UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 15, 2021

Mr. J. Ed Burchfield, Jr.
Vice President, Oconee Nuclear Station
Duke Energy Carolinas, LLC
7800 Rochester Highway
Seneca, SC 29672-0752

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 - ISSUANCE OF AMENDMENT NOS. 421, 423, AND 422 RE: REVISION OF LICENSING BASIS FOR HIGH ENERGY LINE BREAKS OUTSIDE OF CONTAINMENT (EPID L-2019-LLA-0184)

Dear Mr. Burchfield:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 421 to Renewed Facility Operating License (RFOL) No. DPR-38, Amendment No. 423 to RFOL No. DPR-47, and Amendment No. 422 to RFOL No. DPR-55 for Oconee Nuclear Station (Oconee), Units 1, 2, and 3, respectively. The amendments are in response to the application from Duke Energy Carolinas, LLC dated August 28, 2019, as supplemented by letters dated June 15 and September 17, 2020.

The amendments consist of changes to the current licensing basis in the Updated Final Safety Analysis Report (UFSAR) with regards to High Energy Line Breaks outside of the containment building for Oconee, Units 1, 2, and 3.

The NRC has determined that the related safety evaluation (SE) contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations* Section 2.390, "Public inspections, exemptions, requests for withholding." The proprietary information is indicated by text enclosed within double brackets. Accordingly, the NRC staff has also prepared a non-proprietary publicly available version of the SE, which is provided as Enclosure 4. The proprietary version of the SE is provided as Enclosure 5.

A Notice of Issuance will be included in the Commission's monthly *Federal Register* notice.

**Enclosure 5 to this letter contains proprietary information. When separated from Enclosure 5, this document is DECONTROLLED.**

If you have any questions, please contact me at (301) 415-1009 or by e-mail at Shawn.Williams@nrc.gov.

Sincerely,

*/RA/*

Shawn Williams, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, 50-287

Enclosures:
1. Amendment No. 421 to DPR-38
2. Amendment No. 423 to DPR-47
3. Amendment No. 422 to DPR-55
4. Safety Evaluation (non-proprietary)
5. Safety Evaluation (proprietary)

cc w/Enclosures 1, 2, 3, and 4: Listserv

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 421
Renewed License No. DPR-38

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility) Renewed Facility Operating License No. DPR-38 filed by Duke Energy Carolinas, LLC (the licensee), dated August 28, 2019, as supplemented by letters dated June 15 and September 17, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 421, Renewed Facility Operating License No. DPR-38 is hereby amended to authorize revision to the Updated Final Safety Analysis Report (UFSAR), as set forth in the application dated August 28, 2019, as supplemented by letters dated June 15 and September 17, 2020. The licensee shall update the UFSAR to incorporate the changes as described in the application, supplements, and the NRC staff's safety evaluation attached to this amendment and shall submit the revised description authorized by this amendment with the next update of the UFSAR.

3. This license amendment is effective as of the date of its issuance and shall be implemented by the completion of refueling outage 1EC34 (fall 2026). The UFSAR changes shall be implemented in the next periodic update to the UFSAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: March 15, 2021

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 423
Renewed License No. DPR-47

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility) Renewed Facility Operating License No. DPR-47 filed by Duke Energy Carolinas, LLC (the licensee), dated August 28, 2019, as supplemented by letters dated June 15 and September 17, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 423, Renewed Facility Operating License No. DPR-47 is hereby amended to authorize revision to the Updated Final Safety Analysis Report (UFSAR), as set forth in the application dated August 28, 2019, as supplemented by letters dated June 15 and September 17, 2020. The licensee shall update the UFSAR to incorporate the changes as described in the application, supplements, and the NRC staff's safety evaluation attached to this amendment and shall submit the revised description authorized by this amendment with the next update of the UFSAR.

3. This license amendment is effective as of the date of its issuance and shall be implemented by the completion of refueling outage 2EC32 (fall 2025). The UFSAR changes shall be implemented in the next periodic update to the UFSAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: March 15, 2021

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 422
Renewed License No. DPR-55

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Renewed Facility Operating License No. DPR-55 filed by Duke Energy Carolinas, LLC (the licensee), dated August 28, 2019, as supplemented by letters dated June 15 and September 17, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 422, Renewed Facility Operating License No. DPR-55 is hereby amended to authorize revision to the Updated Final Safety Analysis Report (UFSAR), as set forth in the application dated August 28, 2019, as supplemented by letters dated June 15 and September 17, 2020. The licensee shall update the UFSAR to incorporate the changes as described in the application, supplements, and the NRC staff's safety evaluation attached to this amendment and shall submit the revised description authorized by this amendment with the next update of the UFSAR.

3. This license amendment is effective as of the date of its issuance and shall be implemented by the completion of refueling outage 3EC33 (spring 2026). The UFSAR changes shall be implemented in the next periodic update to the UFSAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: March 15, 2021

# ENCLOSURE 4

NON-PROPRIETARY SAFETY EVALUATION FOR

AMENDMENT NO. 421 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 423 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 422 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DUKE ENERGY CAROLINAS, LLC

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

Proprietary information pursuant to Section 2.390 of Title 10 of
the *Code of Federal Regulations* has been redacted from this document.

**Redacted information is identified by blank space enclosed within [[ double brackets ]].**

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION FOR

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AMENDMENT NO. 423 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-47

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DUKE ENERGY CAROLINAS, LLC

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

## 1.0 INTRODUCTION

By application dated August 28, 2019 (Reference 1), as supplemented by letters dated June 15 (Reference 2) and September 17, 2020 (Reference 3), Duke Energy Carolinas, LLC (Duke Energy, the licensee) requested changes to the Updated Final Safety Analysis Report (UFSAR) for the Oconee Nuclear Station (Oconee, ONS), Units 1, 2, and 3. Specifically, the licensee proposed changes that would revise the current licensing basis in the UFSAR with regard to High Energy Line Breaks (HELBs) outside of the containment building for Oconee, Units 1, 2, and 3.

The licensee's license amendment request (LAR) and supplements supersede, in its entirety, previous HELB license amendment requests, including responses to requests for additional information (RAI) to the U.S. Nuclear Regulatory Commission (NRC, the Commission). These prior submittals of information were provided to the NRC in letters dated November 30, 2006 (Reference 4), June 26, 2008 (Reference 5), December 22, 2008 (Reference 6), June 29, 2009 (Reference 7), September 2, 2009 (Reference 8), October 23, 2009 (Reference 9), June 24, 2010 (Reference 10), August 31, 2010 (Reference 11), December 7, 2010 (Reference 12), December 16, 2011 (Reference 13), January 20, 2012 (Reference 14), and March 1, 2012 (Reference 15). Some of the superseded documentation contains information not applicable to this review but that continues to pertain to other licensing actions. Additionally, the application dated August 28, 2019, contains discussions of past commitments related to HELBs that exist or have been closed. However, these past commitments are outside the scope of the current application review. Therefore, approval of this application does not constitute NRC acceptance or completion of the closure of any referenced past commitments.

From February 18 to May 14, 2020, the NRC staff conducted an audit to support its review of the amendment request, as discussed in the NRC staff's audit plan dated February 11, 2020 (Reference 16), and audit summary dated November 3, 2020 (Reference 17).
On September 8, 2020, the NRC staff published a proposed no significant hazards consideration (NSHC) determination in the *Federal Register* (85 FR 55514) for the proposed amendments that included the submittal dated August 28, 2019, and the supplemental letter dated June 15, 2020. The supplemental letter dated September 17, 2020, provided additional information that clarified the application, did not expand the scope of the application as noticed, and did not change the NRC staff's proposed NSHC determination as published in the *Federal Register*.

The NRC staff's safety evaluation (SE) contains proprietary information as originally submitted in the letter dated August 28, 2019, as supplemented by letters dated June 15, 2020 and September 17, 2020. Proprietary information withheld under Title 10 of the *Code of Federal Regulations* Section 2.390 is identified by text enclosed within double brackets as shown here **[[ ]]**.

## 2.0 REGULATORY EVALUATION

### 2.1 Background

Oconee pre-dates the NRC General Design Criteria (GDC) and the plant was licensed to principal design criteria derived from the draft Atomic Energy Commission (AEC) Design Criteria published in the *Federal Register* on July 11, 1967 (32 FR 10213). The GDC that constitute the principal design criteria for Oconee are those described in the Updated Final Safety Analysis Report, Revision 28 (UFSAR) (Reference 18), Chapter 3.1, and in applicable UFSAR sections. These GDC establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety, that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

The Giambusso letter (Reference 19), as modified by the Schwencer letter (Reference 20), provided a list of requested information that would demonstrate with reasonable assurance that the plant could be safely shut down and maintained in a safe shutdown condition following credible failures of piping in high energy systems.

In response to the Giambusso/Schwencer letters, the licensee originally documented its analyses of the dynamic effects resulting from postulated piping breaks outside of the containment building in Duke Energy mechanical design study (MDS) Report No. OS-73.2 dated April 25, 1973 (Reference 21), and supplements dated June 22, 1973 (Reference 22) and March 12, 1974 (Reference 23). In 1998, gaps in documentation with this original HELB analysis were identified and Duke Energy began a project to update the original HELB work. Duke Energy submitted to the NRC proposed mitigation strategies and regulatory commitments for resolution of issues related to HELBs outside of containment on November 30, 2006. By letter dated June 26, 2008, Duke Energy submitted a license amendment request to revise the Oconee licensing basis for HELB mitigation. Subsequently, Duke Energy submitted updated HELB mitigation strategies and regulatory commitments by letter dated November 15, 2017 (Reference 24). The current proposed license amendment request supersedes the license amendment request dated June 26, 2008, and related correspondence.

In its LAR, the licensee requested to establish normal plant systems, protected service water (PSW) system, and/or shutdown standby facility (SSF) as the assured mitigation path following a HELB. Specifically, the licensee proposed the following changes for HELB mitigation based on the location of the piping break:

1. Crediting the PSW system or SSF for HELB mitigation when a HELB results in the loss of plant systems needed for safe shutdown (SSD) inside the turbine building (TB).

2. Crediting normal plant systems (i.e., high pressure injection (HPI) and emergency feedwater (EFW)) or the SSF for HELB mitigation when a HELB results in the loss of plant systems needed for SSD inside the auxiliary building (AB).

3. Crediting normal plant systems for HELB mitigation when a HELB occurs outside of the TB and AB.

4. UFSAR revisions that will incorporate the HELB strategy into the licensing basis.

5. Time critical operator actions (TCAs) associated with the prescribed HELB mitigation strategies.

6. Exclusion of systems whose operating time at high energy (HE) conditions is less than 1 percent of the total unit operating time.

7. Exclusion of systems whose operating time at HE conditions is less than approximately 2 percent of the total system operating time.

8. Elimination of arbitrary intermediate breaks in American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Section III-Class 2 and Class 3 equivalent piping. Intermediate breaks are postulated where calculated longitudinal stress for the applicable load cases (internal pressure, dead weight (gravity), thermal, and seismic (OBE [operating basis earthquake]) conditions) exceed $0.8(S_a + S_h)$.[1]

9. Intermediate breaks in non-rigorously analyzed piping are postulated in accordance with Branch Technical Position (BTP) Mechanical Engineering Branch (MEB) 3-1 Section B.1.c(2)(b)(i).

10. Elimination of critical cracks at the most adverse location in ASME B&PV, Section III - Class 2 and Class 3 equivalent piping. Critical cracks are postulated at axial locations where the calculated stress for the applicable load cases (internal pressure, dead weight (gravity), thermal, and seismic (operating basis earthquake (OBE)) conditions) exceed $0.4(S_a + S_h)$. Critical cracks are not postulated at locations of terminal ends.

[1] $S_h$ is the stress calculated by the rules of NC-3600 and ND-3600 for Class 2 and 3 components, respectively, of the ASME B&PV Code, Section III Winter 1972 Addenda. $S_a$ is the allowable stress range for expansion stress calculated by the rules of NC-3600 of the ASME B&PV Code, Section III, or the USA Standard Code for Pressure Piping, ANSI B31.1.0-1967.

11. The effects of the postulated intermediate breaks bound the effects from critical cracks; therefore, critical cracks are eliminated from evaluation in non-rigorously analyzed piping.

12. Determination of the effective length of jets from a break or critical crack in accordance with NUREG/CR-2913 (Reference 25).

13. Strategies to achieve and maintain Cold Shutdown (CSD) conditions.

14. Elimination of the TCA to cross-connect EFW.

15. Elimination of the TCA to manually start the turbine-driven EFW pump locally.

16. Reactor coolant system (RCS) acceptance criteria, as specified in Attachment 6 of the LAR.

## 2.2 System Design and Operation

In order to maintain an SSD state following a HELB outside containment, operators must maintain certain key safety functions. These functions include reactor core reactivity control, decay heat removal, and RCS inventory control, which supports the reactivity control and decay heat removal functions. For events that could involve a reactor cooldown, reactivity control requires addition of soluble neutron poison (borated water) to the RCS. Decay heat removal requires a supply of water that can be fed to the steam generators (SGs) and a heat sink for the steam generated from decay heat. Finally, reactor coolant inventory control requires high head injection of coolant to compensate for thermal contraction and control of RCS leakage.

The proposed strategy for HELBs resulting in the loss of plant systems inside the TB needed for SSD is to rely on the available SSF and PSW systems, which are located outside the TB. For HELBs within the AB, which includes the East Penetration Room (EPR), the licensee proposed crediting normal plant systems and the SSF for event mitigation. For HELBs in areas outside the TB and AB, the licensee's strategy credits normal plant systems for event mitigation. The following systems perform important functions in mitigation of postulated HELBs.

### 2.2.1 Normal Plant Systems

In Section 2.1.3 of the Enclosure to the LAR, the licensee stated that the normal plant systems and related support systems may remain available for HELB mitigation. These systems can be used for plant cooldown and the establishment of CSD.

The licensee defined the normal modes of operation in Attachment 10 to the LAR as follows:

**MODE 1 (Power Operation)** – Operation of an ONS unit with the following conditions: $k_{eff} \geq 0.99$ and the percent Rated Thermal Power level > 5.

**MODE 2 (Startup)** – Operation of an ONS unit with the following conditions: $k_{eff} \geq 0.99$ and the percent Rated Thermal Power level ≤ 5.

**MODE 3 (Hot Standby)** – Operation of an ONS unit with the following conditions: $k_{eff} < 0.99$ and the average RCS Temperature is ≥ 250°F.

**MODE 4 (Hot Shutdown)** – Operation of an ONS unit with the following conditions: $k_{eff} < 0.99$ and the average RCS Temperature (T) is 250°F > T > 200°F.

**MODE 5 (Cold Shutdown)** – Operation of an ONS unit with the following conditions: $k_{eff} < 0.99$ and the average RCS Temperature is ≤ 200°F. To be in Mode 5, all reactor vessel (RV) head closure bolts must be fully tensioned.

2.2.1.1 High Pressure Injection System

The HPI System is described in Section 9.3.2 of the UFSAR. The HPI System controls the RCS coolant inventory, provides the seal water for the reactor coolant pumps (RCPs), and recirculates RCS letdown for water quality maintenance and reactor coolant boric acid control. The licensee stated that the HPI System consists of two independent trains, each of which splits to discharge into two RCS cold legs, so that there is total of four HPI injection lines. The HPI System contains three HPI pumps; two of the pumps are normally aligned to automatically support HPI train A and the discharge flow path for the third pump is normally aligned to automatically support HPI train B. The discharge flow paths can be manually aligned such that each of the HPI pumps can provide flow to either train. The two HPI trains are designed and aligned such that they are not both susceptible to any single active failure (SAF) including the failure of any power operating component to operate or any single failure of electrical equipment.

2.2.1.2 Emergency Feedwater System

The EFW System is described in Section 10.4.7 of the UFSAR. It automatically supplies feedwater (FDW) to the SGs to remove decay heat from the RCS upon the loss of normal FDW supply. The EFW System includes two Alternating Current (AC) motor-driven pumps and one turbine-driven pump per unit that is independent of AC power. Each motor-driven pump normally serves a separate SG while the turbine-driven pump normally serves both SGs. The discharge header of each EFW System can be cross connected making each system capable of supplying any unit. Following a reactor trip, the EFW System can provide sufficient FDW to maintain Hot Standby for at least 4 hours with or without offsite power.

2.2.2 <u>Protected Service Water System</u>

The PSW system is a high head, moderate volume system that is shared among the three Oconee Units and is described in Section 9.7 of the Oconee UFSAR. The system is designed as a standby system for use under emergency conditions and provides added defense-in-depth protection by serving as a backup to existing safety systems. This provides an alternate means to achieve and maintain SSD conditions for one, two, or three units following postulated scenarios that damage essential systems and components normally used for SSD. In the event of a HELB in the TB, the PSW system is intended to mitigate postulated events that result in the damage to essential equipment located inside the TB that are used to achieve and maintain SSD conditions.

The PSW mechanical system components and piping are located primarily in the AB, and electrical switchgear is primarily located in a separate, shared PSW building. The system provides backup SG FDW capabilities using primary and booster PSW pumps located in the AB and backup electrical power capability for the HPI pumps. The PSW system supplies electrical power to the PSW pumps and backup power to the HPI pumps from independent switchgear in the PSW building. The PSW primary and booster pumps take suction from Unit 2 Condenser Circulating Water (CCW) piping and discharge into the SG for each unit via the EFW piping from just outside containment. The PSW pumps are operated from the Unit 2 control room. For extended operation, the PSW portable pump with a flow path capable of taking suction from the intake canal and discharging into the Unit 2 CCW piping is designed to provide a backup supply of water to the PSW system in the event of loss of CCW and subsequent loss of CCW siphon flow.

The PSW system capabilities are redundant to and diverse from the SSF system. Its mission is to achieve and maintain SSD by maintaining reactivity control (shutdown margin), RCS inventory, and reactor coolant temperature and pressure within acceptable limits by supporting decay heat removal through the SGs. The PSW primary and booster pumps, motor-operated valves, and solenoid valves required to bring the system into service, are controlled from the main control rooms (CRs). Check valves and manual handwheel operated valves are used to prevent back-flow, to accommodate testing, or for system isolation.

2.2.3 Standby Shutdown Facility

The SSF is designed as a standby system for use under certain emergency conditions and is described in Section 9.6 of the Oconee UFSAR. The system provides additional defense-in-depth protection for the health and safety of the public by serving as a backup to existing safety systems. It provides an alternate means to achieve and maintain the unit(s) in Mode 3 with average RCS temperature ≥ 525°F (unless the initiating event causes the unit(s) to be driven to a lower temperature) following a fire, TB flood, and station blackout (SBO) events. The SSF is designed to maintain the reactor(s) in an SSD condition for a period of 72 hours following a fire or TB flood, and for a period of 4 hours following an SBO. The main components of the SSF are the auxiliary service water (ASW) system, SSF Portable Pumping system, SSF Reactor Coolant Makeup (RCMU) system, SSF power system, and SSF instrumentation.

The SSF is a reinforced concrete structure consisting of a diesel generator room, electrical equipment room, mechanical pump room, control room, central alarm station (CAS), and ventilation equipment room. PSW and SSF ASW systems provide cooling water from Lake Keowee via CCW Unit 2 Piping. SSF connection into EFW piping from the SSF ASW pump is located within the AB and connected upstream of SG's and downstream of EFW pumps.

During an SSF Event, the RCS transfers heat from the reactor to the SGs via natural circulation flow. RCS natural circulation flow is used to transfer heat to the SGs during an SSF Event instead of forced RCS flow because the RCPs will be unavailable. RCS natural circulation flow is established when SSF ASW system flow is initiated to a unit's SGs. Once SSF ASW flow is established, the SSF control room Operator will control the SSF ASW system flow rate provided to the SGs based on RCS parameters and steam generator level.

The SSF ASW system is a high head, high volume system designed to provide FDW to the SGs for adequate decay heat removal (DHR) for three units during a loss of normal AC power in conjunction with the loss of the main feedwater (MFDW) and EFW systems. The SSF ASW pump utilizes a suction supply of lake water from the embedded Unit 2 CCW piping.

The SSF Portable Pumping system, which includes a submersible pump and a flow path capable of taking suction from the intake canal and discharging into the Unit 2 CCW line, is designed to provide a backup supply of water to the SSF in the event of loss of CCW and subsequent loss of CCW siphon flow. The SSF Portable Pumping system is installed manually in accordance with procedures. The licensee credits this path for long-term SSF usage by bringing water from the lake to CCW piping, which is the normal inventory supply for the SSF.

The SSF RCMU system is designed to supply makeup to the RCS and RCP seal cooling if normal makeup systems are unavailable. An SSF RCMU pump located in the reactor building (RB) of each unit supplies makeup to the RCS should the normal makeup system flow and seal cooling become unavailable. It provides seal injection flow in the event that HPI (normal seal supply) is unavailable. The SSF requires manual activation and can be activated if emergency systems are not available.

### 2.3 Description of the Proposed Changes

The LAR would modify the plant licensing basis for Oconee Units 1, 2, and 3, by revising the UFSAR to identify the necessary SSCs to implement the HELB mitigation strategies and describe the methodology and results of the analysis performed to evaluate the protection of the reactor's SSCs from the effects of HELBs.

In its LAR, the licensee described conforming actions (previously commitments) and modifications that it was making to the plant to enhance the overall design and safety margin. The licensee stated that it was not applying for NRC approval of the modifications; rather, the licensee determined that the modifications could be performed via 10 CFR 50.59, "Changes, tests, and experiments," without prior NRC approval. The licensee proposed to credit the plant modifications to enhance the plant's capability to withstand the effects of a HELB. The modifications include: installation of the new SSF letdown line; installation/upgrade of SSF control room QA-1 instrumentation; upgrade of the inlet isolation valves for the Unit 1 letdown coolers; upgrade to the heating, ventilation, and air conditioning (HVAC) ducting impacting the control complex; upgrade of TB structural support columns; upgrade of suction valves to the Unit 2A & 2B HPI pumps; elimination of the Control Rod Drive (CRD) cross-connect between units; environmental qualification of SSF-related components located in each unit's AB; installation of HELB protected isolation for the alternate RB cooling system return piping alignment; and TCA validation. In its submittal, the licensee stated that the modifications will be completed prior to implementation of the LAR.

#### 2.3.1 Proposed UFSAR Changes

The licensee proposed to revise sections of the UFSAR, as follows (deletions in double strike-out; additions in double underline).

The licensee proposed to revise Section 3.6, "Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping," of the UFSAR, as follows:

> **3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping**
>
> Main Steam High Energy Line Breaks (MS HELBs) are not synonymous with Main Steam Line Breaks (MSLBs). The analyses and the treatment of MSLBs as described in UFSAR Sections 15.13 and 15.17 were required as part of the initial licensing of the Oconee Nuclear Station (ONS) units. The analyses were completed to evaluate the reactor core response to the resulting overcooling following the MSLB. The postulated break locations for the MSLB analyses described in Chapter 15 were not specified, and as such, damage from the

MSLB was not considered. The Giambusso/Schwencer letters (References 9 and 15) were released as construction of Unit 1 was nearing completion. These letters required that licensees consider damage following a postulated break, including those postulated in the MS system. These breaks were considered for different purposes using different assumptions and acceptance criteria. In cases where the potential damage postulated for a MS HELB was similar to the inputs and assumptions used in the MSLB analyses described in UFSAR Sections 15.13 and 15.17, those analyses were used as surrogates for the MS HELB analyses. In a similar manner, a Main Feedwater (MFDW) HELB is not synonymous with a Main Feedwater Line Break (MFLB). The analyses for a MFLB as discussed in UFSAR Section 10.4.7 were completed to evaluate the reactor core response to the overheating caused by the MFLB. The postulated break locations of the MFLBs described in Chapter 10 were not specified and damage from the MFLBs was not considered. However, for MFDW HELBs, the Giambusso/Schwencer letters required evaluation of specific locations and potential damage.

The licensee proposed to rename Section 3.6.1, as follows:

**3.6.1 Postulated Piping Failures in Fluid Systems Inside ~~and Outside~~ Containment**

The licensee proposed to delete current Section 3.6.1.3, "Protected Service Water (PSW) System."

The licensee proposed to delete current Section 3.6.1.4, "Safety Evaluation."

The licensee proposed to add new Section 3.6.2, "Postulated Piping Failures in Fluid Systems Outside Containment," as follows:

**3.6.2 Postulated Piping Failures in Fluid Systems Outside Containment**

The purpose of this description is to provide a comprehensive strategy for mitigating the potential adverse interactions caused by the Oconee Nuclear Station (ONS) postulated HELBs. The strategy provides an evaluation of the ONS postulated HELBs and describes the (as modified) ONS configuration for the identified HELBs. It also supersedes the analysis provided in the original 1973 ONS HELB analysis (References 1 and 9). The strategy identifies and describes the pathway to a safe shutdown (SSD) condition for any postulated HELB in any unit. HELBs are only postulated to occur during the normal operating configuration of the system with the unit operating at 100% rated thermal power level (full power).

The revised HELB mitigation strategies will be implemented when the following conforming actions are completed: installation of new standby shutdown facility (SSF) letdown line, installation/upgrade of SSF control room QA-1 instrumentation, upgrade of inlet isolation valves to the Unit 1 letdown coolers, upgrade to the heating, ventilation, and air conditioning ducting impacting the control complex, upgrade of turbine building (TB) structural support columns, upgrade of suction valves to the Unit 2A & 2B High Pressure Injection (HPI) pumps, elimination of the Control Rod Drive (CRD) cross connect between units, environmentally qualify SSF related components located in each Unit's auxiliary building (AB), provide HELB protected isolation for Alternate Reactor Building Cooling (RBC) System return piping alignment, and Time Critical Operator Actions (TCA) validation.

### 3.6.2.1 Identification of High Energy Lines

The following criteria are used to identify the high energy piping and the boundaries of the high energy portions of the systems:

- The high energy (piping) lines are those lines that during initial operating conditions, the fluid inside of the pipe has either or both of the following conditions:

  1. A normal operating temperature greater than 200°F.
  2. A normal operating pressure greater than 275 psig.

- The high energy section of any piping run shall extend from component to component. The high energy portion shall not terminate unless there is a termination at a vessel, a pump, a closed valve, or equivalent boundary.

- Piping downstream of a normally closed valve, that is the high energy boundary for a high energy piping run, is not postulated to be high energy due to potential leakage across the closed valve.

- High energy line boundaries are based upon the normal operating configuration of the system with the unit operating at 100% rated thermal power level (full power).

- Gas Systems (e.g. Nitrogen) and oil systems (e.g. Electro Hydraulic Control) are not identified as high energy systems because those systems possess limited energy.

### 3.6.2.2 Identification of High Energy Line Break Locations

The following criteria are used to identify the high energy piping break locations:

- HELBs of any type are not postulated on high energy piping that has a nominal size of (1) inch or less.

- HELBs and critical cracks are not postulated on high energy lines that operate at high energy conditions less than approximately 2% of the total system operating time.

- HELBs and critical cracks are not postulated on high energy lines that operate at high energy conditions less than 1% of the total plant (unit) operating time (Normal Plant Conditions).

- HELBs are postulated at the Terminal Ends of high energy piping runs.

- There is no American Society of Mechanical Engineers (ASME) Boiler and Pressure (B&PV) Code, Section III, Division 1-Class 1 equivalent piping outside of the containment building.

- For ASME B&PV, Section III-Class 2 and Class 3 equivalent piping that is seismically analyzed, HELBs are postulated at axial locations, where the calculated longitudinal stress for the applicable load cases (internal pressure, dead weight (gravity), thermal, and seismic (OBE) conditions) exceeds $0.8(S_a + S_h)$.

- For ASME B&PV, Section III-Class 2 and Class 3 equivalent piping that is seismically analyzed, critical cracks are postulated at axial locations where the calculated stress for the applicable load cases exceed $0.4(S_a + S_h)$. Applicable load cases include internal pressure, dead weight (gravity), thermal, and seismic (OBE). Critical cracks are not postulated at locations of terminal end or intermediate breaks.

- For branch connections where the branch line is included in the seismic stress analysis of the run piping, the stress criteria for seismically analyzed piping lines is used to determine HELBs.

- Breaks and critical cracks at closed valves are postulated as follows. The postulation of terminal end breaks at the first normally closed valve(s) separating portions of a system maintained pressurized during normal operations and portions of a system not maintained pressurized depends on whether the system has a seismic analysis that is continuous across the valve. For systems or portions of systems that are not seismically analyzed, breaks are postulated to occur at all piping girth welds in the system including those that attach to normally closed valves. For systems or portions of systems that are seismically analyzed, and the analysis is continuous across the normally closed valve, such that stresses can be accurately determined, break and crack locations are determined based on comparison to the intermediate break and crack stress thresholds.

- For piping that is not rigorously analyzed or does not include seismic loadings, HELBs are postulated at intermediate break locations as provided in BTP MEB 3-1, Section B.1.c.(2)(b)(i).

- For branches where both the main and branch runs are unanalyzed or where the stress at the branch connection is not accurately known, break locations are postulated on the branch and run sides of the connection.

- For piping that is not rigorously analyzed or does not include seismic loadings, critical cracks are not postulated since the effects of postulated HELBs on these piping runs will bound the effects from critical cracks.

