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January 28, 1999

U. S. Nuclear Regulatory Commission  
Washington, D. C. 20555-0001

ATTENTION: Document Control Desk

Subject: Duke Energy Corporation

McGuire Nuclear Station Units 1 & 2  
Docket Nos. 50-369, 50-370

Catawba Nuclear Station Units 1 & 2  
Docket Nos. 50-413, 50-414

Response to NRC Requests for Additional Information  
on License Amendment Requests for McGuire and  
Catawba Nuclear Stations

This submittal contains information that Duke Energy Corporation considers PROPRIETARY and is being made pursuant to 10CFR 2.790.

By letters dated December 9, 1998 and January 5, 1999 the NRC requested additional information on Duke Energy Corporation's July 22, 1998 license amendment requests (LARs) for the McGuire Nuclear Station, Units 1 & 2; and the Catawba Nuclear Station, Units 1 & 2 Technical Specifications. These LARs would permit use of Westinghouse fuel at McGuire and Catawba. Topical Report DPC-NE-2000P/DPC-NE-2009 was also included in the July 22, 1998 Duke submittal.

The thirteen questions contained in the December 9, 1998 NRC letter, and the corresponding Duke answers, are provided in the attachments to this letter. A proprietary version and a non-proprietary version of the Duke response are attached to this letter.

Some of the information contained in Attachment 1 is considered proprietary. In accordance with 10CFR 2.790, Duke Energy Corporation requests that this information be withheld from public disclosure. An affidavit which attests to the

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proprietary nature of the affected information is included with this letter. A non-proprietary version of the Duke response is included as Attachment 2 to this letter.

Please address any comments or questions regarding this matter to J. S. Warren at (704) 382-4986.

Very truly yours,



M. S. Tuckman

Attachments

xc (w/o Attachment 1):

Mr. L. A. Reyes, Regional Administrator  
U. S. Nuclear Regulatory Commission - Region II  
Atlanta Federal Center  
61 Forsyth St., SW, Suite 23T85  
Atlanta, Georgia 30303

Mr. F. Rinaldi, Senior Project Manager  
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Mr. P. S. Tam, Senior Project Manager  
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Mr. S. M. Shaeffer  
NRC Senior Resident Inspector  
McGuire Nuclear Station

Mr. D. J. Roberts  
NRC Senior Resident Inspector  
Catawba Nuclear Station

AFFIDAVIT

1. I am Executive Vice President of Duke Energy Corporation; and as such have the responsibility for reviewing information sought to be withheld from public disclosure in connection with nuclear power plant ~~licensing~~, and am authorized on the part of said Corporation (Duke) to apply for this withholding.
2. I am making this affidavit in conformance with the provisions of 10CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke's application for withholding, which accompanies this affidavit.
3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.
4. Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.
  - (i) The information sought to be withheld from public disclosure is owned by Duke and has been held in confidence by Duke and its consultants.
  - (ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs relative to a method of analysis that provides a competitive advantage to Duke.



M. S. Tuckman

(Continued)

- (iii) The information was transmitted to the NRC in confidence and under the provisions of 10CFR 2.790, it is to be received in confidence by the NRC.
- (iv) The information sought to be protected is not available in public to the best of our knowledge and belief.
- (v) The proprietary information sought to be withheld in this submittal is that which is marked in the proprietary version of the Duke response to NRC requests for additional information dated December 9, 1998 and January 5, 1999. The subject of these requests for additional information is a Duke license amendment request dated July 22, 1998 and accompanying topical report designated DPC-NE-2009P, *Duke Power Company Westinghouse Fuel Transition Report*. The information of concern is omitted from the non-proprietary version of the Duke response. This information enables Duke to:
  - (a) Respond to Generic Letter 83-11, *Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions*.
  - (b) Perform core design, fuel rod design, and thermal-hydraulic analyses for the Westinghouse Robust Fuel Assembly design.
  - (c) Simulate UFSAR Chapter 15 transients and accidents for McGuire and Catawba Nuclear Stations.
  - (d) Perform safety evaluations per 10CFR50.59.
  - (e) Support Facility Operating Licenses/Technical Specifications amendments for McGuire and Catawba Nuclear Stations.



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M. S. Tuckman

(Continued)

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

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

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

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

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



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M. S. Tuckman

(Continued)

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M. S. Tuckman, being duly sworn, states that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth within are true and correct to the best of his knowledge.

