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10 CFR 50.4
10 CFR 50.54(f)

MNS-16-075
CNS-16-066

October 20, 2016

Attention: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
Renewed License Numbers NPF-35 and NPF-52

McGuire Nuclear Station, Units 1 and 2
Docket Numbers 50-369, 50-370
Renewed License Numbers NPF-9 and NPF-17

Subject: Supplemental Information Regarding Reevaluated Seismic Hazard Screening and Prioritization Results - Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident

References:

1. NRC Letter, Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, dated March 12, 2012 (ADAMS Accession No. ML12053A340)
2. Duke Energy letter, Seismic Hazard and Screening Report (CEUS Sites), Response to NRC 10 CFR 50.54(f) Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, dated March 20, 2014 (ADAMS Accession No. ML14098A421)
3. Duke Energy letter, Seismic Hazard and Screening Report (CEUS Sites), Response to NRC 10 CFR 50.54(f) Request for Additional Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, dated March 31, 2014 (ADAMS Accession No. ML14093A052)
4. Duke Energy letter, Expedited Seismic Evaluation Process (ESEP) Report (CEUS Sites), Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, dated December 17, 2014 (ADAMS Accession No. ML15005A085)

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NRR

5. Duke Energy letter, Catawba Nuclear Station Expedited Seismic Evaluation Process Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, dated December 31, 2014
6. NRC Letter, Screening and Prioritization Results Regarding Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Seismic Hazard Re-Evaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, dated May 9, 2014 (ADAMS Accession No. ML14111A147)
7. NRC Letter, 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, dated October 27, 2015 (ADAMS Accession No. ML15194A015)
8. Nuclear Energy Institute 12-06, Revision 2 "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide", Dated December 2015 (ADAMS Accession No. ML16005A625)

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 located in the Central and Eastern United States (CEUS) to submit a Seismic Hazard Evaluation and Screening Report.

In References 2 and 3, Duke Energy Carolina, LLC (Duke Energy) provided the Seismic Hazard and Screening Reports for McGuire Nuclear Station (MNS), Units 1 and 2, and Catawba Nuclear Station (CNS), Units 1 and 2, respectively. In References 4 and 5, Duke Energy provided the Expedited Seismic Evaluation Process (ESEP) reports for MNS, and CNS, respectively.

In References 6 and 7, the NRC issued the initial industry seismic hazard reevaluation screening and prioritization results and the final seismic risk evaluation determination results, respectively. Following the initial screening and the additional assessment, NRC determined that MNS and CNS "screened-in" to conduct seismic risk evaluations, so as to inform the NRC regulatory decisions on the adequacy of the MNS and CNS current seismic design-bases. A Seismic Probabilistic Risk Assessment (SPRA) is the risk evaluation approach to be performed for MNS and CNS. The SPRA submittal dates for CNS and MNS are September 30, 2019 and December 31, 2019, respectively.

The enclosed report provides supplemental information regarding risk insights associated with the seismic capabilities of the MNS and CNS sites. This information provides additional detailed basis and justification supporting the determination that these sites are low-to-moderate risk sites for seismic hazards; and therefore, do not require additional seismic risk evaluations (i.e. SPRA). This determination is based on additional seismic risk and seismic capacity insights for each site as described in the enclosure to this letter.

By no later than December 31, 2016, Duke Energy respectfully requests NRC reconsider the need for MNS and CNS to conduct seismic risk evaluations (i.e. SPRAs). In the event NRC concludes that MNS and CNS have low-to-moderate risk for seismic hazards and that SPRAs are not warranted, then work on the SPRAs will be suspended. Furthermore, the schedule for MNS and CNS will be accelerated to provide the high frequency evaluation, consistent with Section 4.7 of EPRI 3002004396, by no later than August 31, 2017.

In addition, MNS and CNS plan to complete the seismic mitigating strategies assessment (MSA) in accordance with Reference 8 (Appendix H), Path 4, by August 31, 2017.

This letter contains no new regulatory commitments. For any questions regarding this submittal, please contact Jeff Thomas at 704-382-3438 (jeff.thomas@duke-energy.com) or Paul Guill at 704-382-4753 (paul.guill@duke-energy.com).

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 20, 2016.

Sincerely,

Ernest J. Kapopoulos, Jr.
Vice President, Operations Support

Enclosure:

1. McGuire Nuclear Station, Units 1 and 2 and Catawba Nuclear Station, Units 1 and 2, Supplemental Information Regarding Reevaluated Seismic Hazard Screening and Prioritization Results (35 pages)

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CNS Master File (CN04DM, File CN 801.01)
RGC Date File (CN01RC)

NCMPA-1
PMPA
NCEMC
MNS Master File (MG01DM, File MC 801.01)
ELL (EC2ZF)

ENCLOSURE 1

Supplemental Information Regarding Reevaluated Seismic Hazard Screening and Prioritization Results

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ATTACHMENTS

1.0 INTRODUCTION

The purpose of this enclosure is to provide supplemental information regarding seismic risk and capacity insights to the Nuclear Regulatory Commission (NRC) to support the staff decision-making process for providing relief from the requirement to perform a Seismic Probabilistic Risk Assessment (SPRA) at the McGuire Nuclear Station, Units 1 and 2 (hereafter referred to as "McGuire") and the Catawba Nuclear Station, Units 1 and 2 (hereafter referred to as "Catawba").

Duke Energy has concluded that McGuire and Catawba are low-to-moderate risk sites for seismic hazards and that performance of a Seismic Probabilistic Risk Assessment will not provide significant additional seismic risk insights. This determination is based on insights derived from examination of the significant body of knowledge already available. This body of knowledge consists of the reevaluated seismic hazards and associated seismic risk and capacity information including, but not limited to, previous generic and site-specific seismic risk evaluations, the State-of-the-Art Reactor Consequences Analysis, site-specific Conditional Containment Failure Probability analyses, and the FLEX mitigating strategies for beyond design-basis external events.

2.0 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 NRC established a Near Term Task Force (NTTF) to conduct a systematic review of NRC processes and regulations. The NTTF developed a set of recommendations to enhance protection and mitigation from external events and to strengthen emergency preparedness. On March 12, 2012, the NRC issued a 10 CFR 50.54(f) letter (Reference 1) requesting that licensees reevaluate the seismic hazards at their sites utilizing present day NRC methods and guidance. Licensees were required to submit their reevaluated hazards by March 31, 2014.

By letter dated May 9, 2014 (Reference 2), the NRC provided the initial screening and prioritization results following reviews of updated earthquake hazard information from nuclear power plant sites in the Central and Eastern United States (CEUS). The NRC grouped the "screened-in" and "conditionally screened-in" plants into three groups. Group 1 plants are generally those plants that have the highest reevaluated hazards relative to their original licensing bases in the 1 to 10 Hz frequency range. Group 2 plants have moderate reevaluated hazards relative to their original licensing bases. Group 3 plants have the lowest reevaluated hazards relative to their original licensing bases. NRC categorized McGuire as a Group 3 plant and Catawba as a Group 2 plant and set forth the expectation that both sites should perform a seismic risk assessment, an expedited approach evaluation as an interim action, and limited scope evaluations for high frequency and spent fuel pools.

Following the initial screening and prioritization, the NRC staff performed an additional assessment, examining available information to determine the need for "screened in" plants to perform a seismic risk evaluation. Results from that additional assessment are documented via letter dated October 27, 2015 (Reference 3), which exempted sites with low to moderate reevaluated seismic hazard exceedance above their current deign basis from performing a seismic risk evaluation. Furthermore, the letter documents the NRC determination that McGuire and Catawba should perform SPRAs by December 31, 2019, and September 30, 2019, respectively, to assess the total plant response to the reevaluated hazard.

3.0 GENERAL INFORMATION FOR SPRA RELIEF

McGuire and Catawba are low to moderate risk sites for seismic hazards, as demonstrated by previous seismic risk evaluations. The reevaluated seismic hazard for McGuire and Catawba resulted in a modest exceedance above the current seismic licensing basis for these plants.

3.1 Reevaluation of Seismic Hazard

The 50.54(f) letter was issued, in part, to gather information concerning the seismic hazards at operating reactor sites. The "Required Response" section of Enclosure 1 of the 50.54(f) letter indicated that licensees should provide a Seismic Hazard Evaluation and Screening report within 1.5 years from the date of the letter for CEUS nuclear power plants, and within 3 years for Western United States (WUS) plants. For CEUS plants, the date to submit the report was extended to March 31, 2014, by NRC letter dated May 7, 2013 (Reference 4).

In response to the 50.54(f) letter, McGuire and Catawba utilized the Electric Power Research Institute (EPRI) Report 1025287, Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima NTTF Recommendation 2.1: Seismic (Reference 5), as supplemented by the EPRI Report 3002000704, Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima NTTF Recommendation 2.1: Seismic (Reference 6).

On March 20, 2014 (Reference 7) and March 31, 2014 (Reference 8), Duke Energy submitted the new Ground Motion Response Spectrum (GMRS) along with the results of its screening assessment to the NRC for McGuire and Catawba, respectively. The horizontal GMRS for McGuire and Catawba are described in Section 2.0 of those references. The modest exceedance of the new seismic hazard (i.e. GMRS) above the design basis Safe Shutdown Earthquake (SSE), combined with the seismic risk estimations, indicate that a relatively low to moderate seismic risk is expected for McGuire and Catawba.

3.2 Significance of Design Exceedance - Seismic Design Margin

McGuire and Catawba were designed based on the largest earthquakes expected in the regional and local areas around the plant at the time of licensing, using conservative practices to provide substantial margin to safely withstand large earthquake ground motions. These conservative design practices include:

- Safety factors applied in design calculations
- Use of elastic damping values in dynamic analysis of structures and equipment
- Bounding synthetic time histories for in-structure response spectra calculations
- Broadening criteria for in-structure response spectra
- Response spectra enveloping criteria used in Structures, Systems and Components (SSC) analysis and testing applications
- Response spectra based frequency domain analysis rather than explicit time history based time domain analysis
- Bounding requirements in codes and standards

- Use of minimum strength requirements of structural components instead of actual, tested strength values (concrete and steel) , and
- Bounding testing requirements
- Additional capacity in the primary materials such as steel and reinforced concrete beyond the elastic capacity credited in designs

These conservative design practices for McGuire and Catawba result in additional available capacities, which ensures that the SSCs will continue to fulfill their design functions at ground motions well above the SSE. Additional discussion on how seismic ruggedness is achieved through the design process and demonstrated by earthquake experience is provided in Reference 9.

3.3 Seismic Risk Evaluations and Insights

Seismic Core Damage Frequency (SCDF) point estimates have been made over the years starting with the Individual Plant Examination of External Events (IPEEE) and continuing to the NRC safety/risk assessment results for Generic Issue (GI)-199 (Reference 10). Most-recently, SCDF point estimates for the EPRI fleet-wide risk assessment (Reference 9) for the latest seismic hazards (i.e. GMRS) were completed. The historical estimates for Catawba and McGuire are discussed below.