- Actual stresses used for comparison to the break and crack thresholds are calculated in accordance with the ONS piping code of record, USAS

B31.1.0 (1967 Edition). Allowable stress values $S_a$ and $S_h$ are determined in accordance with USAS B31.1.0 or the USAS B31.7 (February 1968 draft edition with errata) code as appropriate.

- Moderate energy line breaks are not postulated. The HELB requirements for ONS only require compliance to the Giambusso/Schwencer letters. The requirements contained therein do not include postulation of moderate energy line breaks.

- High Energy Piping lines with an internal pressure at atmospheric or below (≤ 0 psig) are excluded from damage assessments due to insufficient energy to create pipe whip of jet impingement forces.

- For the MS penetrations into the containment structure, MS HELBs are postulated to occur at the outside face of the concrete containment structure.

- For the MFDW penetrations into the containment structure, MFDW HELBs are postulated to occur on the outside of the containment structure side of the Main Feedwater terminal/rupture/guard pipe restraint.

- For all other ASME B&PV, Section III-Class 2 equivalent piping penetrations into the containment structure, HELBs are postulated to occur at the outside face of the concrete containment structure.

### 3.6.2.3 Identification of High Energy Break Types

The following criteria are used to identify the high energy break types, required to be postulated at the identified break location in ONS. There are three (3) types of HELBs at ONS. They are circumferential breaks, longitudinal breaks, and critical cracks. The criteria for each break type are as follows:

- Circumferential Breaks are postulated in high energy lines that exceed one (1) inch nominal pipe size.

- Only circumferential breaks are postulated at terminal ends of high energy piping runs (Longitudinal breaks are not postulated at terminal ends).

- Longitudinal breaks are postulated in high energy lines that have a nominal pipe size of four (4) inches or greater.

- Circumferential and longitudinal breaks are not postulated to occur concurrently.

- Longitudinal breaks are not postulated at branch connections.

- Longitudinal breaks are postulated only at intermediate break locations.
- Longitudinal breaks are postulated parallel to the pipe axis and orientated at all points on the pipe circumference.

- The break area of a longitudinal break is equal to the effective cross-sectional flow area of the pipe immediately upstream of the break location.

- Critical Cracks are postulated on seismically analyzed high energy piping that exceeds one (1) inch in nominal pipe size.

### 3.6.2.4 Shutdown Sequence Evaluation Criteria

The following criteria are used to identify the systems and components necessary for HELB mitigation and/or unit shutdown to the cold shutdown condition:

- Equipment used to mitigate postulated HELBs includes those systems and components that are used for detection and isolation of specified HELBs. Equipment that is used for the detection and isolation for an identified HELB is the only detection and isolation equipment required to be targets of that specific HELB.

- Equipment used to meet any of the following shutdown objectives are considered a target of postulated HELBs:

    - Reactivity Control
    - RCS Inventory Control
    - RCS Pressure Control
    - RCS Heat Removal Control
    - Reactor Building (Boundary) Integrity
    - Control Room Habitability (long term)
    - Plant Cooldown

- Both primary and back-up systems, used to achieve the shutdown objectives described above, are included as shutdown equipment and targets of the postulated HELBs.

- Piping, orifices, relief valves, and check valves, are considered passive type components in that they do not require an external power source or manual action to perform their intended function, and these components perform their intended function regardless of the environmental conditions. These components are not identified as required in the shutdown sequence, because they are not subject to single active failures (SAFs). They are, however, HELB targets.

- A SAF is postulated in systems used to mitigate the consequences of the postulated HELBs and Critical Cracks or those systems used to achieve a shutdown objective of the unit. The single active component failure is assumed to occur in addition to those components damaged by the postulated pipe break.
- No SAFs are postulated during the “Plant Cooldown” phase and the “Plant Cooldown to the Cold Shutdown Condition” phase.

- All available systems, including those actuated by operator actions, may be employed to mitigate the consequences of a postulated HELB or critical crack.

- In determining the systems and components available to mitigate the consequences of postulated HELBs, all Shutdown Equipment is assumed to be operable and available at the start of the postulated HELB sequence. It is not necessary to postulate that any systems or components are out of service for maintenance.

- Although a postulated HELB outside of the containment building may ultimately require a cold shutdown, holding at hot standby/shutdown is allowed in order that plant personnel assess the situation and make any necessary repairs to allow the unit to reach cold shutdown.

**3.6.2.5 Interaction Evaluation Criteria**

The following criteria are used to determine the interactions that occur as a result of postulated HELBs with shutdown equipment and the criteria for determining the pathway to cold shutdown for a given postulated HELB:

- The targets of the postulated HELBs are those systems and components required to mitigate the consequences of postulated HELBs and/or are used during the shutdown sequence to safely bring the unit to the cold shutdown condition.

- SSD, Cold Shutdown, and HELB mitigation systems and components directly impacted by a specific postulated HELB are considered to be unavailable to support the Shutdown Objectives for that specific HELB, unless documented otherwise.

- Movement of a ruptured high energy pipe (i.e. pipe whip) is considered for potential interactions. The pipe whip is assumed to occur in the plane defined by the piping geometry.

- The energy level in whipping pipes may be considered insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

- No secondary pipe breaks are postulated due to jet impingement from the source pipe (pipe with postulated HELB).

- The Jet Impingement Forces, Jet Impingement Cone Geometry, and the Jet Impingement Effective Length are determined in accordance with NUREG/CR-2913, “Two Phase Jet Loads,” subject to the pressure and temperature limitations given in the NUREG (i.e. stagnation pressures from 870 psia to 2465 psia, 0 to 126°F sub-cooling, and 0 to 75% steam quality). For jets consisting of steam or subcooled liquid water falling outside of the NUREG limitations, the effective length of the jet is 10 pipe diameters (ID). Similarly, jet lengths from Critical Cracks are limited to 5 pipe diameters (ID).

- Thrust leads for evaluating potential interactions between postulated HELBs and the TB structural components are determined in accordance with ANSI 58.2 (Rev. 2).

- Systems and components, whose only function is to support the cooldown of the unit from an RCS temperature of approximately 250°F to the cold shutdown condition, need not be protected from postulated HELBs.

- A “Loss of Offsite Power” (LOOP) is not postulated unless the initiating break directly causes a LOOP.

- HELB interactions with cables result in the affected component(s) failing in the most undesired state or are evaluated for the effects of the interaction. However, the following exceptions apply. If an electric Load Center (LC) or Motor Control Center (MCC) is affected by interactions, the LC or MCC is considered to be de-energized. Components receiving power from this LC or MCC are considered de-energized and unable to function unless alternate power supplies are available. Valves directly powered from an affected MCC fail “as is” regardless of other interactions.

- The Reactor Trip Breakers and the CRD system can be excluded from the list of Shutdown Equipment components and potential HELB targets because the unit trip function can be considered to be completed prior to any potential degradation of the system due to any gradual adverse environmental effects caused by postulated HELBs.

### 3.6.2.6 DETERMINATION OF SAFE SHUTDOWN SYSTEMS

#### 3.6.2.6.1 HELB Mitigation Strategy

The HELB Mitigation Strategy addresses the level of protection provided to systems, structures, and components (SSCs) necessary to reach SSD from the direct effects (pipe whip and jet impingement) and indirect effects (environmental and flooding) of a given HELB outside of the containment building. The major points of the strategy are as follows:

- Required SSCs located in the TB are not impacted by HELBs postulated to occur in the AB or in the yard.

- Required SSCs located in the AB are not impacted by HELBs postulated to occur in the TB.

- SAFs are imposed for those components required for initial mitigation.
- SAFs are not imposed for those components required to initiate a cooldown of the plant.

- HELBs resulting in the loss of plant systems inside the TB needed for SSD are mitigated by the Protected Service Water (PSW) system (see UFSAR Section 9.7).

- Should the PSW system be unavailable, the SSF (see UFSAR Section 9.6) is credited as an alternate means of achieving and maintaining SSD following HELBs that disable plant systems inside the TB.

- HELBs resulting in the loss of plant systems inside the AB needed for SSD are mitigated by normal plant systems or the SSF.

- As applicable, NUREG/CR-2913 is used for the determination of jet impingement effects following HELBs and critical cracks.

- Exclusion of systems whose operating time at high energy conditions is less than 1% of the total unit operating time.

- Exclusion of systems whose operating time at high energy conditions is less than approximately 2% of the total system operating time.

- Elimination of arbitrary intermediate breaks in ASME B & PV Section III-Class 2 and Class 3 equivalent piping. Intermediate breaks are postulated where calculated longitudinal stress for the applicable load cases (internal pressure, dead weight (gravity) thermal, and seismic (OBE) conditions) exceed $0.8(S_a + S_h)$.

- Intermediate breaks in non-rigorously analyzed piping are postulated in accordance with BTP MEB 3-1, Section B.1.c(2)(b)(i).

- Elimination of critical cracks at the most adverse location in ASME B&PV Section III-Class 2 and Class 3 equivalent piping. Critical cracks are postulated at axial locations where the calculated stress for the applicable load cases (internal pressure, dead weight (gravity), thermal, and seismic (OBE) conditions) exceed $0.4(S_a + S_h)$. Critical cracks are not postulated at locations of terminal ends.

- Elimination of critical cracks at the most adverse location in the non-rigorously analyzed piping. The effects of the postulated intermediate breaks bound the effects from critical cracks.

- HELBs occurring outside of the TB and AB are mitigated by normal plant systems.

#### 3.6.2.6.2 Shutdown Objectives

HELBs outside of the containment building may or may not result in consequences that require an automatic trip of the reactor and main turbine. The operator may elect to trip the reactor and main turbine for personnel and equipment protection. The objective for each shutdown interval is provided below.

The shutdown sequence is divided into four intervals:

1. Shutdown of the Reactor and Main Turbine

   The objective is to place the reactor in a subcritical state to protect the core. The main turbine must be tripped to prevent excessive RCS cooling. With the exception of the MS supply to the turbine driven emergency feedwater pump (TDEFWP), the tripping of the main turbine also separates the MS lines from one another by closure of the main turbine stop valves.

2. Establishment of stable RCS conditions

   The objective is to balance the heat generation in the RCS with the heat being removed by the Steam Generators (SGs) such that RCS temperatures can be controlled. This is accomplished by maintaining RCS inventory control and establishing RCS pressure control such that coupling with the SGs can be restored or maintained. Secondly, feeding and/or steaming of the SGs are controlled in a manner such that the amount of heat generated by core decay heat and Reactor Coolant Pump (RCP) heat (if still running) is balanced with the heat removal from the SGs. Finally, a source of borated water sufficient to maintain the reactor in a subcritical condition is aligned and used to supply the RCS. Depending on the extent of damage from the HELB and the strategy used for mitigation, stable RCS conditions may be maintained up to 72 hours before plant cooldown would be initiated.

3. Initiation of RCS cooldown to approx. 250°F

   The objective of this phase is to initiate a plant cool-down from the point where RCS conditions are stabilized to LPI entry conditions. The SGs are utilized for plant cooldown from normal post reactor trip conditions to approximately 250°F. Typically, plant cooldown would be via forced circulation using any RCP. If all of the RCPs are unavailable, procedures are provided to initiate a natural circulation cooldown.

4. Establishment of the cold shutdown condition (RCS temperature < 200°F)

   The objective of this phase of post-HELB operations is to transition from decay heat removal using the SGs to removing core decay heat using the LPI system. The LPI system, in conjunction with the low pressure service water system, is utilized to cool the RCS from approximately 250°F to less than 200°F.

### 3.6.2.6.3 Functions to meet Safe Shutdown Objectives

This section describes the functions needed to satisfy the shutdown objectives following a postulated HELB outside of the containment building. HELBs outside of the containment building can be divided into three categories: those that result in a loss of heat transfer (loss of SG feedwater), those that result in excessive heat transfer (loss of MS pressure boundary control), and those that result in loss of reactor coolant inventory (letdown line break). Loss of heat transfer scenarios result in a mismatch where more heat is generated in the core than is removed by the secondary system. These scenarios lead to an increase in RCS temperature and pressure. Excessive heat transfer scenarios result in a mismatch where more heat is removed by the secondary system than is generated in the core. These scenarios lead to a decrease in RCS temperature, pressure, and water level (due to reactor coolant shrinkage). Loss of inventory scenarios have a minor effect on the RCS due to the insignificant amount of inventory lost. The systems necessary to reach SSD were selected based on meeting the following Shutdown functions for the categories of HELB:

- Reactivity Control
- RCS Inventory Control
- RCS Pressure Control
- RCS Heat Removal Control
- Reactor Building (Boundary) Integrity
- Control Room Habitability (long term)
- Plant Cooldown
- Process Monitoring
- Support Functions

The licensee proposed to add new Section 3.6.3, “Safety Evaluation,” as follows:

### 3.6.3 Safety Evaluation

Normal plant systems, the PSW system, and the SSF are credited for the mitigation of HELBs outside containment. MFDW HELBs result in overheating transients. MS HELBs result in overcooling transients.

The safety analysis acceptance criteria for each HELB transient are as follows:

Overheating Analysis

- The core must remain intact and in a coolable geometry.

- Minimum departure from nucleate boiling ratio (DNBR) meets specified acceptable fuel design limits.

- RCS pressure must not exceed 2750 psig (110% of design).

Overcooling Analysis

In addition to the criteria specified above, the following criteria are applicable (validated) for the most limiting overcooling analyses:

- The SG tubes remain intact.
- RCS remains within acceptable pressure and temperature limits.

The bounding overheating transient is a MFDW HELB in the TB resulting in a loss of all 4160 VAC power to normal plant systems. The bounding overcooling transient is a double MS HELB in the TB resulting in a loss of all 4160 VAC power to normal plant systems.

The PSW system is credited for the mitigation of HELBs inside the TB when a HELB results in the loss of plant systems needed for SSD. The SSF is credited as an alternate means for mitigation of HELBs inside the TB when a HELB results in the loss of plant systems needed for SSD.

### 3.6.3.1 PSW Response Following a MFDW HELB in the TB

The transient begins with an immediate and complete loss of MFDW from hot full power (HFP) conditions with an initial core power level of 102% of 2568 MW as well as a loss of the 4160 VAC switchgear.  This causes an immediate reactor trip and turbine trip due to the loss of power. The RCPs continue to operate until operator action is taken to trip them either 2 minutes after a loss of indicated subcooled margin, or 3 minutes after a loss of RCP seal cooling. The motor-driven emergency feedwater pumps (MDEFWP) are powered from the 4160 VAC switchgear and are not available.  The TDEFWP is assumed to be unavailable.

Since portions of the integrated control system (ICS) are unprotected from HELB damage, the pressurizer power operated relief valve (PORV) is assumed to be unavailable. The combination of high end of cycle decay heat and delayed PSW flow to the SGs causes a large overheating transient in the primary system and a rapid increase in RCS pressure. RCS pressure increases to the pressurizer safety valve (PSV) lift setting, and the PSVs cycle to control RCS pressure until operators establish PSW flow 14 minutes into the event. PSW is assumed to be available at 14 minutes in the overheating analysis to prevent liquid relief through the PSVs. The peak RCS pressure in the overheating analysis is defined by the pressurizer safety relief valve characteristics since the PORV is not available. With an immediate reactor trip, the rate of RCS pressurization is such that maximum pressure occurs during the first PSV lift. The maximum pressure observed remains below the 2750 psig limit. Thus, the peak RCS pressure results obtained are not contingent on the timing of PSW flow.

Successful mitigation of a HELB condition at ONS shall be defined as ensuring that the integrity of the fuel and RCS remains unchallenged. For the overheating analysis the fuel integrity is ensured by the reactivity added via control rod insertion and maintaining the core covered. A minimum DNBR evaluation is not required for this analysis since the transient does not include a return to power and the DNBR at reactor trip is bounded by the existing UFSAR Chapter 15 analyses. RCS integrity is demonstrated by verifying the RCS pressure remains below the 2750 psig limit.

In summary, the results of the analysis demonstrate that PSW is capable of ensuring peak RCS pressure remains below the 2750 psig limit. Additionally, the results demonstrate there is sufficient decay heat removal (DHR) and primary coolant makeup to keep the core covered and maintain the RCS in Mode 3 for the duration of the scenario.

### 3.6.3.2 SSF Response Following a MFDW HELB in the TB

The transient begins with an immediate and complete loss of MFDW from HFP conditions with an initial core power level of 102% of 2568 MW, as well as a loss of the 4160 VAC switchgear. This causes an immediate reactor trip and turbine trip due to the loss of power. The RCPs continue to operate until operator action is taken to trip them either 2 minutes after a loss of indicated subcooled margin, or 3 minutes after loss of RCP seal cooling. The MDEFWPs are powered from the 4160 VAC switchgear and are not available due to the loss of power. The TDEFWP is assumed to be unavailable.

Since portions of the ICS are unprotected from HELB damage, the pressurizer PORV is assumed to be unavailable. The combination of high end of cycle decay heat and delayed SSF auxiliary service water (ASW) flow to the SGs cause a large overheating transient in the primary system and a rapid increase in RCS pressure. RCS pressure increases to the PSV lift setting, and the PSVs cycle to control RCS pressure until operators establish SSF ASW flow 14 minutes into the event. The peak RCS pressure in the overheating analysis is defined by the pressurizer safety relief valve characteristics since the PORV is not available. With an immediate reactor trip, the rate of RCS pressurization is such that the maximum pressure occurs during the first PSV lift. The maximum pressure observed remains below the 2750 psig limit. Thus, the peak RCS pressure results obtained are not contingent on the timing of SSF ASW flow.

Successful mitigation of a HELB shall be defined as ensuring that the integrity of the fuel and RCS remains unchallenged. For the overheating analysis the fuel integrity is ensured by the reactivity added via control rod insertion and maintaining the core covered. A minimum DNBR evaluation is not required for this analysis since the transient does not include a return to power and the DNBR at reactor trip is bounded by the existing UFSAR Chapter 15 analyses. RCS integrity is demonstrated by verifying the RCS pressure remains below the 2750 psig limit.

In summary, the results of the analysis demonstrate that the SSF is capable of ensuring peak RCS pressure remains below the 2750 psig limit. Additionally, the results demonstrate there is sufficient DHR and primary coolant makeup to keep the core covered and maintain the RCS in Mode 3 for the duration of the scenario.

### 3.6.3.3 PSW Response Following a Double MS HELB in the TB

This analysis determines the plant transient response to a double MS HELB mitigated with PSW equipment and without credit for the automatic feedwater isolation system (AFIS). This analysis assumes an initial core power level of 102% of 2568 MW at HFP conditions. The initiating event causes double MS HELB, an immediate loss of 4160 VAC power, a reactor trip, a turbine trip, and a trip of all condensate and MFDW pumps. The RCPs continue to operate until operator action is taken to trip them either 2 minutes after a loss of indicated subcooled margin, or 3 minutes after a loss of RCP seal cooling. The MDEFWPs are not available due to the loss of 4160 VAC power. To maximize the overcooling, the TDEFWP is assumed to automatically start and run without being throttled until the contents of the upper surge tank (UST) are delivered to the SGs. This scenario is intended to bound the consequences resulting from a double MS HELB.

The primary objective of this analysis is to demonstrate that the minimum DNBR is acceptable and that the plant will achieve a steady state condition where the RCS is in natural circulation flow conditions with PSW providing a heat sink, a PSW powered HPI pump providing seal injection flow, RCS pressure being maintained with the PSW powered pressurizer heaters, and pressurizer level being controlled by operation of the loop high point vents and/or PSW flow. This assures that the core remains intact and in a coolable geometry.

The double MS HELB causes the RCS to depressurize and shrink. As RCS pressure decreases the two CFTs inject additional borated inventory into the RCS. The core remains covered throughout the overcooling transient. The sustained overcooling in the affected loop is not sufficient to result in a return to criticality. The core remains subcritical after the rods insert for the duration of the transient. The core remains covered and cooled for the duration of the transient. The PSW powered HPI pump is started to restore RCP seal cooling and makeup to the RCS. PSW flow is available at 14 minutes, but not delivering flow to the SGs at this time due to the overcooling. The overcooling continues until shortly after the TDEFWP stops feeding the SGs.

After the overcooling has terminated, the RCS begins to slowly reheat and swell, pressurizer level returns on scale, and the PSW powered pressurizer heaters are manually energized. PSW flow is established to the SGs to stabilize RCS temperature and pressurizer level. Saturated conditions are established in the pressurizer and pressurizer heaters are then cycled to maintain RCS pressure stable. Stable subcooled natural circulation conditions are achieved approximately three hours into the transient.

Successful mitigation of a HELB condition at ONS shall be defined as ensuring that the integrity of the fuel and RCS remains unchallenged. For the overcooling analysis the fuel integrity is confirmed by the DNBR analysis.

RCS integrity is demonstrated by determining the limiting SG tube compressive and tensile stresses remain with design limits, and that the RCS pressure and temperature remains within the acceptable cooldown limits during the transient evolution. The time dependent SG tube and SG shell temperatures are determined using a linear average to determine if the temperature differences remain within the SG design limits. The results indicate the SG tube stress remains well within the established limits for the duration of the transient. The cooldown performed through operator control of PSW to below 350°F will provide margin to prevent tube deformation.

This analysis demonstrates that a double MS HELB can be mitigated using PSW equipment. In summary, the overcooling analysis demonstrates that for a double MS HELB scenario, the following acceptance criteria are satisfied:

- The core remains intact and in a coolable geometry,
- Minimum DNBR meets specified acceptable fuel design limits,
- The SG tubes remain intact,
- RCS pressure does not exceed 2750 psig, and
- RCS remains within acceptable pressure and temperature limits.

### 3.6.3.4 SSF Response Following a Double MS HELB in the TB

This analysis determines the plant transient response to a double MS HELB mitigated with SSF equipment and without credit for AFIS. This analysis assumes an initial core power level of 102% of 2568 MW at HFP conditions. The initiating event causes either a single or double MS HELB, an immediate loss of 4160 VAC power, a reactor trip, a turbine trip, and a trip of all condensate and MFDW pumps. The RCPs continue to operate until operator action is taken to trip them either 2 minutes after a loss of indicated subcooled margin, or 3 minutes after a loss of RCP seal cooling. The MDEFWP are not available due to the loss of 4160 VAC power. To maximize the overcooling, the TDEFWP is assumed to automatically start and run without being throttled until the contents of the UST are delivered to the SGs. This scenario is intended to bound the consequences resulting from a MS HELB.

The primary objective of this analysis is to demonstrate that the minimum DNBR is acceptable and that the plant will achieve a steady state condition where the RCS is in natural circulation flow conditions with SSF ASW providing a heat sink, SSF reactor coolant makeup (RCMU) flow providing seal injection flow, RCS pressure being maintained with the SSF powered pressurizer heaters, and pressurizer level being controlled by operation of the SSF letdown line and/or SSF ASW. This assures that the core remains intact and in a coolable geometry.

The double MS HELB causes the RCS to depressurize and shrink. As RCS pressure decreases, the two CFTs inject additional borated inventory into the RCS. The core remains covered throughout the overcooling transient. While a brief recriticality is indicated, the resulting fission power obtained is not significant (less than one watt). The SSF RCMU pump is started to restore RCP seal cooling and makeup to the RCS. SSF ASW flow is available at 14 minutes, but not delivering flow to the SGs at this time due to the overcooling. The overcooling continues until shortly after the TDEFWP stops feeding the SGs.

After the overcooling has terminated, the RCS begins to slowly reheat and swell, pressurizer level returns on scale, and the SSF powered pressurizer heaters are manually energized. SSF ASW flow is established to the SGs to stabilize RCS temperature and pressurizer level. Saturated conditions are established in the pressurizer and pressurizer heaters are then cycled to maintain RCS pressure stable. Stable subcooled natural circulation conditions are achieved approximately three hours into the transient.

Successful mitigation of a HELB condition at ONS shall be defined as ensuring that the integrity of the fuel and RCS remains unchallenged. For the overcooling analysis the fuel integrity is demonstrated by the DNBR analysis.

RCS integrity is demonstrated by determining the limiting SG tube compressive and tensile stresses remain with design limits, and that the RCS pressure and temperature remains within the acceptable cooldown limits during the transient evolution. The time dependent SG tube and SG shell temperatures are determined using a linear average to determine if the temperature differences remain within the SG design limits. The results indicate the SG tube stress remains well within the established limits for the duration of the scenario. The cooldown performed through operator control of SSF ASW to below 350°F will provide margin to prevent tube deformation.

To validate that RCS pressure and temperature remain within limits, these parameters are plotted versus each other to examine the time dependent response. These results indicate significant margin is maintained to the acceptable cooldown limits during the scenario.

This analysis demonstrates that a double MS HELB can be mitigated using SSF equipment. In summary, the overcooling analysis demonstrates that for a double MS HELB scenario, the following acceptance criteria are satisfied:

- The core remains intact and in a coolable geometry,
- Minimum DNBR meets specified acceptable fuel design limits,
- The SG tubes remain intact,
- RCS pressure does not exceed 2750 psig, and
- RCS remains within acceptable pressure and temperature limits.

The licensee proposed to renumber current Section 3.6.2, “References,” to Section 3.6.4 and to make the following changes:

3.6.4 References

2. ~~Duke Power/B&W Report, Oconee Nuclear Station, "Evaluation of Potentially Adverse Environmental Effects on Non-Safety Grade Control Systems", October 5, 1979.~~ USNRC Standard Review Plan (NUREG 0800) Section 3.6.1 Branch Technical Position ASB 3-1.

[…]

9. Clarification Letter (related to the 15 December 1972 letter), dated 17 January 1973, from A. Schwencer (AEC) to A. C. Thies (DPC).

10. HELB Outside Containment Walkdown Criteria & Requirements, ONS, Units 1, 2, & 3.

11. Calculation OSC-8385 – Normal Operating Conditions for High Energy Line Break (HELB) Analysis (ONS Units 1, 2, & 3).

12. OSS-0254.00-00-4017 – Design Basis Specification for the "Pipe Rupture" – ONS Units 1, 2, & 3.

13. NRC Generic Letter 87-11, Relaxation in Arbitrary Intermediate Pipe Rupture Requirements (Rev. 2 of BTP MEB 3-1), June 19, 1987.

14. Duke Energy Calculation, OSC-11769, Analysis of Postulated HELBs Outside of Containment.

15. Letter from A. Giambusso (AEC) to A. C. Thies (Duke Power Company), "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment," dated December 15, 1972.

The licensee proposed to add the following paragraph to the end of Section 5.1.2.4, "Natural Circulation":

> The effects of a HELB may drive a unit to an average reactor coolant temperature less than 525°F. The subsequent minor reduction in RCS temperature required to compensate for the increase in RCS inventory by the SSF RCMU pump during plant stabilization does not constitute a natural circulation cooldown requiring use of the reactor vessel head vent. Refer to Reference 2 for additional information.

The licensee proposed to add the following reference to the new license amendments that are the subject of this SE to Section 5.1.3, "References," of the UFSAR (with information to be updated with the amendment numbers and their date of issuance):

2. License Amendment No. XXX, XXX, and XXX (date of issuance – Month XX, 20XX); HELB Mitigation.

The licensee proposed to add the following text to the end of the third paragraph in Section 9.6.1, "General Description," of the UFSAR:

> A high energy line break (HELB) licensing basis reconstitution effort that began in the early 2000s resulted in the SSF being credited as an alternate means to achieve and maintain safe shutdown following HELBs in the turbine building and auxiliary building.

The licensee proposed to add the following text to the fifth paragraph in Section 9.6.1 of the UFSAR:

> […] The SSF requires manual activation that would occur under adverse fire, flooding, HELB, or […]

The licensee proposed to add the following heading and text in Section 9.6.1 of the UFSAR:

> **HELB Design Criteria**
>
> As a result of a HELB licensing basis reconstitution performed in the early 2000s, the SSF is credited for meeting the design requirements of certain HELB locations. The SSF provides an alternate means to achieve and maintain safe shutdown following HELBs in the turbine building and auxiliary building. See Section 3.6 for more details on the design criteria of HELBs outside containment.

The licensee proposed to revise Section 9.6.5, “Operation and Testing,” of the UFSAR as follows:

> The SSF will be placed into operation to mitigate the consequences of the following events/criterion.
>
> 1. Flooding
> 2. Fire
> 3. Sabotage
> 4. Station Blackout
> 5. High Energy Line Break
>
> […]
>
> For flooding, sabotage, station blackout, high energy line break and those fire events where the SSF is credited for safe shutdown, operators will be sent to the SSF. […]

The licensee proposed to revise Section 9.7.1, “General Description,” of the UFSAR as follows:

> […] For certain scenarios inside the Turbine Building (TB) resulting in loss of 4160V essential power, either the SSF or PSW System is used for reaching safe shutdown. (For HELBs, see UFSAR Section 3.6) […]

The licensee proposed to revise Section 10.4.7.1, “Design Bases,” of the UFSAR as follows:

> […] The effects of High Energy Line Breaks have been analyzed as addressed in UFSAR Section 3.6~~.1.3~~. […]

The licensee proposed to revise Section 10.4.7.3, "Safety Evaluation," of the UFSAR as follows:

> […] Redundancy is provided with separate, full capacity, motor and turbine driven pump subsystems. Except as noted in ~~the subsections that follow~~Section 10.4.7.3.2, failure of either the MDEFWPs or the TDEFWP will not reduce the EFW System below minimum required capacity. Pump controls, instrumentation, and motive power are separate in design.
>
> The transients that require EFW have been evaluated assuming only one MDEFWP is available to deliver the necessary feedwater. Except as noted in ~~the subsections that follow~~Section 10.4.7.3.2, […]

The licensee proposed to revise Section 10.4.7.3.2 of the UFSAR by deleting current Sections 10.4.7.3.2.1, 10.4.7.3.2.2, and 10.4.7.3.2.3, and adding the following:

> **10.4.7.3.2 EFW Response Following a HELB**
>
> Certain HELBs in conjunction with postulated single failure can disable all sources of emergency feedwater. These HELBs are evaluated in applicable analyses as addressed in UFSAR Section 3.6.2. For these cases in which EFW is not available (TDEFWP, MDEFWP, cross-connects), the PSW system and the SSF ASW system provide an additional source of secondary cooling water.

