



**Scott L. Batson**  
Vice President  
Oconee Nuclear Station

**Duke Energy**  
ON01VP | 7800 Rochester Hwy  
Seneca, SC 29672

o: 864.873.3274  
f: 864.873.4208

Scott.Batson@duke-energy.com

ONS-2014-080

10 CFR 50.90

June 30, 2014

Attn: Document Control Desk  
US Nuclear Regulatory Commission  
11555 Rockville Pike  
Rockville, MD 20582-2746

Duke Energy Carolinas, LLC (Duke Energy)

Oconee Nuclear Station (ONS), Units 1, 2 and 3  
Docket Number 50-269, 50-270, and 50-287  
Renewed License Nos. DPR-38, DPR-47, and DPR-55

**Subject: License Amendment Request (LAR) to Reduce Allowed Maximum Rated Thermal Power When High Pressure Injection (HPI) Equipment Is Inoperable  
License Amendment Request No. 2013-03**

In accordance with the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), Duke Energy is submitting a request for an amendment to the Technical Specifications (TS) for ONS, Units 1, 2, and 3. The proposed amendment would revise TS 3.5.2 by reducing the allowed maximum rated thermal power (RTP) at which the unit can operate when select HPI System equipment is inoperable. These revisions were deemed necessary to correct a non-conservative Technical Specification. In accordance with the guidance of U.S. Nuclear Regulatory Commission (NRC) Administrative Letter 98-10, this condition is captured in the ONS Corrective Action Program and proper administrative controls have been established until the TS is revised.

The enclosure to this letter provides an evaluation of the proposed TS change. A regulatory evaluation (including the significant hazards consideration) and environmental considerations are provided in Sections 5 and 6 of the enclosure, respectively. Attachments 1 and 2 provide marked-up TS and TS Bases pages, respectively. Attachments 3 and 4 provide retyped (clean) TS and TS Bases pages, respectively. Attachment 5 contains a listing of acronyms used within the enclosure.

In accordance with Duke Energy administrative procedures that implement the Quality Assurance Program Topical Report, these proposed changes have been reviewed and approved by the Plant Operations Review Committee. A copy of this LAR is being sent to the State of South Carolina in accordance with 10 CFR 50.91 requirements.

A 001  
NRK


U. S. Nuclear Regulatory Commission  
June 30, 2014  
Page 2

Duke Energy requests approval of the proposed LAR by June 30, 2015, effective immediately upon issuance with implementation within 120 days. Duke Energy will also update applicable sections of the ONS Updated Final Safety Analysis Report (UFSAR), as necessary, and submit these per 10 CFR 50.71(e). There are no new regulatory commitments being made as a result of the proposed change.

Inquiries on this proposed amendment request should be directed to Sandra Severance, ONS Regulatory Affairs, at (864) 873-3466.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 30th day of June, 2014.

Sincerely,



Scott L. Batson  
Vice President  
Oconee Nuclear Station

Enclosure: Evaluation of Proposed Changes

Attachments:

- 1 Marked-Up Technical Specifications Pages
- 2 Marked-Up Technical Specification Bases Pages
- 3 Retyped Technical Specifications Pages
- 4 Retyped Technical Specification Bases Pages
- 5 Acronym List

U. S. Nuclear Regulatory Commission  
June 30, 2014  
Page 3

cc w/enclosure and attachments:

Mr. Victor McCree  
Administrator Region II  
U.S. Nuclear Regulatory Commission  
Marquis One Tower  
245 Peachtree Center Ave., NE, Suite 1200  
Atlanta, Georgia 30303-1257

Mr. James R. Hall  
Senior Project Manager  
(by electronic mail only)  
Office of Nuclear Reactor Regulation  
U.S. Nuclear Regulatory Commission  
11555 Rockville Pike  
Mail Stop O-8G9A  
Rockville, Maryland 20852

Mr. Eddy Crowe  
NRC Senior Resident Inspector  
Oconee Nuclear Station

Susan E. Jenkins, Manager, Infectious and Radioactive Waste Management  
Bureau of Land and Waste Management  
Department of Health & Environmental Control  
2600 Bull Street  
Columbia, SC 29201

**ENCLOSURE**

**EVALUATION OF PROPOSED CHANGES**

**LICENSE AMENDMENT REQUEST NO. 2013-03**

Subject: Proposed License Amendment Request to Reduce the Allowed Maximum Rated Thermal Power that the Unit Can Operate at When Select High Pressure Injection Equipment is Inoperable

- 1 SUMMARY DESCRIPTION
- 2 BACKGROUND
- 3 DETAILED DESCRIPTION OF PROPOSED CHANGES
- 4 TECHNICAL EVALUATION
  - 4.1 Partial-Power SBLOCA Analysis for 10 CFR 50.46 Criteria
  - 4.2 Evaluation of Partial-Power SBLOCA Results If ADV Modulation Does Not Occur
  - 4.3 Operator Actions
- 5 REGULATORY EVALUATION
  - 5.1 Significant Hazards Determination
  - 5.2 Applicable Regulatory Requirements/Criteria
  - 5.3 Precedent
  - 5.4 Conclusions
- 6 ENVIRONMENTAL CONSIDERATION
- 7 REFERENCES

## 1 SUMMARY DESCRIPTION

Oconee Nuclear Station (ONS) Technical Specification (TS) 3.5.2, "High Pressure Injection (HPI)," provides requirements for the HPI System when in Modes 1 and 2, and in Mode 3 with Reactor Coolant System (RCS) temperature > 350°F. TS 3.5.2 Action Condition B is entered when an HPI pump, or an HPI discharge crossover valve, is inoperable for longer than 72 hours. TS 3.5.2 Action Condition C is entered when an HPI train is inoperable. For both Action Conditions, currently specified Required Actions are to reduce power to  $\leq 75\%$  of rated thermal power (RTP), and to verify by administrative means that the atmospheric dump valve (ADV) flow path for each steam generator (SG) is operable.

A reanalysis of the small break loss-of-coolant accident (SBLOCA) has been performed for TS 3.5.2, and the maximum power is reduced to 50% RTP to provide margin in the results and to address previously identified deficiencies in the 75% power SBLOCA analysis. In accordance with the guidance of U.S. Nuclear Regulatory Commission (NRC) Administrative Letter 98-10, this condition is captured in the ONS Corrective Action Program, and proper administrative controls have been established until the TS is revised. Duke Energy Carolinas, LLC (Duke Energy) proposes to change the allowed core power level specified in Action Conditions B and C from " $\leq 75\%$  RTP" to " $\leq 50\%$  RTP," in order to remove the non-conservatism from TS 3.5.2. Associated changes to the Bases for TS 3.5.2 and for TS 3.7.4, "Atmospheric Dump Valve (ADV) Flow Paths," as a result of revised ONS SBLOCA analysis, are also provided for information only. These changes accommodate a desired operator action to modulate steam generator pressure at 300 psig; however, from a licensing basis perspective, there is no impact to required operator actions as this modulation is not required to demonstrate post-LOCA long-term core cooling. The operator actions required by the ONS licensing basis remain unchanged.

A detailed description of the proposed changes is provided in Section 3. An evaluation of the proposed changes is provided in Section 4. The marked-up TS and TS Bases pages associated with this LAR are provided in Attachments 1 and 2, respectively, and the retyped (clean) TS and TS Bases pages are provided in Attachments 3 and 4, respectively. Acronyms used within this enclosure are listed in Attachment 5.

Duke Energy requests approval of the proposed LAR by June 30, 2015, effective immediately upon issuance with implementation within 120 days. Duke Energy will also update applicable sections of the ONS Updated Final Safety Analysis Report (UFSAR), as necessary, and submit these per 10 CFR 50.71(e). There are no new regulatory commitments being made as a result of this proposed change.

## 2 BACKGROUND

The ONS HPI System was originally designed with an installed spare pump, and the original TS required that only two HPI pumps be operable so that one pump could be out of service for maintenance. In 1978, Babcock and Wilcox (B&W), the nuclear steam system supplier (NSSS) for ONS, determined that the limiting RCS break location was on the discharge side of the reactor coolant pumps (RCPs). Thus, in order to ensure adequate HPI flow, the TS was revised to require the third HPI pump to be operable (Ref. 1). One HPI pump could be

removed from service for maintenance. If not returned to service within 72 hours, reactor power was to be reduced to < 60% of full power. In 2000, to correct for non-conservative analysis assumptions, the TS was again modified. If the third HPI pump is not returned to service within 72 hours, TS 3.5.2 now requires the unit to reduce power to  $\leq 75\%$  of RTP and that an ADV flow path for each SG be verified operable (Ref. 2 and 3). These requirements are based on the current analysis of record for a SBLOCA initiated at 75% of RTP with one HPI pump out of service, hereafter referred to as 75% partial-power SBLOCA analysis. This analysis is currently described in ONS Updated Final Safety Analysis Report (UFSAR) Section 15.14.4.2, and concludes that one HPI train has sufficient capacity to mitigate SBLOCAs when the initial reactor power is at or below 75% RTP (RTP for each ONS unit is 2568 MWt) and one HPI pump is out of service.

In 2010, two ONS Problem Investigation Program (PIP) Reports were initiated identifying issues with the ONS SBLOCA analysis, and ultimately, TS 3.5.2. A review of ONS Emergency Operating Procedure (EOP) changes identified an issue where the potential exists for core flood tank (CFT) nitrogen gas to enter the RCS during RCS depressurization and affect primary-to-secondary heat transfer in the SG tubes during specific SBLOCAs with inadequate HPI. A review of the issue identified that a new partial-power SBLOCA analysis was required to allow ONS EOPs to include operator actions to isolate the CFTs. Isolating the CFT has the potential to negatively impact the SBLOCA analysis due to the reduction in injection volume of borated water.

The second PIP was initiated after AREVA discovered that the standard axial power shape utilized in the SBLOCA evaluation model (Ref. 4) may not be bounding for the entire fuel cycle, particularly near the end-of-cycle (EOC). The standard axial power shape used in the SBLOCA evaluation model is peaked at the 9.5 foot core elevation. Since the SBLOCA analyses are designed to be independent of time in cycle, a bounding peak should be used to be consistent with the requirements identified in Section I.A of 10 CFR 50 Appendix K. Therefore, a bounding EOC axial power shape peaked at a core elevation of 11 feet was developed by AREVA for use in all SBLOCA analyses.