M. S. Tuckman

M. S. Tuckman, Executive Vice President

Subscribed and sworn to before me this 28<sup>TH</sup> day of  
JANUARY, 1999

Mary P. Melus  
Notary Public

My Commission Expires:  
JAN 22, 2001

SEAL

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bxc (w/o Attachment 1):

L. A. Keller

M. T. Cash

G. D. Gilbert

K. L. Crane

K. E. Nicholson

R. H. Clark

G. B. Swindlehurst

D. E. Bortz

Catawba Owners: NCMPA-1, NCEMC, PMPA, SREC

Catawba Document Control File (T. K. Pasour)

Catawba RGC File 801.01 (T. K. Pasour)

ELL

Attachment 2

Response to NRC Requests for Additional Information Dated December 9, 1998 and  
January 5, 1999 Applicable to Duke Energy Corporation License Amendment Requests  
Dated July 22, 1998

\*\*\* Non-Proprietary Version \*\*\*

**Attachment 2  
(Non-Proprietary)**

1. Section 3.2 of DPC-NE-2009P states that conceptual transition core designs using the Robust Fuel Assembly (RFA) design have been evaluated and show that current reload limits remain bounding with respect to key physics parameters, and that in the event that one of the key parameters is exceeded, the evaluation process described in DPC-NE-3001-PA would be performed.

(a) Describe the evaluation and the result of the conceptual transition core design.

(b) Based on the statement, it appears that the evaluation process described in DPC-NE-3001-PA will not be performed unless one of the key parameters is exceeded. Without actual analysis of the RFA transitional or full cores, how is it determined that any of the key parameters is exceeded?

***Response 1a:***

Conceptual Westinghouse RFA transition core designs were setup and evaluated using NRC approved codes and methods. The evaluation performed considered the effects of partial and full RFA cores. The purpose of the evaluation was to determine the acceptability of the current licensing bases transient analyses. Key safety parameters were calculated for the conceptual core designs and compared against reference values assumed in the Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analyses. Examples of some of the key parameters calculated include Doppler temperature coefficients, moderator temperature coefficients, control bank worth, individual rod worths, boron concentrations, differential boron worths and kinetics data. A summary of the key parameters important to the licensing bases transient analyses are provided in Table 2-1 of DPC-NE-3001. The evaluation demonstrated the expected neutronic similarities between reactor cores loaded with Westinghouse RFA fuel and with Mk-BW fuel, and the acceptability of key safety parameters assumed in the UFSAR Chapter 15 accident analyses.

***Response 1b:***

Key physics parameters important to the UFSAR Chapter 15 accident analyses are calculated for each reload core using NRC approved methodology to confirm that these parameters are bounded by values assumed in the reference UFSAR Chapter 15 accident evaluations. This check is always performed for each new core design. If the key safety parameters assumed in the reference safety analysis are determined to bound the reload core values, then no additional analyses are required. However, if a key physics parameter is not bounded by the reference value, then the affected accidents will be re-analyzed using the new key physics parameter, or the core will be re-designed to produce an acceptable result.

**Attachment 2  
(Non-Proprietary)**

2. To demonstrate that the currently approved CASMO-3/SIMULATE-3P methods and nuclear uncertainties in DPC-NE-1004-PA are applicable to the RFA design, Section 3.2 cites the analyses performed using Sequoyah Unit 2 Cycles 5, 6 and 7, as well as a 10 CFR 50.59 unreviewed safety question (USQ) evaluation. It is stated that the Sequoyah cores were chosen because they are similar to McGuire and Catawba and contained both Integral Fuel Burnable Absorber (IFBA) and Wet Annular Burnable Absorber fuel. Table 3-1 provides the statistical analysis results of nuclear uncertainty factors, which show they are bounded by the uncertainty factors of DPC-NE-1004A.
- (a) Describe any difference between the McGuire/Catawba RFA cores and the Sequoyah cores analyzed. Describe why these differences would not affect the applicability of the analyses of the Sequoyah cores to McGuire and Catawba.
  - (b) Provide the comparison of the analysis results with measured data of boron concentrations, rod worths, and isothermal temperature coefficients.
  - (c) Describe the details and results of the 10 CFR 50.59 USQ evaluation.