IPEEE Seismic Core Damage Frequencies

In response to NRC Generic Letter 88-20, Supplement 4 (Reference 11), Duke Energy utilized existing SPRAs and the guidance in NUREG-1407, Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, June 1991 (Reference 12), to calculate seismic core damage frequencies. See Sections 4.4 and 5.4 herein for McGuire and Catawba, respectively.

GI-199 Safety/Risk Assessment

During the 2000's, the nuclear industry proposed the construction of new nuclear plants co-located with operating plants. As a result, the NRC was requested to review Early Site Permits (ESPs) and Combined Operating Licenses (COLs) for new reactors. As part of those requests, plants submitted updated estimates of new ground motion response spectra which had the potential to differ from existing SSEs at co-located sites. The differences were due to updated seismic source models and ground motion prediction equations. The staff concluded on May 26, 2005 "*that the issue of increased seismic hazard estimates in the CEUS should be examined under the Generic Issue Program*" (Reference 13).

For the safety/risk assessment of GI-199, the NRC calculated SCDF point estimates and documented that work in Reference 10. The point estimates were made by convolving the seismic hazard with the plant-level fragility developed or estimated from plant-specific IPEEE submittals. At the time, there were three current seismic source models to consider: (1) EPRI-1989, (2) Lawrence Livermore National Laboratories (LLNL)-1994 and (3) United States Geological Survey-2008. Furthermore, the NRC used three different methods for the computation: (1) simple average, (2) IPEEE weighted average and (3) weakest link model. Finally, the calculations were made at four different frequencies of 1 Hz, 5 Hz, 10 Hz and Peak Ground Acceleration (PGA). This resulted in numerous calculations documented in Appendix D of the NRC safety/risk assessment. Attachment 2, Table 1 of this document, provides the historical range of SCDF estimates based on PGA.

The conclusions from the GI-199 safety/risk assessment were as follows: *"Results of the Safety/Risk Assessment indicate that there is no immediate concern regarding adequate protection, but that the issue should continue to the Regulatory Analysis Stage of the GIP* [Generic Issue Process] *(for further investigation regarding possible cost-justified backfits)."* (Reference 10)

In March of 2011 the Great Tohoku earthquake and subsequent tsunami resulted in the accident at the Fukushima Dia-ichi Nuclear Plant and GI-199 was subsumed by the Near Term Task Force recommendations.

Seismic Core Damage Frequency Estimates Using GMRS

Prior to the submittal of the site-specific GMRS for the CEUS fleet, EPRI performed a risk assessment for the CEUS fleet using the methods developed under the GI-199 safety assessment. In particular, the simple average method was used to convolve the plant level fragility developed under the GI-199 safety assessment with the new seismic hazards (i.e. GMRS). The EPRI CEUS fleet risk assessment was provided in a March 12, 2014 Nuclear Energy Institute (NEI) letter (Reference 9).

Because the EPRI risk assessment provided an SCDF estimate trend for the CEUS fleet in lieu of actual plant-specific SCDF estimates, Duke Energy requested the SPRA Vendor that performed the original work for EPRI to provide the SCDF point estimates for McGuire and Catawba. The more recent McGuire and Catawba SCDF numbers are presented below.

The overall trend from the EPRI SCDF estimates was that *"...the overall distribution of SCDFs for the fleet indicates that the impact of the updated seismic hazard has been to reduce risk for most plants relative to estimates obtained using either the 2008 USGS or the 1994 LLNL hazard estimates."* In addition:

- *"The range of SCDFs still falls between 1E-7/year and 1E-4/year.*
- *For individual plants, some plant SCDF estimates have increased, but the vast majority have decreased somewhat.*
- *In the case of the sites for which increases were seen, none of the SCDF values approaches 1E-4/year."*

Historical SCDF Point Estimates

Attachment 2, Table 1 of this document provides the historical range of SCDF estimates based on PGA. The estimates at PGA are used because the data in Appendix D shows that PGA controls for McGuire and Catawba (Reference 10). The table also shows that the range of calculated SCDF, considering IPEEE and GI-199 and all the different methods, is as follows:

- 1.1E-5/year < McGuire < 4.7E-5/year
- 1.5E-5/year < Catawba < 4.3E-5/year

Limiting the range to just the simple average method for McGuire and Catawba, which is consistent with the estimates made under the recent EPRI work for the new GMRS, the range reduces to the following:

- 1.5E-5/year < McGuire < 2.8E-5/year
- 1.7E-5/year < Catawba < 2.8E-5/year

The SCDF point estimates for McGuire and Catawba from the recent EPRI work, based on the new GMRS, are as follows:

- McGuire = 2.7E-5/year (using PGA)
- Catawba = 2.8E-5/year (using PGA)

The recent McGuire and Catawba SCDF point estimates using the new seismic hazards (i.e. GMRS) are within the previous range of estimates made under GI-199, considering all methods of computation. In addition, the new SCDF estimates for McGuire and Catawba are within the narrow range computed using just the simple average method. The conclusions reached in the GI-199 Safety/Risk Assessment in 2010 remain valid. Therefore, McGuire and Catawba have margin to withstand potential earthquakes exceeding their original design bases and no concern exists regarding adequate protection.

3.4 State-of-the-Art Reactor Consequences Analysis (SOARCA)

The SOARCA project was initiated in the early 2000's to develop best estimates of the offsite radiological health consequences for potential severe reactor accidents. The initial SOARCA work considered two pilot plants: the Peach Bottom Atomic Power Station in Pennsylvania and the Surry Power Station in Virginia. Peach Bottom is generally representative of U.S. operating reactors using the General Electric boiling-water reactor (BWR) design with a Mark I containment. Surry is generally representative of U.S. operating reactors using the Westinghouse pressurized-water reactor (PWR) design with a large, dry containment. After completing the Peach Bottom and Surry SOARCA analyses, a third pilot plant, Sequoyah, a PWR with an ice condenser containment, similar to McGuire and Catawba, was included.

Sequoyah was added to the study by the NRC because ice condenser containments have a lower design pressure than other U.S. nuclear power plant containment types and are therefore potentially more susceptible to early failure from hydrogen combustion during a severe accident. The more important severe accident scenarios for Sequoyah focused on issues associated with containment response. Hydrogen combustion has long been known to be a potential challenge to the ice condenser containment. The Sequoyah SOARCA analysis examines phenomenology and modeling unique to the ice condenser design including the behavior of hydrogen and the potential for early containment failure from energetic hydrogen combustion.

Duke Energy reviewed an NRC Draft Technical Report entitled State-of-the-Art Reactor Consequence Analyses (SOARCA) Project Sequoyah Integrated Deterministic and Uncertainty Analyses (Reference 14). Based on this review, two notable items from the SOARCA study reinforced risk insights that Duke Energy obtained from previous risk assessments, including the importance of the Turbine-Driven Auxiliary Feedwater Pump (TDAFWP) and the hydrogen igniters.

The Sequoyah SOARCA analysis assumes that a Station Blackout (SBO) is initiated by a low probability severe seismic event because this is an extreme case in terms of timing and equipment failure. For SBO, SOARCA assumes Alternate Current (AC) power is lost but the TDAFWP is available. Use of the TDAFWP to extend core cooling, delays containment failure following the initiating event. The TDAFWP system is very important in extending core cooling and allowing more time for implementation of additional mitigation.

McGuire and Catawba utilize the safety-related TDAFWP as the primary method for supplying feedwater to the steam generators during Phase 1 of an Extended Loss of AC Power (ELAP) event. Each TDAFWP is sized to provide sufficient feedwater flow and remove decay heat post-reactor trip. Moreover, each is designed, constructed and maintained in accordance with seismic Category I requirements. The McGuire and Catawba TDAFWPs were confirmed to be in accordance with the seismic design basis during the NTTF 2.3 walkdowns. Also, the TDAFWPs were examined during the Expedited Seismic Evaluation Process and confirmed to have adequate capacity for the Review Level Ground Motion in the 1 to 10 Hz range.

Furthermore, reflecting these insights on the Duke Energy ice condenser containment designs, each unit at McGuire and Catawba has two redundant trains of hydrogen igniters that are seismically supported and powered from independent, safety-related power supplies. Also, McGuire and Catawba have a unique non-seismic system, the Standby Shutdown System (SSS), which provides alternate AC to one train of hydrogen igniters. In the unlikely event the independent safety-related power supplies and the SSS are lost due to a significant seismic event, McGuire and Catawba have Phase 2 FLEX strategies that can repower the igniters when needed. Due to high redundancy, the igniters are not risk significant. Operator action to initiate the igniters is modeled in the internal events Probabilistic Risk Assessment (PRA). Also, the hydrogen igniters were examined during the Expedited Seismic Evaluation Process and confirmed to have adequate capacity for the Review Level Ground Motion in the 1 to 10 Hz range.

The Sequoyah SBO analysis expands on the SOARCA body of knowledge for ice condenser containments. Beyond expanding the body of knowledge on realistic outcomes of severe accidents, the SOARCA study has complemented and supported other NRC activities to address lessons learned from the Fukushima Dai-ichi accidents, specifically NTTF Recommendation 5.2 (reliable hardened vents for containment designs other than Mark I and Mark II) and NTTF Recommendation 6 (hydrogen control and mitigation inside containment or in other buildings). As documented in SECY-16-0041 (Reference 15), the NRC staff concluded that additional enhancements as a result of NTTF Recommendations 5.2 and 6 would not be justified when evaluated under the criteria in 10 CFR 50.109. The insights from the additional analyses related to ice condenser containments as part of continuing efforts in the SOARCA project provided the bases for concluding that additional study to install hardened containment vents and possible improvements to hydrogen control for ice condenser containments is unlikely to identify the need for regulatory actions beyond those already taken that would provide a substantial safety improvement.

In summary, the SOARCA reinforces the results of past analyses of ice condenser containments and accentuates the importance of the TDAFWP and the hydrogen igniters and their effectiveness in averting early containment failure.

3.5 NTTF 2.1 Spent Fuel Pool Evaluation

Enclosure 1 of the 50.54(f) letter also included a requirement for a Spent Fuel Pool (SFP) evaluation. To assist with the evaluation, the SPID (Reference 5) included comparison criteria, whereby, if the new GMRS exceeded the SSE in the 1 to 10 Hz part of the response spectrum, then a spent fuel pool evaluation was required. This is the case for McGuire and Catawba. Therefore, McGuire and Catawba screened-in for a spent fuel pool integrity evaluation.