As a result of these issues identified in two Oconee PIP reports, Duke Energy concluded that the existing 75% partial-power SBLOCA analysis supporting event mitigation with one HPI pump was non-conservative. The HPI System for each ONS unit is currently considered Operable but Degraded / Nonconforming (OBDN) as a result of these issues. This condition was described in a Duke Energy letter to the NRC dated August 19, 2010 (Ref. 5).

### 3 DETAILED DESCRIPTION OF PROPOSED CHANGES

Duke Energy proposes to modify the TS and TS Bases (for information only). The proposed changes to ONS TS 3.5.2 will align the allowed initial core power level to the new SBLOCA analysis where event mitigation is accomplished with only one HPI pump. The proposed changes are listed below and identified in Attachment 1:

#### TS 3.5.2 Action Condition B

In Required Action B.1, change “≤ 75% RTP” to “≤ 50% RTP.”

#### TS 3.5.2 Action Condition C

- In Required Action C.1, change “≤ 75% RTP” to “≤ 50% RTP.”
- In the NOTE for Required Action C.2, change “≤ 75% RTP” to “≤ 50% RTP.”

#### TS Bases B 3.5.2 Actions B.1, B.2, B.3, and B.4

Change “≤ 75% RTP” to “≤ 50% RTP” in the four locations where “≤ 75% RTP” is specified.

#### TS Bases B 3.5.2 Actions C.1, C.2, and C.3

Change “≤ 75% RTP” to “≤ 50% RTP” in the 12 locations where “≤ 75% RTP” is specified.

#### TS Bases B 3.7.4 Applicable Safety Analysis

- Change “≤ 75% RTP” to “≤ 50% RTP” in the three locations where “≤ 75% RTP” is specified.
- Add the following text directly after the last sentence in this section:

“The 50% partial-power SBLOCA analysis includes a sensitivity case that models an operator action to modulate the main steam pressure at 300 psig via the ADV during the secondary-side depressurization. The purpose of the ADV modulation to maintain steam pressure is to limit Reactor Coolant System (RCS) depressurization, which then prevents the CFTs from completely discharging their liquid contents and introducing nitrogen gas into the RCS during the depressurization. The secondary-side pressure control to preclude significant nitrogen injection is consistent with the generic Emergency Operating Procedure (EOP) guidance for B&W plants provided by AREVA.

To ensure that the new SBLOCA analysis is bounding, the plant must be controlled to a main steam pressure that is less than the value assumed in the 50% partial-power SBLOCA analysis, since less borated water from the CFT would be injected at the higher analyzed value. This ensures that the 50% partial-power SBLOCA analysis remains conservative with respect to actual plant operation. The 50% partial-power SBLOCA analysis modeling the modulation of steam pressure at 300 psig allows operating space within the EOPs such that CFT isolation does not conflict with the applicable safety analysis in terms of isolating the borated water source from the CFTs.

A supplemental SBLOCA analysis demonstrates that long-term core cooling is assured with or without nitrogen gas intrusion for all break sizes. The operator actions required by

the ONS licensing basis remain unchanged. The analyses show that nitrogen gas intrusion does not occur for the small break sizes that rely on steam generator heat removal for a number of hours. In the longer term, core cooling is still assured if the CFTs completely discharge their liquid contents much later because at these longer times following the reactor trip, the lower decay heat levels can be matched by HPI cooling.

Based on the evaluation of impacts to long-term core cooling if ADV modulation does not occur, the operator action modeled in the partial-power SBLOCA analysis to maintain steam generator pressure at 300 psig is considered to be a desired action, and not a required action needed to demonstrate post-LOCA long-term core cooling.”

TS Bases B 3.7.4 LCO

- Change “≤ 75% RTP” to “≤ 50% RTP” in the one location where “≤ 75% RTP” is specified.



## 4 TECHNICAL EVALUATION

### 4.1 Partial-Power SBLOCA Analysis for 10 CFR 50.46 Criteria

AREVA has performed a new partial-power SBLOCA analysis for ONS to support the plant configuration where only one HPI pump is available for event mitigation. The new partial-power SBLOCA analysis supports 24-month operating cycles of full-core of Mk-B-HTP fuel, and includes explicit modeling of the 11-foot (10.811 foot actual) peaked axial power shape to address the SBLOCA evaluation model error (described in Ref. 5). This new partial-power SBLOCA analysis assumes an initial core power level of 52% of 2568 MWt (nominal 50% of 2568 MWt plus 2% calorimetric uncertainty as required per 10 CFR 50 Appendix K), with only one HPI pump available for mitigation. The initial power level was reduced from the previous value of 77% of 2568 MWt (nominal 75% of 2568 MWt plus 2% calorimetric uncertainty) in order to accommodate the more limiting axial power shape. The previous 75% partial-power SBLOCA analysis reported a peak cladding temperature (PCT) of 1788°F, without application of a penalty to address the 11-foot peaked axial power shape (Ref. 5). The reduction in maximum allowed initial core power level to 50% of 2568 MWt results in a PCT value of 1480.2°F. This PCT is less than the full-power SBLOCA analysis for full-core Mk-B-HTP fuel, where a PCT of 1597.5°F is reported (Ref. 6) (also see ONS UFSAR Section 15.14.4.2.3 and Table 15-64). Therefore, the new partial-power SBLOCA analysis is no longer limiting in terms of PCT for SBLOCAs.

To address the potential for CFT nitrogen intrusion into the RCS, the new partial-power SBLOCA analysis includes a new operator action to maintain SG pressure at 300 psig by throttling the ADVs during secondary-side depressurization. SG depressurization from one ADV was credited at 25 minutes after receipt of the engineered safety features actuation system (ESFAS) signal on low RCS pressure. This desired operator action is intended to limit the RCS depressurization, which then prevents the CFTs from completely emptying their liquid content during the depressurization. The secondary-side pressure control to preclude significant nitrogen injection is consistent with the generic Emergency Operating Procedure (EOP) guidance for B&W plants provided by AREVA in Reference 17. In the event the ADV modulation does not occur, the potential for CFT nitrogen intrusion into the RCS is evaluated in Section 4.2 of this enclosure for impacts to long-term core cooling. Proposed EOP guidance for operator action to throttle main steam pressure is discussed in Section 4.3 of this enclosure.

The ONS specific SBLOCA analysis used the NRC-approved methods contained in Volume II of BAW-10192P-A, Revision 0 (Ref. 4). The NRC-approved topical reports identified in BAW-10192P-A are:

- BAW-10162P-A, Rev. 0, TACO3 (Ref. 7).
- BAW-10164P-A, Rev. 3, RELAP5/MOD2-B&W (Rev. 3 of Reference 8).
- BAW-10095-A, Rev. 1, CONTEMPT (Ref. 11).

Since the approval of BAW-10192P-A, Rev. 0, the codes and methods have evolved through approved code revisions and the addition of new methods and error corrections made under 10 CFR 50.46. The following NRC approved topical reports have been added

as part of the evaluation model (EM) for SBLOCA analyses, as noted in the referenced NRC safety evaluations (SEs) for the respective topical reports.

- BAW-10164P-A, Rev. 4, RELAP5/MOD2-B&W (Rev. 4 of Reference 8; NRC SE per Reference 9) - [use of void-dependent cross-flow mode, and supplemental pins]
- BAW-10164P-A, Rev. 6, RELAP5/MOD2-B&W (Ref. 8; NRC SE per Reference 10) - [use of BHTP critical heat flux correlation]
- BAW-10227P-A, Rev. 1, M5 Cladding (Rev. 0 of Reference 12; NRC SE per Reference 13) - [use of M5 cladding, Rev. 1 not necessary for B&W plants]

BAW-10192P-A, Rev. 0 and the three topical reports shown immediately above are currently listed within the applicable Core Operating Limits Reports for ONS Units 1, 2, and 3.

The SBLOCA analysis also used several EM changes made in accordance with 10 CFR 50.46, to assure that 10 CFR 50 Appendix K requirements of that regulation are met. Those 10 CFR 50.46 changes that have not subsequently been approved within a revised topical report are listed below:

- Uncertainty-adjusted core flood tank parameters identified in Preliminary Safety Concern (PSC) 5-94, as reported by Duke Energy to the NRC (Ref. 14).
- SBLOCA reactor coolant pump two-phase degradation modeling for SBLOCAs identified in PSC 2-00. As described in the NRC's SE (Ref. 15), the results of PSC 2-00 are generically applicable to B&W plants.
- A new consideration regarding axial power shapes was developed while performing scoping studies for SBLOCA analyses. It was found that the axial location for the most bounding power shape of 1.7 for any time during the cycle is now found to be at the 11-foot core elevation. This issue was previously reported by Duke Energy to the NRC (Ref. 5). The ONS Mark-B-HTP full-core SBLOCA analysis which support this proposed changed used a skewed end-of-cycle 11-foot axial peak of 1.7.

#### 4.1.1 SBLOCA Peak Cladding Temperature Results

The full SBLOCA break size spectrum was evaluated for the new partial-power analysis. A loss of offsite power is assumed to occur at the time of reactor trip. The reactor coolant pumps are assumed to begin to coast down at the time of reactor trip. The single failure chosen minimizes the delivered emergency core cooling system (ECCS) flow. This assumption involves a power supply failure (4160 V switchgear TC, TD, or TE) that results in only one HPI pump supplying two cold legs, with the SBLOCA occurring on one of these two cold legs. The analysis takes credit for operator action to provide EFW to one steam generator within 20 minutes after receipt of the ESFAS signal on low RCS pressure, and to provide cooldown of one steam generator within 25 minutes of the ESFAS signal. This analysis has been previously reviewed and approved in the NRC Safety Evaluation dated September 6, 2000 (Ref. 3).

The new 50% partial-power SBLOCA analyses include a spectrum of cold leg pump discharge (CLPD) break sizes with a loss-of-offsite power (LOOP) coincident at the time of reactor trip (0.01, 0.04, 0.06, 0.07, 0.072, 0.08, 0.10, 0.13, 0.20, and 0.40 ft<sup>2</sup>). Non-LOOP

CLPD break sizes of 0.3, 0.4, and 0.5 ft<sup>2</sup> were evaluated with RCP trip occurring two minutes after loss of subcooling. Other cases considered include a 0.02464 ft<sup>2</sup> HPI line break and a 0.44 ft<sup>2</sup> CFT line break. The break spectrum is sufficient for identifying the limiting break sizes.

