basis Earthquake (OBE). The DBE was selected to be twice the OBE horizontal ground acceleration of 0.16g, resulting in a high intensity VII (Modified Mercalli) seismic event at the site.

Seismic design requirements are based on an OBE with a horizontal ground acceleration of 0.08g, and DBE with horizontal ground acceleration of 0.16g. The vertical ground accelerations associated with the OBE and DBE were 2/3 and 4/3 of the corresponding OBE horizontal response spectra, respectively. The response spectrum is the smoothed 1940 North-South El Centro spectrum normalized by a factor of 0.08g/0.33g or 0.24 with a corresponding ground velocity of 4 in/sec for OBE. For DBE, the peak ground velocity is 5 in/sec with a peak ground acceleration of 0.16g.

Class I 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 the American Society of Mechanical Engineers (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; American Concrete Institute (ACI) 318-63 Building Code Requirements for Reinforced Concrete; and 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. This was selected to test equipment to the level that it would see in the maximum number of sites. The reduced load factors permitted by ACI 318 (Part IV-B) and the increase in allowable stresses permitted by the AISC Specification for loading combination which include earthquake loads were not used.
- The stresses that resulted from normal loads and design basis loss-of-coolant accident loads combined with the response to the DBE were limited so that no loss of function

occurred and the capability of making a safe and orderly plant shutdown was maintained.

Seismic Class I Instrumentation and Electrical Equipment must perform their safety function before, during and after a seismic event. The original design basis for this equipment required a blanket testing of horizontal 1.5g, vertical 0.5g and a frequency range of 5 to 33 Hz. The standard for seismic acceptance of equipment is Institute of Electrical and Electronics Engineers (IEEE) 344-1971. The standard divides the equipment into four main groups.

- Group A includes instruments and instrumentation and control devices including motor operators for valves.
- Group B includes enclosures, panels, and racks
- Group C includes primary pressure boundary devices
- Group D includes large electric motors

Tests were performed for Groups A and B above and analyses were performed for Groups C and D. Group C components are limited to SSCs which have a primary function of preservation of pressure boundary and are governed by the ASME Code requirements and were not tested. Group D are large electric motors and the analytical method of seismic qualification in accordance with IEEE 344-1971 was used.

Class II structures were generally designed in accordance with procedures of the Uniform Building Code for Zone 1. Class II equipment was designed for a static coefficient of 0.08g. The combined stresses from normal and earthquake loadings were limited to those permitted by applicable standards.

Seismic qualification of motor-operated valve operators has included accelerations in excess of 5g at the point of attachment to the valves and covering the frequency range of 5 to 33 Hz. This exceeds the anticipated acceleration values at the location of any Seismic Class I Limitorque operator and thus meets the requirements of IEEE 344-1971.

For Class I equipment furnished by suppliers other than General Electric (GE)Atomic Power Equipment Department, the seismic criteria considers a combination of normal and seismic loads with appropriate design margin to ensure that, during or immediately following an OBE, interruption or spurious operation shall not occur of controls in the normal or vital mode of operation. Likewise, that immediately following a DBE, interruption of operation of controls shall not occur. Complex equipment was subjected to vibratory motion which conservatively simulates a DBE. The complete system was energized during the test, accelerometers were mounted at various planes and all electrical relays and devices were monitored to detect their proper operation during the test. Analysis of non-complex equipment included determination of inertia and elastic characteristics, natural frequencies and nodal shapes. Either response spectrum technique or time-history approach was used to support the calculation of resulting stresses that determined equipment responses. Together, the testing and analysis ensure the proper performance of Class I equipment with regards to the established seismic criteria.

Revision 3 of the Seismic Qualification Utility Group (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 was used as an alternative to existing methods for the seismic design and verification of modified, new and replacement equipment. This alternative was not used for NRC Regulatory Guide 1.97 equipment unless justified on a case specific basis.

The program for resolution of USI A-46 and response to IPEEE seismic at BSEP was implemented with a single SSEL to address the requirement and guidance of both programs.

3.0 Personnel Qualifications

3.1 Equipment Selection Personnel

All resumes for personnel are included here except plant operations personnel that performed a supplemental role during the SWEL selection process.

3.1.1 Doyle Adams

Doyle G. Adams has over 36 years of engineering experience in both design and construction. This includes over 22 years nuclear experience at an operating nuclear facility. Nuclear experience includes design engineer, design engineering supervisor, SQUG/IPEEE qualifications and walkdown engineer, seismic equipment testing and qualification. He was also the lead responsible civil design engineer for Reactor Building design pressure uprate, Steam Generator and Reactor Head Replacement projects. Doyle Adams has a B.S. degree in Architectural Engineering degree in structures.

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 structures using ASME Section III, ASME/ANSI B31.1 and B31.3, and applicable nuclear plant procedures. Mr. Bamberger holds an A.D. in Mechanical Engineering Technology and has taken additional classes in Mechanical Engineering and Technology.

3.1.3 Leonard Beller (assisting plant operations personnel)

Leonard Beller has almost 30 years of nuclear experience at BSEP. Mr. Beller has Operations experience as a non-licensed operator and progressed to Control Room Supervisor/Senior Reactor Operator. He was Manager of Licensing and Regulatory Programs for seven years, and Operations Training Manager for five years. He is currently the Brunswick Fukushima Response Organization site lead. He has participated in the Institute of Nuclear Power Operations (INPO) Next Level Leadership for Managers, Peer Evaluator for three Operations Training Programs Accreditation Team Visits, and Host Peer Evaluator for Organizational Effectiveness INPO evaluation. Mr. Beller holds a B.A. degree in Geology.

3.1.4 John McIntyre (assisting plant operations personnel)

John McIntyre has over 39 years of experience in the design, construction and maintenance of nuclear power plants. He is currently assigned to the Brunswick Fukushima Response Organization as engineering lead. Work experience includes engineering, supervisory and management positions in Plant Equipment Performance, Mechanical and Structural Seismic Design, Design Control, Nuclear Oversight and Nuclear Plant Construction. Mr. McIntyre holds a B.S. degree in Mechanical Engineering and Associate Aeronautical & Space Engineering Technology.

3.2 Seismic Walkdown Engineers

3.2.1 Doyle Adams

See 3.1.1 above.

3.2.2 Gayuruddin Ahmed

Gayuruddin Ahmed has over 30 years of experience in Civil Engineering. He has experience in design of nuclear power plant structures and components with extensive use of finite element methods of frame analysis. He also has experience in ACI, AWS, and AISC code requirements and related design procedures of power plants for seismic conditions.

3.2.3 Martin Foster

Martin P. Foster has over 35 years of engineering experience most of which with various US nuclear plants. His experience includes seismic equipment qualification and testing for such equipment as control panels, instrument and battery racks, chargers and direct replacement of valves and instruments, seismic design of structures and equipment, seismic piping stress analysis and support design. Martin Foster has a B.S. degree in Civil Engineering.

3.2.4 Nazir Sheikh

Nazir Sheikh is a Registered Professional Engineer and has over 35 years engineering experience with over 30 years nuclear experience. Mr. Sheikh has been associated with nuclear design in nuclear piping design, concrete and steel structures using ACI 318, ACI 349, AISC, mechanical electrical equipment qualification and testing in accordance with IEEE-344. Nazir Sheikh has a B.S. degree in Civil Engineering and studies toward a M.S. degree in Mechanical Engineering.

3.2.5 James Curry

James Curry has over 12 years engineering experience with over six years experience at BSEP. He has nuclear experience as Systems Engineer for heating, ventilation, and air conditioning (HVAC) systems, seismic equipment qualifications, steel and concrete design and is SQUG trained and experienced. He also has four years Nuclear Navy experience in reactor operation and maintenance. Mr. Curry holds a B.S. degree in Civil Engineering.

3.2.6 Daniel Zebroski

Dan Zebroski is a Registered Professional Engineer and has over 30 years engineering experience associated with the design and construction of nuclear power plants. This includes the seismic qualification of equipment. He was responsible for the implementation and resolution of the SQUG verification of the seismic adequacy of all the safety related electrical and mechanical equipment at Ginna Station. At BSEP, he is the primary contact for all seismic issues, including seismic qualification training for other engineering personnel, seismic equipment qualification and dynamic analysis. He holds a B.S. degree in Civil Engineering from Drexel University. He has completed the SQUG walkdown screening and Seismic Evaluation and IPEEE Seismic Add-on courses and is currently registered as a professional engineer in the state of Connecticut.

3.2.7 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 at both AREVA and URS. He was involved in several projects including: Arkansas Nuclear One (ANO) 1 and 2 Control Room expansion, Equipment obsolescence, Dry Fuel Storage, and ANO-2 Containment redesign/design pressure up-rate. As a consultant, he served as the Civil/Structural Engineering Design Lead (EDL) for the U.S. European Pressurized Reactor (EPR) Design Certification (DC) and Combined Operating License (COL) projects providing a technical review of civil based licensing responses to clients or the NRC and project management. More recently, he has served as the Civil-Structural-Architect Discipline Manager for the detailed design phase of the U.S. Advanced Pressurized Water Reactor (US-APWR) including all aspects of the design including the site specific and design control document (DCD) seismic evaluations. Billy Alumbaugh has a M.S. and a B.S. degree in Civil Engineering.

3.3 Licensing Basis Reviewers

3.3.1 Doyle Adams

See 3.1.1 above

3.4 IPEEE Reviewers

3.4.1 Doyle Adams

See 3.1.1 above.

3.5 Peer Reviewers

3.5.1 Branko Galunic

Branko Galunic is the Chief Engineer for the Civil/Structural/Architectural discipline in the URS Nuclear Power Group. He has over 30 years of structural engineering experience in the Nuclear Power Industry. He has expertise in all aspects of static and seismic structural analyses of nuclear power generating structures and components. He is past chairman of Subcommittee B Anchorage, of ACI 349 Nuclear Safety Related Concrete Structures, and member of AISC/ANSI N690, Steel Safety-Related Structures. He is responsible for overall staffing and technical direction of the Civil/Structural/Architectural Discipline in URS/Washington Division's Nuclear Power Group. The staff of approximately 100 engineers and designers in four offices are working on new nuclear plant design for Mitsubishi Heavy Industry's (MHI) US-APWR power plant, modification/repair of the Crystal River containment structure delamination, Steel-Concrete wall design for MHI's Containment Internal structures, Post Fukushima seismic evaluation in accordance with NRC requirements, operating plant modification support and other related projects in the nuclear industry. Mr. Galunic has B.S. and M.S. degrees in Civil Engineering.

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, with over 15 years in management positions associated with construction, maintenance, modifications, including work package control, and operation of Department of Energy and NRC regulated facilities such as Nuclear Power Plants, Vitrification Facilities, Radioactive Waste Facilities, Gaseous Diffusion Facilities, and TRU Waste Characterization and Disposal. Mr. Wade is an American Society for Quality (ASQ) Certified Quality Auditor (CQA) 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 BSEP Unit 1 was based on the EPRI guidance document, Section 3. This selection process was conducted by Equipment Selection Personnel selecting SSCs based on selection criteria. During the process, plant operation staff assisted the Equipment Selection Personnel. The process, as described in the EPRI guidance document, involves the use of 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 one or more of the following five safety functions
    - Reactor reactivity control
    - Reactor coolant pressure control
    - Reactor coolant inventory control
    - Decay heat removal
    - Containment function
- Screen #4: Sample considerations (i.e., systems, major new/replacement, equipment types, environments, IPEEE enhancements)

The list of equipment resulting from Screen #3 is Base List 1. At BSEP, 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 IPEEE program. As discussed in EPRI Report 1025286, the first screen is intended to narrow 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. 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 IPEEE SSEL met the criteria for Screens #1, #2, and #3, and thus using the IPEEE SSEL was an appropriate starting point. Other SSCs were added to the IPEEE SSEL that were modified plant equipment verified to meet the criteria for Screens #1, #2, and #3. The IPEEE SSEL with these additional items served as Base List 1 (i.e., refer to Attachment 1).

Since BSEP is comprised of two nuclear units, there are shared or common equipment between the two units. Though the common equipment is capable of functioning for either unit, the equipment identifier is designated as Unit 1 or Unit 2. For the purposes of this inspection, those SSCs with a Unit 1 designation were included in the Unit 1 SWEL and those with a Unit 2 designator were included in Unit 2 SWEL. There is more common equipment associated with Unit 2 because Unit 2 was built prior to Unit 1. There are some instances, such as the emergency diesel generators, where the equipment might be aligned to serve Unit 1 as the primary function, but have a Unit 2 designator. This equipment was included on the Unit 2 SWEL.

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 (i.e., refer to Attachment 2). Considerations made for the creation of SWEL 1 are detailed in the sections below.

4.1.1 Equipment types/classes

A breakdown of the number of inspected items into the various equipment classes is provided in the following table.

Class No.	Equipment Included	Base List 1	Selected (SWEL 1)
0	Other	28	7
1	Motor Control Centers and Wall-Mounted Contactors	20	3
2	Low Voltage Switchgear and Breaker Panels	3	0*
3	Medium Voltage Metal-Clad Switchgear	6	2
4	Transformers	7	1
5	Horizontal Pumps	7	3
6	Vertical Pumps	4	3
7	Pneumatic-Operated Valves	42	8
8	Motor-Operated and Solenoid Operated Valves	92	18
9	Fans	12	4
10	Air Handlers	9	0*
11	Chillers	3	1
12	Air Compressors	0	0*
13	Motor Generators	2	1
14	Distribution Panels and Automatic Transfer Switches	28	7
15	Battery Racks	4	2
16	Battery Chargers and Inverters	5	2
17	Engine Generators	0	0*
18	Instrument Racks	76	17
19	Temperature Sensors	55	4

Class No.	Equipment Included	Base List 1	Selected (SWEL 1)
20	Instrument and Control Panels	54	15
21	Tanks and Heat Exchangers	14	0*
	Total	471	98

* Items from this class were walked down during the Unit 2 walkdown effort and are common between the two units.

An objective was to obtain equipment in every class; however, due to equipment inaccessibility or located in high radiation areas, some of the common equipment designated Unit 2 was used to meet the objective since it could serve the function for Unit 1 as well. For SWEL 1, five equipment classes, (i.e., 2, 10, 12, 17 and 21) met this need as further explained below. The walkdown information is included in the BSEP Unit 2 report.

- Equipment Class 2 (i.e., Low Voltage Switchgear and Breaker Panels): The one switchgear that was inspected for Unit 1 was a common switchgear to Unit 1 and Unit 2, but with a Unit 2 component designator.
- Equipment Class 10 (i.e., Air Handlers): The one air handling unit that was inspected for Unit 1 was a common air handling unit to Unit 1 and Unit 2, but with a Unit 2 component designator.
- Equipment Class 12 (i.e., Air Compressors): The one air compressor that was inspected for Unit 1 was a common air compressor to Unit 1 and Unit 2, but with a Unit 2 component designator.
- Equipment Class 17 (i.e., Engine Generators): The one emergency diesel generator that was inspected for Unit 1 was a common emergency diesel generator to Unit 1 and Unit 2, but with a Unit 2 component designator.
- Equipment Class 21 (i.e., Tanks and Heat Exchangers): The three tanks and heat exchangers that were inspected for Unit 1 were common tanks and heat exchangers to Unit 1 and Unit 2, but with Unit 2 component designators.

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	SSEL Total	Selected (SWEL 1)
Reactor reactivity control	69	25
Reactor coolant pressure control	88	14
Reactor coolant inventory control	70	20
Decay heat removal	264	58

Safety Function	SSEL Total	Selected (SWEL 1)
Containment function	19	12

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 Building, Control Building, Diesel Generator Building, Service Water Building (Intake Structure), Condensate Storage Tank Yard, and the Fuel Oil Tank Chamber Building. The SWEL 1 in Attachment 2 includes the building and location of each SSC.

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. The SWEL 1 in Attachment 2 includes the environment of each SSC.

4.1.5 Systems

During the SWEL 1 selection process, consideration was given to equipment of varying systems including the Control Rod Drive System, High Pressure Coolant Injection, Residual Heat Removal, and Pneumatic Nitrogen Systems among others. Table B-2 of Appendix E, "Safety Function-System Matrix for BWRs" of the EPRI guidance was consulted to ensure systems to support safety functions were included. Additionally, equipment in the Service Water System Building and equipment associated with the Service Water System that comprises emergency 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 contribution of individual items to overall risk was considered in the selection of the SSCs from the SSEL list for items to include in the SWEL 1. 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 probabilistic risk assessment 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. Some of this equipment was included while maintaining a balance with the other requirements of SWEL equipment selection.

4.1.7 IPEEE vulnerabilities

No seismic vulnerabilities were identified for the IPEEE seismic program. Therefore, no items were added to the SWEL 1 for IPEEE purposes.

4.1.8 Modified, replacement, and new equipment

A review of the plant modifications from 1995 (i.e., the initiation time of IPEEE) to 2002 and Engineering Changes dating 2002 to 2011 was performed to determine significant modifications to the plant. For BSEP Unit 1, 13 significant modifications were identified; three pertained to installing new equipment and 10 were equipment replacements. Since two items were previously identified on the list, 11 modifications were added to Base List 1 and identified as MOD 1 through MOD 11. Plant support personnel (i.e., systems, operations and engineering, etc.) identified the modifications to be included in the walkdowns.

4.1.9 Accessibility

Before and during 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 BSEP Base List 2, Rapid Drain-Down List, and SWEL 2 based on the EPRI guidance document which presents screening criteria to identify specific equipment that is unique to the SFP SSCs. As described in EPRI Report 1025286, Screen #1 and #2 limit SFP SSCs to those which have a Seismic Category I licensing basis and are capable of being visually reviewed in the plant. The equipment that resulted by applying these screens consisted of check and isolation valves on each of the two diffuser return lines in the Fuel Storage Pool Cooling and Filtering System. In addition, two spool pieces shared between the units and part of the Supplemental Spent Fuel Pool Cooling (SSFPC), were added to the Unit 1 list making a total six items in Base List 2 and is included in 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 ten feet above the spent fuel stored in the pool. There were six items identified on the Rapid Drain-Down list that were evaluated, included as Attachment 4. The six SSCs that were found to have a possible rapid drain-down capability include the skimmer surge tanks and skimmer piping, fuel pool cooling and filtering piping, SFP leak chase drains, fuel transfer gates, the SSFPC System, and the refueling canal when in a refueling condition. It was determined that these SSCs do not contain drain-down paths that would meet the SFP Rapid Drain-Down definition. Therefore, the SWEL 2 for Unit 1, included as Attachment 5, was the same as the Base List 2

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 Seismic Walkdown Engineers (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 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, including observations and PASCs, are 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 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.

Seismic walkdowns were performed on each SWEL 1 and 2 item that was accessible at the time of the walkdown effort for BSEP Unit 1. When SWEL items were inaccessible and an appropriate substitute was not available, the item was documented to be inspected at a future date as detailed in Section 5.6.

The seismic walkdowns focused on identifying PASCs for the SSCs listed on the SWEL using the following criteria for adverse anchorage conditions, adverse seismic spatial interactions, or other adverse seismic conditions:

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 10 diameters 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:

- A 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

For BSEP Unit 1, this alternate method was not implemented for any of the items.

The SWEs used engineering judgment to assess whether the anchorage is potentially vulnerable to seismic failure or malfunction (i.e., PASC). 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 component 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 which could occur that would 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 (i.e., either within the room or, for large rooms, within approximately 35 feet 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 and HVAC ducting, housekeeping items that could cause adverse seismic interaction, seismically induced fire and flooding/spray interactions as described below.

