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Edwin I. Hatch Nuclear Plant – Units 1 and 2
Fukushima Near-Term Task Force Recommendation 2.1 Seismic
Limited-Scope High Frequency Confirmation Evaluation

References:

1. NRC Letter, "Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 12, 2012 (ML12053A340)
2. Letter to NRC, "Edwin I. Hatch Nuclear Plant – Units 1 and 2, Seismic Hazard and Screening Report for CEUS Sites," dated March 31, 2014 (ML14092A017).
3. NRC Letter, "Edwin I. Hatch Nuclear Plant – Units 1 and 2, Staff Assessment of Information Pursuant to 10 CFR 50.54(f) Seismic Hazard Reevaluations Relating to Recommendation 2.1 of the NTTF Review of Insights from the Fukushima Dai-ichi Accident," dated April 27, 2015 (ML15097A424).
4. NEI letter, "Request for NRC Endorsement of High Frequency Program: Application Guidance for Functional Confirmation and Fragility Evaluation (EPRI 3002004396)," dated July 30, 2015 (ML15223A100 and ML15223A102).
5. NRC letter, "Endorsement of Electric Power Research Institute Final Draft Report 3002004396, 'High Frequency Program: Application Guidance for Functional Confirmation and Fragility,'" dated September 17, 2015 (ML15218A569)
6. NRC Letter, "Final Determination of Licensee Seismic Probabilistic Risk Assessments under the Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 "Seismic" of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated October 27, 2015 (ML15194A015).

Ladies and Gentlemen:

On March 12, 2012, the Nuclear Regulatory Commission (NRC) issued Reference 1 to all power reactor licensees and holders of construction permits in active or deferred status. Enclosure 1 of Reference 1 requested each addressee in the Central and Eastern United States (CEUS) to submit a Seismic Hazard Evaluation and Screening Report. The requested information was submitted to the NRC for the Edwin I. Hatch Nuclear Plant Units 1 and 2 (Plant Hatch) by Southern Nuclear Operating Company (SNC) on March 31, 2014 (Reference 2). On April 27, 2015, SNC received the NRC's staff assessment of the Plant Hatch seismic hazard and the resulting Ground Motion Response Spectra (GMRS) (Reference 3).

The NRC issued the final determination of licensee seismic probabilistic risk assessment on October 27, 2015 (Reference 6). This NRC letter requested that Plant Hatch perform a limited-scope high frequency confirmation evaluation. By Reference 4, Nuclear Energy Institute (NEI) submitted an Electric Power Research Institute (EPRI) report containing high frequency confirmation guidance that was endorsed by the NRC staff in Reference 5.

The enclosure to this letter provides the 2.1 Seismic Limited-Scope High Frequency Confirmation Evaluation Report for Plant Hatch. As summarized in Section 5 of the report, the components evaluated either had adequate seismic capacity or existing operator actions were already in place. No additional modifications or evaluations are necessary.

This letter is the formal and final response to Fukushima Near-Term Task Force Recommendation 2.1 Seismic limited-scope high frequency evaluation as requested in the NRC's final determination letter (Reference 6).

This letter contains no new NRC commitments. If you have any questions, please contact Matt Euten at 205.992.7673.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 22nd day of August 2017.

Respectfully submitted,

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JTW/MRE/GLS

Enclosure: 2.1 Seismic High Frequency – Hatch

cc: Regional Administrator, Region II
NRR Project Manager – Hatch
Senior Resident Inspector – Hatch
Director, Environmental Protection Division – State of Georgia
RType: CHA02.004

Edwin I. Hatch Nuclear Plant – Units 1 and 2
Fukushima Near-Term Task Force Recommendation 2.1 Seismic
Limited-Scope High Frequency Confirmation Evaluation

Enclosure

2.1 Seismic High Frequency – Hatch

(79 pages)

Executive Summary

The purpose of this report is to provide information as requested by the Nuclear Regulatory Commission (NRC) in its March 12, 2012 letter issued to all power reactor licensees and holders of construction permits in active or deferred status [1]. In particular, this report provides information requested to address the High Frequency Confirmation requirements of Item (4), Enclosure 1, Recommendation 2.1: Seismic, of the March 12, 2012 letter [1].

Following the accident at the Fukushima Dai-ichi nuclear power plant resulting from the March 11, 2011, Great Tohoku Earthquake and subsequent tsunami, the Nuclear Regulatory Commission (NRC) established a Near Term Task Force (NTTF) to conduct a systematic review of NRC processes and regulations and to determine if the agency should make additional improvements to its regulatory system. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. Subsequently, the NRC issued a 50.54(f) letter on March 12, 2012 [1], requesting information to assure that these recommendations are addressed by all U.S. nuclear power plants. The 50.54(f) letter requests that licensees and holders of construction permits under 10 CFR Part 50 reevaluate the seismic hazards at their sites against present-day NRC requirements and guidance. Included in the 50.54(f) letter was a request that licensees perform a "confirmation, if necessary, that SSCs, which may be affected by high-frequency ground motion, will maintain their functions important to safety."

EPRI 1025287, "Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic" [6] provided screening, prioritization, and implementation details to the U.S. nuclear utility industry for responding to the NRC 50.54(f) letter. This report was developed with NRC participation and was subsequently endorsed by the NRC. The SPID included guidance for determining which plants should perform a High Frequency Confirmation and identified the types of components that should be evaluated in the evaluation.

Subsequent guidance for performing a High Frequency Confirmation was provided in EPRI 3002004396, "High Frequency Program, Application Guidance for Functional Confirmation and Fragility Evaluation," [8] and was endorsed by the NRC in a letter dated September 17, 2015 [3]. Final screening identifying plants needing to perform a High Frequency Confirmation was provided by NRC in a letter dated October 27, 2015 [2].

This report describes the High Frequency Confirmation evaluation undertaken for Southern Nuclear Operating Company's (SNC) Edwin I. Hatch Nuclear Plant Units 1 and 2 (Plant Hatch). The objective of this report is to provide summary information describing the High Frequency Confirmation evaluations and results. The level of detail provided in the report is intended to enable NRC to understand the inputs used, the evaluations performed, and the decisions made as a result of the evaluations.

EPRI 3002004396 [8] is used for the Plant Hatch engineering evaluations described in this report. In accordance with Reference [8], the following topics are addressed in the subsequent sections of this report:

- Process of selecting components and a list of specific components for high-frequency confirmation
- Estimation of a vertical ground motion response spectrum (GMRS)
- Estimation of in-cabinet seismic demand for subject components
- Estimation of in-cabinet seismic capacity for subject components
- Summary of subject components' high-frequency evaluations

In performing this High Frequency Confirmation, Southern Nuclear has chosen to use information from the Plant Hatch Seismic Probabilistic Risk Assessment (SPRA). During early discussions between the NRC and SNC concerning possible approaches for this 2.1 Seismic High Frequency submittal, the NRC called attention to the recent Peer Review for this Plant Hatch SPRA. That SPRA has been successfully reviewed against the ASME/ANS PRA standard [13] and meets the Capability Category II requirements of the standard [13]. For details about the Peer Review, please see Appendix C of this report. With respect to high frequency components, the scope of high-frequency components considered in this report were those identified as part of the SPRA which includes and exceeds the scope of components that require a 2.1 Seismic High Frequency evaluation. Therefore, this submittal draws from that SPRA work and supplements it as appropriate to ensure consistency with the requirements of Item (4), Enclosure 1, Recommendation 2.1: Seismic, of the March 12, 2012 letter [1].

In conclusion, Plant Hatch has performed a High Frequency Confirmation evaluation in response to the NRC's 50.54(f) letter [1] using the guidance of EPRI report 3002004396 [8] and no additional actions are necessary. The selection process identified a total of 194 components for evaluation. As summarized in Table B-1 in Appendix B, 179 of the devices have adequate seismic capacity and 15 components did not have adequate seismic capacity following the criteria in Section 4.5 of Reference [8]. The 15 components that did not have adequate seismic capacity are all General Electric (GE) model CFD relays. These relays are known to have a low seismic capacity. Plant Hatch already has operator actions in place that can adequately resolve potential seismic concerns regarding these relays. Therefore, per Section 4.6 of Reference [8], no additional actions are necessary.

1 Introduction

1.1 Purpose

The purpose of this report is to provide information as requested by the NRC in its March 12, 2012 50.54(f) letter issued to all power reactor licensees and holders of construction permits in active or deferred status [1]. In particular, this report provides requested information to address the High Frequency Confirmation requirements of Item (4), Enclosure 1, Recommendation 2.1: Seismic, of the March 12, 2012 letter [1].

1.2 Background

Following the accident at the Fukushima Dai-ichi nuclear power plant resulting from the March 11, 2011, Great Tohoku Earthquake and subsequent tsunami, the Nuclear Regulatory Commission (NRC) established a Near Term Task Force (NTTF) to conduct a systematic review of NRC processes and regulations and to determine if the agency should make additional improvements to its regulatory system. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. Subsequently, the NRC issued a 50.54(f) letter on March 12, 2012 [1], requesting information to assure that these recommendations are addressed by all U.S. nuclear power plants. The 50.54(f) letter requests that licensees and holders of construction permits under 10 CFR Part 50 reevaluate the seismic hazards at their sites against present-day NRC requirements and guidance. Included in the 50.54(f) letter was a request that licensees perform a "confirmation, if necessary, that SSCs, which may be affected by high-frequency ground motion, will maintain their functions important to safety."

EPRI 1025287, "Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic" [6] provided screening, prioritization, and implementation details to the U.S. nuclear utility industry for responding to the NRC 50.54(f) letter. This report was developed with NRC participation and is endorsed by the NRC. The SPID included guidance for determining which plants should perform a High Frequency Confirmation and identified the types of components that should be evaluated in the evaluation.

Subsequent guidance for performing a High Frequency Confirmation was provided in EPRI 3002004396, "High Frequency Program, Application Guidance for Functional Confirmation and Fragility Evaluation" [8], and was endorsed by the NRC in a letter dated September 17, 2015 [3]. Final screening identifying plants needing to perform a High Frequency Confirmation was provided by NRC in a letter dated October 27, 2015 [2].

On March 31, 2014, Plant Hatch submitted a reevaluated seismic hazard to the NRC as a part of the Seismic Hazard and Screening Report [4]. By letter dated October 27, 2015 [2], the NRC transmitted the results of the screening and prioritization review of the seismic hazards reevaluation.

This report describes the High Frequency Confirmation evaluation undertaken for Plant Hatch using the methodologies in EPRI 3002004396, "High Frequency Program, Application Guidance for Functional Confirmation and Fragility Evaluation," as endorsed by the NRC in a letter dated September 17, 2015 [3].

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

In performing this High Frequency Confirmation, Southern Nuclear has chosen to use information from the Plant Hatch Seismic Probabilistic Risk Assessment (SPRA). During early discussions between the NRC and SNC concerning possible approaches for this 2.1 Seismic High Frequency submittal, the NRC called attention to the recent Peer Review for this Plant Hatch SPRA. That SPRA has been successfully reviewed against the ASME/ANS PRA standard [13] and meets the Capability Category II requirements of the standard [13]. For details about the Peer Review, please see Appendix C of this report. With respect to high frequency components, the scope of high-frequency components considered in this report were those identified as part of the SPRA which includes and exceeds the scope of components that require a 2.1 Seismic High Frequency evaluation. Therefore, this submittal draws from that SPRA work and supplements it as appropriate to ensure consistency with the requirements of Item (4), Enclosure 1, Recommendation 2.1: Seismic, of the March 12, 2012 letter [1].

1.3 APPROACH

EPRI 3002004396 [8] is used for the Plant Hatch engineering evaluations described in this report. Section 4.1 of Reference [8] provided general steps to follow for the high frequency confirmation component evaluation. Accordingly, the following topics are addressed in the subsequent sections of this report.

- Plant Hatch SSE and GMRS Information (as well as the Uniform Hazard Response Spectra information that was generated for the 2.1 seismic hazard [4] and used in the SPRA)
- Selection of components and a list of specific components for high-frequency confirmation (via the SPRA)
- Estimation of seismic demand for subject components (via the SPRA)
- Estimation of seismic capacity for subject components
- Summary of subject components' high-frequency evaluations
- Summary of Results

Note that, as discussed in Section 1.2 of this report, Plant Hatch is using the SPRA information in performing the work associated with the High Frequency Confirmation guidance of Reference [8]. Plant Hatch's approach generally follows the steps listed above but uses the SPRA as a supplement.

1.4 PLANT SCREENING

Plant Hatch submitted reevaluated seismic hazard information including GMRS to the NRC on March 31, 2014 [4]. In a letter dated April 27, 2015, the NRC staff concluded that the submitted GMRS adequately characterizes the reevaluated seismic hazard for the Plant Hatch site [12]. Note that the Plant Hatch SPRA seismic hazard is consistent with the seismic hazard documented in Reference [4].

The NRC final screening determination letter [2] concluded that the Plant Hatch GMRS to SSE comparison resulted in a need to perform a High Frequency Confirmation in accordance with the screening criteria in the SPID [6].

2 Selection of Components for High-Frequency Screening

The fundamental objective of the high frequency confirmation review is to determine whether the occurrence of a seismic event could cause credited equipment to fail to perform as necessary. Recently, a Seismic Probabilistic Risk Assessment (SPRA) has been performed for Plant Hatch that meets the requirements of the ASME/ANS PRA Standard in Reference [13], including the closure of all Peer Review Findings and Observations in accordance with NEI 12-13 Appendix X, Reference [17]. The equipment and associated contact control devices selected for the Hatch SPRA envelops the more focused requirements described in EPRI 3002004396, Reference [8]. The EPRI 3002004396 high frequency confirmation approach focuses on achieving a safe and stable plant state following a seismic event. This state is achieved by confirming that key plant safety functions critical to immediate plant safety are preserved (reactor trip, reactor vessel inventory and pressure control, and core cooling) and that the plant operators have the necessary power available to achieve and maintain this state immediately following the seismic event (AC/DC power support systems). The SPRA seismic equipment list (SEL) and associated relay chatter evaluation not only includes all the components that would be included in the optimized evaluation process of Reference [8], but also includes additional components for other systems and safety functions, such as long-term decay heat removal and containment integrity. Therefore, the list of contact control devices from the SPRA was selected for the high frequency confirmation process.

Based on the guidance in EPRI 3002004396, Reference [8], the components that need a high frequency confirmation are the contact control devices subject to intermittent states in seal-in or lockout circuits (SILO). In addition, the SPRA also identified other contact control devices that could potentially leave a component in an undesired state, but do not have SILO circuits. An example is relay contacts that are associated with parameter control circuits, but do not have a SILO function. Accordingly, the objective of the review as stated in Section 4.2.1 of Reference [8] is to determine if seismic induced high frequency relay chatter would prevent the completion of the following key functions:

- Reactor Trip/Scram
- Reactor Coolant System/Reactor Vessel Inventory Control
- Reactor Coolant System/Reactor Vessel Pressure Control
- Core Cooling
- AC/DC Power Support Systems

As mentioned above, Southern Nuclear has chosen to use information from the Plant Hatch SPRA performed for risk informed applications. Documents, including calculations listed in References [15.1 through 15.10], were developed in support of the SPRA. Those documents [15.1 through 15.10] contain detailed information about equipment lists, fragility calculations, and chatter evaluations for the SPRA. The following steps outline the process used by SNC to ensure that the List of Components provided in Table B-1 envelops EPRI 3002004396 [8] requirements.

- Step 1 – Verification that the selection process for the larger SPRA scope of equipment meets and envelops the guidance in EPRI 3002004396, Reference [8].
- Step 2 – Verification that the Relay Chatter Evaluation for the larger SPRA scope meets and envelops the guidance in EPRI 3002004396, Reference [8].
- Step 3 – Development of the List of Components in Table B-1 of this report based on the SPRA.

Step 1 - Verification that the selection process for the larger SPRA scope of equipment meets and envelops the guidance in EPRI 3002004396

The development of the SPRA seismic equipment list (SEL) is described in detail in H-RIE-SEIS-U00-002-001 "Seismic Equipment List and SEL Walkdown Report – Seismic PRA", Reference [15.1]. The development process was compared to the guidance in EPRI 3002004396, Reference [8] to ensure that the SPRA SEL included and enveloped the equipment for the high frequency confirmation process. The Template for the 2.1 High Frequency Confirmation Report, and documentation from industry webcasts was also reviewed. A brief description of the relevant SEL equipment selection process follows.

1. Potential seismic-induced initiating events and consequential events were identified. These included loss of offsite power, loss of coolant accidents (LOCA), line breaks outside containment, and anticipated transient without SCRAM (ATWS).
2. The safety functions that must be fulfilled in response to these initiating events were delineated:
    - Reactivity control
    - Reactor coolant system pressure control
    - Reactor coolant system inventory control
    - Decay heat removal
    - Containment isolation and integrity
3. The associated frontline and support systems that can be used to meet each function for core damage and large early release accident sequences were identified. The selected systems must enable the plant to be in a safe and stable state at the end of the 24-hour mission time.
4. Using the Hatch P&IDs and electrical diagrams and the internal events PRA, the equipment required for system function and pressure boundary integrity was listed.

As can be seen, the safety functions used for the SPRA envelop the safety functions identified for the guidance in EPRI 3002004396, Reference [8]. The AC/DC Power Support Systems are identified with the frontline systems during item 3 above. Based on this evaluation, the selection of equipment for the SPRA was verified to meet the EPRI high frequency confirmation guidance [8].

Step 2 - Verification that the Relay Chatter Evaluation for the larger SPRA scope meets and envelops the guidance in EPRI 3002004396

The evaluation of relay chatter (and similar contact control devices) is described in detail in H-RIE-SEIS-U00-007-001 "Relay Chatter Evaluation Report", Reference [15.8].

For the Hatch SPRA, each of the chatter circuits for equipment in the Seismic Equipment List was evaluated to determine if relay chatter could cause the circuit to actuate the equipment to an undesired state. The steps used to meet the ASME/ANS PRA Standard, Reference [13] were:

- For the equipment identified for the SPRA in the SEL, identify the normal state (position or status) during plant operation, and the desired state (position or status) after the seismic event and loss of offsite power.
- For the equipment, use the elementary and one-line electrical diagrams to identify the contact control devices and chatter circuits that could cause the component to be actuated spuriously to an undesired state.
- For each contact control device and chatter circuit, document the circuit type, including if it is a seal-in or lockout (SILO) circuit.
- For each chatter circuit, determine the impacts on the component state if relays in the circuit contacts were to chatter, and identify the resolution in terms of the potential for relay chatter to impact the component state.
- Identify those relays and contact control devices that are chatter not acceptable, or require operator corrective action.

The resulting evaluation of "chatter acceptable" or "chatter not acceptable" is documented in the SPRA electronic database for all equipment and for each potential chatter circuit. Based on the evaluations, the contact control devices that were "chatter not acceptable" are primarily of three chatter circuit types: seal-in circuit, parameter control circuit, and protection circuits. The seal-in circuits are often part of the standard motor control centers or switchgear equipment, but can also be limit switches for motor- operated valves (MOVs) and other types of components. The parameter control circuits typically control the component based on a parameter such as flow, level, temperature, or pressure. Protection circuits are generally used for breakers for protecting the breaker or component from electrical abnormalities, and often have lockout circuitry.

Based on the SPRA relay chatter evaluation process, the identification of contact control devices that are chatter not acceptable was verified to meet the EPRI high frequency confirmation guidance [8].

Step 3 – Development of the List of Components in Table B-1 based on the SPRA

The list of chatter not acceptable components (contact control devices) is provided in Table B-1 of this report. Note that this list not only includes SILO circuits, but also includes other circuits that could place a component in an undesired state. This list has been verified to meet the requirements of 2.1 Seismic High Frequency, as defined in EPRI 3002004396, "High Frequency Program, Application Guidance for Functional Confirmation and Fragility Evaluation," section 4.2 [8]. Those systems/functions specific to 2.1 Seismic High Frequency assessment are listed at the beginning of Section 2 of this report.

In summary, the list of components in Table B-1 of this report was developed based on the SPRA, and meets the requirements for the high frequency evaluation [8].

2.1 Reactor Trip/SCRAM

The reactor trip/SCRAM function is identified as a key function in Reference [8] to be considered in the High Frequency Confirmation. The same report also states that "the design requirements preclude the application of seal-in or lockout circuits that prevent reactor trip/SCRAM functions" and that "No high-frequency review of the reactor trip/SCRAM systems is necessary."

2.2 Reactor Vessel Inventory Control

The reactor coolant system/reactor vessel inventory control systems were reviewed for contact control devices in circuits that would create a Loss of Coolant Accident (LOCA). The focus of the review was contact control devices that could lead to a significant leak path. Check valves in series with active valves would prevent significant leaks due to mis-operation of the active valve; therefore, SILO circuit reviews were not required for those active valves with series check valves.

The scope of treatment for this function was limited to actuation of those valves that effectively can create loss of coolant type events. This included the safety relief valves (SRVs). The focus of the review was to determine if the circuit designs for these types of valves are such that seal-in or latching circuits exist that would maintain the valves in an undesired state following the seismic event. As an example, if non-throttling motor operated valves are applied to provide a flow isolation function, the associated valve control circuits typically include a seal-in feature such that once energized would remain energized until the valve stroke is completed.

Reactor coolant system/reactor vessel inventory control system reviews were performed for valves associated with the following functions:

- Nuclear Steam Supply Shutoff (main steam and feedwater)
- Reactor Water Clean-Up
- Reactor Core Isolation Cooling
- Residual Heat Removal (low pressure coolant injection)
- Core Spray
- High Pressure Coolant Injection
- Safety/Relief (including automatic depressurization system)
- Reactor Head Vent

2.3 Reactor Vessel Pressure Control

The reactor vessel pressure control function is identified as a key function in Reference [8] to be considered in the High Frequency Confirmation. The same report also states that "required post event pressure control is typically provided by passive devices" and that "no specific high frequency component chatter review is required for this function."

