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ONS-2016-023 10 CFR 50.90

July 21, 2016

U.S. Nuclear Regulatory Commission
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Oconee Nuclear Station Unit Nos. 1, 2, and 3
Docket Nos. 50-269, 50-270 AND 50-287
Renewed License Nos. DPR-38, DPR-47 AND DPR-55

SUBJECT: Application to add the COPERNIC Fuel Performance Code to Oconee Nuclear Station Technical Specifications 2.1.1.1, "Reactor Core Safety Limits," and 5.6.5, "Core Operating Limits Report (COLR)," License Amendment Request No. 2015-07

REFERENCES:

1. BAW-10231P-A, *COPERNIC Fuel Rod Design Computer Code*, Revision 1, Framatome ANP, January 2004 (ML040150701).
2. BAW-10162P-A, *TACO3 - Fuel Pin Thermal Analysis Computer Code*, B&W Fuel Company, October 1989.
3. BAW-10184P-A, *GDTACO - Urania Gadolinia Fuel Pin Thermal Analysis Code*, B&W Fuel Company, February 1995.
4. DPC-NE-2008-PA, Rev. 2, Fuel Rod Mechanical Analysis Methodology Using TACO3 and GDTACO, August 2012.
5. Nuclear Regulatory Commission Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses," Supplement 1.

Pursuant to 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke Energy) proposes to amend Renewed Facility Operating License Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station (ONS), Units 1, 2, and 3. The proposed change revises Technical Specification (TS) 2.1.1.1 to include the COPERNIC fuel performance code for determining maximum fuel pin centerline temperature. ONS TS 2.1.1.1 currently provides similar information for other fuel performance computer codes. The proposed change also adds the COPERNIC topical report (Reference 1) to the list of topical reports in ONS TS 5.6.5.

Duke Energy currently uses the AREVA TACO3 (Reference 2) and GDTACO (Reference 3) fuel performance codes for Oconee fuel rod mechanical analyses using the methodology in DPC-NE-2008 (Reference 4). Per guidelines provided in Nuclear Regulatory Commission (NRC) Generic Letter (GL) 83-11, Supplement 1 (Reference 5), Duke Energy will transition to self-performing fuel rod mechanical analyses (i.e., fuel rod internal pressure, fuel melt, cladding strain, cladding corrosion, and input to the cladding stress, cladding fatigue, and cladding creep collapse analyses) using COPERNIC for ONS Units 1, 2, and 3. Duke Energy has many years

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of experience using fuel performance codes similar to COPERNIC. Duke Energy has self-performed fuel rod mechanical analyses for ONS using TACO3 since 1995 and has used the Westinghouse PAD code to perform fuel rod mechanical analyses for McGuire and Catawba since 1999. In February 2015, AREVA provided formal training to Duke Energy personnel on the COPERNIC code and methodology. AREVA provides quality assurance and change control for the COPERNIC code. COPERNIC is installed and controlled in accordance with Duke Energy software quality assurance procedures. Duke Energy's software quality assurance program is in compliance with 10 CFR 50, Appendix B requirements. Documentation related to the change to COPERNIC will be available for NRC audit.

Duke Energy plans to implement this change beginning with Oconee Unit 2 reload cycle 29 (Fall 2017).

The Enclosure to this submittal provides an evaluation of the proposed changes including the technical evaluation, regulatory evaluation and environmental considerations. Attachment 1 provides markup pages of existing TSs and TS Bases (for information only) to highlight the proposed change. Attachment 2 provides revised (clean) TS pages.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that the proposed changes involve no significant hazards consideration. The bases for these determinations are included in an enclosure to this submittal.

There are no new regulatory commitments contained in this letter. Staff approval of this license amendment application is requested within one year of the date of this submittal. Duke Energy is requesting a 120-day implementation period.

In accordance with 10 CFR 50.91, Duke Energy is notifying the State of South Carolina of this license amendment request by transmitting a copy of this letter and enclosure/attachments to the designated State Official.

Should you have any questions concerning this letter, or require additional information, please contact Stephen C. Newman, Lead Nuclear Engineer, Oconee Regulatory Affairs Group at (864) 873-4388.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 21, 2016.