2.4 Regulatory Requirements and Guidance Documents

The NRC staff considered the following regulatory requirements, licensing and design basis information, and guidance during its review of the proposed changes.

The NRC staff identified the following Oconee UFSAR Plant Design Criteria as being applicable to the proposed amendments.

UFSAR Section 3.1.6, "Criterion 6 - Reactor Core Design (Category A)," states, in part:

> The reactor core shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all off-site power.

UFSAR Section 3.1.9, "Criterion 9 - Reactor Coolant Pressure Boundary (Category A)," states, in part:

> The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

UFSAR Section 3.1.20, “Criterion 20 - Protection Systems Redundancy and Independence (Category B),” states, in part:

> Redundancy and independence designed into Protective Systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protective function. The redundancy provided shall include, as a minimum, two channels of protection for each protective function to be served. Different principles shall be used where necessary to achieve true independence of redundant instrumentation components.

UFSAR Section 3.1.23, “Criterion 23 - Protection against Multiple Disability for Protection Systems (Category B),” states, in part:

> The effects of adverse conditions to which redundant channels or Protective Systems might be exposed in common, either under normal conditions or those of an accident, shall not result in a loss of the protective function.

UFSAR Section 3.1.24, “Criterion 24 - Emergency Power for Protection Systems (Category B),” states, in part:

> In the event of loss of all off-site power, sufficient alternate sources of power shall be provided to permit the required functioning of the Protective Systems.

UFSAR Section 3.1.37, “Criterion 37 - Engineered Safety Features Basis for Design (Category A),” states, in part:

> Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

UFSAR Section 3.1.39, “Criterion 39 - Emergency Power for Engineered Safety Features (Category A),” states, in part:

> Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the on-site power system and the off-site power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

UFSAR Section 3.1.40, “Criterion 40 - Missile Protection (Category A),” states, in part:

> Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

Regulations in 10 CFR 50.49, “Environmental qualification of electric equipment important to safety for nuclear power plants,” require licensees to establish a program for qualifying electric equipment as defined in 10 CFR 50.49(b).

Regulations in 10 CFR 50.49(e)(1), "Temperature and pressure," require that the time-dependent temperature and pressure at the location of the electric equipment important to safety must be established for the most severe design basis accident during or following which this equipment is required to remain functional.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), Section 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," Revision 3 (Reference 26).

NUREG-0800, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," Revision 3 (Reference 27), provides guidance on environmental qualification (EQ) of mechanical and electrical equipment.

NUREG-0800, Chapter 8, "Electric Power," (Reference 28), provides guidance on electrical power systems.

NUREG-0800, Chapter 18, "Human Factors Engineering," Revision 3 (Reference 29), discusses the processes for evaluating operator actions and identifies specific areas of review that are needed for successful integration of human characteristics and capabilities into nuclear power plant design.

NUREG-0800, Section 13.2.1, "Reactor Operator Requalification Program; Reactor Operator Training," Revision 4 (Reference 30), Section 13.2.2, "Non-Licensed Plant Staff Training," Revision 4 (Reference 31), and Section 13.5.2.1, "Operating and Emergency Operating Procedures," Revision 2 (Reference 32).

NUREG-0711, "Human Factors Engineering Program Review Model," Revision 3 (Reference 33), provides acceptance criteria for a human factor engineering design process.

NUREG-1764, "Guidance for the Review of Changes to Human Actions," Revision 1 (Reference 34), provides guidance on the level of review for human actions.

NUREG-0800, Branch Technical Position 3-3, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," Revision 3 (Reference 35).

Technical Specification 3.7.10, "Protected Service Water (PSW) System," requires that the PSW system be operable in operational modes 1 and 2.

Technical Specification 3.10.1, "Standby Shutdown Facility (SSF)," requires that the SSF be operable in operational modes 1, 2, and 3.

Generic Letter (GL) 87-11, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements" (Rev. 2 of BTP MEB 3-1), dated June 19, 1987 (Reference 36).

Letter from A. Giambusso (AEC) to A. C. Thies (Duke Power Company), "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment" (Reference 19), dated December 15, 1972.

Clarification Letter from A. Schwencer (AEC) to A. C. Thies (Duke Power Company), “Errata to General Information Required for Consideration of the Effects of a Piping System Break Outside Containment” (Reference 20), dated January 17, 1973.

The Giambusso/Schwencer letters provided a list of requested information that would demonstrate with reasonable assurance that the plant could be safely shut down and maintained in a safe shutdown condition following credible failures of piping in high energy systems. The NRC staff summarized the guidance in these letters, as updated by GL 87-11, with the following three general strategies for protection against HELBs:

- a postulated break is of very low probability based on the operating conditions of the piping.
- essential mitigating systems are physically separated from the break area and any resulting structural damage.
- essential mitigating systems are designed to withstand the effects of the break, which may include pipe whip restraints, jet shields, and environmental qualification of electrical components.

The Giambusso Letter, as modified by the Schwencer Letter, provided the following review Items as summarized by the NRC staff:

1. Identification of system piping segments needing physical restraint to prevent damage to safe shutdown systems.
2. Criteria for identification of pipe break locations.
3. Criteria to determine pipe break orientation.
4. Summary of dynamic analyses of piping system response to ruptures.
5. Description of measures to protect against pipe whip and jet impingement.
6. Design criteria to evaluate structural response.
7. Design load acceptance criteria for structural elements.
8. Evaluation of structural elements experiencing a reversal in load due to pressurization of compartments by pipe breaks.
9. Evaluation of modified structures to withstand pipe break loads.
10. Verification that the consequences of structural failures would not prevent safe shutdown.
11. Verification that the consequences of pipe breaks would not cause a loss of necessary protective system redundancy for direct effects or loss of function for indirect effects.
12. Assurance of habitability for necessary control capability to achieve safe shutdown.

13. Environmental qualification of equipment that must function after exposure to the indirect effects of high energy pipe breaks.

14. Drawings of MS and FDW lines showing routing.

15. Assessment of flooding potential following HELB.

16. Description of quality control and inspection measures applied to piping systems.

17. Description of the capabilities of necessary leak detection equipment.

18. Summary of emergency procedures.

19. Description of seismic and quality classification of piping systems.

20. Description of assumptions, methods of analysis, and analysis results for pressure and temperature transients in compartments.

21. Description of the method used to demonstrate no adverse effects on primary containment from breaks outside containment.

## 3.0 TECHNICAL EVALUATION

### 3.1 Revised HELB Strategy

The NRC staff's review of plant design for protection against HE piping failures outside containment is to ensure that environmental effects of such failures would not cause the loss of needed functions to ensure that the plant could be safely shut down in the event of such failures. The NRC staff reviewed the HELB methodology to ensure that the criteria were implemented properly.

As noted in the LAR, the Oconee original licensing basis for HELB was contained within the responses to the Giambusso/Schwencer letters. As set out in the letters, a nuclear plant should be designed so that the reactor can be shut down and maintained in an SSD condition in the event of a postulated rupture outside containment of a pipe containing a HE fluid, including the double-ended rupture of the largest pipe in the MS and FDW systems. Plant SSCs important to safety should be designed and located in the facility to accommodate the effects of such a postulated pipe failure to the extent necessary to ensure that an SSD condition of the reactor can be accomplished and maintained.

The overall proposed HELB mitigation strategy proposed by the licensee generally relies on systems and equipment capable of bringing the plant to an SSD condition that are located remotely from the postulated break or crack location(s) or can be shown to not be damaged by the effects of the postulated break. The licensee identified the following actions as a methodology to evaluate potential HELBs outside containment:

1. Identification of the HE systems, the HE lines, and the boundaries of the HE lines on each of those systems.

2. Identification of the postulated HELB locations and break types on each of the HE lines.

3. Determination of the equipment and systems in the ONS units, which could be utilized to mitigate the postulated HELBs.

4. Identification of the targets (SSCs) of each postulated HELB based upon the results of field inspections.

5. Determination of the shutdown equipment that is undamaged by the postulated HELB and can be used for the HELB mitigation and the shutdown of the station. This step is based upon the identification of the targets and the impact of the postulated HELBs on those targets.

6. Identification and recommendation of station physical and/or procedural changes to support the HELB mitigation strategy.

3.1.1 Identification of High Energy Systems

The licensee has specified criteria where evaluation of break effects would not be required due to low contained energy. These criteria are related to system operating parameters. Consistent with Item 1 of the Giambusso letter, the licensee defined the HE (piping) lines as those lines that the fluid inside of the pipe meets either or both of the following conditions during normal operation:

1. A normal operating temperature greater than 200°F.

2. A normal operating pressure greater than 275 psig.

Piping segments meeting these HELB criteria were analyzed within the LAR to ensure that HELBs will not impact the ability to bring the reactor to a shutdown condition. The licensee excluded gas (e.g., nitrogen) and oil (e.g., electro hydraulic control oil) systems from classification as HE because they contain much less internal energy than water systems at similar pressures and temperatures. The licensee also excluded water and steam containing piping that normally operates at sub-atmospheric pressure (e.g., some extraction steam and sealing steam piping) due to insufficient internal energy. Finally, the licensee excluded postulation of HELBs in HE piping that has a nominal size of 1 inch or less, consistent with the licensing basis established in its response to the Giambusso/Schwencer letters.

The licensee excluded other HE systems because the systems operate only a small fraction of time at HE conditions. The licensee defined a small fraction of time as operating at HE conditions less than 1 percent of the total unit operating time (e.g., EFW and RB spray) or operating at HE conditions less than approximately 2 percent of the total system operating time (e.g. low pressure injection (LPI) during the initiation of shutdown cooling with the plant in Hot Shutdown). The licensee postulated no breaks or cracks for systems meeting these limitations.

The licensee proposed that the following systems (and portions of systems) be excluded from HELB postulation based on an evaluation of actual operating time at HE conditions:

- EFW pump discharge piping.
- RB Spray discharge piping.
- ‘C’ (swing) HPI pump discharge piping.
- SSF ASW pump discharge piping.

- LPI system (entire system).
- Condensate recirculation piping to the UST.
- MFDW pump recirculation piping to the main condenser.
- MFDW cleanup piping to the UST.
- MFDW to the SG auxiliary FDW nozzles.
- SG hot blowdown/drain piping.
- Turbine bypass valve discharge piping to the main condenser.

Using the HE line criteria and considering the exclusions described above, the licensee identified the following 12 systems as systems containing HE piping subject to postulated HELBs in each unit:

- Auxiliary Steam
- Condensate
- Extraction Steam
- MFDW
- MS
- Heater Drain
- Heater Vent
- HPI
- Moisture Separator Reheater Drain
- Plant Heating (PH)
- Reverse Osmosis
- Steam Drain
- Steam Seal Header

The above HE systems and associated piping are in the TB and AB of each unit. All the HE systems except for the HPI system are in the TBs, and only the MS, MFDW, HPI, and PH systems are located in the ABs. A segment of MS piping from each unit is in the yard area outside structures.

<u>NRC Staff Evaluation</u>

The NRC staff evaluated the licensee's identification of HE piping segments. The NRC staff finds that the list of HE systems is consistent with the systems listed in Table 3.6.1-2 of SRP Section 3.6.1, with consideration of the identified exclusions. The NRC staff has previously identified that piping systems operated only for short periods of the system operational time at HE may be treated as moderate energy systems in Section B.2 of BTP 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment" (Reference 37). For Oconee, postulated leakage cracks or ruptures of moderate energy piping is not within the facility licensing basis. The lower likelihood of rupture in systems that operate only a short period of time at HE conditions is consistent with the slower development of pipe degradation at low energy conditions and the reduced likelihood that pipe stress would exceed remaining pipe strength because stresses are highest during the short period of operation at HE conditions. Thus, the NRC staff determined that the exclusions reflect previously accepted NRC staff positions.

The NRC staff finds that the exclusions have a substantially lower safety-significance of piping ruptures in systems that are at HE conditions because:

- the small fraction of the plant operating time or,
- they contain little internal energy based on operating fluid, operating conditions, or pipe size.

Based on the above, the NRC staff finds that the list of systems identified as HE is acceptable and consistent with Giambusso/Schwencer Item 1.

3.1.2 Identification of HELB Interaction Zones and Locations

The licensee proposed revised methods of identifying postulated HELB locations using piping stress and location criteria. The interactions due to a HELB event were made up of structural effects (and environmental effects), the mechanism of rupture, and the physical arrangement of adjacent structures, which all contribute to the piping reactions. The licensee used these attributes for each break location to assess the potential HELB interactions.

The licensee identified attributes of the HELB piping reactions in physical regions with significant interactions on other SSCs. The region of impact is the zones of influence (ZoI) affected by the HELB event. The geometry of each ZoI was determined without considering any alteration caused by ‘targets’ located within the zone. Once determined in shape, the ZoI was modified by identified obstructing structures and obstructions. The NRC staff reviewed the orientation of the ZoI as described below.

3.1.2.1 Circumferential Break Zone

Piping that severs at the ends displace axially, under the thermodynamic forces developed by the discharging fluid. The licensee’s calculation considered deflections of the piping ends, caused by the existing fluid energy, at both axial and perpendicular directions, depending on piping/support arrangement and proximity of obstructing structures, and the coordinates of break hinges. The severity of pipe whip displacements, specifically in perpendicular/out-of-plane directions, were dominated by the strength and time lapse of the thermodynamic forces and the attributes of the piping system and environment.

Constrains limit the range or zone of impact of the whipping pipe, due to existing piping restraints or other structures. Without the presence of structures or rupture restraints, adjacent piping to the postulated break determines the path, and the ZoI. The initial blowdown forces are of the magnitude and limited duration that the piping cannot respond elastically, and a localized inelasticity with a plastic hinge would form. The licensee indicated that usually the plastic hinge developed either at the first elbow from the break, or at an in-line point before the first elbow, if determined to be the primary hinge location. The pipe whip path, therefore, remains primarily in a single plane aligned with the axis of the initial jet discharge and the pipe axis at the hinge location. The overall ZoI, which includes the discharging fluid jet, may extend the whip ZoI farther, depending on the orientation of the break plane relative to the hinge position. The NRC staff agrees that the plastic hinge developed at the elbow or a point before and the axis of the jet forms the plane for the ZoI. This is due to the mechanistic degree of freedom for rotation in the plane. Based on the above, the NRC staff finds it acceptable to determine these ZoI’s prior to determining the interactions with obstacles because the boundaries of interaction will be well defined.

3.1.2.2 Longitudinal Break Zone

The licensee indicated that pipe whip due to longitudinal breaks occurs as a reaction from the fluid jet discharging in the opposite direction. The phenomenon of the fluid jet forces is almost identical to those for circumferential breaks. The severity of the pipe whip is limited to a sweeping plane, since the pipe does not sever. Pipe whip displacements are those causing out-of-plane deflections. With a change in direction that is deflected out-of-plane, the ZoI shape takes on a 3-dimensional 'sweep' arc. The interaction zone is close to the ruptured elbow and the boundaries at the two pipe hinges. The ZoI would produce a 2-dimensional shape of a partial elliptical/circular with the segment area between the initial and deflected pipe axes being the impact zone. This is due to the mechanistic degree of freedom for rotation in the plane which would produce a 2-dimensional shape of a partial elliptical/circular with the segment area between the initial and deflected pipe axes being the impact zone. The NRC staff agrees that this is almost identical to that of a circumferential break and, therefore, concludes it is also acceptable.

3.1.2.3 Critical Crack Zone

The licensee stated that HELB cracks are not subjected to pipe whip, since the fluid forces are greatly limited by the reduced discharge/flow area and are, therefore, of insufficient magnitude to appreciably displace the piping which remains fully intact. The NRC staff agrees that the flow through a critical crack is insufficient to cause pipe whip because of the greatly reduced fluid energy associated with the limited discharge/flow area, and thus the reduced impact of the jet compared to those from full-sized breaks. The NRC staff finds this approach acceptable because significant impact from critical crack jets on relatively rugged structures such as structural steel and pressure-boundary piping is not credible.

3.1.2.4 Jet Impingement Interaction Zones

The jet impingement ZoI is based on the jet length and width (diameter) corresponding to the point where the jet pressure at impact has decreased to a threshold value. These 'cutoff' values of impingement pressure are derived from the figures in NUREG/CR-2913, "Two-Phase Jet Loads" (Reference 25).

3.1.2.4.1 Shape of Fluid Jets

The shape of the fluid jet is nearly a circular cone with edges expanding outward from the break plane. The shape of the jet region outboard of the initial jet core is most approximating, and for saturated fluids, reasonably corresponds to a subcooled fluid jet for highly subcooled fluids. Significant radial expansion of the jet occurs in proximity to the break as the jet interacts with the ambient pressure of the surroundings. This is the jet core that extends increasingly from the break as the amount of subcooling increases. An exception to the expanded-cylinder shape is the 'cold-water' jet due to a break in a line carrying pressurized, non-expanding (non-flashing) liquid (i.e., at temperatures less than saturation at ambient conditions (212°F for water)). In this case, the jet does not expand and thus forms a circular cylinder of the same diameter as the break. This is consistent with the thermodynamic description of NUREG/CR-2913. Based on the above, the NRC staff finds it acceptable that the flow can be two-phase with an initial subcooled jet that expands into a two-phase jet.

3.1.2.4.2 Circumferential Break Jet Shape

The double-ended break results in two jets initially directed/opposed toward each other which are assumed to not interfere with each other due to full separation (unless piping is constrained to limit lateral displacement, as described below). The jets initially have the circular cross-section described above, with initial flow area equal to pipe flow area, and directed axially outward from the broken pipe end, in a cylinder-like shape. The jet direction, and as such its impingement zone, can vary significantly as the piping ends displace (whip), unless restrained as described below. Certain piping/support configurations can be shown to preclude gross, unlimited deflection of the severed pipe ends. If the severed pipe end is constrained to significantly limit lateral displacements, such as by piping guides or adjacent rigid structures, and gross axial movement or unlimited distortion about a piping 'hinge' is not stated, the pipe end and the jet axis can be considered fixed. If both severed ends are similarly constrained to limit lateral end-deflections to approximately one pipe wall thickness each, the opposing jets can be taken as fully interfered. In this case, the jet discharges radially outward about the pipe circumference, and the jet impingement zone length is greatly reduced. The NRC staff agrees that, if a double guillotine break's displacement is limited to one pipe thickness, the flow would be in the radial direction with a limited ZoI due to the interference of the opposing flow. Based on the above, the NRC staff finds this acceptable.

3.1.2.4.3 Longitudinal Break Jet Shape

The fluid jet from a longitudinal break discharges perpendicular to the pipe axis, along a line from the centerline of the pipe through the center of the break plane. As such, the postulated location of the break about the pipe circumference defines the direction of the jet. Based on ANSI/ANS-58.2 (Reference 38) methodology, the orientation of the jet can be predicted by the pipe fitting or feature at which the break is postulated. If postulated, break orientations are longitudinal breaks at tees or elbows which occur individually on each side of the fitting, at its center, and oriented perpendicular to the plane of the fitting. Thus, the jet emanates outward in the out-of-plane direction. At welded-on lug attachments, as identified by field walkdown, a longitudinal break is postulated at the centerline of the attachment. The jet direction is radially outward from the center of the attachment. The bases for jet thermodynamic forces are similar to those for circumferential breaks, since the break area is assumed to be based on the pipe ID in both cases. However, the direction of the jet is predictable, since pipe deflection is limited. The NRC staff agrees with the postulated assumption of the out-of-plane flow of the jet because such a configuration has a greater potential to impact a target in proximity and is, therefore, acceptable.

3.1.2.5 Arbitrary Intermediate Breaks and Critical Cracks

The following criteria are used by the licensee to identify the HE break types required to be postulated at the identified break locations at ONS. There are three types of HELBs at ONS—circumferential breaks, longitudinal breaks, and critical cracks. The criteria for each break type are as follows. Circumferential breaks are to be postulated in HE lines that exceed one-inch nominal pipe size. Only circumferential breaks are postulated at terminal ends of HE piping runs. Longitudinal breaks are to be postulated in HE piping that have a nominal pipe size of four inches or greater. Circumferential and longitudinal breaks are not postulated to occur concurrently. Longitudinal breaks are not postulated at branch connections. Longitudinal breaks are postulated only at intermediate break locations. Longitudinal breaks are postulated parallel to the pipe axis and oriented at all points on the pipe circumference. The break area of a longitudinal break is equal to the effective cross-sectional flow area of the pipe immediately upstream of the break location. Critical cracks are to be postulated on seismically analyzed HE piping that exceeds one-inch in nominal pipe size. The Giambusso/Schwencer letters provide criteria to determine pipe break orientation at break locations and specify that longitudinal breaks in piping runs and branch runs be postulated for nominal pipe sizes greater than or equal to four inches. Circumferential breaks are postulated at all terminal ends. The design of existing and potentially new rupture restraints may be used to mitigate the results from such breaks, including prevention of pipe whip and alteration of the break flow. Longitudinal breaks are not postulated at terminal ends. The NRC finds this to be consistent with the guidelines of BTP MEB 3-1 (Reference 36) and is, therefore, acceptable.

3.1.2.5.1 Intermediate Breaks

The licensee postulated circumferential and longitudinal break locations, as follows. Intermediate breaks are postulated in Class 2 or 3 equivalent piping at axial locations where the calculated longitudinal stress for the applicable load cases exceeds $0.8(S_a + S_h)$ for piping that is seismically analyzed. The load cases include internal pressure, dead weight (gravity), thermal, and seismic (OBE). Intermediate breaks are not postulated at locations where the only stress is the thermal expansion stress which does not cause ruptures in pipes. The Giambusso/Schwencer letters contained the guideline to postulate break locations where the actual stress exceeded $0.8(S_a + S_h)$, however, BTP MEB 3-1 includes no such guideline. The licensee concluded that the omission of the thermal stress threshold in BTP MEB 3-1 is recognition by the regulatory authorities that thermal stress, in the absence of primary stress, cannot cause pipe rupture failures. The NRC staff finds this statement to be true, since thermal stresses are self-limiting and cannot cause rupture. Breaks are postulated in accordance with BTP MEB 3-1, Section B.1.c(2)(b)(i) for piping that is not rigorously analyzed or does not include seismic loadings at intermediate locations. The licensee stated that systems and components located in the AB or the SSF are available for mitigation of the effects from the breaks postulated to occur in the TB. Any interactions between non-seismic piping and seismic piping located in the TB will be adequately mitigated with equipment from the AB or the SSF. The identified potential interactions between non-seismic and seismic piping are located in the TB, between the various secondary side HE systems and portions of the following systems: EFW MS branch lines, siphon seal water system, CCW, and the low pressure service water (LPSW) system. The NRC staff acknowledges these interactions in the TB due to their relative proximities.

The licensee stated that terminal ends are vessel/pump nozzles, building penetrations, in-line anchors, and branch-to-run connections that act as essentially rigid constraints to piping thermal expansion. A branch modeled appropriately in a rigorous stress analysis with the run flexibility and applied branch line movements included and where the branch connections stress is accurately known, the stress criteria noted above is used for postulating breaks locations. In order for the branch connection stress to be accurately assessed, the branch line must be included in the stress model of the main run. The NRC staff finds this statement to be in accordance with BTP MEB 3-1, Section B.1.c.(1)(a). For those cases where the branch line is not included in the stress model of the main run, terminal end breaks are postulated on the branch side of the connection. For unanalyzed branch connections or where the stress at the branch connection is not accurately known, break locations are postulated in accordance with BTP MEB 3-1, Section B.1.c(2)(b)(i). The Giambusso/Schwencer letters do not directly address the postulation of terminal end breaks in Class 2 and 3 equivalent piping at isolation valves that separate HE systems or subsystems from non-HE systems or subsystems. The BTP MEB 3-1 footnote expands the definition of terminal ends beyond that provided in Giambusso/Schwencer to include isolation valves that separate piping that is normally maintained at HE conditions from other piping that is not normally maintained at HE conditions. The Giambusso/Schwencer letters provide expectation that the postulation of terminal end breaks at rigidly fixed valves that may act to restrain thermal movement. The licensee indicated that there are no such rigidly fixed isolation valves that serve as a boundary between HE systems or subsystems and non-HE systems or subsystems at Oconee. All similar isolation valves are 'in line valves' that are not independently supported in a way that prohibit piping motion and thermal movement. The licensee does analyze the applicable HE piping systems rigorously for applicable design loads, including seismic. Breaks are postulated at locations where the actual calculated primary stress (longitudinal pressure, gravity, and OBE) and secondary stress exceed the stress thresholds given in BTP MEB 3-1, Section B.1.c(2). For all other systems or subsystems, breaks are postulated to occur at all welds and fittings. The licensee indicated that the reason for not postulating breaks at isolation valves between HE piping and non-HE piping for systems or subsystems that are rigorously analyzed is based on the similarities between a branch connection that is appropriately analyzed in the stress analysis. The licensee also indicated this for a closed isolation valve that is appropriately analyzed in the stress analysis. The NRC staff agrees that an isolation valve can be considered as a branch apart of the main run if the stress analysis shows that it has a significant impact on the main run's behavior and, therefore, needs not be considered as a terminal end. Based on the above, the NRC staff finds this acceptable.

3.1.2.5.2 Postulation of Critical Cracks

For the postulation of critical cracks, the applicability is as follows. For piping that is seismically analyzed, critical cracks are postulated in Class 2 or 3 equivalent piping at axial locations where the calculated longitudinal stress for the applicable load cases exceeds $0.4(S_a + S_h)$. Applicable load cases include internal pressure, dead weight (gravity), thermal, and seismic (OBE). For non-seismically analyzed piping, critical cracks are not postulated, since the effects of postulated circumferential and longitudinal breaks at these locations will bound the effects from critical cracks. Actual stresses used for comparison to the break and crack thresholds noted above are calculated in accordance with the ONS piping code of record, USAS B31.1.0 (1967 Edition). Allowable stress values $S_a$ and $S_h$ are determined in accordance with the USAS B31.1.0 code (Reference 39) or the USAS B31.7 Code (February 1968 Draft Edition with Errata) (Reference 40) and are, therefore, acceptable.

3.1.2.6 Dynamic Analysis

The Giambusso/Schwencer Item 4 requested a summary of the dynamic analyses applicable to the design of Category I piping and associated supports which determine the resulting loadings as a result of a postulated pipe break.

Dynamic analyses were performed in accordance with Section III-Class 2 equivalent piping (MFDW and MS) postulated HELBs in the EPR to determine the internal pressurization of the room. Dynamic analyses were also performed for postulated PH HELBs in the ventilation equipment rooms in the AB to determine internal pressurization of those rooms. Except for two MFDW rupture restraints, located in the EPR, evaluations of the effects of whip and jet impingement associated with postulated HELB locations assumed unrestrained lines.

Dynamic analyses were done for postulated breaks that required a dynamic analysis. Dynamic analyses were not required for breaks postulated to occur in the TB to determine internal pressurization, since the volume of the building is large and contains numerous openings, such that internal pressurization of the building is insignificant. Moreover, breaks within the TB were mitigated by systems credited outside of the TB. However, where the room is small without openings, as with the EPR and the ventilation equipment rooms of the AB, dynamic analyses determined the internal pressurization of the room. The NRC staff agrees that when the volume of the building is large, the pressurization due to a HELB is insignificant since the release is insignificant compared to the volume.

Dynamic analyses of HE Category I piping postulated HELBs and the effect on associated supports were not performed by the licensee, except for two MFDW rupture restraints located in the EPR. The subsequent evaluations of the impact of whip and jet impingement associated with postulated HELBs assumed unrestrained lines. Therefore, the dynamic response for these HELBs did not need to be determined, since none of the supports in the lines were designed to absorb these loads. However, the SSD equipment located in the ZoI of these breaks were assumed to fail.

For the postulated HELBs, jet impingement forces were determined in accordance with ANSI/ANS 58.2-1988. After the jet impingement forces were determined, plastic hinges were postulated, and whip interaction zones were established. After that, surveys were made of the interaction zones to identify any SSD equipment. SSD equipment located within the interaction zones were considered to be damaged and rendered non-functional.

The mitigation strategy for postulated HELBs is the availability of other equipment remote from the postulated HELB location that could be used to bring the affected unit to an SSD state. The number and location of breaks were documented in calculation OSC-11769. The orientation of the breaks was documented in subject calculation. Based on the licensee's summary of the dynamic analyses applicable to the design of Category I piping and associated supports, which determine the resulting loadings as a result of postulated pipe breaks, the NRC staff finds that the licensee's analysis is acceptable.