2.4 Core Cooling

The reactor core isolation cooling (RCIC) system and the high-pressure coolant injection (HPCI) core cooling and decay heat removal system are initially independent of AC power. These systems were reviewed for contact control devices that would prevent these core cooling systems from functioning. In addition, the AC dependent core cooling and decay heat removal systems (RHR/LPCI and core spray) were also included in the assessment of contact control devices, which exceeds the requirements of the EPRI 3002004396 guidance [8].

The relay chatter impacts that could affect this function would be those that would cause the pumps to be tripped, the flow control valves to close and remain closed, or the isolation of the steam for the RCIC and HPCI turbine-driven pumps.

For BWR plants, such as Plant Hatch, the decay heat removal mechanism involves the transfer of mass and energy from the reactor vessel to the suppression pool. This requires the replacement of that mass to the reactor vessel via one of the core cooling systems. As an example, the RCIC system is discussed below. The HPCI, RHR/LPCI, and core spray systems were evaluated in a similar manner. For RCIC, the following functions were reviewed:

- Steam from the reactor pressure vessel to the RCIC turbine and exhausted to the suppression pool.
- Coolant from the condensate storage tank (CST) or suppression pool to the reactor via the RCIC pump.
- Steam from the reactor pressure vessel vented to the suppression pool via the Safety Relief Valves (SRVs). (Note that the SRVs were already reviewed for the RCS/Reactor Vessel Inventory Control function discussed above.)

Only contact devices which could render the RCIC system inoperable were considered, since operation is desired for the core cooling function.

RCIC operation following a seismic event can be compromised by contact chatter leading to a false RCIC isolation signal or false RCIC turbine trip. Chatter in the contacts of the RCIC Isolation Signal Relay or Steam Line High Differential Pressure Time Delay Relay; or coincident chatter in the Turbine Exhaust Diaphragm High Pressure Relays, or Reactor Pressure Relays; may lead to a RCIC Isolation Signal and seal-in of the signal relay. This would cause the RCIC Isolation Valves to close and the RCIC Trip and Throttle Valve to trip. Chatter in the contact control devices that drive those relays could also lead to closure and trip.

Closure of the Trip and Throttle Valve, with the associated RCIC turbine-driven pump trip, could occur due to chatter in the Remote Trip Circuit. Chatter of contact control devices in the Turbine Trip Auxiliary Relay, the Turbine Exhaust High Pressure Relays, the Pump Suction Low Pressure Relay, and the Isolation Signal Relays could isolate the Trip and Throttle valve. Similar chatter in the contact devices that drive those relays could also lead to a turbine trip.

Circuit drawings for valves in the flow-paths from the CST or suppression pool were also reviewed to identify any contact control devices that could cause the valves to be placed in an undesired state.

Based on this relay chatter review, contact control devices that could impact the core cooling and decay heat removal systems (RCIC, HPCI, RHR/LPCI, and core spray) were identified.

2.5 AC/DC Power Support Systems

The AC and DC power support systems were reviewed for contact control devices in circuits that could prevent the availability of DC and AC power sources. The following AC and DC power support systems were reviewed:

- Emergency Diesel Generators
- Battery Chargers and Inverters
- EDG Ancillary Systems
- Switchgear, Load Centers, and Motor Control Centers (MCCs)

Electrical power, especially DC, is necessary to support achieving and maintaining a stable plant condition following a seismic event. DC power relies on the availability of AC power to recharge the batteries. Given an assumed loss of offsite power, the availability of AC power is dependent upon the Emergency Diesel Generators and their ancillary support systems. EPRI 3002004396 [8] requires confirmation that the supply of emergency power is not challenged by a SILO device. The tripping of lockout devices or circuit breakers would require some level of diagnosis to mitigate the fault condition and restore emergency power. Annunciator Response Procedures were reviewed to ensure that the operator actions would be timely and successful.

To ensure contact chatter cannot compromise the emergency power system, the control circuits were analyzed for the Emergency Diesel Generators (EDG), Battery Chargers, Vital AC Inverters, and Switchgear/Load Centers/MCCs as necessary to distribute power from the EDGs to the Battery Chargers and EDG Ancillary Systems. Plant Hatch EDGs provide emergency power for the units. Each unit has two divisions of Class 1E loads with one EDG for each division. In addition there is a "swing" diesel generator that can support either unit. The SPRA has included these systems in the seismic equipment list [15.1], and has evaluated potential chatter of the associated contact control devices [15.8].

The high frequency analysis considers the reactor is operating at power with no equipment failures or LOCA prior to the seismic event. The Emergency Diesel Generators are not operating but are available. The seismic event is presumed to cause a Loss of Offsite Power (LOOP) and a normal reactor SCRAM.

In response to bus under-voltage relaying detecting the LOOP, the Class 1E control systems must automatically shed loads, start the EDGs, and sequentially load the diesel generators as designed. Ancillary systems required for EDG operation as well as Class 1E battery chargers and inverters must function as necessary. The goal of this analysis is to identify any vulnerable contact devices that could chatter during the seismic event, and prevent these systems from performing their intended safety-related function of supplying electrical power during the LOOP.

The following sections contain a description of the analysis for each element of the AC/DC Support Systems. Contact devices are identified by description in this narrative and apply to all divisions.

Emergency Diesel Generators

The analysis of the Emergency Diesel Generators is broken down into the generator protective relaying and diesel engine control. General descriptions of these systems and controls are provided in the UFSAR.

Generator Protective Relaying

The control circuits for the EDG circuit breakers include bus lockout, differential lockout, phase overcurrent protection, field, LOCA signal, and exciter relays. Chatter in any of these relays may prevent closure of the EDG circuit breaker. The generator differential lockout relay may be tripped by chatter in the differential relay. In addition, chatter in the emergency bus differential protection relays, the overcurrent relays, or the exciter relays could lead to the tripping of the bus lockout relay. When tripped, this bus lockout relay prevents closure of the EDG circuit breakers.

Diesel Engine Control

Chatter analysis for the diesel engine control was performed on the start and shutdown circuits of each EDG. The start circuit is blocked by seal-in of the engine trouble shutdown or start failure relays. Chatter of the seal-in contacts of these relays or of the contacts of relays within the coil circuits of these relays may prevent EDG start.

The start failure relay is controlled by time delay relays. The time delay function of these relays prevents momentary chatter in their coil circuits from energizing them.

The engine trouble shutdown relay is controlled by the engine overspeed switch and relay, the emergency stop relay, and the lube oil low pressure switch and relay. Chatter of the contacts of the overspeed relay is blocked by the overspeed switch contacts. The overspeed switch is not vulnerable to chatter. Chatter in the contacts of the emergency stop relay could lead to seal-in of the shutdown relay. Chatter in the emergency stop relay coil circuit is blocked from energizing the emergency stop relay by non-vulnerable control switches. Chatter of the relay contacts of the lube oil pressure circuit could lead to seal-in of the shutdown relay. The lube oil pressure switch is not vulnerable to chatter.

EDG Ancillary Systems

To start and operate the Emergency Diesel Generators, a number of components and systems are required. For the purpose of identifying electrical contact devices, only systems and components which are electrically controlled are analyzed.

Starting Air

Based on Diesel Generator availability as an initial condition, the passive air reservoirs are presumed pressurized and the only active components in this system required to operate are the air start solenoids, which are covered under the EDG engine control analysis above.

Combustion Air Intake and Exhaust

The combustion air intake and exhaust for the Diesel Generators are passive systems which do not rely on electrical control.

Lube Oil

The Diesel Generators utilize engine-driven mechanical lubrication oil pumps which do not rely on electrical control.

Fuel Oil

The Diesel Generator Fuel Oil System is described in the UFSAR. The Diesel Generators utilize engine-driven mechanical pumps and DC-powered auxiliary pumps to supply fuel oil to the engines from the day tanks. The day tanks are re-supplied using the AC-powered fuel oil transfer pumps. The engine-driven mechanical pumps do not rely on electrical control. The control circuits for the electrically-powered auxiliary and transfer pumps were included in the evaluation of contact control devices.

Cooling Water

The Diesel Generator Cooling Water System is described in the UFSAR. This system is cooled by plant service water (PSW). Engine-driven mechanical pumps, which do not rely on electrical control, are credited when the engine is operating. The electric jacket water pump is only used during EDG shutdown periods, and is thus not included in this analysis.

The main PSW pumps provide cooling water to the heat exchangers associated with four of the EDGs. The standby PSW pump provides cooling water to the swing EDG. In automatic mode, these pumps are loaded onto the emergency switchgear via the EDG start signal and load sequencer. Chatter analysis of the EDG start signal is included above. A chatter analysis of the PSW pump circuit breaker control circuits indicates the bus lockout and phase overcurrent relays could prevent automatic (sequential) breaker closure following the seismic event.

Ventilation

The Diesel Generator building ventilation system is described in the UFSAR. Ventilation is provided for the EDG rooms, the switchgear rooms, and the DG battery rooms. Chatter analysis of the control circuits for these fans concluded they do not include SILO devices.

Battery Chargers

The seismic event is presumed to cause a Loss of Offsite Power (LOOP), resulting in an undervoltage signal that initiates load-shed from the 4KV and 600V switchgear. The battery chargers are load-shed (that is, the battery charger breakers are tripped) on a loss of offsite power, which is the desired position for the LOOP scenario. The operators in the main control room can manually load them back onto the emergency switchgear. This operator response was determined to be a highly reliable operator action for the SPRA. Thus, since the battery chargers are desired to be tripped, chatter is not an issue for the presume loss of offsite power scenario.

Inverters

Analysis of schematics for the inverters did not identify contact control devices subject to chatter.

Switchgear, Load Centers, and MCCs

Power distribution from the EDGs to the necessary electrical loads (Battery Chargers, Inverters, Fuel Oil Pumps, and EDG Ventilation Fans) was traced to identify any contact control devices which could lead to a circuit breaker trip and interruption in power. This effort excluded the EDG circuit breakers and the PSW pump breakers which are covered above, as well as component-specific contactors and their control devices, which are covered in the analysis of each component above. The medium- and low-voltage power circuit breakers in switchgear and load centers supplying power to loads identified in this section were included in this evaluation. The molded-case circuit breakers used in the Motor Control Centers are seismically rugged. The DC power distribution is via non-vulnerable disconnect switches.

However, there were two sets of circuit breakers that could be impacted by contact control devices. The first set is the circuit breakers that distribute power from the 4KV emergency buses to the station service transformers, and then to the 600V emergency switchgear load centers. A chatter analysis of the control circuits for these circuit breakers indicates that the phase overcurrent and differential relays could actuate the bus lockout relays, and trip the circuit breakers following the seismic event.

The second set of circuit breakers distribute power from the 600V emergency switchgear to the essential transformers, and then to the essential cabinets. Chatter of the overcurrent relays could trip these breakers.

2.6 Summary of Selected Components

A list of the contact devices selected for evaluation, which envelops the 2.1 Seismic high frequency requirements, is provided in Appendix B of this report. As described above, the list of equipment and the identification of contact control devices that would be chatter unacceptable was based on the SPRA seismic equipment list [15.1] and relay chatter evaluation [15.8]. This SPRA relay chatter identification process was verified to encompass the EPRI high frequency requirements [8].

3 Seismic Evaluation

Upon development of the list of components contained in Appendix B, Table B-1, of this report, the capacity-vs-demand evaluations (as documented in reference [15.11]) were performed. The following sections provide details of the process.

3.1 Horizontal Seismic Demand

Per Reference [8], Sect. 4.3, the basis for calculating high-frequency seismic demand on the subject components in the horizontal direction is the Plant Hatch horizontal ground motion response spectrum (GMRS), which was generated as part of the Plant Hatch Seismic Hazard and Screening Report [4] submitted to the NRC on March 31, 2014 and accepted by the NRC on April 27, 2015 [12]. The same seismic hazard information was used in the development of the Hatch SPRA.

It is noted in Reference [8] that a Foundation Input Response Spectrum (FIRS) may be necessary to evaluate buildings whose foundations are supported at elevations different than the Control Point elevation. Plant Hatch is located on a deep soil site. As part of the Plant Hatch SPRA, UHRS were developed at specific horizons at the 10^{-4} and 10^{-5} hazard levels along with the corresponding strain-compatible soil properties for various buildings and elevations. The Mean Annual Frequency of Exceedance (MAFE) Uniform Hazard Response Spectra (UHRS 10^{-4} [4, 15.1 through 15.10]) were used in developing the input ground motions for the seismic soil-structure interaction (SSI) analyses of Plant Hatch Seismic Category I structures.

The ISRS were developed for the Plant Hatch SPRA project. The Plant Hatch SPRA successfully completed a peer review in October 2016. The peer review process and results are summarized in Appendix C of this report. Since Plant Hatch has ISRS at the locations of all high-frequency component enclosures, there is no need for using generic horizontal response structural amplification factors (AF_{SH}) per EPRI 3002004396 [8]; Plant Hatch specific amplification is included within the ISRS.

Finite element models (FEMs) were produced for the Plant Hatch Seismic Category I structures, and these FEMs were coupled with the site soil to capture soil-structure interaction (SSI) effects. It should be particularly noted, there was minimal high frequency response motion within the structures due to the SSI effects, with the input ground motion having predominant motion occurring at frequencies less than 10 Hz. Nodes closest to the enclosure mounting locations were determined to properly represent the ISRS (demand) that each enclosure would experience.

The ISRS represents the 5% damped envelope of the lower bound (LB), best estimate (BE), and upper bound (UB) ISRS. This envelope of the LB, BE, and UB responses effectively represents the 84-percentile confidence response spectra level which is required when utilizing the Conservative Deterministic Failure Margin (CDFM) approach.

The horizontal GMRS values, in addition to the 10^{-4} and 10^{-5} MAFE UHRS values, are provided in Table 3-1 of this report.

3.2 VERTICAL SEISMIC DEMAND

As described in Section 3.2 of Reference [8], the horizontal GMRS and V/H ratios are used to calculate the vertical GMRS (VGMRS) for calculating high-frequency seismic demand on the subject components in the vertical direction. For this evaluation, Plant Hatch is using seismic demand (ISRS) associated with the Plant Hatch SPRA; the input ground motion is MAFE 10^{-4} UHRS.

Instead of using the generic site soil profiles and subsequent vertical vs. horizontal acceleration (V/H) ratios provided in Section 3.2 of the Reference [8] guidance, specific V/H ratios from the Plant Hatch SPRA were used. A robust, Plant Hatch-site-specific analysis following procedure given in NUREG/CR-6728 [16] was developed for computing mean V/H ratios as part of the Plant Hatch SPRA [15.9, 15.10].

The vertical MAFE 10^{-4} and 10^{-5} UHRS are then calculated by multiplying the mean V/H ratio at each spectral frequency by the horizontal MAFE 10^{-4} and MAFE 10^{-5} UHRS accelerations at the corresponding spectral frequency.

The V/H ratios and vertical MAFE 10^{-4} and MAFE 10^{-5} UHRS values are provided in Table 3-1 of this report.

Figure 3-1 of this report provides a plot of the following for Plant Hatch:

- Ground Response Spectra at Plant Hatch Control Point (EL 129)
  - Horizontal UHRS (10^{-4} HUHRS)
  - Vertical UHRS (10^{-4} VUHRS)
  - GMRS (HGMRS)

Figure 3-2 of this report is a plot of the MAFE 10^{-4} UHRS V/H ratio.

Note: the UHRS 10^{-4} vertical ISRS include in-structure amplification; therefore, no generic vertical response structural amplification factors (AF_{SV}) from Reference 8 are necessary.

Table 3-1: Horizontal and Vertical Ground Motions Response Spectra

		UHRS (MAFE = 10^{-4}), EL 129				UHRS (MAFE = 10^{-5}), EL 129				
Freq. (Hz)		Horiz. (g) (Ref. [4], Table 2.4-1)	V/H (Ref. [15.10], Table 4)	Vert. (g) (Ref. [15.10], Table 5)		Horiz. (g) (Ref. [4], Table 2.4-1)	V/H (Ref. [15.10], Table 4)	Vert. (g) (Ref. [15.10], Table 6)		HGMRS (g) (Ref. [4], Table 2.4-1)
0.100		0.0121	0.543	0.0066		0.0288	0.608	0.0175		0.0145
0.125		0.0168	0.543407803	0.0091		0.0403	0.608292317	0.0245		0.0203
0.167		0.0280	0.543407803	0.0152		0.0691	0.608292317	0.0420		0.0346
0.200		0.0354	0.543407803	0.0192		0.0873	0.608292317	0.0531		0.0437
0.300		0.0476	0.543407803	0.0259		0.1154	0.608292317	0.0702		0.0580
0.400		0.0593	0.543407803	0.0322		0.1475	0.608292317	0.0896		0.0737
0.500		0.0865	0.543407803	0.0470		0.2245	0.608292317	0.1364		0.1113
0.600		0.1142	0.543407803	0.0620		0.2920	0.608292317	0.1776		0.1452
0.700		0.1365	0.543407803	0.0742		0.3390	0.608292317	0.2063		0.1696
0.800		0.1673	0.543407803	0.0909		0.3983	0.608292317	0.2425		0.2009
0.900		0.1847	0.543407803	0.1004		0.4280	0.608292317	0.2607		0.2171
1.000		0.1893	0.543407803	0.1029		0.4342	0.608292317	0.2644		0.2206
1.250		0.2360	0.543407803	0.1284		0.5177	0.608292317	0.3154		0.2654
1.500		0.2563	0.543407803	0.1394		0.5528	0.608292317	0.3367		0.2844
2.000		0.2782	0.543407803	0.1513		0.6173	0.608292317	0.3758		0.3158
2.500		0.2751	0.543407803	0.1497		0.6039	0.608292317	0.3678		0.3096
3.000		0.2695	0.543407803	0.1466		0.5906	0.608292317	0.3598		0.3029
4.000		0.2739	0.543407803	0.1489		0.6006	0.608292317	0.3657		0.3080
5.000		0.2740	0.543407803	0.1489		0.6098	0.608292317	0.3712		0.3118
6.000		0.2807	0.543407803	0.1525		0.6269	0.608292317	0.3815		0.3203
7.000		0.2766	0.543407803	0.1503		0.6196	0.608292317	0.3770		0.3164
8.000		0.2723	0.543407803	0.1480		0.6166	0.608292317	0.3752		0.3142
9.000		0.2678	0.543407803	0.1455		0.6115	0.608292317	0.3722		0.3111
10.000		0.2610	0.543	0.1418		0.5978	0.608	0.3639		0.3039
12.500		0.2351	0.551923315	0.1298		0.5401	0.625139727	0.3377		0.2744
15.000		0.2107	0.558979959	0.1178		0.4839	0.639250858	0.3095		0.2459
20.000		0.1745	0.579990689	0.1013		0.3985	0.678789515	0.2707		0.2027
25.000		0.1538	0.636	0.0978		0.3520	0.768	0.2706		0.1790
30.000		0.1443	0.680182291	0.0981		0.3271	0.866039991	0.2833		0.1666
35.000		0.1378	0.73653814	0.1015		0.3103	0.951659177	0.2954		0.1583
40.000		0.1336	0.79755261	0.1065		0.3002	1.05199686	0.3158		0.1532
45.000		0.1317	0.842491315	0.1103		0.2955	1.14945099	0.3378		0.1508
50.000		0.1291	0.881830401	0.1138		0.2894	1.206133855	0.3492		0.1478
60.000		0.1270	0.895541669	0.1135		0.2842	1.191515232	0.3385		0.1452
70.000		0.1257	0.880952257	0.1106		0.2815	1.158283015	0.3259		0.1438
80.000		0.1248	0.833226566	0.1039		0.2795	1.090275685	0.3049		0.1427
90.000		0.1243	0.769888643	0.0957		0.2784	1.004936266	0.2800		0.1422
100.000		0.1244	0.712	0.0884		0.2784	0.941	0.2620		0.1422