*Response 2a:*

The primary reason for benchmarking the Sequoyah Unit 2 cores was to confirm the fidelity of the CASMO-3/SIMULATE-3 code suite for analyzing reactor cores containing integral fuel burnable absorbers (IFBA). While the introduction of the IFBA product is not considered a major design change, and therefore the benchmarking of this product is not required by the SER requirements of DPC-NE-1004-PA, a conservative approach was adopted to perform benchmark calculations to confirm the acceptability of the current nuclear uncertainty factors. Benchmark calculations were performed using measured data from Sequoyah Unit 2 Cycles 5, 6 and 7.

The Westinghouse Nuclear Design Reports for Sequoyah Unit 2 Cycles 5, 6, and 7, the McGuire and Catawba Updated Final Safety Analysis Report (UFSAR) and Section 2.0 of DPC-NE-2009 were reviewed to determine the differences between the Sequoyah cores analyzed in the benchmark calculations, and McGuire/Catawba RFA core designs. A list of differences is provided below.

- a. The Sequoyah cores modeled and analyzed in the Sequoyah benchmark calculations contained the Westinghouse Vantage-5H (V5H) fuel design. The V5H fuel design is geometrically (ie. pellet diameter, gap and clad thickness and assembly envelope) equivalent to the RFA fuel design to be used in the McGuire and Catawba cores. Differences between the V5H and RFA fuel design are primarily mechanical and, as a result, do not impact the nuclear characteristics of the fuel assemblies. Specific differences between the V5H and RFA fuel design are summarized below.
  - Zirlo™ is used for the fuel rod clad, guide tubes, instrument tubes and mixing grids in the RFA fuel design. The V5H design uses Zr-4 for these components.
  - The RFA fuel design has thicker instrument and guide tubes than the V5H design in order to improve structural stability.
  - The grid design for the RFA design has been modified (optimized vane angles and window size) to improve thermal performance.

**Attachment 2  
(Non-Proprietary)**

- The RFA design Duke intends to use has a pre-oxide coating on the bottom of the fuel rods, longer fuel rod end plugs and a protective bottom grid. The V5H fuel design used at Sequoyah did not have these features.
- The RFA design employs intermediate mid span mixing grids. The V5H design used at Sequoyah did not use mid span mixing grids.

Neutronically, Zirlo™ and Zr-4 are equivalent. The changes in instrument tube and guide tube thickness does not impact core modeling as long as they are accounted for in the generation of cross sections and few group constants. The pre-oxide coating does not impact the modeling of the fuel rod or the neutronic properties of Zirlo™. The fuel rod end plugs are neutronically unimportant because they are located outside of the active fuel region. The mixing grids are specifically accounted for in the neutronics models, therefore, the use of a modified grid design, the addition of the protective bottom grid and mid span mixing grids should not impact model performance. In summary, the differences in the RFA and V5H fuel designs are primarily mechanical and do not impact the nuclear performance of the fuel assembly. Design features that do impact the neutronics (ie. mid span mixing grids) are specifically accounted for in the nuclear models. Therefore, the results and conclusions reached based on the analysis of the Sequoyah core designs are applicable to the RFA fuel design.

- b. The Sequoyah cores that were benchmarked contained both 1.0x and 1.5x IFBAs with rod patterns containing between 48 and 128 IFBA rods. The IFBA loadings (1.0x and 1.5x) and the number of IFBA rods per assembly are representative of the IFBA loadings and the number of IFBA rods expected to be used in McGuire and Catawba core designs. However, the IFBA rod patterns used in the Sequoyah core designs and the IFBA rod patterns that will be used in the McGuire and Catawba core designs are different. The changes in IFBA rod patterns are the result of Westinghouse optimizations that were performed [

] The optimized IFBA rod patterns will be used in the McGuire and Catawba RFA core designs. In addition, all combinations of IFBA loading and rod patterns are explicitly modeled to account for the impact of any design change in the analysis of each reload core design.

The Sequoyah benchmark calculations that were performed demonstrate the acceptability of the CASMO-3/SIMULATE-3 model to accurately calculate core reactivity, reactivity parameters and power distributions for representative IFBA rod loadings and rod configurations. Changes in the IFBA rod configurations primarily affect intra-assembly peaking and not integral and local nodal power distributions which are the parameters that are measured. Consequently, the results from the benchmark analysis are not expected to change as the result of changing the IFBA rod pattern design.