To support these evaluations across the industry, on February 23, 2016 (Reference 16), NEI submitted EPRI Report 3002007148 entitled, Seismic Evaluation Guidance: Spent Fuel Pool Integrity Evaluation (SFP Evaluation Guidance Report) (Reference 17). The SFP Evaluation Guidance Report provides criteria for evaluating the seismic adequacy of a SFP to the reevaluated GMRS hazard levels. This report supplements the guidance in the SPID for plants such as McGuire and Catawba where the GMRS peak spectral acceleration is less than or equal to 0.8g (i.e., low to moderate GMRS sites). Section 3.0 of the EPRI report developed criteria that addresses SFP structural elements (e.g., floors, walls, and supports); non-structural elements (e.g., penetrations); seismic-induced SFP sloshing; and water loses due to heat-up and boil-off. In addition, the EPRI report also provides applicability criteria, which determines if site-specific conditions are within the bounds considered in developing the evaluation criteria.

By letters dated July 20, 2016 (Reference 18) and August 18, 2016 (Reference 19), Duke Energy submitted SFP evaluations for Catawba and McGuire, respectively. The NRC staff promptly reviewed these submittals and concluded that Catawba and McGuire provided sufficient information, as documented in letters dated August 11, 2016 (Reference 20) for Catawba and August 31, 2016 (Reference 21) for McGuire. In addition, the NRC concluded that the SFP structures for Catawba and McGuire are sufficiently robust. These sites can withstand ground motions with peak spectral acceleration less than or equal to 0.8g. NRC determined that Duke Energy acceptably evaluated the Catawba and McGuire non-structural considerations for SSCs whose failure could lead to potential drain-down of the SFP due to a seismic event. Accordingly, Catawba and McGuire SFPs are sufficiently seismically robust to withstand the reevaluated GMRS hazard levels, and the spent fuel stored in the Catawba and McGuire pools is adequately protected from the reevaluated seismic hazards.

4.0 MCGUIRE INFORMATION FOR SPRA RELIEF

This section discusses McGuire-specific seismic risk evaluations, Conditional Containment Failure Probability (CCFP) analyses, and the FLEX mitigating strategies for beyond design-basis external events. McGuire responses to other NTTF seismic activities are also discussed.

4.1 McGuire Ground Motion Response Spectrum

The reevaluated GMRS to SSE comparison is shown in Attachment 1, Figure 1 of this document. From that figure, the design basis SSE exceeds the GMRS below approximately 6 Hz, and the GMRS begins to exceed the McGuire SSE above 6 Hz. In the high frequency range greater than 10 Hz, structural displacements are small and are considered non-damaging. The peak acceleration of the new GMRS is 0.68g at 35 Hz.

According to SPID, the area of concern is in the 1 to 10 Hz range. From Attachment 1, Figure 1, the GMRS-to-SSE ratio is 1.74 in the 1 to 10 Hz range. Installed plant equipment credited in the FLEX strategies was confirmed during the Expedited Seismic Evaluation Process (ESEP) to have adequate capacity to perform its FLEX mitigation function in the 1 to 10 Hz range -- refer to Section 4.3 for additional information regarding the ESEP.

4.2 McGuire NTTF 2.3 Seismic Walkdown

Seismic walkdowns were conducted in accordance with EPRI Report 1025286 (Reference 22) to provide reasonable assurance that seismic equipment configuration control has been maintained consistent with the current seismic licensing basis, including consideration of seismic interaction concerns and equipment degradation. The EPRI document provided instruction and procedures to perform seismic walkdowns as required by the 50.54(f) letter. The EPRI guidance outlined requirements for personnel qualifications, selection of walkdown components, the conduct of the walkdowns, evaluation of potentially adverse conditions against the plant seismic licensing basis, and reporting requirements. The guidance further provided check lists to document the performance of the seismic walkdowns.

The initial walkdown results were reported to the NRC on November 26, 2012 (Reference 23). A total of 30 Potentially Adverse Seismic Conditions (PASC) were identified by the Seismic Walkdowns. The PASC were entered into the Corrective Action Program (CAP). Engineering evaluations were performed and concluded that the conditions were in conformance with the current seismic licensing bases. In some cases work requests or CAP Actions were initiated to resolve minor issues (e.g. loose fastener, add grout, etc.), update design documents, and/or to enhance field equipment clearances. The results of the walkdowns and licensing basis evaluations verified the adequacy of the McGuire monitoring and maintenance procedures with regard to maintaining the seismic licensing basis.

In summary, all PASC identified as a result of the seismic walkdowns were entered into CAP and resolved.

4.3 McGuire Expedited Seismic Evaluation Process (ESEP)

The intent of the ESEP is to demonstrate seismic margin through a review of a subset of the plant equipment that can be relied upon to restore or maintain core cooling and containment function following beyond design basis seismic events. The ESEP is implemented using the methodologies in EPRI Report 3002000704, Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic (Reference 6). Selected equipment with a High Confidence of a Low Probability of Failure (HCLPF) capacity in excess of the calculated Review Level Ground Motion (RLGM) are deemed to have adequate seismic capacity for the reevaluated seismic hazard in the 1 to 10 Hz frequency range.

The procedure for determining the RLGM for the ESEP is described in Section 4 of the EPRI Report. The RLGM is determined by multiplying the spectral acceleration values for the 5%-damped SSE horizontal ground response spectrum by a scale factor. The scale factor is the largest ratio of spectral accelerations between the 5%-damped GMRS and the 5%-damped SSE ground response spectrum at frequencies from 1 Hz to 10 Hz, but not to exceed 2.0. The ratio of the GMRS to the SSE over the 1 to 10 Hz frequency range for McGuire is 1.74.

The selection of equipment to be included on the Expedited Seismic Equipment List (ESEL) was based on plant equipment credited in the FLEX strategies. The scope includes equipment relied upon for the FLEX strategies to sustain the critical functions of core cooling and containment. FLEX recovery actions are excluded from the ESEP scope per EPRI Report 3002000704.

The next step in the process demonstrated that ESEL items have sufficient seismic capacity to meet or exceed the demand characterized by the RLGM. The seismic capacity is characterized as the PGA for which there is a HCLPF. The PGA is associated with a specific spectral shape, in this case the 5%-damped RLGM spectral shape. The HCLPF capacity must be equal to or greater than the RLGM PGA. The criteria for seismic capacity determination are given in Section 5 of EPRI Report 3002000704.

McGuire completed the ESEP and submitted an initial report on December 17, 2014 (Reference 24) and a final summary report on February 4, 2016 (Reference 25). All equipment evaluated for the ESEP was found to have adequate capacity for the McGuire RLGM.

4.4 McGuire Seismic Risk Evaluations and Insights

NRC issued Generic Letter 88-20 on November 23, 1988 (Reference 26), requesting that all licensees perform an Individual Plant Examination (IPE) to identify plant-specific vulnerabilities to severe accidents. At that time, Duke Energy began a program to update an earlier McGuire PRA study to take into account a number of modifications to the plant and to take advantage of plant-specific data and state-of-the-art methods. In November 1991 (Reference 27), Duke Energy submitted a response to the NRC, which included a quantitative assessment of SCDF. The IPE submittal explained that the McGuire PRA is a full-scope Level 3 PRA with complete analysis of external events in addition to internal events. External events have been included in the McGuire PRA studies beginning with the original study.

On June 28, 1991, the NRC issued Generic Letter 88-20, Supplement 4 (Reference 11), requesting that all licensees perform an IPEEE to identify plant-specific vulnerabilities to severe accidents solely due to external events. On June 1, 1994 (Reference 28), Duke Energy submitted its response to the NRC which included an updated assessment of SCDF as well as a qualitative assessment of containment performance. For its seismic assessment, McGuire used a seismic PRA (SPRA) rather than a seismic margins assessment. The SPRA analysis utilized the best data and analysis tools available at that time.

As indicated in Table 3.1 of NUREG-1407 (Reference 12), McGuire was placed in the 0.3g Focused Scope bin. As such, a median ground response spectrum anchored at 0.3g was used for the review level earthquake (RLE) for McGuire as recommended by NUREG-1407. For relay chatter events, Section 3.2.4.2 of the NUREG states that, for non-A-46 plants (e.g., McGuire), focused scope plants are required to locate and evaluate low seismic ruggedness relays or "bad actors" (as found in EPRI NP-7148-SL, Appendix E (Reference 29)).

The McGuire SPRAs were developed using EPRI's Computer Aided Fault Tree Analysis (CAFTA) software program to create a fault tree model to generate the seismic event cut sets. These were then combined with the mean plant seismicity curve and SSC fragilities into a Duke Energy in-house program, Seismic Event Impact Sequence Model (SEISM) which used Monte Carlo simulation to generate the final SCDF. The McGuire IPEEE SCDF was calculated to be 1.1E-05/yr. Several of the dominant accident sequences involve a loss of offsite power followed by a loss of both emergency diesel generators (i.e. SBO).

Because Large Early Relief Frequency (LERF) models had not been developed for McGuire at the time of the IPE and IPEEE submittals, a qualitative analysis was performed for containment performance in response to a seismically induced core damage accident. This was accomplished by examining the containment structure fragility analysis and the containment isolation function.

The entire fragility curve for any mode of failure and its uncertainty can be expressed in terms of best estimate of the median ground acceleration capacity times the product of random variables representing inherent randomness and uncertainty. In estimating these fragility parameters, it is computationally attractive to work in terms of an intermediate random variable called factor of safety. The factor of safety is defined as the ratio of the ground acceleration capacity to the SSE acceleration used in design. The development of seismic safety factors associated with the SSE is based on consideration of several variables. The median factor of safety and its statistical variability for each SSC for McGuire were determined, based on results of existing dynamic models and associated response analyses and evaluations of structures and equipment, supplemented by some limited additional analyses. The resulting median capacities for several of the McGuire structures are greater than 2.5g. The reactor building, the steel containment vessel, and the containment internal structure fall into this category.

Similar to structural fragility, factors of safety and their variabilities are first developed for equipment capacity, earthquake duration, and equipment response. These three factors, along with the factor of safety on structural response, are then multiplied together to obtain an overall factor of safety to be used for the equipment item. McGuire equipment fragility descriptions are based on (1) plant-specific design reports, (2) qualification test reports, (3) generic test or analysis data, (4) knowledge of design specifications and the factors of safety inherent in the governing codes and standards, and (5) engineering judgment and past earthquake experience. Of resulting equipment fragility descriptions, there is an equipment category for those items that possess ground acceleration capacities greater than 2.5g and will not contribute to the overall plant risk. The ice condenser structure and hydrogen igniters fall into this category.

The seismic impact on containment isolation was also evaluated. Piping, valves and supports associated with penetrations which, if failed, could lead to significant release pathways, were determined by the fragility vendor to have median fragilities greater than 2.5g. The cabinets housing the equipment used to generate the containment isolation signals had a median fragility of 1.54g. Likewise, the respective panel boards and Motor Control Centers (MCC) providing power to actuate the valve solenoids and motors were also evaluated for the beyond design basis in-structure levels and found to be adequate. The panel board fragility is 1.66g and the MCC fragility is 1.68g.