5.2.1 Seismically Induced Fire Interactions

The occurrence of a seismic event could create fire in multiple locations, simultaneously degrade fire suppression capabilities, and as a result prevent mitigation of fire damage to multiple safety-related functions.

During the seismic walkdowns, the engineers visually assessed any potential sources of fire (e.g., compressed flammable gas bottles, fuel tanks, other combustible material, etc.) located in the vicinity of the SWEL item to ensure it was adequately restrained. Additionally, potential interactions were assessed to determine if relative motion of high voltage equipment and adjacent support structures that have different foundations can cause high voltage busbars to short out against the grounded bus duct and cause a fire.

5.2.2 Seismically Induced Flood/ Spray Interactions

Seismically induced flooding events can potentially cause multiple failures of safety-related systems. Two examples of potential flooding sources are rupture of piping and vessels. Instances of concern include 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 located in the vicinity of the subject SSC to ensure they had adequate support and, therefore, were not likely to be a source of flooding or spray that could adversely affect the subject item. The items that were identified as potential conditions were documented. Any assessment and disposition of the effects were documented with the subject item. During the walkdowns and walk-bys, spray nozzle clearance with nearby lighting was inspected. It was determined that adequate clearances existed.

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 that criterion, the item was entered into the Corrective Action Program for resolution. Of the 98 SWEL items inspections and 30 area walk-bys, four PASCs were identified. For all PASCs identified, a licensing basis review was completed as stated in Section 6.0 below.

The following table summarizes the condition and status of each item judged as a potentially adverse seismic condition. The SW Building concrete floor condition described in the table below was identified prior to the seismic walkdowns and is currently being repaired. The remaining conditions were found to be in compliance with their seismic licensing basis.

Feature	Condition	Status of Resolution
Service Water (SW) pumps, strainers, and instrumentation	The SW Building concrete floor at the 20 foot elevation is spalled and cracked exposing rebar in many areas. This structural concrete slab, serving as the foundation for much equipment, is a PASC. This condition is documented in Nuclear Condition Report 150706 and the operability of the floor to perform its design function to support floor loading is evaluated as being acceptable in Operability Concern Review (OCR) Task 40, Sub-assignments 1, 2, 3, and 4.	A previously issued Nuclear Condition Report identifies corrective actions to repair the concrete and repairs are currently in progress.
Balance-of-Plant Annunciation Logic Cabinet	A coil of wire hanging down unrestrained inside the top of a cabinet.	Nuclear Condition Report initiated. Work Request initiated and condition corrected.
Small bore pipe near the RHR containment penetration area	Significant sag	Nuclear Condition Report initiated and evaluation of condition satisfactorily completed. Piping stress analysis meets code allowable stresses for applicable loading conditions. This line (i.e., 1-E11-5005-3/4-152) was analyzed in pipe stress analysis calculation SA-ECCS-Vents and was shown to meet code allowable stresses for all applicable loading conditions including seismic.

Feature	Condition	Status of Resolution
Open S-Hooks	"S" hooks securing overhead lighting at instrument racks 1- H21-P004,1-H21-P005 and 1-E11-C001A (1A RHRSW Booster Pump) are not squeezed closed to eliminate any gaps.	This condition is similar to the outlier condition that was resolved under the A-46 program. During A-46 program, Work Requests were initiated and S-Hooks were closed. This specific instance was not covered in the A-46 Program report. An extent of condition investigation was initiated and other open S-Hooks were identified. All cases were determined to be operable but were also closed to eliminate the gaps. Guidance documents for maintenance of fixtures were revised to ensure S-Hooks were closed during future maintenance activities. The extent-of-condition investigation is still underway.

Note: S-Hooks were determined to be PASCs, due to further engineering reviews after completion of the walkdowns.

5.4 Maintenance Assessment

The maintenance assessment, as requested in the 50.54(f) letter, 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 and minor housekeeping problems were noted. Almost all mobile equipment, tables, and tools were either secured properly or located in safe locations away from plant equipment. A few issues were noted with transient items such as cleaning equipment. Contamination was minimal. These indicators suggest that monitoring and maintenance processes and procedures are adequate. No adverse trends were identified in observations.

5.5 Planned or Newly Installed Changes

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

5.6 Inaccessible Items

There are 21 full or partial inspections that will need to be completed. Five of these are equipment that are located inside the primary containment (i.e., drywell) and were inaccessible at power operations. These five will be walked down during a refueling outage. The other 16 items were inaccessible panels and cabinets (i.e., some with anchorage on the interior). These include electrical equipment with doors or panels that were either locked because they either represented a personal safety hazard or contained a potential risk to affect the plant while at power. In these instances, the electrical equipment was walked down and the area walk-bys were performed without opening the panels or doors to inspect the inside. A breakdown of the equipment for future seismic walkdown is provided in the following table.

Item No.	Feature (Equipment ID)	Inspection Date
1	250VDC Motor Control Center (MCC) (1-1XDA)	March 2016
2	480V Motor Control Center (1-1XE)	March 2016
3	480V Motor Control Center (1-1PA)	March 2014
4	Emergency Power System Distribution Panel (1-1A-RX)	March 2014
5	Diesel Generator Building 125VDC Distribution Panel (1-1A-125VDC)	March 2014
6	Emergency Power System Distribution Panel (1-1B-SW)	March 2014
7	Reactor Protection System Power Distribution Panel (1-C71-P001)	March 2014
8	Emergency Power System Distribution Panel (1-1A-120V)	March 2014
9	Partial Winding Heater Cabinet for MCC 1PA (1-SW-PNL-VW8)	March 2014
10	125/250VDC Switchboard 1B (1-1B-250VDC)	March 2014
11	Control Rod Drive Accumulator Monitor Panel Bank 1 & 2 (1-H21-P003)	March 2013
12	Control Rod Drive Accumulator Monitor Panel Bank 3 & 4 (1-H21-P012)	March 2013
13	RHR and SW Pump 1C Motor Stator Winding High Temperature Relay (1-SW-TY-4889)	March 2013
14	125VDC Battery 1A-1 Charger (1-1A-1-125VDC-CHRGR)	March 2014
15	125VDC Battery 1B-1 Charger (1-1B-1-125VDC-CHRGR)	March 2014
16	Primary Steam Line 'D' Safety/Relief Valves (1-B21-F013J)	March 2014
17	HPCI Turbine Steam Supply Inboard Isolation Valve (1-E41-F002)	March 2014
18	Flow Transmitter for B21-F013D Position Indicator PM 84-180 (1-B21-FT-4160)	March 2014
19	Safety Relief Valve B21-F013C Leak Detection Temperature Element (1-B21-TE-N004C)	March 2014
20	Drywell Air Temperature Element (1-CAC-TE-1258-9)	March 2014
21	Distribution Panel Transformer MCC 1PA (1-1A-SW-XFMR)	March 2014

Item No.	Feature (Equipment ID)	Inspection Date
22	Service Water Intake Structure Fan Backdraft Damper (1-VA-1A-BDD-SW-BLDG)	See Note 2
23	Reactor Annunciator Cabinet (1-H12-P630)	See Note 2
24	Balance-of-Plant Annunciator Cabinet (1-XU-34)	See Note 2

Table Notes:

1. Items 3, 4, 5, 6, 9, 13, 14, and 15 were added as a result of Frequently Asked Questions.
2. These items were originally scheduled to be credited for anchorage inspection. However, upon inspection, some of the anchorage was not visible due to interferences. The anchorage that was visible did correlate with the as-built anchorage drawings from the USI A-46 and IPEEE walkdowns that were used during the Fukushima seismic walkdowns. Since 100% of the anchorage could not be inspected for these items they were not credited as part of the minimum 50% anchorage list. These items will be inspected at a future date, if the anchorage is accessible.

The expected inspection date is based on unit outages occurring in March of even years. Following completion of these inspections, the seismic walkdown report will be updated and submitted to the NRC by September 30, 2016.

6.0 Licensing Basis Evaluations

With the exception of the Service Water Building floor that is currently under repair to restore full qualification, potentially adverse seismic conditions that were identified during the seismic walkdowns and area walk-bys were found to meet the plant seismic licensing basis.

7.0 IPEEE Vulnerabilities Resolution Report

No seismic vulnerabilities were identified in the Brunswick IPEEE. Outlier conditions for USI A-46 have been resolved as addressed in a Carolina Power & Light Company letter to NRC on September 11, 1998.

8.0 Peer Review

The Peer Review is included in Attachment 8.

Attachments

Attachment 1: Base List 1

Attachment 2: SWEL 1

Attachment 3: Base List 2

Attachment 4: Rapid Drain-Down List

Attachment 5: SWEL 2

Attachment 6: Seismic Walkdown Checklists **(Attachment Contains Security-Related Information)**

Attachment 7: Area Walk-By Checklists **(Attachment Contains Security-Related Information)**

Attachment 8: Peer Review Report

Attachment 1: Base List 1

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
DG B SUPPL V PLENUM				X	
TURBINE				X	
L.O. COOLER LINE RELIEF VALVE				X	
SCRAM ACCUMULATOR	X				
NITROGEN BOTTLE & REGULATOR	X				
RHR HX 1A RELIEF VALVE				X	
N2 BOTTLES	X	X			
BACKUP PRESSURE RELIEF VALVE		X			
BACKUP N2 DISCHARGE RUPTURE DIAPHRAM	X	X			
BACKUP N2 PRESSURE RELIEF VALVE	X	X			
BACKUP N2 PRESSURE RELIEF VALVE	X	X			
BACKUP PRESSURE RELIEF VALVE	X	X			
NSW PUMP 1A STRAINER PRESS SWITCH				X	
NSW PUMP 1B STRAINER PRESS SWITCH				X	
BACKUP N2 IN LINE FILTER	X				
BACKUP N2 IN LINE FILTER	X				
SP STRAINER			X		
CS STRAINER SUCTION LINE			X		
HPCI/SP STRAINER			X		
NSW PUMP 1A STRAINER				X	
NSW PUMP 1B STRAINER				X	
HPCI COND PUMP DRAIN TO CLEAN RADWASTE ISOL VALVE			X		
HPCI COND PUMP DRAIN TO CLEAN RADWASTE ISOL VALVE			X		
MCC-1XDA			X		
MCC-1XDB				X	
SWITCHBOARD 1A				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
SWITCHBOARD 1B				X	
TRANSFORMER				X	
TRANSFORMER				X	
TRANSFORMER				X	
TRANSFORMER				X	
RHRP-1A				X	
RHRP-1C				X	
HPCI BOOSTER PUMP			X		
RHRSW BOOSTER PUMP 1A				X	
RHRSW BOOSTER PUMP 1B				X	
RHRSW BOOSTER PUMP 1C				X	
RHRSW BOOSTER PUMP 1D				X	
CSP-1A			X		
HPCI MAIN PUMP			X		
NUCLEAR SERVICE WATER PUMP 1A				X	
NUCLEAR SERVICE WATER PUMP 1B				X	
RHR ROOM COOLER RETURN ISOLATION				X	
RHR ROOM COOLER 1B OUTLET ISOLATION				X	
CS ROOM COOLER 1B OUTLET ISOLATION			X		
CS ROOM COOLER RETURN ISOLATION			X		
RHR 1D SEAL COOLER DISCHARGE				X	
RHR 1B SEAL COOLER DISCHARGE				X	
RHR PUMP 1A SEAL COOLING RETURN				X	
RHR PUMP 1C SEAL COOLING RETURN				X	
RHRSW PUMP HX AOV VALVE & SOLENOID				X	
RHRSW PUMP HX AOV VALVE & SOLENOID				X	
RHRSW PUMP HX AOV VALVE & SOLENOID				X	
RHRSW PUMP HX AOV VALVE & SOLENOID				X	
HPCI LO CLR PRESS CONTROL VALVE			X		

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
TURBINE STOP VALVE		X			
TURBINE CONTROL VALVE		X			
RHR HX TORUS CONTROL VALVE				X	
DISCHARGE DAMPER FOR RHR COOLING UNIT A				X	
DISCHARGE DAMPER FOR RHR COOLING UNIT B				X	
RHR DAMPER OPERATOR				X	
RHR DAMPER OPERATOR				X	
SCRAM DISCHARGE VOLUME VENT VALVE	X				
SCRAM DISCHARGE VOLUME VENT VALVE	X				
SCRAM INLET ISOLATION VALVE	X				
SCRAM OUTLET ISOLATION VALVE	X				
SCRAM DISCHARGE VOLUME VENT VALVE	X				X
SCRAM DISCHARGE VOLUME VENT VALVE	X				X
BACKUP N2 DISCHARGE PRESS CONTROL VALVE	X	X			
BACKUP N2 DISCHARGE PRESS CONTROL VALVE	X	X			
SAFETY RELIEF VALVE A & SOLENOID		X			
SAFETY RELIEF VALVE B & SOLENOID		X			
SAFETY RELIEF VALVE C & SOLENOID		X			
SAFETY RELIEF VALVE D & SOLENOID		X			
SAFETY RELIEF VALVE E & SOLENOIO		X			
SAFETY RELIEF VALVE F & SOLENOID		X			
SAFETY RELIEF VALVE G & SOLENOID		X			
SAFETY RELIEF VALVE H & SOLENOID		X			
SAFETY RELIEF VALVE J & SOLENOID		X			
SAFETY RELIEF VALVE K & SOLENOID		X			
SAFETY RELIEF VALVE L & SOLENOID		X			
NSW ROOM COOLER CROSS CONNECT				X	
TURBINE VACUUM BREAKER VALVE		X			
CSW TO RHRSW SUPPLY				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
RHRSW PUMP SUPPLY				X	
TURBINE STEAM SUPPLY VALVE		X			
HPCI TEST LINE/CST RETURN VALVE		X			
ISOLATION VALVE (MOV)			X		
SP COOLING ISOLATION VALVE			X		
SP SPRAY ISOLATION VALVE			X		
SP SUCTION VALVE			X		
ROOM COOLING SW ISOLATION				X	
RHR HT EXCH 1A OUT ISO				X	
RHR HT EXCH 1B OUT ISO				X	
SHUTDOWN CLG OUTBOARD SUCTION ISOL VALVE				X	
LPCI INBOARD INJECTION VALVE				X	
LPCI OUTBOARD INJECTION VALVE				X	
RHR HT EXCH OUT ISO				X	
HPCI INJECTION VALVE			X		
RHR HT EXCH 1A DISCH ISOL				X	
RHRSW PUMP SUCTION XTIE				X	
NSW TO RRCCW TRAIN B				X	
RBCCW SUPPLY ISO				X	
CS OUTBOARD INJECTION VALVE			X		
CS INBOARD INJECTION VALVE			X		
RHR/RWCU MOV (INBOARD)				X	
STEAM SUPPLY INBOARD ISOLATION VALVE			X		X
SHUTDWN CLG INBOARD SUCTION THROTTLE VLV				X	
RHR/RWCU MOV (OUTBOARD)				X	
NSW SUPPLY VALVE				X	
CSW PUMP 1C NSW DISCHARGE				X	
CSW PUMP 1A NSW DISCHARGE				X	
CSW PUMP 1B NSW DISCHARGE				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
NSW 1A DISCHARGE VALVE				X	
NSW 1B DISCHARGE YALVE				X	
RHR HX 1A OUTLET VALVE				X	
MIN FLOW BYPASS VALVE				X	
RHR HX 1A INLET VALVE				X	
RHR HX 1A BYPASS VALVE				X	
CS FULL FLOW TEST BYPASS VALVE			X		
CSW TO RHR PUMPS ISOLATION				X	
CS MIN. FLOW BYPASS VALVE TRAIN A			X		
CSP-1A SP SUCTION VALVE			X		
RHRP-1A SP SUCTION VALVE				X	
RHRP-1C SP SUCTION VALVE				X	
SHUTDOWN COOLING SUCTION VALVE				X	
SHUTDOWN COOLING SUCTION VALVE				X	
SHUTDOWN COOLING SUCTION VALVE				X	
SHUTDOWN COOLING SUCTION VALVE				X	
HPCI/CST SUCTION VALVE			X		
HPCI DISCHARGE VALVE			X		
HPCI TEST LINE/CST RETURN VALVE			X		
MIN FLOW BYPASS VALVE			X		
HPCI/SP SUCTION VALVE			X		
HPCI/SP SUCTION VALVE			X		
HPCI LO COOLING WATER VALVE			X		
TURBINE VACUUM BREAKER VALVE			X		
SP DISCHARGE ISOLATION VALVE			X		
DRVWELL SPRAY OUTBOARD ISOLATION VALVE			X		
SOLENOID VALVE TO DRYWELL INST	X				
SOLENOID VALVE				X	
SCRAM VALVE	X				

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
SCRAM VALVE	X				
SCRAM SOLENOID VALVE	X				X
SCRAM SOLENOID VALVE	X				X
STEAM SUPPLY OUTBOARD ISOLATION VALUE		X			
DRYWELL SOLENOID VALVE		X			X
BACKUP N2 SOLENOID VALVE	X				
BACKUP N2 DISCHARGE SOLENOID VALVE	X				
BACKUP N2 DISCHARGE SOLENOID VALVE	X				
BACKUP N2 SOLENOID VALVE	X				
SP SOLENOID VALVE FOR LSHN015A			X		
SP SOLENOID VALVE FOR LSHN015B			X		
SP SOLENOID VALVE	X				
CONTAINMENT ATMOSPHERE SOLENOID VALVE	X				
CONTAINMENT ATMOSPHERE SOLENOID VALVE	X				
PILOT SOLENOID VALVE FOR CV-F053A	X				
SOLENOID VALVE				X	
SP SOLENOID VALVE	X				
SP SOLENOID VALVE FOR LSHN015B		X			
SP SOLENOID VALVE FOR LSHN015A		X			
DG CELL EXHAUST FAN				X	
DG CELL EXHAUST FAN				X	
DG CELL EXHAUST FAN				X	
DG CELL EXHAUST FAN				X	
DG SUPPLY FAN				X	
DG SUPPLY FAN				X	
DG SUPPLY FAN				X	
DG SUPPLY FAN				X	
COIL FOR CS FAN COOLING UNIT A				X	
COIL FOR CS FAN COOLING UNIT B				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
COIL FOR RHR FAN COOLING UNIT A				X	
COIL FOR RHR FAN COOLING UNIT B				X	
RHRSW PUMP DISCHARGE TEMP				X	
MOTOR GENERATOR SET A	X				
MOTOR GENERATOR SET B	X				
POWER DIST. PNL W/RPS BUS A & B	X				
DISTRIBUTION PANEL 11A				X	
DISTRIBUTION PANEL 11B				X	
DISTRIBUTION PANEL 3A				X	
DISTRIBUTION PANEL 3AB				X	
DISTRIBUTION PANEL 3B				X	
DISTRIBUTION PANEL 1A				X	
DISTRIBUTION PANEL 1B				X	
DISTRIBUTION PANEL 7A				X	
DISTRIBUTION PANEL 7B				X	
BATTERY 1A-1				X	
BATTERY 1-A2				X	
BATTERY 18-1				X	
BATTERY 1B-2				X	
BATTERY CHARGER 1A-1				X	
BATTERY CHARGER 1A-2				X	
BATTERY CHARGER 18-1				X	
BATTERY CHARGER 1B-2				X	
SIGNAL CONVERTER FOR SV-F053A				X	
CST LO WATER LEVEL ACTUATION OF HPCI			X		
CST LO WATER LEVEL ACTUATION OF HPCI			X		
SUPP POOL HI WTR LEVEL ACTUATION OF HPCI		X			
RHR HX 1A PRESS DIFF SWITCH				X	
RHR HX 1A LO PRESSURE SWITCH				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
PRESSURE SWITCH				X	
SP PRESSURE TRANSMITTER		X			
SEAL COOLER LOW FLOW SWITCH				X	
SEAL COOLER LOW FLOW SWITCH				X	
SEAL COOLER LOW FLOW SWITCH				X	
SEAL COOLER LOW FLOW SWITCH				X	
LOW PRESSURE SWITCH				X	
SUPP POOL HI WTR LEVEL ACTUATION OF HPCI		X			
TURBINE EXHAUST RUPTURE DIAPHRAM		X			
TURBINE EXHAUST RUPTURE DIAPHRAM		X			
CORE SPRAY SYSTEM A INSTRUMENT RACK			X		
HPCI INSTRUMENT RACK			X		
RHR CHANNEL A INSTRUMENT RACK				X	
RHR CHANNEL B INSTRUMENT RACK				X	
RECIRC PUMP B INSTRUMENT RACK				X	
HPCI LEAK DETECT SYSTEM A INSTR RACK			X		
CR FLOW INDICATION OF TRAIN B ROOM CLG				X	
CSP RM COOLER DISCHARGE FLOW INDICATOR			X		
FAN/DAMPER LIMIT SWITCH				X	
FAN/DAMPER LIMIT SWITCH				X	
N2 ACCUMULATOR LEVEL SWITCH	X				
N2 ACCUMULATOR PRESSURE SWITCH	X				
JET PUMP INSTRUMENT RACK	X				
JET PUMP INSTRUMENT RACK	X				
CORE SPRAY/HPCI LEAK DETECT INSTR RACK			X		
CORE SPRAY/HPCI LEAK DETECT INSTR RACK			X		
PNS/BACKUP N2 LO PRESSURE SWITCH	X	X			
PNS/BACKUP N2 LO PRESSURE SWITCH	X				
DRYWELL PRESSURE TRANSMITER		X			X