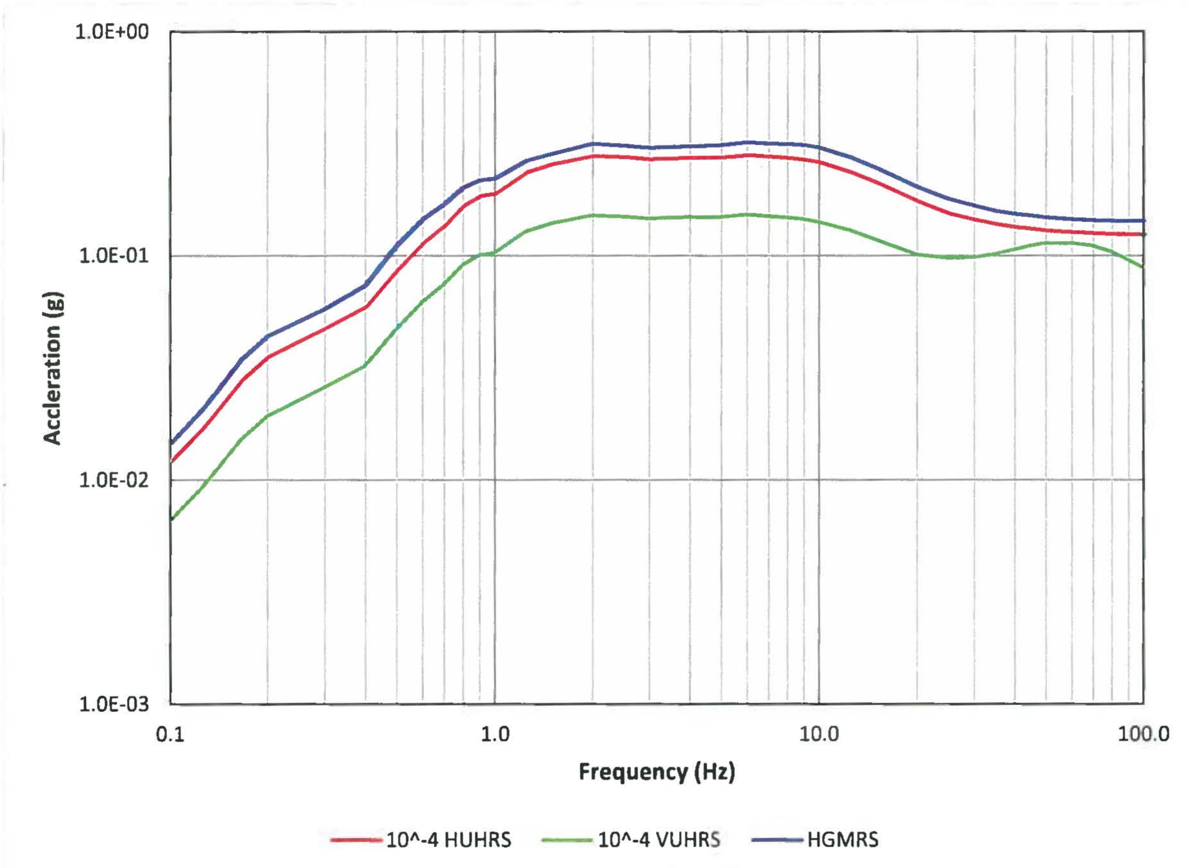


Figure 3-1: Plot of the Mean Annual Frequency of Exceedance (MAFE) 10^{-4} Horizontal Uniform Hazard Response Spectra (HUHRS), MAFE 10^{-4} Vertical Uniform Hazard Response Spectra (VUHRS) and Horizontal Ground Motion Response Spectrum (HGMRS)

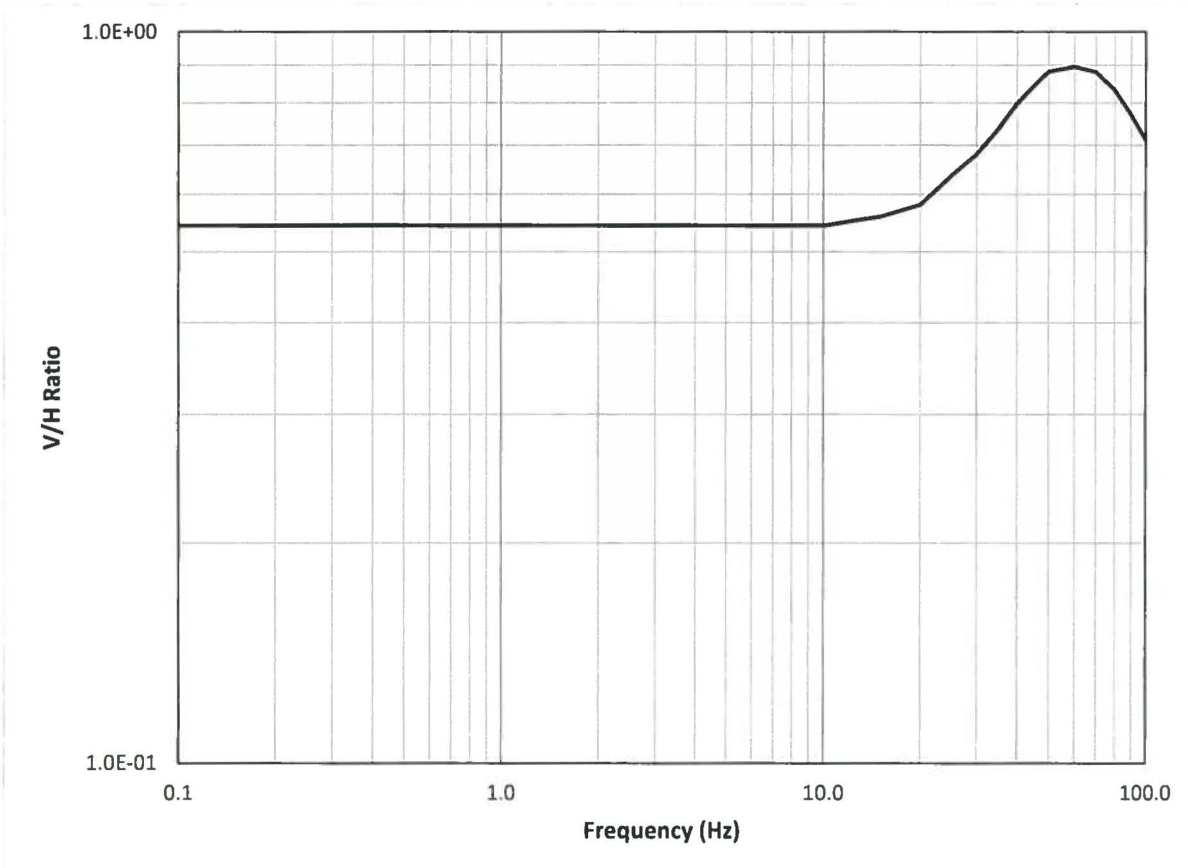


Figure 3-2: Plot of the Mean Annual Frequency of Exceedance (MAFE) 10^{-4} V/H Ratio vs. Frequency

3.3 Component Horizontal Seismic Demand

Using References [15.2] through [15.4] and [15.11] of this report, the demand used to determine the HCLPF capacity of each high-frequency component represents the 10^{-4} UHRS In-Structure Response Spectra (ISRS) multiplied by the Reference [8] in-cabinet amplification factor (AF_c).

The frequency range of interest for the high frequency confirmation is from 15 to 40 Hz. This is consistent with the frequency range used in Section 4.5.3 of EPRI 3002004396 [8]. The ISRS represents the response spectra at the location (floor level) of the enclosure in which the relay device is mounted. Therefore, to capture the demand at the relay device mounting point in the enclosure, an in-cabinet amplification factor (AF_c) must be applied. This in-cabinet amplification factor comes from EPRI 3002004396 [8] for the horizontal direction. All enclosures of the Plant Hatch high-frequency components meet one of the three categories listed in Reference [8]:

- low-amplification, such as Motor Control Centers (MCCs) (AF_C = 3.6);
- high-amplification, such as switchgear (AF_C = 7.2);
- medium-amplification, such as control panels and benchboards (AF_C = 4.5).

3.4 Component Vertical Seismic Demand

As done with the component horizontal seismic demand (see the previous section of this report), vertical ISRS were generated at nodes near to the enclosures for all subject high-frequency components. The peak accelerations of the VGMRS between 15 Hz and 40 Hz was determined.

As done with the component horizontal seismic demand, to capture the demand at the relay device mounting point in the enclosure, an in-cabinet amplification factor (AF_c) is applied. This in-cabinet amplification factor comes from EPRI 3002004396 [8] for the vertical direction, and is 4.7 for all cabinet types.

4 Contact Device Evaluations

Per Reference [8], seismic capacities (the highest seismic test level reached by the contact device without chatter or other malfunction) for each subject contact device are determined by the following procedures:

(1) If a contact device was tested as part of the EPRI High Frequency Testing program [7], then the component seismic capacity from this program is used.
(2) If a contact device was not tested as part of [7], then one or more of the following means to determine the component capacity was used:
    (a) Device-specific seismic test reports (either from the station or from the SQURTS testing program).
    (b) Generic Equipment Ruggedness Spectra (GERS) capacities per [9] and [10].
    (c) Assembly (e.g. electrical cabinet) tests where the component functional performance was monitored.

The high-frequency capacity of each device was evaluated with the component mounting point demand from Section 3 of this report using the criteria in Section 4.5 of Reference [8]. Below is more detailed explanation about the Plant Hatch process for determining capacity:

- A capacity-to-demand ratio for the horizontal and vertical direction was determined.
- The minimum capacity to demand ratio (either horizontal or vertical) was then multiplied times the UHRS 10^{-4} PGA to determine the relay HCLPF capacity in terms of PGA.
- This relay HCLPF capacity was then compared to the GMRS PGA. Dividing the relay HCLPF capacity (which is in terms of PGA) by the GMRS PGA produces this comparison.
- If that ratio is equal to or greater than "1" the relay is acceptable for the high frequency confirmation.

As stated previously, the 10^{-4} UHRS was used as the input motion for calculating the ISRS for the Plant Hatch SPRA. The UHRS 10^{-4} PGA is 0.1244g and the GMRS PGA is 0.1422g. The shapes of the curves are basically identical, as seem in Figure 3-1 of this report. Also, the amplitude of the GMRS to UHRS 10^{-4} is only about 14 to 16% higher; i.e., not a significant change in amplitude. Therefore, it is concluded the calculation of the relay fragility in terms of HCLPF capacity using UHRS 10^{-4} ISRS is applicable for comparison to the GMRS demand.

A summary of the high-frequency evaluation conclusions is provided in Table B-1 in Appendix B.

5 Conclusions

5.1 General Conclusions

Plant Hatch has performed a High Frequency Confirmation evaluation in response to the NRC's 50.54(f) letter [1] using the guidance of EPRI report 3002004396 [8] and no additional actions are necessary.

The selection process identified a total of 194 components for evaluation. As summarized in Table B-1 in Appendix B, 179 of the devices have adequate seismic capacity and 15 components did not have adequate seismic capacity following the criteria in Section 4.5 of Reference [8].

The 15 components that did not have adequate seismic capacity are all General Electric (GE) CFD model relays. These relays are known to have a low seismic capacity. Plant Hatch already has operator actions in place that can adequately resolve potential seismic concerns regarding these relays [Reference 15.11 Attachment M]. Therefore, per Section 4.6 of Reference [8], no additional actions are necessary.

5.2 Identification of Follow-up Actions

No follow-up actions are required as Plant Hatch already has procedural operator actions in place to address the low seismic capacity of the GE CFD model relays.

6 References

1. NRC (E. Leeds and M. Johnson) Letter to All Power Reactor Licensees et al., "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," March 12, 2012, ADAMS Accession Number ML12053A340.
2. NRC (W. Dean) Letter to the Power Reactor Licensees on the Enclosed List. "Final Determination of Licensee Seismic Probabilistic Risk Assessments Under the Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendation 2.1 "Seismic" of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident." October 27, 2015, ADAMS Accession Number ML15194A015.
3. NRC (J. Davis) Letter to Nuclear Energy Institute (A. Mauer). "Endorsement of Electric Power Research Institute Final Draft Report 3002004396, 'High Frequency Program: Application Guidance for Functional Confirmation and Fragility.'" September 17, 2015, ADAMS Accession Number ML15218A569.
4. Seismic Hazard and Screening Report in Response to the 50.54(f) Information Request Regarding Fukushima Near-Term Task Force Recommendation 2.1: Seismic for Plant Hatch dated March 31, 2014, NL-14-0343, ADAMS Accession Number ML14092A017.
5. EPRI 1015109. "Program on Technology Innovation: Seismic Screening of Components Sensitive to High-Frequency Vibratory Motions." October 2007.
6. EPRI 1025287. "Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic." February 2013.
7. EPRI 3002002997. "High Frequency Program: High Frequency Testing Summary." September 2014.
8. EPRI 3002004396. "High Frequency Program: Application Guidance for Functional Confirmation and Fragility Evaluation." July 2015.
9. EPRI NP-7147-SL. "Seismic Ruggedness of Relays." August 1991.
10. EPRI NP-7147 SQUG Advisory 2004-02. "Relay GERS Corrections." September 10, 2004.
11. Procedure for Evaluating Nuclear Power Plant Relay Seismic Functionality EPRI, Palo Alto, CA:1990. NP-7148.
12. NRC (Govan) Letter to Hatch (Pierce). "Edwin I. Hatch Nuclear Plant, Units 1 and 2 – Staff Assessment of Information Provided Pursuant to Title 10 of the Code of Federal Regulations Part 50, Section 50.54(f), Seismic Hazard Reevaluations Relating to Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident (TAC Nos. MF3772 AND MF3773)." April 27, 2015, ADAMS Accession Number ML15097A424.
13. ASME/ANS RA-Sb-2013, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Application," American Society of Mechanical Engineers / American Nuclear Society, ASME/ANS RA-Sb-2013.

14. USNRC, Regulatory Guideline 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2.

15. Station Documents

15.1. H-RIE-SEIS-U00-002-001, Version 3.0. "Seismic Equipment List and SEL Walkdown Report – Seismic PRA", June 30, 2017.

15.2. H-RIE-SEIS-U00-005-007, Version 3.0. "Seismic SSI Analysis – Diesel Generator Building – Hatch Seismic PRA", June 28, 2017.

15.3. H-RIE-SEIS-U00-005-009, Version 3.0. "Seismic SSI Analysis – Reactor Building – Hatch Seismic PRA", June 29, 2017.

15.4. H-RIE-SEIS-U00-005-010 Version 2.0. "Seismic SSI Analysis – Control Building – Hatch Seismic PRA" September 1, 2016.

15.5. H-RIE-SEIS-U00-006-001 Version 3.0. "Fragility Notebook – Hatch SPRA Units 1 and 2", June 30, 2017.

15.6. H-RIE-SEIS-U00-006-002 Version 4.0. "Component Fragility – Control Building – SPRA Hatch Units 1 and 2", May 31, 2017.

15.7. H-RIE-SEIS-U00-006-012 Version 3.0 "Review of Soil Hazards – Special Considerations – Hatch Seismic PRA", June 29, 2017.

15.8. H-RIE-SEIS-U00-007-001, Version 3.0. "Relay Chatter Evaluation Report", June 30, 2017.

15.9. SCNH-13-093 Version 1 "Development of Surface Hazard and Uniform Hazard Response Spectra for Hatch Site", March 19, 2014.

15.10. SCNH-15-057, Version 1. "Development of Horizon-Specific Spectra for MAFE 10^{-4} and 10^{-5} Hazard Levels Using Approach 3", May 18, 2016.

15.11. SCNH-17-007, Version 2.0. "2.1 Seismic High Frequency Confirmation Calculations." August 1, 2017.

16. USNRC, NUREG/CR-6728, "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-consistent Ground Motion Spectra Guidelines", October 2001.

17. NEI-12-13, External Hazards PRA Peer Review Process Guidelines, Revision 0, Nuclear Energy Institute, Washington, DC, August 2012.

A Representative Sample Component Evaluations

This section contains two representative sample component evaluations. These two samples are Attachments to SNC calculation SCNH-17-007, "2.1 Seismic High Frequency Confirmation Calculations" [15.11]. To ensure these samples (Attachments to SCNH-17-007) are easily understood, and to meet the requirements of Section 4.7 of Ref. [8], the main body of the calculation is also provided. This information provides details about the following:

- Purpose
- Design Inputs
- Acceptance Criteria
- Methodology
- Determination of High Frequency seismic demand
- Determination of High Frequency seismic capacity
- Capacity to Demand Check
- Summary of Conclusions

Southern Nuclear Design Calculation

Calculation Number:
SCNH-17-007

Plant: E. I. Hatch Nuclear Plant	**Unit:** (check all that apply) ☒ 1 ☒ 2 ☐ 3 ☐ 4	**Discipline:** Civil/Seismic
Title: 2.1 Seismic High Frequency Confirmation Calculations		**Subject:** SAM Fukushima 2.1 Seismic High Frequency Evaluations
Purpose / Objective: High Frequency Confirmation Calculations in Response to Near Term Task Force (NTTF) 2.1 Seismic Recommendation		
System or Equipment Tag Numbers: Multiple (refer to Attachment B, Table B-1)		

Contents

Nuclear Quality Level

☒ **Safety-Related** ☐ **Safety Significant** ☐ **Non- Safety –Significant**

Version Record

Version No.	Description	Co-Originator Printed Name Initial / Date	Reviewer Printed Name Initial / Date	Approval 1 Printed Name Initial / Date	Approval 2 Printed Name Initial / Date
1.0	Issued	Colter D. Somerville (1) 7/19/2017 Donald P. Moore (2) 7/19/2017 Scott H. Pellet (3) 7/19/2017	Melanie H. Brown 7/19/2017	Kent Johnson 7/19/2017	Kent Johnson 7/19/2017

SCNH-17-007 Title Page Continued

Version No.	Description	Co-Originator Printed Name Initial / Date	Reviewer Printed Name Initial / Date	Approval 1 Printed Name Initial / Date	Approval 2 Printed Name Initial / Date
2.0	See Revision History Log	Colter D. Somerville [(1)] 8/1/17 Donald P. Moore [(2)] 8/1/17 Scott H. Pellet [(3)] 8/1/17	Melanie H. Brown 8-1-17	Kent Johnson 08/01/2017	Kent Johnson 08/01/2017

Note(s): [(1)] Colter D. Somerville co-originated this calculation under the direction of a qualified mentor.
[(2)] Donald P. Moore assisted in the high frequency methodology portions of this calculation under the direction of a qualified mentor.
[(3)] Scott H. Pellet co-originated this calculation as a mentor who is fully qualified to perform calculations.

Southern Nuclear Design Calculations

Plant:	Calculation Number:	Sheet:
Hatch	SCNH-17-007	1

Revision History Log	
Version No.	**Description**
1.0	First issued copy
2.0	1) GE relay model number CR120 was changed to CR120A. This was done in Table B-1 (Row No.'s 7, 69, 71) and throughout Attachment K for clarity and completeness. 2) GE relay model number IAC 54 was changed to IAC 54A per Design Input #16, Page 5 of 16, test group 11. This was done in Table B-1 (Row No.'s 23-25, 30-32, 58, 84-91, 96-103, 123). 3) The capacity for A.O. Smith / Clark Control Div. changed from 5.0g to 4.1g per Design Input #16, page 4 of 16, item number 5. This was done in Table A-1 (Row No.'s 155, 159, 167, 172, 177, 185 – since these rows contained formulas in Excel, all the data in each of the rows was replaced) and throughout Attachment E. 4) Added Design Input #16. This was added under Section 3, Design Inputs. 5) Added Revision History Log on Sheet 1 6) Revised title page and added a second title page 7) Design Input #6 is no longer being used for this calculation and has been striked-through on sheet 2. 8) Design Inputs #8 and #9 went from version to 2 to version 3 on sheet 2

Southern Nuclear Design Calculations

Plant: Hatch	Calculation Number: SCNH-17-007	Sheet: 2

1) Purpose of Calculation:

This calculation was prepared in response to Near-Term Task Force (NTTF) 2.1 Seismic - High Frequency for Plant Hatch Units1 and 2. The requirement for plant Hatch to perform this evaluation is documented in ML#15194A015, NRC Final Determination letter, October 27th 2015.

This calculation will determine the capacity vs demand ratio for the Hatch relay devices listed in Attachment B which were identified by following the Near-Term Task Force (NTTF) 2.1 Seismic "High Frequency Program - Application Guidance for Functional Confirmation and Fragility Evaluation" document (EPRI 3002004396). This calculation focuses on the high frequency range (15Hz – 40Hz) as required by EPRI 3002004396, section 4.

2) Summary of Conclusions:

All of the applicable relay devices identified in Attachment B, Table B-1, are shown to be acceptable. This was done by having a HCLPF capacity / GMRS demand (capacity / demand) ratio greater than or equal to 1, or by having a reliable operator action. The capacity / demand ratios can be seen in Attachment A of this calculation. The results of this calculation will be part of the NTTF 2.1 Seismic High Frequency Submittal report to the Nuclear Regulatory Committee (NRC).

3) Design Inputs:

1. ML#15194A015, "Final Determination of Licensee Seismic Probabilistic Risk Assessments Under the Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendation 2.1 "Seismic" of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident"
2. High Frequency Template, Developed by SSOT, Rev. 2, March 15th, 2017
3. EPRI 3002004396, "High Frequency Program – Application Guidance for Functional Confirmation and Fragility Evaluation", 2015 Technical Report
4. TR-103959, "Methodology for Developing Seismic Fragilities", Technical Report 1994
5. NL-14-0343, "Seismic Hazard and Screening Report for CEUS Sites"
6. ~~SCNH-12-076, "Site Response Analysis for the Hatch Site", Version 1~~ (not used)
7. SCNH-15-057, "Development of Horizon-Specific Spectra for MAFE 10^{-4} and 10^{-5} Hazard Levels Using Approach 3", Version 1
8. H-RIE-SEIS-U00-005-007, "Seismic SSI Analysis – Diesel Generator Building – Hatch Seismic PRA", Version 3
9. H-RIE-SEIS-U00-005-009, "Seismic SSI Analysis – Reactor Building – Hatch Seismic PRA", Version 3
10. H-RIE-SEIS-U00-005-010, "Seismic SSI Analysis – Control Building – Hatch Seismic PRA", Version 2
11. EPRI NP-5223-SLR1, "Generic Seismic Ruggedness of Power Plant Equipment", Revision 1, August 1991
12. EPRI NP-7147-SL, "Seismic Ruggedness of Relays", August 1991
13. EPRI TR-105988-V2, "GERS Formulated Using Data from the SQURTS Program", April 1999
14. EPRI NP-6041-SLR1, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin", Revision 1, August 1991
15. S62067, "Seismic Qualification for Potter and Brumfield Relay", Appendix F

16. ML042610031 EPRI NP-7147 SQUG Advisory 2004-02, "Relay GERS Corrections", September 10, 2004

4) Acronyms

GMRS	Ground Motion Response Spectra
UHRS	Uniform Hazard Response Spectra
PGA	Peak Ground Acceleration
ISRS	In-Structure Response Spectra
CDFM	Conservative Deterministic Failure Margin
SPRA	Seismic Probabilistic Risk Assessment
HCLPF	High Confidence Low Probability of Failure
CEUS	Central and Eastern United States
FEM	Finite Element Model
SSI	Soil-Structure Interaction
GERS	Generic Equipment Ruggedness Spectra
TRS	Test Response Spectra
SQURTS	Seismic Qualification and Reporting Testing Standardization
EPRI	Electric Power Research Institute
SNC	Southern Nuclear Company

5) Acceptance Criteria:

For the relays identified in Attachment B (which will become Table B-1 in the NTTF 2.1 Seismic High Frequency submittal to the NRC), it is desired that the calculated HCLPF capacity be greater than or equal to the calculated GMRS demand in terms of PGA (capacity / demand ratio greater than or equal to 1). If a relay device's established seismic capacity is not shown to be greater than or equal to the expected seismic demand, one of the following resolutions must be taken:

- Perform more detailed analysis to improve the capacity/demand ratio
- Perform additional seismic testing to improve the seismic capacity
- Implement plant modifications
- Implement/Utilize operator actions

6) Methodology:

This calculation follows the EPRI High Frequency Methodology shown in EPRI 3002004396, however, there are two exceptions. The first exception is explained in section 9a and involves using the ISRS developed for the Hatch SPRA (for possible 50.69 applications). The second exception is explained in section 9c and involves developing HCPLF capacities based on the 1E-04 UHRS ISRS for comparison to the GMRS in terms of PGA.

