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RA-19-0301 10 CFR 50.90

July 31, 2019

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
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Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Station (ONS), Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
Renewed Facility Operating License Nos. DPR-38, DPR-47, and DPR-55

Proposed Amendment to the Renewed Facility Operating Licenses Regarding Revisions to the Updated Final Safety Analysis Report Sections Associated with the Oconee Tornado Licensing Basis – Responses to Request for Additional Information

References:

1. Letter to the U. S. Nuclear Regulatory Commission from J. Ed Burchfield, Jr., Vice President, Oconee Nuclear Station, Duke Energy Carolinas, LLC, "Proposed Amendment to the Renewed Facility Operating Licenses Regarding Revisions to the Updated Final Safety Analysis Report Sections Associated with the Oconee Tornado Licensing Basis," dated September 14, 2018 (ADAMS Accession No. ML18264A018).

Pursuant to 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke Energy) submitted a License Amendment Request (LAR) which proposes to revise the ONS Updated Final Safety Analysis Report (UFSAR) regarding the tornado licensing basis (LB) on September 14, 2018 (Reference 1). The new licensing basis would allow the Standby Shutdown Facility (SSF) to mitigate a tornado with the assumed initial conditions of loss of all alternating current power to all units with significant tornado damage to one unit, the use of tornado missile probabilistic methodology, and the elimination of the spent fuel pool to high pressure injection flow path for reactor coolant makeup from the licensing bases.

On February 11, 2019, the U.S. Nuclear Regulatory Commission (NRC) began an audit to support its review of the LAR, as discussed in the NRC staff's audit plan dated February 8, 2019 (ADAMS Accession No. ML19037A005). Based on its review of the application and its audit, the NRC issued requests for additional information (RAIs) on June 28, 2019 (ADAMS Accession No. ML19183A483), in order to complete its review.

Enclosure 2 provides the responses to the RAIs which includes information that is proprietary to Duke Energy. The proprietary information is identified by brackets and is annotated as such. The annotated information has substantial commercial value that provides a competitive advantage. Attachment 1 contains an Affidavit attesting to the proprietary nature of the information in Enclosure 2. Enclosure 1 contains a non-proprietary [redacted] version of this content. Attachment 2 contains the revised marked-up UFSAR pages. No changes to Technical Specifications are proposed.

A053

Enclosure 2 of this letter contains proprietary information. Withhold from Public Disclosure Under 10 CFR 2.390. Upon removal of Enclosure 2, this letter is uncontrolled.

NRR

The responses to the RAIs have been reviewed and determined to not affect the conclusions of the No Significant Hazards Consideration provided in the LAR dated September 14, 2018 (Reference 1).

Attachment 1 of the LAR (Reference 1) provided a schedule for completion of the SSF modifications supporting the new tornado mitigation strategy. The schedule submitted with the LAR was based on the anticipated date of issuance of the Safety Evaluation Report (SER) in the second quarter of 2020. It is requested that the schedule date for the SSF instrumentation modifications be revised to 1EC33 (Fall 2024), 2EC32 (Fall 2025), and 3EC33 (Spring 2026). A request for changes to the other modification schedules is not deemed necessary. The addition and upgrade of the SSF instrumentation requires significant work within containment which is planned to span over multiple outages. Based on the outage related work, the project has started for the Unit 1 SSF instrumentation modifications. Unit 2 and Unit 3 SSF instrumentation modifications would follow with completion in Fall 2025 and Spring 2026, respectively.

Inquiries on this proposed amendment request should be directed to Timothy D. Brown, ONS Regulatory Projects Group, at (864) 873-3952.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 31, 2019.

Sincerely,

J Ed Burchfield

J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Enclosure 1:	Responses to Requests for Additional Information [Non-Proprietary]
Enclosure 2:	Responses to Requests for Additional Information [Proprietary]
Attachment 1	Duke Energy Affidavit
Attachment 2	UFSAR Marked-Up Pages

cc w/enclosure and attachments:

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ATTACHMENT 1

Duke Energy Affidavit

Duke Energy Affidavit
Attachment 1

AFFIDAVIT of Steve Snider

1. I am Vice President of Nuclear Engineering, Duke Energy Carolinas, and as such have the responsibility of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear plant licensing and am authorized to apply for its withholding on behalf of Duke Energy.
2. I am making this affidavit in conformance with the provisions of 10 CFR 2.390 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke Energy's application for withholding which accompanies this affidavit.
3. I have knowledge of the criteria used by Duke Energy in designating information as proprietary or confidential. I am familiar with the Duke Energy information contained in the response to question 8 of the Request for Additional Information (RAI) by the Office of Nuclear Reactor Regulation for Oconee License Amendment request 2018-02 which proposes to update the Updated Final Safety Analysis Report (UFSAR) regarding the Tornado licensing basis.
4. Pursuant to the provisions of paragraph (b)(4) of 10 CFR 2.390, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by Duke Energy and has been held in confidence by Duke Energy and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke Energy. Information is held in confidence if it falls in one or more of the following categories.

      (a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by a vendor or consultant, without a license from Duke Energy, would constitute a competitive economic advantage to that vendor or consultant.

      (b) The information requested to be withheld consist of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage for example by requiring the vendor or consultant to perform test measurements, and process and analyze the measured test data.

      (c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation assurance of quality or licensing of a similar product.

      (d) The information requested to be withheld reveals cost or price information, production capacities, budget levels or commercial strategies of Duke Energy or its customers or suppliers.

(e) The information requested to be withheld reveals aspects of the Duke Energy funded (either wholly or as part of a consortium) development plans or programs of commercial value to Duke Energy.

(f) The information requested to be withheld consists of patentable ideas.

The information in this submittal is held in confidence for the reasons set forth in paragraphs 4(ii)(a) and 4(ii)(c) above. Rationale for this declaration is the use of this information by Duke Energy provides a competitive advantage to Duke Energy over vendors and consultants, its public disclosure would diminish the information's marketability, and its use by a vendor or consultant would reduce their expenses to duplicate similar information. The information consists of analysis methodology details that provides a competitive advantage to Duke Energy.

(iii) The information was transmitted to the NRC in confidence and under the provisions of 10 CFR 2.390, it is to be received in confidence by the NRC.

(iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in the RAI response is that which is marked by brackets in the response to RAI 8. This information is consistent with marked proprietary information in the NRC-approved Duke Energy methodology report DPC-NE-3003-PA. This information enables Duke Energy to:

(a) Support license amendment requests for its Oconee reactors.

(b) Perform transient and accident analysis calculations for Oconee.

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke Energy.

(a) Duke Energy uses this information to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) Duke Energy can sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke Energy.

5. Public disclosure of this information is likely to cause harm to Duke Energy because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring a commensurate expense or allowing Duke Energy to recoup a portion of its expenditures or benefit from the sale of the information.

Duke Energy Affidavit
Attachment 1

Steve Snider affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

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

Executed on July 18, 2019.

Steve Snider

ATTACHMENT 2

UFSAR MARKED-UP PAGES

UFSAR Marked-Up Pages
Attachment 2

n. Reactor Building penetrations and piping through isolation valves.

o. Siphon Seal Water System.

p. Essential Siphon Vacuum System.

q. Electric power for above.

r. Nitrogen supply to the EFW control valves FDW-315 and FDW-316.

Information relating to the seismic design of SSF systems and components is contained in Section 9.6.4.1 and 9.6.4.3. Information relating to the seismic design of the PSW System and its components are contained in Section 9.7.

4. Tornado

Add Insert 1

~~The Reactor Coolant System will not be damaged by a tornado. A loss of Reactor Coolant Pump (RCP) seal integrity was not postulated as part of the tornado design basis. Capability is provided to shutdown safely all three units.~~

~~The Reactor Coolant System, by virtue of its location within the Reactor Building, is protected from tornado damage. A sufficient supply of secondary side cooling water for safe shutdown is assured by Protected Service Water Pumps located in the Auxiliary Building and taking suction from Oconee 2 CCW intake piping. Redundant and diverse sources of secondary makeup water are credited for tornado mitigation. These include: 1) the other units' EFW Systems, 2) the PSW pumps, and 3) the SSF ASW pump.~~

~~Protected or physically separated lines are used to supply cooling water to each steam generator. The sources of power to the PSW pumps are the Keowee Hydro Station and the Central Tie Switchyard via a 100 kV transmission line to a 100/13.8 kV substation.~~

An external source of cooling water is not immediately required due to the large quantities of water stored underground in the intake and discharge CCW piping. The stored volume of water in the intake and discharge lines below elevation 791ft would provide sufficient cooling water for all three units for at least 30 days after trip of the three reactors.

~~Although not fully protected from tornadoes, the following sources provide reasonable assurance that a sufficient supply of primary side makeup water is available during a tornado initiated loss of offsite power.~~

~~a. The SSF Reactor Coolant Makeup Pump can take suction from the Spent Fuel Pool. The pump can be supplied power from the SSF Diesel.~~

~~b. A High Pressure Injection Pump can take suction from either the Borated Water Storage Tank or the Spent Fuel Pool. Either the "A" or "B" High Pressure Injection Pump can be powered from the PSW Switchgear.~~

~~Protection against tornado is an Oconee design criteria, similar to the criteria to protect against earthquakes, wind, snow, or other natural phenomena described in UFSAR section 3.1.2. A specific occurrence of these phenomena is not postulated, nor is all equipment that would be used to bring the plant to safe shutdown comprehensively listed. The statement, "Capability is provided to shutdown safely all three units" is intended to be a qualitative assessment that, after a tornado, normal shutdown systems will remain available or alternate systems will be available to allow shutdown of the plant. It was not intended to imply that specific systems should be tornado proof. As part of the original FSAR development, specific accident analyses were not performed to prove this judgement, nor were they requested by the NRC. Subsequent probabilistic studies have confirmed that the original qualitative assessments were correct. The~~

~~risk of not being able to achieve safe shutdown after a tornado is sufficiently small that additional protection is not required.~~

~~In addition, there was considerable correspondence between Duke and NRC in the years post-TMI discussing Oconee's ability to survive tornado generated missiles. Based upon the probability of failure of the EFW and Station ASW systems combined with the protection against tornado missiles afforded by the SSF ASW system, the NRC concluded that the secondary side decay heat removal function complied with the criterion for protection against tornadoes.~~

3.2.2.1 System Classifications

Plant piping systems, or portions of systems, are classified according to their function in meeting design objectives. The systems are further segregated depending on the nature of the contained fluid. For those systems which normally contain radioactive fluids or gases, the Nuclear Power Piping Code, USAS B31.7 and Power Piping Code USAS, B31.1.0 are used to define material, fabrication, and inspection requirements.

Diagrams for each system are included in the FSAR sections where each system is described.

Fabrication and erection of piping, fittings, and valves are in accordance with the rules for their respective classes. Welds between classes of systems (Class I to II, I to III, or II to III) are performed and inspected in accordance with the rules for the higher class. This preceding sentence does not apply to valves where the class break has been determined to occur at the valve seat, and to pipe with 1" nominal diameter and less.

In-line instrument components such as turbine meters, flow nozzle assemblies, and control valves, etc. are classified with their associated piping unless their penetration area is equal to or less than that of a I inch i.d. pipe of appropriate schedule for the system design temperature and pressure, in which case they are placed in Class III. Definitions of the three classes are listed below:

Class I

This class is limited to the Reactor Coolant System (RCS) and Reactor Coolant Branch lines, as described herein. The Reactor Coolant Branch lines include connecting piping out to and including the first isolation valve. This section of piping is Class I in material, fabrication, erection, and supports and restraints. A Class I analysis of the piping to the first isolation valve has been completed for the following systems:

1. High Pressure Injection (Emergency Injection)
2. High Pressure Injection (Normal Injection)
3. High Pressure Injection (Letdown)
4. Low Pressure Injection (Decay Heat Removal Drop-line)
5. Low Pressure Injection (Core Flood)
6. Reactor Coolant Drain Lines
7. Pressurizer Spray
8. Pressurizer Relief Valve Nozzles

Modifications that affect the Reactor Coolant System and the Class I portion of the branch lines must demonstrate that the impact on the Class I piping is acceptable. The impact may be assessed by performing a Class I analysis or by other conservative techniques to assure Class I allowable limits are not exceeded. Isolation valves can be either stop, relief, or check valves. Piping 1 inch and less is excluded from Class I.

Class II

UFSAR Section 3.2.2, 4. Tornado

Insert 1:

The Reactor Coolant System, by virtue of its location within the Reactor Building, will not be damaged by a tornado. Capability is provided to shutdown safely all three units. Tornado is not considered a design basis event (DBE) or transient for Oconee. Protection against tornado is an Oconee design criterion, similar to the criteria to protect against earthquakes, wind, snow, or other natural phenomena described in UFSAR Section 3.1.2. A specific occurrence of these phenomena is not postulated.

