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October 22 2007

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

Subject: Duke Power Company LLC d/b/a Duke Energy Carolinas, LLC
Oconee Nuclear Site, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
Proposed License Amendment Request to Revise the Technical Specifications for AREVA NP Mark-B-HTP Fuel and for Methodology Report DPC-NE-2015-P "Mark-B-HTP Fuel Transition Methodology"
License Amendment Request No. 2007-12

Duke Power Company LLC d/b/a Duke Energy Carolinas, LLC (Duke) hereby submits a license amendment request (LAR) for the Oconee Nuclear Station Renewed Facility Operating License (FOL) and Technical Specifications (TS) pursuant to 10 CFR 50.90. Specifically, Duke requests NRC review and approval of methodology report DPC-NE-2015-P, "Mark-B-HTP Fuel Transition Methodology" and revisions to Technical Specifications 2.1.1.2 and 5.6.5.b. Associated revisions to associated Technical Specification Bases B.2.1.1 and B.3.4.1 are provided. These revisions will allow the use of the AREVA NP Mark-B-HTP fuel design at the Oconee Nuclear Station beginning with Oconee Unit 2 Cycle 24 in December 2008. The Mark-B-HTP design is currently in use at several B&W design reactors.

To support this new fuel design revision to the Oconee Technical Specifications and to Duke's NRC-approved methodology reports for reload design and non-LOCA safety analyses, NRC review and approval is required. Methodology report DPC-NE-2015-P, "Mark-B-HTP Fuel Transition Methodology," describes the methodology revisions and associated technical justification. Other revisions are also included to enhance the existing methodologies, to delete superseded content, to correct errors, and for editorial clarification. Revisions to the following

The attachments to this letter contain sensitive information
Withhold From Public Disclosure Under 10 CFR 2.390(d)(1).
Upon removal of the attachments, this letter is uncontrolled.

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NRR

seven methodology reports are consolidated within the DPC-NE-2015-P report.

- NFS-1001A - Oconee Nuclear Station Reload Design Methodology (Revision 5)
- DPC-NE-1002-A - Oconee Nuclear Station Reload Design Methodology II (Revision 2)
- DPC-NE-2003P-A – Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01 (Revision 1)
- DPC-NE-2005P-A - Thermal-Hydraulic Statistical Core Design Methodology (Revision 3)
- DPC-NE-2008P-A – Fuel Mechanical Reload Analysis Methodology Using TACO3 (Revision 0)
- DPC-NE-3000-PA – Thermal-Hydraulic Transient Analysis Methodology (Revision 3)
- DPC-NE-3005-PA – UFSAR Chapter 15 Transient Analysis Methodology (Revision 2)

It is Duke's intent to publish approved versions of DPC-NE-2015-P and the above seven methodology reports following NRC approval of DPC-NE-2015-P. Duke requests that the NRC safety evaluation for the DPC-NE-2015-P methodology report indicate that the revisions to these seven reports have also been approved.

This report contains information that is proprietary to Duke and AREVA NP. In accordance with 10 CFR 2.390, Duke requests that this information be withheld from public disclosure. Affidavits are included (Enclosures 2 and 3) from each organization attesting to the proprietary nature of the information in the report. The specific information that is proprietary to each organization is identified in the report. A non-proprietary version of this report will be submitted by separate cover letter following approval.

Duke requests approval of this LAR by September 30, 2008 with the amendment to become effective commencing with Oconee Unit 2 Cycle 24.

In accordance with Duke administrative procedures and the Quality Assurance Program Topical Report, these proposed changes to the license have been reviewed and approved by the Plant Operations Review Committee and Nuclear Safety Review Board. Additionally, a copy of this license amendment request is being sent to the State of South Carolina in accordance with 10 CFR 50.91 requirements.

Inquiries on this proposed amendment request should be directed to Reene' Gambrell of the Oconee Regulatory Compliance Group at (864) 885-3364.

Sincerely,

Bruce Hamilton

B. H. Hamilton, Vice President
Oconee Nuclear Site

Enclosures:

1. Notarized Affidavit of B. H. Hamilton
2. Notarized Affidavit of T. C. Geer
3. Notarized Affidavit of Gayle F. Elliott
4. Evaluation of Proposed Change

Attachments:

1. Technical Specification and Technical Specifications Bases – Mark Up
2. Technical Specification and Technical Specifications Bases – Reprinted Pages
3. DPC-NE-2015-P – Mark-B-HTP Fuel Transition Methodology

bc w/enclosures and attachments:

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NSRB, EC05N
ELL, EC050
File - T.S. Working
ONS Document Management

ENCLOSURE 1

AFFIDAVIT OF B. H. HAMILTON

AFFIDAVIT

Bruce H. Hamilton, being duly sworn, states that he is Vice President, Oconee Nuclear Site, Duke Power Company LLC d/b/a Duke Energy Carolinas, LLC, that he is authorized on the part of said Company to sign and file with the U. S. Nuclear Regulatory Commission this revision to the Renewed Facility Operating License Nos. DPR-38, DPR-47, and DPR-55; and that all statements and matters set forth herein are true and correct to the best of his knowledge.