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
DRYWELL PRESSURE TRANSMITTER		X			X
RNA/BACKUP N2 LO PRESSURE SWITCH	X				
RNA/BACKUP N2 LO PRESSURE SWITCH	X				
RHR HX TEMPERATURE ELEMENT				X	
RX PROTECTION & NSSS INSTR RACK		X	X		
RX PROTECTION & NSSS INSTR RACK		X	X		
BACKUP N2 PRESSURE TRANSMITTER	X	X			
BACKUP N2 PRESSURE TRANSMITTER	X	X			
BACKUP N2 PRESSURE TRANSMITTER	X	X			
BACKUP N2 PRESSURE TRANSMITTER	X	X			
RHR SW PUMPS INLET PRESSURE				X	
RHR SW PUMPS INLET PRESSURE				X	
RHR SW PUMPS INLET PRESSURE				X	
RHR SW PUMPS INLET PRESSURE				X	
RHRSWP DISCHARGE PRESSURE TRANSMITTER				X	
RHRSWP DISCHARGE PRESSURE TRANSMITTER				X	
RHRSWP DISCHARGE PRESSURE TRANSMITTER				X	
RHRSWP DISCHARGE PRESSURE TRANSMITTER				X	
NSW TO RBCCW FLOW TRANSMITIER				X	
CS TRAIN A LOW PRESSURE DSCH SWITCH			X		
DRYWELL PRESSURE TRANSMITTER		X			
SRV A FLOW TRANSMITTER				X	
SRV B FLOW TRANSMITTER		X			
SRV C FLOW TRANSMITTER		X			
SRV D FLOW TRANSMITTER		X	X		
SRV E FLOW TRANSMITTER		X	X		
SRV F FLOW TRANSMITTER		X	X		
SRV G FLOW TRANSMITTER		X	X		
SRV H FLOW TRANSMITTER		X	X		

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
SRV J FLOW TRANSMITTER		X	X		
SRV K FLOW TRANSMITIER		X	X		
SRV L FLOW TRANSMITIER		X	X		
NUCLEAR HDR PRESSURE SWITCH				X	
NUCLEAR HDR PRESSURE TRANSMITTER				X	
AOV FOR STRAINER BACKWASH				X	
AOV FOR STRAINER BACKWASH				X	
SOLENOID VALVE (SV)				X	
SOLENOID VALVE (SV)				X	
DIFFERENTIAL PRESSURE CONTROLLER				X	
DIFFERENTIAL PRESSURE CONTROLLER				X	
RHRSW PUMP DISCHARGE TEMP				X	
RHRSW PUMP DISCHARGE TEMP				X	
RHRSW PUMP DISCHARGE TEMP				X	
TEMPERATURE SENSOR				X	
INDICATOR BRIDGE				X	
TEMPERATURE SENSOR				X	
INDICATOR BRIDGE				X	
CS ROOM TEMPERATURE SENSOR				X	
CS ROOM INDICATING BRIDGE				X	
CS ROOM TEMPERATURE SENSOR				X	
CS ROOM INDICATING BRIDGE				X	
RHR RM TEMPERATURE SWITCH				X	
RHR RM TEMPERATURE SWITCH				X	
RHR RM TEMPERATURE SWITCH				X	
RHR RM TEMPERATURE SWITCH				X	
HPCI RM TEMPERATURE SWITCH				X	
HPCI RM TEMPERATURE SWITCH				X	
RHRSW PUMP DISCHARGE TEMP				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
RHRSW PUMP DISCHARGE TEMP				X	
RHRSW PUMP DISCHARGE TEMP				X	
RHRSW PUMP DISCHARGE TEMP				X	
DRYWELLTEMPERATURE ELEMENT					X
DRYWELL TEMPERATURE ELEMENT					X
DRYWELL TEMPERATURE ELEMENTS					X
DRYWELL TEMPERATURE ELEMENTS					X
DRYWELL TEMPERATURE ELEMENTS					X
DRYWELL TEMPERATURE ELEMENTS					X
DRYWELLTEMPERATURE ELEMENTS					X
DRYWELL TEMPERATURE ELEMENTS					X
TEMPERATURE SENSOR -SRV A		X			
TEMPERATURE SENSOR -SRV B		X			
TEMPERATURE SENSOR -SRV C		X			
TEMPERATURE SENSOR -SRV D		X			
TEMPERATURE SENSOR -SRV E		X			
TEMPERATURE SENSOR -SRV F		X			
TEMPERATURE SENSOR -SRV G		X			
TEMPERATURE SENSOR -SRV H		X			
TEMPERATURE SENSOR -SRV J		X			
TEMPERATURE SENSOR -SRV K		X			
TEMPERATURE SENSOR -SRV L		X			
DRYWELL TEMPERATURE ELEMENTS	X				
DRYWELL TEMPERATURE ELEMENTS	X				
DRYWELL TEMPERATURE ELEMENTS	X				
DRYWELLTEMPERATURE ELEMENTS	X				
DRYWELL TEMPERATURE ELEMENTS	X				
DRYWELL TEMPERATURE ELEMENT	X				
DRYWELL TEMPERATURE ELEMENT	X				

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
DRYWELL TEMPERATURE ELEMENT	X				
SP TEMPERATURE ELEMENT	X				
SP TEMPERATURE ELEMENT	X				
SP TEMPERATURE ELEMENT	X				
SP TEMPERATURE ELEMENT	X				
DRYWELL/SUPP POOL SIGNAL CONVERTER	X				
DRYWELL/SUPP POOL SIGNAL CONVERTER	X				
ENGINEERED SAFEGUARDS VERTICAL BOARD				X	
REACTOR CONTROL PANEL				X	
POWER RANGE NEUTRON MONITORING PANEL	X				
RPS TRIP SYSTEM A		X			
RPS TEST & MONITOR PANEL		X			
RPS TRIP SYSTEM B		X			
FEEDWATER & REACTOR RECIRC INSTR PANEL			X		
PROCESS INSTRUMENTATION CABINET			X		
NSSS TEMP REC & LEAK DET VERTICAL BOARD		X	X		
ROD POSITION INFORMATION SYSTEM CABINET	X				
ROD MANUAL CONTROL PANEL	X				
RHR A RELAY VERTICAL BOARD				X	
RHR RELAY VERTICAL BOARD				X	
HPCI VERTICAL BOARD RELAY			X		
BENCHBOARD AUXILARY RELAY CABINET			X		
CORE SPRAY "A" RELAY VERTICAL BOARD			X		
CORE SPRAY "B" RELAY VERTICAL BOARD			X		
REACTOR ANNUNCIATOR CABINET	X				
TERM CABINET FOR SYSTEMS SW, EB, RCC & BAT				X	
TERM CABINET FOR SYSTEMS SW, EB, RCC & BAT				X	
RX, CONT & TURB HVAC & TURB AUX CTRL PNL				X	
BOP RTG BOARD				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
TERMINATING CABINET DIV-I				X	X
TRIP CALIBRATION CABINET -ECCS DIVISION I				X	
TRIP CALIBRATION CABINET -ECCS DIVISION II				X	
FLUID FLOW DET CAB FOR SRV POSITION IND		X			
TSC/EOF COMPUTER ISOLATOR CABINET				X	
TSC/EOF COMPUTER ISOLATOR CABINET				X	
BOP PROCESS INSTR POWER SUPPLY CABINET				X	
TURBINE CONTROLLER		X			
1XDA-B26/1X-B5 XFER CONTACTOR PANEL				X	
CRD ACCUMULATOR MONITOR PNL BANK 1 & 2	X		X		
CRD ACCUMULATOR MONITOR PNL BANK 3 & 4	X		X		
REMOTE SHUTDOWN PANEL	X				
FLUID FLO OET PREAMP CAB FOR SRV POSITION		X			
MCC 1XA-DHB TO 1XD-DXS TRANSFER PANEL				X	
XFER CONTACTOR PANEL FOR E41-F002-M0		X			
E11-F008-MO ALTERNATOR STARTER PANEL				X	
XFER CONTACTOR PANEL FOR E41-F079-MO		X			
PARTIAL WINDING HTR CAB FOR MCC 1PA				X	
PARTIAL WINDING HTR CAB FOR MCC 1PB				X	
LUBE OIL COOLER				X	
RHR HX 1A				X	
SRV ACCUMULATOR A		X			
SRV ACCUMULATOR B		X			
SRV ACCUMULATOR C		X			
SRV ACCUMULATOR D		X			
SRV ACCUMULATOR E		X			
SRV ACCUMULATOR F		X			
SRV ACCUMULATOR G		X			
SRV ACCUMULATOR H		X			

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
SRV ACCUMULATOR J		X			
SRY ACCUMULATOR K		X			
SRV ACCUMULATOR L		X			
BOP ANNUNCIATOR LOGIC CAB FOR UA-5,-12,-14				X	
ELECTRIC HTR COIL - UNIT 1				X	
MOISTURE CONTROLLER				X	
TEMP CONTROLLER				X	
ELECTRIC HTR				X	
HEATING COIL TIMER				X	
SUPPLY FAN DSCH FLOW SWITCH				X	
SUPPLY FAN LIMIT SWITCH				X	
MCC 1CA				X	
MCC 1CB				X	
MCC 1XA				X	
MCC 1XA-2				X	
MCC 1XB				X	
MCC 1XB-2				X	
MCC 1XC				X	
MCC 1XD				X	
MCC 1X1				X	
MCC 1XM				X	
MCC 1XE				X	
MCC 1XF				X	
MCC 1XG				X	
MCC 1XH				X	
MCC 1PA				X	
MCC 1PB				X	
120/208VAC MAIN UPS DP				X	
480 VOLT UNIT SUBSTATION E5			X	X	X

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
480 VOLT UNIT SUBSTATION E6			X	X	X
50 KVA POWER SUPPLY				X	
50 KVA POWER SUPPLY				X	
SWITCHGEAR ASSEMBLY E1				X	
SWITCHGEAR ASSEMBLY E2				X	
120/24 AC TRANSFORMER				X	
DIST. PANEL TRANSFORMER MCC 1PA				X	
DIST. PANEL TRANSFORMER MCC 1PB				X	
ISOL VALVE				X	
DAMPER 1A-D OPERATOR				X	
DAMPER 1H-D OPERATOR				X	
NSW UNIT I SUPPLY TO DG1 JW				X	
NSW UNIT I SUPPLY TO DG2				X	
DG3 NSW UNIT 2 SUPPLY				X	
DG4 NSW UNIT 2 SUPPLY				X	
SUPPLY FAN SOL VALVES				X	
SOL VALVE FOR KS 1026				X	
AC SUPPLY FAN - UNIT 1				X	
AO DAMPER				X	
STEAM HUMIDIFIER				X	
AIR COOLED CONDENSER				X	
AO DAMPER - UNIT 1				X	
SUPPLY ISOL DAMPER				X	
COOLING COIL - UNIT 1				X	
SUBCOOLING CONDENSER				X	
CB PANEL -120VAC EMERG PWR				X	
480-120/208VAC PANEL				X	
CB PANEL -120VAC EMERG PWR				X	
CB PANEL -12OVAC EMERG PWR				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
480-120/208VAC PANEL				X	
CB PANEL -12OVAC EMERG PWR				X	
CB PANEL -120VAC EMERG PWR				X	
DISTRIBUTION PANEL 1E5				X	
DISTRIBUTION PANEL 1E6				X	
4B0-120/208VAC PANEL				X	
4B0-120/20BVAC PANEL				X	
CB PANEL -120VAC EMERG PWR				X	
CB PANEL -120VAC EMERG PWR				X	
CB PANEL -120VAC EMERG PWR				X	
RB PANEL -120VAC EMERG PWR				X	
RB PANEL -120VAC EMERG PWR				X	
RB PANEL -120VAC EMERG PWR				X	
SW PANEL- 120VAC EMERG PWR				X	
SW PANEL-120VAC EMERG PWR				X	
COOLING UNIT PRESSURE SWITCH				X	
TEMPERATURE TRANSMITTER				X	
TEMPERATURE ELEMENT				X	
TEMPERATURE TRANSMITTER				X	
MAIN CONTROL ROOM RTG BOARD	X				
DG2 ESS LOGIC CABINET				X	
DIV-I TERM CAB FOR XU-2 FOR SYSTEMS EB AND ED				X	
DIV-II TERMINAL CABINET FOR XU-2				X	
RIP TERMINAL CABINET		X			
RIP TERMINAL CABINET DIV-II		X			
RIP TERMINAL CABINET DIV-I		X			
001 ESS LOGIC CABINET				X	
C RM THERMOSTAT		X			
HX				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
1-E11-F020A DOWNSTREAM VENT VALVE				X	
CS PMP 1B SUPP POOL SUCT VLV MOT OP			X		
DG 4 ENG JKT WTR SERVICE WTR INLET ISV				X	
DRYWELL/SUPP POOL TEMP RECORDER	X				
1-VA-1A-EF-SWIS FAN BACKDRAFT DAMPER				X	
B32-F020 PILOT SOLENOID VALVE			X		
DW FLOOR DRN INBRD ISV SOL VLV			X		
CB INTAKE AIR PLENUM CHLORINE DETECTOR				X	
CB INTAKE AIR PLENUM CHLORINE DETECTOR				X	
LOOP A LOCAL DISCHARGE PRES IND			X		
CS PMP 1A SUPP POOL SUCT VLV MOT OP			X		

Attachment 2: SWEL 1

Feature	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
HCU NITROGEN/WATER ACCUMULATOR	0	X					RB	X	X			X		X		CRD	
CRD PORTABLE ACCUM CHARGING SYS N2 TANK	0	X					RB	X	X			X		X		CRD	
250VDC MCC	1			X			RB	X	X			X		X		DC	
HCU SCRAM WTR INLET ISV AIR OPERATOR	7	X					RB	X	X			X		X		CRD	
HCU SCRAM WTR OUTLET ISV AIR OPERATOR	7	X					RB	X	X			X		X		CRD	
SCRAM VALVES PILOT AIR HEADER SOLENOID VALVE	8	X					RB	X	X			X		X		CRD	
SDV VENT & DRAIN PILOT SOL VLV	8	X				X	RB	X	X			X		X		CRD	
SDV VENT & DRAIN PILOT SOL VLV	8	X				X	RB	X	X			X		X		CRD	
CAC-PT-2685, 3341 & 4176 DW SV	8		X			X	RB	X	X			X		X		CAC	
EMERG PWR SYS DISTRIBUTION PANEL	14				X		RB	X	X			X		X		AC	
HCU ACCUMULATOR HIGH LEVEL SW	18	X				X	RB	X	X			X		X		CRD	
DRYWELL PRESSURE TRANSMITTER	18		X				RB	X	X			X		X		CAC	

Feature	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
JET PUMP INSTRUMENT RACK	18	X					RB	X	X			X		X		AUX CONTR BRD	
JET PUMP INSTRUMENT RACK	18	X					RB	X	X			X		X		AUX CONTR BRD	
CORE SPRAY/HPCI LEAK DETECTION RACK	18			X			RB	X	X			X		X		AUX CONTR BRD	
PNEUMATIC N2 DIV I SUP LO PRESS SW	18	X	X				RB	X	X			X		X		PNS	
CRD ACCUMULATOR MONITOR PNL BANK 1 & 2	20	X		X			RB	X	X			X		X		AUX CONTR BRD	
CRD ACCUMULATOR MONITOR PNL BANK 3 & 4	20	X		X			RB	X	X			X		X		AUX CONTR BRD	
B32-F020 PILOT SOLENOID VALVE	8			X			RB	X	X			X		X		REACT RECIRC	X
HPCI BOOSTER PUMP	5			X			RB	X	X			X		X		HPCI	
1-E11-F020A DOWNSTREAM VENT VALVE	8				X		RB	X	X			X		X		RHR	X
DIV II NITROGEN BACKUP SUPPLY PRV	0	X	X				RB	X	X			X		X		IA	

Feature	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
DIV II N2 BACKUP SUP HDR RUPT DIAPHRMS/ DISCS	0	X	X				RB	X	X			X		X		IA	
480V MCC 1XE	1				X		RB	X	X			X		X		AC	
RHR SW BOOSTER PUMP 1C	5				X		RB	X	X			X		X		RHR	
SDV INBD VENT VLV AIR OPERATOR	7	X				X	RB	X	X			X		X		CRD	
SDV OUTBD VENT VLV AIR OPERATOR	7	X				X	RB	X	X			X		X		CRD	
DIV II NITROGEN BACKUP SUPPLY PCV	7	X	X				RB	X	X			X		X		IA	
DIV I NITROGEN BACKUP SUPPLY PCV	7	X	X				RB	X	X			X		X		IA	
HPCI INJECTION VALVE	8			X			RB	X	X			X		X		HPCI	
NUCLEAR HDR TO RBCCW HX ISOL VLV MO	8				X		RB	X	X			X		X		SW	
RBCCW HX SW ISOL VALVE MOTOR OPERATOR	8				X		RB	X	X			X		X		SW	
NUCLEAR SERVICE WATER SUPPLY VALVE	8				X		RB	X	X			X		X		SW	
RX PROTECTION & NSSS INSTR RACK	18		X	X			RB	X	X			X		X		AUX CONTR BRD	