7) Assumptions:

If any assumptions were used, they are clearly documented in the calculation examples in Attachments C through L.

8) References:

None for this calculation

9) Body of Calculation:

High frequency confirmation evaluations depend on two (2) primary variables, a capacity variable and a demand variable. Each variable is described in more detail in the proceeding sections.

a. <u>High Frequency Confirmation – Demand:</u>

The demand used to determine the HCLPF capacity represents the 1E-04 UHRS ISRS multiplied by the in-cabinet amplification factor (AFc). The ISRS represents the 5% damped envelope of the lower bound (LB), best estimate (BE), and upper bound (UB) ISRS. This envelope of the LB, BE, and UB responses effectively represents the 84-percentile confidence response spectra level which is required when utilizing the CDFM approach.

The frequency range of interest for the high frequency confirmation is from 15 to 40 Hz. This is consistent with the frequency range used in section 4.5.3 "Example HF Confirmation Evaluation" of EPRI 3002004396.

The ISRS (design inputs 8, 9, 10) represents the response spectra at the location (floor level) of the cabinet in which the relay device is mounted. Therefore, in order to capture the demand at the relay device mounting point in the cabinet, an in-cabinet amplification factor (AFc) must be applied. This in-cabinet amplification factor comes directly from EPRI 3002004396 for the horizontal and vertical directions.

The ISRS were developed during the Hatch SPRA project for possible 50.69 applications by SNC Risk Informed Engineering department. The Hatch SPRA successfully completed a peer review in October 2016, meeting Capability Category II at a minimum for all high-level requirements (HLRs). Since Hatch has ISRS there is no need for considering building amplification; it is already captured in the ISRS. Finite element models (FEMs) were produced for the Hatch seismic category I structures, and these FEMs were coupled with the site soil in order to capture SSI effects. There was minimal high frequency motion within the structures due to the SSI effects and the input ground motion having predominate motion occurring less than 10 Hz. Nodes closest to the equipment mounting locations were determined to properly represent the ISRS (demand) that each equipment would experience. Each node number that corresponds to a specific ISRS is documented in the Mathcad files (Attachments C-L of this calculation).

The input motion for the ISRS was the 1E-04 UHRS. Understanding the input motion used to generate the ISRS is very important and the reasons why are discussed in the "Capacity to Demand Check", Section 7c of this calculation.

The final demand at device mounting location is defined as follows:

$$Demand_{Final} = \text{ISRS} * AF_C$$

b. High Frequency Confirmation – Capacity:

To meet the requirements of NTTF 2.1 Seismic High Frequency, the high frequency component capacities can be based on one of the following:

- *Test Response Spectra (TRS) Capacity:*

  These capacities represent site-specific tests, such as a SQURTS test or any other seismic test that determines the specific seismic capacity of the relay device (or table limits).

- *Generic Equipment Ruggedness Spectra (GERS) Capacity:*

  These capacities are based on survival statistics and certain criteria / caveats must be confirmed in order to use these capacities. GERS generally represents low frequency capacities; however, these capacities can be extended out into the high frequency region (up to 40 Hz) and this direct extension is considered conservative.

- *EPRI High Frequency Testing Capacity – EPRI 3002002997:*

  These capacities were developed specifically for high frequency evaluations. Note: SNC did not use this option (high frequency testing capacities) in this evaluation, though this option is provided in the calculation for completeness.

There are two factors that are considered when determining the final capacity, which are:

- *Knockdown Factor (F_K):*

  For applicable CDFM knockdown factors, refer to table 4-2 in EPRI 3002004396.

- *Single Axis Correction Factor (F_{MS}):*

  The component capacity can be increased by a factor of 1.2 if the component test represents a multi-axis test and the as-built in-cabinet demand is predominantly single axis. For instance, if the floor motions are dominantly low frequency and the cabinet is reasonably stiff in the side-to-side direction, then most of the in-cabinet response would be front-to-back and the in-cabinet side-to-side and vertical motions would be modest. However, if an argument cannot be made for two of the three direction in-cabinet responses being at different frequencies and different amplitudes, then this factor should be 1.

The final capacity will represent the following:

$$Capacity_{Final} = \left(\frac{\text{'TRS' or 'GERS' or 'HF Data'}}{F_K}\right) * F_{MS}$$

Southern Nuclear Design Calculations

Plant:	Calculation Number:	Sheet:
Hatch	SCNH-17-007	6

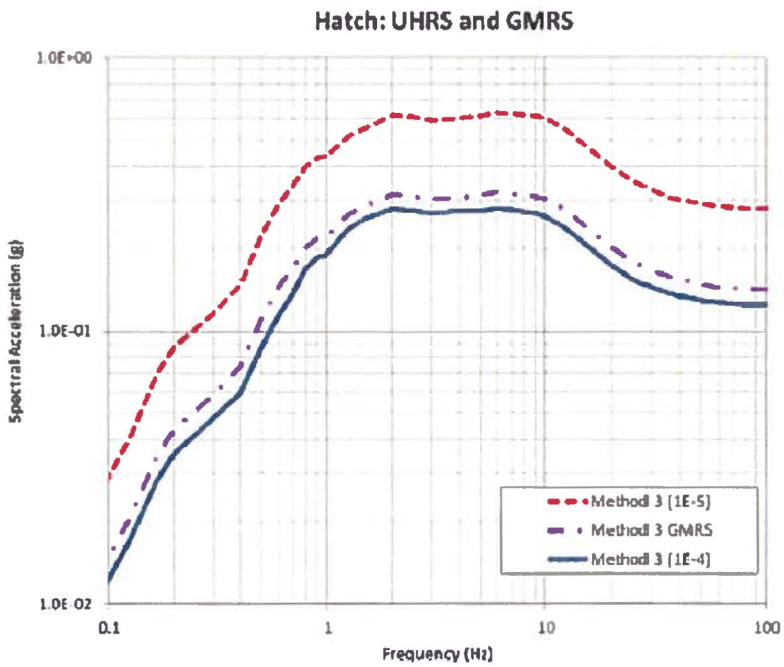
c. Capacity to Demand Check:

Once the final capacities and demands for each relay device are determined, the ratios (capacity / demand) are calculated for the horizontal and vertical directions.

The minimum ratio is then multiplied by the 1E-04 UHRS PGA to obtain the HCLPF capacity of the relay in terms of PGA. The final step is to compare the relay HCLPF capacity in terms of PGA to the demand defined as the GMRS PGA. This is done by calculating the ratio of the capacity (HCLPFpga) to demand (GMRS PGA). This ratio must be greater than or equal to 1 to successfully pass the capacity to demand check. If the ratio is not greater than or equal to one, then one of the steps listed in Section 5 must be taken.

As stated previously, the 1E-04 UHRS was used as the input motion for the ISRS. The 1E-04 UHRS PGA is 0.1244g and the GMRS PGA is 0.1422g. The shapes of the curves are basically identical, as shown below in Figure 1 (Design Input 5 NL-14-0343):

Figure 1: Hatch UHRS and GMRS Curves



Also, the amplitude of the GMRS to 1E-04 UHRS is only about 14-16% higher, i.e., not a significant change in amplitude. Therefore, it is concluded the relay fragility in terms of HCLPF capacity is applicable for the GMRS ground motion.

d. Attachments A and B:
   - Attachment A: High Frequency Confirmation Evaluations – This attachment contains table A-1 which provides the evaluations for the components identified in Attachment B, table B-1. Attachment A is the actual capacity to demand check.
   - Attachment B: Components Identified for High Frequency Confirmation – This attachment contains table B-1 which lists the components required to have a high frequency evaluation. Attachment B will be part of the NTTF 2.1 Seismic High Frequency submittal.

   Understanding the relationship between attachments A and B is very important. Attachment A directly relates to attachment B. Both attachments begin with a column labeled "No." (column furthest to the left). These numerical numbers (1, 2, 3, etc.) under column "No." for both attachments directly relate to each other. For example, the information for item No. 1 in attachment A (table A-1) corresponds to item No. 1 in attachment B (table B-1). In essence, attachment A contains the numerical data (calculation) for the information listed in attachment B, and they are linked by the "No." column.

e. Attachments C through L:
   Attachments C – L contain ten detailed high frequency evaluation examples. These examples came directly from table B-1 in attachment B. Each attachment provides the table B-1 "No." number for that evaluation. For example, attachment C relates to table B-1 No. 1, which is for the Allis Chalmers Size 2 located in the 1R24S011 cabinet.

   These attachments were included to help provide a better understanding of how these high frequency confirmation evaluations were performed. These attachments simply provide more details. It is important to understand that the evaluations and results listed in table A-1 in attachment A are identical to the evaluations and results listed in attachments C through L. Two out of the ten attachments will be included in "A Representative Sample Component Evaluation" of the Hatch 2.1 Seismic High Frequency Confirmation Submittal to the NRC.

f. Attachment M
   This attachment contains the operator procedures that define the operator actions needed for the GE CFD relays. The GE CFD relays listed in Table B-1 in attachment B did not pass the capacity vs. demand check (table A-1, column titled "HCLPF to GMRS Ratio Check"). However, these relays have existing operator actions, which are listed below.
   - 34AB-R22-002-1 "Loss of 4160V Emergency Bus"
     - Pages M1 – M68
   - 31GO-OPS-021-0 "Manipulation of Controls and Equipment"
     - Pages M69 – M100

   The acceptance criteria (section 5 in this calculation) for this evaluation permits the use of operator actions when the capacity to demand check does not pass. Therefore, these relays, according to the acceptance criteria, are considered acceptable. Note: These operator actions were already in place prior to this evaluation for 2.1 Seismic High Frequency.

Southern Nuclear Design Calculations

Plant:	Calculation Number:	Sheet:
Hatch	SCNH-17-007	8

g. <u>Software</u>
Section 4.7 in procedure NMP-ES-039-001 "Calculations – Preparation and Revision" discusses the use of computer software in calculations. For this calculation, Mathcad 15.0 and Microsoft Excel 2010 were used. According to the procedure, Mathcad and Excel spreadsheets are categorized as a 'utility' software. The reviewer must follow the requirements set forth in NMP-ES-039-001 Section 4.7.2.

SCNH-17-007 Attachment J
Detailed High Frequency Confirmation Calculation Example 8
Table B-1 (Attachment B) No. 163 Potter & Brumfield KUEP-7D15-110 located in 1R22S006

High Frequency Confirmation - Potter & Brumfield KUEP-7D15-110 in 1R22S006

Relay Location 'Tag Number':	1R22S006 - Medium Voltage Switchgear
Relay Manufacturer:	Potter & Brumfield
Relay Model Type:	KUEP-7D15-110 (control relay)
Building(s):	Diesel Building (DB)
FEM Node:	DB-4439
Table B-1 No.:	163

High Frequency Evaluation (15Hz - 40Hz):

Seismic Demand:
The seismic demand for this evaluation will represent the following:

- In-structure response spectra (ISRS) is based on the 2014 seismic hazard (1E-04 uniform hazard response spectra (UHRS) ground motion) *[Design Input #5]*
- These ISRS were developed for the Hatch seismic probabilistic risk assessment (SPRA) for possible 50.69 applications by the SNC Risk Informed Engineering group.
- Represents ISRS at the floor at location of panel where the relay is mounted times any cabinet amplification
- 5% damping
- 84% confidence level - CDFM approach
  Represents the envelope of lower-bound, best-estimate, and upper-bound ISRS
- Maximum spectral acceleration between 15Hz - 40Hz
- ISRS clipping factor conservatively assumed to be 1

$ISRS_{Hz} := .199g$ Horizontal spectral acceleration *[Design Input #8]*

$ISRS_{V} := .138g$ Vertical spectral acceleration *[Design Input #8]*

Cabinet Amplification Factor:

Horizontal: *[Design Input #3, section 4.4]*

Motor Control Centers	AF_c = 3.6, horizontal motion only	(4-2a)
Switchgear (flexible panels)	AF_c = 7.2, horizontal motion only	(4-2b)
Control Cabinets (e.g. Control Room electrical panels and benchboards)	AF_c = 4.5, horizontal motion only	(4-2c)

$AFc_H := 7.2$ Horizontal cabinet amplification factor (CDFM) - Flexible Switchgear

SCNH-17-007 Attachment J
Detailed High Frequency Confirmation Calculation Example 8
Table B-1 (Attachment B) No. 163 Potter & Brumfield KUEP-7D15-110 located in 1R22S006

Vertical: *[Design Input #3, section C.4.2]*

Since $AF_{c, CDFM} = AF_{c, HCLPF}$, the clipped level value of broad frequency input spectrum device capacity factor for determining the vertical demand for components mounted in all cabinets is reasonably estimated as $AF_{c, CDFM} = 4.7$. This resulting vertical AF_c for use in CDFM evaluations was determined in the same manner as was done for the horizontal amplification factor for Motor Control Centers and Switchgear, and thus consistently reflects the same conservatism.

$AFc_V := 4.7$ Vertical cabinet amplification factor (CDFM)

Knockdown Factor: *[Design Input #3, table 4-2]*

Table 4-2
CDFM Knockdown Factor for Test Capacity Values

Test Source		F_k
Relay GERS [14] (SQURTS Test [15])	Lowest level without chatter	1.5
	No chatter, Test table capacity	1.2
High Frequency Test Program [12]	Fragility threshold $SA_T = (SA^* + 0.625g)$	1.56
	Function Confirmed, Test table capacity	1.11
Qualification Test (IEEE 344 [25])	No chatter	1.2

Note: References listed in Table 4-2 are provided in EPRI 3002004396, which is Design Input #3 of this calculation

$F_k := 1.2$ CDFM Knockdown Factor (SQURTS, no chatter, test table capacity)

Multi-axis Correction Factor: *[Design Input #4, page 3-68 & Design Input #14, page Q9]*

$F_{MS} := 1.2$ Multi-axis to single-axis correction factor applies due to predominantly single axis excitation

SCNH-17-007 Attachment J
Detailed High Frequency Confirmation Calculation Example 8
Table B-1 (Attachment B) No. 163 Potter & Brumfield KUEP-7D15-110 located in 1R22S006

Capacity vs. Demand:

Capacity:

$SQURTS := 14.62g$ SQURTS qualification level *[Design Input #15, page 10]*

$Capacity := \left(\frac{SQURTS}{F_k}\right) \cdot F_{MS} = 14.62 \cdot g$ HCLPF broad-band component capacity *[Design Input #3, section 4.5.2 Eq. 4-5]*

Demand:

$Demand_H := AFc_H \cdot ISRS_{Hz} = 1.43 \cdot g$ Horizontal demand

$Demand_V := AFc_V \cdot ISRS_V = 0.65 \cdot g$ Vertical demand

Capacity vs. Demand Ratio:

$Ratio_H := \frac{Capacity}{Demand_H} = 10.2$ Horizontal

$Ratio_V := \frac{Capacity}{Demand_V} = 22.54$ Vertical

Peak Ground Accelerations for Hatch: *[Design Input #5, Table 2.4.1]*

$UHRS_{pga} := .1244g$ 1E-04 uniform hazard response spectra (UHRS) peak ground acceleration (PGA). Input motion for ISRS

$GMRS_{pga} := .1422g$ Ground motion response spectra (GMRS) peak ground acceleration (PGA).

SCNH-17-007 Attachment J
Detailed High Frequency Confirmation Calculation Example 8
Table B-1 (Attachment B) No. 163 Potter & Brumfield KUEP-7D15-110 located in 1R22S006

HCLPF to GMRS Ratio/Check:

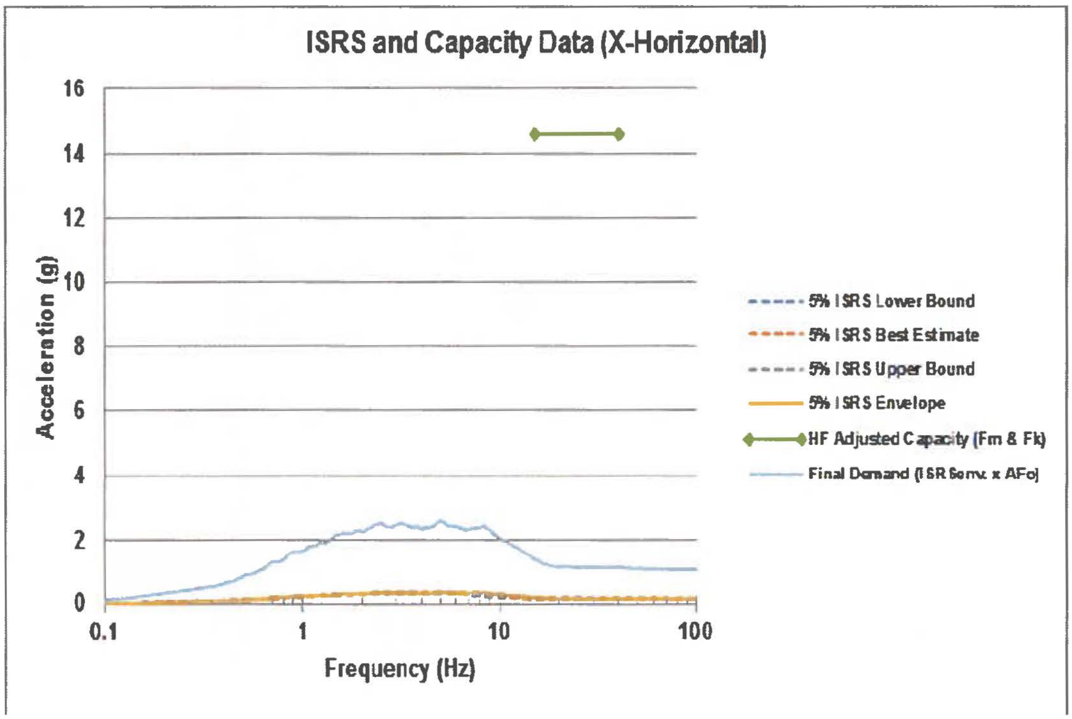
$$HCLPF_{pga} := \min\left(Ratio_H, Ratio_V\right) \cdot UHRS_{pga} = 1.27 \cdot g$$

$$HCLPF\_to\_GMRS\_ratio := \frac{HCLPF_{pga}}{GMRS_{pga}} = 8.93$$

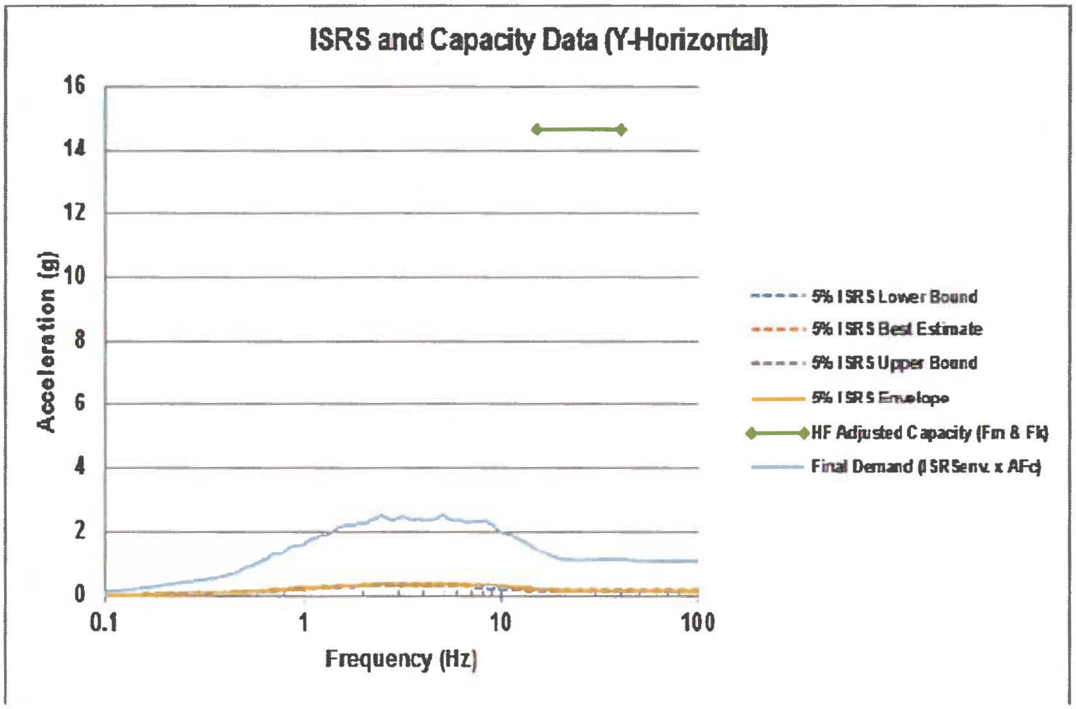
$$HCLPF\_to\_GMRS\_check := \begin{cases} \text{"OK"} & \text{if } HCLPF\_to\_GMRS\_ratio \geq 1 \\ \text{"Not OK"} & \text{otherwise} \end{cases}$$

