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November 15, 2010

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

Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Site, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
Supplement for Proposed License Amendment Request to Revise the Technical Specifications Pursuant to the Use of Gadolinia Integral Burnable Absorber
License Amendment Request Number 2009-12

In a letter dated October 19, 2009, Duke Energy Carolinas, LLC (Duke Energy) submitted a license amendment request (LAR) for the Oconee Nuclear Station (ONS) Renewed Facility Operating License (FOL). Specifically, Duke Energy requested Nuclear Regulatory Commission (NRC) review and approval for usage of gadolinia as an integral burnable neutron absorber in the uranium oxide fuel matrix. The proposed change revised Technical Specification (TS) 2.1.1, Reactor Core Safety Limits and TS 5.6.5.b, the Core Operating Limit Report and Duke Energy's NRC-approved methodology reports for reload design and non-Loss Of Coolant Accident (LOCA) safety analyses to allow use of gadolinia. The TS Bases were also provided.

Duke Energy was notified by the NRC on July 22, 2010 that the review associated with AREVA NP methodology report, BAW-10192, Revision 2 (ML083460314 and ML083460315) had been terminated. As a result, the ECCS analysis for gadolinia fuel, discussed in the AREVA methodology report and used in the gadolinia LAR was no longer acceptable. The NRC requested that ONS supplement the gadolinia LAR.

Enclosure 1 provides the details of the supplement. It includes a revision to the UFSAR Chapter 15 LOCA Evaluation which was previously based on BAW-10192, Revision 2. The revised UFSAR Chapter 15 LOCA Evaluation is based on BAW-10192P-A, Revision 0, BAW-10179P-A, Revision 7 and the NRC approved code RELAP5. The revision supercedes the information that was previously provided in the LAR dated October 19, 2009. The TS and TS Bases are also being re-submitted due to a supplement for the CASMO-4 / SIMULATE-3 LAR dated August 25, 2010. The supplement eliminated the need to provide the CASMO-4 / SIMULATE-3 changes to TS 5.6.5.b. Attachments 1 and 2 contain TS and TS Bases Mark-Ups and Reprinted Pages, respectively. Attachment 3 includes the revised affected methodology report. It supercedes Attachment 3 provided in the October 19, 2009 LAR.

This supplement does not contain any proprietary information for either Duke Energy or AREVA NP.

A001
NRR

Duke requests approval of this LAR by December 31, 2010. Duke will also update the UFSAR to include the revised methodologies and the new analysis results. These revisions will be submitted per 10 CFR 50.71(e). There are no new commitments being made as a result of this proposed change.

A copy of this LAR 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 ONS Regulatory Compliance Group at (864) 873-3364.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 15, 2010.

Sincerely,

T P Gillespie

T. Preston Gillespie, Jr.
Vice President, Oconee Nuclear Station

Enclosures:

1. Supplement for Proposed License Amendment Request to Revise the Technical Specifications Pursuant to the Use of Gadolinia Integral Burnable Absorber

Attachments:

1. Technical Specification and Technical Specifications Bases - Mark Up
2. Technical Specification and Technical Specifications Bases - Reprinted Pages
3. NFS-1001-A - Oconee Nuclear Station Reload Design Methodology (Revision 6a) – Mark Up

bc w/enclosures and attachments:

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ENCLOSURE 1

Supplement for Proposed License Amendment Request to Revise the Technical Specifications Pursuant to the Use of Gadolinia Integral Burnable Absorber License Amendment Request Number 2009-12

In a letter dated October 19, 2009, Duke Energy Carolinas, LLC (Duke Energy) submitted a license amendment request (LAR) for the Oconee Nuclear Station (ONS) Renewed Facility Operating License (FOL). Specifically, Duke Energy requested Nuclear Regulatory Commission (NRC) review and approval for usage of gadolinia as an integral burnable neutron absorber in the uranium oxide fuel matrix. The proposed change revised Technical Specification (TS) 2.1.1, Reactor Core Safety Limits and TS 5.6.5.b, the Core Operating Limit Report and Duke Energy's NRC-approved methodology reports for reload design and non-Loss Of Coolant Accident (LOCA) safety analyses to allow use of gadolinia. The TS Bases were also provided.

Duke Energy was notified by the NRC on July 22, 2010 that the review associated with AREVA NP methodology report, BAW-10192, Revision 2 (ML083460314 and ML083460315) had been terminated. As a result, the ECCS analysis for gadolinia fuel, discussed in the AREVA methodology report and used in this LAR is no longer acceptable. The NRC requested that the gadolinia LAR be supplemented.

The following information is provided as supplemental information to address the concerns associated with the NRC notification.

Methodology Report Revisions

Duke Energy will revise one of the six methodology reports provided in the October 19, 2009 LAR. It is described below and included in Attachment 3. Attachment 3 of this supplement will supercede Attachment 3 that was provided in the October 19, 2009 LAR.

Methodology Report NFS-1001-A, "Oconee Nuclear Station Reload Design Methodology (Revision 6a) will be revised as follows:

- A new section is added, Chapter 8.3, which summarizes how AREVA will model gadolinia in the LOCA analysis.

  The method added in Chapter 8.3 is the method used by AREVA for the other B&W class plants and is documented in the NRC approved topical report BAW-10179P-A, Chapters 9.2.3 and 9.3 (Reference 13).

Enclosure 3 and 4 revisions

The revised ECCS analysis is provided as follows and will supercede that which was previously provided in Enclosures 3 and 4, Section 3.2 of the LAR dated October 19, 2009:

3.2 UFSAR Chapter 15 LOCA Evaluation

The NRC approved LOCA evaluation model (EM) for Oconee is documented in topical report BAW-10192P-A, Rev. 0 (Reference 12) and is based on the NRC approved code RELAP5. The approved LOCA EM is further modified for use with gadolinia fuel in the NRC approved topical BAW-10179P-A, Rev. 7 (Reference 13).