In addition, effects of chatter on relays within the containment isolation circuit were considered. A listing of the affected plant relays was compared against the listing of "bad actor" relays given in EPRI NP-7148-SL mentioned above. Since none of the McGuire relays that would compromise safe shutdown functions qualified as bad actors, this did not become a concern.

Thus, it was determined the containment structure and penetrations are seismically rugged, containment isolation would occur in response to a seismic induced core damage accident, and relays within the containment isolation circuit will function as designed.

Plant walkdowns were performed to support the development of the initial McGuire PRA which included external events. Walkdowns were also performed to support the 1991 IPE submittal.

As part of the IPEEE effort, extensive walkdowns were conducted on both units consistent with the guidelines given in EPRI NP-6041 (Reference 30). The purpose of these walkdowns was to confirm the validity of the equipment fragility assessments, to verify the seismic adequacy of equipment anchorage and to identify any other seismic concerns. Portions of the walkdowns were conducted inside containment for both units, focusing on plant equipment and other containment performance issues.

As a result of the walkdowns (including those for fire), several plant recommended enhancements were implemented as a result of this study. These included the following:

- Adding spacers between the Unit 1 diesel generator batteries and racks
- Adding grout between the Component Cooling heat exchangers saddle base and concrete curb
- Trimming the grating around the Steam Vent Valves
- Replacing missing bolts on the Unit 2 Upper Surge Tanks
- Modifying the Unit 2 Turbine-Driven Auxiliary Feedwater pump control panel to avoid contact with a nearby pipe
- Replacing/cleaning and recoating corroded nuts on the Auxiliary Feedwater Condensate Storage Tank anchor bolts
- Tightening the arc barrier connections inside the Main Control Boards

Overall, no fundamental plant weaknesses or vulnerabilities were identified.

4.5 McGuire Conditional Containment Failure Probability (CCFP)

As part of the McGuire IPE, a containment capacity analysis was performed to establish a "probability of failure" distribution for the containment structures. It was expressed as a distribution of probability of containment failure versus internal containment pressure. The ultimate pressure capacity of the McGuire steel containment vessel was evaluated in a calculation for use in the PRA. This calculation addresses the failure of the vessel shell and all appurtenances. A containment failure distribution is developed in a manner similar to that presented in NUREG/CR-1891, Reliability Analysis of Containment Strength (Reference 31). Insofar as is possible and practical, an assessment was made of all identified potential containment failure modes, failure locations, and failure sizes.

The cumulative CCFPs from the containment capacity analysis are recorded in Table G-4 of the IPE report (Reference 27). Attachment 1, Figure 2 shows the graph of the containment failure probabilities verses containment pressure. The curve combines the pressure fragility from the containment vessel shell, penetrations and anchorage failure modes analyzed. As can be seen from Figure 2, the median pressure capacity of the containment is 77 psi. The HCLPF capacity that corresponds to 1% failure probability of the containment is 56 psi, which is 3.7 times the containment design pressure of 15 psi.

In the past few years, Duke Energy performed plant-specific thermal hydraulic analyses using Modular Accident Analysis Program (MAAP) version 5.01 to develop plant-specific and accident sequence-specific CCFP in support of the Internal Events LERF PRA update. Inputs were taken from the containment capacity analysis which was developed during the IPE (and described above) for the containment failure probabilities. This CCFP analysis was developed to calculate plant-specific CCFP values for various accident sequences instead of using the generic CCFPs developed in NUREG/CR-6595 (Reference 32). The two parameters that represent the highest potential for containment failure are:

1. CCFP with no igniters operating and the Reactor Coolant System (RCS) at high pressure when reactor vessel (RV) failure occurs, and
2. CCFP with no igniters operating and the RCS at low pressure when RV failure occurs.

As used in the description of the CCFP parameters, the RCS pressure is high if Direct Containment Heating (DCH) is a potential threat and low when DCH is not a threat. Revision 1 to NUREG/CR-6595 was published in October of 2004. This revision included new estimates for the two CCFP parameters discussed earlier. The new recommended values for the two parameters are 1.0 and 0.97 (versus 0.2 and 0.1 previously).

The sequences selected for the thermal hydraulic evaluation are common sequences for SBO conditions. The sequences examined various accident conditions. Conditions examined include how long secondary side heat removal is available into the accident progression. Other conditions analyzed are the size of the Reactor Coolant Pump Seal Loss of Coolant Accident (RCPSL) and when the RCPSL occurs. Also, scenarios considered whether the pressurizer relief valves are cycling or stuck in the open position. Inputs also considered what type of mitigation was available for the scenarios. Additionally, the analysis considered if the RCS was at high or low pressure and the time of RV failure.

The CCFP value developed included the contribution from the hydrogen combustion overpressure event following RV failure. In these simulations, the hydrogen igniters are turned off for the time period prior to RV failure. This allows hydrogen generated during core damage to accumulate in the containment. At the time of RV failure, the hydrogen igniters are turned on thus assuring that an ignition source is available. After the accident sequences were analyzed for the peak containment pressure, the containment failure probabilities were estimated from Table G-4 of the IPE report (Reference 27) which provides the containment capacity analysis.

The accident sequences were then binned into containment failure probability bins based on the peak pressures produced in the MAAP analyses. Recovery rules were also developed to ensure that each accident sequence received an appropriate CCFP. Most core damage accident sequences, even without igniters available, resulted in containment peak pressures below or only modestly higher than the containment HCLPF capacity. This is a strong indication that seismic LERF contributions from these accident sequences would be reduced in proportion to notably low CCFPs.

Large Early Release Frequency

Plant-specific thermal hydraulic analyses using MAAP version 5.01 were used to develop CCFPs to support the Internal Events LERF Probabilistic Risk Assessment update. The analysis was developed to use plant-specific CCFP values for various accident sequences instead of the generic CCFPs developed in NUREG/CR-6595 for two parameters that represent the highest potential for containment failure are:

1. CCFP with no igniters operating and the RCS at high pressure when RV failure occurs, and
2. CCFP with no igniters operating and the RCS at low pressure when RV failure occurs.

The sequences selected for the evaluation are common sequences for SBO conditions. As used in the description of these parameters, the RCS pressure is high if DCH is a potential threat and low when DCH is not a threat. Revision 1 to NUREG/CR-6595 was published in October of 2004. This revision included new estimates for the two CCFP parameters discussed earlier. The new recommended values for the two parameters are 1.0 and 0.97 (versus 0.2 and 0.1 previously). After the accident sequences were analyzed for the peak containment pressure, the containment failure probabilities were estimated from Table G-4 of the IPE report (Reference 27) which provides the containment capacity analysis. Attachment 1, Figure 2 shows the graph of the containment failure probabilities verses containment pressure.

The accident sequences were then binned into containment failure probability bins based on the peak pressures produced in the MAAP analyses. Most accident sequences even with igniters assumed to fail result in containment pressure demands below or only modestly higher than the containment HCLPF capacity (~56 psi): a strong indication that Seismic Large Early Release Frequency (SLERF) contributions from these accident sequences are reduced in proportion to notably low CCFPs. Taking credit for igniters maintaining hydrogen concentrations below flammability limits, SLERF contributions from these sequences are even more decreased.

In conclusion, LERF models have been developed for McGuire. Plant-specific peak pressure analyses were performed and integrated into the recent Large Early Release Frequency PRA models. The plant-specific analyses, using MAAP, determined peak pressures during many different accident sequences. The key findings include:

- Most accident sequences, including those without hydrogen igniters, result in containment pressure only slightly higher than the containment HCLPF capacity.
- The resulting CCFPs are lower than those provided in NUREG/CR-6595 for many accident scenarios and result in a lower overall LERF. This is especially true for SBO in which all power to the plant, including the hydrogen igniters, is lost.
- LERF contributions are lower in cases where the hydrogen igniters are available.

4.6 McGuire FLEX

On March 12, 2012 (Reference 33), the NRC issued Order EA-12-049, Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events. The order requires that licensees develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and SFP cooling capabilities following beyond-design-basis external events. A three-phase approach described in the order is a conceptual framework designed to address challenges to these safety functions using installed structures, systems, and components for a coping period until portable mitigating equipment can be used to address those challenges. The actions required by the order provide additional defense-in-depth and diversity for mitigating beyond design-basis external events. The McGuire FLEX strategies are described in the McGuire Final Integrated Plan (FIP) submitted in a letter dated December 7, 2015 (Reference 34), and summarized below.

Steam generator heat removal is achieved during Phase 1 and 2 via the TDAFWP with assured suction from buried Condenser Circulating Water system piping. Later stages of Phase 2 and 3 strategies entail steam generator cooling water make-up via a portable diesel powered FLEX pump with suction from the Standby Nuclear Service Water Pond and discharge aligned to new steam generator FLEX supply connections. The TDAFWP flow control valves and Main Steam Power-Operated Relief Valves are also required to provide steam generator heat-removal capability, their actuators powered by the FLEX Air Tanks via the blackout headers. The Phase 2 steam generator heat removal is achieved via the credited B.5.b connection (primary) or via the new FLEX mechanical connections located in the Auxiliary Building doghouses. The FLEX strategy with steam generators unavailable (i.e., refueling outage) relies on RCS feed-and-bleed for Phase 1 and 2 via the Residual Heat Removal system FLEX connections.

There are no required Phase 1 FLEX actions to maintain containment function. The primary Phase 2 FLEX strategy for containment entails repowering one train of hydrogen igniters and one train of Hydrogen Skimmer fans. Phase 3 entails engaging the Ice Condenser via repowering one train of Containment Air Return fans and two Lower Containment Ventilation Units, utilizing the National SAFER Response Center (NSRC)-supplied portable generators. Later in the Extended Loss of all AC Power (ELAP) event, the Residual Heat Removal system is aligned to maintain containment temperature.

Necessary attendant electrical components primarily entail 600 VAC essential motor control centers, vital batteries, equipment installed to support FLEX electrical connections, and monitoring instrumentation required for core cooling, reactor coolant inventory, and containment integrity. During the latter stages of Phase 3, the 4.16 kV switchgear is energized to support residual heat removal operation, as well as the containment ventilation fans.

The FLEX strategies provide a significant safety enhancement for mitigation of beyond design basis events, including seismic. The FLEX Program provides defense-in-depth with its equipment and strategies, reducing overall risk. By addressing ELAP events, associated SCDF dominant contributors are reduced, which improves the overall plant seismic risk. As discussed in Section 6.1 of this enclosure, the assessment will demonstrate seismic adequacy of the FLEX mitigating strategies for the GMRS from the reevaluated seismic hazard.

5.0 CATAWBA INFORMATION FOR SPRA RELIEF

This section discusses Catawba-specific seismic risk evaluations, site-specific CCFP analyses, and the FLEX mitigating strategies for beyond design-basis external events. Catawba responses to other NTTF seismic activities are also discussed.