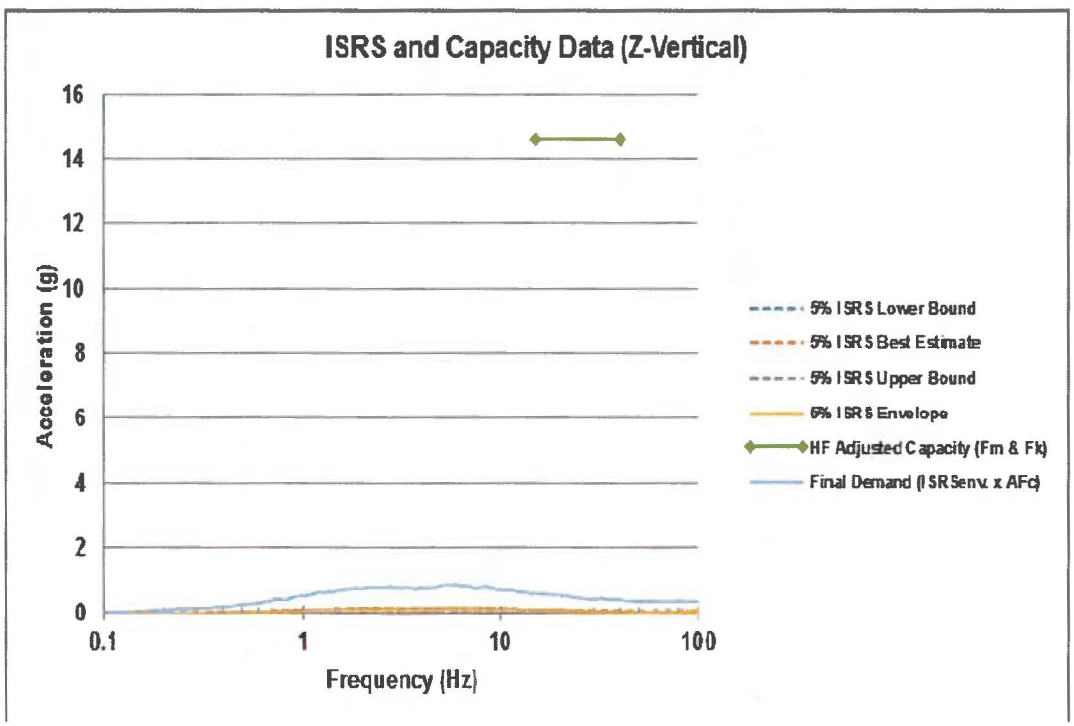
HCLPF_to_GMRS_check = "OK"

ISRS vs. Capacity Curves: *[Design Input #8]*
The following plots show the margin between the seismic demand based on 1E-04 UHRS ISRS and the relay capacity based on GERS or SQURTS



SCNH-17-007 Attachment J
Detailed High Frequency Confirmation Calculation Example 8
Table B-1 (Attachment B) No. 163 Potter & Brumfield KUEP-7D15-110 located in 1R22S006





SCNH-17-007 Attachment K
Detailed High Frequency Confirmation Calculation Example 9
Table B-1 (Attachment B) No. 7 General Electric CR120A located in 1H11P622

High Frequency Confirmation - General Electric CR120A in 1H11P622

Relay Location 'Tag Number':	1H11P622 - Control / Instrumentation Panel
Relay Manufacturer:	General Electric (GE)
Relay Model Type:	CR120A (auxiliary relay)
Building(s):	Control Building (CB)
FEM Node:	CB-11756
Table B-1 No.:	7

High Frequency Evaluation (15Hz - 40Hz):

Seismic Demand:

The seismic demand for this evaluation will represent the following:

- In-structure response spectra (ISRS) is based on the 2014 seismic hazard (1E-04 uniform hazard response spectra (UHRS) ground motion) *[Design Input #5]*
- These ISRS were developed for the Hatch seismic probabilistic risk assessment (SPRA) for possible 50.69 applications by the SNC Risk Informed Engineering group.
- Represents ISRS at the floor at location of panel where the relay is mounted times any cabinet amplification
- 5% damping
- 84% confidence level - CDFM approach
  Represents the envelope of lower-bound, best-estimate, and upper-bound ISRS
- Maximum spectral acceleration between 15Hz - 40Hz
- ISRS clipping factor conservatively assumed to be 1

$ISRS_{Hz} := .296g$ Horizontal spectral acceleration *[Design Input #10]*

$ISRS_{V} := .297g$ Vertical spectral acceleration *[Design Input #10]*

Cabinet Amplification Factor:

Horizontal: *[Design Input #3, section 4.4]*

Motor Control Centers	AF_c = 3.6, horizontal motion only	(4-2a)
Switchgear (flexible panels)	AF_c = 7.2, horizontal motion only	(4-2b)
Control Cabinets (e.g. Control Room electrical panels and benchboards)	AF_c = 4.5, horizontal motion only	(4-2c)

$AFc_{H} := 4.5$ Horizontal cabinet amplification factor (CDFM) - Control & Instrumentation Panel

SCNH-17-007 Attachment K
Detailed High Frequency Confirmation Calculation Example 9
Table B-1 (Attachment B) No. 7 General Electric CR120A located in 1H11P622

Vertical: *[Design Input #3, section C.4.2]*

Since $AF_{c,CDFM} = AF_{c,HCLPF}$, the clipped level value of broad frequency input spectrum device capacity factor for determining the vertical demand for components mounted in all cabinets is reasonably estimated as $AF_{c,CDFM} = 4.7$. This resulting vertical AF_c for use in CDFM evaluations was determined in the same manner as was done for the horizontal amplification factor for Motor Control Centers and Switchgear, and thus consistently reflects the same conservatism.

$AF_{cV} := 4.7$ Vertical cabinet amplification factor (CDFM)

Knockdown Factor: *[Design Input #3, table 4-2]*

Table 4-2
CDFM Knockdown Factor for Test Capacity Values

Test Source		F_k
Relay GERS [14] (SQURTS Test [15])	Lowest level without chatter	1.5
	No chatter, Test table capacity	1.2
High Frequency Test Program [12]	Fragility threshold SA_T = *(SA* + 0.625g)*	1.56
	Function Confirmed, Test table capacity	1.11
Qualification Test (IEEE 344 [25])	No chatter	1.2

Note: References listed in Table 4-2 are provided in EPRI 3002004396, which is Design Input #3 of this calculation

$F_k := 1.5$ CDFM Knockdown Factor (GERS, lowest level, no chatter)

Multi-axis Correction Factor: *[Design Input #4, page 3-68 & Design Input #14, page Q9]*

$F_{MS} := 1.2$ Multi-axis to single-axis correction factor applies due to predominantly single axis excitation

SCNH-17-007 Attachment K
Detailed High Frequency Confirmation Calculation Example 9
Table B-1 (Attachment B) No. 7 General Electric CR120A located in 1H11P622

Capacity vs. Demand:

Capacity:

$GERS := 9g$ GERS qualification level *[Design Input #12, Appendix B, GERS-RLY-AI2.4, pages B-43 - B-46]*

$Capacity := \left(\frac{GERS}{F_k}\right) \cdot F_{MS} = 7.2 \cdot g$ HCLPF broad-band component capacity *[Design Input #3, section 4.5.2 Eq. 4-5]*

Demand:

$Demand_H := AFc_H \cdot ISRS_{Hz} = 1.33 \cdot g$ Horizontal demand

$Demand_V := AFc_V \cdot ISRS_V = 1.4 \cdot g$ Vertical demand

Capacity vs. Demand Ratio:

$Ratio_H := \frac{Capacity}{Demand_H} = 5.41$ Horizontal

$Ratio_V := \frac{Capacity}{Demand_V} = 5.16$ Vertical

Peak Ground Accelerations for Hatch: *[Design Input #5, Table 2.4.1]*

$UHRS_{pga} := .1244g$ 1E-04 uniform hazard response spectra (UHRS) peak ground acceleration (PGA). Input motion for ISRS

$GMRS_{pga} := .1422g$ Ground motion response spectra (GMRS) peak ground acceleration (PGA).

SCNH-17-007 Attachment K
Detailed High Frequency Confirmation Calculation Example 9
Table B-1 (Attachment B) No. 7 General Electric CR120A located in 1H11P622

HCLPF to GMRS Ratio/Check:

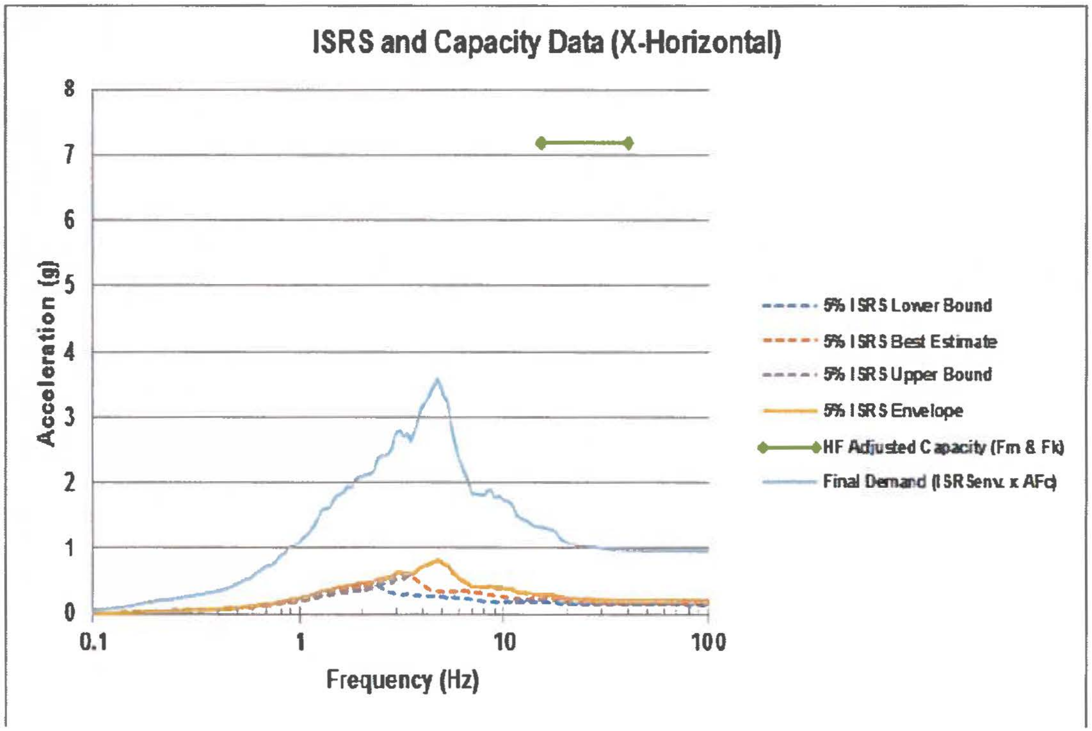
$$HCLPF_{pga} := \min(Ratio_H, Ratio_V) \cdot UHRS_{pga} = 0.64 \cdot g$$

$$HCLPF\_to\_GMRS\_ratio := \frac{HCLPF_{pga}}{GMRS_{pga}} = 4.51$$

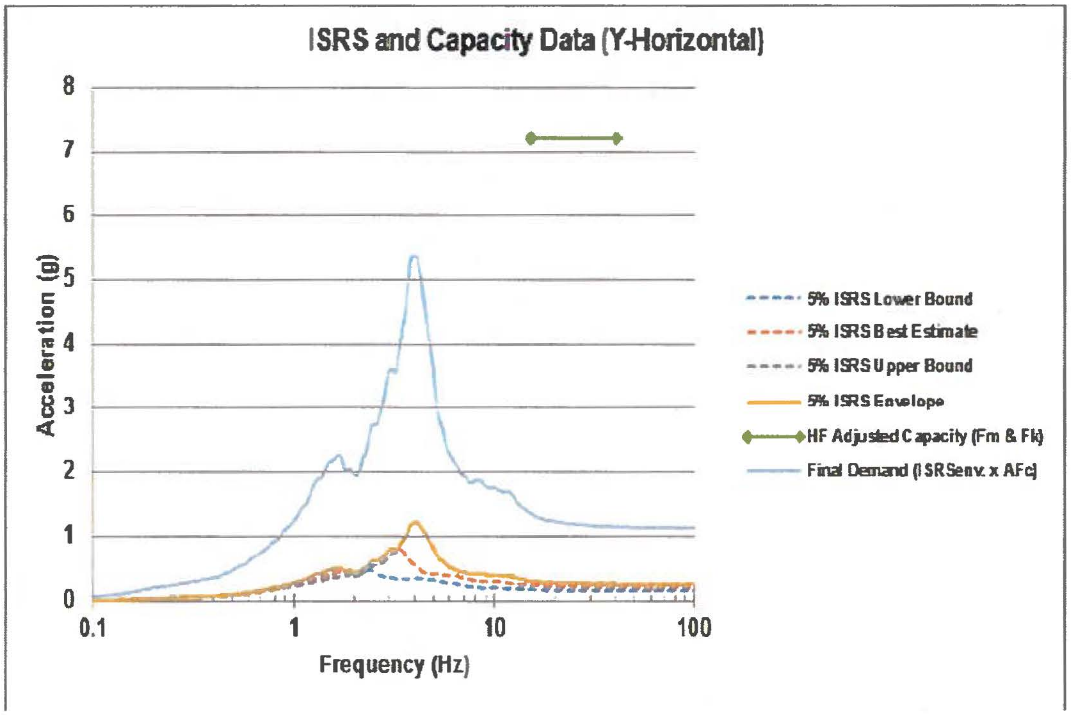
HCLPF_to_GMRS_check := "OK" if HCLPF_to_GMRS_ratio ≥ 1; "Not OK" otherwise

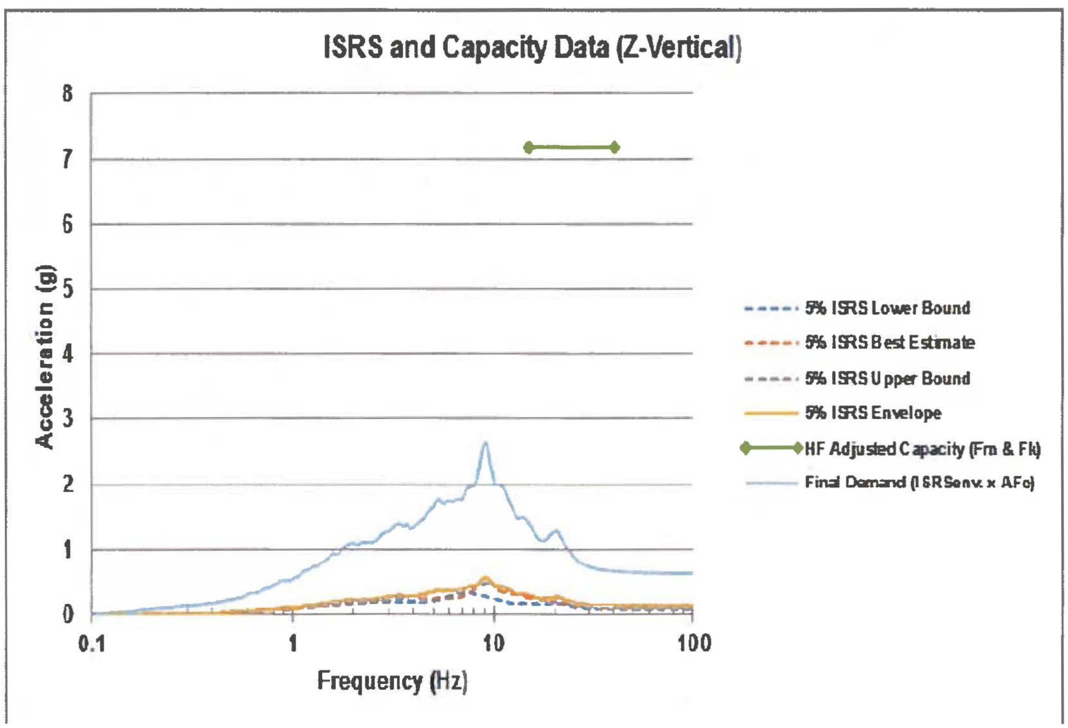
HCLPF_to_GMRS_check = "OK"

ISRS vs. Capacity Curves: *[Design Input #10]*
The following plots show the margin between the seismic demand based on 1E-04 UHRS ISRS and the relay capacity based on GERS or SQURTS



SCNH-17-007 Attachment K
Detailed High Frequency Confirmation Calculation Example 9
Table B-1 (Attachment B) No. 7 General Electric CR120A located in 1H11P622





B Components Identified for High Frequency Confirmation

This section contains a list of the specific components identified for high frequency confirmation including the plant specific component ID, Component type (relay, contactor, etc.) model number, location in the plant (floor elevation, enclosure type), system, and function to which it belongs. In addition, it provides the "Evaluation Result".

As detailed in Section 5.1 of this report, a total of 194 components were evaluated. As summarized in Table B-1, 179 of the devices have adequate seismic capacity and 15 components did not have adequate seismic capacity following the criteria in Section 4.5 of Reference [8].

The 15 components that did not have adequate seismic capacity are all General Electric (GE) CFD model relays. These relays are known to have a low seismic capacity. Plant Hatch already had operator actions in place that can adequately resolve potential seismic concerns with respect to these relays. Therefore, per section 4.6 of Reference [8], no additional actions are necessary.

Note: as described previously in Section 1.2 of this report, the scope of high-frequency components identified as part of the SPRA and provided in Table B-1 on the following pages includes and exceeds the scope of components that require a 2.1 Seismic High Frequency evaluation.