3.1.2.7 NRC Staff Conclusion Regarding HELB Locations and Break Types

The NRC staff reviewed the licensee's method for identifying postulated HELB locations and break types. The jet impingement forces, jet impingement cone geometry, and jet impingement effective length were determined in accordance with NUREG/CR-2913. The thrust loads for evaluating potential interactions between postulated HELBs and structural components were determined in accordance with ANSI/ANS 58.2. The results of the licensee's analyses are consistent with established methodology and regulatory positions. Based on the above, the NRC staff concludes that the licensee's methods for identifying postulated HELB locations and break types is acceptable.

3.1.3 <u>Determination of the Equipment and Systems to Mitigate Postulated HELBs</u>

The licensee identified the following functions to meet safe shutdown objectives in the proposed changes to Oconee UFSAR Section 3.6.2.6.3, "Functions to Meet Safe Shutdown Objectives," as included in Attachment 2 to the LAR:

- Reactivity Control
- RCS Inventory Control
- RCS Pressure Control
- RCS Heat Removal Control
- Reactor Building (Boundary) Integrity

- Control Room Habitability (long term)
- Plant Cooldown
- Process Monitoring
- Support Functions

These functions are performed by a set of normal plant systems for most initiating events that result in a loss of the normal power conversion system. Reactivity control is performed by the control rod system and by addition of neutron poisons using the HPI system, which also contributes to the RCS inventory control function through injection and letdown. The RCS pressure control and heat removal control functions are related and performed by a combination of delivering SG FDW through the EFW system and releasing steam to the atmosphere through the MSRVs or the atmospheric dump valves (ADVs) via the MS system. Pressure control is also supported by use of pressurizer heaters. Plant cooldown to Cold Shutdown involves additional systems for sensible heat removal, including the LPI system for shutdown cooling circulation and LPSW to transfer the heat to an ultimate heat sink. Reactor building integrity, control room habitability, and process monitoring functions, in the context of HELBs, are ensured by appropriate design of structures and supports to protect the SSCs that perform the functions. Electrical power is a key support function for many of the above functions, and electrical distribution systems are particularly vulnerable to the environmental effects produced by HELBs.

The PSW system and the SSF are defense-in-depth systems that perform many of the key functions using alternate equipment. These systems provide alternate sources of FDW for the SGs and alternate electrical distribution systems to support the key safety functions, in addition to other capabilities.

The licensee identified the following three classes of significant HELBs outside containment at Oconee:

- Overheating events that result from a loss of heat transfer (i.e., loss of SG FDW).
- Overcooling events that result from excessive heat transfer (i.e., loss of MS system pressure boundary integrity).
- Loss of reactor coolant inventory (i.e., letdown line break).

To establish the normal plant systems, PSW, and/or SSF as the assured mitigation path following a HELB, the licensee summarized their HELB criteria as follows:

1. The reactor can be shut down and maintained in an SSD condition and subsequently cooled to the CSD condition in the event of a postulated rupture, outside of the containment building, of a pipe containing a HE fluid, including the double-ended rupture of the largest pipe in the MS and FDW systems.

2. Plant SSCs required to safely shut down the reactor and maintain it in an SSD condition should be protected or designed to withstand the effects of such a postulated pipe failure.

The mitigation strategy for breaks or cracks postulated to occur in the TB is based on use of systems and equipment located in the AB or the SSF to reach an SSD condition. For the mitigation strategy for breaks or cracks postulated to occur in the AB, the licensee relies on separation or protection of certain SSCs from HELB effects using either normal plant system or the SSF to reach an SSD condition. For the Oconee mitigation strategy, 'Safe Shutdown' is defined as the transition of an ONS unit from a Mode 1 or Mode 2 condition to a Hot Standby (Mode 3) state with stable RCS conditions and maintaining this state without adversely impacting the health and safety of the public. This is unique to Oconee as compared to other PWRs that would normally go to Mode 4.

3.1.3.1 Shutdown Sequence Evaluation

The licensee addressed Giambusso/Schwencer Item 18 in Attachment 9 to the LAR. This item requests a summary of the emergency procedures that would be followed after a pipe break event or accident, including the automatic and manual operations required to place the reactor unit(s) in a CSD condition. The licensee provided a summary of the actions necessary to safely shut down the affected units for the spectrum of postulated HELB events in Section 3.6, "Operations, Response, Training, and Procedures," of the Enclosure to the LAR. To demonstrate the plant ability to shut down following HELBs in either the TB or AB, the licensee provided bounding scenarios based on consequences of the HELB with respect to operator actions, necessary repairs, manpower requirements, and the associated time limits for performing these actions. The licensee determined that HELBs occurring inside the TB have the potential to create the most bounding overheating scenario involving required operator actions, manpower requirements, and damage repairs. For overcooling scenarios, the bounding scenario is presented as a double MS HELB in the TB resulting in a loss of all AC power and reactor trip. The licensee has identified a number of TCAs as part of the HELB mitigation (e.g., actions to start the PSW system, initiate the SSF, or limit flooding by tripping the CCW pumps), and the NRC staff's evaluation of these actions is provided in Section 3.3 of this SE. A more detailed assessment of each postulated limiting HELB scenario is presented below.

3.1.3.1.1 Overheating Scenarios

The overheating analysis considers a faulted MFDW line while assuming the MS lines remain intact to maximize the overheating. These evaluated events are discussed in Section 3.6.1, "Overheating Scenarios," of the Enclosure to the LAR and are further described in Attachment 9 to the LAR.

Mitigation of these postulated HELBs is divided into four distinct phases. Phase 1 is reactor shutdown and the stabilization of the affected unit(s) in Mode 3 with RC average temperature ≥ 525°F. Phase 2 is plant cooldown from Mode 3 to Mode 4 (< 250°F). Phase 3 is the assessment and repair of SSCs required to transition the unit from Mode 4 (< 250°F) to Mode 5 (< 200°F). Phase 4 is plant cooldown to Mode 5 (< 200°F).

3.1.3.1.1.1 Main Feedwater Line Break in the Turbine Building

The licensee evaluated FDW HELBs in the TB, as described in Attachment 6 to the LAR, and included scenarios where 4160 VAC is maintained throughout the transient, and scenarios where off-site power is lost and 4160 VAC power is restored by the emergency power source, the Keowee Hydroelectric Units.

The postulated MFDW HELBs within the TB could cause a loss of AC power to all three units. When coupled with failures to CCW piping resulting in TB flooding, the licensee considered this scenario would be the bounding overheating scenario for activities necessary to place the units in Mode 5.

The Class 1E electrical system may be damaged by postulated HELBs inside the TB. The direct effects (pipe whip and jet impingement) from some HELBs may result in damage to the 4160 VAC switchgear, 4160 VAC main feeder buses, or associated cabling that may result in loss of the power sources to the 4160 VAC/6900 VAC electrical distribution systems. The effect would be similar to an SBO. To address the loss of these power sources, two alternate means of achieving an SSD condition are available through the PSW system and the SSF. These alternate means of achieving and maintaining an SSD condition are provided to address single active failure for initial mitigation.

HELBs resulting in the loss of plant systems inside the TB needed for SSD are mitigated by the available SSF and PSW system. Should the PSW system be unavailable, the SSF is credited as an alternate means of achieving and maintaining SSD following HELBs that disable plant systems inside the TB. Both PSW and SSF are located outside the TB, so flooding within the TB would not impact PSW or SSF operation.

This postulated MFDW HELB results in an overheating condition for the RCS. The reactor protection system (RPS) will trip the reactor on the loss of MFDW pumps or on high RCS pressure. Operator actions are needed to restore secondary side DHR and RCP seal cooling to establish an SSD condition.

<u>Phase 1: Reactor Shutdown</u>

Reactor shutdown and the stabilization of the affected unit(s) in Mode 3 with RC average temperature ≥ 525°F.

The overheating condition for the RCS will result in trip of the reactor on the loss of MFDW pumps or on high RCS pressure. Emergency procedures direct the operators to initiate both PSW and SSF pathways in parallel. The licensee identified both actions to establish secondary side DHR with the PSW system and with the SSF ASW pump within 14 minutes of a loss of MFDW and EFW as TCAs. Establishing SSF ASW flow within 14 minutes is an existing time critical operator action.

Using the PSW system, the PSW components (located within PSW building) and HPI (located in basement of AB) provide capability for achieving SSD. The PSW is aligned to the affected unit delivering flow to the SGs through EFW piping for inventory control and a PSW powered HPI pump is utilized to provide RCP seal cooling. Operators align PSW power outside the TB to selected pressurizer heaters, and the operators cycle the heaters as necessary to control RCS pressure.

If the PSW system is unavailable, the alternate SSF would be used to bring the reactor to shut down condition. The SSF diesel generator (DG) is emergency started and aligned to the SSF electrical system. The SSF ASW pump is started to provide inventory to SGs. The SSF RCMU pump is started to restore RCP seal cooling. The SSF ASW flow is provided to the SGs to reduce and maintain RCS pressure. The SSF ASW connection to EFW piping is located within the AB and connected upstream of SGs and downstream of EFW pumps. As such, the SSF functions are considered independent of TB HELB effects.

An SSD condition can be maintained from either the main CR using the PSW and HPI systems or from the SSF CR using the SSF ASW and SSF RCMU Systems. There are no required repairs from these postulated HELBs to achieve SSD in Mode 3 using either the PSW system or the SSF.

Phase 2: Plant Cooldown to Mode 4 (< 250°F)

Plant cooldown requires one PSW powered HPI pump to provide makeup capability. The HPI pumps are adequately protected by separation from a HELB in the TB. PSW is used to feed the SGs during plant cooldown. Due to the limited shutdown capability of the SSF, the RCS inventory control, RCP seal cooling, and SG feed functions are transferred from the SSF to the PSW system prior to initiating a cooldown.

The licensee provided a description of the sequence for achieving Mode 4. This sequence utilizes RV head vents, ADVs, PSW flow, HPI, RCS power-operated relief valves, and pressurizer heaters to stabilize the reactor in Mode 4. In this configuration, long-term subcooled natural circulation DHR conditions are maintained with RCS pressure being controlled by cycling of the pressurizer heaters and RC temperature being maintained < 250°F by natural circulation.

Phase 3: Damage Assessment and Repairs Required to Achieve Mode 5

With the plant in Mode 4, HELB damage to systems needed to allow plant cooldown from Mode 4 (< 250°F) to Mode 5 (< 200°F) would be repaired. As outlined in the LAR, key components in the sequence to achieve Mode 5 are the CCW system, the LPSW system, the LPI system, and the associated electrical power to these systems. Postulated loss of AC power to all three units would require restoring power to one CCW pump motor, two LPSW pump motors (one shared by Units 1 and 2, and one for Unit 3), three LPI pump motors (one for each unit), and the decay heat drop line isolation valves for each unit.

The licensee described that the actions that may be necessary to restore power to the pump motors and valves needed for CSD are contained in the site damage repair procedures. The necessary electrical equipment has been identified in these procedures and is available to enable the restoration of power to these motors. Power to the pump motors is provided by 4160 VAC breakers mounted on a portable trailer. Power to the 4160 VAC breaker trailer is provided by a Keowee Hydroelectric Unit via the CT4 transformer. In addition, two LPSW pump motors would need to be replaced if the CCW piping breaks and the TB floods. The licensee stated that two spare LPSW pump motors are maintained available that can be installed using existing station procedures.

Phase 4: Plant Cooldown to Mode 5 (< 200°F)

The guidance to cool down the plant to Mode 5 is contained in site operating procedures. As proposed by the licensee, the HELB strategy allows repairs to be made to any system/components required for CSD. The licensee summarized the sequence of operations to achieve CSD as follows:

1. The CCW system, LPSW systems, and LPI systems are locally aligned for operation.

2. One CCW pump is started locally at the 4160 VAC breaker trailer to supply suction to the LPSW pumps.

3. The two LPSW pumps are started locally at the 4160 VAC breaker trailer to provide cooling water to the Unit 1, Unit 2, and Unit 3 LPI coolers.

4. The decay heat drop line valves are remotely opened from the portable valve control panel. The actions taken to restore power to the drop line valves is contained in the site damage repair procedures.

5. One LPI pump is started locally at the 4160 VAC breaker trailer for each unit.

6. LPSW flow is throttled locally to the LPI coolers to establish the desired cooldown rate.

3.1.3.1.1.2 Main Feedwater Line Break in the East Penetration Room

The postulated MFDW HELB occurs downstream of the check valve in the EPR of the AB. The RPS will trip the reactor on high RCS pressure. The affected SG will completely depressurize following reactor trip resulting in an AFIS actuation that trips the MFDW pumps and isolates main and emergency FDW to the affected SG. The station electrical system is not affected by the HELB, and normal plant equipment is used for mitigation. The motor-driven EFW pump aligned to the intact SG will start automatically on the loss of both MFDW pumps, and the transient would evolve into an overheating scenario with the one motor-driven EFW pump supplying the unaffected SG and four operating RCPs. The licensee identified an existing procedural action to secure one RCP per SG as a new TCA to be completed within 15 minutes of the loss of the MFDW pumps.

This MFDW HELB event downstream of the check valve is further discussed in Section 3.1.3 of Attachment 6 to the LAR. In the evaluation, the EPR MFDW HELB is assumed to damage the instrument air header in the EPR resulting in a loss of normal letdown because the valves fail closed on loss of air. The licensee identified cycling of the RCS high point vents for alternate letdown to control RCS inventory and pressure as a new procedure step and TCA to be completed in 30 minutes for this scenario. The analysis results demonstrate that the peak RCS pressure remains below the 2750 psig limit, as evaluated in Section 3.2 of this SE.

The licensee developed plans to address postulated SAF scenarios. If an SAF prevents the HPI system from providing RCP seal cooling, the SSF RCMU system is used to provide RCP seal cooling and RCS inventory control. If an SAF prevents the EFW system from supplying the unaffected SG, HPI forced cooling (e.g., HPI feed and bleed) is utilized to provide core cooling until SSF ASW feed is established to the unaffected SG. If an SAF results in a failure of the control complex (CR, cable spreading room (CSR), and electrical equipment room) cooling, the affected unit is maintained in an SSD condition using the SSF RCMU system and the SSF ASW system.

Based on a location of the break downstream of the check valve, the normal plant equipment and electrical system will remain available to mitigate the event. Based on the above, the NRC staff finds it acceptable to credit the normal systems for mitigation of MFDW HELBs in the EPR.

3.1.3.1.2 Overcooling Scenarios

The overcooling analysis considers a faulted MS line while assuming the MFDW lines remain intact to maximize the overcooling. Both the overheating and overcooling analyses consider the possibility that the HELB causes damage that may result in the loss of on-site emergency power sources. One objective of the overcooling analysis is to demonstrate adequate core cooling and establish a basis for mitigation strategies for establishing and maintaining SSD conditions following a MS HELB in the TB or EPR. A second objective of the overcooling analysis is to demonstrate that the SG tubes remain intact and that the RCS remains within acceptable pressure and temperature limits. These objectives are evaluated as part of the thermal-hydraulic analysis in Section 3.2 of this SE.

3.1.3.1.2.1 Main Steam Line Break in Turbine Building

The bounding scenario is a MS HELB in the TB resulting in a loss of all AC power and reactor trip. The lack of MS isolation valves and the potential for one or more turbine steam admission valves to not fully close permits steam from both SGs to flow through a single break (i.e., a double MS HELB) that produces a more rapid cooldown than if steam from one steam generator flows through the break. The postulated double MS HELB leads to an overcooling condition for the RCS. The affected unit is stabilized in Mode 3 with RC temperature of approximately 325°F to 350°F.

Mitigation of these postulated HELBs is divided into four distinct phases.

<u>Phase 1: Reactor Shutdown</u>

Reactor shutdown and the stabilization of the affected unit(s) in Mode 3 with RC average temperature ≥ 525°F.

Operator actions are needed to restore secondary side heat removal and RCP seal cooling to establish an SSD condition. The SSF and the PSW systems would remain available to establish and maintain SSD for this MS HELB.

Using the PSW system, the PSW components (located within PSW building and AB) and HPI (located in the basement of the AB) provide capability for achieving SSD. The PSW is aligned to the affected unit to refill the RCS, restore shutdown cooling margin, and recover pressurizer level. The PSW powered HPI pump is utilized to provide RCP seal cooling. PSW power is locally aligned to selected pressurizer heaters and the pressurizer heaters are cycled as required to control RCS pressure. The unit would be placed in Mode 4 within 36 hours with the PSW.

Using the alternate SSF, the SSF ASW and SSF RCMU System are used to bring reactor to a shutdown condition. The SSF DG is emergency started and aligned to the SSF electrical system. The SSF RCMU pump is started to restore RCP seal cooling. RCS boundary valves are closed to isolate potential diversion flow paths. The SSF pressurizer heaters are energized to establish and maintain a ≥ 100°F subcooling margin.

<u>Phases 2, 3, and 4</u>

The activities required to cool down the plant to Mode 4 (< 250°F), perform damage assessment and repairs required to achieve Mode 5 (< 200°F), and cool down the plant to Mode 5 (< 200°F) are the same as the activities and repairs described previously for the overheating scenario above.

Loss of AC power due to the MS HELB will require use of SSF and PSW systems. The SSF is capable of bringing the plant to MODE 3 only. Therefore, PSW (or available normal system) is required to bring the plant to MODE 4, and MODE 5 would be achieved by the repair of AC power and use of plant systems.

By letter dated September 17, 2020, the licensee provided a response to a request for additional information (RAI) addressing the environmental and habitability effects of unisolated steam flow to the TB. The Oconee units have no main steam isolation valves (MSIVs) between the SGs and the turbine stop valves. Also, the licensee described certain locations where steam flow in one header can cross over to the other header. Therefore, steam from either steam generator could pass out the break location until the RCS is cooled to Mode 5.
The licensee determined that the habitability and capability to perform the necessary recovery actions would be acceptable based on the following:

- the long-term nature of the required actions.
- the significant reduction in steam flow over time following the HELB.
- the physical size of the TB structure.
- the options, if necessary, to perform isolation of the break, install barriers, redirect steam release points, or provide ventilation to improve TB access.

The MS HELBs resulting in the loss of plant systems inside the TB are mitigated by the PSW system with the SSF as a backup.

Habitability for operation of the PSW system from the CR and the SSF from the SSF control room is maintained as required to support their SSD capability. Following stabilization, if the unit is being maintained from the SSF, recovery and transfer to the PSW system for extended SSD would be performed. The licensee noted that the NRC staff previously licensed the PSW system for long-term cooling, and the PSW system would be used to maintain the affected units in SSD. The licensee determined that the long-term capability of PSW is adequate to provide ample time for dissipation of the initial steam release as well as time for decay heat to decrease such that any continued steam release in the large open TB structure is localized. These local steam releases may be managed using isolation or placement of barriers to improve TB access. Therefore, habitability conditions due to steam releases would not be expected to prevent access to the TB to recover systems needed to achieve CSD.

The NRC staff evaluated the actions necessary to achieve and maintain SSD following a MS HELB in the TB. The NRC staff finds that the PSW system, with the SSF as backup, provide acceptable mitigation for a MS HELB inside the TB until conditions for CSD can be supported.

3.1.3.1.2.2 Letdown Line Break

There is a postulated terminal end break at the letdown line containment penetration #6 upstream of the outboard containment isolation valve in the EPR. This break does not interact with any other SSD equipment, but the detection and isolation of the letdown line is important because the isolation of the letdown line terminates the loss of RCS inventory. The HPI system has adequate capacity to compensate for the leak rate as RCS pressure and pressurizer level recover, RCS remains subcooled, and RCPs remain in operation.

The letdown line break results in depressurization of the RCS as primary inventory is lost through the break and a reactor trip is initiated on either the low RCS pressure or on variable low pressure trip function. Continued RCS depressurization results in the RCS pressure decreasing to the engineered safeguards (ES) actuation point. The ES system actuation isolates the break by closing the outlet valves for letdown coolers “A” and “B”, which are the parallel inboard containment isolation valves for the penetration. To address a postulated single failure of either of these valves, the licensee identified a new time critical operator action to isolate the corresponding letdown cooler inlet isolation valve within 20 minutes of ES actuation to stop the loss of RCS inventory.

The licensee concluded that the HPI system has adequate capacity to compensate for the leak rate as RCS pressure and pressurizer level recover, RCS subcooling is not lost, and RCPs remain in operation. Following the isolation of the letdown line, unit shutdown would be conducted using the normal shutdown systems, therefore, the NRC staff finds this acceptable.

3.1.3.1.2.3 High Pressure Injection Pump Line Break

The HPI System has HELBs postulated in the EPR, the west penetration room (WPR), and the HPI pump room of each unit. The HPI pump provides RCS makeup and RCP seal cooling. The postulated HPI HELBs in these rooms may create a flooding hazard, but no adverse temperature and pressure environments are generated due to the low temperature (< 110°F) of the borated water storage tank (BWST) and/or the letdown storage tank (LDST) water.

3.1.3.1.2.3.1 HPI HELBs in East Penetration Room

In addition to the RCS letdown line break described above, there are three additional postulated HPI HELBs in the EPR:

1. A terminal end break on the train ‘A’ HPI line at containment penetration #9.

2. A terminal end break on the RCP seal injection line at containment penetration #23A.

3. A terminal end break on the RCP seal injection line at containment penetration #23B.

If a break occurs on the train ‘A’ HPI line, the operators in the CR will be alerted to the break by decreasing LDST level, decreasing pressurizer level, and low HPI header discharge pressure. The break will then be isolated by closing HP-120 (RC volume control valve) from the CR. The component cooling (CC) system remains available and RCP seal cooling will not be lost. If the CC system fails due to an SAF, RCP seal cooling will be restored by the SSF RCMU system. If this leak is not isolated within approximately 10 minutes, the flood level will exceed the top of the curb around the flood outlet device and the water will flow out of the AB to the west yard.

If either one or both seal injection lines break, the operators in the CR will be alerted to the break by RCP seal flow annunciators. The break(s) will then be isolated by closing HP-31 (RCP seal flow control valve) from the CR. If HP-31 fails to close, operators can isolate the break by closing manually operated valves, which are located outside of the break area. The CC system remains available and RCP seal cooling will not be lost. If the CC system fails due to an SAF, RCP seal cooling will be restored by the SSF RCMU system, as described above. If this leak is not isolated in approximately 1 hour and 40 minutes, the flood level will exceed the top of the curb around the flood outlet device and the water will flow out of the AB to the west yard.

For the above postulated HELB scenarios, shutdown to the CSD condition would be performed using normal plant systems, therefore, the NRC staff finds this acceptable.

3.1.3.1.2.3.2 HPI HELBs in West Penetration Room

There are two postulated HPI HELBs in the WPR:

1. A terminal end break on the RCP seal injection line at containment penetration #10A.

2. A terminal end break on the RCP seal injection line at containment penetration #10B.

These breaks would be isolated, as described above, for the seal injection line breaks in the EPR. The CC system remains available and RCP seal cooling will not be lost. If the CC system fails due to an SAF, RCP seal cooling will be restored by the SSF RCMU system. If this leak is not isolated within approximately 1 hour and 4 minutes, the flood level will exceed the top of the flood barrier located in front of the exit door from the WPR to the west yard and the water will flow out of the AB to the west yard.

For the above postulated HELB scenarios, shutdown to the CSD condition would be performed using normal plant systems because they are undamaged, therefore, the NRC staff finds this acceptable.

3.1.3.1.2.3.3 HPI HELBs in the HPI Pump Room

The HPI pump provides RCS makeup and RCP seal cooling. The postulated terminal end break at the discharge nozzle of the HPI pump will result in flooding of the HPI pump room if not isolated. The licensee specifies that mitigation of the discharge nozzle HELB of an operating HPI pump is achieved by diagnosis and isolation of a leak before damage (flooding) occurs. Following a HELB at the discharge nozzle of the operating HPI pump, the immediate response is the loss of discharge flow and the auto-start of the standby HPI pump on low RCP seal flow. Upon start of the second HPI pump, flow is restored to the RCS and the RCPs. The CR operator will receive the HPI pump discharge pressure low annunciator and the RCP seal header flow low annunciator, the standby HPI pump will auto start, and LDST level will rapidly decrease. Once the break location has been identified, the affected HPI pump will be tripped by the CR operator.

The licensee stated that a non-licensed operator would then isolate the leak by closing the remote-operated manual suction valve locally on the affected HPI pump. Operation of the isolation valve will not require entry into the HPI pump room. As indicated in Attachment 1 to the LAR, valves (2HP-103 & 2HP-107) on the individual suction lines to the Unit 2 “A” & “B” HPI pumps will be upgraded to allow the remote operation (operated outside the HPI pump room) of these valves within two refueling outages following issuance of the HELB amendments.

The licensee evaluated the TCA to isolate the break in Item 6 of Attachment 12 to the LAR. Since the HELB is detectable in the CR and ability exists to isolate without entry into the impacted room, it is expected that the plant can mitigate this HELB to avoid flooding of the HPI room. The licensee specifies that isolation of the HPI pump discharge piping break within 39 minutes is required to prevent flooding of the HPI pumps. The licensee indicates validation of this TCA will be completed during implementation and includes the validation as a conforming action in Attachment 1 to the submittal dated August 28, 2019. The licensee further indicated specific guidance to isolate a HELB on the discharge nozzle of an HPI pump will be incorporated into the loss of HPI makeup and RCP seal injection procedure. The NRC staff’s evaluation of the TCAs is in Section 3.3 of this SE.

3.1.4 <u>Identification of the Targets of Each Postulated HELB</u>

To establish the list of targets for postulated HELBs, the licensee determined SSCs required to mitigate the consequences of the postulated HELBs and safely bring the unit to a CSD condition. This list of SSCs can be determined by establishing the shutdown sequence for Oconee. In Attachment 9 to the submittal dated August 28, 2019, the licensee presented the following criteria used to identify the equipment necessary for HELB mitigation and supporting a pathway to CSD for a given postulated HELB:

- Equipment used to mitigate postulated HELBs includes those systems and components that are used for detection and isolation of specified HELBs. Equipment that is used for the detection and isolation for an identified HELB is the only detection and isolation equipment required to be targets of that specific HELB.

- Equipment used to meet any of the following shutdown objectives are considered a target of postulated HELBs:

  - Reactivity Control
  - RCS Inventory Control
  - RCS Pressure Control
  - RCS Heat Removal Control
  - RB (Boundary) Integrity
  - CR Habitability (long term)
  - Plant Cooldown

- Both primary and backup systems, used to achieve the shutdown objectives, described above, are included as shutdown equipment and targets of the postulated HELBs.

- Piping, orifices, relief valves, and check valves are considered passive type components in that they do not require an external power source or manual action to perform their intended function, and these components perform their function regardless of the environmental conditions. These components are not identified as required in the shutdown sequence because they are not subject to SAFs. They are, however, HELB targets.

- A SAF is postulated in systems used to mitigate the consequences of the postulated breaks and critical cracks or those systems used to achieve a shutdown objective of the unit. The single active component failure is assumed to occur in addition to those components damaged by the postulated pipe break.

- No SAFs are postulated during the “Plant Cooldown” phase and the “Plant Cooldown to the CSD Condition” phase.

- All available systems, including those actuated by operator actions, may be employed to mitigate the consequences of a postulated HELB or critical crack.

- In determining the systems and components available to mitigate the consequences of postulated HELBs, all shutdown equipment is assumed to be operable and available at the start of the postulated HELB sequence. It is not necessary to postulate that any systems or components are out of service for maintenance.

- Although a postulated HELB outside of the containment building may ultimately require a CSD, holding at Hot Standby is allowed in order to allow plant personnel to assess the situation and make any necessary repairs to allow the unit to reach CSD.

The NRC staff evaluated these criteria against the guidance in the Giambusso/Schwencer letters and the licensing basis established in the licensee's original response. The NRC staff confirmed that the criteria defined in the first five bullets above are consistent with the guidance in the Giambusso/Schwencer letters. In addition, the use of all available systems, including those actuated by operator actions, to mitigate the effects of a postulated HELB is a criterion consistent with the existing licensing basis and does not represent a licensing basis change. However, the NRC staff reviews each time critical operator action to ensure that the completion of the action in the specified time is reasonably assured, as described in Section 3.3 of this SE. The exclusion of single failures during the "Plant Cooldown" and the "Plant Cooldown to Cold Shutdown" phases is consistent with the precedent established for other plants licensed to the Giambusso/Schwencer letters criteria, where the single failure criteria is only applied to Hot Shutdown systems because cooldown systems for plants of that vintage were not licensed for redundant cooldown capabilities and Hot Shutdown provides reasonable assurance that public health and safety would be adequately protected.

