MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the NRC in accordance with 10 CFR 50.4, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,
2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
3. Control Bank Insertion Limits for Specification 3/4.1.3.6,
4. Axial Flux Difference Limits, target band*, and APL^{ND}* for Specification 3/4.2.1,
5. Heat Flux Hot Channel Factor, F_{Q}^{RTP}, K(Z), W(Z)**, APL^{ND}**, $F_{Q}^{L}(X,Y,Z)$ and $W(Z)_{BL}$** for Specification 3/4.2.2, and
6. Nuclear Enthalpy Rise Hot Channel Factor, $F\Delta H^{L***}(X,Y)$ or, $F_{\Delta H}^{RTP****}$, and Power Factor Multiplier, $MF\Delta H$****, limits for Specification 3/4.2.3.
7. Overtemperature and Overpower Delta T setpoint parameter values for Specification 2.2.1.
8. Boric Acid Storage System and Refueling Water Storage Tank volume and boron concentration limits for Specifications 3/4.1.2.5 and 3/4.1.2.6.
9. Reactor Water Makeup Pump flowrate limits for Specifications 3/4.3.3.12 and 3/4.9.2.

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* Reference 5 is not applicable to target band and APL^{ND}.

** References 4 and 5 are not applicable to W(Z), and APL^{ND}, and $W(Z_{BL})$.

*** Reference 1 is not applicable to $F\Delta H^{L}$.

**** Reference 5 is not applicable to $F_{\Delta H}^{RTP}$ and $MF_{\Delta H}$.

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CORE OPERATING LIMITS REPORT (Continued)

10. Accumulator and Refueling Water Storage Tank boron concentration limits for Specifications 3/4.5.1 and 3/4.5.4.

11. Reactor Coolant System and refueling canal boron concentration limits for Specification 3/4.9.1.

12. Standby Makeup Pump water supply boron concentration limit of Specification 4.7.13.3.

13. Spent Fuel Pool boron concentration limit of Specification 3/4.9.12.

insert

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

   (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 -Nuclear Enthalpy Rise Hot Channel Factor.)

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary).

   (Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology.)

3. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987, (W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

4. BAW-10168PA, Rev. 1, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," January 1991 (B&W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,

3. Control Bank Insertion Limits for Specification 3/4.1.3.6,

4. Axial Flux Difference Limits, target band*, and APL^{ND}* for Specification 3/4.2.1,

5. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(Z), W(Z)**, APL^{ND}**, $F_Q^L(X,Y,Z)$ and $W(Z)_{BL}$** for Specification 3/4.2.2, and

6. Nuclear Enthalpy Rise Hot Channel Factor, $F\Delta H^L$***(X,Y) or, $F_{\Delta H}^{RTP}$****, and Power Factor Multiplier, $MF\Delta H$****, limits for Specification 3/4.2.3.

7. Overtemperature and Overpower Delta T setpoint parameter values for Specification 2.2.1.

8 Boric Acid Storage System and Refueling Water Storage Tank volume and boron concentration limits for Specifications 3/4.1.2.5 and 3/4.1.2.6.

9. Reactor Water Makeup Pump flowrate limits for Specifications 3/4.3.3.12 and 3/4.9.2.

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* Reference 5 is not applicable to target band and APL^{ND}.

** References 4 and 5 are not applicable to W(Z), and APL^{ND}, and $W(Z_{BL})$.

*** Reference 1 is not applicable to $F\Delta H^L$.

****Reference 5 is not applicable to $F_{\Delta H}^{RTP}$ and $MF_{\Delta H}$.

CORE OPERATING LIMITS REPORT (Continued)

10. Accumulator and Refueling Water Storage Tank boron concentration limits for Specifications 3/4.5.1 and 3/4.5.4.

11. Reactor Coolant System and refueling canal boron concentration limits for Specification 3/4.9.1.

12. Standby Makeup Pump water supply boron concentration limit of Specification 4.7.13.3.

13. Spent Fuel Pool boron concentration limit of Specification 3/4.9.12.

added

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A. "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

   (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary).

   (Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology.)

3. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987, (W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

4. BAW-10168PA, Rev. 1, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," January 1991 (B&W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

CORE OPERATING LIMITS REPORT (Continued)

5. DPC-NE-2011P-A, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," March, 1990 (DPC Proprietary).

   (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

6. DPC-NE-3001P-A, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," November 1991 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

7. DPC-NF-2010P-A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, Specification 4.7.13.3 - Standby Makeup Pump Water Supply Boron Concentration, Specification 3.9.1 - RCS and Refueling Canal Boron Concentration, and Specification 3.9.12 - Spent Fuel Pool Boron Concentration.)

8. DPC-NE-3002A, "FSAR Chapter 15 System Transient Analysis Methodology," November 1991.

   (Methodology used in the system thermal-hydraulic analyses which determine the core operating limits)

9. DPC-NE-3000P-A, Rev. 1, "Thermal-Hydraulic Transient Analysis Methodology," November 1991.

   (Modeling used in the system thermal-hydraulic analyses)

10. DPC-NE-1004A, "Design Methodology Using CASMO-3/Simulate-3P," November 1992.

    (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

11. DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," December 1991 (DPC Proprietary).

    (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.2.1 - Axial Flux Difference (AFD), and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor $F\Delta H$ (X,Y).)

12. DPC-NE-2001P-A, Rev. 1, "Fuel Mechanical Reload Analysis Methodology for Mark-BW Fuel," October 1990 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC in accordance with 10 CFR 50.4.

Sample COLR Pages

1.0 Core Operating Limits Report

This Core Operating Limits Report (COLR) has been prepared in accordance with the requirements of Technical Specification 6.9.1.9.