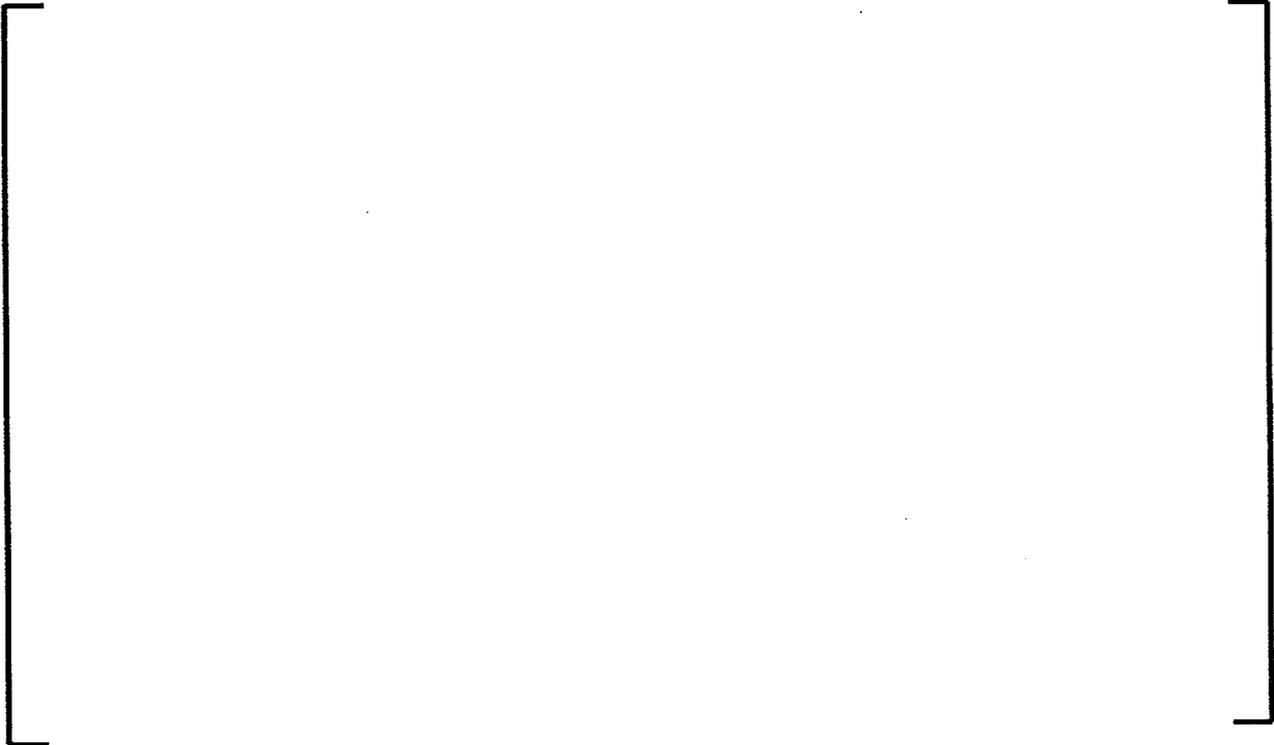
- c. The fuel management strategy (low leakage – ring of fire core designs), the number of fuel assemblies in the reactor core and the core power rating are the same between McGuire, Catawba and Sequoyah. However, there are differences in the reactor coolant flow rate and core inlet temperature. The reactor coolant flow rate at Sequoyah is approximately 3.0% less than at McGuire or Catawba. The core inlet temperature at Sequoyah is ~547°F versus ~555°F at McGuire and Catawba. Core inlet flow and temperature are input variables to the nuclear model and are therefore specifically accounted for. As a result, the performance of the nuclear model and the applicability of the benchmark results are not expected to change due to the aforementioned core inlet flow and temperature differences.