Feature	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
DIV II N2 BACKUP SUPPLY PRESS XMTR	18	X	X				RB	X	X			X		X		IA	
RHR SW PMP 1A SUPP HDR PRESS SWITCH PM 82-129	18				X		RB	X	X			X		X		SW	
RHR SW PMP 1C SUPP HDR PRESS SWITCH PM 82-129	18				X		RB	X	X			X		X		SW	
DIESEL GEN BLDG 125VDC DIST PNL 1A	14				X		DGB	X	X			X		X		DC	
DG 4 ENG JKT WTR SERVICE WTR INLET ISV	8				X		DGB	X	X			X		X		SW	X
RHR SW PMP 1C MOTOR STATOR WDG HIGH TEMP RELAY Q	19				X		DGB	X	X			X		X		SW	
230KV SWYD RLY HSE 125VDC DIST PANEL 7A	14				X		SWYD RLY HSE	X	X			X		X		DC	
NUCLEAR HDR SERV WTR PUMP 1A	6				X		SWB	X	X			X		X		SW	
NUCLEAR HEADER PUMP 1B	6				X		SWB	X	X			X		X		SW	
NUCLEAR SW PUMP 1B STRAINER	0				X		SWB	X	X			X		X		SW	
NUC SW PMP 1A DISCHARGE STRAINER PCV	18				X		SWB	X	X			X		X		SW	
EMERG PWR SYS DIST PANEL	14				X		SWB	X	X			X		X		AC	

Feature	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
NUC SW PMP 1A DISCH STR DIFF PRESS HI	0				X		SWB	X	X			X		X		SW	
NUC SW PMP 1B DISCH STR DIFF PRESS HI	0				X		SWB	X	X			X		X		SW	
480V MCC 1PA	1				X		SWB	X	X			X		X		AC	
NUCLEAR SW PUMP 1B DIFF PRESS CONTROLLER Q	18				X		SWB	X	X			X		X		SW	
NUCLEAR HEADER PRESS TRANSMITTER	18				X		SWB	X	X			X		X		SW	
PARTIAL WINDING HTR CAB FOR MCC 1PA	20				X		SWB	X	X			X		X		SW	
1-VA-1A-EF-SWIS FAN BACKDRAFT DAMPER	9				X		SWB	X	X			X		X		HVAC	X
NUC HDR WTR PUMP A DISCHARGE VALVE MO	8				X		SWB	X	X			X		X		SW	
NUC HDR SW PMP B DISCHARGE VALVE MO	8				X		SWB	X	X			X		X		SW	
125VDC BATTERY 1A-1 CHARGER	16				X		CB	X	X			X		X		DC	
125 VDC BATT FOR 125/250 VDC DISTR SWTCHBRD 1A	15				X		CB	X	X			X		X		DC	
125/250VDC SWITCHBOARD 1B	3				X		CB	X	X			X		X		DC	

Feature	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
125VDC BATTERY 1B-1 CHARGER	16				X		CB	X	X			X		X		DC	
125 VDC BATT FOR 125/250 VDC DISTR SWTCHBRD 1B	15				X		CB	X	X			X		X		DC	
HIGH INERTIA AC MOTOR-GENERATOR SET 'B'	13	X					CB	X	X			X		X		RPS	
STANDBY 50KVA UPS POWER SUPPLY (PM 86-011)	3				X		CB	X	X			X		X		AC	
REACTOR PROTECTION SYS POWER DIST PANEL	14	X					CB	X	X			X		X		RPS	
EMERG PWR SYS DIST PANEL 1A	14				X		CB	X	X			X		X		AC	
REACTOR CONTROL PANEL	20				X		CB	X	X			X		X		RTGB	
ENGINEERED SAFEGUARDS VERTICAL BOARD	20				X		CB	X	X			X		X		RTGB	
DRYWELL/SUPP POOL TEMP RECORDER	20	X				X	CB	X	X			X		X		CAC	X
HPCI VERTICAL RELAY BOARD	20			X			CB	X	X			X		X		RTGB	
TERMINATING CABINET DIV I	20				X	X	CB	X	X			X		X		RTGB	
RIP TERMINAL CABINET	20		X			X	CB	X	X			X		X		RTGB	
NSSS TEMP REC & LEAK DET VERTICAL BOARD	20		X	X			CB	X	X			X		X		RTGB	

Feature	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
FEEDWATER & REACTOR RECIRC INSTR PANEL	20			X			CB	X	X			X		X		RTGB	
DRYWELL/SUPP POOL SIGNAL CONVERTER	20	X				X	CB	X	X			X		X		CAC	
CONTROL BLDG 125VDC DISTRIBUTION PANEL 11B	14				X		CB	X	X			X		X		DC	
REACTOR ANNUNCIATOR CABINET	20	X					CB	X	X			X		X		ANNUNC	
BOP PROCESS INSTR POWER SUPPLY CABINET	20				X		CB	X	X			X		X		RTGB	
BOP ANNUNCIATOR LOGIC CAB FOR UA-5,-12,-14	20				X		CB	X	X			X		X		RTGB	
CORE SPRAY FULL FLOW TEST BYPASS VALVE	8			X			RB	X	X			X		X		CS	
CORE SPRAY SYSTEM A INSTRUMENT RACK	18			X			RB	X	X			X		X		AUX CONTR BRD	
LOOP A LOCAL DISCHARGE PRES IND	18			X			RB	X	X			X		X		CS	X
CS PMP 1A SUPP POOL SUCT VLV MOT OP	8			X			RB	X	X			X		X		CS	X
CORE SPRAY PUMP 1A	6			X			RB	X	X			X		X		CS	

Feature	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
CS PMP RM 1A CLR SW OUTLT VLV AO	7			X			RB	X	X			X		X		SW	
CS PMP 1B SUPP POOL SUCT VLV MOT OP	8			X			RB	X	X			X		X		CS	X
RHR SW BOOSTER PUMP 1B	5				X		RB	X	X			X		X		RHR	
CONV-NUC HDR CROSS-TIE VALVE MO	8				X		RB	X	X			X		X		SW	
RBCCW HX SERVICE WATER INLET FLOW TRANSMITTER	18				X		RB	X	X			X		X		SW	
RHR SW PUMP 1D TEMP SWITCH HIGH	19				X		RB	X	X			X		X		SW	
VA-1D-CU-CB SUBCOOLING CONDENSER	11				X		CB		X	X	X				X	HVAC	
CB INTAKE AIR PLENUM CHLORINE DETECTOR	9				X		CB	X	X			X		X		HVAC	X
CB INTAKE AIR PLENUM CHLORINE DETECTOR	9				X		CB	X	X			X		X		HVAC	X
CB CONTROL ROOM SUPPLY AIR FAN MOTOR	9				X		CB		X	X	X				X	HVAC	
PRI STEAM LINE "D" SAFETY/RELIEF VALVE (ADS)	7		X				RB/DW		X	X			X	X		RX VESSEL & INT	

Feature	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
HPCI TURBINE STEAM SUPPLY INBOARD ISOLATION VALVE	8			X		X	RB/DW		X	X			X	X		HPCI	
FLOW TRANSMITTER FOR B21-F013D POS INDICATOR PM 84-180	18		X	X			RB/DW		X	X			X	X		AUTO DEPRES	
SRV B21-F013C LEAK DETECT TEMPERATURE ELEMENT	19		X				RB/DW		X	X			X	X		AUTO DEPRES	
DRYWELL AIR TEMPERATURE ELEMENT	19					X	RB/DW		X	X			X	X		CAC	
DIST PANEL TRANSFORMER MCC 1PA	4				X		NORTH WALL	X	X			X		X		AC	

Attachment 3: Base List 2

Feature
FUEL STORAGE POOL RECIRCULATION VALVE (DIFFUSER ISOLATION) 1-G41-V10
FUEL STORAGE POOL CLEAN-UP RETURN DIF CHK VLV 1-G41-V24
SUCTION PIPING VORTEX BREAKER
DISCHARGE PIPING VORTEX BREAKER
FUEL STORAGE POOL CLEAN-UP RETURN DIF CHK VLV 1-G41-V8
FUEL STORAGE POOL RECIRCULATION VLV (DIFFUSER ISOLATION) 1-G41-V9

Attachment 4: Rapid Drain-Down List

Feature	Evaluation
Skimmer Surge tanks and Skimmer piping	A seismic induced failure of the Skimmer Surge Tanks, the connected piping, or the scupper drains would not result in drainage of the SFP water below normal pool level.
Fuel Pool Cooling and Filtering System (FPCFS)	The FPCFS removes water from the surface of the pool, runs it through filters and heat exchangers and returns it to the bottom of the pool through two pipes, each of which have a set of check and globe valves located at the pool side as they enter the pool. These design features in combination with their arrangement, piping class, their location with respect to one another and quarterly testing provides the basis for concluding that this system does not provide the possibility of a rapid drain-down event.
Leak Chase Drains	The leak chase drain channel system is located behind the SFP liner plate. The SFP liner is part of the Seismic Class 1 SFP structure and, therefore a seismic induced failure need not be postulated. The system has the capability of detecting leakage and manually isolating for any induced line leak (e.g., puncture, weld defect, etc.) which would be within the make-up capability of the system. This provides the basis for concluding that this system does not provide the possibility of a rapid drain-down event.
Fuel Transfer Gates	The fuel transfer gates are part of the Seismic Class 1 structure and have inflatable seals that are maintained via compressed air with a nitrogen bottle backup supply. The nitrogen is attached to the supply system through a check valve to maintain pressure in the system should air pressure be lost. This redundancy provides the basis for concluding that fuel transfer gates and their associated drain system does not provide the possibly of a rapid drain-down event.
Supplemental Spent Fuel Pool Cooling System (SSFPCS)	Temporarily installed spool pieces are used to connect the permanent suction and return piping and extend six feet and ten feet, respectively, into the spent fuel pool. Both spool pieces have two inch holes in them located just below the normal level of the spent fuel pool and above the minimum Technical Specification water level to prevent inadvertent siphoning of water out of the spent fuel pool. Because of these design attributes, this system is not a candidate for rapid drain down.
Refueling Condition	The Refueling condition was investigated for the time when the reactor well is flooded for refueling. This condition was not considered a rapid drain down possibility due to the protections in place during refueling, the addition of a Class 1 plate for volume protection, the Class I piping, installed drain plugs, and the bellows design.

The items on this list have been evaluated as not having the ability to cause a rapid drain down. Therefore, there are no rapid drain down items to add to the SWEL 2 list.

Attachment 5: SWEL 2

Feature	Building	Rapid Drain Down
FUEL STORAGE POOL RECIRCULATION VALVE (DIFFUSER ISOLATION) 1G41-V10	RB	No
FUEL STORAGE POOL CLEAN-UP RETURN DIF CHK VLV 1-G41-V24	RB	No
SUCTION PIPING VORTEX BREAKER	RB	No
DISCHARGE PIPING VORTEX BREAKER	RB	No
FUEL STORAGE POOL CLEAN-UP RETURN DIF CHK VLV 1-G41-V8	RB	No
FUEL STORAGE POOL RECIRCULATION VLV (DIFFUSER ISOLATION) 1-G41-V9	RB	No

Attachment 8: Peer Review Report

Brunswick Nuclear Power Plant Unit 1 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 Brunswick Nuclear Plant (BNP-U1). 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 BNP-U1 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. Justification was adequately provided for equipment classes that could not have a representative component inspected. 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 vital power, and heat removal systems, etc. being well represented. BNP is a two unit site and some systems are shared between U1 and U2 which were identified as common. The common SSCs typically had U2 equipment numbers which resulted in a slight skewing of equipment on the SWEL and some equipment classes had no representative samples. Review of the U1 and U2 SWEL list did reveal common equipment could be attributed to the U1 SWEL where no specific U1 equipment was identified. Based upon the reviews an adequate number of SWEL items were included for inspection. 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 98 components in SWEL-1 and an additional 6 items in SWEL-2 totaling 104 total items in the combined SWEL. The number of items on SWEL-1 is within the recommended range of 90-120 items. The SWEL was taken from the IPEEE SSEL. The number of items inspected at the site is within the EPRI guidance.

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 Checklists

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 to reflect conditions. Intermediate guidance resulting from the reviews during the inspection process was provided.

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 consistent with expectations and are 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 PASCs were reviewed for the unit and the classification was determined appropriate.

The Peer Review concluded 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 BNP'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 issues were encountered at the BNP-U1 site and the site had already documented the condition within the CAP. The condition of the site and the aggressive identification process resulted in a general impression of the SWEs that 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 BNP-U1 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 with seismic demand for the region.

Brunswick Steam Electric Plant, Unit 2
Seismic Walkdown Report Review by Site Management

The report titled *Brunswick Steam Electric Plant Unit 2 Seismic Walkdown Report* (i.e., Enclosure 4) is provided to the Nuclear Regulatory Commission in response to its request for information. Specifically, by letter dated March 12, 2012, the NRC 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 Brunswick Steam Electric Plant, Unit 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 therein 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 signatures below document site management review of this document:

Signatures	Date
Site Fukushima Program Manager Lenny Beller / L Bell	11/27/12
Site Seismic Engineer Dan Zebroski / DJ JW	11/27/12
Site Design Engineering Manager Joe Price / Hal Sanders for J. Price	11/27/2012

Brunswick Steam Electric Plant, Unit 2
Seismic Walkdown Report

Attachments 6 and 7 Contain Security-Related Information
Withhold in Accordance with 10 CFR 2.390
Upon removal of Attachments 6 and 7, this document is decontrolled.

Brunswick Steam Electric Plant Unit 2 Seismic Walkdown Report

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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 (NTTF) 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 Brunswick Steam Electric Plant (BSEP) Unit 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 NRC, in its letter dated May 31, 2012, endorsed the EPRI guidance document.

The 2.3 Seismic Walkdown inspections performed were non-intrusive visual inspections of primarily plant Seismic Class I structures, systems and components (SSCs). During the inspections, observed degraded, nonconforming, or unanalyzed conditions were identified and addressed through the corrective action program (CAP). Based on the EPRI guidance document, the list of SSCs for inspection were obtained through a systematic selection process to establish a broad, diverse and representative Seismic Walkdown Equipment List (SWEL). The SWEL was made up of two separate lists: SWEL 1 included 113 SSCs from various locations throughout the plant and SWEL 2 included a shorter specific list of four Spent Fuel Pool (SFP) SSCs.

The selection process for the SSCs combined with the inspection checklist attributes assessed design basis 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 (URI) A-46 "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors." Many of the same SSCs inspected for the IPEEE were re-inspected for the current 2.3 Seismic Walkdowns. Most of the SWEL items originated from the IPEEE Safe Shutdown Equipment List (SSEL). These programs occurred in the 1990s. The A-46 program reviewed equipment in the older nuclear plants with start-up dates prior to 1984 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 Assessment (SMA) programs to assess the plants capabilities to perform properly to a larger Review Level Earthquake (RLE). Modifications were also performed as a result, if necessary. For BSEP Unit 2, no IPEEE modifications were required.

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 found in the BSEP Updated Final Safety Analysis Report (UFSAR) and other design documents provides the description of those Systems, Structures, and Components (SSCs) that perform an important-to-safety function both during and after a Design Basis Earthquake (DBE). The following paragraphs provide the development of the site characteristics, earthquake characteristics, the seismic design requirements for SSCs and the various codes and standards used for seismic designs at BSEP Unit 2.

Certain plant structures must remain functional and/or protect vital equipment and systems, both during and following the most severe natural phenomenon postulated to occur at the site. In order to establish the loadings and loading combinations for which each individual structure was designed, buildings and their structural systems were separated into the following classes with respect to seismic design requirements

- Seismic Class I – structures and equipment are categorized as Class I if they are essential for safe shutdown or if failure could result in the release of radiation with dose consequences potentially exceeding the guidelines of 10CFR100.
- Seismic Class II – structures and equipment are categorized as Class II if their failure could not result in the release of radiation with dose consequences in excess of guidelines of 10 CFR100.

Basic geologic and seismic information are presented in Section 2.5.1 of the BSEP UFSAR. The BSEP site is located approximately 2 ½ miles north of Southport and 1 ½ miles west of the Cape Fear River in southeastern North Carolina. Physiographically, the site is located on the Atlantic Coastal Plain about 90 miles southeast of the boundary between the flat lying deposits of the Coastal Plain and the folded formations of the Piedmont and Appalachian regions. This boundary is known as the Fall Line.

In the vicinity of the site, the Coastal Plain consists of approximately 1,500 feet of Cretaceous and younger deposits. In general, hard limestone exists from a depth of approximately 70 feet below existing ground surface and extends to a depth of 230 feet or more. The crystalline or metamorphic basement rock has been broadly warped into a tectonic feature known as the Cape Fear Arch.

From the standpoint of relief or physiography and structural geology, there are good reasons for the relative infrequency of earthquakes in the South Atlantic states. Earthquakes are most common in those areas characterized by narrow coastal plains, with recent mountains rising abruptly from near the coast and having narrow continental shelves extending seaward from the shore. North Carolina and adjoining states along the Atlantic Seaboard have a coastal plain 100 or more miles wide while mountains of ancient geologic origin occur another 100 miles inland. The continental shelf slopes gradually beneath the sea for another 50 to 100 miles before reaching its steeper margin. The world over, this combination of physiographic conditions is indicative of relative seismic stability.

The ultimate heat sink for BSEP Unit 2 is the Cape Fear River with the intake canal being the means by which the water is supplied to the intake structure. Cooling water flows to and from the plant through the canals discussed in Section 2.4.8 of the UFSAR. The circulating water system consists of an open intake canal, the circulating water and service water intake structures, the turbine condensers and piping, and a discharge canal terminating in the Atlantic Ocean. The intake canal is capable of supplying sufficient cooling water for all normal conditions or accident conditions such as a DBE. The only pumps which are required for support of safety-related equipment are the service water header pumps (i.e., intake structure) and the residual heat removal (RHR) service water pumps (i.e., reactor building). The service water intake structure is designed to functionally survive the DBE.

Site structures are founded on Class I backfill. The natural grade around the plant is approximately 20 feet above mean sea level. The area of the plant was excavated to an approximate elevation of -25 feet to dense sand and backfilled to grade to approximately +19 feet above mean sea level. The reactor building is founded on the dense sand and the other buildings are founded on the structural backfill that is founded on the dense sand. Beneath the dense sand is limestone bedrock.

The closest location of large earthquakes is around Charleston, SC located approximately 120 to 150 miles from the BSEP site. Based on possible earthquake effects from the Charleston area and from local faults within 20 to 25 mile radius, a ground acceleration of 0.08g was selected for the Operating Basis Earthquake (OBE). The DBE was selected to be twice the OBE horizontal ground acceleration of 0.16g, resulting in a high intensity VII (Modified Mercalli) seismic event at the site.

Seismic design requirements are based on an OBE with a horizontal ground acceleration of 0.08g, and DBE with horizontal ground acceleration of 0.16g. The vertical ground accelerations associated with the OBE and DBE were 2/3 and 4/3 of the corresponding OBE horizontal response spectra, respectively. The response spectrum is the smoothed 1940 North-South El Centro spectrum normalized by a factor of 0.08g/0.33g or 0.24 with a corresponding ground velocity of 4 in/sec for OBE. For DBE, the peak ground velocity is 5 in/sec with a peak ground acceleration of 0.16g.

Class I 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 the American Society of Mechanical Engineers (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; American Concrete Institute (ACI) 318-63 Building Code Requirements for Reinforced Concrete; and 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. This was selected to test equipment to the level that it would see in the maximum number of sites. The reduced load factors permitted by ACI 318 (Part IV-B) and the increase in allowable stresses permitted by the AISC Specification for loading combination which include earthquake loads were not used.
- The stresses that resulted from normal loads and design basis loss-of-coolant accident loads combined with the response to the DBE were limited so that no loss of function

occurred and the capability of making a safe and orderly plant shutdown was maintained.