The Technical Specifications affected by this report are listed below:

Tech Spec Section	Technical Specifications	COLR Section	COLR Page
2.2.1	Reactor Trip System Instrumentation Setpoints	2.0	5
3/4.1.1.3	Moderator Temperature Coefficient	3.0	7
3/4.1.2.5	Borated Water Source - Shutdown	3.1	9
3/4.1.2.6	Borated Water Source - Operating	3.2	10
3/4.1.3.5	Shutdown Rod Insertion Limit	3.3	10
3/4.1.3.6	Control Rod Insertion Limit	3.4	10
3/4.2.1	Axial Flux Difference	3.5	10
3/4.2.2	Heat Flux Hot Channel Factor	3.6	13
3/4.2.3	Nuclear Enthalpy Rise Hot Channel Factor	3.7	16
3/4.3.3.11	Boron Dilution Mitigation System	3.8	18
3/4.5.1	Accumulators	3.9	18
3/4.5.4	Refueling Water Storage Tank	3.10	18
3/4.9.1	Refueling Operations - Boron Concentration	3.11	19
3/4.9.2	Instrumentation	3.12	19
3/4.9.12	Refueling Operations - Spent Fuel Pool Boron Concentration	3.13	19
4.7.13.3	Standby Makeup Pump Water Supply - Boron	3.14	20

Sample COLR Pages

3.11 Refueling Operations - Boron Concentration (Specification 3/4.9.1)

3.11.1 Minimum boron concentrations for the filled portions of the Reactor Coolant System and refueling canal. Applicable for mode 6 with the reactor vessel head closure bolts less than fully tensioned, or with the head removed.

Parameter	Limit
Refueling boron concentration for the filled portions of the Reactor Coolant System and refueling canal for LCO 3.9.1.b.	2,475 ppm

3.12 Instrumentation (Specification 3/4.9.2)

3.12.1 Reactor Makeup Water Pump Flowrate Limit:

Applicable Mode	Limit
Mode 6	$\leq$ 70 gpm

3.13 Refueling Operations - Spent Fuel Pool Boron Concentration (Specification 3/4.9.12)

3.13.1 Minimum boron concentration limit for the spent fuel pool. Applicable when fuel is stored in the spent fuel pool.

Parameter	Limit
Spent fuel pool minimum boron concentration for LCO 3.9.12	2,475 ppm

1 INTRODUCTION

This submittal represents Duke Power Company's formal request for approval of a license amendment which establishes restricted loading patterns and associated burnup criteria for placement of new and irradiated fuel into the Catawba spent fuel storage pools. Analyses performed in support of this submittal demonstrate that the use of these configurations for storing fuel with initial enrichments of up to 5.00weight% U-235 will maintain sufficient criticality safety margins. This amendment will allow for maximum utilization of the fuel storage racks and will provide additional flexibility in the area of reactor core reload analysis and design.

All enrichments in this submittal are nominal values. As built variations of up to 0.05 weight% U-235 above the nominal enrichments are acceptable. This uncertainty is accounted for in the analyses.

The methodology used to analyze the fuel loading patterns and generate the various burnup criteria discussed above is identical to that submitted for NRC approval in the request for License Amendment for the McGuire Nuclear Station dated June 13, 1994. This methodology was specifically developed for spent fuel burnup credit applications and is based on the CASMO and SIMULATE computer codes. Formal approval of this methodology was requested in the above referenced submittal. Therefore, this License Amendment Request for the Catawba Nuclear Station relies on the approval of Duke Power Company's burnup credit methodology, and as such, references this methodology where appropriate.

In addition, this submittal is similar to the Seabrook Station Unit 1 Amendment No. 6, which was approved by the NRC August 27, 1991. Specifically, both the Seabrook amendment and this amendment request use the CASMO and SIMULATE computer codes for fuel storage burnup credit analysis, establish three fuel types with corresponding loading pattern restrictions and allow high reactivity fuel which is unacceptable for storage in an infinite array to be stored with empty locations or with appropriately qualified low reactivity fuel.

V. GENERAL DESCRIPTION OF AMENDMENT REQUEST/SUBMITTAL

V.1 Introduction

The primary purpose of this submittal is to demonstrate sufficient analytical justification for modifying technical specifications and FSAR sections currently applicable to new and irradiated fuel storage at the Catawba Nuclear Station (CNS). Areas proposed for modification are the limitations and restrictions associated with 1) storage of unirradiated fuel in auxiliary building new fuel storage vaults, and 2) storage of irradiated and un-irradiated fuel in both spent fuel pools.

The fuel storage rack designs for the new fuel storage vaults and the spent fuel storage pools of both CNS units are identical. The spent fuel storage racks are the original storage racks designed and fabricated by Lamco Industries and installed when the plant was constructed between 1975 and 1985. Consequently all analytical methods and results discussed in this document are applicable to either unit as are the resulting procedural and technical specification modifications. Additionally, the most reactive of all fuel types currently in use, or in storage at either the Catawba or McGuire Nuclear Stations, was used in developing the proposed requirements and limitations. Eventual approval of this amendment request should therefore apply to both units.

V.2 New Fuel Storage Vaults

The new fuel storage vaults which are used for temporary dry storage of unirradiated reload fuel are built on 21 inch centers and are currently licensed for maximum fuel enrichments of 4.0weight% U-235. To accommodate anticipated increases in individual and/or batch average enrichments that will exceed this 4.0weight% limit, analytical methods described in Section 4.3.2.6 of the FSAR were used to demonstrate that fuel containing up to 5.00weight% U-235 can be safely stored in these fuel racks. No other restrictions beyond this enrichment limit are applicable to storage in the new fuel vaults for the fuel designs analyzed. Discussion of the methods used to justify this increased limit can be found in section VI.5 and the resulting proposed FSAR modifications can be found in section VIII.2. No technical specifications are applicable to the new fuel storage vaults.

V.3 Spent Fuel Storage Pools

The basic spent fuel storage rack arrangement for units 1 and 2 is shown in figure 5-1 on the following page. A schematic of the individual storage cell configuration is also provided.

VII GENERAL RESULTS OF ANALYSES

This section provides the basic results of the various analyses performed in support of the proposed FSAR and Technical Specification changes. Additional discussion of the supporting analytical techniques can be found in the appendices of the McGuire License Amendment Request, June 13, 1994.

VII.1 New Fuel Vaults

The calculated worst case k_{eff}s for a fuel assembly with the maximum enrichment of $5.00_{weight\%}$ U-235 under fully flooded and optimum moderation conditions in the Catawba new fuel vault are shown below.

Condition	Worst Case k_{eff}
Optimum Moderation	0.95861
Fully Flooded	0.93022

These values were specifically calculated for the Westinghouse Optimized Fuel Assembly (OFA) design which has been shown to be the most reactive fresh fuel in the Catawba new fuel storage vault, of all fuel types stored at any Duke Power facility. These values also include geometrical and material biases and uncertainties at a 95 percent probability and a 95 percent confidence level as required to demonstrate criticality safety. Fuel cage tolerances are included in the geometrical uncertainty.