The statement, "Capability is provided to shutdown safely all three units" was intended to be a qualitative assessment that, after a tornado, normal shutdown systems would remain available or alternate systems would be available to allow shutdown of the plant. It was not intended to imply that specific systems should be tornado proof. As part of the original FSAR development, specific accident analyses were not performed to prove this judgment, nor were they requested by the NRC. Subsequent probabilistic studies confirmed that the original qualitative assessments were correct. The risk of not being able to achieve safe shutdown after a tornado was sufficiently small that additional protection was not required.

In an effort to ensure the risk of not being able to achieve safe shutdown after a tornado is maintained sufficiently small, design criteria are applied to the SSF through physical protection and TORMIS to establish its capability to mitigate a tornado. The overall tornado mitigation strategy utilizes the deterministically tornado protected SSF for secondary side decay heat removal (SSDHR) and reactor coolant makeup (RCMU) following a postulated loss of all normal and emergency systems which usually provide these safety functions.

Successful mitigation of a tornado condition at Oconee is defined in UFSAR Section 9.6, SSF. The SSF and its related equipment have been physically protected to meet tornado requirements or have been evaluated using TORMIS.

In addition to the SSF deterministic capability to mitigate a tornado, the inherent plant design of system redundancy, independence, and diversity is maintained for reasonable assurance that sufficient primary and secondary makeup is available following a tornado. Though all features of the inherent plant design are not tornado proof, their collective capabilities result in high availability and reliability to ensure that system functions are not reliant on any single feature of the design. As such, the high availability and reliability provided by the inherent design of the plant which includes redundancy, independence, and diversity ensures defense in depth is maintained if the SSF and related components become unavailable either prior to or during a tornado. The sources of secondary makeup include: 1) the Emergency Feedwater system including the capability to cross connect from another unit, 2) the PSW system, and 3) the SSF ASW system capable of being powered by the SSF diesel. The sources of primary makeup include: 1) the SSF Reactor Coolant Makeup Pump supplied from the Spent Fuel Pool and capable being powered from the SSF diesel and 2) A High Pressure Injection (HPI) pump supplied from the Borated Water Storage Tank. Note that in addition to their normal and emergency power sources, the "A" and "B" HPI pumps can be powered from the PSW switchgear.

The revised tornado mitigation strategies will be implemented when the SSF letdown line, SSF control room QA-1 instrumentation upgrade, and SSF diesel fuel oil tank fill/vent missile protection conforming modifications are completed.

3.5.1.3 Missiles Generated by Natural Phenomena

For an analysis of missiles created by a tornado having maximum wind speeds of 300 mph, two missiles are considered. One is a missile equivalent to a 12 foot long piece of wood 8 inches in diameter traveling end on at a speed of 250 mph. The second is a 2000 pound automobile with a minimum impact area of 20 square feet traveling at a speed of 100 mph.

For the wood missile, calculations based on energy principle indicate that because the impact pressure exceeds the ultimate compressive strength of wood by a factor of about four, the wood would crush due to impact. However, this could cause a secondary source of missiles if the impact force is sufficiently large to cause spalling of the free (inside) face. The compressive shock wave which propagates inward from the impact area generates a tensile pulse, if it is large enough, will cause spalling of concrete as it moves back from the free (inside) surface. This spalled piece moves off with some velocity due to energy trapped in the material. Successive pieces will spall until a plane is reached where the tensile pulse becomes smaller than the tensile strength of concrete. From the effects of impact of the 8 inch diameter by 12 foot long wood missile, this plane in a conventionally reinforced concrete section would be located approximately 3 inches from the free (inside) surface. However, since the Reactor Building is prestressed, there will be residual compression in the free face, as the tensile pulse moves out and spalling will not occur. Calculations indicate that in the impact area a 2 inch or 3 inch deep crushing of concrete should be expected due to excessive bearing stress due to impact.

For the automobile missile, using the same methods as in the turbine failure analysis, the calculated depth of penetration is ¼ inch and for all practical purposes the effect of impact on the Reactor Building is negligible.

From the above, it can be seen that the tornado generated missiles neither penetrate the Reactor Building wall nor endanger the structural integrity of the Reactor Building or any components of the Reactor Coolant System.

Additional tornado missile requirements were subsequently imposed by NRC post-TMI on Emergency Feedwater Systems. ONS met these requirements based upon the probability of failure of the EFW and station ASW Systems combined with the protection against tornado missiles afforded the SSF ASW System. Subsequently, PSW replaced station ASW relative to this function. See UFSAR Sections 3.2.2 and 10.4.7.3.6 for additional information.

Revision 1 to Regulatory Guide 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," was released in March 2007. Revision 1 to Regulatory Guide 1.76 was ~~incorporated into the plant's licensing basis in the 4th quarter of 2007.~~ The design of new systems (and their associated components and/or structures) that are required to resist tornado loadings will conform to the tornado wind, differential pressure, and missile criteria specified in Regulatory Guide 1.76, Revision 1 or be evaluated by TORMIS.

Add Insert 2

3.5.2 Barrier Design Procedures

The Reactor Building and Engineered Safeguards Systems components are protected by barriers from all credible missiles which might be generated from the primary system. Local yielding or erosion of barriers is permissible due to jet or missile impact provided there is no general failure.

The final design of missile barrier and equipment support structures inside the Reactor Building is reviewed to assure that they can withstand applicable pressure loads, jet forces, pipe

INSERT 2

3.5.1.3.1 TORMIS Methodology

The TORMIS methodology provides an approach to demonstrate adequate protection for existing SSCs that were originally required to be protected from tornado missiles in accordance with the plant design basis but that are not adequately protected due to some oversight. The approved methodology does not allow TORMIS analysis to be used to temporarily or permanently eliminate existing barriers that are credited for providing tornado missile protection.

The TORMIS acceptance criteria are based on the cumulative damage frequency of tornado missile damage to all safety-related SSCs that are not provided positive protection. Therefore, the impacts of all non-conforming items are combined so that the total missile damage frequency is evaluated against the acceptance criterion of 1E-06 per year. If additional new non-conforming SSCs are identified in the future, TORMIS analysis may be used to evaluate these specific plant features and combine their damage impacts with the impacts of SSCs that were previously analyzed using the TORMIS methodology to determine if adequate protection is maintained.

The TORMIS computer code is used to determine the frequency of a damaging tornado missile strike on unprotected plant SSCs that are used to mitigate a tornado. The TORMIS code is an updated version of the original TORMIS code developed for the Electric Power Research Institute (EPRI). The methodologies used in the code to evaluate the frequency of damaging tornado missile strikes are documented in References 9, 10, 11 and 12.

The TORMIS code accounts for the frequency and severity of tornadoes that could strike the plant site, performs aerodynamic calculations to predict the transport of potential missiles around the site, and assesses the annual frequency of these missiles striking and damaging structures and other targets of interest.

The analysis requires the development of input data in three broad areas:

1. development of site tornado hazard information.
2. development of site missile characteristics.
3. development of target size, location, and physical properties.

TORMIS Model Inputs

The TORMIS methodology seeks to demonstrate that the annual probability of a radioactive release in excess of 10 CFR 100 resulting from tornado missile damage to unprotected SSCs used to mitigate a tornado is less than the acceptance criterion of 1E-06/rx-yr. This means that the unprotected SSCs are evaluated collectively against the acceptance criterion rather than individually. For a multi-unit site such as Oconee, this criterion is applied to each unit individually.

For this evaluation, the prevention of a "release in excess of 10 CFR 100" is accomplished by establishing SSD conditions following a tornado strike and maintaining these conditions for up to 72 hours. The following safety functions are required:

- Secondary Side Decay Heat Removal,
- Reactor Coolant Makeup,
- Reactor Coolant System pressure boundary integrity.

Through a process of plant walkdowns and reviews of plant drawings, calculations, and other information, a detailed list of structures and equipment lacking deterministic protection was developed that meets the scope of the TORMIS safety targets described above.

TORMIS Results

A site specific analysis of vulnerable tornado mitigation equipment (SSCs) has been conducted using the TORMIS analysis methodology. This includes a characterization of the site tornado hazard and potential tornado generated missiles developed in a manner consistent with the requirements of the TORMIS User's Manual and other TORMIS reference materials.

For each Oconee unit, the mean annual frequency of a damaging tornado missile strike resulting in a radiological release in excess of 10 CFR 100 limits was determined to be less than the acceptance criteria of 1E-06. The analysis was performed in a manner consistent with the requirements of the EPRI topical reports and with the requirements set forth in the NRC's SER (Reference 14) and RIS 2008-14 (Reference 15).

UFSAR Marked-Up Pages
Attachment 2

250°F with a long term strategy for reactivity, decay heat removal and inventory/pressure control. Long-term subcooled natural circulation decay heat removal is provided by supplying lake water to the steam generators and steaming to atmosphere. The extended coping period at these conditions is based on the significant volume of water available for decay heat removal and reduced need for primary makeup to only match nominal system losses. A stuck rod is not required to be postulated for this event. Initial conditions are 100% power with sufficient decay heat such that natural circulation can be achieved. The hypothesized fire is to be considered an "event", and thus need not be postulated concurrent with non-fire-related failures in safety systems, other plant accidents, or the most severe natural phenomena (Reference 31).

Deleted Paragraph(s) per 2015 update.

Deleted Paragraph(s) per 2012 update.

TURBINE BUILDING FLOOD EVENT

The Turbine Building Flood was one of the events that was identified in the original SSF licensing requirements. The SSF is designed to maintain the reactor in a safe shutdown condition for a period of 72 hours following a TB Flood. No other concurrent event is assumed to occur. The success criteria for this event is to assure natural circulation and core cooling by maintaining the primary coolant system filled to a sufficient level in the pressurizer while maintaining sufficient secondary side cooling. The reactor shall be maintained at least 1% $\Delta k/k$ with the most reactive rod fully withdrawn. (Reference 1, 10)

SECURITY-RELATED EVENT

A Security Related Event was one of the events that was identified in the original SSF licensing requirements. The SSF is designed to achieve and maintain a safe shutdown condition for this event. No other concurrent event is assumed to occur. (Reference 1) The success criteria for this event is to assure the core will not return to criticality, the active fuel will not be uncovered, and long-term natural circulation will not be halted. (Reference 41)

STATION BLACKOUT EVENT

This event was licensed after the design of the SSF was completed and approved by NRC. The SSF w[...] and remains valid. [...]hod the plant would employ to mitigate a SBO event. (References 38 and 39) The success criteria is to maintai[...] overall tornado mitigation strategy utilizes the [...]r 4 hours. No stuck rod is assumed for this event. Initial conditions [...]d 100 days of operation. [...]ce 40)

The design criterion is now extended to SSCs that are a part of the plant which the SSF relies upon and interfaces with for tornado mitigation. This is satisfied either through physical protection or evaluated by TORMIS.

[...]RNADO DESIGN CRITERIA

Originally, the design criterion did

[...] design ~~criterion for the SSF~~ structure that was committed to as part of the original SSF licensing [...]ndence. All parts of the SSF ~~itself~~ that are required for mitigation of the SSF events are [...] to be designed against tornado winds and associated tornado missiles. This [...]ent is satisfied through appropriate design of the SSF structure. ~~This requirement does~~ [...]~~nd to~~ SSCs that were already part of the plant which SSF relies upon and interfaces [...] event ~~mitigation~~. It is important to note that the SSF ~~was not licensed~~ to mitigate a [...] ~~event or a tornado missile event~~ (Reference ~~1~~) 42 Tor[...] design requirements for the [...]lf are addressed in Section 3.2.2. Add Insert 4. A subsequent issue related to crediting SSF ASW as [...]native for EFW tornado missile protection vulnerabilities is discussed below (see EFW [...] Missile Design Criteria).

[...]ISMIC DESIGN CRITERIA (GL 81-14)

During the seismic qualification review of the Oconee EFW system in the 1980s, the NRC postulated that a seismic event could break a pipe and potentially cause a flood of the turbine

Section 9.6.2, SSF Tornado Design Criteria: Insert 4

Successful mitigation of a tornado condition at Oconee shall be defined as meeting the following criteria to ensure that the integrity of the core and RCS remains unchallenged:

- The core must remain intact and in a coolable core geometry during the credited strategy period.
- Minimum Departure from Nucleate Boiling Ratio (DNBR) meets specified acceptable fuel design limits.
- RCS must not exceed 2750 psig (110% of design).

In addition to the criteria specified above, the following criteria are validated for the overcooling analysis to demonstrate acceptable results:

- Steam Generator tubes remain intact.
- RCS remains within acceptable pressure and temperature limits.

The tornado initial conditions are defined for the unit(s) as MODE 1, 102% rated thermal power at end of core life (690 effective full-power days). The tornado is assumed to leave one unit significantly damaged and a loss of all AC power to all three units. Two bounding analyses were performed, overheating and overcooling. For an overheating event, the significantly damaged unit is supplied by SSF ASW. The other two units will be initially supplied by the TDEFWP and subsequently supplied by SSF ASW. For an overcooling event, the TDEFWP is conservatively assumed to run until the contents of the Upper Surge Tank are depleted (to maximize the overcooling). SSF ASW flow is subsequently established to all three units as needed.