Bruce Hamilton

B. H. Hamilton, Vice President
Oconee Nuclear Site

Subscribed and sworn to before me this 22 day of October, 2007

Sheila A Smith
Notary Public

My Commission Expires:

6-12-2013
Date

SEAL

ENCLOSURE 2

AFFIDAVIT OF T. C. GEER

AFFIDAVIT OF THOMAS C. GEER

1. I am Vice President of Duke Energy Corporation, and as such have the responsibility of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear plant licensing and am authorized to apply for its withholding on behalf of Duke.

2. I am making this affidavit in conformance with the provisions of 10 CFR 2.390 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke's application for withholding which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b) (4) of 10 CFR 2.390, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

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

   (ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs, relative to a method of analysis that provides a competitive advantage to Duke.

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

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

   (v) The Duke proprietary information sought to be withheld in the submittal is that which is marked in the proprietary version of the Duke methodology report DPC-NE-2015-P, *Mark-B-HTP Fuel Transition Methodology*. This information enables Duke to:

Thomas C. Geer
T. C. Geer

(Continued)

(a) Support license amendment and Technical Specification revision request for its Oconee reactors.

(b) Perform nuclear design calculations on Oconee reactor cores.

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

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

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

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

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

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

Thomas C. Geer

(Continued)

T. C. Geer

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

Thomas C. Geer

T. C. Geer

Subscribed and sworn to me: October 5, 2007

Date

Freda K. Crump

Notary Public

My Commission Expires: August 17, 2011

FREDA K CRUMP
NOTARY
PUBLIC
LINCOLN COUNTY, NC

SEAL

ENCLOSURE 3

AFFIDAVIT OF GAYLE F. ELLIOTT

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in document DPC-NE-2015-P entitled, "Oconee Nuclear Station Mark-B-HTP Fuel Transition Methodology," dated September 2007 and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 7th

day of September, 2007.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/10
Reg. # 7079129

ENCLOSURE 4

EVALUATION OF PROPOSED CHANGE

1.0 DESCRIPTION

Duke Power Company LLC d/b/a Duke Energy Carolinas, LLC (Duke) requests an amendment to Oconee Nuclear Station Renewed Facility Operating License (FOL) and Technical Specifications (TS) pursuant to 10 CFR 50.90 to support the transition to the AREVA NP Mark-B-HTP fuel assembly design.. This License Amendment Request (LAR) proposes to change Technical Specification 2.1.1.2 and 5.6.5.b and associated bases.

Revisions to NRC-approved Duke methodology reports that are necessary to support this fuel design are provided in the attached DPC-NE-2015-P, Mark-B-HTP Fuel Transition Methodology. NRC review and approval of this methodology report is also requested.

2.0 PROPOSED TECHNICAL SPECIFICATION CHANGES

Technical Specification 2.1.1.2, Reactor Core Safety Limits, will be revised to add the BHTP critical heat flux correlation, the correlation that is applicable to the Mark-B-HTP fuel assembly design, and the 1.132 correlation limit. Technical Specification 5.6.5.b, Core Operating Limits Report (COLR), will be revised to add the AREVA NP topical report BAW- 10164P-A as a reference.

Associated revisions to Technical Specification Bases 2.1.1, 3.4.1, and 5.6.5 are included in this submittal.

3.0 DPC-NE-2015-P – MARK-B-HTP FUEL TRANSITION METHODOLOGY

Methodology report DPC-NE-2015-P, "Mark-B-HTP Fuel Transition Methodology," describes the methodology revisions and associated technical justification. Other revisions are included to enhance the existing methodologies, to delete superseded content, to correct errors, and to provide editorial clarification. Revisions to the current following seven methodology reports are consolidated in the DPC-NE-2015-P report:

- NFS-1001A - Oconee Nuclear Station Reload Design Methodology (Revision 5)
- DPC-NE-1002-A - Oconee Nuclear Station Reload Design Methodology II (Revision 2)
- DPC-NE-2003P-A – Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01 (Revision 1)

- DPC-NE-2005P-A - Thermal-Hydraulic Statistical Core Design Methodology (Revision 3)
- DPC-NE-2008P-A – Fuel Mechanical Reload Analysis Methodology Using TACO3 (Revision 0)
- DPC-NE-3000-PA – Thermal-Hydraulic Transient Analysis Methodology (Revision 3)
- DPC-NE-3005-PA – UFSAR Chapter 15 Transient Analysis Methodology (Revision 2)

The UFSAR will be updated to include the revised methodologies and the new analysis results following NRC approval of the revisions.