Sincerely,

Scott L. Batson
Vice President
Oconee Nuclear Station

Enclosure: Evaluation of Proposed Change
Attachment 1: Markup of existing TSs and TS Bases (for information only)
Attachment 2: Revised (clean) TS pages

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Enclosure

Evaluation of the Proposed Change

Subject: Application to add the COPERNIC Fuel Performance Code to Oconee Nuclear Station Technical Specification (TS) 2.1.1.1, "Reactor Core Safety Limits," and TS 5.6.5, "Core Operating Limits Report."

1. SUMMARY DESCRIPTION

This evaluation supports a request to amend the Operating Licenses DPR-38, DPR-47 and DPR-55 for Oconee Nuclear Station (ONS), Unit Nos. 1, 2, and 3.

The maximum local fuel pin centerline temperature is given by the relationships defined by Technical Specification (TS) 2.1.1.1 for the respective fuel designs and is dependent upon which computer code is used in the analysis. The two equations currently given in TS 2.1.1.1 are applicable to the TACO3 and GDTACO fuel performance codes. Duke Energy proposes to add the maximum local fuel pin centerline temperature equation utilized in the COPERNIC fuel performance code to ONS TS 2.1.1.1. The COPERNIC code has previously been reviewed and approved for use by the Nuclear Regulatory Commission (NRC).

Additionally, Duke Energy proposes to add the COPERNIC topical report to the list of topical reports contained in ONS TS 5.6.5, "Core Operating Limits Report (COLR)."

2. DETAILED DESCRIPTION

2.1 Technical Specification Change Description

Duke Energy currently uses the TACO3 (Reference 1) and GDTACO (Reference 2) fuel performance codes to perform fuel rod mechanical analyses using the NRC-approved methodology in DPC-NE-2008 (Reference 9). In order to utilize the COPERNIC code, Duke Energy is proposing changes to ONS TS 2.1.1.1, Reactor Core Safety Limits. The proposed changes add a provision to TS 2.1.1.1 for the determination of the maximum local fuel pin centerline temperature using the NRC reviewed and approved COPERNIC fuel performance code. Specifically, the existing Oconee TS 2.1.1.1 states the following:

2.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, for UO_2 fuel, the maximum local fuel pin centerline temperature shall be $\leq 4656 - (5.8 \times 10^{-3} \times (\text{Burnup, MWD/MTU})) - 709.04|\text{chi}| - 786.62(\text{chi})^2 + 1087.07(\text{chi})^3$ °F where chi is the quantity oxygen-to-uranium ratio minus 2.0. For gadolinia fuel, the local fuel pin centerline temperature shall be $\leq 4656 - (6.5 \times 10^{-3} \times (\text{Burnup, MWD/MTU}))$ °F. Operation within these limits is ensured by compliance with the Axial Power Imbalance Protective Limits as specified in the Core Operating Limits Report.

The proposed new wording of TS 2.1.1.1 (added or changed text has been underlined) is:

2.1 Reactor Core SLs

2.1.1.1 ----------------------------------NOTE----------------------------------
<u>Following transition to the COPERNIC Fuel Performance Methodology the TACO and GDTACO Fuel Performance Methodologies are not applicable.</u>

In MODES 1 and 2, for UO_2 fuel, the maximum local fuel pin centerline temperature for TACO3 applications shall be ≤ 4656 - (5.8×10^{-3} x (Burnup, MWD/MTU)) - 709.04|chi| - 786.62$(chi)^2$ +1087.07$(chi)^3$ °F where chi is the quantity oxygen-to-uranium ratio minus 2.0. For Gadolinia fuel, the local fuel pin centerline temperature for GDTACO applications shall be ≤ 4656 – (6.5 x 10^{-3} x (Burnup, MWD/MTU)) °F. For COPERNIC applications, the maximum local fuel pin centerline temperature shall be ≤ 4901 - (1.37×10^{-3} x (Burnup, MWD/MTU)) °F. Operation within these limits is ensured by compliance with the Axial Power Imbalance Protective Limits as specified in the Core Operating Limits Report.