Chapter 9.2.3 of BAW-10179 describes the steady-state fuel data input to the LOCA EM. Chapter 9.2.3 states, in its entirety:

> Steady-state fuel rod data, such as local volumetric fuel temperature as a function of LHR, fuel rod internal gas pressure, gap gas composition, and fuel rod dimensions and characteristics, are determined by an NRC-approved steady-state fuel rod computer code. The TACO3 (Reference 11) fuel rod design code is one of the codes that may be utilized to provide steady-state fuel rod input data for UO_2 fuel with either Zircaloy-4 or M5 cladding. The TACO3 predicted best-estimate fuel temperatures are adjusted by an uncertainty factor to ensure that a 95%/95% upper bound tolerance on the volume average temperature is used in the LOCA applications. The EM and steady-state fuel code provide information used to define the uncertainty factors that are applied, since the value of the 95%/95% uncertainty factor is dependent on the bundle or pin that is modeled. BAW-10186P-A, Revision 2, "Extended Burnup Evaluation" approves the use of TACO3 for fuel rod analysis up to a burnup of 62 GWd/mtU, provided that a bias factor is used to account for the reduced fuel thermal conductivity at burnups greater than 40 GWd/mtU. This burnup-dependent fuel thermal-conductivity bias factor increases the 95%/95% uncertainty factor applied to the TACO3 predicted fuel temperatures input in the EM analyses. The GDTACO (Reference 9) fuel rod design code also predicts best-estimate fuel temperatures that are augmented by a 95%/95% upper bound tolerance factor for use in LOCA applications. GDTACO may be utilized for analysis of gadolinia fuel with either Zircaloy-4 or M5 cladding. The fuel thermal-conductivity bias applied in the TACO3 volume-averaged fuel temperatures is also applied in the GDTACO results at burnups greater than 40 GWd/mtU. If no impact on operational limits is expected, fuel data for higher concentrations of gadolinia may be optionally selected to conservatively bound those for a lower concentration.

Chapter 9.3 of BAW-10179 describes the generic LOCA evaluations. The pertinent discussion related to gadolinia and how Areva would model gadolinia for Oconee cores follows:

> LOCA analyses for gadolinia pins are also performed to determine the reduction in allowable LHR limit necessary to account for the decrease in the fuel thermal conductivity compared with a UO_2 fuel rod of the same design. These evaluations are typically performed only at those elevations that have the limiting LOCA margin in the core power distribution analyses. Therefore, analyses that model the gadolinia fuel steady-state data are generally performed with axial peaking at the core inlet and sometimes for the core exit elevations. All burnup ranges and corresponding fuel thermal conductivity inputs based on GDTACO are supplied to RELAP5 in order to determine the LHR limit for the gadolinia pins. Analyses may be performed for each gadolinia concentration, or results obtained for a higher concentration may be conservatively applied to a lower concentration of gadolinia. The gadolinia LHR limit reduction is applied to the UO_2 LOCA LHR limits in order to define the envelope of maximum allowed LOCA LHR limit versus axial elevation and time-in-life for each analyzed gadolinia concentration.

Areva NP will perform the Oconee gadolinia fuel LOCA analyses the same as they do for the other Oconee class plants that also use gadolinia using the NRC approved LOCA EM and the subsequent method changes described above and in Chapter 9 of the NRC approved topical BAW-10179P-A, Rev. 7.

Technical Specification (TS) Revisions

TS 5.6.5.b, COLR, provides the previously approved analytical methods used to determine core operating limits. TS 5.6.5.b will be revised to reflect GDTACO in the title of DPC-NE-2008.

The previous LAR also included a revision to add Methodology Report DPC-NE-1006-P-A to the COLR listing. This revision was included in an August 12, 2010 supplement to the LAR concerning approval of CASMO-4.

The revised TS/TS Bases are being resubmitted in their entirety for clarity and are included in Attachments 1 and 2.

References

The following new references are added as a result of this supplement:

12. BWNT LOCA – BWNT Loss-of-Coolant Accident Analysis for Once Through Steam Generator Plant, BAW-10192-PA, Revision 0, AREVA NP, June 1998.
13. Safety Criteria and Methodology for Acceptable Cycle Reload Analyses, BAW-10179-PA, Revision 7, AREVA NP, January 2008.

ATTACHMENT 1

Technical Specification and Technical Specification Bases – Mark Up

TS 2.1.1.1
TS 5.6.5.b
TSB 2.1.1

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, for UO_2 fuel, the maximum local fuel pin centerline temperature shall be ≤ 465642 - (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 shall be ≤ 4656 – (6.5 x 10^{-3} x (Burnup, MWD/MTU)) °F. Operation within ~~this~~ 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)

6. Nuclear Overpower Flux/Flow/Imbalance and RCS Variable Low Pressure allowable value limits for Specification 3.3.1;

7. RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits for Specification 3.4.1

8. Core Flood Tanks Boron concentration limits for Specification 3.5.1;

9. Borated Water Storage Tank Boron concentration limits for Specification 3.5.4;

10. Spent Fuel Pool Boron concentration limits for Specification 3.7.12;

11. RCS and Transfer Canal boron concentration limits for Specification 3.9.1; and

12. AXIAL POWER IMBALANCE protective limits and RCS Variable Low Pressure protective limits for Specification 2.1.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

(1) DPC-NE-1002-A, Reload Design Methodology II;

(2) NFS-1001-A, Reload Design Methodology;

(3) DPC-NE-2003-P-A, Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01;

(4) DPC-NE-1004-A, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P;

(5) DPC-NE-2008-P-A, Fuel Mechanical Reload Analysis Methodology Using TACO3 and GDTACO;

(6) BAW-10192-P-A, BWNT LOCA - BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants;

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. The dependency of the fuel melt temperature on the as-built oxygen-to-uranium ratio for UO2 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

ATTACHMENT 2

Technical Specification and Technical Specifications Bases - Reprinted Pages

Remove	Insert
TS 2.0-1	TS 2.0-1
TS 5.0-25	TS 5.0-25
TSB 2.1.1-1	TSB 2.1.1-1
TSB 2.1.1-2	TSB 2.1.1-2

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B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

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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.