The limiting full-core Mark-B-HTP 50% full-power SBLOCA PCT was calculated to be 1480.2°F with a 0.072 ft<sup>2</sup> size break at the cold leg pump discharge, with a loss-of-offsite power coincident at the time of reactor trip (see table below). Therefore, 10 CFR 50.46 criterion (b)(1) is satisfied. It is noted that for break sizes which result in the highest PCT, the ADV is opened after the time when PCT is reached.

**ONS 50% Partial-Power SBLOCA PCT versus Break Size**

LOOP / RCP Trip	Break Location	Break Size (ft <sup>2</sup> )	PCT (°F)	Time of PCT (sec)	Time of ADV Opening (sec)
LOOP	CLPD	0.01	711.92	0.04	1673.1
		0.04	711.92	0.02	1549.3
		0.06	1401.5	1562.8	1533.5
		0.07	1446.5	1370.2	1528.7
		<b>0.072</b>	<b>1480.2</b>	1353.6	1527.9
		0.08	1359.1	1113.2	1525.0
		0.10	1288.9	891.5	1519.6
		0.13	1126.4	635.7	> EOT <sup>(2)</sup>
		0.20	756.89	387.3	> EOT <sup>(2)</sup>
	0.40	711.92	0.005	> EOT <sup>(2)</sup>	
	HPI Line	0.02464	711.92	0.02	1577.3
	CFT Line	0.44	712 <sup>(1)</sup>		
Non-LOOP 2-Minute RCP Trip on Loss of Subcooling	CLPD	0.30	712 <sup>(1)</sup>		
		0.40	1010.0	212.0	> EOT <sup>(2)</sup>
		0.50	1090 <sup>(1)</sup>		
	CFT Line	0.44	907 <sup>(1)</sup>		

Note: 1) The reported PCT is an estimated value.

2) " > EOT" means that the parameter occurs at a time greater than the End of the Transient

**4.1.2 SBLOCA Break Sizes**

The minimum SBLOCA break size required to be explicitly analyzed as part of the BWNT Loss-of-Coolant Accident (LOCA) EM (Ref. 4) is the 0.01 ft<sup>2</sup> break. Category 1 break sizes smaller than 0.01 ft<sup>2</sup> typically do not even interrupt natural circulation. These break sizes are more reliant of emergency feedwater (EFW) flow to remove a significant fraction of the core decay heat and maintain RCS pressure near main steam pressure until the HPI is capable of matching and exceeding the core decay heat energy addition. The core remains continuously covered so long as adequate SG heat removal maintains the RCS near main steam pressure.

Category 2 breaks analyzed include the 0.01 ft<sup>2</sup> CLPD break and a break in the HPI line with an area of 0.02464 ft<sup>2</sup>. The Category 2 break sizes present a greater challenge than the Category 1 breaks to the HPI System to replace lost liquid inventory. Both EFW and HPI are important in this category. For the 0.01 ft<sup>2</sup> CLPD break, continuous availability of EFW flow removes the decay heat from the reactor coolant that is not removed via the break. The PCT remained at the maximum initial cladding temperature; therefore, this break size category remains non-limiting in terms of PCT.

For the HPI line break, 0.02464 ft<sup>2</sup>, HPI flow is negligible until the ADV opens and subsequently depressurizes the RCS. It is this same action that brings the RCS to the CFT injection pressure, providing a greatly improved ECCS flow capable of recovering the reactor vessel (RV) liquid inventory. Core uncovering was thus prevented and the peak cladding temperature remained at the maximum initial cladding temperature such that the HPI line break represents a non-limiting break size with respect to the full spectrum of SBLOCA breaks.

Two break sizes were analyzed in Category 3, the 0.04 ft<sup>2</sup> and 0.06 ft<sup>2</sup> CLPD break. This category establishes the transition from preventing clad heat up to undergoing core uncovering. The ADV opens in the 0.04 ft<sup>2</sup> CLPD break case prior to hot channel cladding heat up, and the PCT remained at the maximum initial cladding temperature. The 0.06 ft<sup>2</sup> CLPD break is the first to experience clad heat up above the initial cladding temperature because the ADV opening occurs during hot channel core uncovering.

Category 4 break sizes analyzed are 0.07 ft<sup>2</sup>, 0.072 ft<sup>2</sup>, 0.08 ft<sup>2</sup>, 0.10 ft<sup>2</sup>, 0.13 ft<sup>2</sup>, and 0.20 ft<sup>2</sup>. This category establishes a transition from those break sizes which open the ADV prior to reaching the PCT (smaller breaks), to after the PCT has occurred (larger breaks). As a result, the cladding heat up for the smaller Category 4 breaks is greater than observed in Category 3. The most limiting break size is the 0.072 ft<sup>2</sup> CLPD with a LOOP, which resulted in the limiting PCT of 1480.2°F.

Category 5 break sizes analyzed include breaks in the CLPD with break areas between 0.25 ft<sup>2</sup> and 0.50 ft<sup>2</sup> with either a RCP trip concurrent with a low RCS pressure trip based on an assumed LOOP or an operator-initiated RCP trip two minutes after loss of subcooling margin (LSCM) with offsite power available. At 50% full power, break sizes larger than 0.40 ft<sup>2</sup> undergo departure from nucleate boiling within the first two seconds of the transient, and thus cannot be evaluated using NRC-approved SBLOCA methodology (Ref. 4). For the partial-power analysis, such breaks are therefore estimated based on knowledge derived from full-power SBLOCA analyses. For breaks larger than 0.3 ft<sup>2</sup>, worse consequences are expected if the RCPs were allowed to remain in operation for two minutes following LSCM, as identified in Preliminary Safety Concern 2-00 (Ref. 15) due to RCP two-phase degradation modeling.

Category 6 break sizes are those greater than the Category 5 SBLOCAs (0.50 ft<sup>2</sup> for Mark-B-HTP fuel) up to a full double-ended break of any RCS pipe. These break sizes are of sufficient size to cause the cladding to exceed the critical heat flux upon break initiation, and are considered large break loss-of-coolant accidents (LBLOCAs).

#### 4.1.3 Maximum Local Cladding Oxidation

The limiting full-core Mark-B-HTP 50% full-power SBLOCA maximum local cladding oxidation was calculated to be less than 0.5%. Therefore, 10 CFR 50.46 criterion (b)(2) (i.e., < 17%) is satisfied.

#### 4.1.4 Maximum Hydrogen Generation

The limiting full-core Mark-B-HTP 50% full-power SBLOCA maximum whole core hydrogen generation was calculated to be less than 0.01%. Therefore, 10 CFR 50.46 criterion (b)(3) (i.e.,  $\leq 1\%$ ) is satisfied.

#### 4.1.5 Coolable Core Geometry

The fourth acceptance criterion of 10 CFR 50.46 states that calculated changes in core geometry shall be such that the core remains amenable to cooling. Compliance with this criterion is based on considerations that include the condition of the fuel rods and assembly just prior to the SBLOCA transient, plus, any changes in geometry predicted as a result of the mechanical or thermal effects from the SBLOCA.

The AREVA calculations directly assess the alterations in core geometry from the clad swelling and rupture during a SBLOCA. These calculations demonstrate that the fuel pin is cooled successfully during the short-term phase of the SBLOCA. For the Mark-B-HTP fuel, the hot assembly flow area reduction at rupture is less than 71% for all analyzed SBLOCA cases (full-power and partial-power). Furthermore, the upper limit of possible channel blockage for all SBLOCAs, based on NUREG-0630 (Ref. 16) and AREVA Topical Report BAW-10227P-A (Ref. 12), is 90% since the rupture in a fuel assembly is distributed between the grid spans and does not become coplanar across the assembly. Therefore, the assembly retains a pin-coolant-channel arrangement that is capable of passing coolant along the pin to provide cooling for all regions of the assembly. The consequences of both mechanical and thermal deformation of the fuel assemblies in the core have been assessed. The resultant deformations have been shown to maintain control rod operation and coolable core configurations to successfully demonstrate that the coolable geometry requirements of 10 CFR 50.46 have been met and that the core is shown to remain amenable to core cooling.

#### 4.1.6 Long-Term Core Cooling (LTCC)

The long-term cooling of the core is ensured by maintaining ECCS flow in excess of the decay heat load and by preventing boric acid precipitation by establishing a long-term boron concentration control process. The ONS ECCS design and EOPs accomplish the long-term cooling function and meet these acceptance criteria as described in UFSAR Section 6.3.3.2.1, Boron Precipitation Evaluation.

#### **4.2 Evaluation of Partial-Power SBLOCA Results If ADV Modulation Does Not Occur**

For the purposes of calculating the acceptance to the first three criteria in 10 CFR 50.46 (maximum PCT, local oxidation, and whole core hydrogen generation), the short-term core cooling is the key phase of the event. The break sizes and locations that bypass the largest fraction of ECCS are typically limiting in this short-term phase. Analysis inputs that minimize the CFT pressure and flow during the CFT discharge are also limiting. Short-term core cooling is achieved by successful actuation of the ECCS and EFW systems and the SBLOCA analysis continues until the core decay heat is continuously removed by pumped ECCS injection and EFW heat removal via the SGs. The SBLOCA analysis again continues until the core is covered with a two-phase mixture and any heat up of the fuel pins has been abated. From that point on, short-term core cooling is complete and the remaining transient evaluation is during the LTCC phase.

AREVA performed a supplemental evaluation to address LTCC concerns in the event that the ADV depressurization is not controlled to maintain a steam pressure of 300 psig. Continued depressurization of the secondary-side can induce RCS depressurization that may result in the CFT emptying and nitrogen cover gas intrusion into the RCS. Nitrogen gas intrusion has the potential to interrupt primary-to-secondary heat transfer, and thus degrade LTCC effectiveness for very small break sizes that need SG heat removal. The potential for nitrogen gas intrusion was assessed for all break locations and sizes. These additional evaluations support the new partial-power SBLOCA analysis, and do not change the PCTs summarized in Section 4.1.1 of this enclosure for the new partial-power SBLOCA analysis.