5.1 Catawba Ground Motion Response Spectrum

The reevaluated GMRS to SSE comparison is shown in Attachment 1, Figure 3. From that figure, the design basis SSE exceeds the GMRS below approximately 5.5 Hz, and the GMRS begins to exceed the Catawba SSE above 5.5 Hz. In the high frequency range greater than 10 Hz, structural displacements in this frequency range are small and are considered non-damaging. The peak acceleration of the new GMRS is 0.75g at 30 Hz.

According to SPID, the area of concern is in the 1 to 10 Hz range. From Attachment 1, Figure 3. the GMRS-to-SSE ratio is 1.91 in the 1 to 10 Hz range. Installed plant equipment credited in the FLEX strategies was confirmed during the Expedited Seismic Evaluation Process (ESEP) to have adequate capacity to perform its FLEX mitigation function in the 1 to 10 Hz range -- refer to Section 5.3 for additional information regarding the ESEP.

5.2 Catawba NTTF 2.3 Seismic Walkdown

Seismic walkdowns were conducted in accordance with EPRI Report 1025286 (Reference 22) for each unit on representative seismic equipment types to provide reasonable assurance that seismic equipment configuration control has been maintained consistent with the current seismic licensing basis, including consideration of seismic interaction concerns and equipment degradation. The EPRI document provided instruction and procedures to perform seismic walkdowns as required by the 50.54(f) letter. The EPRI guidance covers selection of personnel, selection of a sample of structures, systems, and components that represent diversity of component types and assures inclusion of components from critical systems/functions; conduct of the walkdowns, evaluation of potentially adverse conditions against the plant seismic licensing basis; and reporting requirements. The guidance also included checklists to be used by the Seismic Walkdown Engineers in the performance of the seismic walkdowns.

The initial walkdown results were reported to the NRC on November 27, 2012 (Reference 35). A total of 40 PASC were identified by the seismic walkdowns. Also, one issue with anchor bolts was identified during the TI-188 walk-down inspection with the regional NRC inspector. All of the PASC were entered into the CAP. Engineering evaluations of the PASC concluded that the conditions were in conformance with the current seismic licensing bases. In some cases, work requests or CAP Actions were initiated to correct minor issues and/or to enhance field equipment clearances. The results of the walkdowns and licensing basis evaluations verified the adequacy of the Catawba monitoring and maintenance procedures with regard to maintaining the current seismic licensing basis.

In summary, all PASC identified as a result of the seismic walkdowns were entered into CAP.

5.3 Catawba Expedited Seismic Evaluation Process (ESEP)

The intent of the ESEP is to demonstrate seismic margin through a review of a subset of the plant equipment that can be relied upon to protect the reactor core following beyond design basis seismic events. The ESEP is implemented using the methodologies in EPRI Report 3002000704, Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic (Reference 6). Selected equipment with a HCLPF capacity in excess of the calculated Review Level Ground Motion (RLGM) are deemed to have adequate seismic capacity for the reevaluated seismic hazard in the 1 to 10 Hz frequency range.

The procedure for determining the RLGM for the ESEP is described in Section 4 of EPRI Report 3002000704. The RLGM is determined by multiplying the spectral acceleration values for the 5%-damped SSE horizontal ground response spectrum by a scale factor. The scale factor is the largest ratio of spectral accelerations between the 5%-damped GMRS and the 5%-damped SSE ground response spectrum at frequencies from 1 Hz to 10 Hz, but not to exceed 2.0. The ratio of the GMRS to the SSE over the 1 to 10 Hz frequency range is 1.91. Therefore, the RLGM is determined by multiplying the SSE ground response spectrum by 1.91.

The selection of equipment to be included on the ESEL was based on installed plant equipment credited in the FLEX strategies. The scope of "installed plant equipment" includes equipment relied upon for the FLEX strategies to sustain the critical functions of core cooling and containment. FLEX recovery actions are excluded from the ESEP scope per EPRI Report 3002000704.

The next step in the process demonstrated that ESEL items have sufficient seismic capacity to meet or exceed the demand characterized by the RLGM. The seismic capacity is characterized as the PGA for which there is a HCLPF. The PGA is associated with a specific spectral shape, in this case the 5%-damped RLGM spectral shape. The HCLPF capacity must be equal to or greater than the RLGM PGA. The criteria for seismic capacity determination are given in Section 5 of EPRI Report 3002000704.

Catawba completed the ESEP and submitted an initial report on December 31, 2014 (Reference 36) and a revised report on November 5, 2015 (Reference 37). All equipment evaluated for the ESEP was found to have adequate capacity for the Catawba RLGM.

5.4 Catawba Seismic Risk Evaluations and Insights

NRC issued Generic Letter 88-20 on November 23, 1988, requesting that all licensees perform an IPE to identify plant-specific vulnerabilities to severe accidents. At that time, Duke Energy began a program to update an earlier Catawba PRA study to take into account a number of modifications to the plant and to take advantage of plant-specific data and state-of-the-art methods. In September 1992 (Reference 38), Duke Energy submitted its response to the NRC which included a quantitative assessment of SCDF. The IPE submittal explained that the Catawba PRA is a full-scope Level 3 PRA with complete analysis of external events in addition to internal events. External events have been included in the Catawba PRA studies beginning with the original study.

In addition to the Catawba PRA studies, Catawba Unit 2 was selected for a trial assessment of the EPRI developed Seismic Margin Methodology, the methodology for assessing the ability of nuclear plants to withstand earthquakes beyond design basis. The assessment established that Catawba would survive earthquake loads up to approximately twice its design basis. This work is documented in EPRI NP-6359 (Reference 39).

On June 28, 1991, the NRC issued Generic Letter 88-20, Supplement 4, requesting that all licensees perform an IPEEE to identify plant-specific vulnerabilities to severe accidents solely due to external events. In June 1994, Duke submitted its response to the NRC (Reference 40) which included an updated assessment of SCDF as well as a qualitative assessment of containment performance. For its seismic assessment, Catawba used primarily a SPRA approach in addition to referencing the SMA approach previously noted. The SPRA analysis utilized the best data and analysis tools available at that time.

As indicated in Table 3.1 of NUREG-1407 (Reference 12), Catawba was placed in the 0.3g Focused Scope bin. As such, a median ground response spectrum anchored at 0.3g was used for the RLE for the Catawba site as recommended by NUREG-1407. For relay chatter events, Section 3.2.4.2 of the NUREG states that, for non-A-46 plants (e.g., Catawba), focused scope plants are required to locate and evaluate low seismic ruggedness relays or "bad actors" (as found in EPRI NP-7148-SL, Appendix E (Reference 29)).

The Catawba SPRAs were developed using EPRI's CAFTA software program to create a fault tree model to generate the seismic event cut sets. These were then combined with the mean plant seismicity curve and SSC fragilities into a Duke Energy in-house program, SEISM, which used Monte Carlo simulation to generate the final SCDF. The Catawba IPEEE SCDF was calculated to be 1.6E-05/yr. Several of the dominant accident sequences involve a loss of offsite power followed by a loss of both emergency diesel generators (i.e. SBO).

Because LERF models had not been developed for Catawba at the time of the IPE and IPEEE submittals, a qualitative analysis was performed for containment performance in response to a seismically induced core damage accident. This was accomplished by examining the containment structure fragility analysis and the containment isolation function.

The entire fragility curve for any mode of failure and its uncertainty can be expressed in terms of best estimate of the median ground acceleration capacity times the product of random variables representing inherent randomness and uncertainty. In estimating these fragility parameters, it is computationally attractive to work in terms of an intermediate random variable called factor of safety. The factor of safety is defined as the ratio of the ground acceleration capacity to the SSE acceleration used in design. The development of seismic safety factors associated with the SSE is based on consideration of several variables. The median of the overall factor of safety is the product of the median safety factors of all the variables. The variabilities of the individual variables also combine to determine that of the overall safety factor. The median factor of safety and its statistical variability for each SSC for the Catawba power plant were determined, based on results of existing dynamic models and associated response analyses and evaluations of structures and equipment, supplemented by some limited additional analyses. The resulting median capacities for several of the Catawba structures are greater than or equal to 2.0g. The reactor building, the steel containment vessel, and the containment internal structure fall into this category.

Similar to structural fragility, factors of safety and their variabilities are first developed for equipment capacity, earthquake duration, and equipment response. These three factors, along with the factor of safety on structural response, are then multiplied together to obtain an overall factor of safety to be used for the equipment item. Catawba equipment fragility descriptions are based on (1) plant-specific design reports, (2) qualification test reports, (3) generic test or analysis data, and (4) engineering judgment and past earthquake experience. Of resulting equipment fragility descriptions, there is an equipment category for those items that possess ground acceleration capacities greater than or equal to 2.0g and will not contribute to the overall plant risk. The ice condenser structure and hydrogen igniters fall into this category.

The seismic impact on containment isolation was also evaluated. Piping, valves and supports associated with penetrations which, if failed, could lead to significant release pathways, were determined by the fragility vendor to have median fragilities greater than 2g. The cabinets housing the equipment used to generate the containment isolation signals had a median fragility of 1.30g. Likewise, the respective panel boards and MCCs providing power to actuate the valve solenoids and motors were also evaluated for the beyond design basis in-structure levels and found to be adequate. The panel board fragility is 1.01g and the MCC fragility is 0.53g.

In addition, effects of chatter on relays within the containment isolation circuit were considered. A listing of the affected plant relays was compared against the listing of "bad actor" relays given in EPRI NP-7148-SL mentioned above. Since none of Catawba relays that would compromise safe shutdown functions qualified as bad actors, this was not a concern.

Thus, it was determined the containment structure and penetrations are seismically rugged, containment isolation would occur in response to a seismic induced core damage accident, and relays within the containment isolation circuit will function as designed.

Plant walkdowns were performed to support the development of the initial Catawba PRA which included external events. Walkdowns were also conducted to support the 1992 IPE submittal. Detailed walkdowns were also conducted for Unit 2 and for items common to both units for the trial plant application of EPRI NP-6359. Plant improvements were implemented as a result of this study. Modifications were made to:

- Add spacers and dummy batteries, and stiffen side rails on the diesel generator battery racks
- Relocate an instrument to avoid a potential seismic interaction with adjacent piping

As part of the IPEEE effort, extensive plant walkdowns were also conducted on both units consistent with the guidelines given in EPRI NP-6041 (Reference 30). The purpose of these walkdowns was to confirm the validity of the equipment fragility assessments, to review equipment with respect to seismic experience caveats, to verify the seismic adequacy of equipment anchorage and to identify any other seismic concerns. Portions of the walkdowns were conducted inside containment for both units, focusing on plant equipment and other containment performance issues. In addition to the seismic fragility validation, enhancements to the plant were recommended and subsequently implemented based upon the IPEEE walkdowns (including those for fire). These included:

- Replacing a service water valve
- Making a procedure enhancement by placing instructions in the pre-fire plan for one of the 4160V AC switchgear areas
- Routing cables for new instrument air compressors to create sufficient redundancy for fire
- Reinstalling missing door bolts on the Auxiliary Shutdown Panel NEMA 4 cabinets

Overall, no fundamental plant weaknesses or vulnerabilities were identified.