4.0 BACKGROUND

Duke has contracted AREVA NP to provide reload core fuel with Mark-B-HTP fuel design beginning with Oconee Unit 2 Cycle 24 in December 2008. This fuel design has been selected primarily for its improved resistance to cladding damage due to flow-induced-vibration. The introduction of the Mark-B-HTP design requires revision to many of the analytical methodologies that Duke employs in the reload design process. These methodologies include core physics, mechanical design, core thermal-hydraulics, and non-LOCA transient and accident analyses. The LOCA analyses are provided by AREVA NP. All of these methodologies have been previously reviewed and approved by the NRC. Duke has consolidated the methodology revisions to seven existing reports into one new methodology report, DPC-NE-2015-P, "Mark-B-HTP Fuel Transition Methodology." This report includes the technical justification for each revision. Some of the revisions are not associated with the change in fuel design, but are included to enhance and maintain the methodology reports. These revisions include improvements, error corrections, deletion of superseded content, and editorial clarification.

The revised methodologies result in minor revisions to the technical specifications and bases. Revisions to Technical Specification 2.1.1.2, Reactor Core Safety Limits, and the associated bases are due to the BHTP critical heat flux correlation that is applicable to the Mark-B-HTP design. Similarly, revisions to the Bases for Technical Specification 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, are also necessary to include the BHTP correlation. The reference for the BHTP correlation is BAW-10241(P)(A), Revision 1, BHTP DNB Correlation Applied with LYNXT, Framatome ANP, July 2005. A revision to Technical Specification 5.6.5.b, Core Operating Limits Report (COLR), is also necessary to add the AREVA NP 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." Revision 6 of this report included the BHTP correlation used for LOCA analysis, and was approved by the NRC in June 2007.

Duke will apply the methodology revisions described in DPC-NE-2015-P in the reload design and non-LOCA safety analyses for Oconee following the introduction of the AREVA NP Mark-B-HTP fuel design beginning with Oconee Unit 2 Cycle 24 in December 2008. AREVA NP will apply Revision 6 to BAW-10164P-A for the LOCA analyses of Mark-B-HTP fuel.

5.0 TECHNICAL ANALYSIS

Section 9.0 of DPC-NE-2015-P describes the Technical Specification and Technical Specification Bases changes and provides technical justifications for each change.

6.0 REGULATORY SAFETY ANALYSIS

6.1 No Significant Hazards Consideration

Pursuant to 10 CFR 50.91, Duke has made the determination that this amendment request does not involve a significant hazards consideration by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed revisions to the technical specifications and to Duke's NRC-approved methodology reports support the use of the AREVA NP Mark-B-HTP fuel design. The methodology will be approved by the NRC prior to plant operation with the new fuel. The proposed safety limit ensures that fuel integrity will be maintained during normal operations and anticipated operational transients. The core operating limits report will be developed in accordance with the approved methodology. The proposed safety limit value does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. There is no impact on the source term or pathways assumed in accidents previously assumed. No analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed safety limit value does not change the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered. The BHTP correlation is not an accident / event initiator. No new initiating events or transients result from the use of the BHTP correlation or the related safety limit change.

3) Involve a significant reduction in a margin of safety.

   The proposed safety limit value has been established in accordance with the methodology for the BHTP correlation to ensure that the applicable margin of safety is maintained (i. e. there is at least 95% probability at a 95% confidence level that the hot fuel rod does not experience DNB). The other reactor core safety limits will continue to be met by analyzing the reload using NRC approved methods and incorporation of resultant operating limits into the Core Operating Limits Report (COLR).

6.2 Applicable Regulatory Requirements/Criteria

The proposed change to the Technical Specifications is based on the forthcoming approval of this License Amendment Request which includes report DPC-NE-2015-P, "Mark-B-HTP Fuel Transition Methodology"

The use of the BHTP DNB correlation has previously been approved for Crystal River Unit 3, Davis Besse Nuclear Power Station, Unit 1, and Arkansas Nuclear One, Unit 1.