As noted above, Safety Limit 2.1.1.1 is modified by a NOTE indicating that following transition to the COPERNIC Fuel Performance Methodology both the TACO and GDTACO Fuel Performance Methodologies are not applicable. In addition, Duke Energy is proposing to add the COPERNIC topical report to the list of topical reports contained in ONS TS 5.6.5.

2.2 Reason for Change

In October 2009, the NRC issued Information Notice (IN) 2009-23 that discusses the impact of irradiation on fuel thermal conductivity. The IN states that "it is well understood that irradiation damage and the progressive buildup of fission products in fuel pellets result in reduced thermal conductivity of the pellets." Thermal performance codes approved by the NRC before 1999 do not include this reduction in thermal conductivity with increasing radiation because early test reactor data was inconclusive as to the significance of the effect.

TACO3 and GDTACO were approved prior to 1999, and in late 2009 AREVA issued a 10 CFR Part 21 Deviation Determination which concluded that the fuel performance codes do not adequately account for fuel thermal conductivity degradation (TCD) with burnup. It was determined the codes may under predict fuel temperatures as burnup increases and under prediction of fuel temperatures can lead to non-conservative transient clad strain (TCS) and centerline fuel melt (CFM) linear heat rate (LHR) limits. Subsequently, AREVA provided new CFM and TCS LHR limits for Duke Energy to use to evaluate the impact of the TCD issue on current operating limits. Based on the information provided by AREVA, Duke Energy determined that there was sufficient margin in the power imbalance limits for the operating cores such that the power imbalance limits remained conservative and no change in plant limits was required.

The NRC approved COPERNIC fuel performance code (Reference 3) directly accounts for TCD with burnup. To directly account for TCD in fuel rod mechanical analyses, Duke Energy proposes to transition to the COPERNIC fuel performance code and use the NRC approved methodology in Chapter 12 of Reference 3.

3. TECHNICAL EVALUATION

3.1 COPERNIC Code

In References 5 and 6, the NRC staff concluded that the use of the COPERNIC code was acceptable for referencing in licensing applications, to the extent specified and under the limitations delineated in the associated topical report and the associated NRC Safety Evaluation. The associated topical report is specified as BAW-10231P. References 5 and 6 applied to Revision 0 of BAW-10231 specifically addressing the advanced M5 cladding material.

3.2 Maximum Local Fuel Pin Centerline Temperature Limit

The intent of the maximum local fuel pin centerline temperature limit is to prevent centerline temperature from reaching the melting point, which conservatively assures that there will be no breach in cladding integrity. The COPERNIC code was approved for use with M5 cladding material. M5 cladding is currently in use in the ONS fuel design.

The centerline melt limit, as presented in COPERNIC, decreases linearly with fuel burnup. The best-estimate limiting fuel melt temperature of new fuel is 4901°F based on the COPERNIC code. The limiting fuel melt temperature is adjusted downward from this temperature depending on the amount of burnup. The downward adjustment is 13.7°F per 10,000 MWD/MTU of burnup.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.36

The license amendment request includes proposed changes to the TSs, the contents of which are controlled by requirements in 10 CFR 50.36, "Technical Specifications." TSs are required to include items in the following five categories related to plant operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls.

4.2 Precedent

The NRC has previously approved changes similar to the proposed changes in this License Amendment Request for the following nuclear power plants:

1. Sequoyah Nuclear Plant, Units 1 and 2: Application dated June 17, 2011 (ADAMS Accession No. ML11172A071); NRC Safety Evaluation dated September 26, 2012, (ADAMS Accession No. ML12249A394) (Reference 7).

   Sequoyah Nuclear Plant, Units 1 and 2 filed an application for a Technical Specification change to allow the use of AREVA's Advanced W17 High Thermal Performance Fuel. As part of that application was a change to the Reactor Core Safety Limit associated with the local fuel pin centerline temperature. The change was based on the use of the COPERNIC computer code. The change proposed in the Sequoyah Safety Limit is similar to the change proposed in this request.

2. Arkansas Nuclear One, Unit 1: Application dated June 11, 2013 (ADAMS Accession No. ML13162A736); Supplement dated December 11, 2013 (ADAMS Accession No. ML13347B236); NRC Safety Evaluation dated July 9, 2014 (ADAMS Accession No. ML14169A475) (Reference 10).