As specified in Section 9.1 of the Standard Review Plan, the maximum k_{eff} value in LWR new fuel storage vaults shall be less than or equal to 0.98 under optimum moderator conditions and less than or equal to 0.95 under fully flooded conditions. The analytical result shown above indicates that these criteria have been met.

VII.2 Fuel Assembly Burnup Requirements

The burnup credit criticality analysis results are summarized by the two burnup curves shown in figure 7-1. These curves define the burnup requirements for the two fuel loading configurations allowed in the Catawba spent fuel pools. Specific data points generated by the criticality analysis and used to create these burnup curves are shown in Table 7-1.

will have no effect on the k_{eff}. A full row of restricted assemblies serving as this boundary would not be acceptable. The boundary restriction is summarized below:

Loading Configuration Interface Restriction	
Unrestricted and Restricted w/ Filler	Row of separation between restricted/filler and unrestricted storage must be a row which contains alternating restricted and filler assemblies

VII.4 Fuel Misloading Accident Analysis

A specific analysis was performed to verify that sufficient margin is provided by the soluble boron to maintain the pool configuration at or below k_{eff} of 0.95 following the misloading of a fresh 5.00weight% assembly in place of a required filler assembly. The results of this analysis showed that the allowed soluble boron provided sufficient criticality safety margin.

Proposed Requirements:

Technical Specification Reference: 3/4.9 Refueling Operations

REFUELING OPERATIONS

3/4.9.12 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.12 The boron concentration in the spent fuel pool shall be within the limit specified in the COLR.

APPLICABILITY:

During storage of fuel in the spent fuel pool.

ACTION:

a. Immediately suspend movement of fuel assemblies in the spent fuel pool and initiate action to restore the spent fuel pool boron concentration to within its limit.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS:

4.9.12. Verify at least once per 7 days that the spent fuel pool boron concentration is within its limit.

Proposed Requirements:

Technical Specification Reference: 3/4.9 Refueling Operations

3/4.9.13 SPENT FUEL ASSEMBLY STORAGE

LIMITING CONDITION FOR OPERATION

3.9.13 New or irradiated fuel may be stored in the Spent Fuel Pool in accordance with these limits:

a. Unrestricted storage of fuel meeting the criteria of Table 3.9-1; or

b. Restricted storage in accordance with Figure 3.9-1, of fuel which does not meet the criteria of Table 3.9-1.

APPLICABILITY:

During storage of fuel in the spent fuel pool.

ACTION:

a. Immediately initiate action to move the noncomplying fuel assembly to the correct location.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS:

4.9.13 Prior to storing a fuel assembly in the spent fuel storage pool, verify by administrative means the initial enrichment and burnup of the fuel assembly are in accordance with Specification 3.9.13.

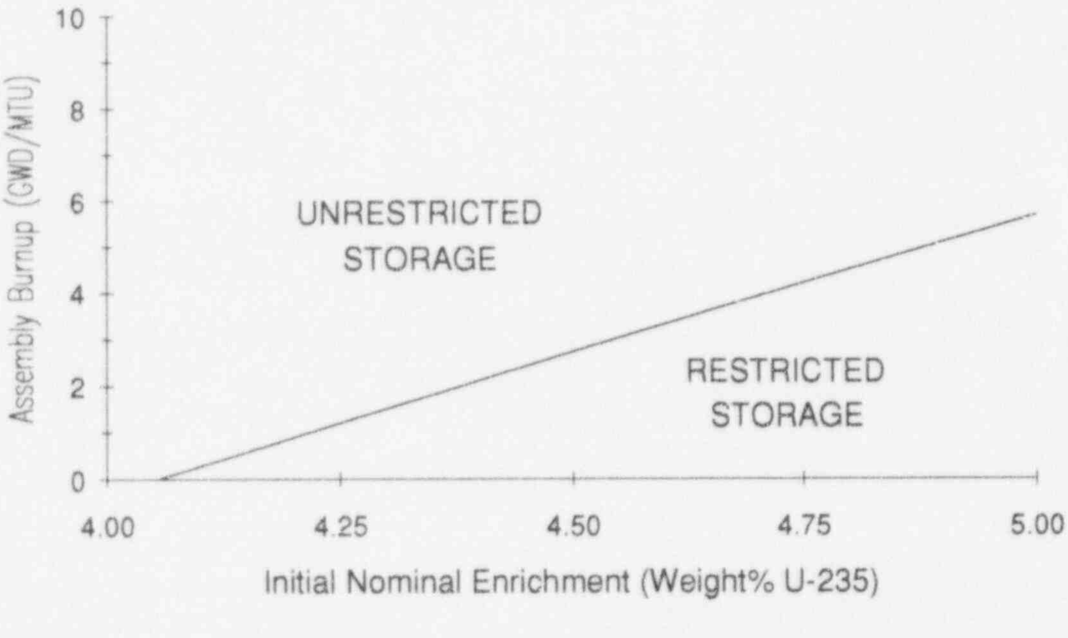
Proposed Requirements:

Technical Specification Reference: 3/4.9 Refueling Operations

Table 3.9-1

Minimum Qualifying Burnup Versus Initial Enrichment for Unrestricted Storage

Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
4.05 (or less)	0
4.50	2.73
5.00	5.67



Fuel which differs from those designs used to determine the requirements of Table 3.9-1 may be qualified for Unrestricted storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Likewise, previously unanalyzed fuel up to 5.0 weight% U-235 may be qualified for Restricted storage by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