7.0 ENVIRONMENTAL CONSIDERATION

Duke has evaluated this license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. Duke has determined that this license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

ATTACHMENT 1

TECHNICAL SPECIFICATION AND TECHNICAL SPECIFICATION BASES – MARK-UP

TS 2.0-1
TS 5.0-26
TSB 2.1.1-1
TSB 2.1.1-4
TSB 3.4.1-1

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be ≤ 4642 - (5.8×10^{-3} x (Burnup, MWD/MTU))° F. Operation within this limit 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, ~~and~~ 1.19 for the BWU correlation, and 1.132 for the BHTP correlation. Operation within ~~this~~ 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 (continued)

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; and

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

(11) BAW-10164P-A RELAP5/MOD2-B&W - AN ADVANCED COMPUTER PROGRAM FOR LIGHT WATER REACTOR LOCA AND NON-LOCA TRANSIENT ANALYSIS

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.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

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), ~~and~~ 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 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), ~~and~~ 1.19 (BWU), and 1.132 (BHTP).

*The BHTP Correlation Applies to the Mark-B-HTP Fuel.

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.

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, 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.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated transients assume initial conditions within the normal steady state envelope. The limits placed on DNB related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The LCO for minimum RCS pressure is consistent with operation within the nominal operating envelope and is above that used as the initial pressure in the analyses. A pressure greater than the minimum specified will produce a higher minimum DNBR. A pressure lower than the minimum specified will cause the unit to approach the DNB limit.

The LCO for maximum RCS coolant loop average temperature is consistent with full power operation within the nominal operating envelope and is lower than the initial loop average temperature in the analyses. A loop average temperature lower than that specified will produce a higher minimum DNBR. A loop average temperature higher than that specified will cause the unit to approach the DNB limit.

The RCS flow rate is not expected to vary during operation with all pumps running. The LCO for the minimum RCS flow rate corresponds to that assumed for the DNBR analyses. A higher RCS flow rate will produce a higher DNBR. A lower RCS flow will cause the unit to approach the DNB limit.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criterion of ≥ 1.18 for BWC correlation, ≥ 1.19 for BWU correlation,* or an equally valid limit when the statistical DNBR limit is employed (SCD methodology). This is the acceptance limit for the RCS DNBR parameters.

* ≥ 1.132 FOR BHTP CORRELATION

ATTACHMENT 2

TECHNICAL SPECIFICATION AND TECHNICAL SPECIFICATION BASES – RETYPED PAGES

Remove	**Insert**
TS 2.0-1	TS 2.0-1
TS 5.0-26	TS 5.0-26
TS5.0-27	TS5.0-27
TSB 2.1.1-1	TSB 2.1.1-1
TSB 2.1.1-2	TSB 2.1.1-2
TSB 2.1.1-4	TSB 2.1.1-4
TSB 3.4.1-1	TSB 3.4.1-1

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be $\leq 4642 - (5.8 \times 10^{-3} \times (\text{Burnup, MWD/MTU}))°$ F. Operation within this limit 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.

5.6.5 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; and

(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 Analyses

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.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance

5.6 Reporting Requirements

Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.10, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG.

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.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the

BASES

BACKGROUND (continued)

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;

f. Flux/Flow Imbalance trip;

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.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated transients assume initial conditions within the normal steady state envelope. The limits placed on DNB related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The LCO for minimum RCS pressure is consistent with operation within the nominal operating envelope and is above that used as the initial pressure in the analyses. A pressure greater than the minimum specified will produce a higher minimum DNBR. A pressure lower than the minimum specified will cause the unit to approach the DNB limit.

The LCO for maximum RCS coolant loop average temperature is consistent with full power operation within the nominal operating envelope and is lower than the initial loop average temperature in the analyses. A loop average temperature lower than that specified will produce a higher minimum DNBR. A loop average temperature higher than that specified will cause the unit to approach the DNB limit.

The RCS flow rate is not expected to vary during operation with all pumps running. The LCO for the minimum RCS flow rate corresponds to that assumed for the DNBR analyses. A higher RCS flow rate will produce a higher DNBR. A lower RCS flow will cause the unit to approach the DNB limit.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criterion of ≥ 1.18 for BWC correlation, ≥ 1.19 for BWU correlation, ≥ 1.132 FOR BHTP correlation, or an equally valid limit when the statistical DNBR limit is employed (SCD methodology). This is the acceptance limit for the RCS DNBR parameters.