Arkansas Nuclear One, Unit 1 filed an application for a Technical Specification change to add a provision for the determination of the maximum local fuel pin centerline temperature using the NRC reviewed and approved COPERNIC fuel performance code. The proposed change to Arkansas Nuclear One, Unit 1 Technical Specification 2.1.1.1, which was approved by the NRC, is similar to the change proposed in this request.

4.3 No Significant Hazards Consideration Determination

Duke Energy is requesting an amendment to the Technical Specifications (TS) for Oconee Nuclear Station (ONS), Units 1, 2, and 3. Specifically, Duke Energy is requesting the Nuclear Regulatory Commission's (NRC) approval to revise TS Sections 2.1.1.1, "Reactor Core Safety Limits (SLs)" and 5.6.5, "Core Operating Limits Report (COLR)." Duke Energy proposes to add the maximum local fuel pin centerline temperature equation utilized in a NRC reviewed and approved fuel performance code to ONS TS 2.1.1.1. Additionally, the topical report for the NRC reviewed and approved fuel performance code is proposed to be added to the list of topical reports in ONS TS 5.6.5.

Duke Energy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

   Response: No.

   The proposed change adds a limit on maximum local fuel pin centerline temperature to ONS Technical Specifications that is based on a NRC reviewed and approved fuel performance code, and does not require a physical change to plant systems, structures or components. Plant operations and analysis will continue to be in accordance with the ONS licensing basis. The peak fuel centerline temperature is the basis for protecting the fuel and is consistent with the safety analysis.

   The proposed change also adds a topical report for a NRC reviewed and approved fuel performance code to the list of topical reports in ONS Technical Specifications, which is administrative in nature and has no impact on a plant configuration or system performance relied upon to mitigate the consequences of an accident. The list of topical reports in the Technical Specifications used to develop the core operating limits does not impact either the initiation of an accident or the mitigation of its consequences.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed change adds a limit on maximum local fuel pin centerline temperature to ONS Technical Specifications that is based on a NRC reviewed and approved fuel performance code, and does not require a physical changes to plant systems, structures or components. Specifying maximum local fuel pin centerline temperature ensures that the fuel design limits are met. Operations and analysis will continue to be in compliance with NRC regulations. The addition of a new fuel pin centerline melt temperature versus burnup relationship does not affect any accident initiators that would create a new accident.

The proposed change also adds a topical report for a NRC reviewed and approved fuel performance code to the list of topical reports in ONS Technical Specifications, which is administrative in nature and has no impact on a plant configuration or on system performance. The proposed change updates the list of NRC-approved topical reports used to develop the core operating limits. There is no change to the parameters within which the plant is normally operated. The possibility of a new or different kind of accident is not created.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change to TS 2.1.1.1 adds a limit on maximum local fuel pin centerline temperature that is based on an NRC reviewed and approved fuel performance code, and does not required a physical change to plant systems, structures or components. Plant operations and analysis will continue to be in accordance with ONS licensing basis.

Adding the local fuel pin centerline temperature and burnup relationship defined by the NRC reviewed and approved fuel performance code to the ONS Technical Specifications, does not impact the safety margins established in the ONS licensing basis.

The proposed change also adds a topical report for a NRC reviewed and approved fuel performance code to the list of topical reports in ONS Technical Specifications, which is administrative in nature and does not amend the cycle specific parameters presently required by the Technical Specifications. The individual Technical Specifications continue to require operation of the plant within the bounds of the limits specified in the Core Operating Limits Report. The proposed change to the list of analytical methods referenced in the Core Operating Limits Report does not impact the margin of safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy concludes that the proposed change 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.