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BASES

BACKGROUND (continued)

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;

f. Flux/Flow Imbalance trip;

ATTACHMENT 3

NFS-1001-A – Oconee Nuclear Station Reload Design Methodology (Revision 6a) - Mark Up

Oconee Nuclear Station
Reload Design Methodology

NFS-1001-~~A~~
Revision ~~6a~~7

~~**June 2009**~~ October 2009

Nuclear Engineering Division
Nuclear Generation Department
Duke Energy Carolinas, LLC

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Appendix B Revision History

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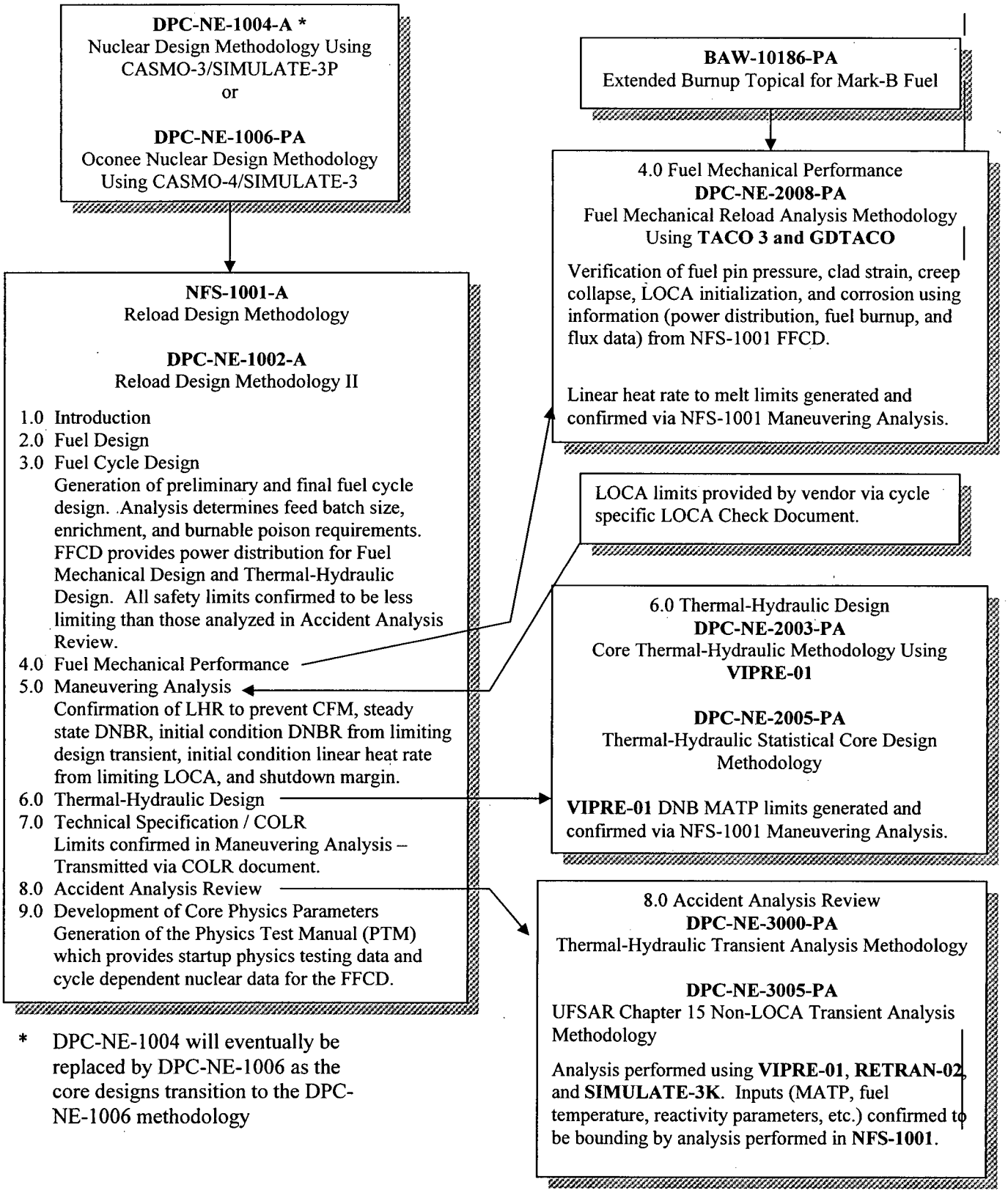
Figure 1-1

Relationship of Reload Methodology Reports



* DPC-NE-1004 will eventually be replaced by DPC-NE-1006 as the core designs transition to the DPC-NE-1006 methodology

3.0 Fuel Cycle Design

3.1 Preliminary Fuel Cycle Design

The purpose of the preliminary fuel cycle design (PFCD) is to determine the number and enrichment(s) of the fresh and possibly burned assemblies to be inserted during the next refueling. A preliminary fuel shuffling scheme is developed and check calculations on certain key parameters are performed.

The input required for the PFCD consists of general ground rules and design bases developed from cycle energy, contract, and operating requirements. The output of the PFCD is the number and enrichment(s) of the feed assemblies.

3.1.1 Overview of Nuclear Calculation System

The nuclear calculation system enables the nuclear designer to numerically model and simulate the nuclear reactor core. The current system in use for Oconee is described in Reference 2. The system that will replace Reference 2 and be used following NRC approval is documented in Reference 17.

3.1.2 Calculations and Results of PFCD

Once the calculation models are prepared for the cycle of interest, the nuclear designer chooses one or more feed enrichments, number of assemblies, and preliminary loading pattern for the reload core. Calculations are performed to verify cycle lifetime and power peaking. The process is iterated until the number and enrichment(s) of feed assemblies as well as a preliminary shuffle scheme has been determined which yield the desired cycle lifetime and a reasonable power distribution.

The preliminary number and enrichment(s) of the feed assemblies must typically be determined eighteen months prior to reactor shutdown for refueling to assure that an adequate quantity of separative work is available. Changes to these preliminary estimates are normally possible up to twelve months prior to reactor shutdown. It is necessary that the results of the PFCD be complete in time to support the fuel order.

3.2 Final Fuel Cycle Design

Having determined the preliminary number and enrichment(s) of the fuel assemblies during the PFCD, the final fuel cycle design (FFCD) concentrates on optimizing the placement of fresh and burned assemblies, control rod groupings, and burnable poison assemblies (if any) to result in an acceptable fuel cycle design. If not already performed during the PFCD, cladding corrosion calculations are performed to ensure licensing limits are met (References 7 and 8). The fuel cycle design is finalized based upon design criteria intended to ensure that the results of the subsequent calculations are acceptable. If unacceptable results are obtained, the fuel cycle design may be revised to obtain a design that produces acceptable results. When appropriate, the calculations performed to support the PFCD are incorporated into the FFCD.