The limiting PCT break sizes and locations as well as CFT inputs are not always limiting for LTCC and have been assessed differently and separately from the short-term core cooling analysis described in Section 4.1 of this enclosure. Challenges to 10 CFR 50.46 acceptance criteria for short-term core cooling analyses are typically maximized, by use of conservative inputs that include the minimum CFT gas pressure and maximum CFT liquid level, because it reduces the CFT pressure during the transient. With lower CFT pressures, the rate of CFT liquid discharge is decreased and the result is the potential for higher PCTs and higher metal-water reaction rates, which in turn maximize the oxidation and hydrogen generation. The AREVA analysis of adequate short-term core cooling is used as the origination for assessing acceptable LTCC. However, in the case where the nitrogen gas intrusion is identified as a potential challenge for maintaining primary-to-secondary heat transfer, the short-term core cooling inputs are neither necessarily limiting nor appropriate for LTCC assessments.

Successful LTCC evaluations demonstrate that there is no core uncovering and heat up following successful short-term core cooling. The LTCC evaluation considers the key CFT inputs and SBLOCA break spectrum (size and location) to show that the RCS pressure can be controlled and that the HPI flow reaching the core continuously matches or exceeds the core decay heat boil-off rates. The RCS pressure is controlled by showing that the break energy relief and SG heat removal matches or exceeds the core energy addition. If the break energy relief is sufficient by itself, no SG heat removal is needed, so smaller break sizes that need continuous SG heat removal to augment the break energy relief are the biggest challenge to this evaluation.

The CFT gas temperature increase due to containment heating during the SBLOCA transient was considered with the maximum CFT gas pressure and minimum CFT liquid volume to maximize the pressure at which the CFT could empty. HPI flow / decay heat match-up and break energy relief state point evaluations were used to determine RCS pressure evolution and if there was sufficient HPI flow at those RCS pressures. Break locations that are limiting for the typical licensing analyses demonstrating compliance with the first three criteria of 10 CFR 50.46, CLPD and HPI line breaks, are found to be not limiting for nitrogen intrusion. Larger break sizes that depressurize the RCS quickly have no reliance on SG cooling during LTCC; hence, pumped ECCS cooling will continue to match decay heat with or without CFT nitrogen intrusion. Smaller break sizes will either not depressurize the RCS sufficiently to empty the CFTs, or the nitrogen gas will nonetheless enter the RCS but it can vent directly through the break without reaching the SG tubes. The hot leg (HL) and cold leg pump suction (CLPS) break locations provide full ECCS flow to the core, thus increasing effective core cooling. However, since these locations do not provide a ready escape path for the nitrogen gas, explicit RELAP5 hot leg break analyses were performed to assure that the smaller HL breaks will not uncover the core or result in CFT emptying. In the event the CFT did empty, the calculations must demonstrate that the RCS will not repressurize to pressures at which the ECCS does not match the core decay heat and provide a challenge to core uncovering and continuous LTCC.

The long-term RCS pressure state point evaluations suggests that for break locations where all of the HPI flow reaches the core (CFT line breaks, HL breaks, and CLPS breaks), only very small break sizes ( $\sim 0.005 \text{ ft}^2$  and less) could result in a substantial RCS repressurization if SG heat removal was lost due to the effects of nitrogen gas from a completely discharged CFT. Therefore, several small HL break sizes with a single HPI pump and combinations of one or two ADV blowdowns were performed with the RELAP5/MOD2-B&W SBLOCA model to determine if the CFTs could completely discharge their liquid contents before the HPI cooling could match the core decay heat at the maximum pressure that the RCS could reach without SG heat removal. Two cases with break sizes of  $0.005 \text{ ft}^2$  and  $0.004 \text{ ft}^2$  were performed with a single ADV blowdown of one SG (the SG in the simulated broken RCS loop which was the loop without the pressurizer) to determine if the RCS reaches pressures at which the CFT completely discharges its liquid content.

The results of these two RELAP5/MOD2-B&W analyses with a single ADV blowdown show that neither of these two tiny HL break sizes analyzed would completely discharge the liquid content of the CFT. The core remains covered throughout the first four hours for both cases. For HL breaks smaller than  $0.005 \text{ ft}^2$ , the completed analysis support the conclusions that the CFTs will not completely discharge their liquid contents in four hours when a single ADV blowdown is credited. Complete CFT discharging for smaller break sizes would either not occur, or would be delayed for a much longer time because the HPI flow matches the break flow at the lower pressures and the RCS does not reach the CFT totally discharged pressure.

The short-term core cooling SBLOCA analysis described in Section 4.1 of this enclosure for the single HPI pump was performed with credit for only one SG blowdown; however, long-term RCS cooldown could be affected if both SGs were depressurized. Therefore, two additional RELAP5/MOD2-B&W analyses for the  $0.004 \text{ ft}^2$  and the  $0.0025 \text{ ft}^2$  breaks on the

bottom of the HL were performed to assess LTCC with blowdown of both SGs. These two SG blowdown cases were also analyzed up to 4 hours. The results of these two RELAP5 analyses with dual SG blowdown show that neither of these small HL break sizes would completely discharge the liquid contents of the CFT. As expected, the larger break size reached lower RCS pressures, but the CFTs did not empty, and there is effective LTCC with the core remaining continuously covered throughout the first four hours for both cases.

Larger break sizes may completely discharge the CFT, but they cannot repressurize above the pressure where HPI can match decay heat. The smaller break sizes will not completely discharge the CFTs because the HPI exceeds the break flow. Based on these calculations and favorable decay heat removal rate comparisons, AREVA concluded that LTCC is assured with or without nitrogen gas intrusion for all break sizes. The RELAP5 calculations show that nitrogen gas intrusion does not occur for the tiny break sizes that rely on SG heat removal for a number of hours. In the longer term, core cooling is still assured if the CFTs completely discharge their liquid contents much later because at these longer times following the reactor trip, the lower decay heat levels can be matched by HPI cooling.

Based on the evaluation of impacts to long-term core cooling if ADV modulation does not occur, the operator action modeled in the partial-power SBLOCA analysis to maintain steam generator pressure at 300 psig as described in Section 4.1 of this enclosure is considered to be a desired action, and not a required action needed to demonstrate post-LOCA long-term core cooling.

#### **4.3 Operator Actions**

ONS's current licensing basis contains a SBLOCA analysis initiated at partial-power conditions, where only one HPI pump is credited for mitigation. This analysis models an operator action to open an ADV in the SG being fed with EFW at 25 minutes following an ESFAS actuation. This LAR does not propose any changes to the timing required to perform this operator action to initiate secondary-side depressurization. This LAR does not propose a change to licensing basis with respect to required operator actions.

The new 50% partial-power SBLOCA analysis supporting this proposed LAR includes a sensitivity case that models a desired operator action to modulate the main steam pressure at 300 psig via the ADV during the secondary-side depressurization. The purpose of the ADV modulation to maintain steam pressure is to limit RCS depressurization, which then prevents the CFTs from completely discharging their liquid contents and introducing nitrogen gas into the RCS during the depressurization. The secondary-side pressure control to preclude significant nitrogen injection is consistent with the generic Emergency Operating Procedure (EOP) guidance for B&W plants provided by AREVA in Reference 17.

Radiological dose analysis determined that the dose received by the operator(s) for this new action will be within regulatory limits. Thus the desired operator action to modulate the main steam pressure at 300 psig has no adverse impact and is considered acceptable.

The plant must be controlled to a main steam pressure that is less than the value assumed in the new 50% partial-power SBLOCA analysis, since less borated water from the CFT would be injected at the higher analyzed value. This will ensure that the new SBLOCA analysis remains conservative with respect to actual plant operation.



Proposed EOP guidance for use in LSCM or Inadequate Core Cooling scenarios would have the operator throttle the ADV to maintain steam pressure in a range of 250 – 275 psig. This range provides some margin to the value of 300 psig that is employed in the new partial-power SBLOCA analysis, and is greater than the expected pressures where the CFTs could empty. The proposed EOP guidance will consider instrument uncertainty to assure actual steam pressure does not exceed 300 psig, and include a band for operator control. The operator is assumed to be controlling steam pressure with the ADVs using the local steam pressure indication in the Turbine Building. The calculated uncertainty in this indication is  $\pm 23.6$  psig, rounded up to  $\pm 25$  psig for the upper end of the control band. In addition, a control band of 25 psig is considered to be acceptable.

Proposed EOP guidance would also include steps to isolate the CFTs, if possible, while the operators are holding steam pressure in the desired range. Isolation of the CFTs to prevent significant nitrogen intrusion into the RCS is consistent with the generic EOP guidance for B&W plants provided by AREVA in Reference 17. The new partial-power SBLOCA analysis modeling the modulation of steam pressure at 300 psig would allow operating space within the EOPS such that CFT isolation would not conflict with the applicable safety analysis in terms of isolating the borated water source from the CFTs.

The proposed EOP guidance to maintain steam generator pressure at 300 psig is considered to be a desired action, and not a required action. This is based on the evaluation of impacts to long-term core cooling if ADV modulation does not occur, as described in Section 4.1 of this enclosure.

## **5 REGULATORY EVALUATION**

### **5.1 Significant Hazards Consideration**

Duke Energy Carolinas, LLC (Duke Energy) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment to Oconee Nuclear Station (ONS) Facility Operating Licenses DPR-38, DPR-47, and DPR-55 by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below.

The requested change will revise the Technical Specifications (TS) by reducing the allowed maximum rated thermal power (RTP) at which the unit can operate when select High Pressure Injection (HPI) System equipment is inoperable. This TS change is described in Section 3 above and detailed markups of the affected TS pages are included in Attachment 1 to this License Amendment Request (LAR).

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed TS changes do not modify the reactor coolant system pressure boundary, nor make any physical changes to the facility design, material, or construction standards. The probability of any design basis accident (DBA) is not affected by this change, nor are the consequences of any DBA affected by this change. The new small break loss-of-coolant accident (SBLOCA) partial-power analysis demonstrates that all 10 CFR 50.46 acceptance criteria are satisfied. Radiological consequences for loss-of-coolant

accident (LOCA) events are evaluated in ONS Updated Final Safety Analysis Report Section 15.15 for the Maximum Hypothetical Accident. The proposed changes will not impact assumptions and conditions previously used in the radiological consequence evaluations for the Maximum Hypothetical Accident. The proposed changes do not involve changes to any structures, systems, or components (SSCs) that can alter the probability for initiating a LOCA event.