5.5 Catawba Conditional Containment Failure Probability (CCFP)

As part of the Catawba IPE, a containment capacity analysis was performed to establish a "probability of failure" distribution for the containment structures. It was expressed as a distribution of probability of containment failure versus internal containment pressure. When possible, direct comparisons were made to the original Catawba Containment Capacity Calculation, which is the basis for Update Final Safety Analysis Report (UFSAR) Ultimate Capacity assessments. When necessary, those calculations were modified to provide a basis consistent with other PRA work. Once all identified failure modes were investigated, a containment failure distribution was developed in a manner similar to that presented in NUREG/CR-1891, Reliability Analysis of Containment Strength (Reference 31). Insofar as is possible and practical, an assessment was made of all identified potential containment failure modes, failure locations, and failure sizes.

The cumulative CCFPs from the containment capacity analysis are recorded in Table G-7 of the IPE report (Reference 38). Attachment 1, Figure 4 shows the graph of those CCFPs verses containment pressure. The curve combines the pressure fragility from the containment vessel shell, penetrations and anchorage failure modes analyzed. As can be seen from Attachment 1, Figure 4, the median pressure capacity of the containment is 84.5 psi. The HCLPF capacity that corresponds to 1% failure probability of the containment is 55 psi, which is 3.7 times the containment design pressure of 15 psi.

Additionally, Duke Energy performed similar plant-specific analyses in support of the Catawba Significance Determination Process (SDP) evaluation. The sequences selected for the evaluation are common sequences for SBO conditions. The analyses were performed with MAAP version 4.0.7. The analysis included five thermal hydraulic cases in regard to containment pressure response in the event of a SBO event that results in reactor vessel failure. Again, two parameters that represents the highest potential for containment failure are:

1. CCFP with no igniters operating and the RCS at high pressure when RV failure occurs, and
2. CCFP with no igniters operating and the RCS at low pressure when RV failure occurs.

The scope of the study was not all inclusive, but is instructive. The CCFP provided is the contribution from the hydrogen combustion overpressure event following RV failure. This allows hydrogen generated during core damage to accumulate in the containment. At the time of RV failure the igniters are turned on thus assuring that an ignition source is available. The results are provided in Attachment 2, Table 2. The CCFP provided is the contribution from the hydrogen combustion overpressure event following RV failure. The value does not include any contributions to containment failure from non-overpressure containment failure modes (e.g., debris contact with the containment steel). Results show that various accident sequences can vary significantly and not all core damage sequences generate sufficient hydrogen to challenge containment integrity, with no igniters operating. In addition, the Catawba LERF model incorporates plant-specific CCFPs into the results. 5.6 Catawba FLEX

On March 12, 2012, the NRC issued Order EA-12-049, Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events. The order requires that licensees develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and SFP cooling capabilities following beyond-design-basis external events. A three-phase approach described in the order is a conceptual framework designed to address challenges to these safety functions using installed SSCs for a coping period until portable mitigating equipment can be used to address those challenges. The actions required by the order provide additional defense-in-depth and diversity for mitigating beyond design-basis external events.

In developing FLEX strategies for protecting the reactor core, spent fuel pool, and containment functions, Catawba employed a three phase approach: (1) initially cope with reliance on installed plant equipment, (2) transition to use of portable on-site equipment, and (3) achieve long term coping using equipment provided from off-site resources. Strategies used to achieve the required functions are described in the FIP for FLEX which was submitted to the NRC with a letter dated February 15, 2016 (Reference 41).

During Phase 1, the core cooling and heat removal function relies on automatically aligning Condenser Circulating Water inventory to provide a qualified supply of feedwater to the TDAFWP which supplies the steam generators. The TDAFWP CCW supply valve has been modified such that it will open automatically to accomplish this function. The steam generators remove heat via steam release through the power operated relief valves.

Upon transition to Phase 2, portable equipment is used to maintain and/or establish required FLEX functions. Water supply from the Standby Nuclear Service Water Pond is provided via portable diesel driven pumps, while electrical power is supplied to various components via portable diesel generators.

Phase 3 utilizes additional equipment from the National Safer Response Centers to provide for long term coping.

The FLEX Phase 2 equipment is stored in a facility designed to withstand all required events, including seismic. This facility is a 144' diameter dome designed to ASCE 7-10 requirements, as specified per NEI 12-06, Revision 0, Diverse and Flexible Coping Strategies (FLEX) Implementation Guide.

Haul paths have been analyzed for liquefaction with an acceptable factor of safety. Debris removal capabilities are included within the FLEX strategies. Staging areas are identified and controlled to ensure they remain accessible.

The FLEX strategies provide a significant safety enhancement for mitigation of beyond design basis events, including seismic. The FLEX Program provides defense-in-depth with its equipment and strategies, reducing overall risk. By addressing Extended Loss of all AC Power (ELAP) events, associated SCDF dominant contributors are reduced, which improves the overall plant seismic risk. As discussed in Section 6.1 of this report, the assessment will demonstrate seismic adequacy of the FLEX mitigating strategies for core cooling, SFP makeup, and containment function strategies for the GMRS from the reevaluated seismic hazard.

6.0 PROSPECTIVE EVALUATION INSIGHTS

Previous sections provided details on the established body of knowledge already in place. Section 6 will cover prospective insights that will be obtained from the Mitigating Strategies Assessment and the High Frequency Evaluations.

6.1 Mitigating Strategies Assessment (MSA) - Path 4

Revision 2 to NEI 12-06 (Reference 42) provides guidance for performing a MSA using the seismic reevaluated hazard. JLD-ISG-2012-01, Revision 1 (Reference 43) endorses the methodologies and guidance described in NEI 12-06, Revision 2.

The approach described in NEI 12-06, Revision 2, for the performance of assessments of the mitigating strategies under the reevaluated seismic and flooding hazards provide an appropriate methodology for licensees to address the reevaluated seismic hazard in a manner that aligns with the proposed Mitigation Beyond Design Basis Events (MBDBE) rulemaking. Specifically, Appendix H to NEI 12-06, Revision 2, discusses a method to assess the results of the seismic hazard reevaluations with respect to the guidance and strategies required by Order EA-12-049. The purpose of this appendix is to provide guidance for a Mitigating Strategies Assessment (MSA) of the impact of reevaluated seismic hazard information and for the modification of FLEX mitigating strategies, if necessary. The reevaluated seismic hazard information is referred to as the Mitigating Strategies Seismic Hazard Information (MSSHI).

Based on the fact that the GMRS for McGuire and Catawba in the 1 to 10 Hz frequency range is greater than the SSE but not more than 2 times, Path 4 will be used to perform an MSA of the impacts of the reevaluated seismic hazard (i.e. GMRS) on FLEX strategies. For Path 4, selected plant equipment relied upon in the FLEX strategies for McGuire and Catawba were previously evaluated under the ESEP up to 1.74 and 1.91 times the SSE, respectively.

The ESEP for McGuire and Catawba provided evaluations that demonstrated seismic adequacy for a single success path for core cooling and containment function strategies for a scaled SSE spectrum that bounded the GMRS in the 1 to 10 Hz range. The ESEP evaluations remain applicable for Path 4 since these evaluations directly addressed the most critical (1 to 10 Hz) part of the new seismic hazard using seismic responses from the scaling of the design basis analyses. The ESEP can therefore be used to demonstrate robustness of SSCs to withstand the reevaluated seismic hazard.

The scope of evaluated SSCs is determined following the guidance in the ESEP and adding the SSCs that were excluded from the ESEP associated with the primary success path. Additional SSC failure modes not addressed under the ESEP will also be evaluated. These failure modes are the seismic interactions that could potentially affect the FLEX strategies.

Successful completion of the MSA Path 4 for McGuire and Catawba will verify successful implementation of FLEX against the reevaluated seismic hazard. Since selected plant components were evaluated up to the GMRS-to-SSE ratio during ESEP, there is high confidence in successful completion of the Seismic MSA Path 4. The MSAs for McGuire and Catawba will be submitted by August 31, 2017.

6.2 High Frequency Evaluation

In the high frequency range (>10 Hz), the GMRS exceeds the SSE for McGuire and Catawba. High-frequency exceedances (>10 Hz) are less damaging to structures due to their small displacements. To support the high-frequency seismic evaluations, EPRI developed a High Frequency Program, which conducted high frequency seismic testing of a diverse set of typical plant control components. The test program used a common test protocol for three-axis high-frequency input motion and a common protocol for the monitoring of device state. The results of this test program were documented in EPRI Report 3002002997 (Reference 44).

High frequency evaluations for McGuire and Catawba will be performed in accordance with EPRI Report 3002004396, High Frequency Program: Application Guidance for Functional Confirmation and Fragility Evaluation (Reference 45). This report provides guidance for performing a high frequency confirmation including identification of the equipment scope to be evaluated, methods for estimating the component demand, evaluating the capacity-to-demand ratio and a method for estimating the vertical GMRS component is also developed.

Beyond design basis in-structure response spectra (ISRS) will be developed, based on the horizontal GMRS and a vertical GMRS computed using the new method provided in the EPRI Report. Horizontal and vertical ISRS will be developed using the structure scale factors documented in the EPRI Report. In cabinet response spectra will be generated using the cabinet scale factors from the EPRI report. Seismic capacity will be derived from several resources, including the EPRI high frequency program. Component adequacy will be evaluated using the HCLPF approach. For high-frequency sensitive components, such as electrical relays, the evaluation will ensure adequate seismic margin exists for those components.

Currently, for those plants performing SPRAs, the high frequency review will be performed as part of the SPRA consistent with the schedule set forth in the NRC letter dated May 9, 2014. The SPRA submittal dates for Catawba and McGuire are currently scheduled for September 30, 2019 and December 31, 2019, respectively. In the event NRC grants SPRA relief, the submittal schedule for McGuire and Catawba will be accelerated to provide the high frequency evaluation, consistent with Section 4.7 of EPRI Report 3002004396, by August 31, 2017.

7.0 SUMMARY

Duke Energy has identified supplemental information regarding seismic risk and capacity insights that supports a request for relief from the requirement to perform a Seismic Probabilistic Risk Assessment at McGuire and Catawba. A summary of that information is provided below:

Seismic Design Margin

McGuire and Catawba have many layers of conservatism built into their seismic design. These layers include safety factors applied to the SSC designs, exacting requirements from accepted engineering codes and standards, and specific, conservative requirements for the strength of materials used to build the plants. Together, these design and construction practices provide margin to failure, even at ground motions well above those associated with the Safe Shutdown Earthquake.