3.2.3 Power Distribution Calculations

For Oconee, emphasis in the FFCD is on radial power distributions both on an assembly basis and on a local rod basis. Power distributions are calculated using the calculation methods described in Reference 2 or Reference 17. Radial pin peaking limits that will result in acceptable DNB and CFM margins are obtained from the accident analyses, thermal and thermal-hydraulic models. These margins are calculated and confirmed during the maneuvering analysis described in Section 5.0 and the accident analysis review described in Section 8.2.

3.2.4 Fuel Burnup Calculations

Current design criteria include limitations on fuel burnup. These limitations may be required as a result of calculations of internal fuel rod pressure, fuel rod growth, cladding corrosion, or licensing limitations. Fuel burnup calculations are performed using the calculation methods described in Reference 2 or Reference 17. Both assembly average and local fuel rod burnups may be calculated using these methods.

3.2.5 Reactivity Coefficients and Deficits

Reactivity coefficients define the reactivity insertion for small changes in reactor parameters such as moderator temperature, fuel temperature, and power level. These parameters are calculated using the methodology described in Reference 2 or Reference 17. These parameters are input to the safety analyses and used in modeling the reactor response during accidents and transients. Whereas reactivity coefficients represent reactivity effects over small changes in reactor parameters, reactivity deficits usually apply to reactivity inserted from larger changes typical of hot full power (HFP) to hot zero power (HZP). An example of a reactivity deficit is the power deficit from HFP to HZP. A different way of looking at the terms is that the coefficient when integrated over a given range yields the deficit, or the coefficient is the partial derivative of reactivity with respect to one specific parameter.

Typically, a nominal case is established at some reference condition. Then one parameter of interest is varied up and/or down by a fixed amount in another calculation and the resulting change in core reactivity divided by the parameter change yields the reactivity coefficient.

3.2.5.1 Doppler Coefficient

The Doppler coefficient or fuel temperature coefficient (FTC) is the change in core reactivity produced by a small change in fuel temperature. The major component of the Doppler coefficient arises from the behavior of the uranium-238 and plutonium-240 resonance absorption cross sections. As the fuel temperature increases, these resonances broaden and increase the chance that a neutron will be absorbed and thus decrease the core reactivity.

3.2.6 Boron Related Parameters

Critical boron concentrations are calculated at a variety of conditions as described in Reference 3.

3.2.7 Xenon Worth

The HFP equilibrium xenon worth may be calculated at BOC (4 EFPD) and at EOC. These values are compared to previous cycle values when a reload report is generated.

Calculations are performed for HFP equilibrium xenon conditions and for no xenon conditions. The difference in reactivities between the equilibrium and no xenon cases is the xenon worth.

3.2.8 Kinetics Parameters

The kinetics behavior of the nuclear reactor is often described in terms of solutions to the Inhour equation for six effective groups of delayed neutrons. Transient and accident analyses often involve kinetic modeling of the reactor core. The rate of change in power from a given reactivity insertion can be calculated by solving the kinetics equations if the six group effective delayed neutron fractions (β), the six group precursor decay constants (λ), and the prompt neutron lifetime are known.

The computer codes used to calculate these parameters are described in References 2, 3 and ~~3~~17. This information is needed for validation of the accident analyses and startup physics testing. The effective delayed neutron fraction (β-effective) for the new reload cycle is compared to that of the previous cycle when a reload report is generated.

5.0 MANEUVERING ANALYSIS

The purpose of a maneuvering analysis is to generate three-dimensional power distributions and imbalances for a variety of rod positions, xenon distributions, and power levels. The maneuvering analysis can be divided into four discrete phases. The first phase is the fuel cycle depletion performed to establish a nominal fuel depletion history. The second phase is the performance of various power maneuvers that conservatively characterize the effect of maldistributed xenon on the power distributions. The third phase is to perform control rod and axial power shaping rod (APSR) scans at the most severe times during the power transients. Each of these phases involves the running of multiple cases and generating three-dimensional power distributions, rod positions, and imbalances for each case. The methodology described in Reference 2 or Reference 17 is used to generate this information. Finally, this data is processed by computer programs which calculate CFM, clad strain, DNB, and LOCA margins to be used to set COLR (see Section 7.0) limits on rod position, axial offset versus power level, and reactor protective system setpoints.

5.1 Fuel Cycle Depletion

If appropriate restart files from the cycle depletion performed during the FFCD are not available, then the fuel cycle depletion is performed as the first step of the maneuvering analysis. Typical depletion steps are 0, 4, 12, 25, 50, 100, 150 ... EFPD. The xenon, power, and exposure data from these cases are saved for use in subsequent analyses.

5.2 Power Maneuvers

Power maneuvers are performed to generate axially skewed xenon distributions for input to the rod scan cases. The power maneuvers are performed near BOC, near EOC, and at least one intermediate burnup.

The first power maneuver is initiated by manipulating the control rods to produce a positive imbalance (with associated equilibrium iodine and xenon distributions) at full power. Control rod group 7 and the APSRs are then inserted to approximately the core midplane and core power reduced accordingly. This control rod insertion generates a large negative imbalance, and in conjunction with the power reduction causes the xenon in the bottom of the core to be depleted while the initial iodine in the top of the core increases the xenon concentration. The power level and rod positions are held constant, and the xenon concentration is allowed to peak over the next several hours. At a timestep near the peak xenon concentration, the xenon distribution is saved for input to the rod scan cases.

Section 7.2.2.1 (continued from previous page in NFS-1001)
The allowable total peaking factor (MAPF) is established by the relation:

$$MAPF = \frac{CFMLHR}{LHR \times FOP}$$

where: LHR is the average full power linear heat rate in the core, and
FOP is the power level expressed as a fraction of rated power.