Table B-1: Components Identified for High Frequency Confirmation

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
1	1	42C [Closure of F004A] (Note 2)	Motor Starter	Core Cooling	Closure of Torus Suction Gate MOV	Allis Chalmers	Size 2	1R24S011	Motor Control Centers	RB	130	GERS	Cap > Dem
2	1	42C [Closure of F004B] (Note 2)	Motor Starter	Core Cooling	Closure of Torus Suction Gate MOV	Allis Chalmers	Size 2	1R24S012	Motor Control Centers	RB	130	GERS	Cap > Dem
3	1	42C [Closure of F004C] (Note 2)	Motor Starter	Core Cooling	Closure of Torus Suction Gate MOV	Allis Chalmers	Size 2	1R24S011	Motor Control Centers	RB	130	GERS	Cap > Dem
4	1	42C [Closure of F004D] (Note 2)	Motor Starter	Core Cooling	Closure of Torus Suction Gate MOV	Allis Chalmers	Size 2	1R24S012	Motor Control Centers	RB	130	GERS	Cap > Dem
5	1	42C [Closure of F001A] (Note 2)	Motor Starter	Core Cooling	Closure of C001A Suction Isolation Gate MOV	Allis Chalmers	Size 2	1R24S011	Motor Control Centers	RB	130	GERS	Cap > Dem
6	1	42C [Closure of F001B] (Note 2)	Motor Starter	Core Cooling	Closure of C001B Suction Isolation Gate MOV	Allis Chalmers	Size 2	1R24S012	Motor Control Centers	RB	130	GERS	Cap > Dem
7	1	(Contact: 5-6) K44	Auxiliary Relay	Cont. Isolation	HPCI Auto Isolation Signal	GE	CR120A	1H11P622	Control and Inst. Panels	CB	164	GERS	Cap > Dem
8	1	42C [Closure of F002] (Note 2)	Motor Starter	Cont. Isolation	Closure of Steam Supply Isolation Gate MOV	Allis Chalmers	Size 2	1R24S011	Motor Control Centers	RB	130	GERS	Cap > Dem
9	1	(Contact: 1-2) K56	Auxiliary Relay	Cont. Isolation	Turbine Exhaust Vacuum Breaker Auto Close (1E41-F104)	GE	CR120A	1H11P622	Control and Inst. Panels	CB	164	GERS	Cap > Dem
10	1	42/C [closure of F104] (Note 2)	Motor Starter	Cont. Isolation	Closure of HPCI Vacuum Breaker Line Gate MOV	Allis Chalmers	Size 2	1R24S011	Motor Control Centers	RB	130	GERS	Cap > Dem
11	1	(Contact: 1-7) K57	Auxiliary Relay	Cont. Isolation	Turbine Exhaust Vacuum Breaker Auto Close (1E41-F111)	GE	HGA	1H11P620	Control and Inst. Panels	CB	164	GERS	Cap > Dem
12	1	42C [Closure of F111] (Note 2)	Motor Starter	Cont. Isolation	Closure of HPCI Vacuum Breaker Line Gate MOV	Allis Chalmers	Size 2	1R24S012	Motor Control Centers	RB	130	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
13	1	(Contact: 5-6) K33	Auxiliary Relay	Cont. Isolation	RCIC Isolation Signal	GE	HFA	1H11P623	Control and Inst. Panels	CB	164	GERS	Cap > Dem
14	1	42C [Closure of F007] (Note 2)	Motor Starter	Cont. Isolation	Closure of Steam Supply Isolation Gate MOV	Allis Chalmers	Size 2	1R24S018B	Motor Control Centers	RB	130	GERS	Cap > Dem
15	1	(Contact: 1-7)K47	Auxiliary Relay	Cont. Isolation	Turbine Exhaust Vacuum Breaker Auto Close (Valve 1E51-F104)	GE	HGA	1H11P621	Control and Inst. Panels	CB	164	GERS	Cap > Dem
16	1	72/RCR [Closure of F104] (Note 2)	Motor Starter	Cont. Isolation	Closure of RCIC Exhaust Line Vacuum Breaker Gate Valve	Allis Chalmers	Size 2	1R24S011	Motor Control Centers	RB	130	GERS	Cap > Dem
17	1	(Contact: 1-7) K48	Auxiliary Relay	Cont. Isolation	Turbine Exhaust Vacuum Breaker Auto Close (Valve 1E51-F105)	GE	HGA	1H11P623	Control and Inst. Panels	CB	164	GERS	Cap > Dem
18	1	R [Closure of F105] (Note 2)	Motor Starter	Cont. Isolation	Closure of RCIC Exhaust Line Vacuum Breaker Gate Valve	Allis Chalmers	Size 2	1R24S012	Motor Control Centers	RB	130	GERS	Cap > Dem
19	1	72RCR [Closure of F524] (Note 2)	Motor Starter	Core Cooling	Closure of RCIC Trip and Throttle Valve	Allis Chalmers	Size 2	1R24S021	Motor Control Centers	RB	110	GERS	Cap > Dem
20	1	1S32-K152-1&3	Protective Relay	AC/DC Power Support Systems	Essential Transformer 1B Overcurrent Relay	GE	IAC 53A	1R23S003	Trans-formers	CB	130	GERS	Cap > Dem
21	1	1S32-K148-1&3	Protective Relay	AC/DC Power Support Systems	Essential Transformer 1C Overcurrent Relay	GE	IAC 53A	1R23S004	Trans-formers	CB	130	GERS	Cap > Dem
22	1	(Contact: 2 and 4) 87-S1CX	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 1C Differential Relay for Breaker #135717 (Contact 2) and 600V Bus 1C Normal Breaker Differential Relay #135811 (Contact 4)	GE	HEA	1H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
23	1	1S32-K231-1	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 1C Overcurrent Relay	GE	IAC 54A	1H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
24	1	1S32-K231-2	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 1C Overcurrent Relay	GE	IAC 54A	1H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
25	1	1S32-K231-3	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 1C Overcurrent Relay	GE	IAC 54A	1H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
26	1	1S32-K232-1	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 1C Differential Relay	Westing-house	HU	1H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
27	1	1S32-K232-2	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 1C Differential Relay	Westing-house	HU	1H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
28	1	1S32-K232-3	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 1C Differential Relay	Westing-house	HU	1H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
29	1	(Contact: 2 and 4) 87-S1DX	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 1D Differential Relay for Breaker #135719 (Contact 2) and 600V Bus 1D Normal Breaker Differential Relay #135814 (Contact 4)	GE	HEA	1H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
30	1	1S32-K233-1	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 1D Overcurrent Relay	GE	IAC 54A	1H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
31	1	1S32-K233-2	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 1D Overcurrent Relay	GE	IAC 54A	1H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
32	1	1S32-K233-3	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 1D Overcurrent Relay	GE	IAC 54A	1H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
33	1	1S32-K234-1	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 1D Differential Relay	Westing-house	HU	1H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
34	1	1S32-K234-2	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 1D Differential Relay	Westing-house	HU	1H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
35	1	1S32-K234-3	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 1D Differential Relay	Westing-house	HU	1H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
36	1	(Contact: 2-10) 81-D1A-X	Protective Relay	AC/DC Power Support Systems	DG 1A Frequency Relay	GE	HEA	1H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
37	1	1R43A-K1	Protective Relay	AC/DC Power Support Systems	DG 1A Exciter, Relaying and Metering Relays	Westing-house	MME-20-25	1R43P001A	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
38	1	1R43A-K3	Protective Relay	AC/DC Power Support Systems	DG 1A Exciter, Relaying and Metering Relays	ROWAN	2190	1R43P001A	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
39	1	1R43A-K4	Protective Relay	AC/DC Power Support Systems	DG 1A Exciter, Relaying and Metering Relays	ROWAN	2190	1R43P001A	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
40	1	1S32-K243-1	Protective Relay	AC/DC Power Support Systems	DG 1A Differential Relay	GE	CFD	1H21P200	Control and Inst. Panels	DGB	130	GERS	Operator Action
41	1	1S32-K243-2	Protective Relay	AC/DC Power Support Systems	DG 1A Differential Relay	GE	CFD	1H21P200	Control and Inst. Panels	DGB	130	GERS	Operator Action
42	1	1S32-K243-3	Protective Relay	AC/DC Power Support Systems	DG 1A Differential Relay	GE	CFD	1H21P200	Control and Inst. Panels	DGB	130	GERS	Operator Action

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
43	BOTH	(Contact: 2, 2C-C, 6-6C, 8) 87-D1BX	Protective Relay	AC/DC Power Support Systems	DG 1B Differential Relay: Breaker #135912 (Contact 2-2C), Breaker #135570 (Contact 6-6C), DG 1B STOP	GE	HEA	1H21P201	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
44	BOTH	1R43B-K1	Protective Relay	AC/DC Power Support Systems	DG 1B Exciter, Relaying and Metering Relays	Westing-house	MME-20-25	1R43P001B	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
45	BOTH	1R43B-K3	Protective Relay	AC/DC Power Support Systems	DG 1B Exciter, Relaying and Metering Relays	ROWAN	2190	1R43P001B	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
46	BOTH	1R43B-K4	Protective Relay	AC/DC Power Support Systems	DG 1B Exciter, Relaying and Metering Relays	ROWAN	2190	1R43P001B	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
47	BOTH	1S32-K253-1	Protective Relay	AC/DC Power Support Systems	DG 1B Differential Relay	GE	CFD	1H21P201	Control and Inst. Panels	DGB	130	GERS	Operator Action
48	BOTH	1S32-K253-2	Protective Relay	AC/DC Power Support Systems	DG 1B Differential Relay	GE	CFD	1H21P201	Control and Inst. Panels	DGB	130	GERS	Operator Action
49	BOTH	1S32-K253-3	Protective Relay	AC/DC Power Support Systems	DG 1B Differential Relay	GE	CFD	1H21P201	Control and Inst. Panels	DGB	130	GERS	Operator Action
50	1	(Contact: 2-10) 81-D1C-X	Protective Relay	AC/DC Power Support Systems	DG 1C Frequency Relay	GE	HEA	1H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
51	1	1R43C-K1	Protective Relay	AC/DC Power Support Systems	DG 1C Exciter, Relaying and Metering Relays	Westing-house	MME-20-25	1R43P001C	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
52	1	1R43C-K3	Protective Relay	AC/DC Power Support Systems	DG 1C Exciter, Relaying and Metering Relays	ROWAN	2190	1R43P001C	Control and Inst. Panels	DGB	130	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
53	1	1R43C-K4	Protective Relay	AC/DC Power Support Systems	DG 1C Exciter, Relaying and Metering Relays	ROWAN	2190	1R43P001C	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
54	1	1S32-K263-1	Protective Relay	AC/DC Power Support Systems	DG 1C Differential Relay	GE	CFD	1H21P202	Control and Inst. Panels	DGB	130	GERS	Operator Action
55	1	1S32-K263-2	Protective Relay	AC/DC Power Support Systems	DG 1C Differential Relay	GE	CFD	1H21P202	Control and Inst. Panels	DGB	130	GERS	Operator Action
56	1	1S32-K263-3	Protective Relay	AC/DC Power Support Systems	DG 1C Differential Relay	GE	CFD	1H21P202	Control and Inst. Panels	DGB	130	GERS	Operator Action
57	1	(Contact: 2, 4 and 5) 87-SICDX	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 1CD Differential Relay for Breaker #135718 (Contact 2), 600V Bus 1D Alternate Breaker Differential Relay #135813 (Contact 4) and 600V Bus 1C Alternate Breaker Differential Relay #135812 (Contact 5)	GE	HEA	1H21P201	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
58	1	1S32-K235-1/2/3	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 1CD Overcurrent Relay	GE	IAC 54A	1H21P201	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
59	1	1S32-K236-1/2/3	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 1CD Differential Relay	Westing-house	HU	1H21P201	Control and Inst. Panels	DGB	130	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
60	1	(Contact: 1-1C) 86	Control Relay	AC/DC Power Support Systems	MCC 1B Station Service Transformer Lockout Relay	GE	HEA	1R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
61	1	1S32-K217-1/2/3/4	Protective Relay	AC/DC Power Support Systems	MCC 1B Supply+F130 Transformer Overcurrent Relay	Westing-house	CO-9 1875282A	1R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
62	1	1S32-K216-1/2/3/4	Protective Relay	AC/DC Power Support Systems	Station Service Transformer 1F2 Overcurrent Relay	Westing-house	CO-9 1875282A	1R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
63	2	42C [Closure of F004A] (Note 2)	Motor Starter	Core Cooling	Closure of Torus Suction Gate MOV	Allis Chalmers	Size 2	2R24S011	Motor Control Centers	RB	130	GERS	Cap > Dem
64	2	42C [Closure of F004B] (Note 2)	Motor Starter	Core Cooling	Closure of Torus Suction Gate MOV	Allis Chalmers	Size 2	2R24S012	Motor Control Centers	RB	130	GERS	Cap > Dem
65	2	42C [Closure of F004C] (Note 2)	Motor Starter	Core Cooling	Closure of Torus Suction Gate MOV	Allis Chalmers	Size 2	2R24S011	Motor Control Centers	RB	130	GERS	Cap > Dem
66	2	42C [Closure of F004D] (Note 2)	Motor Starter	Core Cooling	Closure of Torus Suction Gate MOV	Allis Chalmers	Size 2	2R24S012	Motor Control Centers	RB	130	GERS	Cap > Dem
67	2	F [Closure of F001A] (Note 2)	Motor Starter	Core Cooling	Closure of C001A Suction Isolation Gate MOV	Allis Chalmers	Size 2	2R24S011	Motor Control Centers	RB	130	GERS	Cap > Dem
68	2	F [Closure of F001B] (Note 2)	Motor Starter	Core Cooling	Closure of C001B Suction Isolation Gate MOV	Allis Chalmers	Size 2	2R24S012	Motor Control Centers	RB	130	GERS	Cap > Dem
69	2	(Contact: 5-6) K44	Auxiliary Relay	Cont. Isolation	HPCI Auto Isolation Signal	GE	CR120A	2H11P622	Control and Inst. Panels	CB	164	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
70	2	2R [Closure of F002] (Note 2)	Motor Starter	Cont. Isolation	Closure of Stem Supply Isolation Gate MOV	Allis Chalmers	Size 2	2R24S011A	Motor Control Centers	RB	164	GERS	Cap > Dem
71	2	(Contact: 1-2) K56	Auxiliary Relay	Core Cooling	Turbine Exhaust Vacuum Breaker Auto Close (2E41-F104)	GE	CR120A	2H11P622	Control and Inst. Panels	CB	164	GERS	Cap > Dem
72	2	R [Closure of F104] (Note 2)	Motor Starter	Core Cooling	Closure of HPCI Vacuum Breaker Line Gate MOV	Allis Chalmers	Size 2	2R24S011	Motor Control Centers	RB	130	GERS	Cap > Dem
73	2	(Contact: 1-2) K57	Auxiliary Relay	Core Cooling	Turbine Exhaust Vacuum Breaker Auto Close (2E41-F111)	GE	HGA	2H11P620	Control and Inst. Panels	CB	164	GERS	Cap > Dem
74	2	R [Closure of F111] (Note 2)	Motor Starter	Core Cooling	Closure of HPCI Vacuum Breaker Line Gate MOV	Allis Chalmers	Size 2	2R24S012	Motor Control Centers	RB	130	GERS	Cap > Dem
75	2	(Contact: 5-6) K33	Auxiliary Relay	Cont. Isolation	RCIC Isolation Signal	GE	HFA	2H11P623	Control and Inst. Panels	CB	164	GERS	Cap > Dem
76	2	42C [Closure of F007] (Note 2)	Motor Starter	Cont. Isolation	Closure of Steam Supply Isolation Gate MOV	Allis Chalmers	Size 2	2R24S012B	Motor Control Centers	RB	164	GERS	Cap > Dem
77	2	(Contact: 1-7)K47	Auxiliary Relay	Core Cooling	Turbine Exhaust Vacuum Breaker Auto Close (Valve 2E51-F104)	GE	HGA	2H11P621	Control and Inst. Panels	CB	164	GERS	Cap > Dem
78	2	R [Closure of F104] (Note 2)	Motor Starter	Core Cooling	Closure of RCIC Exhaust Line Vacuum Breaker Gate Valve	Allis Chalmers	Size 2	2R24S011	Motor Control Centers	RB	130	GERS	Cap > Dem
79	2	(Contact: 1-7)K48	Auxiliary Relay	Core Cooling	Turbine Exhaust Vacuum Breaker Auto Close (Valve 2E51-F105)	GE	HGA	2H11P623	Control and Inst. Panels	CB	164	GERS	Cap > Dem
80	2	R [Closure of F105] (Note 2)	Motor Starter	Core Cooling	Closure of RCIC Exhaust Line Vacuum Breaker Gate Valve	Allis Chalmers	Size 2	2R24S012	Motor Control Centers	RB	130	GERS	Cap > Dem
81	2	2R [Closure of F524] (Note 2)	Motor Starter	Core Cooling	Closure of RCIC Trip and Throttle Valve	GE	Size 1 Full Voltage Reversing Starter; GEH-1496F	2R24S021	Motor Control Centers	RB	130	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
82	2	2S32-K838-1/2	Protective Relay	AC/DC Power Support Systems	Essential Transformer 2B Overcurrent Relay	GE	IAC 53A	2R23S003	Trans-formers	CB	130	GERS	Cap > Dem
83	2	2S32-K844-2	Protective Relay	AC/DC Power Support Systems	Essential Transformer 2C Overcurrent Relay	GE	IAC 53A	2R23S004	Trans-formers	CB	130	GERS	Cap > Dem
84	2	2S32-K791-1	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2C Overcurrent Relay	GE	IAC 54A	2H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
85	2	2S32-K791-2	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2C Overcurrent Relay	GE	IAC 54A	2H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
86	2	2S32-K791-3	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2C Overcurrent Relay	GE	IAC 54A	2H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
87	2	2S32-K791-4	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2C Overcurrent Relay	GE	IAC 54A	2H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
88	2	2S32-K792-1	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2D Overcurrent Relay	GE	IAC 54A	2H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
89	2	2S32-K792-2	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2D Overcurrent Relay	GE	IAC 54A	2H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
90	2	2S32-K792-3	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2D Overcurrent Relay	GE	IAC 54A	2H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
91	2	2S32-K792-4	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2D Overcurrent Relay	GE	IAC 54A	2H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
92	2	2S32-K836-1	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2C Differential Relay	Westing-house	HU	2H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
93	2	2S32-K836-2	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2C Differential Relay	Westing-house	HU	2H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
94	2	2S32-K836-3	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2C Differential Relay	Westing-house	HU	2H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
95	2	(Contact: 2 and 4) 87S2C-X	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2C Differential Relay for Breaker #135536 (Contact 2), 600V Bus 2C Normal Breaker Differential Relay #135674 (Contact 4)	GE	HEA	2H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
96	2	2S32-K821-1	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2C Overcurrent Relay	GE	IAC 54A	2H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
97	2	2S32-K821-2	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2C Overcurrent Relay	GE	IAC 54A	2H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
98	2	2S32-K821-3	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2C Overcurrent Relay	GE	IAC 54A	2H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
99	2	2S32-K821-4	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2C Overcurrent Relay	GE	IAC 54A	2H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
100	2	2S32-K822-1	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2D Overcurrent Relay	GE	IAC 54A	2H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
101	2	2S32-K822-2	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2D Overcurrent Relay	GE	IAC 54A	2H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
102	2	2S32-K822-3	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2D Overcurrent Relay	GE	IAC 54A	2H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
103	2	2S32-K822-4	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2D Overcurrent Relay	GE	IAC 54A	2H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
104	2	2S32-K842-1	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2D Differential Relay	Westing-house	HU	2H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
105	2	2S32-K842-2	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2D Differential Relay	Westing-house	HU	2H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
106	2	2S32-K842-3	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2D Differential Relay	Westing-house	HU	2H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
107	2	(Contact: 2 and 4) 87S2D-X	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2D Differential Relay for Breaker #135556 (Contact 2) and 600V Bus 2D Normal Breaker Differential Relay #135684 (Contact 4)	GE	HEA	2H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
108	2	2R43A-K1	Low Voltage Contactor	AC/DC Power Support Systems	DG 2A Exciter, Relaying and Metering Relays	SQUARE D	8508	2R43P001A	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
109	2	2R43A-K3	Protective Relay	AC/DC Power Support Systems	DG 2A Exciter, Relaying and Metering Relays	ITE/ROWAN	2190	2R43P001A	Control and Inst. Panels	DGB	130	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
110	2	2R43A-K4	Protective Relay	AC/DC Power Support Systems	DG 2A Exciter, Relaying and Metering Relays	ITE/ROWAN	2190	2R43P001A	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
111	2	2R43-K758-1	Protective Relay	AC/DC Power Support Systems	DG 2A Differential Relay	GE	CFD	2H21P200	Control and Inst. Panels	DGB	130	GERS	Operator Action
112	2	2R43-K758-2	Protective Relay	AC/DC Power Support Systems	DG 2A Differential Relay	GE	CFD	2H21P200	Control and Inst. Panels	DGB	130	GERS	Operator Action
113	2	2R43-K758-3	Protective Relay	AC/DC Power Support Systems	DG 2A Differential Relay	GE	CFD	2H21P200	Control and Inst. Panels	DGB	130	GERS	Operator Action
114	2	(Contact: 2C-C and 8C-8) 87D2A-X	Protective Relay	AC/DC Power Support Systems	DG 2A Differential Relay: Breaker #135530 (Contact 2C-C), DG 2A STOP (Contact 8C-8)	GE	HEA	2H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
115	2	2R43C-K1	Low Voltage Contactor	AC/DC Power Support Systems	DG 2C Exciter, Relaying and Metering Relays	SQUARE D	8508	2R43P001C	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
116	2	2R43C-K3	Protective Relay	AC/DC Power Support Systems	DG 2C Exciter, Relaying and Metering Relays	ITE/ROWAN	2190	2R43P001C	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
117	2	2R43C-K4	Protective Relay	AC/DC Power Support Systems	DG 2C Exciter, Relaying and Metering Relays	ITE/ROWAN	2190	2R43P001C	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
118	2	2R43-K768-1	Protective Relay	AC/DC Power Support Systems	DG 2C Differential Relay	GE	CFD	2H21P202	Control and Inst. Panels	DGB	130	GERS	Operator Action
119	2	2R43-K768-2	Protective Relay	AC/DC Power Support Systems	DG 2C Differential Relay	GE	CFD	2H21P202	Control and Inst. Panels	DGB	130	GERS	Operator Action