Therefore, the proposed TS changes do not significantly increase the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed TS changes reduce the allowed power level that the unit may be operated at with select HPI equipment out-of-service. The changes do not alter the plant configuration (no new or different type of equipment will be installed) or make changes in methods governing normal plant operation. No new failure modes are identified, nor are any SSCs required to be operated outside the design bases. Therefore, the possibility of a new or different kind of accident from any kind of accident previously evaluated is not created.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed TS changes are supported by SBLOCA analyses which demonstrate that the acceptance criteria of 10 CFR 50.46 are satisfied. These analyses were performed in accordance with the Evaluation Model described in AREVA Topical Report BAW-10192P-A. The new SBLOCA analysis assumes a lower initial core power level (50% of rated thermal power (RTP)) than what was previously analyzed in support of TS 3.5.2 (i.e., 75% of RTP). The resulting peak cladding temperature results for the new SBLOCA analysis are lower than the existing analysis. In addition, a supplemental evaluation demonstrated that failure to perform a desired operator action of maintaining secondary-side pressure at 300 psig by throttling the atmospheric dump valve during a SBLOCA did not result in adverse affects to the new SBLOCA analysis results. Therefore, it is concluded that the proposed amendment request will not result in a significant decrease in the margin of safety.

Based on the above, Duke Energy concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of “no significant hazards consideration” is justified.

## **5.2 Applicable Regulatory Requirements/Criteria**

The regulatory requirements that apply to this LAR, and how ONS satisfies the requirements, are provided in the following table.

Regulatory Requirements/Criteria	How Satisfied
<p>10 CFR 50.36(c)(2)(i) – Limiting conditions for operation (LCOs) are the lowest functional capability or performance levels of equipment required for safe operation of the facility.</p>	<p>Proposed TS 3.5.2 changes ensure that the LCO requirements for HPI are still the lowest functional capability or performance levels of equipment required for safe operation of the facility.</p>
<p>10 CFR 50.36(c)(2)(ii) - A TS LCO of a nuclear reactor must be established for each item meeting one or more of the following criteria:                      ...                      (C) <i>Criterion 3.</i> A SSC that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.</p>	<p>Proposed TS 3.5.2 changes ensure that the LCO requirements for HPI equipment are part of the primary success path and function or actuate to mitigate a SBLOCA that presents a challenge to the integrity of a fission product barrier.</p>
<p>10 CFR 50.46 – The following acceptance criteria shall be met following a LOCA:                      (1) Maximum fuel element cladding temperature is <math>\leq 2200^{\circ}\text{F}</math>;                      (2) Maximum cladding oxidation is <math>\leq 0.17</math> times the total cladding thickness before oxidation;                      (3) Maximum hydrogen generation from a zirconium water reaction is <math>\leq 0.01</math> times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;                      (4) Core is maintained in a coolable geometry;                      and                      (5) Adequate long term cooling capability is maintained.</p>	<p>Proposed TS 3.5.2 change ensures the five acceptance criteria defined in 10 CFR 50.46 are maintained.</p>
<p>10 CFR 50.59(c)(1)(i) - a licensee is required to submit a license amendment pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," if a change to the TS is required. Furthermore, the requirements of 10 CFR 50.59 necessitate that the NRC approve the TS changes before the TS changes are implemented.</p>	<p>This LAR meets the requirements of 10 CFR 50.59(c)(1)(i) and 10 CFR 50.90.</p>

### 5.3 Precedent

A review of the NRC's Agencywide Documents Access and Management System (ADAMS) for prior similar or related LARs at other B&W plants since January 1, 2000 resulted in no documents of precedence. Prior Duke Energy LARs or licensing actions involving the HPI System are provided in the following documents:

- Letter, W. R. McCollum, Jr. (Duke Energy) to USNRC, Oconee Nuclear Station Units 1, 2, and 3, Proposed Amendment to the Facility Operating License Regarding the High Pressure Injection System Requirements Technical Specification Change No. 98-13, December 16, 1998 (Ref. 2).
- NRC Letter to Duke Energy, Oconee Nuclear Station, Units 1, 2 and 3 RE: Issuance of Amendments (TAC NOS. MA4451, MA4452, AND MA4453), September 6, 2000 (Ref. 3).

### 5.4 Conclusions

In Section 5.1, Duke Energy made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by NRC regulations in 10 CFR 50.92. The regulatory requirements and guidance applicable to this LAR are identified in Section 5.2.

## 6 ENVIRONMENTAL CONSIDERATION

Duke Energy Carolinas, LLC (Duke Energy) has evaluated this License Amendment Request (LAR) against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. Duke Energy has determined that this LAR meets the criteria for a categorical exclusion as set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that the amendment meets the following specific criteria:

- The amendment involves no significant hazard consideration as demonstrated in Section 5.1.
- There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite. The principal barriers to the release of radioactive materials are not modified or affected by this change and no significant increases in the amounts of any effluent that could be released offsite will occur as a result of this change.
- There is no significant increase in individual or cumulative occupational radiation exposure. Because the principal barriers to the release of radioactive materials are not modified or affected by this change, there will be no significant increase in individual or cumulative occupational radiation exposure resulting from this change.

Therefore, no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment pursuant to 10 CFR 51.22(b).

## 7 REFERENCES

1. Letter, R. W. Reid (USNRC) to W. O. Parker, Jr. (Duke Power Co.), Issuance of Amendments 65, 65 and 62 for Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2, and 3, October 23, 1978 [ADAMS Accession No. ML011980166].
2. Letter, W. R. McCollum, Jr. (Duke Energy) to USNRC, Oconee Nuclear Station Units 1, 2, and 3, Proposed Amendment to the Facility Operating License Regarding the High Pressure Injection System Requirements Technical Specification Change No. 98-13, December 16, 1998 [ADAMS Legacy Library Accession No. 9812230267].
3. USNRC Letter to Duke Energy, Oconee Nuclear Station, Units 1, 2 and 3 RE: Issuance of Amendments (TAC NOS. MA4451, MA4452, AND MA4453), September 6, 2000 [ADAMS Accession No. ML003748730].
4. AREVA Topical Report BAW-10192P-A, Revision 0, "BWNT LOCA - BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants", June 1998 [ADAMS Legacy Library Accession No. 9808250157].
5. Letter, T. C. Geer (Duke Energy) to USNRC, "30-Day Report Pursuant to 10 CFR 50.46, Changes to or Errors in an Evaluation Model", August 19, 2010 [ADAMS Accession No. ML102360485].
6. Letter, R. M. Glover (Duke Energy) to USNRC, "30-Day Report Pursuant to 10 CFR 50.46, Changes to or Errors in an Evaluation Model", December 8, 2011 [ADAMS Accession No. ML11347A193].
7. AREVA Topical Report BAW-10162P-A, Revision 0, "TACO3 – Fuel Pin Thermal Analysis Code", October 1989 [ADAMS Legacy Library Accession No. 9001030124].
8. AREVA Topical Report BAW-10164P-A, Revision 6, "RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis Code", June 2007.
9. Letter, L. W. Barnett (USNRC) to J. F. Mallay (Framatome ANP), Safety Evaluation of Framatome Technologies Topical Report BAW-10164P Revision 4, "RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analyses" (TAC Nos. MA8465 and MA8468), April 9, 2002 [ADAMS Accession No. ML013390204].
10. Letter, H. K. Nieh (USNRC) to R. Gardner (AREVA NP), Final Safety Evaluation for AREVA NP Topical Report BAW-10164(P), Revision 6, RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis (TAC No. MD2187), June 25, 2007 [ADAMS Accession No. ML102360485].
11. AREVA Topical Report BAW-10095-A, Revision 1, "CONTEMPT – Computer Program for Predicting Containment Pressure-Temperature Response to a LOCA", April 1978.
12. AREVA Topical Report BAW-10227P-A, Revision 1, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel", June 2003.

13. Letter, S. A. Richards (USNRC) to T. A. Coleman (Framatome Comega Fuels), Revised Safety Evaluation (SE) for Topical Report BAW-10227P: "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel" (TAC No. M99903), February 4, 2000 [ADAMS Accession Nos. ML003681479 and ML003681490].
14. Letter, M. S. Tuckman (Duke Energy) to USNRC, "Report Pursuant to 10 CFR 50.46, Changes to or Errors in an Evaluation Model", February 2, 1998 [ADAMS Legacy Library Accession No. 9802110094].
15. Letter, S. A. Richards (USNRC) to J. S. Holm (Framatome ANP), Request for Amendment of Safety Evaluation for "Report of Preliminary Safety Concern (PSC) 2-00 Related to Core Flood Line Break with 2-Minute Operator Action Time" (TAC No. MA9973), January 10, 2005 [ADAMS Accession No ML043550355].
16. NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis," April 1980 [ADAMS Accession No. ML053490337].
17. AREVA Document 74-1152414-11, "Emergency Operating Procedures Technical Bases Document", March 31, 2012; Applicable to B&W Sites.