Seismic Class I Instrumentation and Electrical Equipment must perform their safety function before, during and after a seismic event. The original design basis for this equipment required a blanket testing of horizontal 1.5g, vertical 0.5g and a frequency range of 5 to 33 Hz. The standard for seismic acceptance of equipment is Institute of Electrical and Electronics Engineers (IEEE) 344-1971. The standard divides the equipment into four main groups.

- Group A includes instruments and instrumentation and control devices including motor operators for valves.
- Group B includes enclosures, panels, and racks
- Group C includes primary pressure boundary devices
- Group D includes large electric motors

Tests were performed for Groups A and B above and analyses were performed for Groups C and D. Group C components are limited to SSCs which have a primary function of preservation of pressure boundary and are governed by the ASME Code requirements and were not tested. Group D are large electric motors and the analytical method of seismic qualification in accordance with IEEE 344-1971 was used.

Class II structures were generally designed in accordance with procedures of the Uniform Building Code for Zone 1. Class II equipment was designed for a static coefficient of 0.08g. The combined stresses from normal and earthquake loadings were limited to those permitted by applicable standards.

Seismic qualification of motor-operated valve operators has included accelerations in excess of 5g at the point of attachment to the valves and covering the frequency range of 5 to 33 Hz. This exceeds the anticipated acceleration values at the location of any Seismic Class I Limitorque operator and thus meets the requirements of IEEE 344-1971.

For Class I equipment furnished by suppliers other than General Electric (GE) Atomic Power Equipment Department, the seismic criteria considers a combination of normal and seismic loads with appropriate design margin to ensure that, during or immediately following an OBE, interruption or spurious operation shall not occur of controls in the normal or vital mode of operation. Likewise, that immediately following a DBE, interruption of operation of controls shall not occur. Complex equipment was subjected to vibratory motion which conservatively simulates a DBE. The complete system was energized during the test, accelerometers were mounted at various planes and all electrical relays and devices were monitored to detect their proper operation during the test. Analysis of non-complex equipment included determination of inertia and elastic characteristics, natural frequencies and nodal shapes. Either response spectrum technique or time-history approach was used to support the calculation of resulting stresses that determined equipment responses. Together, the testing and analysis ensure the proper performance of Class I equipment with regards to the established seismic criteria.

Revision 3 of the Seismic Qualification Utility Group (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 was used as an alternative to existing methods for the seismic design and verification of modified, new and replacement equipment. This alternative was not used for NRC Regulatory Guide 1.97 equipment unless justified on a case specific basis.

The program for resolution of USI A-46 and response to IPEEE seismic at BSEP was implemented with a single SSEL to address the requirement and guidance of both programs.

3.0 Personnel Qualifications

3.1 Equipment Selection Personnel

All resumes for personnel are included here except plant operations personnel that performed a supplemental role during the SWEL selection process.

3.1.1 Doyle Adams

Doyle G. Adams has over 36 years of engineering experience in both design and construction. This includes over 22 years nuclear experience at an operating nuclear facility. Nuclear experience includes design engineer, design engineering supervisor, SQUG/IPEEE qualifications and walkdown engineer, seismic equipment testing and qualification. He was also the lead responsible civil design engineer for Reactor Building design pressure uprate, Steam Generator and Reactor Head Replacement projects. Doyle Adams has a BS Architectural Engineering degree in structures.

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 structures using ASME Section III, ASME/ANSI B31.1 and B31.3, and applicable nuclear plant procedures. Mr. Bamberger holds an A.D. in Mechanical Engineering Technology and has taken additional classes in Mechanical Engineering and Technology.

3.1.3 Leonard Beller (assisting plant operations personnel)

Leonard Beller has almost 30 years of nuclear experience at BSEP. Mr. Beller has Operations experience as a non-licensed operator and progressed to Control Room Supervisor/Senior Reactor Operator. He was Manager of Licensing and Regulatory Programs for seven years, and Operations Training Manager for five years. He is currently the Brunswick Fukushima Response Organization site lead. He has participated in Institute of Nuclear Power Operations (INPO) Next Level Leadership for Managers, Peer Evaluator for three Operations Training Programs Accreditation Team Visits, and Host Peer Evaluator for Organizational Effectiveness INPO evaluation. Mr. Beller holds a B.A. degree in Geology.

3.1.4 John McIntyre (assisting plant operations personnel)

John McIntyre has over 39 years of experience in the design, construction and maintenance of nuclear power plants. He is currently assigned to the Brunswick Fukushima Response Organization as engineering lead. Work experience includes engineering, supervisory and management positions in Plant Equipment Performance, Mechanical and Structural Seismic Design, Design Control, Nuclear Oversight and Nuclear Plant Construction. Mr. McIntyre holds a B.S. degree in Mechanical Engineering and Associate Aeronautical & Space Engineering Technology.

3.2 Seismic Walkdown Engineers

3.2.1 Doyle Adams

See 3.1.1 above.

3.2.2 Guyuruddin Ahmed

Gayuruddin Ahmed has over 30 years of experience in Civil Engineering. He has experience in design of nuclear power plant structures and components with extensive use of finite element methods of frame analysis. He also has experience in ACI, AWS, and AISC code requirements and related design procedures of power plants for seismic conditions.

3.2.3 Martin Foster

Martin P. Foster has over 35 years of engineering experience most of which with various US nuclear plants. His experience includes seismic equipment qualification and testing for such equipment as control panels, instrument and battery racks, chargers and direct replacement of valves and instruments, seismic design of structures and equipment, seismic piping stress analysis and support design. Martin Foster has a B.S. degree in Civil Engineering.

3.2.4 Nazir Sheikh

Nazir Sheikh is a Registered Professional Engineer and has over 35 years engineering experience with over 30 years nuclear experience. Mr. Sheikh has been associated with nuclear design in nuclear piping design, concrete and steel structures using ACI 318, ACI 349, AISC, mechanical electrical equipment qualification and testing in accordance with IEEE-344. Nazir Sheikh has a B.S. degree in Civil Engineering and studies toward a M.S. degree in Mechanical Engineering.

3.2.5 James Curry

James Curry has over 12 years engineering experience with over six years experience at BSEP. He has nuclear experience as Systems Engineer for heating, ventilation, and air conditioning (HVAC) systems, seismic equipment qualifications, steel and concrete design and is SQUG trained and experienced. He also has four years Nuclear Navy experience in reactor operation and maintenance. Mr. Curry holds a B.S. degree in Civil Engineering.

3.2.6 Daniel Zebroski

Dan Zebroski is a Registered Professional Engineer and has over 30 years engineering experience associated with the design and construction of nuclear power plants. This includes the seismic qualification of equipment. He was responsible for the implementation and resolution of the SQUG verification of the seismic adequacy of all the safety related electrical and mechanical equipment at Ginna Station. At BSEP, he is the primary contact for all seismic issues, including seismic qualification training for other engineering personnel, seismic equipment qualification and dynamic analysis. He holds a B.S. degree in Civil Engineering from Drexel University. He has completed the SQUG walkdown screening and Seismic Evaluation and IPEEE Seismic Add-on courses and is currently registered as a professional engineer in the state of Connecticut.

3.2.7 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 at both AREVA and URS. He was involved in several projects including: Arkansas Nuclear One (ANO) 1 and 2 Control Room expansion, Equipment obsolescence, Dry Fuel Storage, and ANO-2 Containment redesign/design pressure up-rate. As a consultant, he served as the Civil/Structural Engineering Design Lead (EDL) for the U.S. European Pressurized Reactor (EPR) Design Certification (DC) and Combined Operating License (COL) projects providing a technical review of civil based licensing responses to clients or the NRC and project management. More recently, he has served as the Civil-Structural-Architect Discipline Manager for the detailed design phase of the U.S. Advanced Pressurized Water Reactor (US-APWR) including all aspects of the design including the site specific and design control document (DCD) seismic evaluations. Billy Alumbaugh has a M.S. and a B.S. degree in Civil Engineering.

3.3 Licensing Basis Reviewers

3.3.1 Doyle Adams

See 3.1.1 above.

3.4 IPEEE Reviewers

3.4.1 Doyle Adams

See 3.1.1 above.

3.5 Peer Reviewers

3.5.1 Frederick Ogden

Frederic Ogden is a Registered Professional Engineer and has over 39 years experience in the power industry, the majority in the analysis and design of new nuclear generating stations or modification to existing stations. His experience also includes Communication, Industrial and Transmission and Distribution projects. He is responsible for the technical and administrative direction, supervision and oversight of the department personnel in the Princeton, New Jersey office, the Fermi Site Office in Monroe, MI, and the Warrenville, Illinois office. Mr. Ogden has a B.S. degree in Civil Engineering, Drexel University and a M.S. degree in Civil and Urban Engineering, University of Pennsylvania.

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, with over 15 years in management positions associated with construction, maintenance, modifications, including work package control, and operation of Department of Energy and NRC regulated facilities such as Nuclear Power Plants, Vitrification Facilities, Radioactive Waste Facilities, Gaseous Diffusion Facilities, and TRU Waste Characterization and Disposal. Mr. Wade is an American Society for Quality (ASQ) Certified Quality Auditor (CQA) 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 BSEP Unit 2 was based on the EPRI guidance document, Section 3. This selection process was conducted by Equipment Selection Personnel selecting SSCs based on selection criteria. During the process, plant operation staff assisted the Equipment Selection Personnel. The process, as described in the EPRI guidance document, involves the use of 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 one or more of the following five safety functions
    - Reactor reactivity control
    - Reactor coolant pressure control
    - Reactor coolant inventory control
    - Decay heat removal
    - Containment function
- Screen #4: Sample considerations (i.e., systems, major new/replacement, equipment types, environments, IPEEE enhancements)

The list of equipment resulting from Screen #3 is Base List 1. At BSEP, 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 IPEEE program. As discussed in EPRI Report 1025286, the first screen is intended to narrow 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. 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 IPEEE SSEL met the criteria for Screens #1, #2, and #3, and thus using the IPEEE SSEL was an appropriate starting point. Other SSCs were added to the IPEEE SSEL that were modified plant equipment verified to meet the criteria for Screens #1, #2, and #3. The IPEEE SSEL with these additional items served as Base List 1 (i.e., refer to Attachment 1).

Since BSEP is comprised of two nuclear units, there are shared or common equipment between the two units. Though the common equipment is capable of functioning for either unit, the equipment identifier is designated as Unit 1 or Unit 2. For the purposes of this inspection, those SSCs with a Unit 2 designation were included in the Unit 2 SWEL and those with a Unit 1 designator were included in Unit 1 SWEL. There is more common equipment associated with Unit 2 because Unit 2 was built prior to Unit 1. There are some instances, such as the emergency diesel generators, where the equipment might be aligned to serve Unit 1 as the primary function, but have a Unit 2 designator. This equipment was included on the Unit 2 SWEL.

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 (i.e., refer to Attachment 2). Considerations made for the creation of SWEL 1 are detailed in the sections below.

4.1.1 Equipment types/classes

A breakdown of the number of inspected items into the various equipment classes is provided in the following table.

Class No.	Equipment Included	Base List 1	Selected (SWEL 1)
0	Other	82	9
1	Motor Control Centers and Wall-Mounted Contactors	25	3
2	Low Voltage Switchgear and Breaker Panels	4	2
3	Medium Voltage Metal-Clad Switchgear	6	2
4	Transformers	7	1
5	Horizontal Pumps	15	3
6	Vertical Pumps	4	3
7	Pneumatic-Operated Valves	49	13
8	Motor-Operated and Solenoid Operated Valves	102	19
9	Fans	10	0*
10	Air Handlers	14	1
11	Chillers	4	0*
12	Air Compressors	1	1
13	Motor Generators	2	2
14	Distribution Panels and Automatic Transfer Switches	31	6
15	Battery Racks	4	2
16	Battery Chargers and Inverters	9	4
17	Engine Generators	4	1
18	Instrument Racks	78	15
19	Temperature Sensors	57	4
20	Instrument and Control Panels	67	17
21	Tanks and Heat Exchangers	39	5
	Total	614	113

* Items from this class were walked down during the Unit 1 walkdown effort and are common between the two units.

An objective was to obtain equipment in every class; however, due to equipment inaccessibility or located in high radiation areas, some of the common equipment designated Unit 1 was used to meet the objective since it could serve the function for Unit 2 as well. For SWEL 1, two equipment classes, (i.e., 9 and 11) met this need as further explained below. The walkdown information is included in the BSEP Unit 1 report.

- Equipment Class 9 (i.e., Fans): The one fan that was inspected for Unit 2 was a common fan to Unit 1 and Unit 2, but with a Unit 1 component designator.
- Equipment Class 11 (i.e., Chillers): The one chiller that was inspected for Unit 2 was a common chiller to Unit 1 and Unit 2, but with a Unit 1 component designator.

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	SSEL Total	Selected (SWEL 1)
Reactor reactivity control	53	27
Reactor coolant pressure control	92	25
Reactor coolant inventory control	97	29
Decay heat removal	404	70
Containment function	30	8

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 Building, Control Building, Diesel Generator Building, Service Water Building (Intake Structure), Condensate Storage Tank Yard, and the Fuel Oil Tank Chamber Building. The SWEL 1 in Attachment 2 includes the building and location of each SSC.

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. The SWEL 1 in Attachment 2 includes the environment of each SSC.

4.1.5 Systems

During the SWEL 1 selection process, consideration was given to equipment of varying systems including the Control Rod Drive System, High Pressure Coolant Injection, Residual Heat Removal, and Pneumatic Nitrogen Systems among others. Table B-2 of Appendix E "Safety Function-System Matrix for BWRs" of the EPRI guidance was consulted to ensure systems to support safety functions were included. Additionally, equipment in the Service Water System Building and equipment associated with the Service Water System that comprise emergency 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 contribution of individual items to overall risk was considered in the selection of the SSCs from the SSEL list for items to include in the SWEL 1. 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 probabilistic risk assessment 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 flood systems, emergency cooling water systems, and 1E electrical switchgear are identified as critical equipment that have a higher risk profile. Some of this equipment was included while maintaining a balance with the other requirements of SWEL equipment selection.

4.1.7 IPEEE vulnerabilities

No seismic vulnerabilities were identified for the IPEEE seismic program. Therefore, no items were added to the SWEL 1 for IPEEE purposes.

4.1.8 Modified, replacement, and new equipment

A review of the plant modifications from 1995 (i.e., the initiation time of IPEEE) to 2002 and Engineering Changes dating 2002 to 2011 was performed to determine significant modifications to the plant. For BSEP Unit 2, 19 significant modifications were identified; four pertained to installing new equipment and 15 were equipment replacements. Since one item was previously identified on the list, 18 modifications were added to Base List 1 and identified as MOD 1 through MOD 18. Plant support personnel (i.e., systems, operations and engineering, etc.) identified the modifications to be included in the walkdowns.

4.1.9 Accessibility

Before and during 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 BSEP Base List 2, Rapid Drain-Down List, and SWEL 2 based on the EPRI guidance document which presents screening criteria to identify specific equipment that is unique to the SFP SSCs. As described in EPRI Report 1025286, Screen #1 and #2 limit SFP SSCs to those which have a Seismic Category I licensing basis and are capable of being visually reviewed in the plant. The equipment that resulted by applying these screens consisted of check and isolation valves on each of the two diffuser return lines in the Fuel Storage Pool Cooling and Filtering System. In addition, two spool pieces shared between the units and part of the Supplemental Spent Fuel Pool Cooling (SSFPC), were added to the Unit 2 list making a total six items in Base List 2 and is included in 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 ten feet above the spent fuel stored in the pool. There were six items identified on the Rapid Drain-Down list that were evaluated, included as Attachment 4. The six SSCs that were found to have a possible rapid drain-down capability include the skimmer surge tanks and skimmer piping, fuel pool cooling and filtering piping, SFP leak chase drains, fuel transfer gates, the Supplemental Spent Fuel Pool Cooling System, and the refueling canal when in a refueling condition. It was determined that these SSCs do not contain drain down paths that would meet the SFP Rapid Drain-Down definition. The spool pieces which are considered part of the SSFPC are shared between the units, they were only inspected one time and this occurred during the BSEP Unit 1 inspections. Therefore, the SWEL 2 for Unit 2 did not include these and contains four items, 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 Seismic Walkdown Engineers (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 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, including observations and PASCs, are 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 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.

Seismic walkdowns were performed on each SWEL 1 and 2 item that was accessible at the time of the walkdown effort for BSEP Unit 2. When SWEL items were inaccessible and an appropriate substitute was not available, the item was documented to be inspected at a future date as detailed in Section 5.6.

The seismic walkdowns focused on identifying PASCs for the SSCs listed on the SWEL using the following criteria for adverse anchorage conditions, adverse seismic spatial interactions, or other adverse seismic conditions:

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 10 diameters 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:

- A 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

For BSEP Unit 2, this alternate method was not implemented for any of the items.

The SWEs used engineering judgment to assess whether the anchorage is potentially vulnerable to seismic failure or malfunction (i.e., PASC). 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 component 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 which could occur that would 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 (i.e., either within the room or, for large rooms, within approximately 35 feet 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 and 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

The occurrence of a seismic event could create fire in multiple locations, simultaneously degrade fire suppression capabilities, and as a result prevent mitigation of fire damage to multiple safety-related functions.

During the seismic walkdowns, the engineers visually assessed any potential sources of fire (e.g., compressed flammable gas bottles, fuel tanks, other combustible material, etc.) located in the vicinity of the SWEL item to ensure it was adequately restrained. Additionally, potential interactions were assessed to determine if relative motion of high voltage equipment and adjacent support structures that have different foundations can cause high voltage busbars to short out against the grounded bus duct and cause a fire.

5.2.2 Seismically Induced Flood/ Spray Interactions

Seismically induced flooding events can potentially cause multiple failures of safety-related systems. Two examples of potential flooding sources are rupture of piping and vessels. Instances of concern include 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 located in the vicinity of the subject SSC to ensure they had adequate support and, therefore, were not likely to be a source of flooding or spray that could adversely affect the subject item. The items that were identified as potential conditions were documented. Any assessment and disposition of the effects were documented with the subject item. During the walkdowns and walk-bys, spray nozzle clearance with nearby lighting was inspected. It was determined that adequate clearances existed.

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 that criterion, the item was entered into the Corrective Action Program for resolution. Of the 113 SWEL items inspections and 40 area walk-bys, three PASCs were identified. For all PASCs identified, a licensing basis review was completed as stated in Section 6.0 below.

The following table summarizes the condition and status of each item judged as a potentially adverse seismic condition. The SW Building concrete floor condition described in the table below was identified prior to the seismic walkdowns and is currently being repaired. The remaining conditions were found to be in compliance with their seismic licensing basis.

Feature	Condition	Status of Resolution
Service Water (SW) pumps, strainers, and instrumentation	The SW Building concrete floor at the 20 foot elevation is spalled and cracked exposing rebar in many areas. This structural concrete slab, serving as the foundation for much equipment, is a PASC. This condition is documented in Nuclear Condition Report 150706 and the operability of the floor to perform its design function to support floor loading is evaluated as being acceptable in Operability Concern Review (OCR) Task 40, Sub-assignments 1, 2, 3, and 4.	A previously issued Nuclear Condition Report identifies corrective actions to repair the concrete and repairs are currently in progress.
Condensate Storage Tank	Insulation package not secure and large gaps.	Work Request initiated.
	Frame support and nearby equipment had excessive surface rust.	Nuclear Condition Report initiated. Work Request initiated.