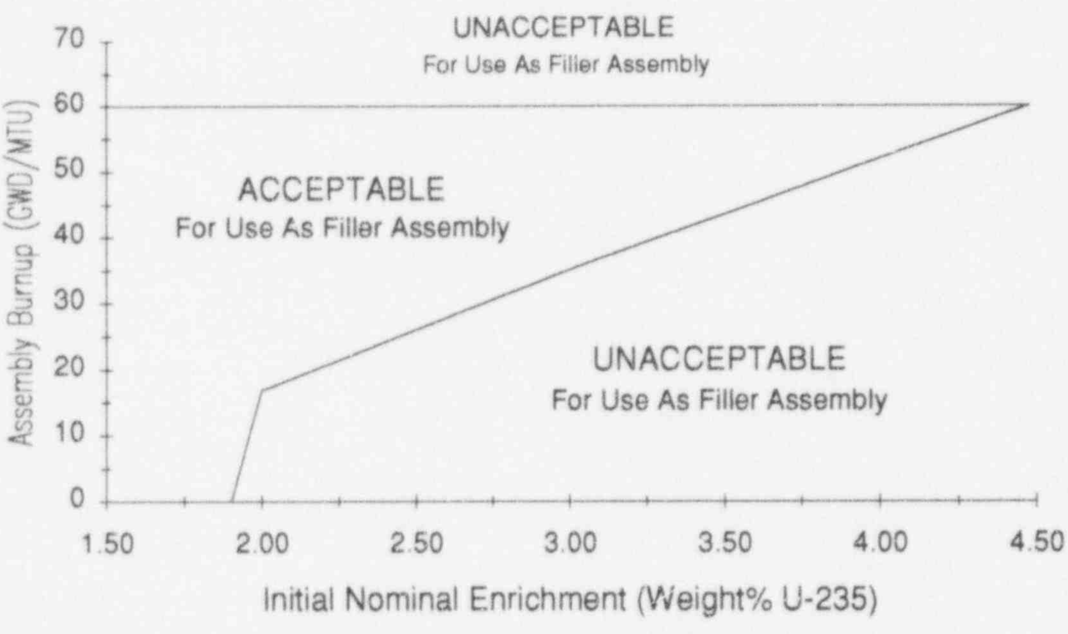
Proposed Requirements:

Technical Specification Reference: 3/4.9 Refueling Operations

Table 3.9-2

Minimum Qualifying Burnup Versus Initial Enrichment for Filler Assemblies

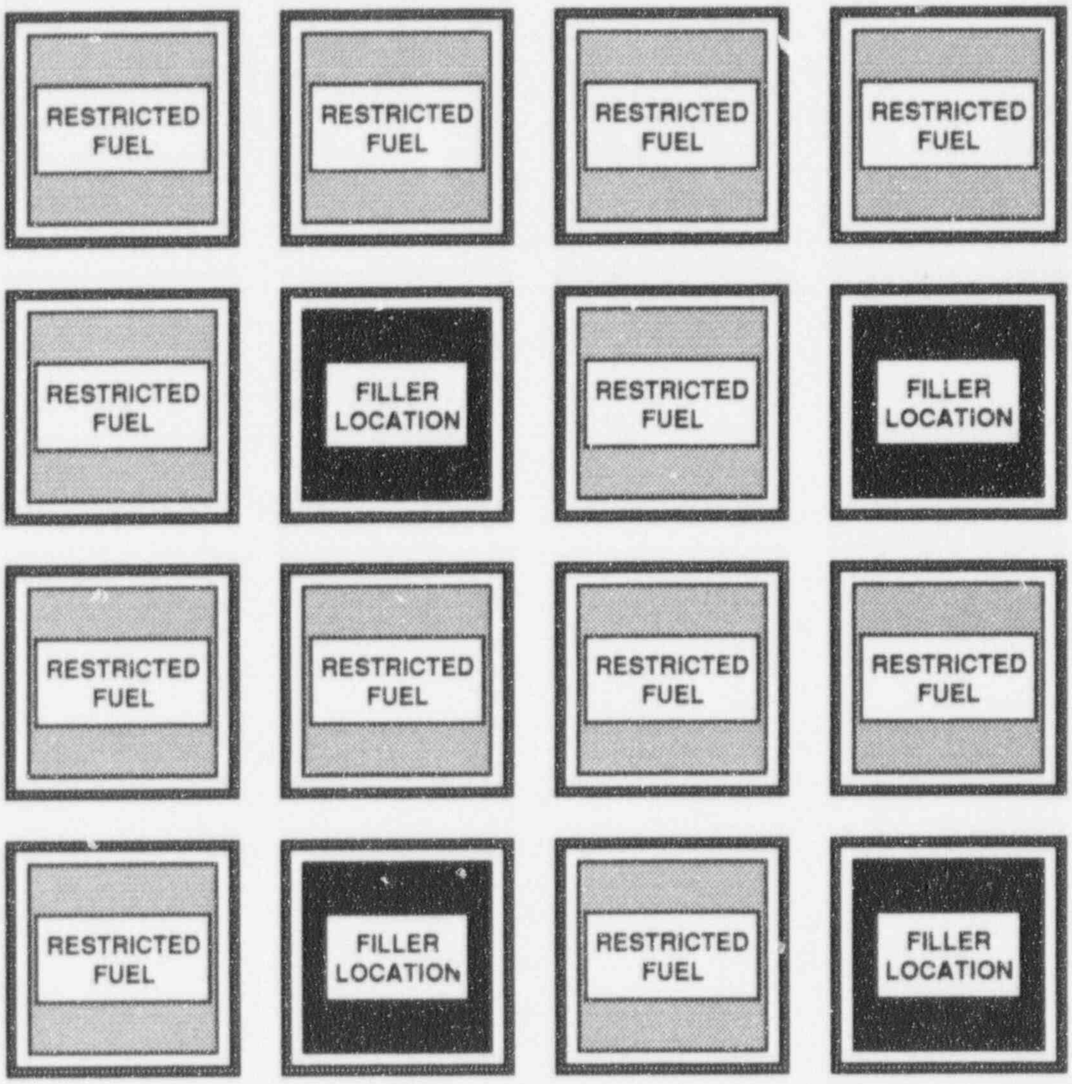
Initial Nominal Enrichment (Weight% U-235)	Assembly Burnup (GWD/MTU)
1.90 (or less)	0
2.00	16.83
2.50	26.05
3.00	35.11
3.50	43.48
4.00	51.99
4.48	60.00



Fuel which differs from those designs used to determine the requirements of Table 3.9-2 may be qualified for use as a Filler Assembly by means of an analysis using NRC approved methodology to assure that k_{eff} is less than or equal to 0.95.

Figure 3.9-1

Required 3 out of 4 Loading Pattern for Restricted Storage



Restricted Fuel: Fuel defined for Restricted Storage in Table 3.9-1. (Fuel defined for Unrestricted Storage in Table 3.9-1, or non-fuel components, or an empty location may be placed in restricted fuel locations as needed)

Filler Location: Either fuel which meets the minimum burnup requirements of Table 3.9-2, or an empty cell.

Boundary Condition: Any row bounded by an Unrestricted Storage Area shall contain a combination of restricted fuel assemblies and filler locations arranged such that no restricted fuel assemblies are adjacent to each other.
Example: In the figure above, row 1 or column 1 can not be adjacent to an Unrestricted Storage Area, but row 4 or column 4 can be.