3.1.5 Determination of the Shutdown Equipment Undamaged by the Postulated HELB

The licensee evaluated the bounding damage states for postulated HELBs and described the limiting conditions for the HELB scenarios in Section 3.6 of the Enclosure to the LAR. The licensee provided the following description of the analysis to determine the equipment affected by the postulated HELBs:

> All of the postulated HELBs outside of the containment building are described in calculation entitled, "Analysis of Postulated HELBs Outside of Containment."
>
> HELB mitigation is dependent on the location and magnitude of the HELB as well as its interactions with SSD equipment. The consequences of the HELB interactions were reviewed to determine if one HELB could be found that was bounding with respect to operator actions, necessary repairs, manpower requirements and the associated time limits for performing these actions. It was found that HELBs occurring inside the TB have the potential to create the most bounding scenario involving required operator actions, manpower requirements and damage repairs.

The licensee described the basic HELB mitigation strategy in the proposed new Section 3.6.2.6.1, "HELB Mitigation Strategy," of the ONS UFSAR, which the licensee provided in Attachment 2 to the submittal dated August 28, 2019. The licensee identified the following major points of the strategy:

- Required SSCs located in the TB are not impacted by HELBs postulated to occur in the AB or the yard.
- Required SSCs located in the AB are not impacted by HELBs postulated to occur in the TB.
- SAFs are imposed for those components required for initial mitigation.
- SAFs are not imposed for those components required to initiate a cooldown of the plant.

- HELBs resulting in the loss of plant systems inside the TB needed for SSD are mitigated by the PSW system.
- Should the PSW system be unavailable, the SSF is credited as an alternate means of achieving and maintaining SSD following HELBs that disable plant systems inside the TB.
- HELBs resulting in the loss of plant systems inside the AB needed for SSD are mitigated by normal plant systems or the SSF.

For postulated HELBs within the TB, the licensee relied on systems separated from the TB for the initial mitigation to stabilize and maintain the plant in Hot Shutdown. The licensee identified the PSW system and the SSF as the primary and backup systems for a TB HELB, thereby accommodating an SAF. For plant systems needed to cooldown and place the plant in Cold Shutdown, the licensee proposed to repair or replace damaged components in the TB, if necessary. For postulated HELBs in the AB, the licensee relied on separation or protection of certain SSCs from HELB effects.

3.1.5.1 Direct Effects from Pipe Whip and Jet Impingement

The licensee described the process to evaluate the direct pipe whip and jet impingement effects of postulated HELBs on equipment in Attachment 9 to the LAR. The licensee stated that, except for the MFDW pipe rupture restraints located at the two MFDW containment penetrations in each unit's EPR, the evaluations assumed unrestrained lines. For each unrestrained line, the licensee determined ZoI based on jet impingement loads and resulting pipe whip. The licensee performed surveys of each of these break locations to identify SSD equipment within the ZoI. The licensee considered any SSD equipment located within the ZoI as damaged and non-functional. Therefore, except for the MFDW break at the containment penetration in the EPR, the licensee relied on SSCs remote from the break location for HELB mitigation to achieve and maintain Hot Shutdown conditions. The MFDW pipe rupture restraints have been included in the HELB licensing basis, and the licensee did not propose a change to their function or design.

The licensee determined that MS and MFDW HELBs in the TB have the potential to directly cause damage to equipment that provides essential support functions for normal plant systems, equipment necessary to place a plant in Cold Shutdown, and structural components of the TB. The essential support equipment includes the switchgear and electrical cables in the TB, and the licensee characterized the effect of the damage as equivalent to an SBO. Accordingly, the licensee proposed to place the affected units in Hot Standby using the PSW system, which is supported by a separate electrical distribution system and includes no components located in the TB. If the PSW system is unavailable, the SSF provides a redundant capability, both independent of the normal electrical distribution system and located outside the TB, to place the affected units in Hot Standby without reliance on equipment located in the TB. To address the potential for damage to systems and components necessary to place affected units in CSD, the licensee proposed staging equipment and developing procedures to promptly repair the LPSW system and supporting electrical switchgear. In addition, the licensee determined that certain line breaks could cause damage to structural columns in the TB that has the potential to fail the columns. The licensee proposed implementation of modifications to prevent the potential failure of the TB structural columns to withstand postulated pipe whip loads three refueling outages after issuance of the HELB license amendments, as outlined in Attachment 1 to the LAR.

The licensee identified a number of postulated HELBs within the AB. For each unit, the licensee identified the following AB HELBs as limiting with respect to mitigation actions:

- Two MFDW terminal end break locations, one at each containment penetration in the EPR downstream of the check valve, with pipe whip restraint and guard pipe to prevent direct interaction.
- One MS terminal end break at the EPR containment penetration determined not to directly interact with SSD systems.
- A letdown line terminal end break at the EPR containment penetration determined not to directly interact with SSD systems.
- One HPI injection line terminal end break in the EPR determined not to interact with other SSD systems.
- An HPI pump discharge line terminal end break at the discharge nozzle of the A or B HPI pumps determined not to interact with SSD systems.
- Four RCP seal injection line terminal end breaks at the containment penetration (i.e., two RCP seal injection lines in the EPR and two RCP seal injection lines in the WPR) determined not to directly interact with SSD systems.

The NRC staff reviewed the analysis of the protection against direct effects for the TB and AB HELB identified locations as specified in Items 1 and 5 of the Giambusso/Schwencer letters. This included a review of licensee documents via an audit process, as documented in the regulatory audit plan and summary (Reference 17). Through a sampling of documents, the NRC staff verified that the licensee had performed surveys of the physical location of postulated HELBs to assess the effects of the postulated breaks. The licensee found that many analyzed break locations did not impact any SSD function, while other break locations were determined to impact functions required for SSD of the plant requiring further analysis. These interactions were analyzed further to assess the impacted functions in detail and ensure that Cold Shutdown is achievable. With this additional analysis, the licensee also defined SSD equipment and operator actions that are necessary to mitigate the event, stabilize the RCS in Hot Standby, and perform RCS cooldown.

<u>Turbine Building HELBs</u>

The licensee described a protection strategy for HELBs in the TB that relies on the PSW system or, if the PSW system is unavailable, the SSF for initial mitigation of the HELB. The NRC staff evaluated the separation provided for the PSW system and the SSF using plant arrangement drawings and the descriptions of the PSW system and the SSF. The NRC staff concluded that the location of PSW system and SSF components are separated adequately and provide redundant capability to permit operators to place the plant in the Hot Standby condition. The limiting conditions for operation of the PSW system and the SSF are specified in each of the Oconee unit’s Technical Specifications (TSs) as TS 3.7.10, “Protected Service Water (PSW) System,” and TS 3.10.1, “Standby Shutdown Facility (SSF),” and provide reasonable assurance that these systems would be available for mitigation of a HELB as considered in the licensee’s analysis. The NRC staff finds that the physical separation of the PSW system and the SSF provides acceptable protection against pipe whip and jet impingement for HELBs within the TB, thereby satisfying Item 1 of the Giambusso/Schwencer letters.

The NRC staff also considered protection from direct effects for essential SSD systems used for placing the affected unit(s) in CSD. These systems provide for long-term mitigation of HELBs within the TB. The licensee’s proposed strategy permits repair of SSCs necessary to place the plant in CSD and, consequently, excludes consideration of single failures affecting those SSCs. The licensee determined that the direct effects of HELBs in the TB could damage certain structural support columns, electrical cables, electrical switchgear, and service water and CCW piping. With respect to SSD functions, these damage conditions have the potential to directly or indirectly affect the availability to use the LPSW system located in the TB and electrical power for a CCW pump and LPI system components to provide subcooled decay heat removal to place and maintain the units in CSD.

The licensee addressed the damage states associated with TB HELBs. In Attachment 1 to the submittal dated August 28, 2019, the licensee listed conforming actions associated with the proposed HELB licensing basis amendment. These actions include modifications to several TB structural columns to prevent potential column failure when exposed to the design pipe whip load. The list of conforming actions in the TB also includes modifications to ensure that necessary LPSW system isolations can be made to enable operation of the alternate reactor building cooling system, considering the potential for direct HELB damage to several LPSW system valves. The licensee also proposed a conforming change to eliminate cross-connections between units for the CRD system to ensure that a loss of power to the unit results in an immediate trip of the affected unit by ensuring an immediate loss of power to the CRDs. To ensure that these modifications are completed prior to implementing the revised HELB mitigation, the licensee proposed to update the UFSAR by stating that the revised HELB mitigation strategy will be implemented once the conforming actions of Attachment 1 to the submittal dated August 28, 2019, are completed for each unit.

The NRC staff requested additional information regarding the potential for damage to the CCW and LPSW systems from MFDW HELBs in the TB and the consequential indirect effect of TB basement flooding that could result from that damage. The licensee addressed this in its response to RAI-16 by letter dated September 17, 2020 (Reference 3). The licensee identified several locations in the TB where direct effects of a HELB could cause damage to service water and CCW piping in the TB. However, the licensee explained that the rates of leakage, additional water removal capability resulting from installation of a large drain from the turbine building basement, and improved flood protection between the TB and AB would permit substantial time to reduce the rate of in-leakage and make repairs. The available time for HELB operator response exceeded previous TCAs to reduce flow from CCW non-HELB flooding sources as a result of the improved drain capacity. In addition, the PSW system, backed-up by the SSF, provides adequate capability to maintain the plant in Hot Shutdown for a period adequate to implement repairs to isolate the leak and prepare for cooldown to CSD. Therefore, the NRC staff concludes that protection from the direct effects of HELBs on the LPSW and CCW systems is acceptable and consistent with Item 1 of the Giambusso/Schwencer letters.

<u>Auxiliary Building HELBs</u>

The licensee described a protection strategy for HELBs in the AB that relies on normal plant systems or, if normal plant systems are unavailable, the SSF for initial mitigation of the HELB. The EPR of the AB contains the containment penetrations for piping systems that provide FDW to the SGs (including MFDW, EFW, PSW, and SSF FDW), HPI to some RCP seals, and other functions necessary for normal operation, accident mitigation, or SSD. The WPR of the AB contains the containment penetrations for piping systems that provide HPI to the remaining RCPs and other functions necessary for normal operation, accident mitigation, or SSD.

The licensee also evaluated the effects of postulated terminal end breaks at containment penetrations on containment integrity. The licensee stated that the RB penetrations for all HE lines were designed to withstand the forces and moments applied to the terminal end that could occur from postulated breaks located either inside or outside of the containment building. The MS and MFDW RB penetrations differ from the other RB penetrations because they include structural anchors installed adjacent to the RB penetrations to restrain the piping. The MS anchors are located inside the RB, while the MFDW anchors are in the EPR. These anchors are designed to absorb the large forces and moments that could occur in the aftermath of either a postulated MS or MFDW break. The MS and MFDW anchors consist of a collar wrapped around the outside diameter of the piping. The collar is connected at both ends to the piping via two circumferential fillet welds. The collar is in turn welded to a series of structural wide flange members that span back to the RB wall. The wide flange members are then welded to embedded structural tees located in the RB wall. The licensee provided a simplified sketch of the MFDW anchor in Attachment 9 to the submittal dated August 28, 2019.

The NRC staff evaluated the HELB restraints and the separation provided between postulated break locations and the systems necessary for mitigation of the break. The NRC staff reviewed plant piping arrangement drawings for the EPR and the licensee evaluations of direct HELB effects from postulated breaks in the AB. The NRC staff determined that the location of the postulated MS terminal end HELB in the EPR (the other MS line exits the RB outside other structures directly into the yard area) was adequately separate from other systems, thereby supporting the licensee conclusion that direct interaction would not reasonably occur. The two postulated MFDW terminal end HELB locations at the penetrations within each EPR were suitably restrained and jet impingement effects directed by a guard pipe such that the direct effects would not damage normal systems necessary for mitigation. The SSF components within the WPRs were similarly protected from the direct effects of the terminal end HELBs at the containment penetrations. The limiting conditions for operation of the normal mitigation systems and the SSF are specified in the Oconee TSs and provide reasonable assurance that these systems would be available for mitigation of a HELB as considered in the licensee's analyses. The NRC staff finds that the physical separation between the normal mitigation systems and the one MS penetration within the EPR and the MFDW pipe restraints and guard pipe provide acceptable protection against pipe whip and jet impingement for the limiting terminal end HELBs within the EPR, thereby satisfying Giambusso/Schwencer Items 1, 5, and 21.

The NRC staff also evaluated the other lower-energy terminal end HELBs postulated in the AB. The licensee determined that the letdown line, HPI, and RCP seal injection line breaks in the AB would not directly interact with other piping systems to the extent that safety functions would be affected. Based on its review of the area drawings for the arrangement of piping in the EPR and evaluation of the systems, the NRC staff finds the licensee's determination acceptable and sufficient to address Giambusso/Schwencer Items 1 and 21.

3.1.5.2 Indirect Environmental Effects

Consistent with the Giambusso/Schwencer letters, the licensee also evaluated environmental and other indirect effects of the postulated HELBs. These effects included the following considerations:

- Compartment pressure and temperature response to HELB.
- Structural response to compartment pressurization and other HELB effects.
- Flooding impacts induced by the HELB.
- Maintenance of required redundancy in protective systems.
- Habitability impacts on the control room.
- Environmental qualification of essential electrical mitigation equipment in HELB areas.

3.1.5.2.1 Pressure and Temperature Response to HELB and Structural Effects

The licensee evaluated the pressure and temperature response to HELB in the TB. The licensee determined that the large volume and open construction of the TB would preclude any significant pressurization, therefore, the licensee did not conduct a detailed analysis of the building pressure and temperature response. The licensee concluded that the TB structure would experience only localized damage to structural columns as a result of pipe whip and jet impingement, and the licensee has identified conforming modifications to the columns to ensure that SSD conditions can be achieved and maintained.

The licensee performed a more detailed analysis of HELBs in the AB. In Attachment 9 to the LAR, the licensee described an analysis addressing limiting breaks in the EPR and WPR. The EPR and WPR, which are adjacent to the RB, contain piping terminal ends at containment penetrations and would experience more significant pressure and temperature transients in response to HELBs because of the limited volume of the rooms. By report dated April 25, 1973 (Reference 21), in the response to the Giambusso/Schwencer letters, the licensee described the installation of light weight blow-out panels in selected walls of each unit's EPR to reduce the peak pressure in these rooms and prepared an analysis of the pressure and temperature response. The licensee has since updated this analysis to use the GOTHIC Code and revised blowdown data for the MS and MFDW HELBs and made additional modifications to the EPR.

The licensee provided additional information regarding the updated EPR analysis in its RAI response letter dated September 17, 2020 (Reference 3). The licensee described that the pipe movement and jet direction/length for each postulated HELB was determined based on information collected during walkdowns of the break locations. The MS terminal end break was postulated to occur at a weld a short distance from the RB wall. From the postulated break location, the MS pipe bends downward at an elbow adjacent to the break location and back to horizontal at a second elbow about 7.5 feet below the first elbow. The licensee determined that this pipe configuration would result in the primary jet from the SG in the RB pointing outward from the RB wall toward blow-out panels and pipe movement away from the break location, which directs the secondary, and much weaker jet upward. Thus, the pipe motions and jet effects are directed away from essential SSD equipment in the EPR.

In its letter dated September 17, 2020, the licensee also described the GOTHIC model of the EPR, which used four volumes, as follows:

- Volume 1 represented the narrow volume between a column line and the WPR wall and was modeled by lumped parameters with junctions at the top and bottom to the main volume (Volume 2) to represent recirculation flow through the mostly open boundary.
- Volume 2, which represented the main EPR volume between the North-South column line by the WPR and an East-West column line, was subdivided into a fine calculational mesh to support accurate modeling of the flow patterns in the EPR and the orientation of the various postulated pipe breaks.
- Volume 3 represented a lumped parameter volume consisting of the remainder of the EPR that has a higher floor elevation than Volumes 1 and 2 and that largely acts as a flow path between several junctions with Volume 2 and some blow-out panels.
- Volume 4 represented the atmosphere outside the blow-out panels and was connected to the EPR by junctions representing the blow-out panels.
- The junctions between the volumes represented large openings and were modeled with low loss coefficients to reflect the large flow area. For the junctions experiencing

significant flow due to pressurization, the licensee selected somewhat larger values to reflect the effect of rapid pressurization.

- Several junctions between the EPR volumes and the atmosphere represented the blow-out panels as quick-opening valves with setpoints between 0.4 to 6.8 psid.

In Attachment 9 to the submittal dated August 28, 2019, the licensee described how operator actions and protective instrumentation actuations were modeled in the analysis. For MS HELBs, the analysis assumes that no operator action is taken within the first 10 minutes. The MFDW pumps are assumed to be tripped by AFIS when the actuation setpoint is reached, but the MFDW control valve is assumed to fail open. MFDW is assumed to continue feeding the faulted SG via the condensate booster pumps until MFDW is isolated by the operator at 10 minutes. The analysis models that EFW is automatically stopped by AFIS when the actuation setpoint is reached. For MFDW HELBs, the analysis assumes MFDW continues until the condensate inventory is depleted (i.e., no operator action assumed to isolate MFDW).

The licensee evaluated the pressure and temperature effects of the MS terminal end break, the limiting MFDW terminal end break, and the limiting MFDW critical crack break in each unit's EPR using the above model. The MS terminal end break produced the highest temperature and pressure in the EPR, with peak pressure as high as 4.12 psig and a peak temperature of 482°F in each room. The MS primary jet orientation directed some of the flow into Volume 3, and the pressurization of Volume 3 also resulted in the immediate opening of blow-out panels between Volume 3 and the atmosphere. Therefore, the licensee concluded that the lumped parameter modeling did not impact the behavior of the blow-out panels in Volume 3. The licensee also determined that Volume 1 was sufficiently distant from the MS terminal end break that its lumped parameter modeling also did not impact the analysis.

The licensee provided the following statement regarding structural analysis of the EPR for the loading imposed by the MS HELB in its response letter dated September 17, 2020:

> An analysis of the EPR structure was performed with pressure responses that bound those developed for the MS HELB pressures listed above. This analysis is contained in [a licensee calculation]. All analyzed reinforced concrete and steel structural components for each of the penetration rooms met either the elastic or plastic acceptance criteria. In all cases, the ceiling structure meets either elastic or plastic criteria and demonstrates that gross failure of the ceiling structure will not occur following the postulated MS HELB.

The licensee provided additional descriptions of the AB and TB response to HELB pressure effects in Attachment 9 to the LAR under Giambusso/Schwencer Items 6, 8, 10, and 20. The licensee determined that only the EPR and WPR ceilings would be subjected to significant reversal of load. The ceiling structures are normally loaded with equipment and dead loads. Pressurization of the EPR and WPR due to postulated MFDW and MS HELBs located in the EPR would exert an upward load on the ceiling structure followed by a reestablishment of the equipment and dead loads. The licensee stated that the evaluation results show that the loadings would be within the acceptable limits. The CRs, cable spreading rooms, and control battery room are separated from the penetration rooms by reinforced walls. However, several unreinforced masonry walls are expected to crack and potentially fail. The NRC staff discusses the effect of these masonry wall failures in the subsection addressing habitability below because they separate CR ductwork from the EPR.

The NRC staff reviewed the supporting HELB pressure and temperature response analysis as part of its audit activities and evaluated the licensee's response providing modeling details, initial conditions, assumptions, and results for the analysis. The NRC staff determined that the modeling, initial conditions, and assumptions were reasonable for the postulated HELB analysis. The NRC staff finds that the analysis results and other information provide reasonable assurance that the TB and EPR structures would withstand the pressurization resulting from postulated HELBs without significant damage that could challenge the completion of SSD functions and would be sufficient to address Giambusso/Schwencer Items 8, 10, and 20.

3.1.5.2.2 Flooding Impacts of Postulated HELBs

Giambusso/Schwencer Item 15 requests a discussion of the potential for flooding of safety-related equipment in the event of failure of a FDW line or any other line carrying a HE fluid. The licensee provided this information in Attachment 9 to its submittal dated August 28, 2019. Giambusso/Schwencer Item 17 requests a discussion of the leak detection equipment capabilities associated with any flood mitigation modification.

Auxiliary Building

The licensee determined that postulated pipe failures in the MFDW system can lead to flooding inside the EPR, and the licensee installed flood protection modifications in these rooms, including barriers and flood outlet devices. The licensee stated that this design ensures that flood water from the MFDW line breaks is released to the outside at a rate sufficient to prevent submergence of the electrical penetrations in the EPR. The resulting water level inside the EPR is limited to 2 feet, and the licensee determined that the AB floor structure can sustain a 2-foot flood height without failure. Flood impoundment walls were installed in each EPR to limit flood water from being released to other areas of the AB. Any water released to other areas of the AB could eventually reach the HPI pump rooms. The flood impoundment walls protect the HPI pump rooms from flooding caused by line breaks inside the EPR.

The licensee also identified that postulated pipe ruptures on the discharge of the 'A' or 'B' HPI pumps could lead to flooding of the HPI pump rooms. The licensee determined that sufficient time exists for the operators to diagnose and isolate the break to preclude the loss of all HPI pumps due to flooding. To address failures of active components, the licensee described conforming modifications to support isolation of the faulted pump discharge while keeping two HPI pumps and two HPI flow paths available for achieving and maintaining an SSD condition. This modification would maintain the HPI function assuming an SAF.

As discussed previously, postulated HELBs in the TB can result in a loss of 4160 VAC power. A loss of 4160 VAC power results in a loss of spent fuel cooling and heat removal through greatly increased evaporation from the spent fuel pool (SFP). The licensee described that condensed water vapor from the SFP could drain to the first floor of the AB and flood the safety related HPI pumps. The licensee stated that procedures are in place to vent water vapor from the spent fuel building and to block the spent fuel building floor drains to prevent HPI pump flooding from occurring.

The NRC staff evaluated the installed provisions and operator actions to mitigate potential flooding within the AB as a result of a HELB. The NRC staff determined that reasonable modifications have been made and appropriate operator actions have been identified to accommodate flooding effects associated with postulated AB HELBs. Based on the above, the

NRC staff finds that the licensee addressed Giambusso/Schwencer Item 15 sufficiently for the AB.

Turbine Building

The licensee identified several piping segments in the LPSW and CCW systems that could be damaged by pipe whip effects in the TB and result in flooding. Floods in the TB can be identified by a flood detection system, which provides the CR alarms to warn of a flood. Systems and components located in the TB basement are not protected from flooding by barriers, but the licensee enhanced the drain capacity. The licensee credits existing flood protection measures to ensure that TB flooding does not propagate to affect the AB. Damage repair guidelines are credited to terminate the source of flooding and repair those systems and components necessary to reach CSD (e.g., the LPSW system).

By letter dated September 17, 2020, the licensee responded to an RAI by addressing the potential for TB flooding in greater detail. The licensee stated that both direct and indirect causes of flooding had been evaluated. For direct effects, the licensee identified seven postulated MFDW HELB locations across the three units where the licensee determined that LPSW piping was within the pipe whip impact zone, and thereby considered a flood source in addition to the flooding from the MFDW system. Although the licensee identified multiple LPSW pipe segments within certain MFDW pipe whip zones, the licensee determined that the maximum equivalent cross-sectional area, if all pipes in the area were ruptured, was bounded by the design-basis TB flood associated with CCW expansion joint failure. The licensee also described potential indirect effects from localized structural damage that could result in increased flooding. The licensee assumed that the localized structural damage could result in collapse that would impale or crush piping, including the 78-inch nominal diameter CCW piping, however, conforming modifications to the TB columns would reduce the likelihood of localized structural damage and the consequent flooding. The licensee determined this type of damage could result in flood rates exceeding the design-basis TB flood but found by analysis that the modification to install the TB drain would allow a longer time period for operator action to trip the CCW pumps than assumed for the design-basis TB flood. Regardless of flood cause/source and the action to trip the CCW pumps, the licensee stated that additional actions could be necessary to completely stop the flooding, potentially involving either isolation of the damaged piping section or lowering of the Keowee Lake level. Once the flooding source is stopped, the floodwater would recede through the TB drain and permit completion of repairs within several days.

The licensee described the capability of TB flood detection instrumentation for HELB-related flood mitigation. The TB water level system provides alarms at two separate TB sump water levels on annunciators in the Unit 2 CR and computer alarms in the Unit 3 CR. The licensee evaluated the path of these alarm inputs from the level switches in each unit to the control panel outside the Unit 2 CR.

In its letter dated September 17, 2020, the licensee addressed the capability of the PSW system, as backed-up by the SSF, to support recovery from TB flooding. The PSW system is located outside of the TB. The licensee stated that the damage to the service water piping in the TB would not affect the long-term cooling capability of the PSW system. The PSW pump does not rely on the service water piping as a water source because it takes suction from the Unit 2 embedded CCW piping and a submersible pump installed in the intake canal provides makeup to the embedded CCW piping.

The NRC staff evaluated the effect of TB flooding on the capability to achieve and maintain SSD. The proposed mitigation strategy relies on the PSW system with the SSF as backup, which provides the capability to maintain SSD for an extended period. This period in Hot Shutdown would allow for necessary repairs to support further cooldown to CSD involving electrical switchgear, LPSW pump motors, and damaged piping. Based on the above, the NRC staff finds the response sufficient to address Giambusso/Schwencer Item 15 for the TB.

3.1.5.2.3 Maintenance of Required Redundancy in Protective Systems

The licensee addressed Giambusso/Schwencer Item 11 in Attachment 9 to the LAR. This item specifies that rupture of a pipe carrying HE fluid should not directly or indirectly result in either:

a. Loss of required redundancy in any portion of the protection system, Class 1E electrical system, ES equipment, cable penetrations, or their interconnecting cables required to mitigate the consequences of that accident and place the reactor(s) in a CSD Condition; or
b. Environmental induced failures caused by a leak or rupture of the pipe which would not of itself result in protective action but does disable protection functions. In this regard, a loss of redundancy is permitted but a loss of function is not permitted. For such situations plant shutdown is required.

The licensee described that the core protection systems at ONS consist of the RPS and ES systems. The RPS trips the reactor to prevent exceeding acceptable fuel damage limits. The ES system automatically initiates the HPI and LPI Systems on either a low RCS pressure or high containment pressure. The cabinets for the RPS and ES systems are physically located inside the control complex and, therefore, are protected from the effects of postulated HELBs outside of the containment building. HELBs outside of the containment building can lead to either inadequate heat transfer or excessive heat transfer in the RCS. Inadequate heat transfer results in high RCS pressure conditions. Excessive heat transfer results in low RCS pressure conditions. The RPS trip on high RCS pressure is credited for inadequate heat transfer. The RPS trip on low RCS pressure or variable low RCS pressure is credited for excessive heat transfer. The ES system is expected to be actuated following excessive heat transfer or letdown line breaks. There are RCS pressure and temperature instruments that feed the RPS. These instruments are located inside of the containment building. The containment pressure instruments are not required for HELBs outside of the containment building. There are RCS pressure transmitters that feed into the ES system. These instruments are also located inside of the containment building. The cabling from these instruments for RPS and ES systems are routed through the penetration rooms to the CSR. The electrical penetrations and the associated cabling for the instruments are qualified for the environmental conditions inside the penetration room, and these cables are not impacted directly by any postulated HELBs in the penetration room. Therefore, there is no expected loss of required redundancy due to the effects of postulated HELBs outside of the containment building.

The licensee provided the following description of capabilities to maintain essential functions following environmental effects of HELBs:

> The Class 1E electrical system may be damaged by postulated HELBs inside the TB. The direct effects (pipe whip and jet impingement) from some HELBs may result in damage to the 4160 VAC switchgears, 4160 VAC main feeder buses, or associated cabling that may result in loss of the power sources to the 4160 VAC/6900 VAC electrical distribution systems. The effect would be similar to a

SBO. To address the loss of these power sources, two alternate means of achieving a SSD condition are available through the PSW system and the SSF. Two alternate means of achieving and maintaining a SSD condition are provided to address SAFs.

The PSW system is capable of maintaining the SSD condition from the unit CR. Electrical power to the PSW system is provided from either the 100 kV power line or from a Keowee Hydroelectric unit outside the TB. The PSW pump can be started from the CR to feed either or both SGs to maintain secondary side heat removal. The PSW pump also supplies cooling water to the HPI pump motors. Power for one HPI pump can be restored from the PSW electrical system as well as selected motor-operated valves to align pump suction to the BWST and control flow to the RCS via the ‘A’ injection header and RCP seal injection. RCS pressure can be controlled by using pressurizer heaters powered from the PSW electrical system. Finally, the control batteries serving the 125 VDC [volts direct current] and 125 VAC Vital I&C systems can be recharged from the battery chargers powered from the PSW electrical system.

The SSF is capable of maintaining the SSD condition from the SSF CR. The SSF power system includes 4160 VAC, 600 VAC, 208 VAC, 120 VAC and 125 VDC power. It consists of switchgear, a LC, MCCs, panelboards, remote starters, batteries, battery chargers, inverters, a DG [diesel generator], relays, control devices, and interconnecting cable supplying the appropriate loads. The SSF power system provides electrical isolation of SSF equipment from non-SSF equipment. The SSF 125 VDC power system provides a reliable source of power for DC loads needed to black start the DG. The DC power system consists of two 125 VDC batteries and associated chargers, two 125 VDC distribution centers (DCSF, DCSF-1), and a DC power panelboard (DCSF). The SSF power system is provided with standby power from a dedicated DG.