Feature	Condition	Status of Resolution
Open S-Hooks	Open item for Unit 2 based on extent-of-condition review associated with Unit 1 observation.	This condition is similar to the outlier condition that was resolved under the A-46 program. During A-46 program, Work Requests were initiated and S-Hooks were closed. This specific instance was not covered in the A-46 Program report. An extent of condition investigation was initiated and other open S-Hooks were identified. All cases were determined to be operable but were also closed to eliminate the gaps. Guidance documents for maintenance of fixtures were revised to ensure S-Hooks were closed during future maintenance activities. The extent-of-condition investigation is still underway.

Note: S-Hooks were determined to be PASCs, due to further engineering reviews after completion of the walkdowns.

5.4 Maintenance Assessment

The maintenance assessment, as requested in the 50.54(f) letter, 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 and minor housekeeping problems were noted. Almost all mobile equipment, tables, and tools were either secured properly or located in safe locations away from plant equipment. A few issues were noted with transient items such as cleaning equipment. Contamination was minimal. These indicators suggest that monitoring and maintenance processes and procedures are adequate. No adverse trends were identified in observations.

5.5 Planned or Newly Installed Changes

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

5.6 Inaccessible Items

There are 23 full or partial inspections that will need to be completed. Four of these are equipment that are located inside the primary containment (i.e., drywell) and were inaccessible at power operations. These four will be walked down during a refueling outage. The other 16 items were inaccessible panels and cabinets (i.e., some with anchorage on the interior). These include electrical equipment with doors or panels that were either locked because they either represented a personal safety hazard or contained a potential risk to affect the plant while at power. In these instances, the electrical equipment was walked down and the area walk-bys were performed without opening the panels or doors to inspect the inside. A breakdown of the equipment for future seismic walkdown is provided in the following table.

Item No.	Feature (Equipment ID)	Inspection Date
1	250VDC Motor Control Center (MCC) (2-2XDB)	March 2015
2	250VDC MCC (2-2XDA)	March 2017
3	Control Rod Drive Accumulator Monitor Panel Bank 1 & 2 (2-H21-P003)	March 2013
4	Partial Winding Heater Cabinet for MCC 2PB (2-SW-PNL-VW7)	December 2013
5	Diesel Generator 1 Excitation Panel (2-DG1-EXCIT-PNL)	December 2013
6	480V Unit Substation E7 (2-E7)	March 2017
7	Diesel Generator 1 Control Panel (2-DG1-GEN-CTRL-PNL)	December 2013
8	125/250VDC Switchboard 2A (2-2A-250VDC)	March 2017
9	125/250VDC Switchboard 2B (2-2B-250VDC)	March 2015
10	Reactor Protection System Power Distribution Panel (2-C72-P001)	March 2013
11	125VDC Battery 2A-2 Charger (2-2A-2-125VDC-CHRGR)	March 2013
12	125VDC Battery 2B-1 Charger (2-2B-2-125VDC-CHRGR)	March 2013
13	Control Building Battery Room 2A Sup Fan 2C Circuit Bkr (2-2CA-C21-52)	August 2016
14	RHR Service Water Pump 2A Motor High Temperature Relay (2-SW-TY-4887)	December 2013
15	Diesel Generator Building 125VDC Distribution Panel (2-2A-125VDC)	March 2013
16	4160V Switchgear E4 Air Intake Fire Damper (2-VA-DG-FDMP-52)	See Note 2
17	Condensate Storage Tank Low Water Level (2-E41-LSL-N002)	See Note 2
18	Control Building U-1 & U-2 Control Room HVAC Pneumatic Control Panel (2-VA-M1-CB)	March 2013
19	Engineered Safeguards Vertical Board (2-H12-P601)	See Note 2
20	Main Control Room RTG Board (2-XU-2)	March 2013
21	Fluid Flow Detection Cabinet for Safety Relief Valve Position Indication (2-XU-73)	March 2013

Item No.	Feature (Equipment ID)	Inspection Date
22	Reactor Control Panel (2-H12-P603)	See Note 2
23	PRI Steam Line 'B' Safety/Relief Valve (ADS) PM 80-085 (2-B21-F013C)	March 2013
24	Safety Relief Valve G & Solenoid (2-B21-F013G)	March 2013
25	Drywell Air Temperature Element (2-CAC-TE-1258-6)	March 2013
26	Drywell Air Temperature Element (2-CAC-TE-1258-5)	March 2013
27	Distribution Panel 480-208/120V 30KVA Transformer (2-2A-SW-XFMR)	March 2017

Table Notes:

1. Items 4, 5, 7, 10, 11, 12, 13, 14, 15, and 19 were added as a result of Frequently Asked Questions.
2. These items were originally scheduled to be credited for anchorage inspection. However, upon inspection, some of the anchorage was not visible due to interferences. The anchorage that was visible did correlate with the as-built anchorage drawings from the USI A-46 and IPEEE walkdowns that were used during the Fukushima seismic walkdowns. Since 100% of the anchorage could not be inspected for these items they were not credited as part of the minimum 50% anchorage list. These items will be inspected at a future date, if the anchorage is accessible.

The expected inspection dates are based on unit outages occurring in March of odd years. Following completion of these inspections, the seismic walkdown report will be updated and submitted to the NRC by September 30, 2017.

6.0 Licensing Basis Evaluations

With the exception of the Service Water Building floor that is currently under repair to restore full qualification, potentially adverse seismic conditions that were identified during the seismic walkdowns and area walk-bys were found to meet the plant seismic licensing basis.

7.0 IPEEE Vulnerabilities Resolution Report

No seismic vulnerabilities were identified in the Brunswick IPEEE. Outlier conditions for USI A-46 have been resolved as addressed in a Carolina Power & Light Company letter to NRC on September 11, 1998.

8.0 Peer Review

The Peer Review is included in Attachment 8.

Attachments

Attachment 1: Base List 1

Attachment 2: SWEL 1

Attachment 3: Base List 2

Attachment 4: Rapid Drain-Down List

Attachment 5: SWEL 2

Attachment 6: Seismic Walkdown Checklists **(Attachment Contains Security-Related Information)**

Attachment 7: Area Walk-By Checklists **(Attachment Contains Security-Related Information)**

Attachment 8: Peer Review Report

Attachment 1: Base List 1

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
TURBINE				X	
RHR DAMPER OPERATOR				X	
RHR DAMPER OPERATOR				X	
SCRAM ACCUMULATOR	X				
NITROGEN BOTTLE & REGULATOR	X				
RHR HX 2A RELIEF VALVE				X	
N2 B0TTLES	X				
N2 BOTTLES	X				
BACKUP N2 DISCHARGE RUPTURE DIAPHRAGM	X	X	X		
BACKUP N2 DISCHARGE RUPTURE DIAPHRAGM	X	X	X		
STRAINER 2A DP SWITCH			X		
STRAINER 2B DP SWITCH			X		
BACKUP N2 INLINE FILTER	X	X	X		
BACKUP N2 INLINE FILTER	X	X	X		
SP STRAINER			X		
CS STRAINER SUCTION LINE			X		
HPCI/SP STRAINER			X		
NSW 2A SELF CLEANING STRAINER, MO				X	
MCC-2XDA			X	X	
MCC-2XDB			X	X	
BACKUP N2 DISCHARGE PRESS CONTROL VALVE	X				
BACKUP N2 DISCHARGE PRESS CONTROL VALVE	X				
BACKUP PRESSURE RELIEF VALVE	X			X	
SWITCHBOARD 2A			X	X	
SWITCHBOARD 2B			X	X	
TRANSFORMER				X	
TRANSFORMER				X	
TRANSFORMER				X	
TRANSFORMER				X	
RHRP-2A				X	
RHRP-2C				X	
HPCI BOOSTER PUMP			X		
RHRSW BOOSTER PUMP 2A				X	
RHRSW BOOSTER PUMP 2B				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
RHRSW BOOSTER PUMP 2C				X	
RHRSW BOOSTER PUMP 2D				X	
CSP-2A			X		
HPCI MAIN PUMP			X		
NSW PUMP 2A				X	
NSW PUMP 2B				X	
RHR ROOM COOLER 2B ISO				X	
HPCI LO CLR PRESS CONTROL VALVE			X		
CS PUMP ROOM 20 ISO				X	
CS ROOM COOLER NSW OUT ISO				X	
RHR PUMP 2B SEAL COOLER OUTLET				X	
RHRP 2A SEAL COOLING DISCHARGE				X	
RHRP 2C SEAL COOLING DISCHARGE				X	
RHR PUMP 20 SEAL COOLER OUTLET				X	
RHRSW PUMP HX AOV VALVE & SOLENOID				X	
RHRSW PUMP HX AOV VALVE & SOLENOID				X	
RHRSW PUMP HX AOV VALVE & SOLENOID				X	
RHRSW PUMP HX AOV VALVE & SOLENOID				X	
HPCI COND PUMP DRAIN TO CRW ISOL VALVE			X		
HPCI COND PUMP DRAIN TO CRW ISOL VALVE			X		
LO. COOLER LINE RELIEF VALVE				X	
TURBINE STOP VALVE		X			
TURBINE CONTROL VALVE		X			
RHR PUMP 2A ROOM COOLER OUT ISO				X	
RHR HX/TORUS CONTROL VALVE				X	
DISCHARGE DAMPER FOR RHR COOLING UNIT A				X	
DISCHARGE DAMPER FOR RHR COOLING UNIT B				X	
SCRAM DISCHARGE VOLUME DRAIN VALVE	X				
SCRAM DISCHARGE VOLUME DRAIN VALVE	X				
SCRAM INLET ISOLATION VALVE	X				
SCRAM OUTLET ISOLATION VALVE	X				
SCRAM DISCHARGE VOLUME VENT VALVE	X				
SCRAM DISCHARGE VOLUME VENT VALVE	X				
BACKUP N2 PRESSURE RELIEF VALVE	X	X			
BACKUP N2 PRESSURE RELIEF VALVE	X	X			
BACKUP N2 PRESSURE RELIEF VALVE	X				

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
SAFETY RELIEF VALVE A & SOLENOID		X			
SAFETY RELIEF VALVE B & SOLENOID		X			
SAFETY RELIEF VALVE C & SOLENOID		X			
SAFETY RELIEF VALVE D & SOLENOID		X			
SAFETY RELIEF VALVE E & SOLENOID		X			
SAFETY RELIEF VALVE F & SOLENOID		X			
SAFETY RELIEF VALVE G & SOLENOID		X			
SAFETY RELIEF VALVE H & SOLENOID		X			
SAFETY RELIEF VALVE J & SOLENOID		X			
SAFETY RELIEF VALVE K & SOLENOID		X			
SAFETY RELIEF VALVE L & SOLENOID		X			
DG SUPPLY ISOL, NSW				X	
CONTAINMENT ATMOSPHERE SOLENOID VALVE					X
SCRAM VALVE	X				
SCRAM VALVE	X				
SCRAM SOLENOID VALVE	X				
SCRAM SOLENOID VALVE	X				
RHRSW HEAT EXCHANGER DISCH ISOLATION				X	
NSWP 2A DISCH ISOL				X	
NSWP 2B DISCH ISOL				X	
SP SUCTION VALVE			X		
CSP-2A SP SUCTION VALVE			X		
ROOM COOLING NSW XTIE				X	
RHR HX 2A OUTLET VALVE				X	
MIN FLOW BYPASS VALVE				X	
RHR HX 2A INLET VALVE				X	
RHR HX 2A BYPASS VALVE				X	
CS FULL FLOW TEST BYPASS VALVE			X		
HPCI TEST LINE/CST RETURN VALVE			X		
ROOM COOLING NSW ISO				X	
CS MIN FLOW BYPASS VALVE- TR A			X		
RHRP-2A SP SUCTION VALVE				X	
HPCI DISCHARGE VALVE			X		
HPCI TEST LINE/CST RETURN VALVE			X		
MIN FLOW BYPASS VALVE			X		
HPCI LO COOLING WATER VALVE			X		

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
TURBINE VACUUM BREAKER VALVE			X		
SP COOLING ISOLATION VALVE			X		
SP SPRAY ISOLATION VALVE			X		
SP DISCHARGE ISOLATION VALVE			X		
DRYWELL SPRAY OUTBOARD ISOLATION VALVE			X		
ROOM COOLING NSW ISO				X	
RHR HT EXCH 2A OUT ISO				X	
RHR HX 2B NSW OUT FLOW CTL				X	
SHUTOWN CLG OUTBOARD SUCTION ISOL VALVE				X	
LPCI INBOARD INJECTION VALVE	X			X	
LPCI OUTBOARD INJECTION VALVE	X			X	
ISOLATION VALVE (MOV)			X		
HPCI INJECTION VALVE			X		
RHR HX 2B NSW OUT ISO				X	
RBCCW X TIE, TR A				X	
RHRSW PUMP SUCTION ISO				X	
RHRSW PUMP SUCT XTIE				X	
NSW TO RBCCW				X	
RBCCW HX SERVICE WATER ISOL VALVE				X	
CS OUTBOARD INJECTION VALVE			X		
CS INBOARD INJECTION VALVE			X		
RHR/RWCU MOV (INBOARD)				X	
STEAM SUPPLY INBOARD ISOLATION VALVE		X			
SHUTOWN CLG INBOARD SUCTION THROTTLE VLV				X	
RHR/RWCU MOV (OUTBOARD)				X	
CONV SW PUMP 2A N HDR ISO				X	
CONV SW PUMP 2B N HDR ISO				X	
CONV SW PUMP 2C N HDR ISO				X	
RHRP-2C SP SUCTION VALVE				X	
SHUTDOWN COOLING SUCTION VALVE				X	
SHUTDOWN COOLING SUCTION VALVE				X	
TURBINE STEAM SUPPLY VALVE		X			
STEAM SUPPLY OUTBOARD ISOLATION VALVE		X			
HPCI/CST SUCTION VALVE			X		
HPCI/SP SUCTION VALVE			X		
HPCI/SP SUCTION VALVE			X		

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
TURBINE VACUUM BREAKER VALVE		X			
SP SOLENOID VALVE			X		
SP SOLENOID VALVE FOR LSH-N0I5A			X		
SP SOLENOID VALVE FOR LSH-N0I5A			X		
SP SOLENOID VALVE FOR LSH-N015B			X		
CONTAINMENT ATMOSPHERE SOLENOID VALVE					X
SOLENOID VALVE				X	
SOLENOID VALVE				X	
SP SOLENOID VALVE			X		
PILOT SOLENOID VALVE FOR CV-F053A				X	
DRYWELL SOLENOID VALVE			X		
BACKUP N2 DISCHARGE SOLENOID VALVE	X				
BACKUP N2 DISCHARGE SOLENOID VALVE	X				
BACKUP N2 SOLENOID VALVE	X				
BACKUP N2 SOLENOID VALVE	X				
SOLENOID VALVE FOR DRYWELLINST			X		
SP SOLENOID VALVE FOR LSH-N015B			X		
FAN FOR CS FAN COOLING UNIT A			X		
FAN FOR CS FAN COOLING UNIT B			X		
FAN FOR RHR FAN COOLING UNIT A				X	
FAN FOR RHR FAN COOLING UNIT B				X	
COIL FOR CS FAN COOLING UNIT A			X		
COIL FOR CS FAN COOLING UNIT B			X		
COIL FOR RHR FAN COOLING UNIT A				X	
COIL FOR RHR FAN COOLING UNIT B				X	
MOTOR GENERATOR SET A	X	X	X	X	
MOTOR GENERATOR SET B	X	X	X	X	
RPS POWER DIST. PNL -RPS A AND B	X	X			
DISTRIBUTION PANEL 12A			X		
NODE HZS -DISTRIBUTION PANEL 12B			X		
NODE H23 -DISTRIBUTION PANEL 4A				X	
NODE H37 -DISTRIBUTION PANEL 4AB				X	
NODE H24 -DISTRIBUTION PANEL 4B				X	
DISTRIBUTION PANEL 2A		X	X	X	
DISTRIBUTION PANEL 2B		X	X	X	
DISTRIBUTION PANEL 8A		X	X	X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
DISTRIBUTION PANEL 8B		X	X	X	
BATTERY 2A-1		X	X	X	
BATTERY 2A-2		X	X	X	
BATTERY 2B-1		X	X	X	
BATTERY 2B-2		X	X	X	
BATTERY CHARGER 2A-1		X	X	X	
BATTERY CHARGER 2A-2		X	X	X	
BATTERY CHARGER 2B-1		X	X	X	
BATTERY CHARGER 2B-2		X	X	X	
SIGNAL CONVERTER FOR SV-F053A		X			
RX PROTECTION & NSSS INSTRUMENT RACK	X	X			
RHRSW PUMP INLET PRESSURE				X	
RHRSW PUMP INLET PRESSURE				X	
RHRSW PUMP INLET PRESSURE				X	
CST LO WATER LEVEL ACTUATION OF HPCI			X		
CST LO WATER LEVEL ACTUATION OF HPCI			X		
LOW PRESSURE SWITCH				X	
RHR HX 2A PRESS DIFF SWITCH				X	
PRESSURE SWITCH				X	
SP PRESSURE TRANSMITTER				X	
SEAL COOLER LOW FLOW SWITCH				X	
SEAL COOLER LOW FLOW SWITCH				X	
SEAL COOLER LOW FLOW SWITCH				X	
SEAL COOLER LOW FLOW SWITCH				X	
SUPP POOL HI WATER LVL ACTUATION OF HPCI			X		
SUPP POOL HI WATER LVL ACTUATION OF HPCI			X		
TURBINE EXHAUST RUPTURE DIAPHRAM			X		
TURBINE EXHAUST RUPTURE DIAPHRAM			X		
CORE SPRAY SYSTEM A INSTRUMENT RACK			X		
HPCI INSTRUMENT RACK			X		
RHR CHANNEL A INSTRUMENT RACK				X	
RHR CHANNEL B INSTRUMENT RCK				X	
HPCI LEAK DETECTION SYSTEM A INSTR RACK			X		
ROOM COOLER DISCHARGE FLOW				X	
ROOM COOLER FLOW TRANSMITIER				X	
FAN/DAMPER LIMIT SWITCH				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
FAN/DAMPER LIMIT SWITCH				X	
N2 ACCUMULATOR LEVEL SWITCH	X				
N2 ACCUMULATOR PRESSURE SWITCH	X				
JET PUMP INSTRUMENT RACK			X		
JET PUMP INSTRUMENT RACK			X		
DRYWELL PRESSURE TRANSMITTER					X
DRYWELL PRESSURE TRANSMITTER					X
RNA/BACKUP N2 LO PRESSURE SWITCH	X				
RX PROTECTION & NSSS INSTRUMENT RACK	X				
RHRSW PUMP INLET PRESSURE				X	
BACKUP N2 PRESSURE TRANSMITTER	X				
PNS/BACKUP N2 LO PRESSURE SWITCH	X				
PNS/BACKUP N2 LO PRESSURE SWITCH	X				
BACKUP N2 PRESSURE TRANSMITTER	X	X			
BACKUP N2 PRESSURE TRANSMITTER	X	X		X	
BACKUP N2 PRESSURE TRANSMITTER	X	X			
RHRSWP DISCHARGE PRESSURE TRANSMITTER				X	
RHRSWP DISCHARGE PRESSURE TRANSMITTER				X	
RHRSWP DISCHARGE PRESSURE TRANSMITTER				X	
RHRSWP DISCHARGE PRESSURE TRANSMITTER				X	
NSW TO RBCCW FLOW TRANSMITTER				X	
CS TRAIN A LO DSCH PRESSURE SWITCH			X		
DRYWELL PRESSURE TRANSMITTER					X
SRV A FLOW TRANSMITTER		X			
SRV B FLOW TRANSMITTER		X			
SRV C FLOW TRANSMITTER		X			
SRV D FLOW TRANSMITTER		X			
SRV E FLOW TRANSMITTER		X			
SRV F FLOW TRANSMITTER		X			
SRV G FLOW TRANSMITTER		X			
SRV H FLOW TRANSMITTER		X			
SRV J FLOW TRANSMITTER		X			
SRV K FLOW TRANSMITTER		X			
SRV L FLOW TRANSMITTER				X	
NUCLEAR HDR PRESSURE SWITCH				X	
NUCLEAR HDR PRESSURE TRANSMITTER				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
AOV FOR STRAINER BACKWASH				X	
AOV FOR STRAINER BACKWASH				X	
SOLENOID VALVE (SV)				X	
SOLENOID VALVE (SV)				X	
DIFFERENTIAL PRESSURE CONTROLLER				X	
DIFFERENTIAL PRESSURE CONTROLLER				X	
RHRSW PUMP DISCHARGE TEMP				X	
RHRSW PUMP DISCHARGE TEMP				X	
RHRSW PUMP DISCHARGE TEMP				X	
RHRSW PUMP DISCHARGE TEMP				X	
CS RM TEMPERATURE SENSOR			X		
CS RM INDICATING BRIDGE			X		
CS RM TEMPERATURE SENSOR			X		
CS RM INDICATING BRIDGE			X		
RHR RM TEMPERATURE SWITCH				X	
RHR RM TEMPERATURE SWITCH				X	
TEMPERATURE SENSOR				X	
INDICATION BRIDGE				X	
RHR RM TEMPERATURE SWITCH				X	
RHR RM TEMPERATURE SWITCH				X	
HPCI RM TEMPERATURE SWITCH			X		
HPCI RM TEMPERATURE SWITCH		X			
TEMPERATURE SENSOR				X	
INDICATION BRIDGE				X	
RHR HX 2A TEMPERATURE ELEMENT				X	
RHRSW PUMP DISCHARGE TEMP				X	
RHRSW PUMP DISCHARGE TEMP				X	
RHRSW PUMP DISCHARGE TEMP				X	
RHRSW PUMP DISCHARGE TEMP				X	
DRYWELL TEMPERATURE ELEMENT					X
DRYWELL TEMPERATURE ELEMENT					X
DRYWELL TEMPERATURE ELEMENT					X
DRYWELL TEMPERATURE ELEMENT					X
DRYWELL TEMPERATURE ELEMENT					X
DRYWELL TEMPERATURE ELEMENT					X
DRYWELL TEMPERATURE ELEMENT					X