The maneuvering analysis (Section 5.0) establishes the maximum calculated total peaking factors for various core conditions (power levels, xenon conditions, control rod positions and burnups). These calculated maximum total peaking factors are increased by several conservative factors to obtain the worst case expected total peaking factor corresponding to each condition. The individual conservative factors are as follows:

1) Nuclear uncertainty factor as specified in Reference 2 or Reference 17.
2) Spacer grid effect factor of 1.026, which is only applicable when utilizing assemblies with inconel intermediate spacer grids.
3) Engineering hot channel factor of 1.014 for UO_2 fuel and 1.0145 for gadolinia-bearing fuel
4) Densification power spike factor which varies with axial location of the peak in the core (Reference 15).
5) Fuel assembly bow factor.
6) Fuel rod bow factor.
7) ~~Lumped b~~Burnable poison manufacturing tolerance factor.

The nuclear uncertainty factor accounts for the uncertainty in the calculated peak due to the limitations of the analytical models. The spacer grid effect factor accounts for the flux distortion caused by inconel spacer grids (no spacer grid effect factor is required for zircaloy spacer grids). The engineering hot channel factor accounts for the manufacturing tolerances of critical fuel rod design parameters (pellet enrichment, pellet density, pellet diameter, etc.). The densification power spike factor accounts for the local flux enhancement resulting from gaps in the fuel column induced by fuel densification. The effect of fuel assembly bow on the pin power distribution is accounted for by a penalty factor that is dependent on the location of the pin within the assembly. A burnup dependent peaking penalty consistent with topical reports BAW-10147-PA (Reference 10) and BAW-10186-PA (Reference 7) is applied to account for the potential power peaking enhancement due to fuel rod bow. The ~~lumped~~ burnable poison manufacturing tolerance factor accounts for the effect of the variance in the as-built enrichment of the lumped burnable poison (LBP) pellets in LBP rods or the gadolinia in gadolinia-bearing fuel rods. The statistical combination of these factors is described in Reference 15.

The worst case expected maximum total peaking factors calculated in this manner for different power levels are compared to the respective allowable total peaking factors, and the CFM margin for each condition can be determined. The margin at a particular power level is given by:

$$Margin\ (\%) = \frac{(allowable\ total\ peak - worst\ case\ expected\ maximum\ total\ peak)}{allowable\ total\ peak} \times 100$$

Core conditions which correspond to non-negative margins are acceptable conditions, and core conditions which correspond to negative margins cannot be permitted. In order to preclude core conditions with negative margins, limits should be established on acceptable values of power peaking conditions for each power level, and corresponding reactor trip setpoints should be established so as to trip the reactor when conditions approach unacceptable values. Since power peaking cannot directly be measured by the RPS, power peaks are first correlated with the RPS-measurable axial offset for each power level. The outputs of the maneuvering calculations include the maximum total peaking factor in the core, its location and the corresponding core axial offset. In order to determine the axial offset limits that correspond to an acceptable margin for a particular power level, the margin for each calculated maximum total peak for that power level is plotted against the corresponding axial offset. These plots define a relationship between core offset and margin. The value of offset at the zero margin intercept defines the offset limit for that particular set of reactor conditions. Figure 7-2 provides an example of the analysis for the 100% full power (FP) case.

In practice, detailed calculations typically are performed for the 100% FP case. Limits for other power levels may be determined by conservatively extrapolating the 100% FP limits to other power levels by using the power feedback effect on peaking factors and by validating these limits by comparison with results of a limited number of maneuvering calculations at these power levels. Offset limits are typically established for power levels of 110% FP and 100% FP.

7.2.2.2 Calculation of Power-Power Imbalance Limits for DNBR Criterion

The power-power imbalance limits based on the DNBR criterion are determined by a synthesis of the results of the thermal-hydraulic analysis and the results of the maneuvering analysis.

The thermal-hydraulic analysis establishes the maximum allowable total peaking (MATP) factors as a function of core elevation for various axial flux shapes to prevent violation of the DNBR criterion. The maneuvering analysis generates the power distribution in the core (including the maximum total peaking factor and the associated axial peaking factor for each fuel assembly, typically in a quarter-core representation, and the core axial offset) for various design conditions and for various times in the cycle. For each power distribution, the calculated maximum total peaking factors of each of the assemblies is increased by the radial nuclear uncertainty factor, by a ~~lumped~~ burnable poison manufacturing tolerance factor, and by a factor to account for the effect of fuel assembly bow, and the resulting adjusted peak is compared to the allowable peaking factor for that axial peaking factor and axial peak location. The statistical combination of these factors is described in Reference 15. Application of the radial nuclear uncertainty is not necessary when the allowable peaking factor is determined using the statistical core design methodology described in Reference 6 (which accounts for the radial nuclear uncertainty in developing the allowable peaking factor). The DNBR margin is then obtained as:

$$DNBR\ Margin\ (\%) = \frac{(allowable\ total\ peak - adjusted\ maximum\ total\ peak)}{allowable\ total\ peak} \times 100$$

Section 7.4.1 (continued from previous page in NFS-1001)
The power peaking factor in the core changes with fuel burnup, axial imbalance, full length control rod position, and part length control rod (APSR) position. In addition, the peaking factor is influenced by the existence of any quadrant power tilt and non-equilibrium xenon conditions. Therefore, allowable ranges of these core operation parameters would have to be established in order for the maximum operating peaking factors at the designated axial locations to be within the allowable values. Although the fuel densification phenomenon has the potential for enhancing power peaks, no explicit allowance is required for power spikes associated with this phenomenon in the LOCA power distribution limits on the basis that the densification power spikes do not enhance the local heat flux.

The effect of a positive quadrant power tilt on the peaking factors is quantified either on a cycle-specific basis as a function of assembly location and burnup statepoints (using the methods described in Reference 2 or Reference 17), or by application of a conservative generic factor. The quadrant tilt power peaking factors are calculated as the percentage change in peak per percent change in quadrant tilt for each symmetric assembly. Specifically, a series of cases are executed with each unique control rod location modeled as a dropped rod. The associated increase in peaking and tilt in the opposite quadrant is tabulated for each symmetric assembly location. The largest ratio of percent change in peak per percent change in tilt is saved for each symmetric assembly location. These cycle-specific 'tilt factors' typically range from 0.8% to 1.4% increase in peaking factor (depending on the assembly location) per percent positive quadrant tilt. The conservative tilt factor may be as high as 1.5% increase in peaking factor per percent positive quadrant tilt. Technical Specifications permit reactor operation with a positive quadrant tilt as specified in the COLR. A tilt limit of 5.0% would typically amount to a 4.0% to 7.0% increase in peaking factor when using the cycle-specific tilt factors, or a 7.5% increase in peaking factor when using the conservative generic factor. Therefore, the allowable peaking factor would have to be reduced by 4.0% to 7.0%, or by 7.5%, whichever is applicable, to account for the permitted quadrant tilt condition.