Seismic Risk Evaluations and Insights

Numerous seismic risk evaluations have been previously performed for McGuire and Catawba including the IPEEE, the NRC Safety Assessment for GI-199, and the recent EPRI fleet-wide risk assessment for the reevaluated seismic hazards (i.e., GMRS). All of these assessments have produced SCDF point estimates within the range of 1E-7/year to 1E-4/year, which is the range computed under GI-199. Specifically, the recent McGuire and Catawba SCDF point estimates based on the reevaluated seismic hazards, using the simple average method for Peak Ground Acceleration are 2.7E-5/year for McGuire and 2.8E-5/year for Catawba. This indicates that McGuire and Catawba have margin to withstand potential earthquakes exceeding their original design bases and, consistent with the conclusions from GI-199, the reevaluated seismic hazards are not a concern.

State-of-the-Art Reactor Consequences Analysis

In addition to the above, previous seismic risk analyses have identified insights consistent with the NRC technical report regarding the Sequoyah State-of-the-Art Reactor Consequence Analyses. Specifically:

- Loss of AC power is the principal risk contributor to severe reactor accidents
- The Turbine-Driven Auxiliary Feedwater Pump plays a vital role in extending core cooling and allowing additional time for accident mitigation

- The hydrogen igniter system is effective in averting early containment failure

In light of these insights, McGuire and Catawba have verified that the Turbine-Driven Auxiliary Feedwater Pumps and hydrogen igniters have adequate seismic margin. Furthermore, McGuire and Catawba have implemented Phase 2 FLEX strategies that will ensure the Turbine-Driven Auxiliary Feedwater Pumps provide core cooling and repower the hydrogen igniters when needed.

Conditional Containment Failure Probability

Although the site-specific GMRS at McGuire and Catawba exceeds the Safe Shutdown Earthquake in the 5 to 10 Hz range, there is confidence that Seismic Category 1 structures will remain functional for GMRS demands. This is based, in part, on containment capacity analyses which were performed to establish a "probability of failure" distribution for the containment structures in support of the McGuire and Catawba IPEEE. Moreover, MAAP was used to determine peak containment pressures and plant-specific and accident sequence-specific CCFPs. The McGuire and Catawba containment structures, containment penetrations/isolation valves, and containment response to external events were evaluated from several perspectives, including failure modes, fragilities, and relay chatter. Collectively, these analyses identified no significant impact on containment performance. The ultimate pressure capacities for the McGuire and Catawba containments were found to be notably higher than their design pressure. Furthermore, the analyses found that most accident sequences, including those without hydrogen igniters, result in containment pressure only slightly higher than the containment HCLPF capacity. This results in lower failure probabilities for the majority of accident sequences than previously thought and is instrumental in effectively mitigating severe accident scenarios initiated by earthquakes.

NTTF 2.3 Seismic Walkdowns

In response to NTTF 2.3, McGuire and Catawba conducted seismic walkdowns on representative seismic equipment types to provide reasonable assurance that seismic equipment configuration control has been maintained, including consideration of seismic interaction concerns and equipment degradation. These walkdowns confirmed that the evaluated equipment is being maintained consistent with the seismic licensing basis.

Expedited Seismic Evaluation Process

McGuire and Catawba utilized the Expedited Seismic Evaluation Process to confirm that a subset of plant equipment that can be relied upon to provide protection from beyond design basis events has sufficient seismic capacity to meet or exceed the demand Review Level Ground Motion. All equipment evaluated under the Expedited Seismic Evaluation Process was confirmed to have adequate capacity for the Review Level Ground Motion in the 1 to 10 Hz frequency range.

FLEX Mitigating Strategies

McGuire and Catawba have implemented FLEX Order EA-12-049 which requires licensees to develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and spent fuel pool cooling. The FLEX mitigating strategies utilize a three-phased approach which increases defense-in-depth and diversity for beyond-design-basis scenarios, including seismic, that involve an extended loss of AC power and loss of normal access to the ultimate heat sink occurring simultaneously at all units on a site. Furthermore, by addressing extended loss of AC power events, associated SCDF dominant contributors are reduced, which improves the overall plant seismic risk.

Spent Fuel Pool Evaluations

McGuire and Catawba utilized the guidance in EPRI Report 3002007148 to evaluate their spent fuel pools to asses structural and non-structural elements, seismic-induced sloshing, and water losses due to heat-up and boil-off. These evaluations confirmed that the spent fuel pools are sufficiently robust to withstand the reevaluated GMRS hazard levels, and that spent fuel stored in the pools is adequately protected.

High Frequency Evaluations and Mitigating Strategies Assessment

McGuire and Catawba have sufficient margin to bound the site reevaluated seismic hazards in the lower frequency range below 10 Hz. Thus, the focus for new knowledge is in the areas of High Frequency Evaluations and Mitigating Strategies Assessments.

In the high frequency range (>10 Hz), the GMRS at McGuire and Catawba exceeds the Safe Shutdown Earthquake. Thus, a limited scope high frequency evaluation will be performed to provide confirmation that structures, systems, and components that may be affected by high-frequency ground motion will maintain their functions important to safety.

The Mitigating Strategies Assessments will verify implementation of FLEX against the reevaluated seismic hazard. Since selected plant components were evaluated up to the Review Level Ground Motion during the Expedited Seismic Evaluation Process, there is high confidence that McGuire and Catawba will successfully complete the seismic Path 4 Mitigating Strategies Assessments.

Schedule

If the NRC grants McGuire and Catawba SPRA relief, then both sites will submit the High Frequency Evaluations and the Mitigating Strategies Assessments by August 31, 2017. This proposed path forward allows for earlier closure of NTTF Recommendation 2.1: Seismic for McGuire and Catawba.

8.0 CONCLUSION

Duke Energy has evaluated available relevant information and concludes that McGuire and Catawba are low to moderate risk sites for seismic hazards and that performance of a Seismic Probabilistic Risk Assessment will not provide significant additional seismic risk insights.

9.0 REFERENCES

1. NRC Letter, Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, dated March 12, 2012
2. NRC letter, Screening and Prioritization Results Regarding Information Pursuant to Tile 10 of the Code of Federal Regulations 50.54(f) Regarding Seismic Hazard Re-Evaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident, dated May 9, 2014
3. NRC letter, 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, dated October 27, 2015
4. NRC Letter, Electric Power Research Institute Final Draft Report XXXXXX, Seismic Evaluation Guidance: Augmented Approach for the Resolution Fukushima of Near-Term Task Force Recommendation 2.1: Seismic, as an Acceptable Alternative to the March 12, 2012, Information Request for Seismic Reevaluations, dated May 7, 2013
5. EPRI Report 1025287, Seismic Evaluation Guidance, Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic, dated February 2013
6. EPRI Report 3002000704, Seismic Evaluation Guidance Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1 – Seismic, dated May 2013
7. Duke Energy letter, Seismic Hazard and Screening Report (CEUS Sites), Response to NRC 10 CFR 50.54(f) Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, dated March 20, 2014,
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11. NRC Generic Letter 88-20, Supplement 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10CFR50.54(f), dated June, 28, 1991

12. NUREG-1407; Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities; June 1991
13. NRC Memo, Identification of a Generic Seismic Issue, Dated May 26, 2005
14. NRC Draft Technical Report, State-of-the-Art Reactor Consequence Analyses (SOARCA) Project Sequoyah Integrated Deterministic and Uncertainty Analyses
15. SECY-16-0041, Closure of Fukushima Tier 3 Recommendation Related to Containment Vents, Hydrogen Control and Enhanced Instrumentation, dated March 31, 2016
16. NEI Letter, transmits EPRI Report 3002007148 for NRC endorsement, dated February 23, 2016
17. EPRI Report 3002007148, Seismic Evaluation Guidance Spent Fuel Pool Integrity Evaluation, February 2016
18. Duke Energy letter, Spent Fuel Pool Evaluation Supplemental Report, Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, dated July 20, 2016
19. Duke Energy letter, Spent Fuel Pool Evaluation Supplemental Report, Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, dated August 18, 2016
20. NRC letter, Catawba Nuclear Station Units 1 and 2 - Staff review of Spent Fuel Pool Evaluation associated with Reevaluated Seismic Hazard Implementing Near-Term Task Force Recommendation 2.1, dated August 11, 2016
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22. EPRI Report 1025286, Seismic Walkdown Guidance For Resolution of Fukushima Near-Term Task Force Recommendation 2.3: Seismic, dated May 2012
23. Duke Energy letter, Seismic Walkdown Information Requested by NRC Letter, Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50. 54(f) Regarding Recommendations 2.1,2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident; dated November 26, 2012
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34. Duke Energy letter, Final Notification of Full Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design Basis External Events and with Order EA-1 2-051, Order to Modify Licenses With Regard To Reliable Spent Fuel Pool Instrumentation for McGuire Nuclear Station, dated December 7, 2015
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37. Duke Energy letter, Errata Letter for 10 CFR 50.54(f) Submittal of Expedited Seismic Evaluation Process (ESEP) Report, Revision 2, dated November 5, 2015
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41. Duke Energy letter, Final Notification of Full Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies

for Beyond Design Basis External Events and with Order EA-1 2-051, Order to Modify Licenses With Regard To Reliable Spent Fuel Pool Instrumentation for Catawba Nuclear Station, dated February 15, 2016,