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
DRYWELL TEMPERATURE ELEMENT					X
TEMPERATURE SENSOR -SRV A		X			
TEMPERATURE SENSOR -SRV B		X			
TEMPERATURE SENSOR -SRV C		X			
TEMPERATURE SENSOR -SRV D		X			
TEMPERATURE SENSOR -SRV E		X			
TEMPERATURE SENSOR -SRV F		X			
TEMPERATURE SENSOR -SRV G		X			
TEMPERATURE SENSOR -SRV H		X			
TEMPERATURE SENSOR- SRV J		X			
TEMPERATURE SENSOR- SRV K		X			
TEMPERATURE SENSOR- SRV L		X			
DRYWELL TEMPERATURE ELEMENT					X
DRYWELL TEMPERATURE ELEMENT					X
DRYWELL TEMPERATURE ELEMENT					X
DRYWELL TEMPERATURE ELEMENT					X
DRYWELL TEMPERATURE ELEMENT					X
DRYWELL TEMPERATURE ELEMENT					X
DRYWELL TEMPERATURE ELEMENT					X
DRYWELL TEMPERATURE ELEMENT					X
SP TEMPERATURE ELEMENT					X
SP TEMPERATURE ELEMENT					X
SP TEMPERATURE ELEMENT					X
SP TEMPERATURE ELEMENT					X
MIAN STEAM LEAK DETECTION CABINET	X				
DIVISION I SPTMS SIGNAL RELAY DEVICE					X
DIVISION II SPTMS SIGNAL RELAY DEVICE					X
ENGINEERED SAFEGUARDS VERTICAL BOARD				X	
REACTOR CONTROL PANEL		X			X
POWER RANGE NEUTRON MONITORING PANEL	X				
RPS TRIP SYSTEM A	X	X			
RPS TEST & MONITOR PANEL		X			
RPS TRIP SYSTEM B		X			
FEEDWATER & REACTOR RECIRC INSTR PANEL			X		
PROCESS INSTRUMENTATION CABINET				X	
NSSS TEMP REC & LEAK DETECT VERT BOARD		X			

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
ROD POSITION INFORMATION SYSTEM CABINET	X				
ROD MANUAL CONTROL PANEL	X				
RHR A RELAY VERTICAL BOARD				X	
RHR B RELAY VERTICAL BOARD				X	
HPCI VERTICAL RELAY PANEL			X		
NSSS INBOARD VALVE REAL Y BOARD		X			
NSSS OUTBOARD VALVE RELAY BOARD		X			
BENCHBOARD AUXILIARY RELAY CABINET			X		
CORE SPRAY A RELAY VERTICAL BOARD			X		
CORE SPRAY B RELAY VERTICAL BOARD			X		
REACTOR ANNUNCIATOR CABINET	X				
TERMINAL CAB FOR SYSTEMS SW.EB.RCC & BAT				X	
MAIN CONTROL ROOM RTG BOARD		X	X		
TERMINAL CAB FOR SYSTEMS SW,EB,RCC & BAT				X	
RX.DG & CTRL BLOGS HVAC DIV-I TERMINAL CABINET				X	
RX.OG & CTRL BLOGS HVAC DIV-II TERMINAL CABINET				X	
RX CONT & TURB BLOG HVAC & TURB AUX CONT PNL				X	
RIP TERMINAL CABINET		X			
BOP RTG BOARD				X	
TERMINATING CABINET DIV-I		X			
TERMINATING CABINET DIV-II		X			
RIP TERMINAL CABINET DIV-II		X			
RIP TERMINAL CABINET DIV-I		X			
TRIP CALIBRATION CABINET -ECCS DIVISION I				X	
TRIP CALIBRATION CABINET -ECCS DIVISION II				X	
FLUID FLOW DETECT CAB FOR SRV POSITION IND		X			
POST-ACCIDENT MISC INSTRUMENT CABINET. DIV-I				X	
TSC/EOF COMPUTER ISOLATOR CABINET				X	
TSC/EOF COMPUTER ISOLA TOR CABINET				X	
POST ACCIDENT MISC INSTRUMENT CABINET				X	
BOP PROCESS INSTR POWER SUPPLY CABINET				X	
TURBINE CONTROLLER		X			
RECIRC PUMP B INSTRUMENT RACK			X		
CRD ACCUMULATOR MONITOR PANEL BANK 1 & 2	X				
MCC 2XDA-B26 TO 2XDB-350 XFER CONTACTOR				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
XFER C0NTACT0R PANEL FOR E41-F079-MO				X	
RHR SUCT OUTBD ISV ASSD STARTER PANEL				X	
PARTIAL WINDING HTR CAB FOR MCC 2PB				X	
PARTIAL WINDING HTR CAB FOR MCC 2PA				X	
LUBE OIL COOLER				X	
RHR HX 2A				X	
SRV A ACCUMULATOR		X			
SRV B ACCUMULATOR		X			
SRV C ACCUMULATOR		X			
SRV D ACCUMULATOR		X			
SRV E ACCUMULATOR		X			
SRV F ACCUMULATOR		X			
SRV G ACCUMULATOR		X			
SRV H ACCUMULATOR		X			
SRV J ACCUMULATOR		X			
SRV K ACCUMULATOR		X			
SRV L ACCUMULATOR		X			
NSW 2B SELF CLEANING STRAINER, MO				X	
RECIRCULATION PUMP A INSTRUMENT RACK	X		X		
FIRE DAMPER				X	
DIV-II TERM CAB FOR RTGB XU-2				X	
DIV-I TERM CAB FOR EB & ED SYSTEMS				X	
DG4 ESS LOGIC CABINET				X	
DG3 ESS LOGIC CABINET				X	
DG4 PIPE TRENCH HIGH WATER LEVEL SWITCH				X	
DG4 PIPE TRENCH HIGH WATER LEVEL SWITCH				X	
DG3 PIPE TRENCH HIGH WATER LEVEL SWITCH				X	
DG3 PIPE TRENCH HIGH WATER LEVEL SWITCH				X	
DG2 PIPE TRENCH HIGH WATER LEVEL SWITCH				X	
DG2 PIPE TRENCH HIGH WATER LEVEL SWITCH				X	
DGI PIPE TRENCH HIGH WATER LEVEL SWITCH				X	
DG1 PIPE TRENCH HIGH WATER LEVEL SWITCH				X	
DG B DELAY VALVE PIT FLDGD SWITCH				X	
DG4 PIPE TRENCH HIGH WATER LEVEL SWITCH				X	
DG3 PIPE TRENCH HIGH WATER LEVEL SWITCH				X	
DG2 PIPE TRENCH HIGH WATER LEVEL SWITCH				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
DG1 PIPE TRENCH HIGH WATER LEVEL SWITCH				X	
DG4 TANK ROOM HIGH FLOOD LEVEL SWITCH				X	
DG3 TANK ROOM HIGH FLOOD LEVEL SWITCH				X	
DG1 TANK ROOM HIGH FLOOD LEVEL SWITCH				X	
DG2 TANK ROOM HIGH FLOOD LEVEL SWITCH				X	
SUPPLY FAN LIMIT SWITCH				X	
SUPPLY FAN LIMIT SWITCH				X	
SUPPLY FAN LIMITS WITCH				X	
AC SUPPLY FAN -UNIT 1 & 2				X	
AC SUPPLY FAN -UNIT 2				X	
AIR COOLED CONDENSER				X	
ELECTRICHTR COIL - UNIT2				X	
120/24 AC TRANSFORMER				X	
ISOLVALVE				X	
ISOLVALVE				X	
C RM THERMOSTAT		X			
TEMPERATURE TRANSMITTER				X	
TEMPERATURE ELEMENT				X	
TEMP CONTROLLER				X	
SOLENOID FOR FV916B				X	
SOL VALVE FOR KS 1028				X	
SUPPLY FAN SOL VALVES				X	
SOL VALVE FOR KS 1027				X	
SUPPLY FAN SOL VALVES				X	
INSTRUMENT AIR LOW PRESS				X	
B AIR COMPRESSOR PRESS SWITCH				X	
A AIR COMPRESSOR PRESS SWITCH				X	
COOLING UNIT PRESSURE SWITCH				X	
COOLING UNIT PRESSURE SWITCH				X	
COOLING UNIT PRESSURE SWITCH				X	
COOLING UNIT PRESSURE SWITCH				X	
COOLING UNIT PRESSURE SWITCH				X	
MOISTURE CONTROLLER				X	
CONTROL PANEL	X			X	
TEMPERATURE TRANSMITTER				X	
HEATING COIL TIMER				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
HEATING COIL TIMER				X	
SUPPLY ISOL DAMPER				X	
DAMPER 2A-D OPERATOR				X	
DAMPER 21-D OPERATOR				X	
SUPPLY FAN DSCH FLOW SWITCH				X	
SUPPLY FAN DSCH FLOW SWITCH				X	
FIRE DAMPER				X	
FIRE DAMPER				X	
FIRE DAMPER				X	
SUBCOOLING CONDENSER				X	
HX				X	
AIR COOLED CONDENSER				X	
SUBCOOLING CONDENSER				X	
HX				X	
STEAM HUMIDIFIER				X	
ROLL TYPE FILTER				X	
AO DAMPER				X	
COOLING COIL - UNIT 2				X	
COOLING COIL-UNITS 1 & 2				X	
ELECTRIC HTR COIL - UNIT 1 & 2				X	
AO DAMPER -UNIT 2				X	
50 KVA POWER SUPPLY				X	
50 KVA POWER SUPPLY				X	
DG4 NSW UNIT 2 SUPPLY				X	
DG3 NSW UNIT 2 SUPPLY				X	
NSW UNIT 1 SUPPLY TO DG2				X	
NSW UNIT 1 SUPPLY TO DG1 JW				X	
DG1 NSW JW PRESSURE				X	
DG2 NSW PRESSURE SWITCH				X	
DG3 NSW PRESSURE SWITCH				X	
DG4 NSW PRESSURE SWITCH				X	
MCC DGB				X	
DG4 JACKET WATER TCV				X	
DG4 JACKET WATER TCV				X	
DG3 JACKET WATER TCV				X	
DG2 JACKET WATER TCV				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
DG2 JACKET WATER TCV				X	
DG3 JACKET WATER TCV				X	
DG1 JACKET WATER TCV				X	
DG1 JACKET WATER TCV				X	
DG4 JACKET WATER COOLER				X	
DG3 JACKET WATER COOLER				X	
DG2 JACKET WATER COOLER				X	
DG1 JACKET WATER COOLER				X	
DG1 JACKET WATER EXP TANK				X	
DG4 JACKET WATER EXP TANK				X	
DG3 JACKET WATER EXP TANK				X	
DG2 JACKET WATER EXP TANK				X	
DG4 LUBE OIL STRAINER				X	
DG1 LUBE OIL STRAINER				X	
DG3 LUBE OIL STRAINER				X	
DG2 LUBE OIL STRAINER				X	
DG4 LUBE OIL TCV				X	
DG1 LUBE OIL TCV				X	
DG4 LUBE OIL COOLER				X	
DG3 LUBE OIL COOLER				X	
DG2 LUBE OIL COOLER				X	
DG1 LUBE OIL COOLER				X	
DG3 LUBE OIL TCV				X	
DG2 LUBE OIL TCV				X	
DAY TANK LEVEL SWITCH				X	
4 DAY TANK LEVEL SWITCH				X	
DAY TANK LEVEL SWITCH				X	
4 DAY TANK 1 LEVEL SWITCH				X	
4 DAY TANK FOR DG1				X	
DG4 4 DAY TANK				X	
DG3 4 DAY TANK				X	
DG2 4 DAY TANK				X	
FUEL OIL TRANSFER PUMP 4B				X	
FUEL OIL TRANSFER PUMP 4A				X	
FUEL OIL TRANSFER PUMP 3B				X	
FUEL OIL TRANSFER PUMP 3A				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
FUEL OIL TRANSFER PUMP 2A				X	
FUEL OIL TRANSFER PUMP 2B				X	
FUEL OIL TRANSFER PUMP 1B				X	
FUEL OIL TRANSFER PUMP 1A				X	
480 VOLT UNIT SUBSTATION E8			X	X	X
480 VOLT UNIT SUBSTATION E7			X	X	X
SWITCHGEAR ASSEMBLY E4				X	
SWITCHGEAR ASSEMBLY E3				X	
DG1 START AIR TANK & FILTER				X	
DG1 START AIR TANK & FILTER				X	
DG4 START AIR TANK & FILTER				X	
DG4 START AIR TANK & FILTER				X	
DG3 START AIR TANK & FILTER				X	
DG3 START AIR TANK & FILTER				X	
DG2 AIR START TANK & FILTER				X	
DG2 AIR START TANK & FILTER				X	
DG1 AIR INTAKE FILTER				X	
DG4 AIR INTAKE SILENCER				X	
DG3 AIR INTAKE SILENCER				X	
DG2 AIR INTAKE SILENCER				X	
DG1 AIR INTAKE SILENCER				X	
DG4 AIR EXHAUST SILENCER				X	
DG3 AIR EXHAUST SILENCER				X	
DG2 AIR EXHAUST SILENCER				X	
DG1 AIR EXHAUST SILENCER				X	
MCC DGB				X	
DG1 START AIR SOLENOID				X	
EMERGENCY DIESEL GENERATOR 2				X	
MCC DGC				X	
CONTROL PANEL F-9776(17)				X	
DG SUMP PMP DGB-2G-1&2 ALTERNATOR PANEL				X	
DG4 AIR INTAKE FILTER				X	
DG3 AIR INTAKE FILTER				X	
DG2 AIR INTAKE FILTER				X	
MCC DGA				X	
DG4 CRANKCASE VACUUM BLOWER				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
DG4 START AIR SOLENOID				X	
DIESEL GEN 4 CONTROL PANEL				X	
EMERGENCY DIESEL GENERATOR 4				X	
EXCITATION PANEL				X	
DG4 ENGINE CONTROL PANEL				X	
DG3 CRANKCASE VACUUM BLOWER				X	
DG3 START AIR SOLENOID				X	
DG3 START AIR SOLENOID				X	
DIESEL GEN 3 CONTROL PANEL				X	
EMERGENCY DIESEL GENERATOR 3				X	
EXCITATION PANEL				X	
DG3 ENGINE CONTROL PANEL				X	
DG2 CRANKCASE VACUUM BLOWER				X	
DG2 START AIR SOLENOID				X	
DG2 START AIR SOLENOID				X	
DIESEL GEN 2 CONTROL PANEL				X	
EXCITATION PANEL				X	
DG2 ENGINE CONTROL PANEL				X	
DG1 CRANKCASE VACUUM BLOWER				X	
DIESEL GEN 1 CONTROL PANEL				X	
EMERGENCY DIESEL GENERATOR 1				X	
EXCITATION PANEL				X	
DG1 ENGINE CONTROL PANEL				X	
CB PANEL -120VAC EMERG PWR				X	
CB PANEL -120VAC EMERG PWR				X	
CB PANEL -120VAC EMERG PWR				X	
MCC 2XD				X	
MCC 2XM				X	
MCC 2XL				X	
MCC2XH				X	
MCC 2XG				X	
MCC 2XF				X	
MCC 2XE				X	
MCC 2XC				X	
MCC 2XB-2				X	
MCC 2XB				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
MCC 2XA-2				X	
MCC 2XA				X	
MCC 2PB				X	
MCC 2PA				X	
DISTRIBUTION PANEL 2E6				X	
DISTRIBUTION PANEL 2E7				X	
CB PANEL -120VAC EM ERG PWR				X	
MCC 2CB				X	
MCC 2CA				X	
CB PANEL -120VAC EMERG PWR				X	
DIST. PANEL TRANSFORMER MCC 2PB				X	
SW PANEL-120VAC EMERGE PWR				X	
RB PANEl-120VAC EMERG PWR				X	
CB PANEL -120VAC EMERG PWR				X	
120/208VAC MAIN UPS DP				X	
DIST. PANEL TRANSFORMER MCC 2PA				X	
SW PANEL-120VAC EMERGE PWR				X	
RB PANEL-120VAC EMERG PWR				X	
RB PANEL-12OVAC EMERG PWR				X	
CB PANEL -120VAC EMERG PWR				X	
CB PANEL -120VAC EMERG PWR				X	
DG4 START AIR SOLENOID				X	
DG1 START AIR SOLENOID				X	
CB BATTERY ROOM 2A SUP FAN 2C CKT BKR				X	
SW HDR OUTBOARD SUPPLY VALVE TO TURB BLDG				X	
SERVICE WATER TO CW PUMPS BEARINGS INBOARD ISOLATION VALVE				X	
LOOP A LOCAL DISCHARGE PRES IND				X	
CAD INJECTION LINE RELIEF VALVE			X		
VENT VALVE DOWNSTREAM OF 2-E21-FO-D002B				X	
INBOARD SUPPRESSION POOL PURGE EXHAUST VALVE OPEN POSITION SWITCH				X	
DIESEL GENERATOR INSTRUMENT TEST RACK CELL #1				X	
HPCI LUBE OIL EAST STRAINER DRAIN VALVE			X		
DG 3 ENG JKT WTR CLR SERVICE WTR INLET ISV				X	
JET ASSIST CONTROL TIME DELAY RLY				X	