The effect of non-equilibrium xenon conditions on peaking factors is quantified by the analysis of the power peaking factors occurring during various power maneuvers. Power redistribution caused by transient xenon in the power maneuver leads to peaking and offsets being explicitly accounted for in the setting of LOCA limits.

The remaining core parameters which influence the maximum operating power peaks are the full-length control rod position, part length control rod (APSR) position, axial imbalance, and core burnup. The permissible values of these quantities are to be determined such that the resulting power peaks, after accounting for any uncertainties, would be within the maximum allowable power peaks. The maneuvering analysis establishes the relationship of operating peaking factors at various axial locations with the core imbalance and control rod positions. The maneuvering analysis calculations include part length control rod scans inducing a range of values of core axial offset for different full length control rod positions. The calculations are performed for various power levels and for the full range of core burnups. The calculations yield the values of the maximum peaking factor at the different axial planes corresponding to various full-length control rod positions, various axial offsets, and for different part length rod positions, and these calculations also yield the variations of the maximum peaking factor with axial offset.

The calculated maximum peaks at each axial plane are increased by the following factors to obtain the worst case operating peaking factors. In addition, a power level uncertainty factor, as specified in References 3 and 15, is applied as a bias to the calculated maximum peaks.

1) Nuclear uncertainty factor as specified in Reference 2 or Reference 17.
2) Spacer grid effect factor of 1.026, which is only applicable when utilizing assemblies with Inconel intermediate spacer grids.
3) Engineering hot channel factor of 1.014 for UO_2 fuel and 1.0145 for gadolinia-bearing fuel
4) Fuel assembly bow factor.
5) Fuel rod bow factor.
6) ~~Lumped b~~Burnable poison manufacturing tolerance factor.

The nuclear uncertainty factor accounts for the uncertainty in the calculated peak due to the limitations of the analytical models. The spacer grid effect factor accounts for the flux distortion caused by Inconel spacer grids (no spacer grid effect factor is required for Zircaloy spacer grids). The engineering hot channel factor accounts for the manufacturing tolerances of critical fuel rod design parameters (pellet enrichment, pellet density, pellet diameter, etc.). The effect of fuel assembly bow on the pin power distribution is accounted for by a penalty factor that is dependent on the location of the pin within the assembly. A burnup dependent peaking penalty consistent with the topical reports BAW-10147-PA (Reference 10) and BAW-10186-PA (Reference 7) is applied to account for the potential power peaking enhancement due to fuel rod bow. The ~~lumped~~ burnable poison manufacturing tolerance factor accounts for the effect of the variance in the as-built enrichments of the lumped burnable poison (LBP) pellets in LBP rods or the gadolinia in gadolinia-bearing fuel rods. The statistical combination of these terms is described in Reference 15.

To determine the allowable values of full-length and part-length (APSR) control rod positions and the axial offsets, first an operating range for the full-length control rod position is chosen and then the ranges of axial offsets and part-length control rod positions for which the worst case operating peaking factors at the designated axial planes are less than or equal to their respective allowable values are determined. If the resulting ranges of axial offset and part-length control rod position are acceptable from the standpoint of operational flexibility, the assumed full-length control rod position ranges and the calculated range of axial offset and part-length control rod position are taken as their operating limits. If, however, the resulting ranges of axial offsets and part-length control positions are unacceptable from the standpoint of operational flexibility, a more restrictive full-length control rod bank position is selected and the corresponding axial offset and part-length control rod position limits are established.

8.0 ACCIDENT ANALYSIS REVIEW

8.1 Introduction

A major aspect of the safety consideration of a reactor is the analysis of postulated accidents. These safety analyses enable one to confirm that the reactor system is designed to mitigate such events and that the resulting consequences of such events are acceptable. The most important considerations affecting the calculated consequences of the various postulated accidents are (a) the values of plant parameters assumed in the analysis, (b) the performance characteristics of the mitigating systems assumed in the analysis, and (c) the analytical models used. In general, the accident analyses documented in the UFSAR (Reference 1) are based on values of plant parameters that correspond to bounding conditions, are based on conservative performance characteristics of the mitigating systems, and were performed utilizing generally accepted analytical methodology. The non-LOCA accident analysis methodology of DPC-NE-3005 (Reference 3) and DPC-NE-3000 (Reference 16) is used. The LOCA analysis methodology described in UFSAR Section 15.14 as modified for gadolinia fuel described in Section 8.3 below will also be used.

The primary goal of the safety analyses during the reload design process is to ensure the continued safe operation of the facility with the refueled core. The reference safety analyses and facility Technical Specifications establish the bases and conditions for safe operation of the core. An equivalent level of safety for the refueled core is established when it is determined that the reload design satisfies the analysis bases and conditions. In particular, the accident analyses contained in the licensing basis safety analyses remain valid if a reload design predicts steady-state and transient parameters that lie within the ranges of the values assumed in the reference analyses. Thus, reload safety analysis consists of verifying that the core physics, fuel performance, thermal-hydraulic, and mechanical design parameters for the reload design are bounded by the licensing basis analysis values.

8.2 Overview of Accident Analysis Review

The role of accident analysis review in a typical Oconee reload design consists of a systematic review of the reference analyses of all postulated accidents. In this review each accident is examined by comparing the values of important plant parameters and RPS trip functions and trip setpoints assumed in the reference accident analyses to the corresponding values predicted for the fuel cycle under consideration. The safety parameters of interest for the reload cycle are obtained from appropriate nuclear design, thermal-hydraulic design, and fuel performance analyses. If the safety analysis review confirms that all pertinent plant parameters and RPS trip functions and trip setpoints for the reload cycle are conservative with respect to their values assumed in the accident analyses, it is concluded that the reference accident analyses continue to be valid for the fuel cycle, and therefore in these situations no reanalyses of accidents are performed. If, however, one or more plant parameters or RPS trip functions or trip setpoints assumed in the reference accident analyses are found to be non-conservative for the fuel cycle, a reanalysis of affected accidents is performed. This process is shown schematically in Figure 8-1.