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ATTACHMENT 1

FIGURES

Figure 1, McGuire Nuclear Station SSE Versus GMRS

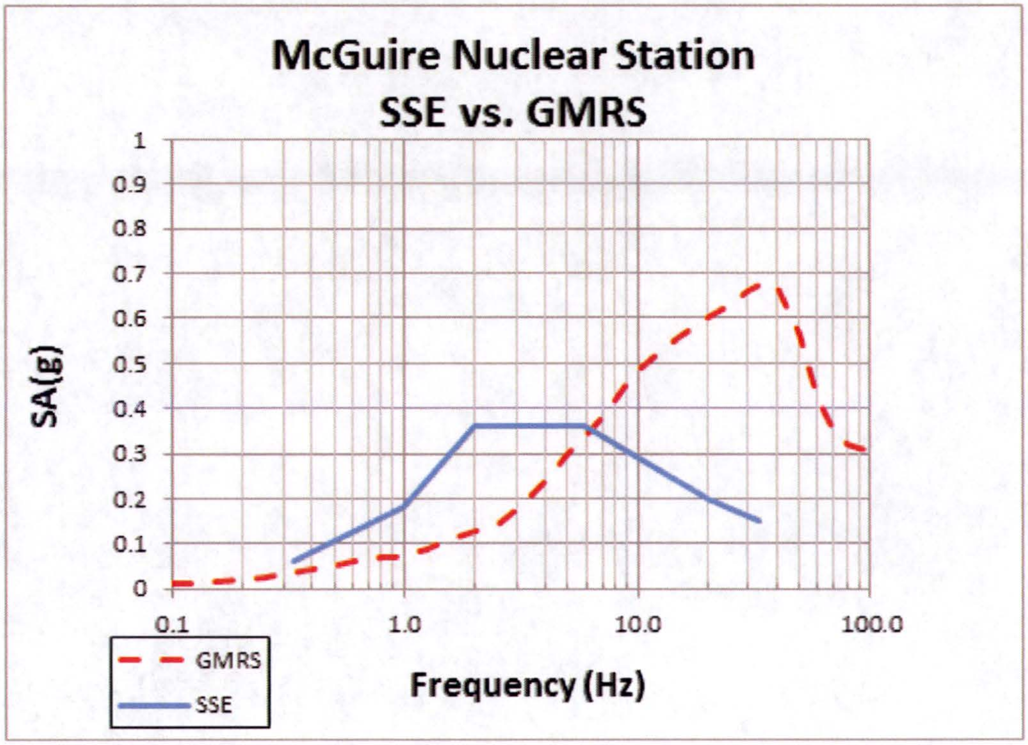


Figure 2, McGuire Containment Failure Probabilities Versus Containment Pressure

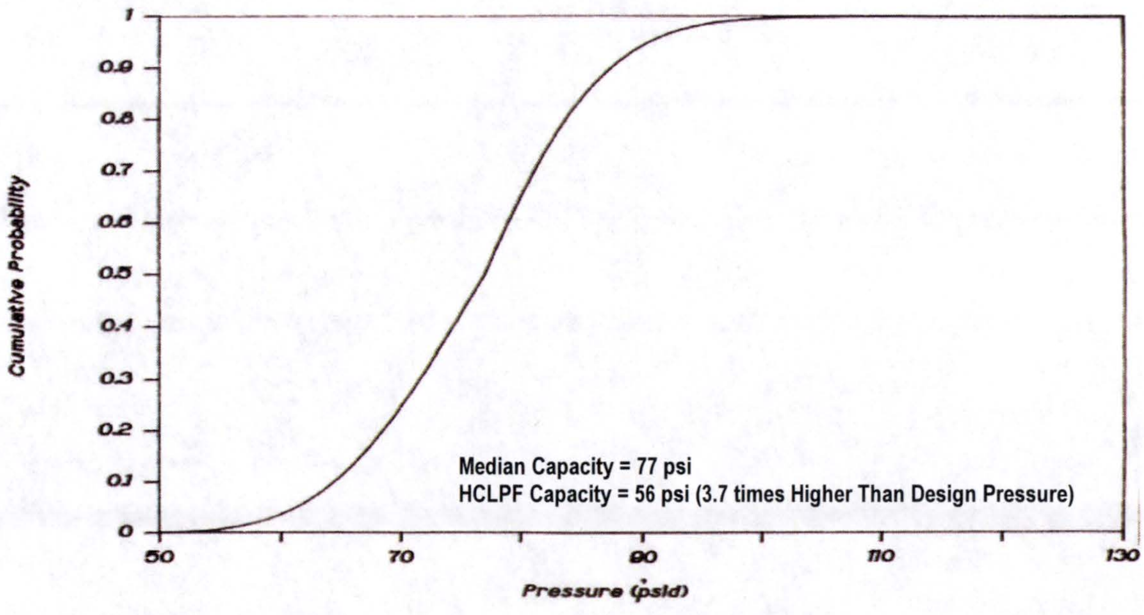


Figure 3, Catawba Nuclear Station SSE Versus. GMRS

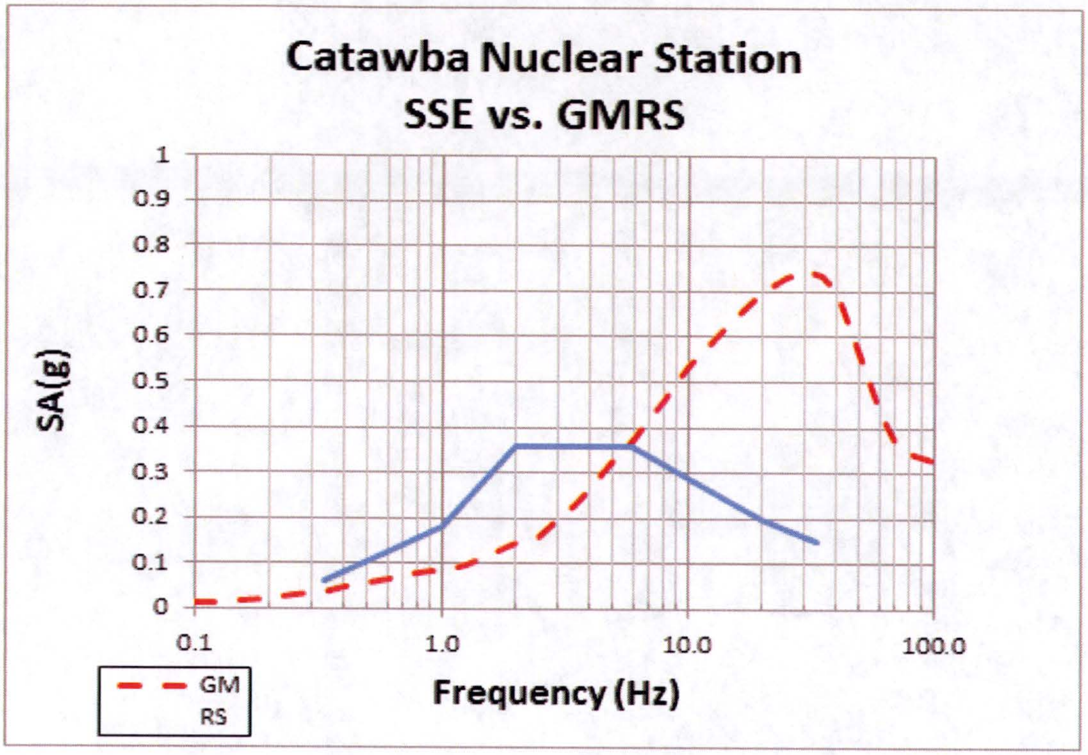
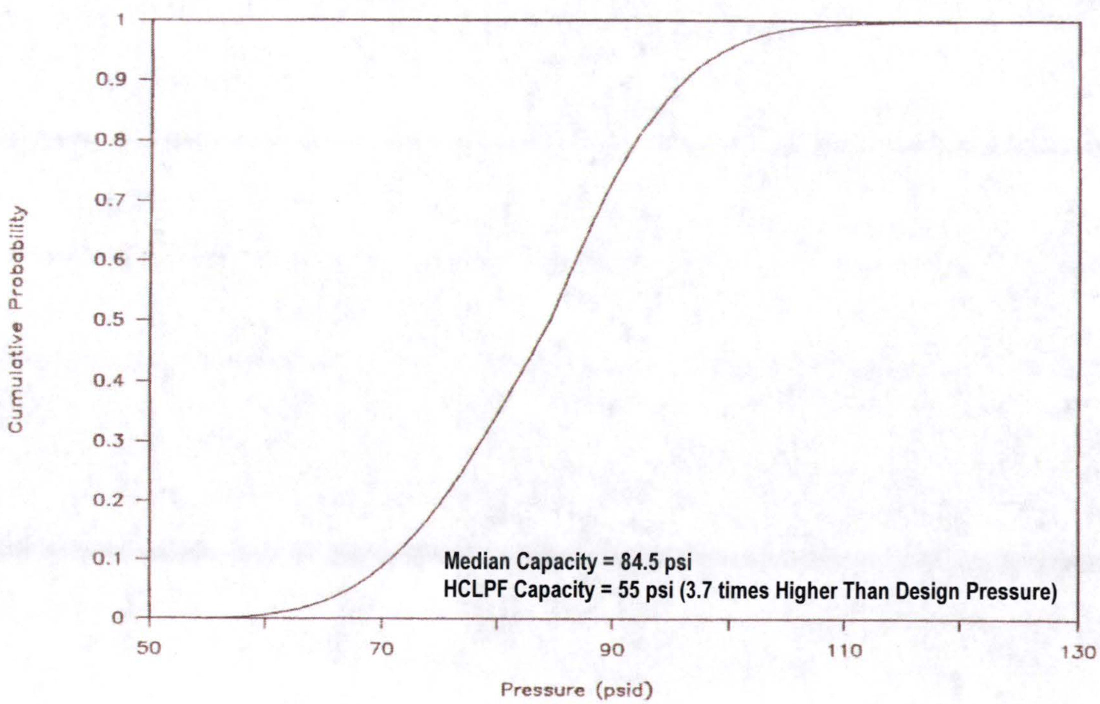


Figure 4, Catawba Containment Failure Probabilities Versus Containment Pressure



ATTACHMENT 2

TABLES

Table 1, Historical Range of SCDF for McGuire and Catawba

	Input	Method [1]	SCDF Catawba	SCDF McGuire
IPEEE/SPAR	NUREG 0098	PGA [2]	1.6E-05	1.1E-05
GI-199 SA	2008 USGS	Simple Average	2.7E-05	2.2E-05
		IPEEE Weighted Avg.	2.6E-05	2.0E-05
		Weakest Link Model	3.7E-05	3.1E-05
	1989 EPRI	Simple Average	1.7E-05	1.5E-05
		IPEEE Weighted Avg.	1.5E-05	1.3E-05
		Weakest Link Model	3.0E-05	2.8E-05
	1994 LLNL	Simple Average	2.8E-05	2.8E-05
		IPEEE Weighted Avg.	2.5E-05	2.5E-05
		Weakest Link Model	4.3E-05	4.7E-05
Range	All	All - Lower Bound	1.5E-05	1.1E-05
		All - Upper Bound	4.3E-05	4.7E-05
Range	All	Simple Average Lower Bound	1.7E-05	1.5E-05
		Simple Average Upper Bound	2.8E-05	2.8E-05
CEUS GMRS	EPRI 2012	Simple Average [3]	2.8E-05	2.7E-05

[1] PGA Controls

[2] PGA hazard was typically used for these risk estimates convolved with fragilities referenced to the PGA

[3] EPRI used simple average of risks generated from the four frequencies as the comparison for the fleet risk

Table 2, Catawba CCFP Contributions for Various SBO Sequences

Sequence (Catawba)	% Clad Reacted In Vessel	Peak Pressure (psia)	CCFP
SBO event with cycling safety relief valves with a start failure of the turbine driven auxiliary feedwater pump, RCS at DCH relevant pressure	43.11	58.4	0.00
SBO event with cycling safety relief valves with a start failure of the turbine driven auxiliary feedwater pump, RCS at non DCH relevant pressure due to hot leg creep rupture	43.40	66.4	0.00
SBO event with cycling safety relief valves with a 12 hour available run time for the turbine driven auxiliary feedwater pump, RCS at DCH relevant pressure	43.93	60.6	0.00
SBO event with cycling safety relief valves with a 12 hour available run time for the turbine driven auxiliary feedwater pump, RCS at non DCH relevant pressure due to hot leg creep rupture	60.26	106	0.78
SBO event with a 250 gpm/pump reactor coolant pump seal LOCA at time equals 0. Secondary side heat removal (SSHR) via the turbine driven auxiliary feedwater pump is available during the entire event, RCS at DCH relevant pressure	29.64	47.3	0.00