4.4 <u>Conclusion</u>

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

5. ENVIRONMENTAL CONSIDERATION

The proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, and would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of an effluent that may be released offsite, and (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6. REFERENCES

1. BAW-10162P-A, *TACO3 - Fuel Pin Thermal Analysis Computer Code*, B&W Fuel Company, October 1989.
2. BAW-10184P-A, *GDTACO - Urania Gadolinia Fuel Pin Thermal Analysis Code*, B&W Fuel Company, February 1995.
3. BAW-10231P-A (Revision 1), *COPERNIC Fuel Rod Design Computer Code*, FRAMATOME ANP, January 2004.
4. NRC letter, *Final Safety Evaluation for Topical Report BAW-10231P, "COPERNIC Fuel Rod Design Code" Chapter 13, MOX Applications (TAC No. MB7547)*, January 14, 2004 (ADAMS Accession No. ML040150701).
5. NRC letter, *FRAMATOME ANP Topical Report BAW-10231, "COPERNIC Fuel Rod Design Computer Code" - Correction of Error in Safety Evaluation (TAC No. MA6792)*, June 14, 2002 (ADAMS Accession No. ML021360461).
6. NRC letter, *Safety Evaluation (SE) of FRAMATOME ANP Topical Report BAW-10231P, "COPERNIC Fuel Rod Design Computer Code" (TAC No. MA6792)*, April 18, 2002 (ADAMS Accession No. ML020070158).
7. NRC letter, *Sequoyah Nuclear Plant, Units 1 and 2 - Issuance of Amendments to Revise the Technical Specification to Allow Use of AREVA Advanced W17 High Thermal Performance Fuel (TS-SQN-2011-07) (TAC NOS. ME6538 and ME6539)*, September 26, 2012 (ADAMS Accession No. ML12249A394).
8. NRC Staff Requirements Memorandum, *SECY-92-223 - Resolution of Deviations Identified During the Systematic Evaluation Program*, September 18, 1992 (ADAMS Accession No. ML003763736).
9. DPC-NE-2008-PA, Rev. 2, Fuel Mechanical Reload Analysis Methodology Using TACO3 and GDTACO, August 2012
10. NRC letter, *Arkansas Nuclear One, Unit 1 - Issuance of Amendment RE: Revision to Technical Specification 2.1.1.1, Reactor Core Safety Limits (TAC No. MF2277)*, July 9, 2014 (ADAMS Accession No. ML14169A475).

ATTACHMENT 1

Markup of existing TSs and TS Bases (for information only)

- TS page 2.0-1
- TS page 5.0-27
- TS Bases 2.1.1 (Info only)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 ---NOTE---
Following transition to the COPERNIC Fuel Performance Methodology the TACO and GDTACO Fuel Performance Methodologies are not applicable.

In MODES 1 and 2, for UO_2 fuel, the maximum local fuel pin centerline temperature for TACO applications shall be $\leq 4656 - (5.8 \times 10^{-3} \times (\text{Burnup, MWD/MTU})) - 709.04|\text{chi}| - 786.62(\text{chi})^2 + 1087.07(\text{chi})^3$ °F where chi is the quantity oxygen-to-uranium ratio minus 2.0. For G~~g~~adolinia fuel, the local fuel pin centerline temperature for GDTACO applications shall be $\leq 4656 - (6.5 \times 10^{-3} \times (\text{Burnup, MWD/MTU}))$ °F. For COPERNIC applications the maximum local fuel pin centerline temperature shall be $\leq 4901 - (1.37 \times 10^{-3} \times (\text{Burnup, MWD/MTU}))$ °F. Operation within these limits is ensured by compliance with the Axial Power Imbalance Protective Limits as specified in the Core Operating Limits Report.

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limit of 1.18 for the BWC correlation, 1.19 for the BWU correlation, and 1.132 for the BHTP correlation. Operation within these limits is ensured by compliance with the Axial Power Imbalance Protective Limits and RCS Variable Low Pressure Protective Limits as specified in the Core Operating Limits Report.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2750 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour.

2.2.2 In MODE 1 or 2, if SL 2.1.2 is violated, restore compliance within limits and be in MODE 3 within 1 hour.

2.2.3 In MODES 3, 4, and 5, if SL 2.1.2 is violated, restore RCS pressure to ≤ 2750 psig within 5 minutes.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

(7) DPC-NE-3000-P-A, Thermal Hydraulic Transient Analysis Methodology;

(8) DPC-NE-2005-P-A, Thermal Hydraulic Statistical Core Design Methodology;

(9) DPC-NE-3005-P-A, UFSAR Chapter 15 Transient Analysis Methodology :

(10) BAW-10227-P-A, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel;

(11) BAW-10164P-A, RELAP 5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and non-LOCA Transient Analysis; and

(12) DPC-NE-1006-P-A, Oconee Nuclear Design Methodology Using CASMO-4/SIMULATE-3 (Revision 0, May 2009).