Following a tornado induced overcooling event the unit may experience a minor return to power of short duration. There are no consequences associated with the return to power due to the very low power level generated. The SSF is not required to meet the single failure criterion or the postulation of the most reactive rod stuck fully withdrawn. Failures in the SSF system will not cause failures or inadvertent operations in other plant systems. The SSF requires manual activation and can be activated if emergency systems are not available.

UFSAR Marked-Up Pages
Attachment 2

such that it would be expected to withstand the design basis earthquake. The piping from the hotwell to the MDEFWPs is seismically qualified.

Portions of the EFW System are vulnerable to tornado missiles. Thus, the plant relies upon diverse means to provide feedwater to the SGs in the event of a tornado. These diverse means include the SSF ASW System and the PSW System.

The Emergency Feedwater System was not designed to withstand the effects of internally generated missiles. If such an event were to occur and if main feedwater were unavailable, the single train SSF ASW System would provide an assured means of providing heat removal from the SGs. A detailed evaluation of the capability of the existing EFW System to withstand missiles was not considered necessary (Reference 2).

The effects of High Energy Line Breaks have been analyzed as addressed in UFSAR Section 3.6.1.3.

Provisions for water hammer events are considered unnecessary due to the use of Once Through Steam Generators (OTSG) (Reference 9).

Portions of the Emergency Feedwater system are credited to meet the Extensive Damage Mitigation Strategies (B.5.b) commitments, which have been incorporated into the Oconee Nuclear Station operating license Section H - Mitigation Strategy License Condition.

10.4.7.1.1 Deleted Per 2002 Update

10.4.7.1.2 Deleted Per 2002 Update

10.4.7.1.3 Deleted Per 2002 Update

10.4.7.1.4 Deleted Per 2002 Update

10.4.7.1.5 Deleted Per 2002 Update

10.4.7.1.6 Deleted per 1996 Revision

10.4.7.1.7 Deleted Per 2002 Update

10.4.7.1.8 Deleted Per 2002 Update

1. requiring portions of the EFW System (defined in UFSAR Section 3.2.2) to be capable of withstanding a MHE, and
2. providing alternative seismically qualified means of decay heat removal with the SSF ASW System and the HPI System.

10.4.7.3.6 EFW Response Following Tornado Missiles

Reference 7 concludes that the Standard Review Plan probabilistic criterion is met based upon the probability of failure of the EFW and station ASW Systems combined with the protection against tornado missiles afforded the SSF ASW System. Subsequently, PSW replaced station ASW relative to this function.

10.4.7.3.7 EFW Response Following a SBO

This event is similar to the LMFW with LOOP analysis with the additional assumption that the onsite emergency AC power sources have been lost. This results in the loss of the MDEFWPs. The TDEFWP should be available for 2 hours during this event because of its AC power independence. The SSF ASW System; however, is credited to remove the decay heat in this event. The SBO event, which is not a design basis event, is described in UFSAR Section 8.3.2.2.4.

10.4.7.3.8 Initiation of SSF ASW, PSW, and HPI Forced Cooling

The SSF ASW System, PSW, and HPI forced cooling serve as alternate means of decay heat removal for some of the EFW design events described in Section 10.4.7.3.

Once the control room decides to use the SSF ASW system, the system can be aligned within 14 minutes, consistent with the assumptions in the safety analyses. The SSF ASW flow rate provided to each unit's steam generators is controlled using the motor operated valves on each unit's SSF ASW supply header. The SSF contains adequate instrumentation to maintain the plant in a safe shutdown condition. The SSF ASW System is described in Section 9.6 of the UFSAR.

The Protected Service Water (PSW) system is designed as a standby system for use under emergency conditions. The PSW System is powered from either the Central Tie Switchyard via a 100 kV transmission line to a 100/13.8 kV substation or the Keowee Hydroelectric Station. The PSW System is provided as an alternate means to achieve and maintain safe shutdown for one, two, or three units.

The PSW System is capable of cooling each unit's RCS to approximately 250°F and maintaining this condition for an extended period. Failures in the PSW System will not cause failures or inadvertent operations in existing plant systems. The PSW System is operated from the Main Control Rooms (MCRs) when existing diverse emergency systems are not available. The power to PSW controlled pressurizer heaters must be manually aligned outside the MCR.

Please note that information associated with powering the pressurizer heaters and vital I&C battery chargers will not be effective until completion of Milestone 5, but is being included in the UFSAR for completeness.

If feedwater is unavailable, operator action is taken on high RCS pressure or pressurizer level to initiate HPI forced cooling. These actions are from the control room and include starting HPI pumps, opening the PORV, and throttling HPI flow as necessary. HPI forced cooling is initiated within 5 minutes of exceeding the initiation criteria. The HPI System is described in Section 6.3 of the UFSAR.

ENCLOSURE 1

RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

[NON-PROPRIETARY]

Enclosure 1

Responses to Requests for Additional Information [Non-Proprietary]

RAI 1

Regarding Section 3.8 of the application dated September 14, 2018 (i.e., the license amendment request (LAR)), the NRC staff requests the licensee to provide a discussion of the meaning of "Passive Civil Features" with respect to the Oconee licensing basis and Technical Specification operability.

RAI 1 Response:

Tornado is a design criterion that applies to structures, systems, and components (SSCs) credited in the Oconee Current Licensing Basis (CLB) for the mitigation of tornadoes. Similar to the treatment of other design criteria in the CLB, the operability/functionality process is entered when aspects of a particular design criterion are found deficient. Since there are no Technical Specifications applicable to tornado, functionality is assessed according to station procedures to determine if a non-conforming condition exists.

To control design features that are credited in the mitigation of tornadoes, Oconee maintains a passive design features control process. The process is described in Site Directive 3.2.16, "Control of Passive Design Features." The objective of this process is to maintain civil structures to protect important systems and components from both internal and external events described in the Updated Final Safety Analysis Report (UFSAR). The directive allows tornado protection barriers to be temporarily taken out of service to allow maintenance or changes to the facility. However, the directive requires in those cases that compensatory actions be available should a tornado watch/warning be declared by the National Weather Service. The compensatory actions may take different forms including: (1) Having the means, such as tools, equipment, labor, etc. in place to restore the original barrier back to its design configuration; (2) Having the means to install a temporary barrier equivalent to the normal barrier. Also, the directive provides guidance that intentional breaches of tornado related barriers should be accomplished in periods of the year where there is a lower risk of a tornado impacting the site.

If a passive design feature is found to be out of service without Operation's approval, then Operations shall be contacted immediately. A Nuclear Condition Report must be initiated, operability/functionality assessed, and action taken to restore the design feature. Appropriate contingency actions must be established until the passive design feature is restored. The LAR credits the Standby Shutdown Facility (SSF) as the tornado mitigation system.

RAI 2

Section 3.8 of the LAR states, "Because a tornado is a design criterion and does not constitute a design basis accident or transient as described in 10 CFR 50.36(c)(2)(ii), degradation of passive civil features protecting the SSF will not apply to operability under TS LCO 3.10.1, 'Standby Shutdown Facility.'" The SSF satisfies criterion 4 of 10 CFR 50.36(c)(2)(ii), which is not related to design basis accidents or transients. Criterion 4 is for SSCs which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. The NRC staff requests the licensee to provide a more substantial discussion and justification for why degradation of passive civil features protecting the SSF will not apply to operability under TS LCO 3.10.1, "Standby Shutdown Facility."

RAI 2 Response:

TS Limiting conditions for operations (LCOs) are defined in 10 CFR 50.36(c)(2) as "...the lowest functional capability or performance levels of equipment required for safe operation of the facility." 10 CFR 50.36(c)(2)(ii) notes that "A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria." Criterion 4 of 10 CFR 50.36(c)(2)(ii) states "A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety." The NRC Policy Statement associated with technical specification improvements (No. 93-102, dated July 23, 1993) refers to "unique plant vulnerabilities" that a plant specific probabilistic safety assessment (PSA) has shown to be significant to public health and safety. Oconee agrees that the SSF meets criterion 4 for scenarios to which it was originally designed and licensed and not specifically to the mitigation of tornadoes or the maintenance of tornado passive design features associated with the SSF. The plant specific PSA shows that the importance of SSF is primarily associated with the mitigation of fire events and its vulnerabilities are associated with human reliability and equipment availability/reliability and not the maintenance of tornado passive design features.

Oconee maintains that tornado is not a design basis accident or transient. The SSF was originally licensed to mitigate fires postulated to occur in the TB, internal flooding in the TB due to failures of CCW, and security events. The facility met 10 CFR 50.36 criterion 4 based on the risk associated with these events, not tornado.

Oconee has recognized the importance of managing and maintaining passive design features in protecting the station from natural phenomena. To that end, a site administrative directive was created to manage the barriers to facilitate modifications to and maintenance of the station. The site directive (SD 3.2.16, "Control of Passive Design Features") is described in the response to RAI-1. The site directive requires that contingency actions be established prior to the intentional or planned breach of a given barrier associated with the SSF. This provides assurance that a given barrier can be quickly restored should deteriorating weather develop. It also requires entry into the operability process should a given barrier be discovered impaired. This provides assurance that should a given barrier be found deficient, functionality will be assessed, and corrective actions implemented in a timely manner to restore the barrier. Given the NRC-recognized low probability of a tornado impacting a particular nuclear station, the actions described above collectively assure that the SSF associated tornado passive design features will be in place and functioning should a tornado impact the site.

Oconee concludes that the potential degradation of SSF related tornado passive design features does not apply to the operability of the SSF as defined in TS 3.10.1 "Standby Shutdown Facility." This is due to the insignificant contribution of passive design features to the overall risk profile of the SSF, the NRC-recognized low probability of a tornado impacting a particular nuclear station, and the robust control and maintenance of tornado passive design features as described in the site directive.

RAI 3

The LAR does not include references to official calculations that document support for the assumptions made in the success criteria for the TORMIS Boolean logic. These references need to be provided on the docket to support the decision of the safety evaluation and to provide traceability, auditability, and inspectability.

A. The NRC staff requests the licensee to provide a reference (i.e., title(s), revision number(s), and date(s)) for the official calculation (or calculations) that support the following statement from the LAR concerning the CCW surge lines [emphasis added]:

   The surge piping targets are evaluated for crushing or crimping failure that would prevent adequate vent flow. <u>An evaluation showed that only 44% of the flow area of only one of the 24 inch pipe (one of two for success) is required to provide an adequate vent path.</u>

B. The NRC staff requests the licensee to provide a reference (i.e., title(s), revision number(s), and date(s)) for the official calculation (or calculations) that support the following statement from the LAR concerning the Main Steam Relief Valves:

   The assumed success criteria for the MSRVs for tornado mitigation is that one of two lowest pressure relief valves opens (either 1/2/3MS-8 on the 'A' Header or 1/2/3MS-16 on the 'B' Header), and that one relief valve (any one of eight) on the opposite header opens for overpressure protection.

RAI 3 Response:

3A The source of the statement described originates from OSC-8860 (Evaluation of Tornado Missile Damage Frequency for Oconee Unit 3), Revision 5, approved June 10, 2018. The evaluation performed to determine the minimum flow area of the surge line is contained within OSC-11622 (Oconee Tornado Strategy Utilizing the Standby Shutdown Facility (SSF)), Revision 2, approved October 30, 2018.

3B The source of the statement described originates from OSC-8860 (Evaluation of Tornado Missile Damage Frequency for Oconee Unit 3), Revision 5, approved June 10, 2018; OSC-9307 (Evaluation of Tornado Missile Damage Frequency for Oconee Unit 1), Revision 2, approved June 10, 2018; and OSC-9308 (Evaluation of Tornado Missile Damage Frequency for Oconee Unit 2), Revision 2, approved June 10, 2018.

The thermal hydraulic analysis within OSC-11638 (RCS Response to a Loss of Main Feedwater and 4160VAC Power with PSW or SSF Recovery (High Energy Line Break Analysis / Tornado Analysis)), Revision 5, approved May 15, 2019 documents that the required initial SSF ASW flow rate to a unit from the most limiting of the main steam relief valve (MSRV) sensitivity cases performed is 408 gpm (refer to case 1f1). OSC-4171 (SSF ASW Design Inputs Calculation), Revision 37, approved April 30, 2018 documents that the available SSF ASW flow rate provided to a single unit is 711.9 gpm (refer to case 55.1) assuming the most limiting variation of the described TORMIS MSRV success criteria. As seen by a comparison of the analysis results, the available SSF ASW flow significantly exceeds the required SSF ASW flow from the thermal hydraulic analysis. The success criteria documented within OSC-8860, OSC-9307, and OSC-9308 related to the available MSRVs ensures that the assumptions of the thermal hydraulic analysis contained within OSC-11638 and the SSF ASW flow analysis contained within OSC-4171 are met.