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
120	2	2R43-K768-3	Protective Relay	AC/DC Power Support Systems	DG 2C Differential Relay	GE	CFD	2H21P202	Control and Inst. Panels	DGB	130	GERS	Operator Action
121	2	(Contact: 2C-C and 8C-8) 87D2C-X	Protective Relay	AC/DC Power Support Systems	DG 2A Differential Relay: Breaker #135540 (Contact 2C-C), DG 2C STOP (Contact 8C-8)	GE	HEA	2H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
122	2	(Contact: 2, 4 and 5) 87 S2CDX	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2CD Differential Relay for Breaker #135546 (Contact 2), 600V Bus 2D Alternate Breaker Differential Relay #135680 (Contact 4) and 600V Bus 2C Alternate Breaker Differential Relay #135670 (Contact 5)	GE	HEA	2H21P201	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
123	2	2S32-K847-1/2/3	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2CD Overcurrent Relay	GE	IAC 54A	2H21P201	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
124	2	2S32-K848-1/2/3	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer 2CD Differential Relay	Westing-house	HU	2H21P201	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
125	2	2S32-K818-1/2/3/4	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer SWGR Bus 2F1 Differential Relay	Westing-house	CO-9 1875282A	2R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
126	2	2S32-K819-1/2/3/4	Protective Relay	AC/DC Power Support Systems	Emergency Station Service Transformer SWGR Bus 2F Differential Relay	Westing-house	CO-9 1875282A	2R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
127	1	1S32-K201-1/2/3	Protective Relay	Core Cooling	Plant Service Water Pump 1A Overcurrent Relay	Westing-house	CO5	1R22S005	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
128	1	1S32-K201-4	Protective Relay	Core Cooling	Plant Service Water Pump 1A Overcurrent Relay (Neutral)	Westing-house	CO9	1R22S005	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
129	1	86 [No contact for C001A] (Note 4)	Control Relay	Core Cooling	Plant Service Water Pump 1A Lockout Relay	Westing-house	WL	1R22S005	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
130	1	1S32-K224-1/2/3	Protective Relay	Core Cooling	Plant Service Water Pump 1B Overcurrent Relay	Westing-house	CO5	1R22S007	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
131	1	1S32-K224-4	Protective Relay	Core Cooling	Plant Service Water Pump 1B Overcurrent Relay (Neutral)	Westing-house	CO9	1R22S007	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
132	1	86 [No contact for C001B] (Note 4)	Control Relay	Core Cooling	Plant Service Water Pump 1B Lockout Relay	Westing-house	WL	1R22S007	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
133	1	1S32-K212-1/2/3	Protective Relay	Core Cooling	Plant Service Water Pump 1C Overcurrent Relay	Westing-house	CO5	1R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
134	1	1S32-K212-4	Protective Relay	Core Cooling	Plant Service Water Pump 1C Overcurrent Relay (Neutral)	Westing-house	CO9	1R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
135	1	86 [No contact for C001C] (Note 4)	Control Relay	Core Cooling	Plant Service Water Pump 1C Lockout Relay	Westing-house	WL	1R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
136	1	1S32-K210-1/2/3	Protective Relay	Core Cooling	Plant Service Water Pump 1D Overcurrent Relay	Westing-house	CO5	1R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
137	1	1S32-K210-4	Protective Relay	Core Cooling	Plant Service Water Pump 1D Overcurrent Relay (Neutral)	Westing-house	CO9	1R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
138	1	86 [No contact for C001D] (Note 4)	Control Relay	Core Cooling	Plant Service Water Pump 1D Lockout Relay	Westing-house	WL	1R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
139	2	2S32-K795-1/2/3	Protective Relay	Core Cooling	Plant Service Water Pump 2A Overcurrent Relay	Westing-house	CO5	2R22S005	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
140	2	2S32-K795-4	Protective Relay	Core Cooling	Plant Service Water Pump 2A Overcurrent Relay (Neutral)	Westing-house	CO9	2R22S005	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
141	2	86 [No contact for C001A] (Note 4)	Control Relay	Core Cooling	Plant Service Water Pump 2A Lockout Relay	Westing-house	WL	2R22S005	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
142	2	2S32-K827-1/2/3	Protective Relay	Core Cooling	Plant Service Water Pump 2B Overcurrent Relay	Westing-house	CO5	2R22S007	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
143	2	2S32-K827-4	Protective Relay	Core Cooling	Plant Service Water Pump 2B Overcurrent Relay (Neutral)	Westing-house	CO9	2R22S007	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
144	2	86 [No contact for C001B] (Note 4)	Control Relay	Core Cooling	Plant Service Water Pump 2B Lockout Relay	Westing-house	WL	2R22S007	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
145	2	2S32-K810-1/2/3	Protective Relay	Core Cooling	Plant Service Water Pump 2C Overcurrent Relay	Westing-house	CO5	2R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
146	2	2S32-K810-4	Protective Relay	Core Cooling	Plant Service Water Pump 2C Overcurrent Relay (Neutral)	Westing-house	CO9	2R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
147	2	86 [No contact for C001C] (Note 4)	Control Relay	Core Cooling	Plant Service Water Pump 2C Lockout Relay	Westing-house	WL	2R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
148	2	2S32-K806-1/2/3	Protective Relay	Core Cooling	Plant Service Water Pump 2D Overcurrent Relay	Westing-house	CO5	2R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
149	2	2S32-K806-4	Protective Relay	Core Cooling	Plant Service Water Pump 2D Overcurrent Relay (Neutral)	Westing-house	CO9	2R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
150	2	86 [No contact for C001D] (Note 4)	Control Relay	AC/DC Power Support Systems	Plant Service Water Pump 2D Lockout Relay	Westing-house	WL	2R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	GERS	Cap > Dem
151	1	(Contact: 2) 40-DIAX	Protective Relay	AC/DC Power Support Systems	DG 1A Field Relay	GE	HEA	1H21P230	Control and Inst. Panels	DGB	130	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
152	1	(Contact: 2 and 8) 87-DIAX	Protective Relay	AC/DC Power Support Systems	DG 1A Differential Relay: Breaker #135911 (Contact 2), DG 1A STOP (Contact 8)	GE	HEA	1H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
153	1	(Contact: 4-7) 1R22-K001 and 1R22-K019	Control Relay	AC/DC Power Support Systems	Interposing Relays for Closing of Breaker #135711 and #135712 (Bus 1E)	Potter & Brumfield	KUEP-7D15-110	1R22S005	Medium Voltage Metal Clad Switch-gear	DGB	130	SQURTS Report	Cap > Dem
154	1	(Contact: 8-2) 51-DIAX	Protective Relay	AC/DC Power Support Systems	DG 1A Overcurrent Relay	GE	HGA	1H21P230	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
155	1	R43 SDR [No contact/MPL # for DG 1A] (Note 4)	Control Relay	AC/DC Power Support Systems	DG 1A Shutdown Relay	A.O. Smith Clark Control Div.	4U 3NO3NC	1R43P001A	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
156	1	(Contact: 2 and 6-6C) 40-DIBX	Protective Relay	AC/DC Power Support Systems	DG 1B Field Relay: DG 1B Breaker #135912 (Contact 2) and DG 1B Breaker #135570 (Contact 6-6C)	GE	HEA	1H21P231	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
157	1	(Contact: 2, 2C-C, 6-6C, 8) 87-D1BX	Protective Relay	AC/DC Power Support Systems	DG 1B Differential Relay: Breaker #135912 (Contact 2-2C), Breaker #135570 (Contact 6-6C), DG 1B STOP (Contact 8)	GE	HEA	1H21P201	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
158	1	(Contact: 3-4, 7-8) 51-DIBX	Protective Relay	AC/DC Power Support Systems	DG 1B Overcurrent Relay: Breaker #135912 (Contact 3-4) and Breaker #135570 (Contact 7-8)	GE	HFA	1H21P231	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
159	1	R43 SDR [No contact/MPL # for DG 1B] (Note 4)	Control Relay	AC/DC Power Support Systems	DG 1B Shutdown Relay	A.O. Smith Clark Control Div.	4U 3NO3NC	1R43P001B	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
160	1	(Contact: 3-4) 1R43K832 (Note 3)	Control Relay	AC/DC Power Support Systems	DG 1B LOCA-B1X2 Relay	GE	HFA	2H21P231	Control and Inst. Panels	DGB	130	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
161	1	(Contact: 3-4) 1R43K833 (Note 3)	Control Relay	AC/DC Power Support Systems	DG 1B LOCA-B1X3 Relay	GE	HFA	2H21P231	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
162	1	(Contact: 3) 86/F (Note 3)	Control Relay	AC/DC Power Support Systems	DG 1B Lockout Relay	GE	HEA	2H11P652	Control and Inst. Panels	CB	164	GERS	Cap > Dem
163	1	(Contact: 4-7) 1R22-K021 and 1R22-K039	Control Relay	AC/DC Power Support Systems	Interposing Relays for Closing of Breaker #135713 and #135714 (Bus 1F)	Potter & Brumfield	KUEP-7D15-110	1R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	SQURTS Report	Cap > Dem
164	1	(Contact: 2) 40-DICX	Protective Relay	AC/DC Power Support Systems	DG 1C Field Relay	GE	HEA	1H21P232	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
165	1	(Contact: 2 and 8) 87-DICX	Protective Relay	AC/DC Power Support Systems	DG 1C Differential Relay: Breaker #135913 (Contact 2) and DG 1C STOP (Contact 8)	GE	HEA	1H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
166	1	(Contact: 8-2) 51-DICX	Protective Relay	AC/DC Power Support Systems	DG 1C Overcurrent Relay	GE	HGA	1H21P232	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
167	1	R43 SDR [No contact/MPL # for DG 1C] (Note 4)	Control Relay	AC/DC Power Support Systems	DG 1C Shutdown Relay	A.O. Smith Clark Control Div.	4U 3NO3NC	1R43P001C	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
168	1	(Contact: 4-7) 1R22-K045 and 1R22-K061	Control Relay	AC/DC Power Support Systems	Interposing Relays for Closing of Breaker #135715 and #135716 (Bus 1G)	Potter & Brumfield	KUEP-7D15-110	1R22S007	Medium Voltage Metal Clad Switch-gear	DGB	130	SQURTS Report	Cap > Dem
169	2	(Contact: 2C-2) 40-D2AX	Protective Relay	AC/DC Power Support Systems	DG 2A Field Relay	GE	HGA	2H21P230	Control and Inst. Panels	DGB	130	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
170	2	(Contact: 2C-2 and 8C-8) 87-D2AX	Protective Relay	AC/DC Power Support Systems	DG 2A Differential Relay: Breaker #135530 (Contact 2C-2), DG 1A STOP (Contact 8C-8)	GE	HEA	2H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
171	2	(8-2) 51-D2AX	Protective Relay	AC/DC Power Support Systems	DG 2A Overcurrent Relay	GE	HGA	2H21P230	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
172	2	SDR [No contact/MPL # for DG 2A] (Note 4)	Control Relay	AC/DC Power Support Systems	DG 2A Shutdown Relay	A.O. Smith Clark Control Div.	4U 4NO4NC	2R43P001A	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
173	2	(Contact: 4-7) 2R22-K001 and 2R22-K017	Control Relay	AC/DC Power Support Systems	Interposing Relays for Closing of Breaker #135554 and #135544 (Bus 2E)	Potter & Brumfield	KUEP-7D15-110	2R22S005	Medium Voltage Metal Clad Switch-gear	DGB	130	SQURTS Report	Cap > Dem
174	2	(Contact: 2 and 6-6C) 40-DIBX (Note 3)	Protective Relay	AC/DC Power Support Systems	DG 1B Field Relay: DG 1B Breaker #135912 (Contact 2) and DG 1B Breaker #135570 (Contact 6-6C)	GE	HEA	1H21P231	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
175	2	(Contact: 2, 2C-C, 6-6C, 8) 87-D1BX (Note 3)	Protective Relay	AC/DC Power Support Systems	DG 1B Differential Relay: Breaker #135912 (Contact 2-2C), Breaker #135570 (Contact 6-6C), DG 1B STOP (Contact 8)	GE	HEA	1H21P201	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
176	2	(Contact: 3-4, 7-8) 51-DIBX (Note 3)	Protective Relay	AC/DC Power Support Systems	DG 1B Overcurrent Relay: Breaker #135912 (Contact 3-4) and Breaker #135570 (Contact 7-8)	GE	HFA	1H21P231	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
177	2	SDR [No contact/MPL # for DG 1B] (Note 4)	Control Relay	AC/DC Power Support Systems	DG 2B Shutdown Relay	A.O. Smith Clark Control Div.	4U 3NO3NC	1R43P001B	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
178	2	(Contact: 3-4) 86F (Note 3)	Control Relay	AC/DC Power Support Systems	DG 1B Lockout Relay	GE	HEA	1H11P652	Control and Inst. Panels	CB	164	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
179	2	(Contact: 9-10) 2R43K832	Control Relay	AC/DC Power Support Systems	DG 1B LOCA-B1X2 Relay	GE	HFA	2H21P231	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
180	2	(Contact: 9-10) 2R43K833	Control Relay	AC/DC Power Support Systems	DG 1B LOCA-B1X3 Relay	GE	HFA	2H21P231	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
181	2	(Contact: 4-7) 2R22-K023 and 1R22-K041	Control Relay	AC/DC Power Support Systems	Interposing Relays for Closing of Breaker #135574 and #135564 (Bus 2F)	Potter & Brumfield	KUEP-7D15-110	2R22S006	Medium Voltage Metal Clad Switch-gear	DGB	130	SQURTS Report	Cap > Dem
182	2	(2C-2) 40-D2CX	Protective Relay	AC/DC Power Support Systems	DG 2C Field Relay	GE	HEA	2H21P232	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
183	2	(2C-2 and 8C-8) 87-D2CX	Protective Relay	AC/DC Power Support Systems	DG 1C Differential Relay: Breaker #135540 (Contact 2) and DG 1B STOP (Contact 8C-8)	GE	HEA	2H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
184	2	(8-2) 51-D2CX	Protective Relay	AC/DC Power Support Systems	DG 2C Overcurrent Relay	GE	HGA	2H21P232	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
185	2	SDR [No contact/MPL # for DG 2C] (Note 4)	Control Relay	AC/DC Power Support Systems	DG 2C Shutdown Relay	A.O. Smith Clark Control Div.	4U 4NO2NC	2R43P001C	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
186	2	(Contact: 4-7) 2R22-K045 and 1R22-K061	Control Relay	AC/DC Power Support Systems	Interposing Relays for Closing of Breaker #135594 and #135584 (Bus 2G)	Potter & Brumfield	KUEP-7D15-110	2R22S007	Medium Voltage Metal Clad Switch-gear	DGB	130	SQURTS Report	Cap > Dem
187	1	(Contact: 2) 86E	Control Relay	AC/DC Power Support Systems	DG 1A Lockout Relay	GE	HEA	1H11P652	Control and Inst. Panels	CB	164	GERS	Cap > Dem

No	Unit	Component						Enclosure		Location		Component Evaluation	
		ID	Type	System	Function	Manuf.	Model No.	ID	Type	Bldg. (Note 1)	Floor EL. (ft)	Basis for Capacity	Evaluation Result
188	1	(Contact: 2) 86F	Control Relay	AC/DC Power Support Systems	DG 1B Lockout Relay	GE	HEA	1H11P652	Control and Inst. Panels	CB	164	GERS	Cap > Dem
189	1	(Contact: 2) 86G	Control Relay	AC/DC Power Support Systems	DG 1C Lockout Relay	GE	HEA	1H11P652	Control and Inst. Panels	CB	164	GERS	Cap > Dem
190	2	(Contact: 2-2T) 86E	Control Relay	AC/DC Power Support Systems	DG 2A Lockout Relay	GE	HEA	2H11P652	Control and Inst. Panels	CB	164	GERS	Cap > Dem
191	2	(Contact: 2-2T) 86F	Control Relay	AC/DC Power Support Systems	DG 1B Lockout Relay	GE	HEA	2H11P652	Control and Inst. Panels	CB	164	GERS	Cap > Dem
192	2	(Contact: 2-2T) 86G	Control Relay	AC/DC Power Support Systems	DG 2C Lockout Relay	GE	HEA	2H11P652	Control and Inst. Panels	CB	164	GERS	Cap > Dem
193	1	(Contact: 2 and 8) 87D1A-X	Protective Relay	AC/DC Power Support Systems	DG 1A Differential Relay: Breaker #135911 (Contact 2), DG 1A STOP (Contact 8)	GE	HEA	1H21P200	Control and Inst. Panels	DGB	130	GERS	Cap > Dem
194	1	(Contact: 2 and 8) 87D1C-X	Protective Relay	AC/DC Power Support Systems	DG 1C Differential Relay: Breaker #135913 (Contact 2), DG 1C STOP (Contact 8)	GE	HEA	1H21P202	Control and Inst. Panels	DGB	130	GERS	Cap > Dem

NOTES:

1. Building Name abbreviations are as follows:
    a. CB: Control Building
    b. DGB: Diesel Generator Building
    c. RB: Reactor Building
2. Motor Starter Relays do not have a specific MPL number or contact number.
3. This relay is associated with the Swing Diesel Generator DG 1B that controls one unit and is located physically in the other unit.
4. This relay does not have a specific MPL number or contact number.

C Summary of Seismic PRA Peer Review and Assessment of PRA Technical Adequacy

This Appendix has two purposes:

1. Provide a summary of the SPRA peer review
2. Provide the bases for the technical adequacy of the SPRA.

The Hatch SPRA was subjected to an independent peer review against the pertinent requirements in Part 5 of Addendum B of the ASME/ANS PRA Standard [Ref. C-1]. The dispositions of findings and observations from this review were subjected to an independent Facts and Observations (F&O) technical review and Focused scope peer review as per the guidance. The F&Os dispositions were assessed for technical adequacy by reviewing the actions taken by Southern Nuclear to close them out. This was performed per the guidance provided in the Appendix X of NEI 12-13 [Ref. C-5]. F&O dispositions were reviewed and it was determined that the F&Os had been adequately addressed through this technical review process. They are considered "closed" and are no longer relevant to the current SPRA model. The independent assessment concluded that all the F&Os of record were 100% resolved and 0% remain open.

The information presented here establishes that the SPRA has been peer reviewed by a team with adequate credentials to perform the assessment, establishes that the peer review process followed meets the intent of the peer review characteristics and attributes in Table 16 of RG1.200 R2 [Ref. C-2] and the requirements in Section 1-6 of the ASME/ANS PRA Standard [Ref. C-1], and presents the significant results of the peer review.

C.1. Overview of Peer Review

The peer review assessment [Ref. C-3] is summarized in this Appendix. The scope of the review encompassed the set of technical elements and supporting requirements (SR) for the SHA (seismic hazard), SFR (seismic fragilities), and SPR (seismic PRA modeling) elements for seismic CDF and LERF. The peer review therefore addressed the full set of SRs identified in Tables 6-4 through 6-6 of the SPID [Ref. C-4].

The Hatch SPRA peer review was conducted during the week of October 17, 2016. As part of the peer review, a walk-down of portions of the Plant Hatch was performed on October 18, 2016 by members of the peer review team who had the appropriate SQUG training.

C.2. Summary of the Peer Review Process

The peer review was performed against the requirements in Part 5 (Seismic) of Addenda B of the PRA Standard [Ref. C-1], using the peer review process defined in NEI 12-13 [Ref. C-5]. The review was conducted over a four-day period, with a summary and exit meeting on the morning of the fifth day.

The NEI 12-13 SPRA peer review process [Ref. C-5] involves an examination by each reviewer of their assigned PRA technical elements against the requirements in the Standard to ensure the robustness of the model relative to all the requirements.

Implementing the review involves a combination of a broad scope examination of the PRA elements within the scope of the review and a deeper examination of portions of the PRA elements based on what is found during the initial review. The supporting requirements (SRs) provide a structure which, in combination with the peer reviewers' PRA experience, provides the basis for examining the various PRA technical elements. If a reviewer identifies a question or discrepancy, that leads to additional investigation until the issue is resolved or a Fact and Observation (F&O) is written describing the issue and its potential impacts, and suggesting possible resolution.

For each technical element, i.e., SHA, SFR, SPR, a team of two or three peer reviewers were assigned, one having lead responsibility for that area. For each SR reviewed, the responsible reviewers reached consensus regarding which of the Capability Categories defined in the Standard that the PRA meets for that SR, and the assignment of the Capability Category for each SR was ultimately based on the consensus of the full review team. The Standard also specifies high level requirements (HLR). Consistent with the guidance in the Standard, capability Categories were not assigned to the HLRs, but a qualitative assessment of the applicable HLRs in the context of the PRA technical element summary was made based on the associated SR Capability Categories.

As part of the review team's assessment of capability categories, F&Os are prepared. There are three types of F&Os defined in NEI 12-13 [Ref. C-5]: Findings, which identify issues that must be addressed in order for an SR (or multiple SRs) to meet Capability Category II; Suggestions, which identify issues that the reviewers have noted as potentially important but not requiring resolution to meet the SRs; and Best Practices, which reflect the reviewers' opinion that a particular aspect of the review exceeds normal industry practice.

C.3. Peer Review Team Qualifications

The members of the peer review team were Mr. Paul Amico of Jensen Hughes, Dr. Glenn Rix of Geosyntec, Dr. Annie Kammerer of Annie Kammerer Consulting, Dr. Mayasandra Ravindra of MK Ravindra Consulting, Mr. Wen Tong of Simpson Gumpertz & Heger, Mr. Lawrence Mangan of FirstEnergy Nuclear Operating Company, Mr. Clement Littleton of Entergy, and Mr. Philip Tarpinian of Exelon Generation. The peer review team members met the peer review independence criteria in NEI 12-13 [Ref. C-5] and had no involvement in the development of the Plant Hatch SPRA.

Mr. Paul Amico, the team lead, is a nuclear engineer with almost forty years of experience in the performance and management of domestic and international programs involving risk and reliability technology and its application to the design and operation of nuclear plants. He has been involved in seismic PRA since 1981 and is currently very active in seismic PRA standards development and the performance of seismic PRAs.

Dr. Glenn Rix was the lead for the Seismic Hazard Analysis (SHA) technical element. He has almost 30 years of experience in geotechnical earthquake engineering and engineering seismology (particularly for the eastern and central U.S.), and seismic hazard assessment and risk mitigation for civil infrastructure. He was assisted in the hazard review by Dr. Annie Kammerer. Dr. Kammerer is a seismic hazard and risk specialist with over 15 years of consulting and nuclear regulatory experience. Dr. Kammerer specializes in integrated seismic hazard and risk evaluations and performance-based, risk-informed engineering. Dr. Kammerer is the lead of the seismic hazard working group for the ASME/ANS external event PRA Standard. Dr. Kammerer is an author of the current US NRC guidance for performing PSHA.

Dr. Mayasandra Ravindra was the lead for the Seismic Fragility Analysis (SFR) technical element. He has over 45 years of experience in risk and reliability analyses, probabilistic design code development, and evaluation of natural and man-made hazards for buildings, power plants and chemical facilities. Dr. Ravindra serves as Chair of Writing Group on "External Events" of the ASME/ANS PRA Standard. He was assisted by Mr. Wen Tong. Mr. Tong has over thirty years of experience in seismic structural dynamics analysis and evaluations of seismic capacities of building structures and equipment. He has performed seismic fragility and seismic margins evaluations for more than thirty nuclear power plants.

Mr. Lawrence Mangan was the lead for the Seismic Plant Response (SPR) technical element. Mr. Mangan has 8 years of experience developing and maintaining Probabilistic Risk Assessment (PRA) models for the Perry Nuclear Power Plant, including Seismic, Internal Flooding, and Internal Events. He was assisted by Mr. Clement Littleton and Mr. Philip Tarpinian. Mr. Littleton has over 25 years of PRA experience covering a broad range of PRA applications on a day-to-day basis. Mr. Tarpinian has sixteen (16) years of experience in the specific field of PRA and is the Exelon corporate PRA engineer responsible for overall direction and oversight for all risk analyses associated with NRC Near Term Task Force (NTTF) Recommendations 2.1 and 2.3. This includes development of SPRA models, and risk support for NTTF Recommendations 4.2 (Mitigation (FLEX) Strategies) and 5.1 (Severe Accident Containment Vent).

Working observers were Mr. Winston Stewart of Duke Energy Carolinas and Mr. Habib Shtaih of Energy Northwest. Other observers include Mr. Jeffrey Clarke (Exelon Generation), Ms. Sara Lyons (US Nuclear Regulatory Commission), and Mr. Brett Titus (US Nuclear Regulatory Commission).

C.4. Summary of the Peer Review Conclusions

The review team's assessment of the SPRA elements is excerpted from the peer review report [Ref. C-3] as follows. Where the review team identified issues, these are captured in peer review findings.