Feature	Reactor Reactivity Control	Reactor Coolant Pressure Control	Reactor Coolant Inventory Control	Decay Heat Removal	Containment Function
NUCLEAR HEADER WTR PMP A DISCHARGE VALVE				X	
CB INTAKE AIR PLENUM CHLORINE SENSOR				X	
CB INTAKE AIR PLENUM CHLORINE SENSOR				X	
RX BLDG CLASS A HVAC DUCTS... WITH SUPPORTS				X	
VA-2D-CU-CB SUBCOOLING CONDENSER				X	
VA-2E-CU-CB SUBCOOLING CONDENSER				X	
DG1 ENG LUBE OIL LOW PRESS TRIP SW				X	

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Feature	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
RECIRCULATION PUMP A INSTRUMENT RACK	20	X		X			RB	X		X		X		X		AUX CONTR BRD	
CORE SPRAY SYSTEM A INSTRUMENT RACK	18			X			RB	X		X		X		X		AUX CONTR BRD	
HCU SCRAM WTR INLT ISV AIR OPERATOR	7	X					RB	X		X		X		X		CRD	
SCRAM VALVES PILOT AIR HEADER SOLENOID VALVE	8	X					RB	X		X		X		X		CRD	
250VDC MCC	1			X	X		RB	X		X		X		X		DC	
250VDC MCC	1			X	X		RB	X		X		X		X		DC	
HCU ACCUMULATOR HIGH LEVEL SWITCH	18	X					RB	X		X		X		X		CRD	
HCU ACCUMULATOR N2 PRESSURE SWITCH	18	X					RB	X		X		X		X		CRD	
JET PUMP INSTRUMENT RACK	18			X			RB	X		X		X		X		AUX CONTR BRD	
JET PUMP INSTRUMENT RACK	18			X			RB	X		X		X		X		AUX CONTR BRD	

Brunswick Steam Electric Plant Unit 2 Seismic Walkdown Report
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Feature	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
CRD ACCUMULATOR MONITOR PANEL BANK 1 & 2	20	X					RB	X		X		X		X		AUX CONTR BRD	
RHR SW BOOSTER PUMP 2A	5				X		RB	X		X		X		X		RHR	
DIV I N2 BACKUP BOTTLE RACK	0	X					RB	X		X		X		X		IA	
RHR SW PUMP 2C SUPPLY HDR PRESSURE SWITCH	16				X		RB	X		X		X		X		SW	
RX PROTECTION & NSS SYSTEM INSTRUMENT RACK	16	X	X				RB	X		X		X		X		AUX CONTR BRD	
HPCI INJECTION VALVE PM 87-260	8			X			RB	X		X		X		X		HPCI	
DIV I N2 BACKUP SUPPLY PRESS CONTR VALVE	7	X	X				RB	X		X		X		X		IA	
RHR SW PUMP C COOLER INLET VALVE	7				X		RB	X		X		X		X		SW	
SDV INBD VENT VLV AIR OPERATOR	7	X				X	RB	X		X		X		X		CRD	
DIV I N2 BACKUP SUPPLY RELIEF VALVE	7	X	X				RB	X		X		X		X		IA	
NUCLEAR HDR TO RBCCW HX ISOL VALVE	8				X		RB	X		X		X		X		SW	

Feature	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
NON INTERR IA PRESS SW IN SUPPLY HEADER PNS-225-2-170 PM 87-170	18	X					RB	X		X		X		X		PNS	
TEMP SWITCH FOR RHR SW PUMP C DISCH	19				X		RB	X		X		X		X		SW	
RBCCW HX SERVICE WATER ISOL VALVE	8				X		RB	X		X		X		X		SW	
DIV II NITROGEN BACKUP SUPPLY PRV	7	X	X		X		RB	X		X		X		X		IA	
DIV II NITROGEN BACKUP SUPPLY PCV	7	X	X				RB	X		X		X		X		IA	
DIV II NITROGEN BACKUP ISV	8	X	X				RB	X		X		X		X		IA	
VENT VALVE DOWNSTREAM OF 2-E21-FO-D002B	8			X	X		RB	X		X		X		X		CS	X
CONV HDR FEED VLV SW-V101 MOTOR OPERATOR	8				X		RB	X		X		X		X		SW	
RHR SERVICE WATER CROSSTIE INJECTION VLV	8			X			RB	X		X		X		X		RHR	
DIV II NITROGEN BACKUP SUPPLY RELIEF VALVE	7	X	X				RB	X		X		X		X		IA	
DIV II NITROGEN BACKUP FILTER	0	X	X	X			RB	X		X		X		X		IA	

Feature	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
DIV II NITROGEN BACKUP SUPPLY PRESS XMTR	18	X	X		X		RB	X		X		X		X		IA	
INBOARD SUPPR POOL PURGE EXHST VALVE OPEN POSITION SWITCH	14				X	X	RB	X		X		X		X		CAC	X
CS PMP ROOM B CLR OUTLET VLV AO	7				X		RB	X		X		X		X		SW	
RHR PUMP 2A DISCHARGE TO MIN FLOW LINE	8			X			RB	X		X		X		X		RHR	
HPCI LUBE OIL EAST STRAINER DRAIN VALVE	8			X			RB	X		X		X		X		HPCI	X
RHR HX 2A SW DISCHARGE VALVE	8				X		RB	X		X		X		X		RHR	
DIV II N2 BACKUP HDR RUPTURE DISC	0	X	X	X			RB	X		X		X		X		IA	
LOOP A LOCAL DISCHARGE PRES IND	14			X	X		RB	X		X		X		X		CS	X
CORE SPRAY PUMP 2A	6			X			RB	X		X		X		X		CS	
DRYWELL PRESSURE TRANSMITTER	18					X	RB	X		X		X		X		CAC	
CS PMP RM A CLR SW OUTLET VLV AO	7				X		RB	X		X		X		X		SW	

Feature	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
CRD PORTABLE ACCUM CHARGING SYS N2 TANK	0	X					RB	X		X		X		X		CRD	
RBCCW HX SERV WATER INLET FLOW TRANSM	18				X		RB	X		X		X		X		SW	
NUCLEAR HEADER SW PUMP B DISCHARGE VALVE PM 89-026	8				X		SWB	X		X		X		X		SW	
CONV SW PMP A DISCHARGE VALVE MO	8				X		SWB	X		X		X		X		SW	
CONV DISCH VLV NUC HDR MOT OP	8				X		SWB	X		X		X		X		SW	
SW TO CW PUMPS BEARINGS INBOARD ISOLATION VALVE	8				X		SWB	X		X		X		X		SW	X
NUCLEAR HEADER WTR PMP A DISCHARGE VALVE	8				X		SWB	X		X		X		X		SW	X
NUCLEAR SERVICE WATER PUMP 2A	6				X		SWB	X		X		X		X		SW	
NUCLEAR SERVICE WATER PUMP 2B	6				X		SWB	X		X		X		X		SW	
NUCLEAR HEADER PRESSURE TRANSMITTER	18				X		SWB	X		X		X		X		SW	
PARTIAL WINDING HTR CAB FOR MCC 2PB	20				X		SWB	X		X		X		X		SW	

Feature	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
SELF CLEANING STRAINER FOR NUCLEAR HDR SW 2B	21				X		SWB	X		X		X		X		SW	
NUC SW PMP 2A DISCHARGE STRAINER PCV	18				X		SWB	X		X		X		X		SW	
NUC SW PUMP 2A DISCH STRAINER BLOWDOWN SOLENOID VALVE	18				X		SWB	X		X		X		X		SW	
NUCLEAR SW PUMP 2A DIFF PRESS CONTROLLER	18				X		SWB	X		X		X		X		SW	
DG1 JKT WTR CLR OUTLET TEMP CTRL VLV	0				X		DGB	X		X		X		X		DG JKT & DEMIN WTR	
DG1 PIPE TRENCH HIGH WTR LEVEL SWITCH	0				X		DGB	X		X		X		X		DGB	
480V MCC DGA BUS	1				X		DGB	X		X		X		X		AC	
DG ENG STARTING AIR RIGHT HEADER SOL VALVE	8				X		DGB	X		X		X		X		DSA	
DG1 EXCITATION PANEL	20				X		DGB	X		X		X		X		DG	
DIESEL GEN 1 CONTROL PANEL	20				X		DGB	X		X		X		X		DG	
DG3 STARTING AIR TANK A NC #185725, NB #46739	21				X		DGB	X		X		X		X		DSA	
DG2 ENG LUBE OIL COOLER	21				X		DGB	X		X		X		X		LO	

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Feature	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
DG3 ENG JKT WTR CLR SW INLET ISV	8				X		DGB	X		X		X		X		SW	X
DIESEL GENERATOR NO 2	17				X		DGB	X		X		X		X		DG	
125/250VDC SWITCHBOARD 2A	3			X	X		CB	X		X		X		X		DC	
125/250VDC SWITCHBOARD 2B	3			X	X		CB	X		X		X		X		DC	
HIGH INERTIA AC MOTOR-GENERATOR SET 'A'	13	X	X	X	X		CB	X		X		X		X		RPS	
HIGH INERTIA AC MOTOR-GENERATOR SET 'B'	13	X	X	X	X		CB	X		X		X		X		RPS	
REACTOR PROTECTION SYS POWER DISRIBUTION PANEL	14	X	X				CB	X		X		X		X		RPS	
DIST SWBRD 2A DIV-I 125VDC BAT	15		X	X	X		CB	X		X		X		X		DC	
DIST SWBRD 2B DIV-II 125VDC BAT	15		X	X	X		CB	X		X		X		X		DC	
125VDC BATTERY 2A-2 CHARGER	16		X	X	X		CB	X		X		X		X		DC	
125VDC BATTERY 2B-1 CHARGER	16		X	X	X		CB	X		X		X		X		DC	
CB BATTERY ROOM 2A SUP FAN 2C CKT BKR	2				X		CB	X		X		X		X		AC	X

Feature	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
DG1 FUEL OIL TRANSFER PUMP 1B	5				X		DGB	X		X		X		X		FOD	
DG4 FUEL OIL TRANSFER PUMP 4B	5				X		DGB	X		X		X		X		FOD	
DG3 FUEL OIL 4-DAY STORAGE TANK	21				X		DGB	X		X		X		X		FOD	
SW HDR OUTBOARD SUPPLY VALVE TO TB	8				X		RWB	X		X		X		X		SW	X
DG3 LUBE OIL INLT DUPLEX STR W/XFER VALVE	0				X		DGB	X		X		X		X		LO	
480 VOLT UNIT SUBSTATION E7	2			X	X	X	DGB	X		X		X		X		AC	
DG3 ENG JKT WTR OUTLET TEMP CTRL VLV	7				X		DGB	X		X		X		X		DG JKT & DEMIN WTR	
DG2 STARTING AIR TANK A NC #185727, NB #46740	12				X		DGB	X		X		X		X		DSA	
RHR SW PMP 2A MOTOR HIGH TEMP RELAY	19				X		DGB	X		X		X		X		SW	
DIESEL GEN BLDG 125VDC DIST PANEL	14		X	X	X		DGB	X		X		X		X		DC	
4160V SWGR E4 AIR INTAKE FIRE DAMPER	10				X		DGB	X		X		X		X		HVAC	

Feature	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
CONDENSATE STORAGE TANK LOW WTR LVL	18			X			CST		X	X	X				X	HPCI	
CB U-1 &2 CTRL RM HVAC PNEUMATIC CTRL PNL	18	X			X		CB	X		X		X		X		HVAC	
VA-2D-CU-CB SUBCOOLING CONDENSER	0				X		CB		X	X	X				X	HVAC	X
VA-2E-CU-CB SUBCOOLING CONDENSER	0				X		CB		X	X	X				X	HVAC	X
VA-2E-CU-CB SUBCOOLING CONDENSER HEAT EXCH	21				X		CB		X	X	X				X	HVAC	
CTRL BLDG 125VDC DIST PNL	14				X		CB	X		X		X		X		DC	
CTRL BLDG 125VDC DIST PNL	14				X		CB	X		X		X		X		DC	
ENGINEERED SAFEGUARDS VERTICAL BOARD	20				X		CB	X		X		X		X		RTGB	
TERMINAL CAB FOR SYS SW, EB, BAT & RCC	20				X		CB	X		X		X		X		RTGB	
MAIN CONTROL ROOM RTG BOARD	20		X	X			CB	X		X		X		X		RTGB	
FLUID FLOW DET CAB FOR SRV POSITION IND	20		X				CB	X		X		X		X		RTGB	
ROD MANUAL CONTROL PANEL	20	X					CB	X		X		X		X		RTGB	

Feature	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
POST-ACCIDENT MISC INSTRUMENT CAB, DIV I	20				X		CB	X		X		X		X		RTGB	
MAIN STEAM LEAK DETECTION CABINET	20	X				X	CB	X		X		X		X		RX VESSEL & INT	
RPS TRIP SYSTEM A, MAIN CONTROL ROOM PNL	20	X	X				CB	X		X		X		X		RTGB	
TERMINATING CABINET DIV II	20		X				CB	X		X		X		X		RTGB	
REACTOR CONTROL PANEL	20		X			X	CB	X		X		X		X		RTGB	
FEEDWATER & REACTOR RECIRC INSTR PANEL	20			X			CB	X		X		X		X		RTGB	
BENCHBOARD AUXILIARY RELAY CABINET	20			X			CB	X		X		X		X		RTGB	
PRI STM LINE 'B' SAFETY/RELIEF VLV (ADS) PM 80-085	7		X				RB/DW		X	X			X	X		RX VESSEL & INT	
SAFETY RELIEF VALVE G & SOLENOID	7		X				RB/DW		X	X			X	X		RX VESSEL & INT	
DRYWELL AIR TEMPERATURE ELEMENT	19					X	RB/DW		X	X			X	X		CAC	
DRYWELL AIR TEMPERATURE ELEMENT	19					X	RB/DW		X	X			X	X		CAC	

Feature	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
DIST PNL 480-208/120V 30KVA XFMR .	4					X	SWB		X	X			X	X		AC	

Attachment 3: Base List 2

Feature
FUEL STORAGE POOL RECIRCULATION VALVE (DIFFUSER ISOLATION) 2-G41-V10
FUEL STORAGE POOL CLEAN-UP RETURN DIF CHK VLV 2G41-V24
SUCTION PIPING VORTEX BREAKER
DISCHARGE PIPING VORTEX BREAKER
FUEL STORAGE POOL CLEAN-UP RETURN DIF CHK VLV 2-G41-V8
FUEL STORAGE POOL RECIRCULATION VLV (DIFFUSER ISOLATION) 2-G41-V9

Attachment 4: Rapid Drain-Down List

Feature	Evaluation
Skimmer Surge tanks and Skimmer piping	A seismic induced failure of the Skimmer Surge Tanks, the connected piping, or the scupper drains would not result in drainage of the SFP water below normal pool level.
Fuel Pool Cooling and Filtering System (FPCFS)	The FPCFS removes water from the surface of the pool, runs it through filters and heat exchangers and returns it to the bottom of the pool through two pipes, each of which have a set of check and globe valves located at the pool side as they enter the pool. These design features in combination with their arrangement, piping class, their location with respect to one another and quarterly testing provides the basis for concluding that this system does not provide the possibility of a rapid drain-down event.
Leak Chase Drains	The leak chase drain channel system is located behind the SFP liner plate. The SFP liner is part of the Seismic Class 1 SFP structure and, therefore a seismic induced failure need not be postulated. The system has the capability of detecting leakage and manually isolating for any induced line leak (e.g., puncture, weld defect, etc.) which would be within the make-up capability of the system. This provides the basis for concluding that this system does not provide the possibility of a rapid drain-down event.
Fuel Transfer Gates	The fuel transfer gates are part of the Seismic Class 1 structure and have inflatable seals that are maintained via compressed air with a nitrogen bottle backup supply. The nitrogen is attached to the supply system through a check valve to maintain pressure in the system should air pressure be lost. This redundancy provides the basis for concluding that fuel transfer gates and their associated drain system does not provide the possibility of a rapid drain-down event.
Supplemental Spent Fuel Pool Cooling System (SSFPCS)	Temporarily installed spool pieces are used to connect the permanent suction and return piping and extend six feet and ten feet, respectively, into the spent fuel pool. Both spool pieces have two inch holes in them located just below the normal level of the spent fuel pool and above the minimum Technical Specification water level to prevent inadvertent siphoning of water out of the spent fuel pool. Because of these design attributes, this system is not a candidate for rapid drain down.
Refueling Condition	The Refueling condition was investigated for the time when the reactor well is flooded for refueling. This condition was not considered a rapid drain down possibility due to the protections in place during refueling, the addition of a Class 1 plate for volume protection, the Class I piping, installed drain plugs, and the bellows design.

The items on this list have been evaluated as not having the ability to cause a rapid drain down. Therefore, there are no rapid drain down items to add to the SWEL 2 list.

Attachment 5: SWEL 2

Feature	Building	Rapid Drain-Down
FUEL STORAGE POOL RECIRCULATION VALVE (DIFFUSER ISOLATION) 2-G41-V10	RB	No
FUEL STORAGE POOL CLEAN-UP RETURN DIF CHK VLV 2-G41-V24	RB	No
FUEL STORAGE POOL CLEAN-UP RETURN DIF CHK VLV 2-G41-V8	RB	No
FUEL STORAGE POOL RECIRCULATION VLV (DIFFUSER ISOLATION) 2-G41-V9	RB	No

Attachment 8: Peer Review Report

Brunswick Nuclear Power Plant Unit 2 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 Brunswick Nuclear Plant (BNP-U2). 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 BNP-U2 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. Justification was adequately provided for equipment classes that could not have a representative component inspected. 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 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 113 components in SWEL-1 and an additional 4 items in SWEL-2 totaling 117 selected items for the combined SWEL. The number of items on SWEL-01 was within the EPRI recommended range of 90-120 items. The SWEL was taken from the A-46 SSEL. The number of items inspected at the site is within the EPRI guidance.

The process used to select the SWEL items, inclusion of Operations Personnel into the selection of the items, IPEEE 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 to reflect conditions. Intermediate guidance resulting from the reviews during the inspection process was provided.

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 expectations and are 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 CAP 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 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 PASCs were reviewed for the unit and the classification was determined appropriate.

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 condition into BNP/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 issues were encountered at the BNP-U2 site and the site had already documented the condition within the CAP. The condition of the site and the aggressive identification process resulted in a general impression of the SWEs that 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 BNP-U2 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 with seismic demand for the region.

List of Regulatory Commitments

The following table identifies the actions in this document to which the Brunswick Steam Electric Plant (BSEP) has committed. Statements in this submittal, with the exception of those in the table below, are provided for information purposes and are not considered commitments. Please direct questions regarding these commitments to Mr. Lee Grzeck, Manager - Regulatory Affairs, at (910) 457-2487.

Commitment	Completion Date
Complete seismic walkdown inspections for the 21 inaccessible BSEP Unit 1 equipment items listed in Section 5.6 (i.e., Items 1 through 21) of the seismic walkdown report and submit updated BSEP Unit 1 seismic walkdown report.	September 30, 2016
Complete seismic walkdown inspections for the 23 inaccessible BSEP Unit 2 equipment items listed in Section 5.6 (i.e., Items 1 through 15, 18, 20, 21, and 23 through 27) of the seismic walkdown report and submit updated BSEP Unit 2 seismic walkdown report.	September 30, 2017