RAI 4

The application is crediting many conservatisms in the TORMIS modeling that offset the simplification and limitations of TORMIS computer code. One source of conservatism is the choice of worst case missiles (i.e., concrete block) to derive damage velocity values for the CCW surge lines. The application states:

> The finite element analysis supporting the damage velocity values for the CCW surge lines for concrete block, wood plank, and metal siding missiles are based on missile impacts at the worst-case location and at the worst-case angle of incidence. This combination represents only a small fraction of the potential missile interactions and is very conservative for estimating the frequency of damage to the CCW surge lines.

According to OSC-11760, "FINITE ELEMENT ANALYSIS [FEA] OF ONS CCW SURGE PIPES," the concrete block missile is modeled with smooth particle hydrodynamics (SPH) as opposed to finite elements. However, the distance chosen between particles can affect the failure property of the aggregate structure hence the actual energy delivered to the target and the subsequent deformation. The NRC staff requests the licensee to explain the basis for how the particle distances are chosen and benchmarked.

RAI 4 Response:

The SPH particle spacing was selected based on prior experience conducting concrete missile impact analyses against other targets. In general, as a smaller particle size is used a more accurate answer is obtained and less damage is imparted to the target. This is because smaller particles allow the material to fracture into smaller pieces, with the potential to spread the applied impulse over larger areas. The best approach in general is to run analyses with varied particle spacing and demonstrate that the predicted damage (in this case crimping) has converged or to demonstrate that the spacing selected is conservative. This approach has been taken for other concrete missile geometries in previous analyses against other targets and this trend is seen. It was not taken here. But, based on these previous analyses, selection of smaller particles would likely result in equal or less crimping. Consideration was also taken for the specific concrete block geometry such that sufficient resolution (10 particles through the thickness of each wall of the concrete block) was used to resolve through-thickness and bending stresses.

RAI 5:

The defined missile types in the current licensing basis, as defined in UFSAR Table 9-17, includes a utility pole, which is usually the most conservative in terms of damage. Section 5.3 of the LAR does state the dominant missile types striking safety targets are wood plank and metal siding types; however, the NRC staff notes that the wood plank may not bound a utility pole. While the TORMIS analysis contains defined missile types of about 23 missiles, the FEA analysis for OSC-11760 includes site-specific missiles (i.e., concrete block and aluminum siding). The NRC staff requests the licensee to justify why the dominant missiles do not bound the utility pole and explain the justification for the missile set chosen for the FEA.

RAI 5 Response:

The utility pole and other missile types similar to those defined in the UFSAR are conservatively modeled and are shown in the TORMIS missile simulations to have a very low contribution to the CCW surge line damage frequency.

The tornado missile parameters defined in the CLB are conservatively defined to ensure that plant physical barriers can withstand worst case tornado missile impacts to provide adequate protection of safety related SSCs. The impact frequency for missile types comparable in weight and velocity to these deterministic missiles is inherently very low because they are defined at the upper range of these parameters to create the highest potential for barrier damage. In contrast, the subject TORMIS targets are not designed to withstand deterministic missile strikes and are susceptible to damage from lower velocity and lighter weight missile types that are more aerodynamic and more likely to strike the targets. Accordingly, the TORMIS missile types are chosen to represent the highest damage frequency potential rather than the highest potential physical damage.

For the analysis of the circulating cooling water (CCW) surge lines, a set of preliminary TORMIS simulations were made to collect data on missile hits on the surge lines. The data showed that the majority of hits came from wood planks, aluminum siding, and concrete blocks. These missile types were then evaluated using detailed analysis (FEA or SPH) to estimate the minimum impact velocity (VDAM) required to cause damage to the CCW surge lines. The final TORMIS analysis runs utilized the estimated VDAM values for the wood plank, aluminum siding, and concrete block and applied a conservative VDAM value of zero (hit=damage) for all other missile types including the utility pole.

RAI 6:

The LAR requests "Approval for elimination of the Spent Fuel Pool (SFP) to High Pressure Injection (HPI) flow path for Reactor Coolant Makeup (RCMU)," which implies that the flow path will be physically removed. The technical justification for this change in Section 3.7, "Elimination of SFP Suction for HPI," of the Enclosure to the LAR, could also be read that the line is being physically removed as it is no longer necessary. However, in Section 4.3, "No Significant Hazards Consideration," of the Enclosure to the LAR, it states, "The spent fuel pool suction path to the HPI system currently described in UFSAR Section 3.2.2 is being deleted from the licensing basis. The existing piping configuration that connects the spent fuel pool suction path to the HPI system will remain but will no longer be credited." The NRC staff requests the licensee confirm that this request is specifically to eliminate the flow path from the tornado licensing basis and not to make any changes to the plant itself. If this is not the case, then the NRC staff requests the licensee provide information on when this flow path is currently used and if it is credited in any analysis.

RAI 6 Response:

Though it will not be credited in the tornado licensing basis nor in plant operating procedures, the existing piping configuration that connects the SFP suction path to the HPI system will remain in the plant for beyond design basis response.

RAI 7

Page 21 of the Enclosure to the LAR states that the tornado initial conditions are defined for the unit(s) as MODE 1, 102% rated thermal power at end of core life (690 effective full-power days).

Given that this initial condition may not be bounding, the NRC staff requests the licensee to provide justification for assuming a single initial condition and no consideration of other initial conditions (i.e., low power/low decay heat) which could be more limiting for overcooling events.

RAI 7 Response:

The LAR described mitigation of a tornado assuming the units are at 100% power (the RCS thermal hydraulic analyses were performed assuming 102% power). As described in Section 3.1 of the SSF Thermal Margin LAR dated October 20, 2017, all SSF events are not subject to consideration for off nominal modes of operations. For SSF mitigated station blackout (SBO) events, initial reactor power assumptions are defined as 100% power and at least 100 days of operation at this power level (Regulatory Guide 1.155, Station Blackout, Section 3.2.1). A similar approach was used in the thermal hydraulic analysis of fire events, most recently NFPA 805, and historically Appendix R. Based on the low risk of off nominal modes of operations associated with the SSF usage during a Tornado, the SSF is being treated similarly to the initial conditions associated with SBO events (i.e. full power conditions) as described within 10 CFR 50.63.

As described within the SSF Thermal Margin LAR dated October 20, 2017, off nominal conditions represent those times in which the plant is not at 100% power for a minimum of approximately 4 days. These low power or low decay heat conditions were not deemed to result in an appreciable contribution to overall plant risk. A similar conclusion is reached for off nominal conditions for the SSF with respect to tornado mitigation. This is further supported by the NRR memorandum dated February 28, 2019 (Closeout of Low Safety Significant/Low Risk Concerns – Tornado-Generated Missile Protection), which stated that tornado missile scenarios that may lead to core damage are generally very low probability events.

RAI 8

Page 20 of the Enclosure to the LAR states, "The ONS RELAP5/MOD2-B&W model and analysis methods are described in Duke Energy's NRC approved methodology report DPC-NE-3003-PA (Reference 15) and have been modified, as described in Attachment 5, to include additional detail and features required to perform these analyses." On page 2 of the March 15, 1995 SE in Reference 15, it states, "RELAP5/MOD2-B&W has been reviewed by the NRC staff and is the subject of a safety evaluation (Ref.: Letter from A. Thadani to J. Taylor, dated April 18, 1990). The NRC staff found the code acceptable for use, subject to specified limitations, for calculation of transient response for reload analyses of large and small break LOCAs and operational transients *for plants having recirculating steam generators. The NRC staff is currently evaluating its use, for those purposes, for once-through steam generator (OTSG) plants* [emphasis added]."

Given that the approved methodology report is for use in the Oconee Updated Final Safety Analysis Report (UFSAR) Chapter 6 Loss of Coolant Accident (LOCA) mass and energy release analyses, the NRC staff requests the licensee to provide details on the approval of the RELAP5/MOD2-B&W code for use in analyzing overcooling (main steam line break) and overheating (loss of feedwater) transients. If the code has not been approved for use for these transients, provide justification for its use. In addition, the NRC staff requests the licensee describe any limitations and conditions as well as how they are met for use of the code for the selected transients.

LAR Reference 15: Duke Energy Methodology Report DPC-NE-3003-PA, Revision 1, "Mass and Energy Release and Containment Response Methodology," dated September 2004 (Safety Evaluations dated March 15, 1995; September 24, 2003, ADAMS Accession No. ML050320034)

RAI 8 Response:

RELAP5/MOD2-B&W SER for non-LOCA analyses

RELAP5/MOD2-B&W has been approved for non-LOCA analyses in the BAW-10193NP-A SER dated October 15, 1999 [Reference ML003682985]. This is a Framatome topical report for the B&W-designed nuclear steam supply system (NSSS). Based on the safety evaluation report (SER), RELAP5/MOD2-B&W version 19.0 is used in the BAW-10193 submittal.

The BAW-10193 topical report presents benchmarks of RELAP5/MOD2-B&W calculations to data from test facilities and plant transients, as well as comparisons to computer code predictions from CADDS and TRAP2, to demonstrate that RELAP5/MOD2-B&W properly predicts the phenomena exhibited by Babcock and Wilcox (B&W) designed Pressurized Water Reactors (PWRs) during non-LOCA events. The benchmarks and comparisons include an overheating and an overcooling event.

BAW-10193 Section 5.2 provides a benchmark of Three Mile Island (TMI)-2 Loss of Feedwater (LOFW) Event. The benchmark of the TMI-2 LOFW event shows that RELAP5/MOD2-B&W is appropriate for analyzing overheating events on B&W-designed PWRs, as stated in Section 2.2.1 of the NRC Safety Evaluation for BAW-10193.

BAW-10193 Section 6.2 provides a code to code comparison of a main steam line break (MSLB) between RELAP5/MOD2-B&W and TRAP2. The RELAP5/MOD2-B&W and TRAP2 comparisons of the MSLB events demonstrate that, given conservative initial and boundary conditions, RELAP5/MOD2-B&W produces conservative results, similar to those predicted by TRAP2 as stated in Section 2.3.2 of the NRC Safety Evaluation for BAW-10193.

The Safety Evaluation does not provide limitations and restrictions for the use of the topical report, but Framatome included Appendix A in response to staff question #1. Appendix A describes the noding details to be used to model the NSSS for various accidents and lists the options for constitutive models and correlations.

Review of BAW-10193 Appendix A

The discussion provided in BAW-10193 Section A.2 relates to the code options available in RELAP5/MOD2-B&W deal with interface drag inputs for the OTSG, and heat transfer correlation adjustments made to the nucleate boiling, critical heat flux (CHF) and post-CHF correlations. These options are either not available in the code version used by Duke Energy or are not adjusted from the base coding values in the Oconee RELAP5 model.

The Duke Energy Oconee RELAP5 model is similar to the large detail model described in BAW-10193 Appendix A. The Oconee model has a similar number of steam generator (SG) secondary nodes, uses the same approach for the high elevation auxiliary feedwater (AFW) model, and has a finer nodalization in the reactor core and pressurizer.

Duke Proprietary Information

- Proprietary bracketing is consistent with DPC-NE-3003-PA

- Reactor core, SG secondary, and high elevation AFW model are based on DPC-NE-3003-PA
- Pressurizer and SG tube wetting percentage are based on the Oconee Tornado analyses

BAW-10193 non-proprietary information obtained from ML003682985

	BAW-10193 App A large detail model	Oconee RELAP5 model Tornado analyses
Reactor Core	3 nodes	[ ]a,c
SG secondary	11 nodes	[ ]a,c
Pressurizer	11 nodes	[ ]a,c
High elevation AFW model	2 radial regions	[ ]a,c
SG tubes % wetted	10%	[ ]a,c

Comparison of RELAP5/MOD2-B&W code versions

The Duke Energy methods described in DPC-NE-3003-PA were submitted on August 11, 1993, a couple of years before Framatome submitted BAW-10193, RELAP5/MOD2-B&W for Safety Analysis of B&W Designed PWRs. This time difference is the source of the DPC-NE-3003 Revision 0 SER text, which is highlighted in the first paragraph of RAI 8. At the time the DPC-NE-3003-PA Revision 0 SER was being written, the approved version of BAW-10164P-A addressed recirculating SGs.

RELAP5/MOD2-B&W is described in BAW-10164P-A and approved for use with OTSGs in Revision 3. This code provides the basis for the B&W plant safety analysis capability described in BAW-10193.

Duke Energy methods use RELAP5/MOD2-B&W version 13.0 which roughly corresponds to BAW-10164 Revision 1. The topical revision record on page v of the BAW-10164 Revision 4 [ML030220134] provides a description of the changes included in the various revisions of BAW-10164. This information has been transcribed below for convenience. Based on the BAW-10193 SER (ML003682985), RELAP5/MOD2-B&W version 19.0 is used in the BAW-10193 Revision 0.