8.3 LOCA Analyses

The NRC approved LOCA evaluation model for Oconee is documented in Reference 18. Areva has modified the model for use with gadolinia fuel as described in Sections 9.2.3 and 9.3 of Reference 19, which is also NRC approved. The pertinent text from Reference 19 is repeated below.

8.3.1 Steady-State Fuel Data Input to LOCA EM

Steady-state fuel rod data, such as local volumetric fuel temperature as a function of LHR, fuel rod internal gas pressure, gap gas composition, and fuel rod dimensions and characteristics, are determined by an NRC-approved steady-state fuel rod computer code. The TACO3 (Reference 4) fuel rod design code is one of the codes that may be utilized to provide steady-state fuel rod input data for UO_2 fuel with either Zircaloy-4 or M5 cladding. The TACO3 predicted best-estimate fuel temperatures are adjusted by an uncertainty factor to ensure that a 95%/95% upper bound tolerance on the volume average temperature is used in the LOCA applications. The EM and steady-state fuel code provide information used to define the uncertainty factors that are applied, since the value of the 95%/95% uncertainty factor is dependent on the bundle or pin that is modeled. Reference 7 approves the use of TACO3 for fuel rod analysis up to a burnup of 62 GWd/mtU, provided that a bias factor is used to account for the reduced fuel thermal conductivity at burnups greater than 40 GWd/mtU. This burnup-dependent fuel thermal-conductivity bias factor increases the 95%/95% uncertainty factor applied to the TACO3 predicted fuel temperatures input in the EM analyses. The GDTACO (Reference 4) fuel rod design code also predicts best-estimate fuel temperatures that are augmented by a 95%/95% upper bound tolerance factor for use in LOCA applications. GDTACO may be utilized for analysis of gadolinia fuel with either Zircaloy-4 or M5 cladding. The fuel thermal-conductivity bias applied in the TACO3 volume-averaged fuel temperatures is also applied in the GDTACO results at burnups greater than 40 GWd/mtU. If no impact on operational limits is expected, fuel data for higher concentrations of gadolinia may be optionally selected to conservatively bound those for a lower concentration.

8.3.2 Generic LOCA Evaluation for Gadolinia Fuel

LOCA analyses for gadolinia pins are also performed to determine the reduction in allowable LHR limit necessary to account for the decrease in the fuel thermal conductivity compared with a UO_2 fuel rod of the same design. These evaluations are typically performed only at those elevations that have the limiting LOCA margin in the core power distribution analyses. Therefore, analyses that model the gadolinia fuel steady-state data are generally performed with axial peaking at the core inlet and sometimes for the core exit elevations. All burnup ranges and corresponding fuel thermal conductivity inputs based on GDTACO are supplied to RELAP5 in order to determine the LHR limit for the gadolinia pins. Analyses may be performed for each gadolinia concentration, or results obtained for a higher concentration may be conservatively applied to a lower concentration of gadolinia. The gadolinia LHR limit reduction is applied to the UO_2 LOCA LHR limits in order to define the envelope of maximum allowed LOCA LHR limit versus axial elevation and time-in-life for each analyzed gadolinia concentration.

9.0 DEVELOPMENT OF CORE PHYSICS PARAMETERS

Upon completion of the reload design, a variety of physics parameters have been generated primarily for HFP and some HZP conditions. The purpose of this stage of developing core physics parameters is to provide additional calculations to supplement those already performed. These calculations are performed using the methodology described in Reference 2 or Reference 17. The results of these calculations are used for startup test predictions and core physics parameters throughout the cycle. Changes to the startup test procedures, plant operations, or particular core designs may change the physics parameters that are required. The following descriptions are typical of current requirements.

9.1 Startup Test Predictions

After each refueling, the reactor undergoes a startup test program aimed at verifying that the reactor core is correctly loaded, that control rods are in the correct locations and are functioning properly, and that reactor behavior is accurately predicted by the nuclear models which were used in generating the data used in the plant's safety analyses.

9.1.1 Critical Boron Concentrations and Boron Worths

Critical boron concentrations and boron worths are typically calculated at a variety of rod configurations, at HZP and HFP, as a function of boron concentration, at different xenon concentrations, and at different times in the fuel cycle. The calculation model is capable of critical boron searches and when critical boron concentrations are desired is usually run in this mode. An acceptable alternative, however, is to not search on critical boron but to correct the input boron concentration to the critical boron concentration using a calculated boron worth and the calculated reactivity.

Both HFP and HZP critical boron calculations are normally performed for startup physics tests. Soluble boron worths are usually calculated at HFP and HZP for startup physics tests. The boron worths are usually calculated by running two similar cases except that the soluble boron concentration is varied. The differential boron worth is calculated by subtracting the reactivities and dividing by the boron difference. Differential boron worths are usually quoted in $\%\Delta\rho/100$ ppmb or in ppmb/$\%\Delta\rho$ (the latter term is sometimes referred to as the inverse boron worth).

Critical boron concentration is calculated as a function of cycle burnup. These predictions may be provided in tabular form.

Differential boron worth may be calculated as a function of boron concentration and also as a function of cycle burnup. These predictions may also be provided in tabular form.

10.0 REFERENCES

1. Oconee Nuclear Station, Units 1, 2, and 3, Updated Final Safety Analysis Report, Docket Nos. 50-269, -270, and -287.

2. Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004-A, Revision 1a, Duke Energy Carolinas, January 2009.

3. UFSAR Chapter 15 Transient Analysis Methodology, DPC-NE-3005-PA, Revision ~~3a~~4, Duke Energy Carolinas, ~~February 2009~~***publication date***.

4. Fuel Mechanical Reload Analysis Methodology using TACO3 and GDTACO, DPC-NE-2008-PA, Revision ~~1a~~2, Duke Energy Carolinas, ~~December 2008~~***publication date***.