(13) BAW-10231P-A, COPERNIC Fuel Rod Design Computer Code, FRAMATOME ANP, January 2004.

The COLR will contain the complete identification for each of the Technical Specifications referenced topical reports used to prepare the COLR (i.e., report number, title, revision number, report date or NRC SER date, and any supplements).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) and Main Feeder Bus Monitor Panel (MFPMP) Report

When a report is required by Condition B or G of LCO 3.3.8, "Post Accident Monitoring (PAM) Instrumentation" or Condition D of LCO 3.3.23, "Main Feeder Bus Monitor Panel," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring (PAM only), the cause of the inoperability, and the plans and schedule for restoring the

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

ONS Design Criteria (Ref. 1) require that reactor core SLs ensure specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated transients. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that the fuel centerline temperature stays below the melting temperature.

DNB is not a directly measurable parameter during operation, but neutron power and Reactor Coolant System (RCS) temperature, flow and pressure can be related to DNB using a critical heat flux (CHF) correlation. The BWC (Ref. 2), the BWU (Ref. 4), and the BHTP (Ref. 5) CHF correlations have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The BWC correlation applies to Mark-BZ fuel. The BWU correlation applies to the Mark-B11 fuel. The BHTP correlation applies to the MARK-B-HTP fuel. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.18 (BWC), 1.19 (BWU) and 1.132 (BHTP).

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. <u>The maximum allowable fuel centerline temperatures are given by the relationships defined in Reactor Core SL 2.1.1.1 and are dependent on whether TACO3, GDTACO, or COPERNIC analyses are</u>

BASES

BACKGROUND (continued)

utilized. For TACO3 applications, t~~T~~he dependency of the fuel melt temperature on the as-built oxygen-to-uranium ratio for UO_2 fuel pins is provided by the fuel vendor. For gadolinia fuel pins, there is no dependence on the oxygen-to-uranium ratio.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding-water (zirconium-water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and main steam relief valves (MSRVs) prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and anticipated transients. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience fuel centerline melting.

The RPS setpoints (Ref. 3), in combination with all the LCOs, are designed to prevent any analyzed combination of transient conditions for RCS temperature, flow and pressure, and THERMAL POWER level that would result in a DNB ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following:

a. RCS High Pressure trip;

b. RCS Low Pressure trip;

c. Nuclear Overpower trip;

d. RCS Variable Low Pressure trip;

e. Reactor Coolant Pump to Power trip;

BASES

APPLICABLE SAFETY ANALYSIS (continued)

f. Flux/Flow Imbalance trip;

g. High Core Outlet Temperature trip; and

h. MSRVs.

The SL represents a design requirement for establishing the RPS trip setpoints identified previously.

SAFETY LIMITS

SL 2.1.1.1 and SL 2.1.1.2 ensure that fuel centerline temperature stays below the melting point and that the minimum DNBR is not less than the safety analyses limit. Safety Limit 2.1.1.1 is modified by a NOTE indicating that following transition to the COPERNIC Fuel Performance Methodology both the TACO and GDTACO Fuel Performance Methodologies are not applicable.

The SLs are preserved by monitoring process variables, AXIAL POWER IMBALANCE and Variable Low RCS Pressure, to ensure that the core operates within the fuel design criteria. AXIAL POWER IMBALANCE and Variable Low RCS Pressure protective limits are provided in the COLR. The trip setpoints are derived by adjusting the measurement system independent AXIAL POWER IMBALANCE and Variable Low RCS Pressure protective limits given in the COLR to allow for measurement system observability and instrumentation errors.