With the unit(s) being maintained in a SSD condition, there is no immediate need for plant cooldown. Damage repair guidelines will continue to be credited to restore power to systems and components needed for plant cooldown to CSD conditions. As part of the damage repair procedures a portable valve control panel would be installed and wired to allow closure of the core flood outlet valves (CF-1 & CF-2) when conditions permit their closure. In addition, the portable valve control panel would allow the opening of the decay heat drop line isolation valves (LP-1 & LP-2) when entry conditions for normal DHR are established.

Some ES equipment may be lost due to possible flooding inside the TB basement, specifically the LPSW pumps. The LPSW pumps for all three units are located in the TB basement. Some postulated HELBs inside the TB may result in ruptures to the CCW piping. ES equipment (HPI and LPI) located inside the AB are protected from the effects of flooding inside the TB by the existing flood protection measures/barriers and the TB drain located at the south end of the TB. The EFW pumps, although not classified as ES equipment, are also located in the TB basement. TB flooding can result in the loss of LPSW and EFW on all three units. Damage repair guidelines are credited to restore the LPSW Systems once the source of flooding has been isolated to enable a plant cooldown to CSD conditions. Replacement motors and associated cabling for the LPSW pumps are stored in a protected warehouse.

SAFs are not postulated in establishing plant cooldown and the establishment of CSD.

Although certain normally essential support systems may be lost as a result of the environmental effects of postulated HELBs, the licensee maintains acceptable alternate capability (i.e., the PSW system and SSF) that is protected from the effects of the postulated HELBs that the PSW system and SSF are credited to mitigate. The NRC staff finds the above strategy sufficient to address Giambusso/Schwencer Item 11.

3.1.5.2.4 Habitability Impacts on the Control Room

Giambusso/Schwencer Item 12 specifies that assurance be provided that the CR will be habitable and its equipment functional after a MS or MFDW line break or that the capability for shutdown and cooldown of the unit(s) will be available in another habitable area.

The CRs, CSRs, and electrical equipment rooms are provided with air conditioning systems, described in UFSAR Section 9.4.1, to maintain a suitable environment for personnel and equipment. Chilled water is supplied to the HVAC systems from the CR ventilation chilled water system as described in UFSAR Section 9.2.5. Electrical power to the HVAC systems as well as the chilled water system itself is vulnerable to the effects of HELBs inside the TB. Following a HELB in the TB that results in a loss of cooling to the CRs, CSRs, and electrical equipment rooms, the licensee stated that an alternate cooling system would be placed in operation to ensure that these areas remain habitable. The alternate cooling system is located outside the TB and would remain free of HELB damage.

The licensee determined that postulated MS and MFDW line breaks inside the EPR do not result in a direct loss of CR habitability, but CR habitability may be subject to indirect degradation through the failure of block walls separating the CR ductwork from the EPR. The integrity of the CR is protected by a reinforced concrete wall between it and the EPR, however, the CR HVAC system ductwork serving the Unit 1 and 2 CR is partially located inside a duct shaft adjacent to the EPR. The unreinforced masonry walls of the duct shaft may crack and potentially fail due to the compartmental pressurization effects following either a MS or MFDW HELB postulated inside the EPR. The CSR is also protected by a combination of HELB blast walls and doors, as well as a reinforced concrete wall between it and the EPR. However, the HVAC duct work serving the CSR has a discharge register into the stairwell adjacent to the EPR, and the licensee determined that an unreinforced masonry wall separating the stairwell from the EPR may crack and potentially fail due to the compartmental pressurization effects following either a MS or MFDW HELB postulated inside the EPR. A failure of this wall could fill the stairwell with steam from the postulated HELBs inside the EPR. The discharge register to the stairwell is equipped with a fire damper that should close if it were subjected to a steam environment, but this capability does not satisfy the applicable design basis. Furthermore, for Units 1 and 2, the HVAC system serving the electrical equipment room, which is directly below the CSR, uses the same duct shaft that is adjacent to the EPR. In Attachment 1 to the submittal dated August 28, 2019, the licensee identified conforming modifications to the CR, CSR, and electrical equipment room (the control complex) cooling systems for all three units that address the potential for indirect effects on the control complex through the above described interactions.

The control complex for each unit is protected from direct HELB effects, and the licensee has proposed conforming modifications to the cooling systems that address the potential for indirect effects. Based on the above, the NRC staff finds that licensee's response is sufficient to address Giambusso/Schwencer Item 12.

3.1.5.2.5 Environmental Qualification and Electrical Review

Giambusso/Schwencer Item 13 requested that EQ of equipment that must function after exposure to the indirect effects of pipe break be demonstrated. The NRC staff evaluated whether electrical equipment and components would remain bounded by the existing EQ due to the proposed changes in MS and MFDW HELB mitigation.

The licensee stated in Attachment 9 of its submittal dated August 28, 2019,that environmental profiles were evaluated for the MFDW and MS HELBs postulated to occur in the TB and EPR of the AB. Specifically, the licensee stated that the temperature and pressure profiles for components inside the EPR and WPR rooms were changed. The licensee provided a list of the components that were impacted by the proposed changes in Attachment 9. The NRC staff determined that performing an audit was necessary to verify key assumptions, analyses, and test reports used to support the basis for the LAR. The NRC staff selected a representative sample of electrical components in the licensee's EQ program to audit based on a risk-informed approach. The NRC staff selected the representative sample from a list of components affected by the proposed changes based on risk significance and importance to defense-in-depth. The NRC staff reviewed documents and confirmed that the selected components remain qualified as a result of the new environmental profiles, as documented in the audit report dated November 3, 2020 (Reference 17).

The licensee explained in Attachment 9 of its submittal dated August 28, 2019, that the cabling from instruments for the RPS and ES system are routed through the penetration rooms to the CSR. The licensee further stated that the electrical penetrations and the associated cabling for the instruments are qualified for the environmental conditions inside the penetration room, and these cables are not adversely impacted by any postulated HELBs in the penetration room. As documented in the audit report, the NRC staff reviewed the EQ of the cables and electrical penetration assemblies and confirmed that the components remain qualified as a result of the proposed changes (i.e., temperature and pressure changes).

The licensee explained in Attachment 9 that the Class 1E electrical system may be damaged by postulated HELBs inside the TB. The licensee further stated that the effect would be similar to an SBO, as such, SSD can be achieved through the PSW system and the SSF. UFSAR Table 8-4 addresses the loss of 4160 VAC main feeder buses, 4160 VAC stand-by power buses and feeder circuit breakers, and 4160 VAC auxiliary switchgear bus sections, and indicates that there would be no consequence since there are redundant systems. UFSAR Section 8.3.1.1.3 notes that upon loss of the normal source of power to the 4160 VAC main feeder buses, standby power is available from the Keowee hydroelectric units or supplied from the 100 kilovolt (kV) transmission line, hence, there are standby sources of power available upon loss of the normal source of power for the 4160 VAC system. UFSAR Section 8.3.2.2.4 indicates that the alternate AC power source, the SSF DG, would be available within 10 minutes. In UFSAR Section 8.3.1.1.1, the licensee stated that the SSF electrical power system supplies power necessary to maintain the reactor of each unit in an SSD condition in the event of loss of power from all other power systems. Thus, there would not be a loss of function for the electrical system, as there is redundancy, as well as backup and standby power sources available, including the SSF and the PSW system.

Based on its review of the information in the LAR and its confirmation by audit that the EQ of electrical equipment remains bounded, the NRC staff finds that the proposed changes will have no adverse impact on the licensee's EQ Program and, therefore, the licensee sufficiently addressed Item 13 of the Giambusso/Schwencer letter.

3.1.6 Inspection Program

Giambusso/Schwencer Item 16 specifies that a description be provided of the quality control and inspection program that will be required or has been utilized for piping systems outside containment.

To address this item, the licensee stated that it has instituted an inspection program that ensures that the girth and accessible attachment welds of the MS and MFDW piping in the AB are inspected, at least once, during each 10-year inservice inspection interval of ASME Code, Section XI. The licensee examines the girth welds for internal weld flaws and weld thickness. The attachment welds are inspected for surface indications. The licensee also stated that it has completed the initial inspections of the MS and MFDW girth and attachment welds located in the AB.

In addition, the licensee inspected the accessible terminal end welds inside the respective rupture restraint guard pipe on the 'A' and 'B' MFDW trains. The licensee has initiated a program to inspect these terminal end welds during each 10-year inservice inspection interval of ASME Code, Section XI. The licensee has included the inspections of the piping base metal downstream of the respective MFDW isolation valves within the weld inspection program or the station's flow accelerated corrosion inspection program.

With respect to postulated critical cracks, the licensee stated that it would implement an inspection program that ensures that critical cracks located at welds and in the base metal away from welds, for other HE lines located in the AB, receive an inspection, at least once, during each 10-year inservice inspection interval. The licensee has determined that no critical crack locations at welds or at base metal locations away from welds for other HE lines exist in the AB and, therefore, such an inspection program is not needed at this time.

The licensee stated that each MFDW guard pipe encloses the postulated MFDW break location(s). The licensee identified inaccessible girth welds on an elbow that are enclosed by the MFDW guard pipes adjacent to the RB penetrations #25 and #27. The guard pipes form part of the MFDW rupture restraints. The licensee also explained that the inaccessible girth welds are present in Units 1 and 2, but not in Unit 3. For Units 1 and 2, the licensee volumetrically inspected the accessible portion of the elbow girth weld in the MFDW A and B header(s) once during each 10-year inservice inspection interval. The Unit 3 headers contain no such elbows, and as such there are no girth welds enclosed by the MFDW rupture restraint guard pipe. The licensee further stated that because these elbow girth welds are adjacent to the postulated break location inside the guard pipe, a break at the inaccessible weld(s) would result in no greater consequences than those that would occur for break(s) previously postulated inside the guard pipe.

The NRC staff finds that the licensee has identified the relevant welds in the MS and MFDW line outside of containment in Units 1, 2, and 3 that are required to be inspected. The licensee provided the identification numbers of the welds, weld type, header (A or B), and inspection method (ultrasonic testing or penetrant testing). The NRC staff notes that the licensee has performed the initial inspections of these welds. Based on its review, the NRC staff determined that the licensee adequately described the programs that examine these welds and base metals in the future inservice inspection intervals, including the examinations specified in the weld and flow accelerated corrosion inspection programs. Based on the above, the NRC staff finds that the licensee sufficiently addressed Giambusso/Schwencer Item 16.

### 3.1.7 NRC Staff Conclusion Regarding HELB Mitigation Strategy

The NRC staff determined that the licensee identified appropriate systems to provide initial mitigation for postulated HELBs that are protected from the effects of the HELB by separation or designed to withstand the effects of the HELB. The NRC staff based this determination on the review of plant drawings, the audit of assessments and calculations, and the review of the LAR. The NRC staff finds that proposed alternate SSD capabilities (i.e., the SSF and PSW systems) have reliability and testability commensurate with the low likelihood of major TB HELBs that would necessitate their use. The repair of potential damage to the LPSW system and supporting electrical distribution to support placing the units in CSD is acceptable considering the low likelihood of major TB HELBs that would damage these systems and the continually decreasing secondary DHR water requirements. Furthermore, the NRC staff finds that existing equipment, the schedule for conforming modifications, and procedure changes to identify TCAs would provide acceptable and reasonably timely assurance of defense-in-depth for mitigation of postulated HELBs. Therefore, the NRC staff concludes that the proposed final HELB strategy is acceptable.

## 3.2 Thermal Hydraulic Analysis

The purpose of the RCS thermal hydraulic (T-H) analyses is to evaluate and confirm mitigation of HELBs and SSD of reactors following HELBs postulated to occur in the TB and the EPR of the AB. The systems used for this purpose are either the normal plant systems, SSF, or PSW system depending on their availability during the HELB scenario.

The licensee performed T-H analyses for the HELB scenarios that result in RCS overheating during the MFDW HELB scenarios, and RCS overcooling during the MS HELB scenarios. The analyses evaluate the response of the ONS unit experiencing a HELB to each of the scenarios in establishing the heat removal from the RCS by the SGs using the normal plant systems, SSF equipment, or the PSW system. The purpose of the T-H analyses is to demonstrate that for the scenario analyzed, the credited systems meet the proposed HELB mitigation acceptance criteria given in Section 3.2.1 below and the SSD of the reactor is achieved.

To confirm that the acceptance criteria is met, the licensee analyzed the HELB scenarios by using the following methodologies: (a) RELAP5/MOD2-B&W ONS model documented in NRC approved topical report (TR) DPC-NE-3003-PA (Reference 41), (b) RETRAN-3D ONS model documented in NRC approved TRs DPC-NE-3000-PA (Reference 42), DPC-NE-3003-PA (Reference 41), and DPC-NE-3005-PA (Reference 43), (c) Simulate-3K (S3K) and Simulate-3P (S3P) methodologies documented in NRC approved TRs DPC-NE-1006-PA (Reference 44), DPC-NE-3005-PA (Reference 43), and NFS-1001-A (Reference 45), and (d) Versatile Internals and Component Program for Reactors; Electric Power Research Institute (EPRI) (VIPRE) methodology documented in NRC approved TR DPC-NE-3000-PA (Reference 42).

3.2.1 HELB Mitigation Acceptance Criteria

HELB mitigation requires an SSD of the reactor while the reactor, RCS, and SGs remain within their design limits. The licensee specified the following acceptance criteria for the RCS overheating and RCS overcooling T-H analyses.

3.2.1.1 RCS Overheating Analysis Acceptance Criteria for MFDW HELBs

- The core must remain intact and in a coolable core geometry.
- Minimum departure from nucleate boiling ratio (DNBR) meets specified acceptable fuel design limits.
- RCS pressure must not exceed 2750 psig (110 percent of design pressure).

The NRC staff finds the above criteria acceptable because for overheating of the RCS, the core should remain intact and maintain coolable geometry, the minimum DNBR should meet the acceptable fuel design limits, and RCS pressure increase should not exceed design pressure.

3.2.1.2 RCS Overcooling Analysis Acceptance Criteria for MS HELBs

- The SG tubes remain intact.
- RCS remains within acceptable pressure and temperature limits.

The NRC staff finds the above criteria acceptable because for the overcooling of the RCS, the pressure/temperature of the RCS should remain within the specified limits and SG tube stresses should remain within the design limits to maintain the tubes intact.

3.2.2 T-H Analysis Methodology for MFDW and MS HELBs

The licensee used the RELAP5/MOD2-B&W methodology for the MFDW HELBs as well as some of the MS HELBs T-H analysis of the RCS. UFSAR Section 15.1.3 states that the RELAP5/MOD2-B&W code, developed by AREVA NP, is used for the best-estimate and licensing transient simulation of pressurized water reactors (PWRs). The original RELAP5/MOD2 was developed by the Idaho National Engineering Laboratory for the NRC. The NRC has approved the RELAP5/MOD2-B&W methodology in the Framatome NP (renamed as AREVA NP) TR BAW-10164P-A (Reference 46). Duke Energy modified the RELAP5/MOD2-B&W model in TR DPC-NE-3003-PA and denominated it as RELAP5/MOD2-B&W ONS. The licensee further enhanced the RELAP5/MOD2-B&W ONS by including the following features to make it suitable for the proposed HELB analyses:

(1) Included the following features:

- Ambient heat losses from the pressurizer after the time of peak RCS pressure.
- Improved its capability to model thermal stratification of fluid in the pressurizer region.

(2) In addition to (1) above, for the analysis of HELB in EPR, mitigation by the normal plant systems, added the following features:

- Portions of the condensate and MFDW system piping to represent the fluid volumes.
- Loop high point vents and RV head vent modeling.

(3) In addition to (1) above, for the analysis of HELB in TB, mitigation by the normal plant systems, added the following feature:

- Loop high point vents and RV head vent modeling.

(4) In addition to (1) above, for the analysis of HELB in TB, mitigation by the SSF, added the following features:

- Elimination of the MFDW piping to conservatively minimize liquid added to the SGs.
- Modeling of the SSF letdown line.

(5) In addition to (1) above, for the analysis of HELB, mitigation by the PSW system, added the following features:

- Elimination of the MFDW piping to conservatively minimize liquid added to the SGs.
- Loop high point vents and RV head vent modeling.

The following subsections describe the features added by the licensee in the RELAP5/MOD2-B&W ONS base model for a more realistic and/or conservative analysis.

3.2.2.1 Ambient Heat Losses from the Pressurizer

The licensee stated that in the RELAP5/MOD2-B&W ONS base model, the exterior heat structures on the RCS and pressurizer components are modeled **[[**
**]]** To model the ambient heat losses from the RCS, the associated heat structures are converted to **[[**

**]]** In its response to RAI-2 by letter dated June 15, 2020, the licensee evaluated the heat transfer coefficients by testing during steady state Mode 1 RCS conditions using measured values of heat input (from heater voltage, current, and power) for all operating pressurizer heaters according to plant procedures.

Within the RELAP5/MOD2-B&W ONS models used for the RCS overheating and overcooling analysis for a HELB scenario mitigated by normal plant equipment, the SSF, or the PSW system, the ambient heat losses are **[[**
**]]**

Consistent with the approach in DPC-NE-3000-PA (Reference 42), DPC-NE-3003-PA (Reference 41), and DPC-NE-3005-PA (Reference 43), the licensee did not model ambient heat losses from other RCS structures. The NRC staff finds this change in the RELAP5/MOD2-B&W ONS base model acceptable because the licensee enhanced the existing models by realistically modeling the pressurizer heat losses.

3.2.2.2 RV Head Axial Conduction

The RELAP5/MOD2-B&W ONS base model does not contain heat structures to represent the internal metal structures within the RV upper head nodes. In the nodalization scheme given in Figure 2.1-2 of DPC-NE-3003-PA, the RV upper head region is effectively a dead-ended volume. This is non-physical because the buoyancy effects would result in mixing of the RCS fluid in this region. In its response to RAI-3 by letter dated June 15, 2020, the licensee stated that in the RCS overcooling analysis, to eliminate the non-physical behavior attributed to nodalization, **[[**

**]]** This feature is not used in the overheating analysis.

The NRC staff finds this change in the RELAP5/MOD2-B&W ONS base model acceptable for the RCS overcooling analysis because it mitigates thermal stratification in a voided RV head, which can develop due to the absence of axial heat conduction in the base model. Not using this feature is conservative for the RCS overheating analysis.

3.2.2.3 Pressurizer Nodalization for Thermal Stratification

In the RELAP5/MOD2-B&W base model the pressurizer is modeled with large height nodes. To increase the resolution of axial temperature gradients, the licensee **[[**

**]]** as well as improved predictions of thermal stratification of the liquid region during in-surges, out-surges, and large pressure drops.

The licensee provided the following rationale for this change:

> The pressurizer plays a significant role in regulating RCS pressure during [overheating and overcooling] transients and experiences several important phenomena for both overcooling and overheating conditions.
>
> In general, for overheating scenarios, there is an initial insurge of subcooled liquid into the pressurizer from thermal expansion of the RCS inventory. If the overheating transient is short lived, the presence of subcooled liquid in the pressurizer has little impact on the immediate response. This is because there is little mixing in the fluid region under these conditions and buoyancy (density) effects cause the colder liquid in the pressurizer to remain near the bottom of the vessel, while the hotter (originally saturated) liquid remains near the top of the water column and in contact with the vapor space. Thermal stratification of the pressurizer liquid helps limit the amount of steam condensation that occurs at the steam-liquid interface during these pressure excursions.
>
> For RCS overcooling transients, saturated liquid in the pressurizer flashes to steam, expands, and limits the depressurization rate of the RCS. Subsequently when the pressurizer refills, in-surges of subcooled liquid to the pressurizer can limit the ability of the pressurizer to regulate subsequent depressurizations of the RCS. For more severe overcooling scenarios, the pressurizer may empty as a result of the initial overcooling, but subcooled liquid will refill the pressurizer once operators restore RCS pressure or pressurizer level to the specified operating range. In the long-term recovery phase, operator actions to stabilize pressurizer level and energize pressurizer heaters allows the fluid in the pressurizer to re-saturate and restore RCS pressure to a desired range.
>
> For scenarios mitigated with limited pressurizer heater capacity, the ability to re-saturate the subcooled liquid in the pressurizer is greatly diminished. Additionally, pressurizer ambient heat losses can cause condensation of the vapor space on internal structural surfaces. Continued condensation of the

> vapor space leads to a reduction in RCS pressure and increases in pressurizer level. As the vapor space collapses, the continual in-surge of subcooled liquid challenges the ability of the pressurizer heaters to re-saturate the fluid. Should the pressurizer eventually refill to a water-solid condition, RCS pressure control is provided by balancing makeup and letdown flow with either the SSF letdown line, pressurizer PORV or loop high point vents.

The NRC staff finds the licensee's rationale for **[[ ]]** in the RELAP5/MOD2-B&W ONS base model acceptable because it is more realistic modeling of the heat transfer within the pressurizer fluid and from the pressurizer to the ambient.

#### 3.2.2.4 Main Feedwater and Condensate System Nodalization

The RELAP5/MOD2-B&W ONS base model nodalization includes the MFDW piping between the last check valve and the SG. It is conservative to minimize the amount of FDW that can enter the SGs for the RCS overheating scenarios and maximize the amount of FDW that can enter the SGs for the RCS overcooling scenarios.

For the RCS overheating analysis of the HELB upstream of the last MFDW check valve, the licensee conservatively removed the MFDW piping included in the RELAP5/MOD2-B&W ONS model to minimize liquid added to the SGs.

For the EFW mitigated RCS overheating analysis of a break downstream of the last MFDW check valve in the EPR, the licensee added the portions of the MFDW piping required to appropriately model the flow and break boundary conditions.

For the RCS overcooling scenarios, the licensee assumed conservatively that the MFDW control valves open to allow the maximum amount of FDW to enter the SG.

The NRC staff finds the above changes in the RELAP5/MOD2-B&W ONS base model acceptable because they would result in conservative RCS overheating and overcooling analyses.

#### 3.2.2.5 Main Steam System Nodalization

The RELAP5/MOD2-B&W ONS base model represents the MS piping from the SG to the turbine with a single volume (node) for each loop. This nodalization is conservative for the RCS overheating analysis for the MFDW HELBs because the turbine stop valves are immediately closed upon break initiation and the turbine bypass valves are assumed to be unavailable and therefore a single node would maximize RCS overheating.

For RCS overcooling scenarios, the licensee described the change in the ONS base model and its rationale as follows:

> For overcooling scenarios, additional phenomena are present that potentially impact the ability to remove heat from the SGs. These phenomena are associated with the rapid depressurization due to postulated MS piping breaks. The rapid depressurization will initially cause a liquid level swell and entrainment due to high steam velocities. The SG outlet nozzles installed in the Replacement Once Through Steam Generators (OTSGs) serve to limit the blowdown mass

flow rate. Entrained liquid droplets in the steam flow may become de-entrained in the vertical portions of the steam line piping downstream of the SGs. Modeling the vertical piping enables a liquid level in this section of steam line that could impact conditions within the SG. Additional detail that preserves flow area and elevation change is included in the steam line nodalization used for the HELB overcooling analysis to allow the analysis to capture these effects.

The NRC staff finds the existing single volume nodalization in the RELAP5/MOD2-B&W ONS base model acceptable with no changes because it represents a conservative approach of heat transfer and steam release from the SGs for the RCS overheating scenarios.

For the RCS overcooling analysis, the NRC staff finds the change in the RELAP5/MOD2-B&W ONS base model acceptable because it captures the effect of the entrained liquid droplets in the steam flow that may become de-entrained in the vertical portions of the steam lines which increases heat transfer conservatively from the RCS to SGs and, therefore, results in an increased RCS overcooling.

3.2.2.6 Steam Generator Modeling

The RELAP5/MOD2-B&W ONS base model describes the SG modeling approach. The RELAP5/MOD2-B&W EFW heat transfer model, described in BAW-10164P-A (Reference 46), is used to model flow through the SG upper nozzles for the RCS overheating and overcooling analyses. The SG upper nozzles are used by a variety of flow sources including EFW, SSF ASW, and PSW system flow depending on the event scenarios. The licensee stated that the EFW model consists of **[[**

**]]** To use this model, the SG tubes are modeled with **[[**

**]]**

For the RCS overheating scenarios, the licensee stated that the limiting peak RCS pressure occurs prior to the SSF ASW or PSW system being aligned to the SGs for cooling and, therefore, the RELAP5/MOD2-B&W ONS base model modified **[[**
**]]** will not affect the peak RCS pressure. For the RCS overcooling scenarios, the modified model conservatively maximizes the heat transfer from the RCS in the SG.

The NRC staff finds the SG modeling approach in the RELAP5/MOD2-B&W ONS base model acceptable because it is conservative for the RCS overcooling analysis, and it does not affect the RCS peak pressure in the analysis of RCS overheating scenarios.

3.2.2.7 Boundary Condition Modeling

The RELAP5/MOD2-B&W ONS base model does not include specific boundary conditions for the HELB scenarios, therefore, the licensee added the following features in the RELAP5/MOD2-B&W ONS base model:

- SSF letdown line

- Loop high point vents and RV head vent
- Steam line ADVs
- SSF ASW
- PSW system
- Turbine-driven EFW
- Secondary steam loads

The modeling also includes the impact on the boundary conditions due to MS HELB which introduces asymmetry in the SG loops.

The NRC staff finds the added boundary conditions for the RELAP5/MOD2-B&W ONS base model acceptable because these modeling features are applied to ensure appropriate boundary conditions are specified for each specific analysis.

3.2.2.8 NRC Staff Evaluation

This RELAP5 methodology is suitable for the proposed T-H analyses because it provides accurate results for the two-phase conditions that potentially could exist in the RCS during the postulated HELB events. The MFDW HELB and some MS HELB scenarios can result in sustained two-phase conditions in the RCS. The NRC staff finds the use of RELAP5/MOD2-B&W ONS methodology for the proposed analyses of the HELB scenarios acceptable because of its ability to predict accurate results for the two-phase conditions in the RCS. The features described in Subsections 3.2.2.1 through 3.2.2.7 added in the RELAP5/MOD2-B&W ONS base model are acceptable.

3.2.3 T-H Analysis Methodologies for MS HELBs

The licensee used the following T-H analysis methodologies for the postulated MS HELB scenarios that result in the overcooling of the RCS:

- RETRAN-3D ONS
- RELAP5/MOD2-B&W ONS
- S3K
- S3P
- VIPRE

Section 15.1.3 of the UFSAR, states that the RETRAN-3D code was developed by Computer Simulation & Analysis, Inc. for EPRI to enhance and extend the simulation capabilities of the RETRAN-02 code. RETRAN-02 has the flexibility to model any general fluid system by partitioning the system into a one-dimensional network of fluid volumes and connecting junctions. Most of the capabilities of the RETRAN-02 code have been retained within RETRAN-3D as options, except for a limited number of models and correlations that were not in use.

The RETRAN-3D ONS is Duke Energy's modified version of the RETRAN-3D methodology described in TRs DPC-NE-3000-PA, DPC-NE-3003-PA, and DPC-NE-3005-PA. The RETRAN-3D ONS methodology is not suitable for analyzing MS HELB scenarios in which sustained two-phase conditions exist in the RCS as it does not predict accurate results for these conditions. For these MS HELB scenarios, the licensee used RELAP5/MOD2-B&W ONS methodology which has been accepted to predict accurate results for two-phase conditions in the RCS.

RETRAN-3D code has the following core neutronics model options: (a) point kinetics, (b) 1-D kinetics, and (c) 3-D kinetics. The licensee stated that using a point kinetics model option generally provides a more conservative core power response relative to the 3-D kinetics model option.

To ensure that an appropriate transient reactivity is calculated for the RCS overcooling analysis, the licensee also used S3P core model to determine conservative reactivity calculations (i.e., higher return to power or less subcritical) obtained by using RETRAN-3D ONS. The process used is described in the MSLB analysis methodology described in DPC-NE-3005-PA. The RETRAN-3D ONS point kinetics model option reactivity calculation provided conservative and bounding results relative to the results obtained using S3P.

The licensee used RELAP5/MOD2-B&W ONS and S3K for the reactivity calculation for MS HELB scenarios in which two-phase conditions occur in the RCS. The RELAP5/MOD2-B&W ONS calculation provided conservative and bounding results relative to the results obtained using S3K. The licensee selected S3K instead of S3P for comparing results with RELAP5/MOD2-B&W ONS calculation based on the anticipation of voiding at the limiting return to power state point. The S3K model can calculate voiding and its impact on reactivity and power distributions which are not included in the PWR version of S3P.

S3K and S3P are Studsvik Scandpower's best-estimate 3-dimensional core kinetics codes for neutronics and T-H analysis over a wide range of applications for PWRs. The licensee's analysis using S3K and S3P followed the guidance described in the methodology defined in DPC-NE-1006-PA (Reference 44), DPC-NE-3005-PA (Reference 43), and NFS-1001-A (Reference 45).

VIPRE is a T-H code to evaluate reactor core safety limits including minimum DNBR, critical power ratio, fuel and clad temperatures, and RCS state in normal operation and assumed accident conditions. The licensee's analysis using VIPRE followed the guidance described in the methodology in DPC-NE-3000-PA.

For the DNBR calculation during MS HELB scenarios, the licensee used methodology described in DPC-NE-3000-PA to demonstrate that a significant departure from nucleate boiling (DNB) margin exists for the state point at the peak heat flux.