License Amendment Request No. 2013-03  
June 30, 2014

**ATTACHMENT 1**

**MARKED-UP TECHNICAL SPECIFICATIONS PAGES**  
**[2 pages follow this cover page]**

**NOTE:** This attachment contains marked-up TS Pages 3.5.2-2 and -3.

Change "75%" to "50%"

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to $\leq 75\%$ RTP.	12 hours
	AND	
	B.2 Verify by administrative means that the ADV flow path for each steam generator is OPERABLE.	12 hours
	AND	
	B.3 Restore HPI pump to OPERABLE status.	30 days from initial entry into Condition A
	AND	
	B.4 Restore HPI discharge crossover valve(s) to OPERABLE status.	30 days from initial entry into Condition A

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
<p>C. One HPI train inoperable.</p>	<p>C.1 -----NOTE----- Only required when inoperable HPI train is incapable of automatic actuation and incapable of actuation through remote manual alignment.</p> <p>Reduce THERMAL POWER to <math>\leq 75\%</math> RTP.</p> <p>AND <span style="border: 1px solid black; padding: 2px;">Change "75%" to "50%"</span></p> <p>C.2 -----NOTE----- Only required when THERMAL POWER <math>\leq 75\%</math> RTP.</p> <p>Verify by administrative means that the ADV flow path for each steam generator is OPERABLE.</p> <p>AND</p> <p>C.3 Restore HPI train to OPERABLE status.</p>	<p>3 hours</p> <p>3 hours</p> <p>72 hours</p>	
<p>D. HPI suction headers not cross-connected.</p>	<p>D.1 Cross-connect HPI suction headers.</p>	<p>72 hours</p>	
<p>E. HPI discharge headers cross-connected.</p>	<p>E.1 Hydraulically separate HPI discharge headers.</p>	<p>72 hours</p>	

(continued)

License Amendment Request No. 2013-03  
June 30, 2014

**ATTACHMENT 2**

**MARKED-UP TECHNICAL SPECIFICATION BASES PAGES**  
**[7 pages follow this cover page]**

**NOTE:** This attachment contains the following marked-up TS Bases pages:

B 3.5.2-7	B 3.7.4-2
B 3.5.2-8	Insert A for B 3.7.4-2
B 3.5.2-9	B 3.7.4-3
B 3.5.2-10	

BASES

APPLICABILITY (continued) Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation – High Water Level," and LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation – Low Water Level."

ACTIONS

A.1 and A.2

With one HPI pump inoperable, or one or more HPI discharge crossover valve(s) (i.e., HP-409 and HP-410) inoperable, the HPI pump and discharge crossover valve(s) must be restored to OPERABLE status within 72 hours. The HPI System continues to be capable of mitigating an accident, barring a single failure. The 72 hour Completion Time is based on NRC recommendations (Ref. 4) that are based on a risk evaluation and is a reasonable time for many repairs.

In the event HPI pump "C" becomes inoperable, Condition C must be entered as well as Condition A. Until actions are taken to align an HPI pump to HPI train "B," HPI train "B" is inoperable due to the inability to automatically provide injection in response to an ESPS signal.

This Condition permits multiple components of the HPI System to be inoperable concurrently. When this occurs, other Conditions may also apply. For example, if HPI pump "C" and HP-409 are inoperable coincidentally, HPI train "B" is incapable of being automatically actuated or manually aligned from the Control Room. Thus, Required Action C.1 would apply.

In order to utilize another HPI pump to supply HPI train "B" when HPI pump "C" is inoperable, HP-116 must be opened. This action results in cross-connecting the HPI discharge headers; thus, Condition E must be entered. HP-115 may be closed to provide hydraulic separation provided that pump minimum flow requirements are maintained. However, two operating pumps would be required for this configuration, one to provide makeup flow and one to provide seal injection flow.

B.1, B.2, B.3, and B.4

Replace "75%" with "50%"

If the Required Action and associated Completion Time of Condition A is not met, THERMAL POWER of the unit must be reduced to  $\leq 75\%$  RTP within 12 hours. The 12 hour Completion Time is reasonable, based on operating experience, to reach the required unit condition from full power conditions in an orderly manner and without challenging unit systems. This time is less restrictive than the Completion Time for Required Action C.1,

BASES

ACTIONS B.1, B.2, B.3, and B.4 (continued)

because the HPI System remains capable of performing its function, barring a single failure.

Two HPI trains are required to mitigate specific small break LOCAs, if no credit for enhanced steam generator cooling is assumed in the accident analysis. However, if equipment not qualified as QA-1 (i.e., an atmospheric dump valve (ADV) flow path for a steam generator) is credited for enhanced steam generator cooling, the safety analyses have determined that the capacity of one HPI train is sufficient to mitigate a small break LOCA on the discharge of the reactor coolant pumps if reactor power is  $\leq 75\%$  RTP.

Required Actions B.2, B.3, and B.4 modify the HPI pump and discharge crossover valve OPERABILITY requirements to permit reduced requirements at power levels  $\leq 75\%$  RTP for an extended period of time. Required Action B.2 provides a compensatory measure to verify by administrative means that the ADV flow path for each steam generator is OPERABLE within 12 hours. This compensatory measure provides additional assurance regarding the ability of the plant to mitigate an accident. Compliance with this requirement can be established by ensuring that the ADV flow path for each steam generator is OPERABLE in accordance with LCO 3.7.4, "Atmospheric Dump Valve (ADV) Flow Paths."

Replace "75%" with "50%"

Required Actions B.3 and B.4 require that the HPI pump and discharge crossover valve(s) be restored to OPERABLE status within 30 days from initial entry into Condition A. The 30-day time period limits the time that the plant can operate while relying on non QA-1 ADVs to provide enhanced steam generator cooling to mitigate small break LOCAs. The 30-day time period is acceptable, because:

1. Without crediting an ADV flow path, the HPI System remains capable of performing the safety function, barring a single failure;
2. If credit is taken for an ADV flow path for a steam generator, the safety analysis has demonstrated that only one HPI train is required to mitigate the consequences of a small break LOCA when THERMAL POWER is  $\leq 75\%$  RTP. Thus, for this case, the HPI System would be capable of performing its safety function even with an additional single failure;

BASES

ACTIONS B.1, B.2, B.3, and B.4 (continued)

3. OPERABILITY of the ADV flow path for each steam generator is required to be confirmed by Required Action B.2 within 12 hours. Additional defense-in-depth is provided, because the ADV flow path for only one steam generator is required to mitigate the small break LOCA; and
4. A risk-informed assessment (Ref. 7) concluded that operating the plant in accordance with these Required Actions is acceptable.

ACTIONS C.1, C.2, and C.3

If the plant is operating with THERMAL POWER  $> 75\%$  RTP, two HPI pumps capable of providing flow through two HPI trains are required. One HPI train is required to provide flow automatically upon receipt of an ESPS signal, while flow through the other HPI train must be capable of being established from the Control Room within 10 minutes. Thus, if the plant is operating at  $> 75\%$  RTP, and one HPI train is inoperable and incapable of being automatically actuated or manually aligned from the Control Room to provide flow post-accident, the HPI System would be incapable of performing its safety function. For this Condition, Required Action C.1 requires the power to be reduced to  $\leq 75\%$  RTP within 3 hours. Required Action C.1 is modified by a Note which limits its applicability to the condition defined above. The 3 hour Completion Time is considered reasonable to reduce the unit from full power conditions to  $\leq 75\%$  RTP in an orderly manner and without challenging unit systems. The time frame is more restrictive than the Completion Time provided in Required Action B.1 for the same action, because the condition involves a loss of safety function.

If the plant is operating with THERMAL POWER  $> 75\%$  RTP and the inoperable HPI train can be automatically actuated or manually aligned to provide flow post-accident, Required Action C.3 permits 72 hours to restore the HPI train to an OPERABLE status.

If enhanced steam generator cooling is not credited in the accident analysis, two HPI trains are required to mitigate specific small break LOCAs with THERMAL POWER  $\leq 75\%$  RTP. However, if equipment not qualified as QA-1 (i.e., an ADV flow path for a steam generator) is credited for enhanced steam generator cooling, the safety analyses have determined that the capacity of one HPI train is sufficient to mitigate a small break LOCA on the discharge of the reactor coolant pumps if THERMAL POWER is  $\leq 75\%$  RTP. In order to permit an HPI train to be inoperable regardless of the reason when THERMAL POWER is  $\leq 75\%$  RTP, Required Action C.2 provides a compensatory measure to verify by administrative means that the ADV flow path for each steam generator is

Replace "75%" with "50%"

BASES

ACTIONS

C.1, C.2, and C.3 (continued)

OPERABLE within 3 hours. This Required Action is modified by a Note which states that it is only required if THERMAL POWER is  $\leq 75\%$  RTP. This compensatory measure provides assurance regarding the ability of the plant to mitigate an accident while in the Condition and THERMAL POWER  $\leq 75\%$  RTP. Compliance with this requirement can be established by ensuring that the ADV flow path for each steam generator is OPERABLE in accordance with LCO 3.7.4, "Atmospheric Dump Valve (ADV) Flow Paths."

With one HPI train inoperable, the inoperable HPI train must be restored to OPERABLE status within 72 hours. This action is appropriate because:

1. With THERMAL POWER  $\leq 75\%$  RTP, the safety analysis demonstrates that only one HPI train is required to mitigate the consequences of a small break LOCA assuming credit is taken for the ADV flow path for one steam generator. The OPERABILITY of the ADV flow path for each steam generator is confirmed by Required Action C.2 within 3 hours. This provides additional defense-in-depth. Additionally, a risk-informed assessment (Ref. 7) concluded that operating the plant in accordance with this Required Action is acceptable.
2. With THERMAL POWER  $> 75\%$  RTP, the remaining OPERABLE HPI train is capable of automatic actuation, and the inoperable train can be manually aligned by operator action to cross-connect the discharge headers of the HPI trains. This manual action was approved by the NRC in Reference 6.

Replace "75%" with "50%"

D.1

With the HPI suction headers not cross-connected, the HPI suction headers must be cross-connected within 72 hours. The HPI System continues to be capable of mitigating an accident, barring a single failure. The 72 hour Completion Time is based on NRC recommendations (Ref. 4) that are based on a risk evaluation and is a reasonable time for many repairs.

An argument similar to that utilized for Required Actions B.2, B.3, and B.4 could have been made for operating the HPI System with the suction headers not cross-connected for an extended period of time. However, this action was not considered prudent, due to the potential of damaging two HPI pumps in the event HP-24 or HP-25 failed to open in response to an ESPS signal while the HPI suction headers were not cross-connected.

BASES

Replace "75%"  
with "50%"

APPLICABLE  
SAFETY ANALYSIS

The SGTR analysis credits operator action to depressurize the steam generators by opening both ADV flow paths (i.e., the ADV flow path for each steam generator) within 40 minutes of identifying the ruptured steam generator. Within this 40-minute time period, the operators are only required to open the bypass valve, the block valve, and the throttle valve. However, later in the event, the analysis also assumes that the operators will open the isolation valves in each ADV flow path.