Proposed Requirements:

Technical Specification Reference: 3/4.9 Refueling Operations

BASES

3/4.9.12 and 3/4.9.13 SPENT FUEL POOL BORON CONCENTRATION and SPENT FUEL ASSEMBLY STORAGE

The requirements for spent fuel pool boron concentration specified in Specification 3.9.12 ensure that a minimum boron concentration is maintained in the pool. The requirements for spent fuel assembly storage specified in Specification 3.9.13 ensure that the pool remains subcritical. The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence the design of the spent fuel storage racks is based on the use of unborated water, which maintains the spent fuel pool in a subcritical condition during normal operation with the pool fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 4) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the accidental misloading of a fuel assembly. This could increase the reactivity of the spent fuel pool. To mitigate this postulated criticality related accident, boron is dissolved in the pool water.

Tables 3.9-1 and 3.9-2 allow for specific criticality analyses for fuel which does not meet the requirements for storage defined in these tables. These analyses would require using NRC approved methodology to ensure that $k_{eff} \leq 0.95$ with a 95 percent probability at a 95 percent confidence level as described in Section 9.1 of the FSAR. This option is intended to be used for fuel not included in previous criticality analyses. Fuel storage is still limited to the configurations defined in TS 3.9-13. The use of specific analyses for qualification of previously unanalyzed fuel includes, but is not limited to, fuel assembly designs not previously analyzed which may be as a result of new fuel designs or fuel shipments from another facility. Currently analyzed fuel designs include the Babcock and Wilcox MkBW design, and the Westinghouse Standard and Optimized fuel designs. Another more likely, and expected use of this specific analysis provision would be to analyze movement and storage of individual fuel pins as a result of reconstitution activities.

In verifying the design criteria of $k_{eff} \leq 0.95$, the criticality analysis assumed the most conservative conditions, i.e. fuel of the maximum permissible reactivity for a given configuration. Since the data presented in Specification 3.9.13.a and 3.9.13.b represents the maximum reactivity requirements for acceptable storage, substitutions of less reactive components would also meet the $k_{eff} \leq 0.95$ criteria. Hence an empty cell, or a non-fuel component may be substituted for any designated fuel assembly location. These, or other substitutions which will decrease the reactivity of a particular storage cell will only decrease the overall reactivity of the spent fuel storage pool.

Proposed Requirements:

Technical Specification Reference: 3/4.9 Refueling Operations

If both restricted and unrestricted storage is used, an additional criteria has been imposed to ensure that the boundary row between these two configurations would not locally increase the reactivity above the required limit.

The action statement applicable to fuel storage in the spent fuel pool requires that action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. Prior to the resumption of fuel movement, the requirements of the LCOs must be met. This requires restoring the soluble boron concentration and the correct fuel storage configuration to within the corresponding limits. This does not preclude movement of a fuel assembly to a safe position.

The surveillance requirements ensure that the requirements of the two LCOs are satisfied, namely boron concentration and fuel placement. The boron concentration in the spent fuel pool is verified to be greater than or equal to the minimum limit. The fuel assemblies are verified to meet the subcriticality requirement by meeting either the initial enrichment and burnup requirements of Table 3.9-1 and 3.9-2, or by using NRC approved methodology to ensure that $k_{eff} \leq 0.95$. By meeting either of these requirements, the analyzed accidents are fully addressed.

The fuel storage requirements and restrictions discussed here and applied in specification 3.9.13 are based on a maximum allowable fuel enrichment of 5.0 weight% U-235. The enrichments listed in Tables 3.9-1 and 3.9-2 are nominal enrichments and include uncertainties to account for the tolerance on the as built enrichment. Hence the as built enrichments may exceed the enrichments listed in the tables by up to 0.05 weight% U-235. Qualifying burnups for enrichments not listed in the tables may be linearly interpolated between the enrichments provided. This is because the reactivity of an assembly varies linearly for small ranges of enrichment.

REFERENCES

1. "Regulatory Guide 1.13: Spent Fuel Storage Facility Design Basis", U.S. Nuclear Regulatory Commission, Office of Standards Development, Revision 1, December 1976.

2. "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations", American Nuclear Society, ANSI N210-1976/ANS-57.2, April 1976.

3. FSAR, Section 9.1.

4. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

Current Requirements:

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,

3. Control Bank Insertion Limits for Specification 3/4.1.3.6,

4. Axial Flux Difference Limits, target band*, and APL^{ND*} for Specification 3/4.2.1,

5. Heat Flux Hot Channel Factor, F_Q^{RTP} , K(Z), W(Z)**, APL^{ND**}, $F_Q^L(X,Y,Z)$ and $W(Z)_{BL}$** for Specification 3/4.2.2, and

6. Nuclear Enthalpy Rise Hot Channel Factor, $F\Delta H^{L***}(X,Y)$ or, $F_{\Delta H}^{RTP****}$, and Power Factor Multiplier, $MF\Delta H^{****}$, limits for Specification 3/4.2.3.

7. Overtemperature and Overpower Delta T setpoint parameter values for Specification 2.2.1.

8. Boric Acid Storage System and Refueling Water Storage Tank volume and boron concentration limits for Specifications 3/4.1.2.5 and 3/4.1.2.6.

9. Reactor Water Makeup Pump flowrate limits for Specifications 3/4.3.3.12 and 3/4.9.2.

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* Reference 5 is not applicable to target band and APL^{ND}.

** References 4 and 5 are not applicable to W(Z), and APL^{ND}, and $W(Z_{BL})$.

*** Reference 1 is not applicable to $F\Delta H^L$.

****Reference 5 is not applicable to $F_{\Delta H}^{RTP}$ and $MF_{\Delta H}$.

Current Requirements:

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (Continued)

10. Accumulator and Refueling Water Storage Tank boron concentration limits for Specifications 3/4.5.1 and 3/4.5.4.

11. Reactor Coolant System and refueling canal boron concentration limits for Specification 3/4.9.1.

12. Standby Makeup Pump water supply boron concentration limit of Specification 4.7.13.3.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

   (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary).

   (Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology.)

3. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987, (W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

4. BAW-10168PA, Rev. 1, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," January 1991 (B&W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

CORE OPERATING LIMITS REPORT (Continued)

5. DPC-NE-2011P-A, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," March, 1990 (DPC Proprietary).

   (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

6. DPC-NE-3001P-A, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," November 1991 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

7. DPC-NF-2010P-A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, Specification 4.7.13.3 - Standby Makeup Pump Water Supply Boron Concentration, Specification 3.9.1 - RCS and Refueling Canal Boron Concentration, and Specification 3.9.12 - Spent Fuel Pool Boron Concentration.)