SHA

- As required by the Standard, the frequency of occurrence of earthquake ground motions at the site was based on a probabilistic seismic hazard analysis (PSHA). The seismic source characterization (SSC) inputs to the PSHA are based on the Central and Eastern U.S. (CEUS) regional SSC model published in NUREG-2115 (i.e., the "CEUS-SSC" model). The ground motion characterization (GMC) inputs to the PSHA are based on an updated model published in 2013 by EPRI's the CEUS ground motion update project. The seismic hazard analysis for the Hatch site also accounts for the effects of local site response.
- The Senior Seismic Hazard Analysis Committee (SSHAC) methodology defines a process of structured expert interaction (elicitation) that is considered a minimum technical requirement for conduct of a PSHA. The SSHAC process (NUREG/CR-6372 and NUREG-2117) of conducting a PSHA was used to develop both the SSC and GMC models used as inputs to

the analysis. Use of the SSHAC methodology ensures that data, methods and models supporting the PSHA are fully incorporated and that uncertainties are fully considered in the process at sufficient depth and detail necessary to satisfy scientific and regulatory needs. The SSHAC-related guidance documents define and describe four "levels." The level of study is not mandated in the Standard; however, both the SSC and the GMC parts of the PSHA were developed as a result of SSHAC Level 3 analyses. In the case of the GMC, a SSHAC Level 2 analysis was carried out to update a prior Level 3 study. These Level 3 studies satisfy the requirements of the Standard related to the method of conduct of the PSHA generally, as well as addressing several individual requirements related to data collection, data evaluation and model development, and quantification of uncertainties supporting HLR-A to HLR-D.

- As a first step to performing a PSHA, the Standard requires that an up-to-date database, including regional geological, seismological, geophysical data, and local site topography, and a compilation of information on surficial geologic and geotechnical site properties. These data include a catalog of relevant historical, instrumental, and paleoseismic information within 320 km of the site. The CEUS-SSC study involved an extensive data collection effort that satisfies the requirements of the Standard as it relates to developing a regional-scale seismic source model.
- In the implementation of the CEUS-SSC model for the Hatch site, all distributed seismic sources in the CEUS-SSC model were included in the PSHA calculations. By including these seismic sources in the analysis, the contribution of "near-" and "far-field" earthquake sources to ground motions at the Hatch site were considered. An effort was made to identify local seismic sources that may not have been included in the regional model. Additional information pertinent to the site response analyses was collected and assessed.
- The CEUS-SSC and EPRI regional models discussed above were used for the Hatch site PSHA. Even though the PSHA conducted was performed specifically for the Hatch site, the underlying models were existing models and the seismicity database that underpins significant aspects of the CEUS-SSC only includes earthquakes through 2008. According to SHA-H1, if an existing model is used, a data collection and evaluation effort should be conducted to determine (1) whether new information has become available since the data was compiled for the existing model and, if so, (2) whether any new information challenges the validity of the technical basis of the existing study. It is not the case that identification of new data automatically requires an update to the PSHA existing model. Rather, an evaluation of the new data determines whether or not the existing model is appropriate for its continued use in the intended application. Supporting requirement SHA-H1 was found to be not met for the Hatch site for the reasons discussed below.
- In the case of the PSHA for the Hatch site, this data collection and review would be performed to support the continued use of the CEUS-SSC as a basis for the PSHA, particularly given that the CEUS-SSC catalog was finalized in 2008. However, as part of the Hatch site PSHA, there was no effort to compile new (relative to the data used in the CEUS SSC study) pertinent earth science information or evaluate any new information for its potential to challenge the ongoing viability of the regional study. This would include the development of an updated seismicity catalog that could be quantitatively assessed to ensure that (1) assumptions regarding the distribution of the maximum magnitude are not violated and (2) no new data exists that undermines the rate of seismicity of sources in the CEUS-SSC model important to the seismic hazard at the Hatch site. In some cases, an evaluation may include consideration of the mechanism (e.g., strike-slip, reverse, normal, or

oblique) of new seismicity. However, the CEUS-SSC model does not account for mechanism and so this consideration would not apply to the Hatch site.

- Additionally, the Standard addresses all sources that can potentially cause important vibratory ground motion at the Hatch site. The CEUS-SSC model used to assess vibratory ground motions explicitly removes non-tectonic earthquakes, which is appropriate because the underlying causation is different from tectonic earthquakes and is non-stationary (i.e., it may change over relatively short time periods). However, human-induced seismicity (e.g., earthquakes from wastewater injection) can produce damaging ground motions in some cases. While induced seismicity cannot (and should not) be incorporated into the tectonic SSC model, a separate seismicity catalog can be compiled and evaluated via a screening process. If necessary, a hazard analysis can be performed. The PSHA documentation does not indicate an effort to collect and assess information that would provide insight into the possible vibratory ground motion hazard from induced or triggered earthquakes (e.g., the presence or absence of injection wells in the area).
- The PSHA results are provided over an appropriately wide range of spectral frequencies and annual frequencies of exceedances. Uncertainties on the rock hazard are quantified, analyzed and reported as required in the standard. The lower-bound magnitude chosen for the analysis is consistent with standard practice. The results include fractile, median and mean hazard curves, and uniform hazard response spectra.
- The seismic hazard analysis for the Hatch site included a site response analysis. As part of the characterization of the site, historical, site-specific shear-wave velocity measurements and information from a nearby deep well were used to inform the site response analysis. The analysis includes the effects of site topography, surficial geologic deposits, and site geotechnical properties on ground motions at the site. The Standard requires that spectral shapes be based on a site-specific evaluation taking into account the contributions of deaggregated magnitude-distance results of the probabilistic seismic hazard analysis. The PSHA fully accounted for the "near-" and "far-field" source spectral shapes. The horizontal UHRS used in the SPRA is based on site-specific results and incorporates analysis results for all spectral frequencies. To ensure that the spectral shape captures potential site response effects, the results for the seven spectral frequencies used to calculate the hard-rock hazard were supplemented with an additional 31 spectral frequencies via interpolation and extrapolation. The interpolation is based on single- and double-corner spectral shapes for the CEUS from NUREG/CR-6728 and the extrapolation at low frequencies is based on accepted practice.
- Vertical to horizontal (V/H) ratios are used to calculate vertical response spectra. The approach employed to calculate the V/H ratios uses the results from NUREG/CR-6728 and modifies them to adjust for (i) soil site conditions (as compared to rock) and (ii) CEUS spectral shapes. This approach is considered to represent the state-of-the-practice for recent nuclear applications.
- Both the aleatory and epistemic uncertainties have been addressed in characterizing the seismic sources, ground motion models, and site response analyses. However, per the method allowed in EPRI (2013), the epistemic and aleatory uncertainties were prematurely combined to calculate a total uncertainty in the soil amplification factors. Although the estimate of the mean surface hazard is not affected, the epistemic uncertainty in the surface hazard curves is underestimated using this approach. Because the mean surface hazard is used to calculate the mean CDF and LERF, this shortcoming does not impact the mean CDF and LERF. The estimate of uncertainty in CDF and LERF, however, may be affected. The

premature combination of uncertainties resulted in a finding. However, because the mean values of CDF and LERF are unaffected, supporting requirement SHA-E2 was met.

- Notwithstanding the above premature combination of uncertainties, both epistemic and aleatory uncertainties are assessed in the site response analysis. Specifically, epistemic uncertainty is represented by three shear wave velocity profiles, two sets of modulus reduction and damping curves, and three values of kappa. Aleatory variability is represented by 60 random realizations of each profile, including random variations in shear wave velocity, modulus reduction and damping, kappa, layer thickness, and depth to rock. In general, the parameters selected to model each type of uncertainty are consistent with values recommended in EPRI (2013b). Correlation between properties is modeled when appropriate. The use of a damping adjustment factor to ensure the correct value of kappa is obtained in simulated profiles is considered to be a best practice.
- The rock hazard calculations are based on the CEUS-SSC and EPRI GMC models. During the development of these models, uncertainties in the seismic sources and ground motion prediction equations were included and appropriate sensitivity analyses were performed to demonstrate the sensitivity of the results to uncertainties in key model parameters. These sensitivity analyses are also documented in the associated reports. However, the effects of local site response for Plant Hatch are based entirely on site-specific calculations. No sensitivity analyses appear to have been performed to evaluate the sensitivity of the calculated soil amplification factors and/or surface hazard curves to variations in site response parameters (e.g., the shear wave velocity profile; modulus reduction and damping curves; and the near-surface, low-strain damping parameter kappa). Examining the difference in mean amplification factors provides qualitative insight into the effect of epistemic uncertainties in site response parameters, but a more formal sensitivity analysis is required by the Standard and would be helpful for understanding the relative importance of various site-specific parameters.
- Supporting requirement SHA-I1 addresses the bases and methodology used for any screening of the seismic hazards other than vibratory ground motion. The screening level analysis of other soil-related hazards including liquefaction potential, ground settlements, and slope stability/deformation references analyses presented in the Hatch IPEEE and provides a largely qualitative analysis of how changes in practice since the IPEEE study was completed in 1996 would affect the conclusions of these analyses. That analysis focuses on the changes in seismic demand and capacity relative to the IPEEE analyses and adjustments for conservatisms that were used in the IPEEE analyses. While the conclusion that other soil-related hazards screen out from further consideration may be valid, the approach taken results in considerable uncertainty about the veracity of that conclusion. It would have been more convincing to have updated the analyses using modern methods for liquefaction triggering, seismically induced ground settlement, and seismic slope stability/deformation.
- In addition to the above general finding, liquefaction triggering analyses performed for the FLEX travel paths at the for borings T-8 and T-9 indicate that the factors of safety against liquefaction (FS) are 1.0 and 1.3, respectively, for a groundwater table at Elevation 77 ft. Although the FS are 1.0 or above, analyses (e.g., Idriss and Boulanger, 2010) show that the probability of liquefaction associated with FS = 1.0 is approximately 15%, i.e., not negligible. Both soil borings are in proximity to the Reactor Building. The layer(s) with the lowest FS are at a depth of 60-70 ft below ground surface (bgs); the foundation of the Reactor Building is at a depth of 54 ft bgs. On this basis alone, liquefaction should not have been screened out from further consideration. Liquefaction-related ground failure should screen in, and

additional analyses should be performed to assess the "frequency of hazard occurrence and the magnitude of hazard consequences" per SHA-I2.

- Supporting requirement SHA-I2 addresses the hazard assessment requirements for non-vibratory seismic hazards that screen in. Because liquefaction-related ground failure could be an important contributor to the overall seismic risk profile, this model limitation needs to be addressed.
- The Standard requires that documentation of the PSHA that supports the PRA applications, peer review and potential future upgrades of the seismic hazard analysis be provided. This requirement establishes a high standard for documentation of the PSHA that allows for examination of the PSHA methodology, its implementation, and the PSHA results to evaluate whether the approach is appropriate, the analyses were performed correctly, and the results are reasonable. While the documentation related to SHA-J1 and SHA-J2 are sufficient, the Hatch PSHA documentation related to uncertainty does not fully satisfy the requirements of the Standard and therefore supporting requirement SHA-J3 is not met. There are two key issues with SHA-J3. The first is that the sensitivity studies related to site response were not provided in the hazard documentation. Secondly, there is no effort to document the uncertainties in the CEUS-SSC model, the EPRI GMC model, and the site response in one place. While the CEUS-SSC and EPRI GMC models broadly discuss uncertainties, nowhere are the various uncertainties related to the Hatch site specifically synthesized and put into context along with the site-specific site response study. The scattered and non-site-specific nature of the current documentation of the uncertainties that impact the hazard analysis for the Hatch site does not meet supporting requirement SHA-J3.

SFR

- The Standard requires that all the structures, systems and components (SSCs) that play a role in the seismic PRA be identified as candidates for subsequent seismic fragility evaluation. This was performed through the development of the Seismic Equipment List (SEL). As permitted by the Standard, extremely seismically rugged items in the list were screened out – i.e., no seismic fragility evaluation is required for these items. The Standard requires that the seismic-fragility evaluation be based on realistic seismic response that the SSCs experience at their failure levels. New 3-D building models were developed for all structures and used for this purpose. Since all the safety-related structures of Hatch are either founded on or embedded in the soil, the effects of soil-structure interaction (SSI) were considered in the seismic response analysis across three soil cases (lower bound, best estimate, and upper bound) to obtain median-centered and 84th percentile structure response and in-structure response spectra. The ground motions used for the SSI response analysis were the 1E-4 hazard level outcrop UHRS at the foundation levels of the different buildings that were calculated in the site response analysis.
- A series of walkdowns, focusing on the anchorage, lateral seismic support, functional characteristics, and potential systems interactions were conducted on Unit 1 and common areas and documented appropriately in support of the fragility analysis. The walkdowns also identified the potential for seismic-induced fires and floods. Subsequently, walkdown reviews of the Unit 2 SSCs were performed to either confirm their similarity to the Unit 1 SSCs or to document their differences. However, discussion of the Unit 2 SSC walkdown review observations and findings were not found in either the walkdown report or the Fragility Notebook. Based on the walkdown review, the SRT ranked each of the SSCs as

Rugged, High, Medium, or Low seismic capacity. The walkdown observations and capacity ranking were subsequently incorporated in the seismic fragility evaluations.

- The SPRA identifies all the relevant failure modes for the SSCs through a review of plant design documents, earthquake experience data, and walkdowns. Subsequently, seismic-fragility evaluations were performed only for the critical failure modes of the SSCs. Conservative generic seismic fragilities were calculated based on the screening level of Table 2-4 of EPRI NP-6041-SL, Revision 1 for those SSCs ranked as having high seismic capacity in the seismic walkdown review. Realistic seismic fragilities were developed for the SSCs ranked as having low seismic capacity or identified as potential risk contributors using the "separation of variables" method and the plant-specific data and the realistic in-structure response spectra at the mounting locations of the SSCs.
- The Standard requires that the seismic-fragility parameters be based on plant-specific data supplemented as needed by earthquake experience data, fragility test data, and generic qualification test data. The review team found that this requirement was generally satisfied.
- Seismic fragilities of all the SSCs, except the NSSS components and the building, were initially calculated using the seismic responses associated with the 2012 seismic hazard. The median seismic capacities of these SSCs were converted to that associated with the 2014 seismic hazard by scaling. This scaling approach was considered to be adequate due to similarity of the 2012 and the 2014 UHRS shapes and the corresponding ISRS, and the small effects of the revised soil properties (2014 seismic hazard) on the transfer functions as demonstrated in the Fragility Notebook.

SPR

- The seismic PRA model was developed by modifying the Full Power Internal Events (FPIE) PRA model to incorporate specific aspects of seismic analysis that are different from the FPIE. The logic model appropriately includes seismic-caused initiating events and other failures including seismic-induced SSC failures, non-seismic-induced unreliability and unavailability failure modes (based on the FPIE model), and human errors.
- The Hatch SPRA was one of the first in the industry to credit the use of FLEX provisions in a PRA model. While the PRA modeling was performed prior to the release of NEI 16-06 describing how to model such provisions, the work performed by Hatch was found to largely meet the requirements of NEI 16-06. Through their efforts Hatch should be recognized as an industry leader in the use of FLEX provisions in PRA.
- The Hatch SPRA used a multiplier method to account for adjustments to the Performance Shaping Factors (PSFs) to modify the internal events Human Error Probabilities (HEPs). New HEPs for seismic-specific scenarios such as recovery from relay chatter or to address FLEX and ELAP considerations were developed from a FPIE perspective and then similarly modified for Seismic effects using the same multipliers. The resulting Seismic HEPs appear to be unrealistically low in many cases. Some additional refinements to the Human Reliability Analysis (HRA) should be performed to develop HEPs that are more realistic.
- The work performed to address seismically induced fires and floods was judged to be incomplete. While some fire and flood sources were identified and included in the walkdowns and evaluations, it was found that a number of potential fire sources were not considered. Furthermore, the process to identify and screen internal flood sources was not clear and it could not be determined if all potential flood sources were adequately addressed.

- A number of sensitivities were performed to understand the impact of the various modeling and screening assumptions. In these aspects, the quantification of the Hatch SPRA is judged to meet the PRA Standard.
- It was noted that the Hatch Seismic Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) were much lower than has been found at other sites. Some of this is due to the lower hazard frequencies, as well as the robustness of the plant and significant SSCs. However, the low HEP values for many significant operator actions may be a non-conservative and artificially reducing CDF and LERF. Further work to refine the HRA analysis could possibly result in an increase in HEPs and an increase in CDF and LERF, but even if not, it will result in a more robust and realistic justification of the results that is currently missing.
- In conclusion, the seismic PRA model integrates the seismic hazard, the seismic fragilities, and the systems-analysis aspects appropriately to quantify core damage frequency and large early release frequency, albeit with the above noted deficiencies.
- The seismic-PRA analysis was extensively documented in a manner that facilitates applying and updating the SPRA model.

C.5. Summary of the Peer Review Assessment of Supporting Requirements and Findings

The PRA Standard [Ref. C-1] has 77 individual SRs for seismic PRA technical elements, and referenced SRs. The Peer Review included all of the individual SRs and most of the referenced SRs (not all SRs or referenced SRs are considered applicable). There were 4 SRs considered "not applicable" to the Hatch SPRA Peer Review. Of the 73 assessed PRA Standard SRs, 69 were supportive of Capability Category II or greater.

Based on the summary of Facts and Observations (F&Os) provided by the review team, 23 were categorized as Findings, 4 were categorized as Suggestions, and 4 were categorized as Best Practice. An independent assessment to review close out of F&Os of record was performed and is summarized in the next section.

C.6. Summary of the Technical Review of F&Os Disposition and Focused-Scope Peer Review

An independent assessment was performed in accordance with Appendix X, NEI 12-13 [Ref. C-5, Ref. C-6, and Ref. C-7] to review close out of F&Os of record from prior SPRA Peer Review against the ASME/ANS PRA Standard [Ref. C-1]. In addition, Focused-Scope Peer Review was performed to review a SPRA upgrade to the requirements in Part 5 of the ASME/ANS PRA Standard [Ref. C-1] using the peer review process defined in NEI 12-13 [Ref. C-5]. The review process was based on completed SPRA Peer Review. The F&O Technical Review and Focused-Scope Review was conducted on May 22, 2017 through June 30, 2017.

The Technical Team consisted of five team members. All members of the Hatch F&O Technical and Focused-Scope Review Team were experienced PRA personnel, each with appropriate experience in the areas of PRA that they were reviewing. The reviewers met the peer review independence criteria in NEI 12-13 [Ref. C-5] and had no involvement in the development of the Plant Hatch SPRA.

The F&O Technical Review Team's evaluation concluded that all the 27 F&Os (23 Findings and 4 Suggestions) were 100% resolved and 0% remain open. F&O dispositions were reviewed and it was determined that the F&Os had been adequately addressed through this technical review process. They are considered "closed" and no longer relevant to the current SPRA model.

A peer review was conducted on the Hatch Seismic PRA prior to this focused peer review. Since that review, an alternative approach was implemented to part of the PRA that meets the conditions of a PRA Upgrade as defined by the standard. The upgrade affects one high level requirement (HLR) under the technical element SHA, namely HLR SHA-I. Therefore, this focused peer review addressed the Supporting Requirements (SRs) under that HLR. The Focused-Scope Peer Review Team's evaluation concluded that 100% of the assessed SRs were met and there were no findings.

The F&O Technical Review and the Focused-Scope Peer Review Assessment is summarized in the report H-RIE-U00-010-002 [Ref. C-8].

C.7. Summary of Technical Adequacy of the SPRA for the 50.54(f) Response

The set of supporting requirements from the ASME/ANS PRA Standard [Ref. C-1] that are identified in Tables 6-4 through 6-6 of the SPID [Ref. C-4] define the technical attributes of a PRA model required for a SPRA. The conclusions of the peer review discussed above and summarized in this submittal demonstrates that the Hatch SPRA model meets the expectations for PRA scope and technical adequacy as presented in RG 1.200, Revision 2 [Ref. C-2] as clarified in the SPID [Ref. C-4].

Detailed archival information for the Hatch SPRA consistent with the listing in Section 4.1 of RG 1.200 Rev. 2 [Ref. C-2] is available if required to facilitate the NRC staff's review of this submittal.

The peer review observations and conclusions noted in Section C.4, the summary of the peer review assessment of SRs and F&Os discussed in Section C.5, the results of the technical review of F&Os disposition and focused-scope peer review summarized in Section C.6, and the discussion in Section C.7 demonstrate that the Hatch SPRA is technically adequate in all aspects for this submittal.

References:

C-1) ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, including Addenda B, 2013, American Society of Mechanical Engineers, New York, September 30, 2013.

C-2) Regulatory Guide 1.200, Revision 2, "An Approach For Determining The Technical Adequacy Of Probabilistic Risk Assessment Results For Risk-Informed Activities," U.S. Nuclear Regulatory Commission, March 2009.

C-3) Southern Nuclear Operating Company, H-RIE-SEIS-U00-010-001, Version 1.0, Seismic PRA – Hatch Units 1 and 2, Seismic PRA Peer Review Report Using ASME/ANS PRA Standard Requirements, August 2017.

C-4) EPRI 1025287, Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic. Electric Power Research Institute, Palo Alto, CA: February 2013.

C-5) NEI-12-13, External Hazards PRA Peer Review Process Guidelines, Revision 0, Nuclear Energy Institute, Washington, DC, August 2012.

C-6) U.S. Nuclear Regulatory Commission, Staff Expectations for an Industry Facts and Observations Independent Assessment Process, May 1, 2017 (ML17121A271).

C-7) U.S. Nuclear Regulatory Commission, Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-12, and 12-13, Close-Out of Facts and Observations (F&Os), May 3, 2017 (ML17079A427).

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