Operator action to depressurize a steam generator via its ADV flow path is credited in the analysis of certain small break LOCAs with THERMAL POWER  $\leq$  75% RTP and the plant operated with a degraded HPI System. This event credits operator action to open one ADV flow path within 25 minutes of an Engineered Safeguards Protective System (ESPS) actuation.

If enhanced steam generator cooling is not credited in the small break LOCA analysis, two HPI trains are required to mitigate specific small break LOCAs. However, if equipment not qualified as QA-1 (i.e., an ADV flow path for a steam generator) is credited for enhanced steam generator cooling, the safety analyses have determined that the capacity of one HPI train is sufficient to mitigate a small break LOCA on the discharge of the reactor coolant pumps if THERMAL POWER is  $\leq$  75% RTP.

The analysis for degraded HPI credits an ADV flow path for one steam generator as a compensatory measure in the event an HPI train is inoperable and THERMAL POWER is  $\leq$  75% RTP. During this situation, the ADV flow path for one steam generator is credited during certain small break LOCAs to depressurize the steam generator and enhance primary-to-secondary heat transfer. This is done in conjunction with the EFW System providing cooling water to the steam generator. The ADV flow path is comprised of manual valves. Operator action is credited for establishing the ADV flow path within 25 minutes of an ESPS signal.

Additionally, the ADV flow path for each steam generator is credited as a compensatory measure in TS 3.5.2, "High Pressure Injection (HPI)." Typically, single failures are not considered once the plant has entered a condition defined in the TS. However, the Completion Time permitted when the HPI system is degraded, is an extended period of time. In the event an accident occurred during this extended Completion Time and a single failure were to occur in the degraded HPI system, the ability of a plant to mitigate the consequences of specific small break LOCAs continues to be assured by the ADV flow path for one steam generator.

The ADV flow paths satisfy Criterion 3 of 10 CFR 50.36 (Ref. 1).

Add Insert A

**Insert A (for TS Bases Page 3.7.4-2)**

"The 50% partial-power SBLOCA analysis models a new operator action to modulate the main steam pressure at 300 psig via the ADV during the secondary-side depressurization. The purpose of the ADV modulation to maintain steam pressure is to limit Reactor Coolant System (RCS) depressurization, which then prevents the CFTs from completely discharging their liquid contents and introducing nitrogen gas into the RCS during the depressurization. The secondary-side pressure control to preclude significant nitrogen injection is consistent with the generic Emergency Operating Procedure (EOP) guidance for B&W plants provided by AREVA.

To ensure that the new SBLOCA analysis is bounding, the plant must be controlled to a main steam pressure that is less than the value assumed in the 50% partial-power SBLOCA analysis, since less borated water from the CFT would be injected at the higher analyzed value. This ensures that the 50% partial-power SBLOCA analysis remains conservative with respect to actual plant operation. The 50% partial-power SBLOCA analysis modeling the modulation of steam pressure at 300 psig allows operating space within the EOPs such that CFT isolation does not conflict with the applicable safety analysis in terms of isolating the borated water source from the CFTs.

A supplemental SBLOCA analysis demonstrates that long-term core cooling is assured with or without nitrogen gas intrusion for all break sizes. The analyses show that nitrogen gas intrusion does not occur for the small break sizes that rely on steam generator heat removal for a number of hours. In the longer term, core cooling is still assured if the CFTs completely discharge their liquid contents much later because at these longer times following the reactor trip, the lower decay heat levels can be matched by HPI cooling.

Based on the evaluation of impacts to long-term core cooling if ADV modulation does not occur, the operator action modeled in the partial-power SBLOCA analysis to maintain steam generator pressure at 300 psig is considered to be a desired action, and not a required action needed to demonstrate post-LOCA long-term core cooling."



**BASES**

---

**LCO**

The ADV flow path for each steam generator is required to be OPERABLE. The failure to meet the LCO can result in the inability to depressurize the steam generators following a SGTR.

The ADV flow path for each steam generator is required to be OPERABLE. Failure to meet the LCO can result in the inability to depressurize a steam generator following a small break LOCA. This function is required to support operation with a degraded HPI System when THERMAL POWER is  $\leq$  75% RTP.

Replace "75%" with "50%"

An ADV flow path is considered OPERABLE when it is capable of providing a controlled relief of the main steam flow, and each valve which comprises the ADV flow path is capable of opening and closing.

---

**APPLICABILITY**

The ADV flow path for each steam generator is required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal. In MODE 4, steam generators are relied upon for heat removal whenever an RCS loop is required to be OPERABLE or operating to satisfy LCO 3.4.5, "RCS Loops - MODE 4" or available to transfer decay heat to satisfy LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled." The steam generators do not contain a significant amount of energy in MODE 4 when the unit is not relying upon a steam generator for heat transfer, and MODES 5 and 6; therefore, the ADV flow paths are not required to be OPERABLE in these MODES and condition.

With the ADV flow paths required to be OPERABLE at all times that the steam generators are being relied upon for heat removal, it is assured that the ADV flow paths will be available for use for mitigation of a SBLOCA and a SGTR. These are the only two conditions in which the use of the ADV flow paths is credited in the analyses of any accident.

License Amendment Request No. 2013-03  
June 30, 2014

**ATTACHMENT 3**

**RETYPE TECHNICAL SPECIFICATIONS PAGES**  
**[2 pages follow this cover page]**

**NOTE:** This attachment contains retyped TS Pages 3.5.2-2 and -3.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to $\leq 50\%$ RTP.	12 hours
	<u>AND</u>	
	B.2 Verify by administrative means that the ADV flow path for each steam generator is OPERABLE.	12 hours
	<u>AND</u>	
	B.3 Restore HPI pump to OPERABLE status.	30 days from initial entry into Condition A
	<u>AND</u>	
	B.4 Restore HPI discharge crossover valve(s) to OPERABLE status.	30 days from initial entry into Condition A

(continued)



License Amendment Request No. 2013-03  
June 30, 2014

**ATTACHMENT 4**

**RETYPE TECHNICAL SPECIFICATION BASES PAGES**  
**[6 pages follow this cover page]**

**NOTE:** This attachment contains the following retyped TS Bases pages:

B 3.5.2-7	B 3.7.4-2
B 3.5.2-8	B 3.7.4-3
B 3.5.2-9	
B 3.5.2-10	

BASES

---

APPLICABILITY  
(continued)

Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled."  
MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation – High Water Level," and LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation – Low Water Level."

---

ACTIONS

A.1 and A.2

With one HPI pump inoperable, or one or more HPI discharge crossover valve(s) (i.e., HP-409 and HP-410) inoperable, the HPI pump and discharge crossover valve(s) must be restored to OPERABLE status within 72 hours. The HPI System continues to be capable of mitigating an accident, barring a single failure. The 72 hour Completion Time is based on NRC recommendations (Ref. 4) that are based on a risk evaluation and is a reasonable time for many repairs.

In the event HPI pump "C" becomes inoperable, Condition C must be entered as well as Condition A. Until actions are taken to align an HPI pump to HPI train "B," HPI train "B" is inoperable due to the inability to automatically provide injection in response to an ESPS signal.

This Condition permits multiple components of the HPI System to be inoperable concurrently. When this occurs, other Conditions may also apply. For example, if HPI pump "C" and HP-409 are inoperable coincidentally, HPI train "B" is incapable of being automatically actuated or manually aligned from the Control Room. Thus, Required Action C.1 would apply.

In order to utilize another HPI pump to supply HPI train "B" when HPI pump "C" is inoperable, HP-116 must be opened. This action results in cross-connecting the HPI discharge headers; thus, Condition E must be entered. HP-115 may be closed to provide hydraulic separation provided that pump minimum flow requirements are maintained. However, two operating pumps would be required for this configuration, one to provide makeup flow and one to provide seal injection flow.

B.1, B.2, B.3, and B.4

If the Required Action and associated Completion Time of Condition A is not met, THERMAL POWER of the unit must be reduced to  $\leq 50\%$  RTP within 12 hours. The 12 hour Completion Time is reasonable, based on operating experience, to reach the required unit condition from full power conditions in an orderly manner and without challenging unit systems. This time is less restrictive than the Completion Time for Required Action C.1,

BASES

---

ACTIONS

B.1, B.2, B.3, and B.4 (continued)

because the HPI System remains capable of performing its function, barring a single failure.

Two HPI trains are required to mitigate specific small break LOCAs, if no credit for enhanced steam generator cooling is assumed in the accident analysis. However, if equipment not qualified as QA-1 (i.e., an atmospheric dump valve (ADV) flow path for a steam generator) is credited for enhanced steam generator cooling, the safety analyses have determined that the capacity of one HPI train is sufficient to mitigate a small break LOCA on the discharge of the reactor coolant pumps if reactor power is  $\leq 50\%$  RTP.

Required Actions B.2, B.3, and B.4 modify the HPI pump and discharge crossover valve OPERABILITY requirements to permit reduced requirements at power levels  $\leq 50\%$  RTP for an extended period of time. Required Action B.2 provides a compensatory measure to verify by administrative means that the ADV flow path for each steam generator is OPERABLE within 12 hours. This compensatory measure provides additional assurance regarding the ability of the plant to mitigate an accident. Compliance with this requirement can be established by ensuring that the ADV flow path for each steam generator is OPERABLE in accordance with LCO 3.7.4, "Atmospheric Dump Valve (ADV) Flow Paths."

Required Actions B.3 and B.4 require that the HPI pump and discharge crossover valve(s) be restored to OPERABLE status within 30 days from initial entry into Condition A. The 30-day time period limits the time that the plant can operate while relying on non QA-1 ADVs to provide enhanced steam generator cooling to mitigate small break LOCAs. The 30-day time period is acceptable, because:

1. Without crediting an ADV flow path, the HPI System remains capable of performing the safety function, barring a single failure;
2. If credit is taken for an ADV flow path for a steam generator, the safety analysis has demonstrated that only one HPI train is required to mitigate the consequences of a small break LOCA when THERMAL POWER is  $\leq 50\%$  RTP. Thus, for this case, the HPI System would be capable of performing its safety function even with an additional single failure;

BASES

---

ACTIONS

B.1, B.2, B.3, and B.4 (continued)

3. OPERABILITY of the ADV flow path for each steam generator is required to be confirmed by Required Action B.2 within 12 hours. Additional defense-in-depth is provided, because the ADV flow path for only one steam generator is required to mitigate the small break LOCA; and
4. A risk-informed assessment (Ref. 7) concluded that operating the plant in accordance with these Required Actions is acceptable.