8. DPC-NE-3002A, "FSAR Chapter 15 System Transient Analysis Methodology," November 1991.

   (Methodology used in the system thermal-hydraulic analyses which determine the core operating limits)

9. DPC-NE-3000P-A, Rev. 1, "Thermal-Hydraulic Transient Analysis Methodology," November 1991.

   (Modeling used in the system thermal-hydraulic analyses)

Current Requirements:

ADMINISTRATIVE CONTROLS

10. DPC-NE-1004A, "Design Methodology Using CASMO-3/Simulate-3P," November 1992.

    (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

11. DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," December 1991 (DPC Proprietary).

    (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.2.1 - Axial Flux Difference (AFD), and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}$ (X,Y).)

12. DPC-NE-2001P-A, Rev. 1, "Fuel Mechanical Reload Analysis Methodology for Mark-BW Fuel," October 1990 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC in accordance with 10 CFR 50.4.

Proposed Requirements:

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,
2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
3. Control Bank Insertion Limits for Specification 3/4.1.3.6,
4. Axial Flux Difference Limits, target band*, and APL^{ND*} for Specification 3/4.2.1,
5. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(Z), W(Z)**, APL^{ND**}, $F_Q^L(X,Y,Z)$ and $W(Z)_{BL}$** for Specification 3/4.2.2, and
6. Nuclear Enthalpy Rise Hot Channel Factor, $F\Delta H^{L***}(X,Y)$ or, $F_{\Delta H}^{RTP****}$, and Power Factor Multiplier, $MF\Delta H^{****}$, limits for Specification 3/4.2.3.
7. Overtemperature and Overpower Delta T setpoint parameter values for Specification 2.2.1.
8. Boric Acid Storage System and Refueling Water Storage Tank volume and boron concentration limits for Specifications 3/4.1.2.5 and 3/4.1.2.6.
9. Reactor Water Makeup Pump flowrate limits for Specifications 3/4.3.3.12 and 3/4.9.2.

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* Reference 5 is not applicable to target band and APL^{ND}.

** References 4 and 5 are not applicable to W(Z), and APL^{ND}, and $W(Z_{BL})$.

*** Reference 1 is not applicable to $F\Delta H^L$.

****Reference 5 is not applicable to $F_{\Delta H}^{RTP}$ and $MF_{\Delta H}$.

Proposed Requirements:

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (Continued)

10. Accumulator and Refueling Water Storage Tank boron concentration limits for Specifications 3/4.5.1 and 3/4.5.4.

11. Reactor Coolant System and refueling canal boron concentration limits for Specification 3/4.9.1.

12. Standby Makeup Pump water supply boron concentration limit of Specification 4.7.13.3.

13. Spent Fuel Pool boron concentration limit of Specification 3/4.9.12.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

   (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary).

   (Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology.)

3. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987, (W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

4. BAW-10168PA, Rev. 1, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," January 1991 (B&W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

Proposed Requirements:

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (Continued)

5. DPC-NE-2011P-A, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," March, 1990 (DPC Proprietary).

   (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

6. DPC-NE-3001P-A, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," November 1991 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

7. DPC-NF-2010P-A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, Specification 4.7.13.3 - Standby Makeup Pump Water Supply Boron Concentration, Specification 3.9.1 - RCS and Refueling Canal Boron Concentration, and Specification 3.9.12 - Spent Fuel Pool Boron Concentration.)

8. DPC-NE-3002A, "FSAR Chapter 15 System Transient Analysis Methodology," November 1991.

   (Methodology used in the system thermal-hydraulic analyses which determine the core operating limits)

9. DPC-NE-3000P-A, Rev. 1, "Thermal-Hydraulic Transient Analysis Methodology," November 1991.

   (Modeling used in the system thermal-hydraulic analyses)

Proposed Requirements:

ADMINISTRATIVE CONTROLS

10. DPC-NE-1004A, "Design Methodology Using CASMO-3/Simulate-3P," November 1992.

    (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

11. DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," December 1991 (DPC Proprietary).

    (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.2.1 - Axial Flux Difference (AFD), and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}$ (X,Y).)

12. DPC-NE-2001P-A, Rev. 1, "Fuel Mechanical Reload Analysis Methodology for Mark-BW Fuel," October 1990 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC in accordance with 10 CFR 50.4.

Current Requirements:

Sample COLR Pages

1.0 Core Operating Limits Report

This Core Operating Limits Report (COLR) has been prepared in accordance with the requirements of Technical Specification 6.9.1.9.

The Technical Specifications affected by this report are listed below:

Tech Spec Section	Technical Specifications	COLR Section	COLR Page
2.2.1	Reactor Trip System Instrumentation Setpoints	2.0	5
3/4.1.1.3	Moderator Temperature Coefficient	3.0	7
3/4.1.2.5	Borated Water Source - Shutdown	3.1	9
3/4.1.2.6	Borated Water Source - Operating	3.2	10
3/4.1.3.5	Shutdown Rod Insertion Limit	3.3	10
3/4.1.3.6	Control Rod Insertion Limit	3.4	10
3/4.2.1	Axial Flux Difference	3.5	10
3/4.2.2	Heat Flux Hot Channel Factor	3.6	13
3/4.2.3	Nuclear Enthalpy Rise Hot Channel Factor	3.7	16
3/4.3.3.11	Boron Dilution Mitigation System	3.8	18
3/4.5.1	Accumulators	3.9	18
3/4.5.4	Refueling Water Storage Tank	3.10	18
3/4.9.1	Refueling Operations - Boron Concentration	3.11	19
3/4.9.2	Instrumentation	3.12	19
4.7.13.3	Standby Makeup Pump Water Supply - Boron	3.13	20

Current Requirements:

Sample COLR Pages

3.11 Refueling Operations - Boron Concentration (Specification 3/4.9.1)

3.11.1 Minimum boron concentrations for the filled portions of the Reactor Coolant System and refueling canal. Applicable for mode 6 with the reactor vessel head closure bolts less than fully tensioned, or with the head removed.