BAW-10164 Documentation Revision	Description	Program Version
0	Original Issue	8.0
1	Typographical corrections Replace CSO correlation with Condie-Bengston IV	10.0
2	SBLOCA modifications Miscellaneous corrections	18.0
3	EM Pin Enhancements Filtered flows for hot channel heat transfer Rupture area enhancement for surface heat transfer OTSG improvements and benchmarks using the Becker CHF, Slug Drag, and Chen Void Ramp	19.0
4	Zirconium-based allow pin model changes Option for multiple pin channels in a single core fluid channel Void-dependent core cross flow option Zirconium-based alloy rupture temperature	24.0

The modifications made to RELAP5/MOD2-B&W in BAW-10164 Revisions 2, 3 and 4 are primarily to facilitate the LOCA peak cladding temperature (PCT) calculations required for BAW-10192 LOCA analyses. Revision 4 of BAW-10164 is referenced in the latest revision of DPC-NE-3003-PA as the basis for the RELAP5/MOD2-B&W code due to the information contained in the BAW-10164 Revision 4 SER on the applicability of RELAP5/MOD2-B&W to the B&W designed lowered loop plants. The BAW-10164 revision 4 topical report includes the Revision 3 SER and includes information on the benchmarks performed for the B&W plants.

The high elevation heat transfer model described in BAW-10164 is used in the Duke Energy's Oconee RELAP5 model. The details of this model are provided starting on page 5-354 of BAW-10164 Revision 4 which can be found in ML030410278. Review and approval for usage of the high elevation heat transfer model is included in the DPC-NE-3003-PA methods.

BAW-10164 Limitations and Restrictions

The limitations and restrictions from BAW-10164 Revisions 1 through 4 are provided below. Following each limitation or restriction is a paragraph describing the applicability to the Duke Energy DPC-NE-3003-PA based methods used for the Oconee tornado analyses.

<u>BAW-10164 Revision 1</u>

The following limitations and restrictions are obtained from the BAW-10164 Revision 1 SER (ML030220205).

1. The Chen-Sundaram-Ozkaynak film-boiling correlation in the core heat transfer model and the B&W auxiliary feedwater model for OTSGs were not reviewed and, therefore, should not be used in licensing calculations without prior review and approval by the NRC.

Duke Energy applicability

The core heat transfer model identified is not selected for use in the Duke methods. The B&W AFW model is part of the approved DPC-NE-3003-PA methods.

2. Prerupture cladding swell is not modeled because BWFC indicated that the swell is generally less than 20 percent with insignificant flow diversion effects. The acceptability of neglecting the effects of prerupture swelling is part of the LOCA EM review based on BWFC's analysis of the flow diversion effects. The SER on report BAW-10168P will address the resolution of this matter.

   Duke Energy applicability

   The prerupture cladding swell option is not used.

3. The built-in kinetics data for decay heat calculations in the RELAP5/MOD2-B&W code are based on the 1973 and 1979 standards of the American Nuclear Society (ANS). Because Appendix K requires the use of a value that is 1.2 times the 1971 ANS standard for decay heat calculation, BWFC should ensure that the decay heat used in licensing LOCA analysis complies with Appendix K.

   Duke Energy applicability

   The Oconee tornado analyses use input designed to replicate ANS-79 based decay heat loads. The Duke Energy Oconee applications are not required to comply with Appendix K.

4. The LOCA assessments of the Extended Henry-Fauske and Moody critical flow models were based on the use of the static properties as input to the critical flow tables. The LOCA licensing calculations should be performed accordingly.

   Duke Energy applicability

   The Extended Henry-Fauske and Moody critical flow models are not used.

5. The interphase drag model of the RELAP5/MOD2-B&W code tends to overpredict interphase drag. This overprediction may cause nonconservative predictions of loop seal clearing phenomena in that liquid is cleared even when the steam flow is not sufficiently high to drag the-liquid out of the loop seal. Therefore, this model may not accurately calculate the core uncovery and the PCT. A resolution requiring a sensitivity study to choose a proper loop seal nodalization that results in the highest PCT calculation will be addressed in the LOCA EM review.

   Duke Energy applicability

   The Duke Energy applications are not used for determining PCT. The loop seal clearing phenomena described in the limitation is applicable to recirculating SG plants. The internal reactor vessel vent valves in the B&W plant eliminate this phenomena during LOCAs.

6. Even though noncondensible gases are not modeled in the small break LOCA (SBLOCA) system analysis, BWFC demonstrated negligible effect that all sources of noncondensible gases will have on the overall response of the system for the range of SBLOCAs. However, BWFC noted that a 50 psi increase above the steam generator

control pressure of 1150 psia could result from a worst case release of noncondensible gases. The staff believes that this pressure increase generally would not substantially reduce the injection capabilities of the charging and safety injection (SI) systems. However, because the performance characteristics of the SI pumps vary widely in the plants, verification should be made on a plant-specific basis to ensure that a 50 psi pressure increase will not greatly reduce SI flow such that the PCT would increase by more than 500°F. Otherwise, additional information should be provided to justify neglect of noncondensible gases, or the effect of the pressure increase caused by noncondensible gases should be included in the analysis.

Duke Energy applicability

The Duke Energy applications are not used for determining PCT, and noncondensible gases are not modeled. The concern identified in the limitation is not present in the Oconee tornado analyses.

7. For a complete safety analysis, an approved core thermal hydraulic code and CHF correlation should be used with the RELAP5/MOD2-B&W code. The noding details and inputs should be justified on a plant-specific basis. The choice of constitutive models including the empirical models and correlations should be justified to ensure their use is within the ranges of applicability.

Duke Energy applicability

The Duke Energy methods use an approved core thermal hydraulic code and CHF correlation to evaluate the core thermal response using transient results from the RELAP5/MOD2-B&W code.

A departure from nucleate boiling ratio (DNBR) evaluation is performed for the overcooling analysis using VIPRE and the EPRI and Modified Barnett CHF correlations. The VIPRE methodology used is described in the Duke Energy NRC approved methodology report DPC-NE-3000-PA. The EPRI CHF correlation is used to identify the limiting critical heat flux and DNBR statepoints. The Modified Barnett CHF correlation is then used to evaluate the limiting statepoints identified with the EPRI correlation and the peak heat flux statepoint. The Modified Barnett correlation is the current licensed correlation used for low pressure (steam line break) events for Oconee and B-HTP fuel.

Reference for DPC-NE-3000-PA

Duke Energy Methodology Report DPC-NE-3000-PA, Oconee Nuclear Station, McGuire Nuclear Station, Catawba Nuclear Station, Thermal-Hydraulic Transient Analysis Methodology, Revision 5. (Safety Evaluations for Oconee Nuclear Station dated August 8, 1994 (Accession Number ML16293A840); October 14, 1998 (Accession Number 9810190223); September 24, 2003 (Accession Number ML032670816); October 29, 2008 (Accession Number ML082800408); and July 21, 2011 (Accession Number ML11137A150)).

BAW-10164 Revision 2 and Revision 3

The following limitations and restrictions are obtained from the BAW-10164 Revision 2 and 3 SER (ML030410278). Note the code version used for the Duke Energy calculations does not include all of the options included in this version of BAW-10164.

Modifications made to RELAP5/MOD2-B&W as described in Revisions 2 and 3 of BAW-10164P have been reviewed and evaluated. Based on the benchmarks presented, the staff finds that the models described in version 19 of RELAP5/MOD2-B&W to be acceptable for LOCA and non-LOCA analysis for PWRs with recirculating and OTSGs subject to the following limitations:

1. Use of the Wallis and UPTF parameters at the tube bundle and steam generator plenum inlet are acceptable. The parameters used in the CCFL model for any other application must be validated, and the validation reviewed and approved by the staff for that application (see section 3.1.3 of this evaluation).

   Duke Energy applicability
   These options are not used. The CCFL model input addressed by this limitation is not available in the code version used by Duke Energy.

2. The BWUMV correlation is limited to pressures above 1300 psia.

   Duke Energy applicability

   The BWUMV correlation option is not used.

3. For large break LOCA emergency core cooling system (ECCS) evaluation model calculations, form losses due to ruptured cladding should not be excluded using the user option described in Section 3.2.4 of this evaluation.

   Duke Energy applicability

   The cladding rupture options are not used.

4. The value of the user specified parameters listed in Table 1 of this evaluation (i.e. those used for the benchmark calculations) are the only acceptable values for LOCA licensing calculations.

   Duke Energy applicability

   A review of the parameters appearing in Table 1 shown on pages 5-392 to 5-394 indicates these options are not used. In addition, the Duke Energy methods are not used for LOCA licensing calculations to determine fuel cladding PCT.

BAW-10164 Revision 4

The following limitations and restrictions are obtained from the BAW-10164 Revision 4 SER (ML030220258). Note the code version used for the Duke Energy calculations does not include the options included in this version of BAW-10164. The text referring to Section 2 below, is copied from the BAW-10164 Revision 4 SER.

Based on reviews discussed in Section 2, the staff finds the following Framatome proposed methodology changes (BAW-10164P, Revision 4) acceptable within the stated terms and limitations:

1. A change that will model the hot channel modeling to treat the hot pin and the hot assembly as two heat structures for large break LOCA (LBLOCA) evaluations of RSG and OTSG plants.

Duke Energy applicability

The Duke Energy methods do not include hot channel modeling and are not used for LOCA licensing calculations to determine fuel cladding PCT.

2. A change to the initial fuel stored energy uncertainty that will apply a lower uncertainty in the initial fuel stored energy, derived from TACO3, to the hot assembly and core average heat structures for LBLOCA evaluations of RSG and OTSG plants.

   Duke Energy applicability

   The Duke Energy methods use an initial core average fuel temperature selected based on the transient objectives. For the overcooling analysis, the predominant concern is the potential for a return to criticality. A high initial fuel temperature is assumed to maximize the Doppler feedback during the transient. For the overheating analysis, a high initial fuel temperature is assumed to maximize the initial stored energy in the core.

   The limitation is intended to ensure appropriate inputs are selected for the initial fuel stored energy for analyses that determine fuel cladding PCT. The Duke Energy methods are not used for LBLOCA licensing calculations to determine fuel cladding PCT.

3. A change to automate the void dependent crossflow model and to interpolate the inter-channel void-dependent cross-flow for SBLOCA evaluations for OTSG plants.

   Duke Energy applicability

   The Duke Energy methods do not include the void dependent crossflow model and are not used for SBLOCA licensing calculations to determine fuel cladding PCT.

4. Automation of the core heat BEACH blockage limitation that will automate the flow blockage limit in BEACH, used for LBLOCA and SBLOCA analyses of RSG and OTSG plants.

   Duke Energy applicability

   The Duke Energy methods do not include hot channel modeling and are not used for LOCA licensing calculations to determine fuel cladding PCT.

For reasons discussed in Section 2, in its review of future changes to the LBLOCA and SBLOCA methodologies beyond the context discussed in this safety evaluation, the staff will closely examine the impacts of the proposed changes with respect to the TACO3 stored energy model, the hot channel modeling changes, and the cross-flow model discussed in this safety evaluation.

Duke Energy applicability

The Duke Energy methods are not used for LOCA licensing calculations to determine fuel cladding PCT.

Review of Overcooling Transient Phenomena

The following is a review of the major phenomena in the Oconee tornado overcooling transients. The initiating event is a loss of the secondary system pressure boundary that leads to an RCS temperature and pressure decrease, pressurizer level decreases, and a reactor trip. The

Reactor Coolant Pumps (RCPs) are either tripped by a loss of power or by the operator in accordance with procedural guidance, causing a flow coastdown. Liquid flashing in the condensate and feedwater piping occurs. Sustained two-phase conditions develop in the RCS. RCS makeup flow refills the RCS and restores pressurizer level. Steam line relief valves must be modeled and high elevation heat transfer occurs in the SGs as the heat sink is restored. With the exception of the initial increase in SG heat transfer due to decreasing SG pressure, each of these phenomena are present in the large and small break LOCA analyses using the approved methods described in DPC-NE-3003-PA.

- Initial increase in SG heat transfer due to decreasing SG pressure. In the tornado analyses the initial overcooling is due to the available SG heat removal exceeding the core decay heat load. The main feedwater pumps are assumed to trip, liquid flashing in the condensate/feedwater piping is aided by extended turbine driven emergency feedwater (EFW) flow to provide the sustained heat removal capability.
- Reactivity feedback due to RCS temperature decrease. The LOCA analyses include moderator density feedback that will account for temperature changes as well as voiding. Reactivity feedback due to density changes is modeled for the tornado analyses. The overcooling analyses use a significantly more negative moderator temperature coefficient than that assumed in the LOCA analysis to maximize the reactivity feedback.
- Pressurizer level decreases offscale low, then recovers. ECCS restores pressurizer level for the smaller SBLOCAs. The pressurizer level increase is a function of the RELAP5 state equations and the mass and energy balance for the event.
- Reactor trip and turbine trip. Both are modeled in the SBLOCA analyses.
- RCP coastdown. LOCAs model RCP coastdown on either loss of power or by operator action on the loss of subcooled margin.
- High elevation SG heat transfer. At Oconee, main feedwater auto-swaps from the main nozzles to the upper nozzles on RCP trip. All other feedwater sources, such as EFW, SSF ASW, PSW, or Flex equipment, are aligned to the upper nozzles. Thus, when the heat sink is restored, the high elevation heat transfer model is used.