ACTIONS

C.1, C.2, and C.3

If the plant is operating with THERMAL POWER > 50% RTP, two HPI pumps capable of providing flow through two HPI trains are required. One HPI train is required to provide flow automatically upon receipt of an ESPS signal, while flow through the other HPI train must be capable of being established from the Control Room within 10 minutes. Thus, if the plant is operating at > 50% RTP, and one HPI train is inoperable and incapable of being automatically actuated or manually aligned from the Control Room to provide flow post-accident, the HPI System would be incapable of performing its safety function. For this Condition, Required Action C.1 requires the power to be reduced to  $\leq 50\%$  RTP within 3 hours. Required Action C.1 is modified by a Note which limits its applicability to the condition defined above. The 3 hour Completion Time is considered reasonable to reduce the unit from full power conditions to  $\leq 50\%$  RTP in an orderly manner and without challenging unit systems. The time frame is more restrictive than the Completion Time provided in Required Action B.1 for the same action, because the condition involves a loss of safety function.

If the plant is operating with THERMAL POWER > 50% RTP and the inoperable HPI train can be automatically actuated or manually aligned to provide flow post-accident, Required Action C.3 permits 72 hours to restore the HPI train to an OPERABLE status.

If enhanced steam generator cooling is not credited in the accident analysis, two HPI trains are required to mitigate specific small break LOCAs with THERMAL POWER  $\leq 50\%$  RTP. However, if equipment not qualified as QA-1 (i.e., an ADV flow path for a steam generator) is credited for enhanced steam generator cooling, the safety analyses have determined that the capacity of one HPI train is sufficient to mitigate a small break LOCA on the discharge of the reactor coolant pumps if THERMAL POWER is  $\leq 50\%$  RTP. In order to permit an HPI train to be inoperable regardless of the reason when THERMAL POWER is  $\leq 50\%$  RTP, Required Action C.2 provides a compensatory measure to verify by administrative means that the ADV flow path for each steam generator is



BASES

---

ACTIONS

C.1, C.2, and C.3 (continued)

OPERABLE within 3 hours. This Required Action is modified by a Note which states that it is only required if THERMAL POWER is  $\leq 50\%$  RTP. This compensatory measure provides assurance regarding the ability of the plant to mitigate an accident while in the Condition and THERMAL POWER  $\leq 50\%$  RTP. Compliance with this requirement can be established by ensuring that the ADV flow path for each steam generator is OPERABLE in accordance with LCO 3.7.4, "Atmospheric Dump Valve (ADV) Flow Paths."

With one HPI train inoperable, the inoperable HPI train must be restored to OPERABLE status within 72 hours. This action is appropriate because:

1. With THERMAL POWER  $\leq 50\%$  RTP, the safety analysis demonstrates that only one HPI train is required to mitigate the consequences of a small break LOCA assuming credit is taken for the ADV flow path for one steam generator. The OPERABILITY of the ADV flow path for each steam generator is confirmed by Required Action C.2 within 3 hours. This provides additional defense-in-depth. Additionally, a risk-informed assessment (Ref. 7) concluded that operating the plant in accordance with this Required Action is acceptable.
2. With THERMAL POWER  $> 50\%$  RTP, the remaining OPERABLE HPI train is capable of automatic actuation, and the inoperable train can be manually aligned by operator action to cross-connect the discharge headers of the HPI trains. This manual action was approved by the NRC in Reference 6.

D.1

With the HPI suction headers not cross-connected, the HPI suction headers must be cross-connected within 72 hours. The HPI System continues to be capable of mitigating an accident, barring a single failure. The 72 hour Completion Time is based on NRC recommendations (Ref. 4) that are based on a risk evaluation and is a reasonable time for many repairs.

An argument similar to that utilized for Required Actions B.2, B.3, and B.4 could have been made for operating the HPI System with the suction headers not cross-connected for an extended period of time. However, this action was not considered prudent, due to the potential of damaging two HPI pumps in the event HP-24 or HP-25 failed to open in response to an ESPS signal while the HPI suction headers were not cross-connected.

BASES (continued)

---

APPLICABLE  
SAFETY ANALYSIS

The SGTR analysis credits operator action to depressurize the steam generators by opening both ADV flow paths (i.e., the ADV flow path for each steam generator) within 40 minutes of identifying the ruptured steam generator. Within this 40-minute time period, the operators are only required to open the bypass valve, the block valve, and the throttle valve. However, later in the event, the analysis also assumes that the operators will open the isolation valves in each ADV flow path.

Operator action to depressurize a steam generator via its ADV flow path is credited in the analysis of certain small break LOCAs with THERMAL POWER  $\leq$  50% RTP and the plant operated with a degraded HPI System. This event credits operator action to open one ADV flow path within 25 minutes of an Engineered Safeguards Protective System (ESPS) actuation.

If enhanced steam generator cooling is not credited in the small break LOCA analysis, two HPI trains are required to mitigate specific small break LOCAs. However, if equipment not qualified as QA-1 (i.e., an ADV flow path for a steam generator) is credited for enhanced steam generator cooling, the safety analyses have determined that the capacity of one HPI train is sufficient to mitigate a small break LOCA on the discharge of the reactor coolant pumps if THERMAL POWER is  $\leq$  50% RTP.

The analysis for degraded HPI credits an ADV flow path for one steam generator as a compensatory measure in the event an HPI train is inoperable and THERMAL POWER is  $\leq$  50% RTP. During this situation, the ADV flow path for one steam generator is credited during certain small break LOCAs to depressurize the steam generator and enhance primary-to-secondary heat transfer. This is done in conjunction with the EFW System providing cooling water to the steam generator. The ADV flow path is comprised of manual valves. Operator action is credited for establishing the ADV flow path within 25 minutes of an ESPS signal.

Additionally, the ADV flow path for each steam generator is credited as a compensatory measure in TS 3.5.2, "High Pressure Injection (HPI)." Typically, single failures are not considered once the plant has entered a condition defined in the TS. However, the Completion Time permitted when the HPI system is degraded, is an extended period of time. In the event an accident occurred during this extended Completion Time and a single failure were to occur in the degraded HPI system, the ability of a plant to mitigate the consequences of specific small break LOCAs continues to be assured by the ADV flow path for one steam generator. The ADV flow paths satisfy Criterion 3 of 10 CFR 50.36 (Ref. 1).

The 50% partial-power SBLOCA analysis includes a sensitivity case that models an operator action to modulate the main steam pressure at 300 psig via the ADV during the secondary-side depressurization. The purpose of the ADV modulation to maintain steam pressure is to limit Reactor

BASES

---

APPLICABLE  
SAFETY ANALYSIS  
(continued)

Coolant System (RCS) depressurization, which then prevents the CFTs from completely discharging their liquid contents and introducing nitrogen gas into the RCS during the depressurization. The secondary-side pressure control to preclude significant nitrogen injection is consistent with the generic Emergency Operating Procedure (EOP) guidance for B&W plants provided by AREVA.

To ensure that the new SBLOCA analysis is bounding, the plant must be controlled to a main steam pressure that is less than the value assumed in the 50% partial-power SBLOCA analysis, since less borated water from the CFT would be injected at the higher analyzed value. This ensures that the 50% partial-power SBLOCA analysis remains conservative with respect to actual plant operation. The 50% partial-power SBLOCA analysis modeling the modulation of steam pressure at 300 psig allows operating space within the EOPs such that CFT isolation does not conflict with the applicable safety analysis in terms of isolating the borated water source from the CFTs.

A supplemental SBLOCA analysis demonstrates that long-term core cooling is assured with or without nitrogen gas intrusion for all break sizes. The operator actions required by the ONS licensing basis remain unchanged. The analyses show that nitrogen gas intrusion does not occur for the small break sizes that rely on steam generator heat removal for a number of hours. In the longer term, core cooling is still assured if the CFTs completely discharge their liquid contents much later because at these longer times following the reactor trip, the lower decay heat levels can be matched by HPI cooling.

Based on the evaluation of impacts to long-term core cooling if ADV modulation does not occur, the operator action modeled in the partial-power SBLOCA analysis to maintain steam generator pressure at 300 psig is considered to be a desired action, and not a required action needed to demonstrate post-LOCA long-term core cooling.

---

LCO

The ADV flow path for each steam generator is required to be OPERABLE. The failure to meet the LCO can result in the inability to depressurize the steam generators following a SGTR.

The ADV flow path for each steam generator is required to be OPERABLE. Failure to meet the LCO can result in the inability to depressurize a steam generator following a small break LOCA. This function is required to support operation with a degraded HPI System when THERMAL POWER is  $\leq$  50% RTP.

An ADV flow path is considered OPERABLE when it is capable of providing a controlled relief of the main steam flow, and each valve which comprises the ADV flow path is capable of opening and closing.

License Amendment Request No. 2013-03  
June 30, 2014

**ATTACHMENT 5**

**ACRONYM LIST**

**[1 page follows this cover page]**

**Acronym List**

ADV	Atmospheric Dump Valve
B&W	Babcock and Wilcox
CFT	Core Flood Tank
CLPD	Cold Leg Pump Discharge
CLPS	Cold Leg Pump Suction
DBA	Design Basis Accident
ECCS	Emergency Core Cooling System
EM	Evaluation Model
EOC	End of Cycle
EOP	Emergency Operating Procedures
EOT	End of Transient
ESFAS	Engineered Safety Features Actuation System
HL	Hot Leg
HPI	High Pressure Injection
LBLOCA	Large Break Loss of Coolant Accident
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LSCM	Loss of Subcooling Margin
LTCC	Long Term Core Cooling
MWt	Mega Watts thermal
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam System Supplier
OBD/N	Operable but Degraded/Non-conforming
ONS	Oconee Nuclear Station
PCT	Peak Cladding Temperature
PIP	Problem Investigation Program
PSC	Preliminary Safety Concern
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RTP	Rated Thermal Power
RV	Reactor Vessel
SBLOCA	Small Break Loss of Coolant Accident
SE	Safety Evaluation
SG	Steam Generator
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report