For the MS HELB RCS overcooling analysis, the licensee added the following features in RETRAN-3D ONS base model by including boundary conditions for the HELB scenarios that are not described in DPC-NE-3000-PA or DPC-NE-3005-PA:

- additional break junctions
- turbine-driven EFW
- secondary steam loads
- additional steam line nodes to facilitate turbine control modeling
- normal letdown line

The above features are used for the analysis of MS HELB scenarios using normal plant, SSF, or PSW system equipment to ensure appropriate boundary conditions are specified for each analysis. The licensee also modeled the asymmetry of the SG loops occurring due to HELBs.

The limitations and conditions listed in the generic safety evaluation report (SER) for the use of RETRAN-3D have been evaluated in Appendix C of DPC-NE-3000-PA. This evaluation demonstrates that the RETRAN-3D ONS T-H model is within the RETRAN-3D SER conditions and limitations.

3.2.3.1 NRC Staff Evaluation

This RETRAN-3D ONS methodology is suitable for the proposed T-H analyses for some of the MS HELB events in which the potential of two-phase conditions would not exist in the RCS. The licensee added acceptable features to the RETRAN-3D ONS model by including the specific boundary conditions and the asymmetry in the SG loops due to HELB for the RCS overcooling analyses. The NRC staff finds the use of the RETRAN-3D point kinetics model option acceptable and more suitable relative to its 3-D kinetics model option because the former results in a conservative core power response.

3.2.4 T-H Analysis of MFDW HELB Scenarios

The postulated MFDW HELB outside the containment results in the TB flooding, creates overheating of the RCS, and makes it necessary to bring the units to an SSD condition. The licensee analyzed the following MFDW HELB scenarios:

- 4160 VAC Power Available, Mitigation by Normal Plant Equipment
  - MFDW HELB in the TB
  - MFDW HELB in the EPR, upstream of check valve
  - MFDW HELB in the EPR, downstream of check valve
- 4160 VAC Power Unavailable due to HELB, SSF Mitigation
- 4160 VAC Power Unavailable due to HELB, PSW Mitigation

The evaluation of the RCS T-H overheating analysis for each of these scenarios is given below.

3.2.4.1 MFDW HELB in the TB, Mitigation by Normal Plant Equipment

For the analysis of MFDW HELB scenario in the TB, the 4160 VAC ES switchgear is assumed to be available and, therefore, normal plant equipment is used for HELB mitigation and for bringing the plant to an SSD condition. The transient results in overheating of the RCS while the two motor-driven EFW pumps and all four RCPs remain operating. The RELAP5/MOD2-B&W ONS model is used for the analysis. The Initial conditions, assumptions, operator actions, and results of the analysis are summarized below.

Initial Conditions

- 4160 VAC power is available.
- Reactor thermal power: 102 percent of 2568 megawatts thermal (MWt) at hot full power (HFP) conditions.
- Complete loss of MFDW.
- Single active failure.

Assumptions

- Decay heat is according to American Nuclear Society (ANS)-79 (Reference 47) with a $+2\sigma$ uncertainty.
- The RPS trips the reactor following the loss of MFDW on high RCS pressure.
- SG pressure is unaffected due to the presence of MFDW check valves.
- The MS lines are assumed to remain intact with only the MSRVs lifting and controlling MS pressure to maximize the RCS heatup.
- Conservatively, two motor-driven EFW pumps and all four RCPs are operating for overheating of RCS.
- EFW system is available for SG heat removal.
- Conservatively, pressurizer sprays are not credited.
- The pressurizer safety valves (PSVs) are credited to maintain RCS pressure below its safety limit.

Operator Actions

No operator actions are credited prior to reaching the peak RCS pressure.

Results

- Integrity of the fuel is ensured by control rod insertion.
- DHR from both SGs using EFW system keeps the core intact and maintains a coolable core geometry.
- The RCS pressure and temperature increase because of insufficient heat removal by the SGs while the two EFW pumps and four RCPs are in operation. The increase in pressure dominates and, therefore, increases the RCS subcooling in the core to suppress any potential DNB. Increase in the RCS temperature results in a decrease in core power because of the negative moderator temperature coefficient. The minimum DNBR is not limiting as it is bounded by the uncontrolled bank withdrawal analysis state point included in the normal ONS reload analysis.
- The maximum RCS pressure is determined to be 2737.4 psia which is below its safety limit of 2750 psig (2764.7 psia) and occurs during the initial PSV cycle. The PSV (setpoint 2500 psig) maintains the RCS pressure below the safety limit. The pressurizer does not become water solid.

The NRC staff finds that the analysis is based on conservative initial conditions and valid assumptions and meets the acceptance criteria given in Section 3.2.1.1. Therefore, the NRC staff finds that the results of this analysis are acceptable.

3.2.4.2 MFDW HELB in the EPR, Upstream of Check Valve

For the analysis of the MFDW HELB scenario in the EPR upstream of the MFDW check valve, the 4160 VAC ES switchgear is assumed to be available and, therefore, normal plant equipment is used for HELB mitigation and for bringing the plant to an SSD condition. The transient results in an overheating of the RCS with two motor-driven EFW pumps and all four RCPs operating. The RELAP5/MOD2-B&W ONS model is used for the T-H analysis. Initial conditions, assumptions, operator actions, and the results of the analysis are given below.

Initial Conditions

- 4160 VAC power is available.
- Reactor thermal power: 102 percent of 2568 MWt at HFP conditions.
- Complete loss of MFDW.
- Single active failure.

Assumptions

- Decay heat is according to ANS-79 with a $+2\sigma$ uncertainty.
- The RPS trips the reactor following the loss of MFDW on high RCS pressure.
- SG pressure is unaffected because the break is upstream of the MFDW line check valves
- EFW system is available for SG heat removal.
- The HPI system is available for normal makeup and RCP seal cooling.
- The RPS, EFW, and HPI are sufficient to achieve and maintain reactor in SSD condition.
- The ADVs, in addition to HPI and EFW, are credited to cool the plant down to LPI entry conditions.

Operator Actions

No operator actions are credited prior to reaching the peak RCS pressure.

Results

- Reactor does not return to criticality.
- DNBR at reactor trip is bounded by the existing UFSAR Chapter 15 analyses.
- Peak RCS pressure remains below the 2750 psig limit.
- DHR and primary coolant makeup continues to keep the core covered and maintains a coolable core geometry.
- Reactor remains in Mode 3 for the duration of the scenario.
- Integrity of the fuel is ensured by the control rod insertion.

The NRC staff finds that the analysis is based on conservative initial conditions and valid assumptions and meets the acceptance criteria given in Section 3.2.1.1. Therefore, the NRC staff finds that the results of this analysis are acceptable.

3.2.4.3 MFDW HELB in the EPR, Downstream of Check Valve

For the analysis of the MFDW HELB scenario in the EPR downstream of the MFDW check valve, the 4160 VAC ES switchgear is assumed to be available and, therefore, normal plant equipment is used for HELB mitigation and for bringing the plant to an SSD condition. The transient results in an overheating of the RCS with two motor-driven EFW pumps and all four RCPs operating. The RELAP5/MOD2-B&W ONS model is used for the T-H analysis. Initial conditions, assumptions, operator actions, and the results of the analysis are summarized below.

Initial Conditions

- 4160 VAC power is available.
- Reactor thermal power: 102 percent of 2568 MWt at HFP conditions.

- Break area is limited to 0.54 $ft^2$ by a guard pipe.
- Single active failure.

Assumptions

- Decay heat is according to ANS-79 with a $+2\sigma$ uncertainty.
- RPS trips the reactor following the break on high reactor pressure.
- Limited break area allows a fraction of the initial FDW flow to reach the SGs.
- Reduced initial FDW flow to the SGs results in a rapid heatup of the RCS.
- The affected SG completely depressurizes following the reactor trip resulting in an AFIS actuation.
- On low steam line pressure AFIS will trip both MFDW pumps and trip or block the turbine-driven EFW pump.
- Conservatively, initially two motor-driven EFW pumps and all four RCPs are operating for overheating of RCS.
- Conservatively, pressurizer sprays are not credited.
- Normal RCS letdown is lost.
- PORV is not available for pressure control.

Operator Actions

- Reduce the number of operating RCPs to one pump operation in each loop at 15 minutes to limit the RCS pressure below the PSV lift setpoint.
- Open one loop high point vent (HPV) to control RCS pressure between 2150-2350 psig after HPVs become available at 30 minutes.

Results

- Reactor does not return to criticality.
- DNBR at reactor trip is bounded by the existing UFSAR Chapter 15 analyses.
- Peak RCS pressure remains below the 2750 psig limit.
- DHR and primary coolant makeup continues to keep the core covered and maintains a coolable core geometry.
- Reactor remains in Mode 3 for the duration of the scenario.
- Integrity of the fuel is ensured by the control rod insertion.

The NRC staff finds that the analysis is based on conservative initial conditions and valid assumptions and meets the acceptance criteria given in Section 3.2.1.1. Therefore, the NRC staff finds that the results of this analysis are acceptable.

3.2.4.4 MFDW HELB in the TB, SSF Mitigation

For the analysis of MFDW HELB scenario in the TB, the SSF ASW is used for providing an alternate means of establishing SG heat removal if the EFW system is unavailable. This analysis evaluates the RCS response to a rupture in the MFDW piping with a loss of the 4160 VAC ES switchgear due to a TB HELB. This break location is upstream of the MFDW line check valves such that a break in this location results in a complete loss of MFDW to both SGs. The RELAP5/MOD2-B&W ONS model is used for the T-H analysis. Initial conditions, assumptions, operator actions, and the results of the analysis are summarized below.

Initial Conditions

- Loss of 4160 VAC ES switchgear due to TB HELB.
- Reactor thermal power: 102 percent of 2568 MWt at HFP conditions.
- Rupture in the MFDW piping in the TB.
- Break location is upstream of the MFDW line check valves, resulting in a complete loss of MFDW to both SGs.

Assumptions

- Decay heat is according to ANS-79 with a $+2\sigma$ uncertainty.
- Conservatively assume reactor does not trip coincident with loss of power.
- Motor-driven EFW pumps are unavailable due to loss of 4160 VAC power.
- Turbine-driven EFW pumps are assumed unavailable.
- SG SSF ASW flow is controlled by procedure to maintain an RCS pressure at 2100 psig approximately.
- Portions of the integrated control system (ICS) are unprotected from HELB damage, therefore, the pressurizer PORV is assumed to be unavailable.
- RCS pressure increases to the PSV lift setting, and the PSVs cycle to control RCS pressure until operators establish SSF ASW flow.
- Pressurizer is predicted not to become water solid prior to SSF ASW aligned to SG.

Operator Actions

- Trip the operating RCPs either 2 minutes after a loss of indicated subcooled margin, or 3 minutes after loss of RCP seal cooling.
- Align SSF ASW to establish steam generator cooling at 14 minutes.
- Establish RCP seal cooling and RCS makeup from SSF RCMU at 20 minutes.
- Energize pressurizer heaters from the SSF standby power supply for RCS pressure control at 20 minutes.
- Throttle SSF ASW to maintain RCS pressure band until SG level develops, then maintain SG level band afterwards.

Results

- Reactor does not return to criticality.
- DNBR at reactor trip is bounded by the existing UFSAR Chapter 15 analyses.
- Peak RCS pressure remains below the 2750 psig limit and is determined by the PSV valve characteristics as the PORV is not available.
- DHR and sufficient primary coolant makeup keeps the core covered and maintains the RCS in Mode 3 for the duration of the scenario.
- Integrity of the fuel is ensured by the control rod insertion and the core remains covered.
- Pressurizer does not become water solid before SSF ASW flow is initiated to the SGs at 14 minutes, therefore, liquid relief does not take place through the PSVs.

The NRC staff finds that the analysis is based on conservative initial conditions and valid assumptions and meets the acceptance criteria given in Section 3.2.1.1. Therefore, the NRC staff finds that the results of this analysis are acceptable.

3.2.4.5 MFDW HELB in the TB, PSW Mitigation

For the analysis of the MFDW HELB scenario in the TB, the PSW system is used as an alternate means of establishing SG heat removal when both the EFW system and SSF ASW are unavailable. The analysis evaluates the RCS response to a loss of MFDW with a loss of the 4160 VAC ES switchgear due to a TB HELB. The break location is upstream of the MFDW line check valves resulting in a complete loss of MFDW to both SGs. The RELAP5/MOD2-B&W ONS model is used for the T-H analysis. Initial conditions, assumptions, operator actions, and the results of the analysis are given below.

Initial Conditions

- Loss of 4160 VAC ES switchgear due to TB HELB.
- Reactor thermal power: 102 percent of 2568 MWt at HFP conditions.
- Rupture in the MFDW piping in TB, resulting in a complete loss of MFDW to both SGs.

Assumptions

- Decay heat is according to ANS-79 with a $+2\sigma$ uncertainty.
- Motor-driven EFW pumps are unavailable due to loss of power.
- Turbine-driven EFW pumps are assumed unavailable.
- SG PSW flow is controlled by procedure to maintain an RCS pressure at about 2100 psig.
- Portions of the ICS are unprotected from HELB damage, therefore, the pressurizer PORV is assumed to be unavailable.
- RCS pressure increases to the PSV lift setting, and the PSVs cycle to control RCS pressure until operators establish PSW flow.

Operator Actions

- Trip the operating RCPs either at 2 minutes after a loss of indicated subcooled margin, or 3 minutes after loss of RCP seal cooling.
- Align PSW to establish SG cooling at 14 minutes (from loss of MFDW) from the CR.
- Establish RCP seal cooling and RCS makeup from one HPI pump at 20 minutes.
- Energize pressurizer heaters from the PSW switchgear for RCS pressure control at 2 hours.
- Throttle PSW to maintain an RCS pressure band until SG level develops, then maintain SG level band afterwards.

Results

- Reactor does not return to criticality.
- DNBR at reactor trip is bounded by the existing UFSAR Chapter 15 analyses.
- Peak RCS pressure remains below the 2750 psig limit and is determined by the PSV valve characteristics as the PORV is not available.
- DHR and sufficient primary coolant makeup keeps the core covered and maintains the RCS in Mode 3 for the duration of the scenario.
- Integrity of the fuel is ensured by the control rod insertion.
- Pressurizer is predicted not to become water solid prior to PSW flow is initiated to the SGs, therefore, liquid relief is not predicted through the PSVs.

The NRC staff finds that the analysis is based on conservative initial conditions and valid assumptions and meets the acceptance criteria given in Section 3.2.1.1. Therefore, the NRC staff finds that the results of this analysis are acceptable.

### 3.2.5 T-H Analysis for MS HELB Scenarios

The postulated MS HELB outside the containment results in overcooling of the RCS and makes it necessary to bring the units to an SSD condition. The licensee analyzed the following MS HELB scenarios:

- 4160 VAC Power Available, Mitigation by Normal Plant Equipment
  - Single MS HELB in the TB, 4160 VAC Power Available
  - Double MS HELB in the TB, 4160 VAC Power Available
  - Double MS HELB, LOOP, 4160 VAC Power Available
- 4160 VAC Power Unavailable, SSF Mitigation
  - Single MS HELB in the TB
  - Double MS HELB in the TB
- 4160 VAC Power Unavailable, PSW Mitigation
  - Single MS HELB in the TB
  - Double MS HELB in the TB

The evaluation of the RCS overcooling T-H analysis for each of these scenarios is given below.

#### 3.2.5.1 Single MS HELB in the TB, 4160 VAC Power Available

The analysis assumes a double-ended break of a single MS line in the TB mitigated by normal plant equipment as the normal 4160 VAC power remains available. In its response to RAI-10 by letter dated June 15, 2020, the licensee stated that single MS HELB is not specifically analyzed for this LAR because the existing MSLB analyses described in UFSAR Sections 15.13 and 15.17 similarly assume that 4160 VAC power is not affected by the breaks. The UFSAR Chapter 15 analyses were performed using the RETRAN-3D methodology.

In Attachment 6 to the submittal dated August 28, 2019, Section 4.1.1, the licensee stated, in part:

> These analyses assume an initial core power level of 102 percent of 2568 MW and HFP conditions. RCS integrity is demonstrated by determining the limiting SG tube compressive and tensile stresses remain within design limits, and that the RCS pressure and temperature remains within the acceptable cooldown limits during the transient evolution. The maximum tensile stress resulting from a single MSLB is significantly less than the limiting tensile stress that results from a large break LOCA [loss-of-coolant accident]. With 4160 VAC power available, normal plant equipment is able to maintain the plant within limits during the cooldown.

In its response to RAI-8 and by letter dated June 15, 2020, the licensee stated, in part:

> The initial conditions for the UFSAR Chapter 15.13 … and UFSAR Chapter 15.17 analyses performed to evaluate the limiting minimum DNBR are defined by the statistical core design (SCD) methodology. This method specifies that

parameters important to the DNBR are set to nominal values and incorporates the associated uncertainty in the DNBR evaluation.

The analysis results in a limiting tube-to-shell ΔT of -233.3°F. Therefore, the SG tube stresses which are based on tube-to-shell ΔT of -233.3°F are bounded by the SG tube stresses based on ΔT of -374°F induced during a large break LOCA and are, therefore, below their design limits.

The NRC staff finds it acceptable that the SG tube stresses for single MS HELB are bounded by the SG tube stresses induced during a large break LOCA and, therefore, are below their design limits. The NRC staff also finds it acceptable that the minimum DNBR meets the specified acceptable fuel design limits because the limiting minimum DNBR from the UFSAR Chapter 15 analyses are included in the normal reload core design process. Based on the above and the MS HELB mitigation acceptance criteria in Section 3.2.1.2 being met, the NRC staff finds that the single MS HELB in the TB can be mitigated using the normal plant equipment.

3.2.5.2 Double MS HELB in the TB, 4160 VAC and 6900 VAC Power Available

The analysis assumes a double-ended guillotine break in both MS lines in the TB mitigated by normal plant equipment as the normal 4160 VAC power remains available. The licensee analyzed this scenario using the RETRAN-3D ONS methodology. The licensee stated that the boundary conditions for this scenario are identical to the UFSAR Section 15.13.3, MSLB analysis for the case with offsite power available and with a double-ended guillotine break of one of the two main steam lines inside the containment. The main difference is that both SGs experience an uncontrolled depressurization in the double MS HELB analysis.

<u>Initial Conditions</u>

- 4160 VAC and 6900 VAC power is available.
- Reactor thermal power: 102 percent of 2568 MWt at HFP conditions.
- Single active failure.

<u>Assumptions:</u>

- Decay heat is according to ANS-79 with a +2$\sigma$ uncertainty.
- Conservatively assume immediate reactor trip to maximize overcooling effect.
- Conservatively, no credit is taken for automatic isolation of MFDW or EFW during this scenario. The condensate and FDW systems are assumed to continue feeding the SGs until the condensate inventory is depleted.
- The turbine-driven EFW pump is assumed to operate to ensure a conservative overcooling is obtained.

In its response to RAI-13 and by letter dated June 15, 2020, the licensee stated that the time frame during which the DNBR is of concern is prior to the completion of control rod insertion. The scenarios in which RCPs are assumed to lose power and coast down, bound the scenarios in which they remain in operation. The licensee stated:

> For scenarios where offsite power remains available and the RCPs remain operating, the mass flow entering the core increases as the core inlet temperature decreases. The potential decrease in core exit pressure is limited by the liquid saturation temperature. The limiting DNBR during this time frame

occurs when the core inlet mass flow decreases [due to higher temperature (lower density) core inlet flow in the RCPs trip and coast down scenario] while core discharge pressure decreases, before the heat flux from the fuel has time to decay.

The limiting case that challenges a return to criticality for the double MS HELB with the 4160 VAC equipment available is with the RCPs running.

Operator Actions

- Trip the operating RCPs at 2 minutes after a loss of indicated subcooled margin.
- Isolate EFW flow to the affected steam generator at 10 minutes.

Results

- Reactor does not return to criticality.
- RCS pressure and temperature remains within the acceptable cooldown limits during the transient. RCS subcooling as a function of time indicates that the subcooled margin remains less than 200°F for the duration of the limiting case, which is less than the 250°F subcooled screening criteria.
- The analysis results in a limiting tube-to-shell ΔT of -269.8°F. Therefore, the SG tube stresses which are based on tube-to-shell ΔT of -269.8°F are bounded by the SG tube stresses based on ΔT of -374°F induced during a large break LOCA and are, therefore, below their design limits.

The NRC staff finds it acceptable that by using NRC accepted methodology, conservative initial conditions, assumptions, and accepted operator actions, the licensee demonstrated that: (a) RCS integrity is maintained because the SG tube compressive and tensile stresses are within their design limits, and (b) minimum DNBR meets the acceptable fuel design limits. Based on the above and the MS HELB mitigation acceptance criteria in Section 3.2.1.2 being met, the NRC staff finds that the double MS HELB in the TB can be mitigated using the normal plant equipment.

3.2.5.3 Double MS HELB in the TB, LOOP, 4160 VAC Power Available

The analysis assumes a double-ended guillotine break of both MS lines in the TB mitigated by normal plant equipment as the emergency 4160 VAC power remains available. In this HELB scenario, LOOP is assumed to coincide with the double MS HELBs and 4160 VAC power is available. As stated in UFSAR Section 15.13.4, LOOP trips the reactor, causes the RCPs to coast down, and flow is lost while depressurization is occurring. The minimum DNBR state point occurs within the first few seconds of the RCP coastdown; therefore, the duration of the DNBR analysis is 5 seconds. The licensee analyzed this scenario using RETRAN-3D ONS methodology. The licensee stated that this scenario is similar to the description provided in UFSAR Section 15.13.4 for MSLB analysis with respect to the initial and boundary conditions for the case without offsite power available. The primary difference is both SGs experience an uncontrolled depressurization in the MS HELB analysis.

Initial Conditions

- 4160 VAC power is available.

- Reactor thermal power: 102 percent of 2568 MWt at HFP conditions.
- Single active failure.

Assumptions

- Decay heat is according to ANS-79 with a $+2\sigma$ uncertainty.
- LOOP is coincident with the double MS HELB in the TB.
- A 5-second transient analysis immediately after reactor trip is sufficient to determine the minimum DNBR because this case is considered to be bounded by the double MS HELB case evaluated in Section 3.2.5.2 above for the effects on core reactivity and the potential of return to criticality.

Operator Actions

- Trip the operating RCPs 2 minutes after a loss of indicated subcooled margin.
- Isolate EFW flow to the affected steam generator at 10 minutes.

Results

- The reactor does not return to criticality.
- Transient results for RCS subcooling show subcooled margin less than 200°F for the duration of the limiting case which is less than the subcooled screening criteria of 250°F.
- The minimum DNBR is within acceptable limits.
- RCS pressure and temperature remain within the acceptable cooldown limits during the transient.
- The analysis results in a limiting tube-to-shell $\Delta T$ of -269.8°F. Therefore, the SG tube tensile stresses which are based on tube-to-shell $\Delta T$ of -269.8°F are bounded by the SG tube tensile stresses, based on $\Delta T$ of -374°F, induced during a large break LOCA and are, therefore, below their design limits.

The NRC staff finds it acceptable that by using NRC accepted methodology, conservative initial conditions, assumptions, and accepted operator actions, the licensee demonstrated that: (a) RCS integrity is maintained because the SG tube compressive and tensile stresses are within their design limits, and (b) minimum DNBR meets the acceptable fuel design limits. Based on the above and the MS HELB mitigation acceptance criteria in Section 3.2.1.2 being met, the NRC staff finds that the double MS HELB in the TB in the presence of LOOP can be mitigated using the normal plant equipment.

3.2.5.4 Single MS HELB in the TB, 4160 VAC Power Unavailable, SSF Mitigation

In the analysis of single MS HELB in the TB with a double-ended guillotine break scenario, the 4160 VAC power is assumed to be unavailable. The licensee used RELAP5/MOD2-B&W ONS methodology for the core power response, and VIPRE methodology for the DNBR analysis.

Initial Conditions

- 4160 VAC power is unavailable.
- Reactor thermal power: 102 percent of 2568 MWt at HFP conditions.

Assumptions

- Decay heat is according to ANS-79 with a $+2\sigma$ uncertainty.
- No credit for AFIS.
- Immediate reactor and turbine trip to obtain the limiting core response.
- Trip of all condensate and MFDW pumps.
- The motor-driven EFW pumps are not available due to the loss of 4160 VAC power.
- To conservatively maximize overcooling, the turbine-driven EFW pump is assumed to automatically start and run without being throttled until the contents of the UST are delivered to the SGs.

Sensitivity Case

The licensee also analyzed a sensitivity case that does not credit boron added by the SSF RCMU pump.

Operator Actions

The following operator actions are credited in the analysis:

- RCPs secured at 3 minutes after loss of RCP seal cooling or 2 minutes after a loss of indicated subcooling.
- SSF RCMU pump is credited to restore RCP seal cooling at 20 minutes.
- Isolation of normal letdown is credited at 20 minutes.
- The SSF powered pressurizer heaters are available at 20 minutes but are not energized until pressurizer level recovers sufficiently **[[ ]]**
- **[[ ]]**
- Initiate SSF operation as described in Section 3.6 of the LAR.

Results

- The core remains covered, intact, and subcooled during the return to power with adequate DNB margin.
- Minimum DNBR result is acceptable.
- Large DNB margin exists for the state point at the peak heat flux during the return to critical portion of the transient.
- RELAP5/MOD2-B&W ONS core power response calculation is conservative (i.e., higher return to power) compared to S3K calculation.
- The SG tubes remain intact.
- RCS remains within acceptable pressure and temperature limits.
- Minimum cold leg temperatures do not approach the nil ductility temperature (NDT) curves.
- The maximum SG tube compressive stresses are based on tube-to-shell ΔT of 245.5°F and 58.3°F for loops A and B respectively which are well below the compressive stress limit based on ΔT of 343°F.
- The maximum SG tube tensile stresses are based tube-to-shell ΔT of -169.4°F and -40.0°F for loops A and B respectively which are well below the tensile stress limit based on ΔT of -374°F.
- The results of the sensitivity case described above showed that maximum core power level reached to 2.4 percent power at 1501 seconds. The core exit subcooling between 1200 and

1800 seconds is greater than 120 °F, and consistently greater than 60°F subcooled during the return to power.

The NRC staff finds it acceptable that by using NRC accepted methodology, conservative initial conditions, assumptions, and accepted operator actions, the licensee demonstrated that: (a) RCS integrity is maintained because the SG tube compressive and tensile stresses are within their design limits, and (b) minimum DNBR meets the acceptable fuel design limits. Based on the above and the MS HELB mitigation acceptance criteria in Section 3.2.1.2 being met, the NRC staff finds that the RCS overcooling due to single MS HELB in the TB can be mitigated using the SSF equipment.

3.2.5.5 Double MS HELB in the TB, 4160 VAC Power Unavailable, SSF Mitigation

In the analysis of the double MS HELB in the TB with double-ended guillotine break in both lines, the 4160 VAC power is assumed to be unavailable. The licensee used RELAP5/MOD2-B&W ONS methodology for the core power response, and VIPRE methodology for the DNBR analysis.

<u>Initial Conditions</u>

- 4160 VAC power is unavailable.
- Reactor thermal power: 102 percent of 2568 MWt at HFP conditions.

<u>Assumptions</u>

- Decay heat is according to ANS-79 with a $+2\sigma$ uncertainty.
- No credit for AFIS.
- Immediate reactor and turbine trip gives the limiting core response.
- Trip of all condensate and MFDW pumps.
- The motor-driven EFW pumps are not available due to the loss of 4160 VAC power.
- To conservatively maximize overcooling, the turbine-driven EFW pump is assumed to automatically start and run without being throttled until the contents of the UST are delivered to the SGs.

<u>Operator Actions</u>

The following operator actions are credited in the analysis:

- RCPs secured at 3 minutes after loss of RCP seal cooling or 2 minutes after a loss of indicated subcooling.
- SSF RCMU pump is credited to restore RCP seal cooling at 20 minutes.
- Isolation of normal letdown is credited at 20 minutes.
- The SSF powered pressurizer heaters are available at 20 minutes but are not energized until pressurizer level recovers sufficiently **[[ ]]**
- Operators initially control SSF ASW flow to stabilize pressurizer level, which prevents the pressurizer from developing into a water solid condition. The licensee stated that the operator guidance assumed in the analysis is to stabilize the RCS temperature between 325°F to 350°F and its pressure between 650 psig to 700 psig. The licensee stated the following advantages in maintaining these conditions:

- The RCS would be in natural circulation with a subcooled margin consistent with the normal natural circulation guidance (150°F indicated subcooling).
- SSF ASW would be controlled to maintain a constant cold leg temperature with pressurizer heaters and SSF letdown available to control RCS pressure.
- Below 350°F, RCP seal integrity would not be readily challenged should seal injection flow be interrupted.
- Remaining above 600 psi allows time to isolate the core flood tanks (CFTs) and prevent nitrogen injection.
- Should the pressurizer become water solid at these conditions, there is a significant amount of margin to lifting the pressurizer code safety valves (2500 psig setpoint).
- The compressive tube stress analytical limit is defined by the RCS at 550°F and the SG shell at 212°F. The cooldown to below 350°F will provide margin to prevent tube deformation.
- During the cooldown, sufficient boron is added to ensure that the core remains subcritical down to 200°F without credit for Xenon.