The overcooling transient phenomena are well within the capabilities of the RELAP5/MOD2-B&W code. The overcooling transient benchmark results provided in BAW-10193 demonstrate the code capability, and the acceptability is provided by the NRC approval of BAW-10193 and BAW-10164 for performing non-LOCA analyses for the B&W-designed NSSS.

Review of Overheating Transient Phenomena

The following is a review of the major phenomena in the Oconee tornado overheating transients. The initiating event is a loss of SG heat transfer that leads to an RCS temperature and pressure increase, pressurizer level increase, and a reactor trip. The RCPs are either tripped by a loss of power or by the operator causing a flow coastdown. Pressurizer and steam line relief valves must be modeled, and high elevation heat transfer occurs in the SG(s) as the heat sink is restored.

In general overheating transients are not as challenging from a code simulation perspective, with the potential for sustained two-phase conditions resulting in the selection of RELAP5 for the

Oconee tornado analyses. With the exception of the RCS pressure and temperature increase, resulting in a pressurizer level increase, each of these phenomena are present in the large and small break LOCA analyses using the approved methods described in DPC-NE-3003-PA.

- Loss of SG heat transfer due to loss of feedwater flow. In a larger break LOCA heat transfer is initially lost due to the depressurization of the RCS and limiting the ability to release steam in the SG secondary. Feedwater flow is also terminated at the beginning of the LOCA transient. The heat transferred from the RCS to the SG steam environment is modeled in a portion of the SBLOCA analyses.
- RCS pressure/temperature increase and Pressurizer level swell. The RCS temperature increase is the result of a mismatch in heat transfer between the core in the steam generator. The ability to transfer heat from the core is established in the LOCA analyses. The RCS pressure and Pressurizer level increase are functions of the RELAP5 state equations and the mass and energy balance for the event.
- Reactor trip and turbine trip. Both are modeled in the SBLOCA analyses.
- RCP coastdown. LOCAs model RCP coastdown on either loss of power or by operator action on the loss of subcooled margin.
- High elevation SG heat transfer. At Oconee, main feedwater auto-swaps from the main nozzles to the upper nozzles on RCP trip. All other feedwater sources, EFW, SSF ASW, PSW, or Flex equipment, are aligned to the upper nozzles. Thus, when the heat sink is restored, the high elevation heat transfer model is used.

The overheating transient phenomena are well within the capabilities of the RELAP5/MOD2-B&W code. The overheating transient benchmark results provided in BAW-10193 demonstrate the code capability, and the acceptability is provided by the NRC approval of BAW-10193 and BAW-10164 for performing non-LOCA analyses for the B&W-designed NSSS.

RAI 9

Page 17 of the Enclosure to the LAR describes revisions to the SSF Tornado Design Criteria in UFSAR, Section 9.6.2, and lists the following five criteria to ensure that the integrity of the core and RCS remains unchallenged:

- The core must remain intact and in a coolable core geometry during the credited strategy period.
- RCS must not exceed 2750 psig (110% of design).
- Minimum Departure from Nucleate Boiling Ratio (DNBR) meets specified acceptable fuel design limits.
- Steam Generator tubes remain intact.
- RCS remains within acceptable pressure and temperature limits.

Page 21 of the Enclosure to the LAR states, "In addition to the criteria specified above, the following criteria are validated for the overcooling analysis to demonstrate acceptable results:

- The steam generator tubes remain intact.
- RCS remains within acceptable pressure and temperature limits."

The above implies that these last two criteria are examined only for the overcooling analysis and not the overheating analysis. The NRC staff requests the licensee clarify whether the last two criteria are validated for the overheating analysis. If they are not, then the NRC staff requests the licensee to also justify why it used different criteria between the two analyses (overheating and overcooling) and why this distinction is not made clear in the proposed revisions to the UFSAR section.

RAI 9 Response:

The acceptance criteria identified in LAR Section 2.6 for UFSAR Section 9.6.2 description on page 17 of the LAR is not consistent with the content of LAR Section 3.2 and will be revised. See Attachment 2 for revised markup.

The overcooling analysis includes two acceptance criteria not applicable to the overheating analysis. The difference in the acceptance criteria provided in LAR Section 3.2 does reflect the criteria used in the respective analyses, as described in LAR Attachment 7, Section 2.0. The LAR Section 2.6 description of changes for UFSAR Section 9.6.2 has been modified to reflect the LAR Section 3.2 description of the acceptance criteria.

The two additional criteria validated in the overcooling analysis recognize the thermal stress induced on the RCS and SG materials during the transient evolution. These criteria ensure the thermal stress induced on the RCS materials during the transient evolution does not challenge the integrity of the RCS pressure boundary. The first criteria is required by the OTSG design. The second criteria is validated to ensure the transient response remains within analyzed limits.

During the initial stages of an overcooling event, the SG tubes become cooler than the surrounding SG shell. Because the upper and lower tubesheets are constrained at the edge by the SG shell, a tensile load develops due to the decreasing temperature in the SG tubes relative to the SG shell temperature. Later in the recovery phase of the event a compressive load can develop if the SG tube temperatures increase due to RCS temperatures returning to a normal zero power condition with a depressurized SG secondary allowing the SG shell temperature to decrease. This compressive stress develops as the SG shell cools by steam cooling and ambient heat losses through the insulation.

The Oconee replacement OTSGs have a compressive tube stress analytical limit with a limiting tube-to-shell temperature difference of +343°F (defined by the RCS at 555°F and the SG shell at 212°F), and a tensile tube stress analytical limit with a limiting tube-to-shell temperature difference of -375°F (defined by the large break LOCA tube stress analysis). The overcooling analyses demonstrate adequate margin to these values.

The generic pressurized thermal shock guidance for the B&W designed lowered loop design currently requires the operator to stabilize the plant and perform a one hour "soak" to allow thermal gradients to normalize. This guidance was developed by the vendor considering normal plant equipment is available. The intent of this guidance is incorporated into the tornado MSLB analyses.

For overheating events with an intact SG secondary, such as the limiting feedwater line break (FWLB) cases evaluated for the tornado LAR, RCS and SG tube temperatures tend to remain within normal bounds. Validating the thermal stress limits is not required for these events.

For overheating events with an intact SG secondary such as the limiting FWLB cases evaluated for the tornado LAR, RCS and SG tube temperatures do increase but not sufficiently to approach the compressive temperature limit. Similarly, the tensile limits are not approached for an event where RCS temperatures are increasing.

Consider a scenario where the SG shell temperature is controlled by the saturated steam temperature, and RCS temperatures limited by saturated liquid temperatures at the pressurizer safety valve lift setpoint. Assuming a SG saturation temperature of 550°F, RCS temperatures would need to approach 890°F before a compressive tube load issue was present or decrease to 175°F to reach the tensile tube load. These conditions are well outside of the expected response for a tornado-induced overheating event.

Therefore, the thermal stress related criteria are not used for overheating transients.

RAI 10

Chapter 9.6.1 of the Oconee UFSAR states that the SSF is designed to:

1. Maintain a minimum water level above the reactor core, with an intact Reactor Coolant System, and maintain Reactor Coolant Pump Seal cooling.
2. Assure natural circulation and core cooling by maintaining the primary coolant system filled to a sufficient level in the pressurizer while maintaining sufficient secondary side cooling water.
3. Transfer decay heat from the fuel to an ultimate heat sink.
4. Maintain the reactor 1% shutdown with the most reactive rod stuck fully withdrawn, after all normal sources of RCS makeup have become unavailable, by providing makeup via the Reactor Coolant Makeup Pump System which always supplies makeup of a sufficient boron concentration.

The above criteria are different than the acceptance criteria given on Page 17 of the Enclosure to the LAR. The NRC staff requests the licensee clarify whether the current UFSAR SSF criteria are applicable and met by the existing analysis. If the current criteria are not applicable to tornado events, then the NRC staff requests the licensee to also justify why these criteria are no longer needed.

RAI 10 Response:

The criteria described in UFSAR 9.6.1 are specific to the SSF for fire and TB flood and are not applicable to the SSF for tornado. As described within UFSAR 9.6.2, the SSF was not originally licensed to mitigate a tornado. The SSF ASW system was later identified after Three Mile Island as an alternative for EFW tornado missile protection vulnerabilities. UFSAR 9.6.2 documents that this licensing action did not specify a tornado missile event or define a tornado missile mitigation strategy. Using a probabilistic approach, it solely focused on ensuring that a secondary side heat removal path is adequately designed to withstand the effects of tornado missiles.

As part of the SSF licensing history, the acceptance criteria for the SSF have varied based on the event or scenario. The varying acceptance criteria for the events and scenarios associated with the SSF are detailed within UFSAR 9.6.2. As part of this licensing action, the tornado acceptance criteria for the SSF was modeled after the SBO success criteria. As described within UFSAR 9.6.2 for SBO, "The success criteria is to maintain the core covered..."

Specific to the analyses performed for tornado, the acceptance criteria are:

- Verify the core remains intact and in a coolable core geometry and verify minimum DNBR limits ensures sufficient core coverage.

- Verify that the peak allowable RCS pressure is not exceeded; the steam generators remain intact; and the RCS remains within acceptable pressure and temperature limits ensures the capability to establish long term core cooling, thereby maintaining core coverage.

The individual system specific design parameters for the SSF subsystems (seal injection, decay heat removal, pressurizer level control, etc.) were not detailed in the acceptance criteria as the bounds of their limits and operation may vary based on the overcooling or overheating scenarios evaluated. The results from the various safety analyses demonstrate the SSF systems and operator guidance can be used to successfully mitigate a Tornado by maintaining core cooling and coverage.

RAI 11

Page 6 of Attachment 6 to the LAR states, "The steam line ADVs [Atmospheric Dump Valves] (or other steam flow paths) are included in the overcooling analysis for examining long term recovery actions for single MSLB cases, and are not credited in the mitigation phase of the analysis."

Previously (in 2008) one of the licensee commitments was to protect the ADVs from tornado, but the commitment was withdrawn. LAR reference No. 41 (i.e., Letter to the U.S. Nuclear Regulatory Commission from Thomas D. Ray, Vice President, Oconee Nuclear Station, Duke Energy Carolinas, LLC, "Revision to Tornado/HELB Mitigation Strategies and Regulatory Commitments," dated November 15, 2017" (ML17333A120)) clarifies the Duke Energy decision to not install MSIVs nor tornado protect the ADVs. In addition, the ADVs are not included in the TORMIS analysis to justify probabilistically. The NRC staff requests the licensee to justify use of the ADVs in the analysis given they are not protected from tornado or considered in the TORMIS analysis.

RAI 11 Response:

The statement in Attachment 6 to the LAR was intended to mean that even though the ADVs are included in the TORMIS model, they were not used or needed in the analysis to demonstrate that the unit could be placed and maintained in a safe shutdown (SSD) condition from the SSF following a single MSLB.

RAI 12

Page 4 of Attachment 7 to the LAR states that the goal of the operator guidance assumed in the analysis is to stabilize the plant by maintaining RCS temperature between 325°F - 350°F and pressure between 650 psig - 700 psig. The NRC staff requests the licensee to confirm if this can be accomplished without the use of the ADVs, as they are unprotected and may be damaged in the tornado (see RAI-11 above). If these conditions cannot be met without use of the ADVs, then the NRC staff requests the licensee to also explain how the RCS would respond, what conditions would be achieved, and how the acceptance criteria are still met.

RAI 12 Response:

Page 4 of Attachment 7 is referring to an overcooling scenario. For overcooling, the plant can be stabilized without the use of ADVs. In addition, the ADVs are not required for any other tornado scenarios.

RAI 13

The licensee's LAR summary describes the future use of TORMIS. The licensee notes that modifications are being performed under 10 CFR 50.59 and that their approval is not a part of this LAR. The licensee credits some plant modifications to be physically protected or evaluated in the TORMIS model. The licensee appears to have proposed allowing the option to evaluate the future plant modifications in the TORMIS model. However, the TORMIS methodology is only to be used on existing plant structures' and components' nonconformances. RIS 2008-14 states, "TORMIS acceptance criteria are based on the cumulative effects of tornado missile damage to all safety-related SSCs that are not provided positive protection. Therefore, when using TORMIS to address any additional tornado missile vulnerabilities that are identified in the future, the analysis should include those SSCs that were previously analyzed."

The TORMIS safety evaluation report (SER) (i.e., Letter from L. S. Rubenstein (U. S. NRC) to F. J. Miraglis (U. S. NRC), "Safety Evaluation Report -- Electric Power Research Institute (EPRI) Topical Reports Concerning Tornado Missile Probabilistic Risk Assessment (PRA) Methodology," dated October 26, 1983, ADAMS Accession No. ML080870291) stated that the use of TORMIS should be limited to the evaluation of specific plant features where additional costly tornado missile protective barriers or alternative systems are under consideration.