5. Oconee Nuclear Station Core Thermal-Hydraulic Methodology using VIPRE-01, DPC-NE-2003-PA, Revision ~~2a~~3, Duke Energy Carolinas, ~~December 2008~~***publication date***.

6. Thermal-Hydraulic Statistical Core Design Methodology, DPC-NE-2005-PA, Revision 4a, Duke Energy Carolinas, December 2008.

7. Extended Burnup Evaluation, BAW-10186-PA, Revision ~~1~~2, AREVA NP, ~~April 7, 2000~~June 2003.

8. Letter from D. L. LaBarge (NRC) to W. R. McCollum, Jr. (ONS), Use of Framatome Cogema Fuels Topical Report on High Burnup – Oconee Nuclear Station, Units 1, 2, and 3 (TAC Nos. MA0405, MA0406, MA0407), Docket Nos. 50-269, 50-270, and 50-287; March 1, 1999.

9. Letter from M. S. Tuckman to Document Control Desk, Duke Energy Corporation's Use of FCF's Extended Burnup Evaluation Topical Report, BAW-10186-PA, August 25, 1999.

10. Fuel Rod Bowing in Babcock and Wilcox Fuel Designs, BAW-10147-PA, Revision 1, AREVA NP, June 28, 1983.

11. Letter from H. N. Berkow to M. S. Tuckman, Subject: Duke Power use of CROV Computer Code, June 19, 1995.

12. Fuel Reconstitution Analysis Methodology, DPC-NE-2007-PA, Revision 0, Duke Power Company, October 1995.

13. Letter from D. E. LaBarge to M. S. Tuckman, Subject: Oconee Nuclear Station, Units 1, 2, and 3, RE: Topical Report DPC-NE-2003, Rev. 1 (TAC Nos. MA8234, MA8235, MA8236), June 23, 2000.

14. Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel, BAW-10227-PA, AREVA NP, February 11, 2000.

15. Oconee Nuclear Station Reload Design Methodology II, DPC-NE-1002-A, Revision ~~3b~~4, Duke Energy Carolinas, ~~June 2009~~***publication date***.

16. Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision 45, Duke Energy Carolinas, ~~October 2008~~***publication date***.

17. Oconee Nuclear Station Reload Design Methodology Using CASMO-4/SIMULATE-3, DPC-NE-1006-P, Revision 0, Duke Energy Carolinas, May 2009

18. BWNT LOCA – BWNT Loss-of-Coolant Accident Analysis for Once Through Steam Generator Plant, BAW-10192-PA, Revision 0, AREVA NP, June 1998

19. Safety Criteria and Methodology for Acceptable Cycle Reload Analyses, BAW-10179-PA, Revision 7, AREVA NP, January 2008

Appendix B

Revision History

Date	Occurrence
April 23, 1979	Revision 0 of NFS-1001 submitted to the NRC
May 20, 1980	Revision 1 of NFS-1001 submitted to the NRC
October 16, 1980	First NRC RAI issued
November 13, 1980	First Duke response to the first RAI submitted
January 28, 1981	Second Duke response to the first RAI and Revision 2 of NFS-1001 submitted to the NRC
March 18, 1981	Third Duke response to the first RAI submitted
April 22, 1981	Revision 3 of NFS-1001 submitted to the NRC
June 2, 1981	Second NRC RAI issued
June 16, 1981	Duke response to the second RAI and Revision 4 of NFS-1001 submitted to the NRC
July 29, 1981	NRC SER issued
December 22, 1999	Revision 5 of NFS-1001 submitted to the NRC
May 24, 2000	NRC RAI issued
August 23, 2000	Duke response to the RAI submitted
December 8, 2000	NRC SER issued
October 22, 2007	DPC-NE-2015 submitted to the NRC
September 17, 2008	Duke response to the (emailed) RAI submitted
October 29, 2008	NRC SER issued for DPC-NE-2015
June 2009	Revision 6a of NFS-1001 approved
publication date	Revision 7 of NFS-1001 published

NFS-1001 was originally submitted to the Nuclear Regulatory Commission (NRC) in April 1979. It was approved by the NRC in July 1981. In between the original submittal and the approval were two NRC Requests for Additional Information (RAI), four separate Duke responses to the RAIs, and four revisions of the report, as shown in the table above. It was Duke's practice at that time to issue a new revision of the report if the RAI response modified the report in any way. The original NRC Safety Evaluation Report (SER) for NFS-1001 was dated July 29, 1981 and it approved Revision 4 of the report.

Revision 5 of NFS-1001 was submitted to the NRC in December 1999 and it was approved by the NRC with the SER dated December 8, 2000.

Revision 6 of NFS-1001 was submitted to the NRC via DPC-NE-2015 (Oconee Nuclear Station Mark-B-HTP Fuel Transition Methodology) in October 2007. This transition report was approved by the NRC with the SER dated October 29, 2008.

Revision 6a of NFS-1001 was approved in June 2009. This revision contained the following significant changes. The first three changes were approved by the NRC in DPC-NE-2015, and the remaining changes were implemented via 10CFR50.59.

1) The Mark-B-HTP fuel design was added to the list of fuel assembly designs in Section 2.1.

2) The fuel densification power spike factor, which was specified as a value of 1.08, was replaced with an axially-dependent factor.

3) A fuel assembly bow penalty factor and a lumped burnable poison manufacturing tolerance factor were applied to CFM, DNB and LOCA margins.

4) The target average core moderator temperature was corrected from approximately 580 °F to approximately 579 °F.

5) The start of the average moderator temperature plateau was changed from approximately 15% full power to an approximate range of 15-20 % full power based on the efficiency of the steam generators.

Revision 7 of NFS-1001 was published in ***mmm 2010***. This revision updated the reference methodology report DPC-NE-2008 due to the inclusion of GDTACO in the title. It added DPC-NE-1006 to the list of references and approved methodologies and updated the DNB, CFM, and LOCA margins discussion to add a description of the burnable poison manufacturing tolerance factor for gadolinia as it is included in the SCUF equations in DPC-NE-1002. It also added the engineering hot channel factor for HTP with gadolinia and added Section 8.3 (and References 18 and 19) to address changes to the LOCA analyses due to the presence of gadolinia.