Operation within these limits is ensured by compliance with the AXIAL POWER IMBALANCE and Variable Low RCS Pressure protective limits preserved by their corresponding RPS setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," as specified in the COLR. The AXIAL POWER IMBALANCE protective limits are separate and distinct from the AXIAL POWER IMBALANCE operating limits defined by LCO 3.2.2, "AXIAL POWER IMBALANCE Operating Limits." The AXIAL POWER IMBALANCE operating limits in LCO 3.2.2, also specified in the COLR, preserve initial conditions of the safety analyses but are not reactor core SLs.

APPLICABILITY

SL 2.1.1.1 and SL 2.1.1.2 only apply in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The MSRVs, or automatic protection actions, serve to prevent RCS heatup to reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1.
In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER.

BASES (continued)

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the requirement to go to MODE 3 places the unit in a MODE in which these SLs are not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where these SLs are not applicable and reduces the probability of fuel damage.

REFERENCES

1. UFSAR, Section 3.1.
2. BAW-10143P-A, "BWC Correlation of Critical Heat Flux," April 1995.
3. UFSAR, Chapter 15.
4. BAW-10199P, "The BWU Critical Heat Flux Correlations," Addendum 1, April 2000
5. BAW-10241(P)(A), Revision 1, BHTP DNB Correlation Applied with LYNXT, Framatome ANP, July 2005.

ATTACHMENT 2

Revised (clean) TS pages

- TS page 2.0-1
- TS page 5.0-27

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 ----------NOTE----------
Following transition to the COPERNIC Fuel Performance Methodology the TACO and GDTACO Fuel Performance Methodologies are not applicable.

In MODES 1 and 2, for UO_2 fuel, the maximum local fuel pin centerline temperature for TACO applications shall be $\leq$ 4656 - (5.8 x 10^{-3} x (Burnup, MWD/MTU)) – 709.04|chi| - 786.62(chi)^2 + 1087.07(chi)^3 °F where chi is the quantity oxygen-to-uranium ratio minus 2.0. For Gadolinia fuel, the local fuel pin centerline temperature for GDTACO applications shall be $\leq$ 4656 – (6.5 x 10^{-3} x (Burnup, MWD/MTU)) °F. For COPERNIC applications the maximum local fuel pin centerline temperature shall be $\leq$ 4901 – (1.37 x 10^{-3} x (Burnup, MWD/MTU)) °F. Operation within these limits is ensured by compliance with the Axial Power Imbalance Protective Limits as specified in the Core Operating Limits Report.

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limit of 1.18 for the BWC correlation, 1.19 for the BWU correlation, and 1.132 for the BHTP correlation. Operation within these limits is ensured by compliance with the Axial Power Imbalance Protective Limits and RCS Variable Low Pressure Protective Limits as specified in the Core Operating Limits Report.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained $\leq$ 2750 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour.

2.2.2 In MODE 1 or 2, if SL 2.1.2 is violated, restore compliance within limits and be in MODE 3 within 1 hour.

2.2.3 In MODES 3, 4, and 5, if SL 2.1.2 is violated, restore RCS pressure to $\leq$ 2750 psig within 5 minutes.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

(7) DPC-NE-3000-P-A, Thermal Hydraulic Transient Analysis Methodology;

(8) DPC-NE-2005-P-A, Thermal Hydraulic Statistical Core Design Methodology;

(9) DPC-NE-3005-P-A, UFSAR Chapter 15 Transient Analysis Methodology :

(10) BAW-10227-P-A, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel;

(11) BAW-10164P-A, RELAP 5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and non-LOCA Transient Analysis; and

(12) DPC-NE-1006-P-A, Oconee Nuclear Design Methodology Using CASMO-4/SIMULATE-3 (Revision 0, May 2009).

(13) BAW-10231P-A, COPERNIC Fuel Rod Design Computer Code, FRAMATOME ANP, January 2004.

The COLR will contain the complete identification for each of the Technical Specifications referenced topical reports used to prepare the COLR (i.e., report number, title, revision number, report date or NRC SER date, and any supplements).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) and Main Feeder Bus Monitor Panel (MFPMP) Report

When a report is required by Condition B or G of LCO 3.3.8, "Post Accident Monitoring (PAM) Instrumentation" or Condition D of LCO 3.3.23, "Main Feeder Bus Monitor Panel," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring (PAM only), the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.