Therefore, to evaluate whether the use of TORMIS will be consistent with the position that was stated in the TORMIS SER, the NRC staff requests the licensee to:

(1) confirm that TORMIS will not be used to temporarily or permanently eliminate existing barriers that are credited for providing tornado missile protection,

(2) confirm that the use of TORMIS will be limited to demonstrating adequate protection for existing SSCs that were originally required to be protected from tornado missiles in accordance with the plant design basis due to some oversight, are not adequately protected,

(3) describe how the cumulative effects of newly found non-conforming SSCs will be incorporated into TORMIS, and

(4) provide draft updates to the UFSAR based on these responses.

RAI 13 Response:

From the viewpoint of the new deterministic SSF mitigation pathway including the associated enhancement modifications, the TORMIS methodology is used to evaluate nonconformances of the tornado missile protection for the SSF and related structures and components. TORMIS is not and will not be used to temporarily or permanently eliminate existing barriers that are credited for providing tornado missile protection.

Physical Tornado missile protection is provided for SSF components within the SSF structure and containment structure. The TORMIS analysis addresses areas of the Cask Decontamination Tank Room (CDTR) and West Penetration Room (WPR) where some of the existing SSF components are located. The cabling for the enhancement modifications will be within the same target area as the existing SSF cabling. Therefore, the TORMIS analysis for the CDTR and WPR will similarly apply to the cabling of the enhancement modifications that routes through these same areas. TORMIS will not be used for future modifications beyond this LAR.

For non-conforming SSCs identified, TORMIS analysis may be used to evaluate the specific plant features where additional costly tornado missile protective barriers or alternative systems are under consideration. However, the impact of any new tornado missile vulnerabilities will be considered in combination with the impact of other SSCs that were previously analyzed using the TORMIS methodology. This approach ensures that the cumulative impact of tornado missile damage to all unprotected SSCs is evaluated against the TORMIS acceptance criteria consistent with the approved methodology.

The following paragraphs are added to the beginning of Section 3.5.1.3.1 of the proposed UFSAR changes to clarify the application of the TORMIS methodology for Oconee Nuclear Station. See Attachment 2 for markups.

> The TORMIS methodology provides an approach to demonstrate adequate protection for existing SSCs that were originally required to be protected from tornado missiles in accordance with the plant design basis but that are not adequately protected due to some oversight. The approved methodology does not allow TORMIS analysis to be used to temporarily or permanently eliminate existing barriers that are credited for providing tornado missile protection.
>
> The TORMIS acceptance criteria are based on the cumulative damage frequency of tornado missile damage to all safety-related SSCs that are not provided positive protection. Therefore, the impacts of all non-conforming items are combined so that the total missile damage frequency is evaluated against the acceptance criterion of 1E-06 per year. If additional non-conforming SSCs are identified in the future, TORMIS analysis may be used to evaluate these specific plant features and combine their damage impacts with the impacts of SSCs that were previously analyzed using the TORMIS methodology to determine if adequate protection is maintained.

RAI 14

In the NRC staff's TORMIS SER dated October 26, 1983 (ADAMS Accession No. ML080870281), Section III, "Conclusion," states, "Further, use of the EPRI PRAs or any tornado missile probabilistic study should be limited to the evaluation of specific plant features where additional costly tornado missile protective barriers or alternative systems are under consideration." RIS 2008-14 identifies issues raised by NRC staff during reviews of TORMIS applications, including that licensees did not fully address the fifth point identified in the SER nor explain how the methodology was implemented when the parameters used differed from those specified in the TORMIS methodology (e.g., inappropriately limiting the number of targets modeled).

In its application, the licensee provided a list of the revised tornado licensing basis and committed modifications, which include:

1. LAR Item 2.5.1 references Commitment 18T identified in the Tornado/HELB Commitment letter submitted to the NRC on November 15, 2017.
2. LAR Item 2.5.2 references commitment 19T previously identified in the Tornado/HELB Commitment letter submitted to the NRC on November 15, 2017
3. LAR Item 2.5.4 indicates to provide missile protection for the outdoor SSF diesel fuel oil tank fill and vent lines to prevent shear/perforation of the piping and subsequent rain water intrusion into the underground tank.
4. LAR Item 2.5.5 and 2.5.6 indicates to credit new pulsation dampener and letdown line.
5. LAR Item 2.5.7 references new QA-1 instrumentation to provide.

Other commitments or noted changes discussed in the application include:

6. Various instrumentation is credited for tornado damage stabilization (Section 3.2 table), but LAR Attachment 1 commitments are not yet installed. The Table located in Section 3.2 of LAR contains a note stating, "Note this will be upgraded or newly installed instrumentation."
7. In response to RIS 2008-14 documented in LAR Attachment 4, Section 8.3, Item 3.c, "proposing plant modifications," the licensee clarified that TORMIS is not being used as a justification to modify plant features to reduce, eliminate, or otherwise engineer the design of existing or new tornado missile protection features. However, the licensee indicated that additional modifications will be implemented that could impact the TORMIS analysis and results further when it stated, "Duke is enhancing the SSF capabilities through modifications implemented by 10 CFR 50.59. The routing of those modifications has been or will be included in the TORMIS evaluation as required."
8. RIS 2008-14 provides a discussion for including limiting the number of targets modeled, as indicated in Attachment 1, "Regulatory Commitments" of the LAR. The licensee proposed to provide missile protection for the outdoor SSF diesel fuel oil tank fill and vent lines to prevent shear/perforation of the piping and subsequent rain water intrusion into the underground tank. The licensee has proposed to complete this within 3 years after issuance of the SER. Therefore, the tank and main tank capacity will remain unprotected until the commitment is completed.

While these proposed modifications are presented as commitments in the LAR, the NRC staff requests the licensee to clarify whether and how these commitments are credited or may impact TORMIS results for the NRC staff to decide whether to escalate the commitments to requirements (e.g., new license conditions). Additionally, the NRC staff requests the licensee to discuss whether and how any of these unprotected components will be required or credited for a plant shutdown.

RAI 14 Response:

Each of the commitments are detailed below with respect to how they are credited or may impact TORMIS results, as well as whether and how any of these unprotected components will be required or credited for a plant shutdown.

1. LAR Item 2.5.1 is associated with the revision and clarification of the UFSAR, including incorporation of the TORMIS methodology, as provided in the LAR. The subject item represents implementation of the revised licensing basis upon receipt of the NRC safety evaluation. As such, the activity is not deemed to impact the TORMIS results nor will it introduce any additional credited components for a plant shutdown beyond those already presented within the LAR.

2. LAR Item 2.5.2 is associated with the revision and clarification of TS Bases relative to passive civil features as provided in the LAR. The subject item represents implementation of the revised licensing basis upon receipt of the NRC safety evaluation. As such, the activity is not deemed to impact the TORMIS results nor will it introduce any additional credited components for a plant shutdown beyond those already presented within the LAR.

3. LAR Item 2.5.4 represents a modification to provide missile protection for the outdoor SSF diesel fuel oil tank fill and vent lines to prevent shear/perforation of the piping and

subsequent rain water intrusion into the underground tank. The TORMIS analysis was performed from the perspective that the modifications for deterministic protection of these components were completed. Therefore, the frequency of tornado missile impact to the diesel fuel oil tank fill and vent lines were not included in the TORMIS analysis. Inclusion of these components without physical protection would be expected to increase the frequency of tornado missile impacts within the TORMIS results.

The proposed SSF deterministic mitigation strategy credits operation of the SSF diesel as a power source. Preventing shearing or perforation of the piping that connects to the SSF diesel fuel tank was the chosen method for assuring the diesel fuel source is not adversely affected by the introduction of potential rain water.

The modifications to the SSF diesel fuel oil tank fill and vent lines are to be performed under 10 CFR 50.59 to conform to the new licensing basis as described in the LAR. The new licensing strategy will become effective as the enhancement modifications are completed.

Until the enhancement modifications are completed, the existing Tornado licensing basis will be maintained during the implementation period. Oconee remains in compliance with its existing Tornado licensing basis of redundancy, diversity and separation as defined in UFSAR Chapter 3 to fulfill:

- Secondary Side Decay Heat Removal (SSDHR).
- RCMU.
- Reactor Coolant System Pressure Boundary Integrity.

Oconee has redundant systems normally available for SSDHR which are EFW, SSF ASW, and the PSW system which is an enhanced replacement for the original Station ASW system. The RCMU function can be provided either by SSF RCMU or by the HPI system. Both systems can provide RCP Seal Cooling while providing RCS makeup.

4. LAR Item 2.5.5 represents a modification to provide a new pulsation dampener on each unit's SSF RCMU system. The addition of new pulsation dampeners has been completed on all three units. The new pulsation dampeners are located within containment and are deterministically protected from the effects of tornado missiles. Therefore, their inclusion is not required within the TORMIS analysis and the TORMIS results are not impacted. The proposed SSF deterministic mitigation strategy credits the new SSF RCMU pulsation dampener to accommodate operation of the SSF RCMU system at lower range RCS pressures, which could potentially occur with a loss of the secondary side pressure boundary.

   LAR Item 2.5.6 represents a modification to replace the SSF letdown line on each unit's SSF RCMU system. Prior to the modification, the existing SSF letdown line valve cabling is unprotected. The new SSF letdown line valve cabling will also be unprotected but routed within the footprint of analyzed missile strikes associated with the unprotected portions of the SSF pathway detailed in the TORMIS analyses. The TORMIS conclusions are provided in the LAR. Therefore, although inclusion of the unprotected portions of the new SSF letdown line valve cabling is required within the TORMIS analyses, the TORMIS results are not impacted by the planned modifications. The proposed SSF deterministic mitigation strategy credits the new SSF letdown line to accommodate operation of the SSF RCMU system at lower range RCS pressures, which could potentially occur with a loss of the secondary side pressure boundary.

The modifications for the new SSF letdown lines are to be performed under 10 CFR 50.59 to the existing licensing basis. Following receipt of the SER, the new SSF letdown line is to be credited as described in the LAR.

As previously described above for Item 3 (LAR Item 2.5.4), until the enhancement modifications are completed, the existing Tornado licensing basis will be maintained during the implementation period. Oconee remains in compliance with its existing Tornado licensing basis of redundancy, diversity and separation as defined in UFSAR Chapter 3.

5. LAR Item 2.5.7 represents a modification to provide new QA-1 instrumentation in the SSF CR for SG pressure, nuclear instrumentation, core exit thermocouples, pressurizer temperature, and temperature compensated pressurizer level. The portions of the new SSF instrumentation which are unprotected are routed within the footprint of analyzed missile strikes associated with the unprotected portions of the SSF pathway detailed within the TORMIS analyses and represented by the TORMIS conclusions provided in the LAR. Therefore, although inclusion of the unprotected portions of the new SSF instrumentation is required within the TORMIS analyses, the TORMIS results are not impacted by the planned modifications. The proposed SSF deterministic mitigation strategy credits the new SSF instrumentation to provide similar instrumentation in the SSF as is provided in the Main Control Room for operation with a potential loss of the secondary side pressure boundary.

   The modifications for the new SSF instrumentation are to be performed under 10 CFR 50.59 to conform to the new licensing basis as described in the LAR. The new licensing strategy will become effective on a staggered per unit basis as the enhancement modifications are completed.

   As previously described above for Item 3 (LAR Item 2.5.4), until the enhancement modifications are completed, the existing Tornado licensing basis will be maintained during the implementation period. Oconee remains in compliance with its existing Tornado licensing basis of redundancy, diversity and separation as defined in UFSAR Chapter 3.

6. Section 3.2 of the LAR is related to the new SSF instrumentation described in LAR Item 2.5.7. The requested information for LAR Item 2.5.7 is provided within Item 5 above.

7. The statement included in Attachment 4 of the LAR which reads "The routing of those modifications has been or will be included in the TORMIS evaluation as required," is clarified to state "The routing of the unprotected portions of the SSF components associated with the new SSF tornado mitigation strategy has been included in the TORMIS evaluation described within this LAR."

8. The modifications to provide protection for the outdoor SSF diesel fuel oil tank fill and vent lines to prevent/perforation of the piping and subsequent rain water intrusion into the underground tank are described in LAR Item 2.5.4. The requested information for LAR Item 2.5.4 is provided within Item 3 above.

Note that the following information is added to UFSAR section 3.2.2 to describe the conforming modifications:

> The revised tornado mitigation strategies will be implemented when the SSF letdown line, SSF control room QA-1 instrumentation upgrade, and SSF diesel fuel oil tank fill/vent missile protection conforming modifications are completed.

RAI 15

UFSAR Section 3.2.2, "Tornado," clearly requires ability to shut down all three units in the event of a tornado. The UFSAR states that the Reactor Coolant System will not be damaged by a tornado, a loss of Reactor Coolant Pump (RCP) seal integrity was not postulated as part of the tornado design basis, and capability is provided to shutdown safely all three units. The UFSAR further states, "Capability is provided to shutdown safely all three units," which was intended to be supported by a qualitative assessment that, after a tornado, normal shutdown systems will remain available, or alternate systems will be available to allow shutdown of the plant.