Parameter	Limit
Refueling boron concentration for the filled portions of the Reactor Coolant System and refueling canal for LCO 3.9.1.b.	2,475 ppm

3.12 Instrumentation (Specification 3/4.9.2)

3.12.1 Reactor Makeup Water Pump Flowrate Limit:

Applicable Mode	Limit
Mode 6	$\leq$ 70 gpm

Proposed Requirements:

Sample COLR Pages

1.0 Core Operating Limits Report

This Core Operating Limits Report (COLR) has been prepared in accordance with the requirements of Technical Specification 6.9.1.9.

The Technical Specifications affected by this report are listed below:

Tech Spec Section	Technical Specifications	COLR Section	COLR Page
2.2.1	Reactor Trip System Instrumentation Setpoints	2.0	5
3/4.1.1.3	Moderator Temperature Coefficient	3.0	7
3/4.1.2.5	Borated Water Source - Shutdown	3.1	9
3/4.1.2.6	Borated Water Source - Operating	3.2	10
3/4.1.3.5	Shutdown Rod Insertion Limit	3.3	10
3/4.1.3.6	Control Rod Insertion Limit	3.4	10
3/4.2.1	Axial Flux Difference	3.5	10
3/4.2.2	Heat Flux Hot Channel Factor	3.6	13
3/4.2.3	Nuclear Enthalpy Rise Hot Channel Factor	3.7	16
3/4.3.3.11	Boron Dilution Mitigation System	3.8	18
3/4.5.1	Accumulators	3.9	18
3/4.5.4	Refueling Water Storage Tank	3.10	18
3/4.9.1	Refueling Operations - Boron Concentration	3.11	19
3/4.9.2	Instrumentation	3.12	19
3/4.9.12	Spent Fuel Pool Boron Concentration	3.13	19
4.7.13.3	Standby Makeup Pump Water Supply - Boron	3.14	20

Proposed Requirements:

Sample COLR Pages

3.11 Refueling Operations - Boron Concentration (Specification 3/4.9.1)

3.11.1 Minimum boron concentrations for the filled portions of the Reactor Coolant System and refueling canal. Applicable for mode 6 with the reactor vessel head closure bolts less than fully tensioned, or with the head removed.

Parameter	Limit
Refueling boron concentration for the filled portions of the Reactor Coolant System and refueling canal for LCO 3.9.1.b.	2,475 ppm

3.12 Instrumentation (Specification 3/4.9.2)

3.12.1 Reactor Makeup Water Pump Flowrate Limit:

Applicable Mode	Limit
Mode 6	≤ 70 gpm

3.13 Spent Fuel Pool Boron Concentration (Specification 3/4.9.12)

3.13.1 Minimum boron concentration limit for the spent fuel pool. Applicable when fuel is stored in the spent fuel pool.

Parameter	Limit
Spent fuel pool minimum boron concentration for LCO 3.9.12	2,475 ppm

VIII.2 Proposed FSAR Modifications

Sections 4.3, Nuclear Design, and 9.1, Fuel Storage and Handling, are the only sections of the Catawba FSAR which will require modification as a result of the proposed technical specification changes detailed in this license amendment submittal. Specific language will be developed upon approval of this amendment request. In general however, changes will focus on the following sections:

4.3.2.6 Criticality of the Reactor During Refueling and Criticality of Fuel Assemblies

The description of the codes and methodology used in the criticality analysis would be updated to reflect the use of both the CASMO-3/TABLES-3/SIMULATE-3 and SCALE/KENO Va methodologies. The use of burnup credit will also be discussed.

9.1.1.3 Safety Evaluation

The descriptions of the assemblies pertaining to the enrichment will be changed to reflect the increase in the upper enrichment limit to 5.00weight%.

9.1.2.3.1 Criticality Analysis

Assumption number 5 in the listing of those made in evaluating criticality safety will be changed to reflect both the increase in the upper enrichment limit to 5.00weight%, and the use of burnup credit for assemblies greater than 4.05weight% U-235.

The calculated worst case k_{eff} values will be updated to reflect the increase in enrichment. The discussion of postulated accidents will be revised to include accidents which could increase reactivity.

9.1.3.1.1 Spent Fuel Pool Cooling

While the spent fuel cooling system will not be modified as a result of this submittal, the normal and abnormal heat load assumptions will change as a result of the higher anticipated discharge burnups. The results of an analysis based on these higher heat loads will be reflected in this section.

ATTACHMENT 3

MODIFICATIONS TO NO SIGNIFICANT HAZARDS ANALYSIS
AND
ENVIRONMENTAL IMPACT ANALYSIS

CATAWBA NUCLEAR STATION

ATTACHMENT II
NO SIGNIFICANT HAZARDS ANALYSIS

Duke Power Company has reviewed the proposed changes utilizing the criteria specified in 10CFR50.92 and has determined that the proposed changes do not involve a Significant Hazards Consideration pursuant thereto, for the reasons discussed below.

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

There is no increase in the probability or consequences of an accident in the new fuel vault since the only credible accidents for this area are criticality accidents and it has been shown that calculated, worst case K_{eff} for this area is ≤ 0.95 for fully flooded conditions and $K_{eff} \leq 0.98$ under optimum moderation conditions. This is in accordance with current licensing criteria. Likewise, there is no increase in the probability or consequences of an accident in the Spent Fuel Pool since, for criticality accidents, analyses have shown that $K_{eff} \leq 0.95$ under all conditions is being maintained.

There is also no increase in the probability or consequences of a fuel drop accident in the Spent Fuel Storage Pool. Since the mass of an assembly will not be affected by t' increase in fuel enrichment, the probability of an accident is not increased, and since the fission product inventory of individual fuel assemblies will not change significantly as a result of increased enrichment, the consequences of a fuel rupture accident remain unchanged.

The likelihood of other accidents, previously evaluated and described in Section 9.1.2 of the FSAR, is also not affected by the proposed changes. In fact, it could be postulated that since the increase in fuel enrichment will allow for extended fuel cycles, there will be a decrease in fuel movement and the probability of an accident may likewise be decreased.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not create the possibility of a new or different kind of accident since fuel handling accidents(fuel drop and misplacement) are not new or different kinds of accidents. Fuel handling accidents are already discussed in the FSAR for fuel with enrichments up to 4.0 weight %. As described in Section VI.9 of Attachment IV, additional analyses have been performed for fuel with enrichment up to 5.00 weight %. Worst case misloading accidents associated with the new loading patterns were evaluated. It was shown that the negative reactivity provided by soluble boron maintains $k_{eff} \leq 0.95$ under all conditions.