- **[[ ]]**

<u>Results</u>

- A short return to criticality occurs after 1500 seconds which results in a fission power increase to 1 watt approximately.
- The core remains covered, intact, and subcooled during the return to power with adequate DNB margin.
- Minimum DNBR result is acceptable.
- RELAP5/MOD2-B&W ONS core power response calculation is conservative (i.e., higher return to power) compared to S3K calculation.
- The SG tubes remain intact.
- RCS remains within acceptable pressure and temperature limits with significant margin.
- Minimum cold leg temperatures do not approach the NDT curves.
- The maximum SG tube compressive stresses are based on tube-to-shell ΔT of 105.1 °F and 157.9°F for loops A and B respectively. The stress values are well below the compressive stress limit based on ΔT of 343°F.
- The maximum SG tube tensile stresses are based on tube-to-shell ΔT of -151.3°F and -142.2°F for loops A and B respectively. These values are well below the tensile stress limit based on ΔT of -374°F.

The NRC staff finds it acceptable that by using NRC accepted methodology, conservative initial conditions, assumptions, and accepted operator actions, the licensee demonstrated that: (a) RCS integrity is maintained because the SG tube compressive and tensile stresses are within their design limits, and (b) minimum DNBR meets the acceptable fuel design limits. Based on the above and the MS HELB mitigation acceptance criteria in Section 3.2.1.2 being met, the NRC staff finds that the double MS HELB in the TB can be mitigated using the SSF equipment.

3.2.5.6 Single MS HELB in the TB, 4160 VAC Power Unavailable, PSW Mitigation

In the analysis of the single MS HELB in the TB with a double-ended guillotine break, the 4160 VAC power is assumed to be unavailable. The licensee used RELAP5/MOD2-B&W ONS and

S3K methodologies for the core power response, and VIPRE methodology for the DNBR analysis.

Initial Conditions

- 4160 VAC power is unavailable.
- Reactor thermal power: 102 percent of 2568 MWt at HFP conditions.

Assumptions

- Decay heat is according to ANS-79 with a $+2\sigma$ uncertainty.
- PSW provides the heat sink.
- No credit for AFIS.
- Immediate reactor and turbine trip for the limiting core response.
- Trip of all condensate and MFDW pumps.
- The motor-driven EFW pumps are not available due to the loss of 4160 VAC power.
- To conservatively maximize overcooling, the turbine-driven EFW pump is assumed to automatically start and run without being throttled until the contents of the UST are delivered to the SGs.
- The PSW powered HPI pump provides RCS makeup flow and restores RCP seal injection flow from the BWST.

Operator Actions

The following operator actions are credited in the analysis:

- RCPs secured at 3 minutes after loss of RCP seal cooling.
- Isolate the normal letdown flowpath at 20 minutes and start a PSW powered HPI pump to restore RCP seal cooling at 20 minutes.
- PSW pump flow is available at 14 minutes, but not aligned until RCS temperatures recover to control core exit thermocouples (CETCs) to the desired setpoint of 350°F.
- The PSW powered pressurizer heaters are available at 20 minutes but are not energized until pressurizer level recovers.
- The PSW letdown path from the RCS is through the reactor vessel head vent (RVHV) and loop HPV.
- The RVHV is opened **[[ ]]**
- **[[ ]]** and is subsequently cycled to maintain a 20-inch pressurizer level band.

Results

- The core remains covered, cooled, and subcritical after the control rods insertion.
- RCS remains within acceptable pressure and temperature limits.
- Minimum DNBR result is acceptable.
- RELAP5/MOD2-B&W ONS core power response calculation is conservative (i.e., higher return to power) compared to S3K calculation.
- Minimum cold leg temperatures do not approach the NDT curves.

- The maximum SG tube compressive stresses are based on tube-to-shell ΔT of 54.7°F and 58.8°F for loops A and B respectively. These values are well below the compressive stress limit based on ΔT of 343°F.
- The maximum SG tube tensile stresses are based on tube-to-shell ΔT of -152.0°F and -67.8°F for loops A and B respectively. These values are well below the tensile stress limit based on ΔT of -374°F.

The NRC staff finds it acceptable that by using NRC accepted methodology, conservative initial conditions, assumptions, and accepted operator actions, the licensee demonstrated that: (a) RCS integrity is maintained because the SG tube compressive and tensile stresses are within their design limits, and (b) minimum DNBR meets the acceptable fuel design limits. Based on the above and the MS HELB mitigation acceptance criteria in Section 3.2.1.2 being met, the NRC staff finds that the single MS HELB in the TB can be mitigated using the PSW system.

3.2.5.7 Double MS HELB in the TB, 4160 VAC Power Unavailable, PSW Mitigation

In the analysis of the double MS HELB in the TB with double-ended guillotine break in both lines, the 4160 VAC power is assumed to be unavailable. The licensee used RELAP5/MOD2-B&W ONS methodology for the core power response, and VIPRE methodology for the DNBR analysis.

<u>Initial Conditions</u>

- 4160 VAC power is unavailable.
- Reactor thermal power: 102 percent of 2568 MWt at HFP conditions.

<u>Assumptions</u>

- Decay heat is according to ANS-79 with a $+2\sigma$ uncertainty.
- PSW provides the heat sink.
- No credit for AFIS.
- Immediate reactor and turbine trip for the limiting response.
- Trip of all condensate and MFDW pumps.
- The motor-driven EFW pumps are not available due to the loss of 4160 VAC power.
- To conservatively maximize overcooling, the turbine-driven EFW pump is assumed to automatically start and run without being throttled until the contents of the UST are delivered to the SGs.
- The PSW powered HPI pump provides RCS makeup flow and restore RCP seal injection flow from the BWST.
- The pressurizer heaters are used to maintain 150°F core exit temperature (CET) subcooling.

<u>Operator Actions</u>

The following operator actions are credited in the analysis:

- RCPs secured at 3 minutes after loss of RCP seal cooling.
- Isolate the normal letdown flowpath at 20 minutes and start a PSW powered HPI pump to restore RCP seal cooling at 20 minutes.
- PSW pump flow is available at 14 minutes, but not aligned until RCS temperatures recover to control CETCs to the desired setpoint of 350°F.

- **[[ ]]** and is subsequently cycled to maintain a 20-inch pressurizer level band.
- Operators initially control PSW flow to stabilize CET. The licensee stated that the operator guidance assumed in the analysis is to stabilize the RCS temperature between 325°F to 350°F and its pressure between 650 psig to 700 psig. The licensee stated the following advantages in maintaining these conditions:

    - The RCS would be in natural circulation with a subcooled margin consistent with the normal natural circulation guidance (150°F indicated subcooling).
    - PSW would be controlled to maintain a constant cold leg temperature with pressurizer heaters, and RCS head and loop vent valves available to control RCS pressure.
    - Below 350°F, RCP seal integrity would not be readily challenged should seal injection flow be interrupted.
    - Remaining above 600 psi allows time to isolate the CFTs and prevent nitrogen injection.
    - Should the pressurizer become water solid at these conditions, there is a significant amount of margin to lifting the pressurizer code safety valves (2500 psig setpoint).
    - The compressive tube stress analytical limit is defined by the RCS at 550°F and the SG shell at 212°F. The cooldown to below 350°F will provide margin to prevent tube deformation.
    - During the cooldown, sufficient boron is added to ensure that the core remains subcritical down to 200°F without credit for Xenon.

- The PSW powered pressurizer heaters are available at 20 minutes but are not energized until pressurizer level recovers sufficiently **[[ ]]**
- The RVHV is opened **[[ ]]**

<u>Results</u>

- The core remains covered, cooled, and subcritical after the control rods insertion.
- RCS remains within acceptable pressure and temperature limits.
- Minimum DNBR result is acceptable.
- RELAP5/MOD2-B&W ONS core power response calculation is conservative (i.e., higher return to power) compared to S3K calculation.
- Minimum cold leg temperatures do not approach the NDT curves.
- The maximum SG tube compressive stresses are based on tube-to-shell ΔT of 81.4°F and 96.7°F for loops A and B respectively. The stress values are well below the compressive stress limit based on ΔT of 343°F.
- The maximum SG tube tensile stresses are based on tube-to-shell ΔT of -198.0°F and -149.7°F for loops A and B respectively. The stress values are well below the tensile stress limit based on ΔT of -374°F.

The NRC staff finds it acceptable that by using NRC accepted methodology, conservative initial conditions, assumptions, and accepted operator actions, the licensee demonstrated that: (a) RCS integrity is maintained because the SG tube compressive and tensile stresses are within their design limits, and (b) minimum DNBR meets the acceptable fuel design limits. Based on the above and the MS HELB mitigation acceptance criteria in Section 3.2.1.2 being met, the NRC staff finds that the double MS HELB in the TB can be mitigated using the PSW system.

## 3.3 Human Factors

NUREG-1764 contains a risk-informed process for scaling the review of operator manual actions using quantitative or qualitative risk information for the human actions (HA). The licensee's LAR was not submitted as a risk-informed LAR; therefore, following the guidance in NUREG-1764, the NRC staff utilized a combination of the "Estimated Importance Method" and the "Generic HA Method" in Section 2.4.3 to identify the appropriate amount of review for the operator actions described in the LAR. A Level II or Level III review for the proposed TCAs was determined to be appropriate for all TCAs. The NRC staff notes that the level of detail provided by the licensee was generally consistent with a Level II review. This is consistent with NUREG-1764, which states, "Licensees may choose to use the Level II guidance to address HFE [human factors engineering] considerations for HAs that fall into Level III." Therefore, using the information provided by the licensee, the NRC staff performed a Level II review of the new TCAs below.

### 3.3.1 Evaluation of Operator Actions

The licensee provided two attachments to the LAR that discuss TCAs. In its submittal dated August 28, 2019, Attachment 11, "Time Critical Operator Actions," contains both existing and new TCAs and Attachment 12, "Feasibility Assessment for New Proposed Time Critical Operator Actions," discusses the new TCAs from Attachment 11.

The existing TCAs in Attachment 11 have been reviewed and approved previously by the NRC, and documented in OSS 0254.00-00-4005, "Design Basis Specification for the DBEs," therefore, the NRC staff reviewed the six new TCAs.

#### 3.3.1.1 HELB Initiated Flooding Within the Turbine Building

The licensee postulated an MFDW HELB that, with a loss of AC power to all three units and in conjunction with failures to the CCW piping, would result in the TB flooding. This would create an overheating condition for the RCS and affect activities necessary for SSD. Mitigation of these HELBs is divided into four distinct phases. The first phase, Phase 1, "Reactor Shutdown," is reactor shutdown and the stabilization of the affected unit(s) in Mode 3 with reactor coolant average temperature ≥ 525°F.

The licensee proposed two new TCAs in Phase 1. The second new TCA is discussed in Section 3.3.2 of this SE. Phase 1 has two pathways and the emergency procedures (EPs) direct operators to initiate both pathways in parallel. Pathway 1 states the actions taken for SSD using the PSW system. Pathway 2 states the actions taken for SSD using the SSF. In the event of an MFDW HELB in the TB, the licensee proposed first, a new TCA for operators to trip all four CCW pumps on all three units within 45 minutes.

<u>NRC Staff Evaluation</u>

The licensee proposed a new TCA in Phase 1 to prevent the maximum flood height from exceeding the limit. The licensee stated that there is a similar existing TCA to control a design basis TB flood within 20 minutes by tripping the CCW pumps and closing the CCW pump discharge valves from the CR. The guidance for this existing action can be found in the TB flood abnormal procedures (APs) for each ONS unit and operators receive periodic classroom and simulator training on this scenario. Operators will enter the AP when the turbine basement water emergency high level annunciator is received. This annunciator is located in the Unit 1 CR. Unit 1 and Unit 2 share a combined CR, therefore, the operators should immediately recognize that the entry conditions for the TB flood AP are being met. For Unit 3 operators, the turbine basement water emergency high level computer alarm will be the entry condition. Additionally, the Unit 2 TB flood AP directs the Unit 2 CR operator to notify the Unit 1 and Unit 3 CR operators to enter the TB flood AP. The location of the four CCW pump control switches used by the operators are on an auxiliary control board located immediately behind the main control board and within the main CR envelope (e.g., no intervening panels or walls that would impede access to the control switches). The sump switches are also clearly labeled. During the most recent re-validation of the current TCA, the CCW pumps were tripped in 3 minutes. During implementation, the licensee stated that it will validate the proposed TCA, add guidance for operators to secure the CCW pumps to the TB HELB mitigation procedure, and train licensed operators via classroom and simulator training.

The NRC staff finds that the proposed TCA to trip all CCW pumps on all three ONS units within 45 minutes of an MFDW HELB initiated flooding within the TB to be acceptable because it is similar to an existing TCA where operators must trip the CCW pumps. In addition, recent testing of the operators for the existing TCA provided adequate results. The licensee will validate the proposed TCA, update the procedure, and conduct training during the implementation process thereby providing additional assurance that operators have the knowledge, skills, and abilities to successfully complete this action. Based on the above, the NRC staff finds this TCA acceptable.

3.3.1.2 Loss of Main Feedwater and Emergency Feedwater

Certain postulated MFDW HELBs in the TB would result in a reactor trip subsequent to the loss of MFDW to both SGs. Because AC power is also lost during this event, EFW will be lost resulting in a loss of secondary side DHR. In order to restore secondary side DHR and RCP seal cooling to establish an SSD condition, the licensee has proposed another new TCA in Phase 1, Pathway 1, “SSD Using PSW Systems,” for operators to establish PSW flow to the SGs within 14 minutes. Moreover, the EPs will direct operators to use the SSF (Pathway 2) in parallel.

NRC Staff Evaluation

The licensee proposed the new TCA to replace the current TCA that requires operators to locally start the turbine-driven EFW pump and locally cross-connect EFW from an unaffected unit. The licensee stated that the new action is identical to an existing TCA—operators must establish PSW flow to the SGs following a loss of secondary side DHR. The guidance for the existing action is in the emergency operating procedures (EOPs) of each ONS unit and the operators are periodically evaluated on their ability to successfully accomplish this action during simulator evaluations. During the most recent re-validation of the current TCA, the licensee stated the PSW flow was established to the SGs in 10 minutes. The licensee will validate the new TCA in accordance with Operation’s EOP/AP validation procedure to ensure that it can be consistently accomplished with margin. During implementation, the licensee stated that it will train the licensed operators on the revised procedural guidance via classroom and simulator training.

The NRC staff finds the proposed TCA to establish secondary side DHR with the PSW system within 14 minutes of loss of MFDW and EFW to be acceptable because it is essentially the same as an existing TCA in the ONS EOPs. The licensee described recent testing of the operators for the existing TCA with adequate results. During implementation, the licensee will validate the proposed TCA and conduct training on the revised procedural guidance thereby providing additional assurance that operators have the knowledge, skills, and abilities needed to successfully complete this action. Based on the above, the NRC staff finds this TCA acceptable.

3.3.1.3 Loss of Main Feedwater

The licensee postulated an MFDW HELB occurring downstream of the check valve in the EPR of the AB. This will cause the following events to occur: the RPS will trip due to high RCS pressure; the affected SG will completely depressurize resulting in an AFIS actuation; AFIS actuation trips the MFDW pumps and isolates MFDW and EFW to the affected SG; and the motor-driven EFW pump aligned to the intact SG will auto start on the loss of both MFDW pumps. The transient rapidly evolves into an overheating scenario because one motor-driven EFW pump is supplying the unaffected SG and all 4 RCPs are operating.

The licensee stated that the action for operators to secure one RCP per SG to limit heat input into the RCS is in the ONS EOPs. However, the licensee proposed to turn this action into a TCA that must be completed within 15 minutes.

<u>NRC Staff Evaluation</u>

The licensee stated that for the existing operator action, the operators currently receive periodic classroom and simulator training on the mitigation of overheating scenarios and are evaluated on their ability to complete this action. The location of the RCP control switches used by the operators are on the auxiliary control board immediately adjacent to the front control board and are clearly labeled. During implementation, the licensee will validate the new TCA in accordance with Operation's EOP/AP validation procedure to ensure that the TCA can be accomplished consistently with margin and to train the licensed operators on the revised procedural guidance via classroom and simulator training.

The NRC staff finds the proposed TCA to secure one RCP per SG within 15 minutes of a loss of MFDW to be acceptable because it is the same action that operators are currently trained on, however, it will now be credited. During implementation, the licensee will validate the proposed TCA and conduct training on the revised procedural guidance thereby providing additional assurance that operators have the knowledge, skills, and abilities to successfully complete this action. Based on the above, the NRC staff finds this TCA acceptable.

3.3.1.4 Maintaining RCS Pressure during Loss of Main Feedwater

For the overheating scenario discussed in Section 3.1.3 of this SE, an MFDW HELB occurring downstream of the check valve in the EPR of the AB, it is assumed that the PORVs are unavailable to provide RCS pressure control.

The licensee stated that the current ONS EOPs have an existing procedural action for operators to reduce RCP heat to the RCS to limit heat input during mitigation of an overheating scenario. However, the licensee proposed a new TCA to utilize the RCS HPV valves. Specifically, one set of RCS HPV valves is cycled as required to maintain an alternate RCS letdown flow path to be completed within 30 minutes.

Once the affected unit has restored a steam bubble in the pressurizer and RC letdown has been restored, the RCS HPV valves are closed, and the unit is cooled down to Mode 5 using normal plant systems and procedures.

NRC Staff Evaluation

The licensee stated that there is guidance in the EOPs for each ONS unit that has the same action for operators to open one set of RCS HPV valves as an alternate RCS letdown path. This action is taken for PSW mitigated overheating events; however, it is not in the EOPs for mitigating overheating events with normal plant systems. For overheating scenarios with PSW, the licensee stated that the operators receive periodic classroom and simulator training specifically on operating the RCS HPV valves, and operators are evaluated on their ability to mitigate overheating scenarios with PSW using the RCS HPV valves in simulator evaluations. The location of the control switches for the two sets of valves (one switch per valve) used by the operator are on the front control board and clearly labeled; each RCS hot leg (A and B) is equipped with two in-series solenoid operated HPV valves and both valves must be opened to establish flow. During implementation, the licensee will validate the new TCA in accordance with Operation's EOP/AP validation procedure to ensure that the TCA can be consistently accomplished with margin and to train the licensed operators on the revised procedural guidance via classroom and simulator training.

The NRC staff finds the proposed TCA to open the RCS high point vents within 30 minutes of a loss of MFDW to maintain RCS pressure below the pressurizer code safety valve lift setpoint to be acceptable because it is essentially the same action taken during the mitigation of an overheating scenario using PSW. During implementation, the licensee will validate the proposed TCA and conduct training on the revised procedural guidance thereby providing additional assurance that operators have the knowledge, skills, and abilities to successfully complete this action. Based on the above, the NRC staff finds this TCA is acceptable.

3.3.1.5 Isolate a Letdown Line Break HELB Within 20 Minutes

The licensee postulated an HELB at the letdown line containment penetration in the EPR upstream of the outside containment isolation valve. This will cause the following events to occur: a reactor trip will be initiated on either the low RCS pressure or on a variable low pressure trip function; continued RCS depressurization will fall to the ES actuation point; and the ES system actuation will isolate the break by closing valves HP-3 (A letdown cooler outlet and containment isolation valve) and HP-4 (B letdown cooler outlet and containment isolation valve). If a single active failure of either HP-3 or HP-4 occurs (failure to close), the procedural guidance will direct the operators to close HP-1 (A letdown cooler inlet isolation valve) or HP-2 (B letdown cooler inlet isolation valve) to isolate the break. However, the break does not interact with any other SSD equipment and the HPI system has adequate capacity to compensate for the leak rate as RCS and pressurizer level recover, RCS remains subcooled, and the RCPs remain in operation.

The licensee has identified the following TCAs that are needed to isolate the primary leakage outside of the containment building so that it limits the radiological effluent release:

1. Isolate the letdown line break within 20 minutes following ES actuation. This is a new TCA.
2. Start the control room ventilation system booster fans within 30 minutes of ES actuation. This is an existing TCA.

NRC Staff Evaluation

The licensee stated that there is existing guidance in the ES actuation verification enclosure of all the ONS EOPs for the action to isolate the letdown line break with HP-1 or HP-2, however, it is not a TCA. The location of the control switches for HP-1, 2, 3, and 4 used by the operators to complete this action are on the front control board adjacent to the HPI pump control switches. During implementation, the licensee will validate the new TCA in accordance with Operation's EOP/AP validation procedure to ensure that the TCA can be consistently accomplished with margin and to train the licensed operators on the revised procedural guidance. Specifically, the existing procedure steps in the EOP stated above will be rearranged, as necessary.

The NRC staff finds the proposed TCA to isolate a letdown line break HELB within 20 minutes following failure of ES actuated valve to close to be acceptable because it is essentially the same as an existing action where operators must isolate the letdown line break with HP-1 or HP-2, and action found in the ONS EOP's ES actuation verification enclosure. During implementation, the licensee will validate the proposed TCA and conduct training on the revised procedural guidance thereby providing additional assurance that operators have the knowledge, skills, and abilities to successfully complete this action. Based on the above, the NRC staff finds this TCA acceptable.

3.3.1.6 Isolate High Pressure Injection Pump Discharge Piping Break

The licensee postulated a terminal end break at the discharge nozzle of the 1, 2, 3A or 1, 2, 3B HPI pump which could result in flooding the HPI pump room if it is not isolated. The HPI pump provides RCS makeup and RCP seal cooling. The licensee stated that following an HELB at the discharge nozzle of the operating HPI pump, the immediate response is the loss of discharge flow and the auto-start of the standby HPI pump on low RCP seal flow. Upon start of the second HPI pump, flow will be restored to the RCS and the RCPs. The CR operator will be alerted to this event because they will receive the HPI pump discharge pressure low annunciator and the RCP seal header flow low annunciator. Once the break location has been identified, the affected HPI pump will be tripped by the CR operator. At the same time, a non-licensed operator (NLO) will locally isolate the leak by closing the remote-operated manual suction valve on the affected HPI pump. This is a new TCA for the NLO that must be completed within 39 minutes.

NRC Staff Evaluation

The licensee stated that the proposed action has a similar existing TCA. The existing action requires the operator to terminate flooding in the AB due to a pipe break on a raw service water system within 45 minutes. During the most recent re-validation of the current TCA, the break was isolated in 33 minutes. The guidance for the existing TCA can be found in the unit specific RCS leakage AP which will: (1) have the operators trend AB waste tank levels and containment sump levels on the operator aid computer to assist in the identification of leaks in the AB or inside containment; (2) direct the operators to perform a loss of HPI makeup and RCP seal injection after the leak in the HPI system is identified; (3) isolate the HPI system leak.

Like the existing TCA, the operators will enter the unit specific RCS leakage AP and loss of HPI makeup and RCP seal injection following an HELB at the discharge nozzle of an operating HPI pump scenario. However, specific guidance to isolate an HELB on the discharge nozzle of an

HPI pump is not in the loss of HPI makeup and RCP seal injection procedure. The licensee plans to update this during implementation of the proposed amendments.

With respect to the new TCA to isolate the break, the NLO must get to the valve handwheels (HP-103, HP-107, and HP-993), which are located on the deck immediately above the HPI pump room. Therefore, entering the HPI pump room is not required. The deck, which is in an easily accessible area, and the HPI pump room are located on the 771' elevation of the AB. The CR is located on the 822' elevation of the AB. There is a stairwell near each ONS CR access door for the NLO to get to that area. The licensee stated that the stairwells, corridors, and HPI pump room hatch are well-lit and the valves are clearly labeled. The remote-operated manual valves are operated by a handwheel connected to a reach rod, and there are no special tools or equipment required for the NLO to operate this valve.

The licensee stated that communication between the CR and the NLO is not required. However, there is a wall mounted telephone available that is located in the primary side sample hood area, immediately adjacent to the HPI pump room hatch area. The NLOs are also given portable radios.

There is a staffing requirement of a minimum of 9 NLOs to be on-site for all three ONS units for Modes 1-4. The operations management procedures require one qualified NLO responsible for implementing AP and EOP actions to be assigned to each ONS unit to ensure that requisite staff is always available. The licensed operators and NLOs will be trained for their respective actions for the postulated event via classroom and simulator training. The NLOs will also have this action added to their TCA qualification card. To ensure that the TCA can be consistently accomplished within margin, the licensee will validate this action in accordance with Operation's EOP/AP validation procedure.

The NRC staff finds that the proposed TCA to isolate HPI pump discharge piping break by closing HPI pump suction valve within 39 minutes is acceptable because it is similar to an existing TCA where operators will isolate an HPI break in order to terminate flooding in the AB due to a pipe break on a raw service water system. The recent testing of the operators for the existing TCA provided adequate results. Validation of the proposed TCA, training for both operators, and updating the procedure will be conducted during implementation providing additional assurance that operators have the knowledge, skills, and abilities to successfully complete this action. The licensee also considered the physical impediments for the NLO to complete the proposed action and the potential issues for communication and staffing. Based on the above, the NRC staff finds this TCA acceptable.

#### 3.3.1.7 NRC Staff Conclusion

The NRC staff reviewed the proposed TCAs and finds them to be acceptable because the actions were either identical or similar to existing actions that have already been validated. The licensee stated for the new TCAs that the guidance in the procedures will be updated and both licensed operators and NLOs will be trained appropriately. During implementation, the licensee will validate the proposed TCAs thereby providing additional assurance that operators will have the knowledge, skills, and abilities to successfully complete their actions. There is reasonable assurance that the operators can successfully accomplish the actions for their respective TCA within the time available, and the action is not expected to create any undue burden for the operators. Based on the above, the NRC staff finds the proposed TCAs acceptable.

3.3.2 Procedure Development

The licensee provided a discussion on procedures and the verification and validation (V&V) process in Section 3.6.7, "Procedures and Verification," of the Enclosure to the LAR. The licensee stated that all changes to the ONS EOPs and APs undergo the V&V process governed by operations administrative procedures and that they are validated using a table-top setting, in the field, and/or on the training simulator, including the SSF simulator for the SSF procedure. The licensee stated that the V&V process will ensure that the procedures used to mitigate and correct abnormal and emergency conditions meet criteria including written correctness, accurate technical content, usability, and operational correctness. The licensee stated that validation will provide assurance that the procedures contain sufficient and understandable operator information and are compatible with plant response, equipment accessibility, plant hardware, shift manpower, and ensures that the TCAs can be completed within the required time.

The proposed TCAs have been reviewed by (1) the senior reactor operator (SRO) responsible for managing the operations TCA program and (2) the previously licensed SRO responsible for maintaining the APs and EOPs. The qualitative assessment included the following: a review of the TCA to the applicable operating procedure; the time available before the action is required; the time required to complete the action; the required staffing; the complexity/feasibility of the action; the plant condition at the time that the action is required; and the adequacy of existing operator skill and knowledge to perform the TCA.

Following approval of a TCA, it will be managed in accordance with an administrative procedure that provides guidance on how to identify TCAs and control these actions to ensure that the required times can be met. TCAs without excess margin are re-validated approximately every five years to verify the ability to accomplish the actions with margin.

The NRC staff finds that the proposed changes to the ONS procedures are acceptable because there is a V&V process for the procedures for all ONS EOPs and Aps. All the proposed changes will be validated because they are EPs. The licensee has adequately considered the need for validating procedures and there is an administrative control in place following the approval of a TCA. There is reasonable assurance that the proposed guidance in the procedures will be sufficient to allow the operators to successfully complete their respective new TCAs. Based on the above, the NRC staff finds the proposed changes acceptable.

3.3.3 Training Program Development

The licensee provided a discussion of operator training in Section 3.6.6, "Training," of the Enclosure to the LAR. The licensee stated that during the initial licensed operator training program, operators receive classroom, simulator (including SSF simulator), and on-the-job training for the EPs and APs. The licensee requires the operators to maintain their proficiency with these procedures and their skill in placing the plant in an SSD condition using the simulator through the continuing training program. The NLOs also receive training on their EP and AP related tasks during their initial and continuing training program. Both operators and NLOs may be evaluated on SSF time critical tasks using job performance measures during their annual operating examination, which is part of the operator requalification program.

For implementation of this LAR, the licensee stated that both licensed operators and NLOs will receive classroom and simulator training for the proposed actions and training on the revised procedural guidance.

The NRC staff finds that the ONS training program is acceptable to support the LAR because the licensee will provide an appropriate level of training through the classroom and simulator environment for both licensed operators and NLOs. There is reasonable assurance that the operators will be appropriately trained to have the knowledge, skills, and abilities to successfully complete their respective new TCAs. Based on the above, the NRC staff finds the proposed changes acceptable.

## 4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments on December 12, 2020. On December 12, 2020, the State official confirmed that the State of South Carolina had no comments.

## 5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration published in the *Federal Register* on September 8, 2020 (85 FR 55514) and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs be prepared in connection with the issuance of the amendments.

## 6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date of Issuance: March 15, 2021

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 - ISSUANCE OF AMENDMENT NOS. 421, 423, AND 422 RE: REVISION OF LICENSING BASIS FOR HIGH ENERGY LINE BREAKS OUTSIDE OF CONTAINMENT (EPID L-2019-LLA-0184) DATED MARCH 15, 2021

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