In its application, the licensee requests approval for crediting the SSF as the assured mitigation path following a tornado with the assumed initial conditions of loss of all Alternating Current (AC) power to-all units with significant tornado damage to one unit.

A. The NRC staff requests the licensee to define "significant damage" and to clarify whether a damaged unit includes failure of all unprotected components on an affected unit, or a single worst case/bounding failure of exposed components on a damaged unit.

B. The NRC staff requests the licensee to describe any failure from a tornado event that might impact safe shutdown of all units. If one exists, the NRC staff requests the licensee to describe how it is analyzed.

C. The current licensing basis provides the option to credit other undamaged units for secondary makeup. The NRC staff requests the licensee to discuss whether any function or feature of the undamaged units would be credited to assist the degraded state of damaged unit for tornado mitigation recovery.

D. The NRC staff requests the licensee to discuss whether any systems shared between units are modeled in TORMIS and how an impact on multiple units is accounted for in TORMIS.

RAI 15 Response:

15A For the significantly damaged unit, significant tornado damage includes failure of all unprotected components directly associated with that unit that are not either physically protected or evaluated within the TORMIS methodology described within the LAR.

15B With the assumed initial conditions of loss of all AC power to all units with significant tornado damage to one unit, there are no failures from a tornado that might impact SSD of all three units.

15C While significant damage is only assumed to a single unit, all units are assumed to be impacted by the tornado. A station blackout is assumed for all three units. In response to the station blackout, secondary makeup would be required for all three units. The SSF ASW system is credited to provide secondary makeup to the significantly damaged unit. The margins associated with the SSF ASW decay heat removal to the significantly damaged unit are discussed within the response to RAI 3. The turbine driven EFW pump is assumed available for the initial secondary makeup to the units experiencing a blackout (i.e. non-significantly damaged units). Based on UFSAR 3.2.2, redundant and diverse sources of secondary makeup water include the EFW system, the other unit's EFW system, and the PSW system.

Beyond the shared systems, no function or feature of the undamaged units are credited to assist the degraded state of the significantly damaged unit for tornado mitigation. Treatment of the shared systems between the units are discussed below.

15D The unprotected portions of shared systems and features are modeled within TORMIS. The missile damage frequency for unprotected shared features are counted against each applicable unit. For example, the shared SSF ASW System takes suction from the large embedded CCW pipes. Although it is unlikely for tornado effects to impact a vent path to the embedded pipes that lead back to the CCW intake, the Unit 2 CCW surge lines were credited as providing the vent path for the SSF ASW system suction. The tornado missile damage frequency was calculated for the Unit 2 CCW surge lines but was applied beyond Unit 2. Given that the SSF ASW system was credited for any unit that may have experienced significant damage, the tornado missile damage frequency result for the Unit 2 CCW surge line was included as part of the total missile damage frequency result for Unit 1, Unit 2, and Unit 3.

RAI 16

The existing licensing basis defined in the UFSAR for tornado mitigation following a tornado provides redundancy, independence, and diversity with reliance on the combined capabilities of the tornado-protected station ASW system, EFW from the unaffected units, and the SSF ASW system. However, the proposed tornado mitigating strategy includes reliance on the use of SSF ASW alone, which degrades the level of defense-in-depth and subsequently increases risk. Additional risk results from the TSs allowing the SSF to be inoperable for 45 days.

The defense-in-depth philosophy has traditionally been applied in plant design and operation to provide multiple means to accomplish safety functions. System redundancy, independence, and diversity result in high availability and reliability of the function and also help ensure that system functions are not reliant on any single feature of the design. In the event of tornado, damage (ie. fallen trees, blocked access roads, etc...) could occur at the site resulting in limited ability for movement throughout the site. By relying only on a manually-operated SSF ASW system as the assured means of providing SSHR following a tornado, redundancy and diversity are lost. Eliminating the redundancy and diversity of the SSHR capability and RCMU makeup path provided under the existing licensing basis for tornado mitigation eliminates defenses-in-depth and increases risk related to achieving safe shutdown (SSD) following a damaging tornado. While the combination of physical protection and use of TORMIS to justify SSF meets the criteria for a fully protected system, additional information is needed to demonstrate that the planned use of the SSF alone is enough to overcome the loss of redundancy and diversity that would result from the proposed change.

The NRC staff requests the licensee to:

A. Discuss actions to retain tornado mitigation capability during the SSF 45-day inoperable periods of maintenance.

B. Discuss whether SSF is the only credited method to mitigate a tornado and shutdown all units. Discuss other protected systems or methods available to mitigate impacts of tornado.

C. Describe how defense-in-depth is maintained in the event of the SSF and related components being unavailable.

D. Describe post-72-hour actions and the long-term strategy.

RAI 16 Response:

16A and 16C The existing licensing basis defined in the UFSAR for tornado mitigation following a tornado provides redundancy, independence, and diversity. It is recognized that the system redundancy, independence, and diversity result in high availability and reliability of the function and also help ensure that system functions are not reliant on any single feature of the design. With the exception of crediting the SFP suction path to the HPI system, there are no changes proposed to eliminate the redundancy, independence, and diversity of the existing SSDHR capability and RCMU paths. Furthermore, there are no changes proposed to eliminate the tornado design criteria applied to the various systems, structures, or components as described in the UFSAR. As such, the high availability and reliability provided by the inherent design of the plant which includes redundancy, independence, and diversity ensures defense in depth is maintained if the SSF and related components become unavailable either prior to (i.e. SSF 45-day inoperable periods of maintenance) or during a tornado.

One of the redundant, diverse paths for makeup to the RCS is the BWST flowpath to the suction of the HPI pumps. The BWST has since been modified to withstand tornado missiles defined in UFSAR Section 3.8.4 (Table 3-23), such that the SFP is not expected to be needed as a redundant or diverse suction path for the HPI pumps. FLEX equipment is also available as a viable beyond design basis event mitigation option.

16B A new mitigation strategy is being defined for Oconee that is deterministic. Although diverse means of primary makeup, secondary decay heat removal, and electrical power may remain available during a tornado, only one protected deterministic strategy will be credited within the plant licensing. The SSF is credited for establishing and maintaining SSDHR and RCMU up to 72 hours following a damaging tornado. Currently, the CLB is a combination of probabilistic, diversity, and defense-in-depth strategies addressing the capability to provide SSD of the ONS units. The establishment of a tornado deterministic path provides clarity with respect to the licensing basis. In addition, the high availability and reliability which is provided by the inherent design of the plant which includes redundancy, independence, and diversity is not changed. Furthermore, there are no changes proposed to eliminate the tornado design criteria applied to the various systems, structures, or components as described in the UFSAR. The sources of secondary makeup include: 1) the Emergency Feedwater system including the capability to cross connect from another unit, 2) the PSW system, and 3) the SSF ASW system capable of being powered by the SSF diesel. The sources of primary makeup include: 1) the SSF Reactor Coolant Makeup Pump supplied from the Spent Fuel Pool and capable being powered from the SSF diesel and 2) An HPI pump supplied from the Borated

Water Storage Tank. FLEX equipment is also available as a viable beyond design basis event mitigation option.

16D The SSF is designed to maintain a SSD condition for a period of 72 hours following a fire or turbine building flood, and for a period of 4 hours following an SBO. Consistent with other SSF scenarios, the tornado mitigation strategy credits the SSF for establishing and maintaining SSD up to 72 hours following a tornado. The existing licensing basis defined in the UFSAR for tornado mitigation following a tornado provides redundancy, independence, and diversity. The systems which provide inherent plant redundancy, independence, and diversity include such capabilities as long-term decay heat removal, reactor coolant makeup, and plant cooldown. With the exception of the SFP suction path to the HPI system, there are no changes proposed to eliminate the redundancy, independence, and diversity of the existing SSDHR capability and RCMU makeup paths. There are also no changes proposed to eliminate the tornado design criteria applied to the various systems, structures, or components as described in the UFSAR. Redundancy, independence, and diversity of the plant provide means of long term response for tornado mitigation.

The proposed changes to the UFSAR have been revised as follows to provide clarification based on the responses to RAI 16:

- *Add the following paragraph to UFSAR Section 3.2.2 at the end of Insert 1:* "In addition to the SSF deterministic capability to mitigate a tornado, the inherent plant design of system redundancy, independence, and diversity is maintained for reasonable assurance that sufficient primary and secondary makeup is available following a tornado. Though all features of the inherent plant design are not tornado proof, their collective capabilities result in high availability and reliability to ensure that system functions are not reliant on any single feature of the design. As such, the high availability and reliability provided by the inherent design of the plant which includes redundancy, independence, and diversity ensures defense in depth is maintained if the SSF and related components become unavailable either prior to or during a tornado. The sources of secondary makeup include: 1) the Emergency Feedwater system including the capability to cross connect from another unit, 2) the PSW system, and 3) the SSF ASW system capable of being powered by the SSF diesel. The sources of primary makeup include: 1) the SSF Reactor Coolant Makeup Pump supplied from the Spent Fuel Pool and capable being powered from the SSF diesel and 2) A High Pressure Injection (HPI) pump supplied from the Borated Water Storage Tank. Note that in addition to their normal and emergency power sources, the "A" and "B" HPI pumps can be powered from the PSW switchgear."
- *Eliminate the addition to UFSAR 9.6.2 section "EFW Tornado Missile Design Criteria".*
- *Eliminate the revision to UFSAR 10.4.7.1.*
- *Eliminate the revision to UFSAR 10.4.7.3.6.*

RAI 17

Section III, "Conclusion," of the NRC staff's SER dated October 26, 1983, ADAMS Accession No. ML080870291), appoving the TORMIS methodology states: "... Further, use of the EPRI PRAs or any tornado missile probabilistic study should be limited to the evaluation of specific plant features where additional costly tornado missile protective barriers or alternative systems are under consideration." As discussed in RIS 2008-14, the NRC staff noted that licensees did not fully address the fifth point identified in the SER nor explain how the methodology was implemented when the parameters used differed from those specified in the TORMIS methodology.

Section 3.1, "RCS T-H Analysis," of the LAR states, "The Main Feedwater and Main Steam piping located outside containment are not protected from tornado missiles. Therefore, these piping systems may or may not remain intact following a tornado strike."

Section 5.2 "Determination of Safety Targets," of Attachment 4 of the LAR references the following components that are "Unprotected SSCs that if damaged could fail the SSF Mitigation Strategy":

- Main Steam Relief Valves (MSRVs) - damage preventing adequate steam relief for SSDHR,
- Main Steam header in EPR - damage causing pipe rupture affecting SSF equipment in WPR, and
- Main Feedwater headers in EPA - damage causing pipe rupture affecting SSF equipment in WPR.

The NRC staff requests the licensee to discuss whether the complete portion of the exposed components on the main steam and feedwater systems referenced in Section 3.1 are included and analyzed in the TORMIS analysis.

RAI 17 Response:

The portions of main steam and feedwater piping located inside the East Penetration Room (EPR) are included in TORMIS models for each respective unit. A tornado induced rupture of main steam or main feedwater piping inside the EPR is postulated to fail the barrier separating the EPR and WPR potentially exposing SSF equipment for that unit to adverse environmental conditions. Damage to the portions of main steam and main feedwater piping outside the EPRs will dissipate the energy to either the turbine building or the outside environment and is not postulated to cause damage to the West Penetration Room barrier or pose an adverse environmental condition to SSF equipment located there. Therefore, the portions of main steam and main feedwater piping outside the EPRs are not included in the Oconee TORMIS model.

Additional Requested Information

Please note that section 2.6, pages 17 and 18 of the Tornado LAR dated September 14, 2018 have the revised UFSAR description for UFSAR Section 9.6.2. Some information was inadvertently omitted and some has been deleted with the response to RAI 9 above.

The information in UFSAR Section 9.6.2 (page 17, section 2.6 of the LAR Enclosure) should match the information contained in the red marked UFSAR section in Attachment 2 and should read as follows:

> "The tornado initial conditions are defined for the unit(s) as MODE 1, 102% rated thermal power at end of core life (690 effective full-power days). The tornado is assumed to leave one unit significantly damaged and a loss of all AC power to all three units. Two bounding analyses were performed.....

> Following a tornado induced overcooling event the unit may experience a minor return to power of short duration. There are no consequences associated with the return to power due to the very low power level generated. The SSF is not required to meet the single failure criterion or the postulation of the most reactive rod stuck fully withdrawn…."

In section 2.6, page 19 of the Tornado LAR dated September 14, 2018, it was proposed that UFSAR section 9.7.1 be revised to add the following information from UFSAR Section 3.2.2:

> "An external source of cooling water is not immediately required due to the large quantities of water stored underground in the intake and discharge CCW piping. The stored volume of water in the intake and discharge lines below elevation 791 ft would provide sufficient cooling water for all three units for at least 30 days after trip of the three reactors."

This statement will not be relocated as part of the Tornado LAR package. It will be left in UFSAR section 3.2.2. The revised section is provided in Attachment 2.