3. The proposed changes do not involve a significant reduction in the margin of safety.

The proposed change does not involve a significant reduction in the margin of safety since, a $k_{eff} \leq 0.95$ is being maintained for fully flooded conditions and a $k_{eff} \leq 0.98$ under optimum moderation conditions. The specification of $k_{eff} \leq 0.98$, for optimum moderation conditions in the new fuel vault, is an addition to the existing specification requirements of $k_{eff} \leq 0.95$ for fully flooded conditions. Although previous analyses have been performed to demonstrate that this requirement could be met, there was no licensing requirement to do so. Addition of this specification brings the specification more in line with current STS requirements and, in fact, may increase the margin of safety since, compliance with this requirement was not previously required.

ATTACHMENT III
ENVIRONMENTAL IMPACT ANALYSIS

Pursuant to 10CFR51.22 (b), an evaluation of the proposed amendments has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10CFR51.22 (c)9 of the regulations. The proposed amendment does involve changes in the use of facility components located within the restricted area as defined in 10CFR20, and changes some surveillance requirements however, the proposed amendment does not involve; 1) a significant hazards consideration, 2) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or 3) a significant increase in individual or cumulative occupational radiation exposure.

In drawing these conclusions, Duke Power Company has reviewed the published Notice of Environmental Assessment and Finding of No Significant Impact for Extended Burnup Fuel Use in Commerical LWR's in the Federal Register (53 FR 6040), dated February 29, 1988. We have determined that this notice is applicable to the changes for Catawba Nuclear Station, Units 1 & 2, and it's fuel. In the above cited notice, the staff concluded that the environmental impacts summarized in Table S-4 of 10CFR51.52 for the transportation of fuel up to 4.0 weight percent enriched Uranium-235 and a burnup level of 33,000 MWT/MTU are conservative and bound the corresponding impacts for transportation of fuel with a burnup level up to 60,000 MWD/MTU and enrichments up to 5.0 wight percent. In addition, we have reviewed the Staff's environmental assessment for the Shearon Harris Nuclear Power Plant for enrichment and extended fuel burnup increases pursuant to 10CFR51.21, 51.32, and 51.35 and a finding of no significant impact as published in the Federal Register (53 FR 30355) on August 11, 1988. We have determined that this evaluation is equally applicable to these changes, as well.
In summary, we believe this amendment request meets the criteria set forth in 10CFR51.22 (c) 9 of the regulations for categorical exclusion from an environmental impact statement.

ATTACHMENT I
PROPOSED TECHNICAL SPECIFICATION CHANGES

This section contains the proposed modifications to the CNS Technical Specifications. In general, these changes increase the initial fuel enrichment limit and establish restricted loading patterns, and associated burnup criteria, for qualifying fuel in the Catawba Spent Fuel Pools. These changes are necessary to improve core reload designs and increase operational flexibility, while at the same time maintaining acceptable criticality safety margin and decay heat removal capabilities. In addition, several administrative changes have been included in order to provide clarity to the Specifications and bring them more in line with STS format. A description of each of the changes being requested is given below.

The accompanying FSAR changes will be incorporated at the next annual revision following approval of this submittal. These changes are identified and discussed in Section VIII of Attachment IV.

1. The Technical Specification Index is being changed to add Specifications 3/4.9.12 and 3/4.9.13, accompanying Tables 3.9-1 and 3.9-2, and Figure 3.9-1. This change is purely administrative in nature.

2. Specification 3/4.9.12, Spent Fuel Pool (SFP) Boron Concentration is being added to establish an explicit SFP boron concentration limit, where one does not exist, as well as to establish a consistent LCO for SFP boron concentration for all modes operation. By allowing this limit to be specified in the COLR, the change reduces the possibility of a dilution event since all other potential sources of borated water to the SFP are also specified in the COLR. It also provides consistency with other operational, cycle specific limits. This change is being made in order to accommodate the more complex SFP storage requirements and provide clarity to the Specifications, as well as increase operational flexibility. The change also provides more consistency with STS format.

3. Specification 3/4.9.13, accompanying Tables 3.9-1 and 3.9-2, and Figure 3.9-1 are being added to establish restricted loading patterns (with appropriate interface restrictions) for spent fuel storage and associated burnup criteria. The proposed changes are necessary to increase the efficiency of fuel storage while at the same time ensuring that acceptable criticality safety margin and decay heat removal capabilities are maintained. The format of these changes is also more in line with STS format. The technical basis for these changes and the associated criticality analysis are described in detail in Attachment IV.

4. The BASES for Sections 3/4.9.12 and 3/4.9.13 of the Technical Specifications has been added to reflect the addition of the corresponding Specifications and to more fully explain the basis for each LCO, Action Statement and Surveillance Requirement covered by these Specifications. Paragraph 2 of the BASES explains the provisions to use specific analysis for fuel types not previously analyzed. In addition the specific fuel designs used to develop the limits in Specification 3/4.9.13 are discussed. Paragraph 3 of the BASES explains the acceptability of using less reactive fuel components or non-fuel components in designated fuel assembly locations and non-fuel components in empty cell locations, as this would ensure the reactivity limits are met while increasing operational flexibility. In addition, the last paragraph specifies the limit for maximum fuel

enrichment, 5.00 weight %, as the basis for all fuel storage requirements imposed by Technical Specification 3/4.9.13 and to describe appropriate methods for interpolating the data provided in Tables 3.9-1 to 3.9-5. The proposed modifications to the BASES Section are also more consistent with those in STS.

5. Technical Specification 5.6, Fuel Storage, has been changed to reflect appropriate limits for criticality analysis for fuel storage. This change allows increased operational flexibility, while maintaining acceptable criticality safety margin. In addition the Specifications have been reformatted to bring them more in line with STS format.

a. Specification 5.6.1 has been changed to allow for use of $k_{eff} \leq 0.98$ under optimum moderation conditions in the rack design criteria for new fuel storage racks. Actual calculations have shown that $k_{eff} \leq 0.95$, under all storage conditions however, this change allows increased flexibility when performing criticality analyses and is consistent with the criteria currently specified in ANSI-ANS57.3, 1983 and STS.

6. Specification 6.9 has been changed to require specification of the SFP boron concentration limit in the Core Operating Limits Report (COLR). This change is being made to ensure consistency with corresponding changes to Specifcation 3/4.9.12 and is administrative in nature.