**Attachment 2  
(Non-Proprietary)**

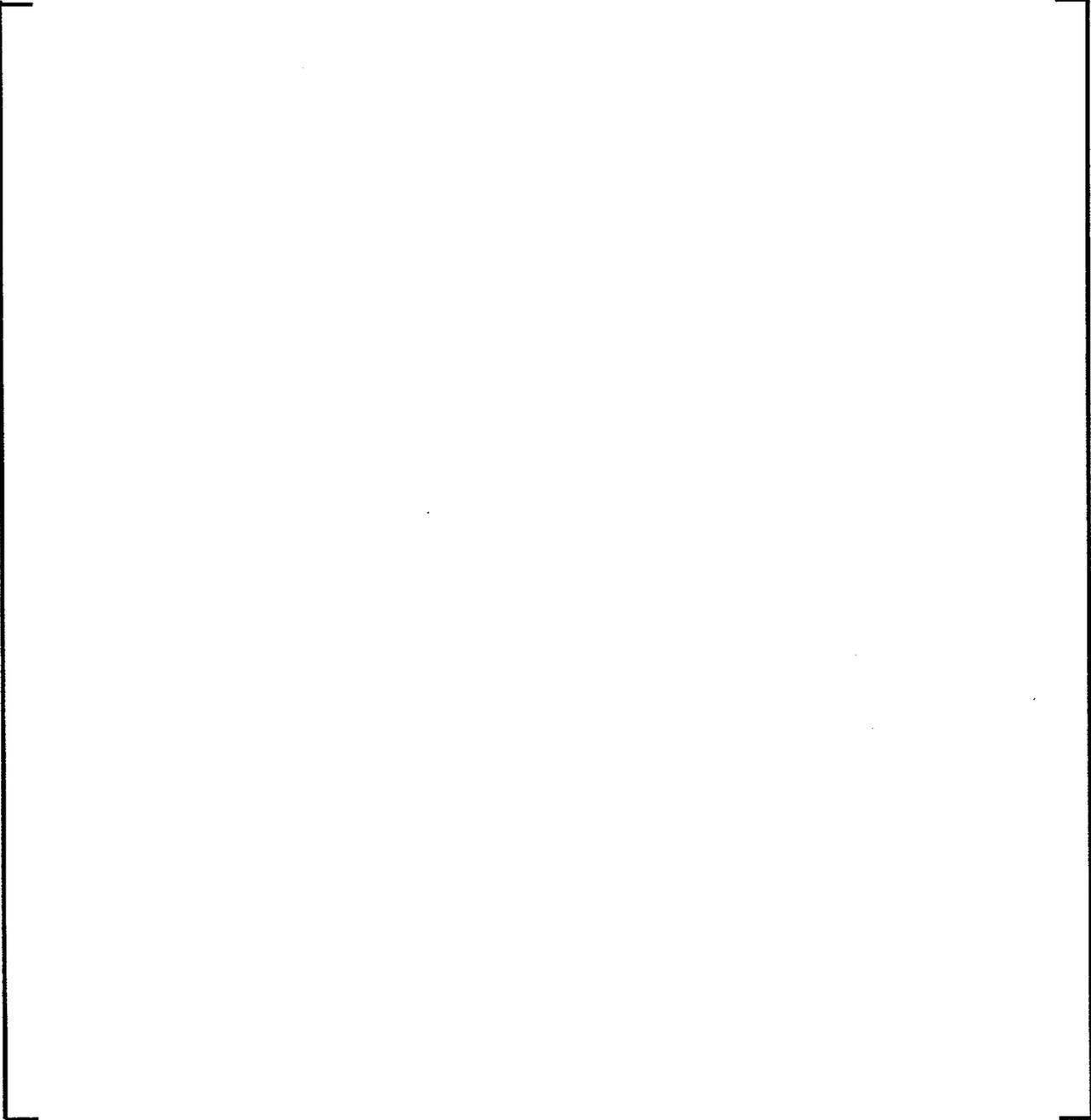
- d. The Control Bank (Bank D) for Sequoyah Unit 2 is comprised of 9 control rods versus 5 control rods for McGuire and Catawba. Since control bank locations are specifically modeled, and because during normal operation control banks are positioned near all rods out (ARO), the impact of this difference on the results of the benchmark analysis is negligible.

***Response 2b:***

Comparisons between Duke predicted and measured zero power physics testing (ZPPT) results are shown below for Sequoyah Unit 2 Cycles 5, 6 and 7. The ZPPT results included comparisons of critical boron concentrations, control rod worths and isothermal temperature coefficients. Excellent agreement between predicted and measured results is generally observed. The large percent differences between predicted and measured control rod worths for Control Bank A in cycles 5 and 6 is primarily the result of the low worth of these banks and to a lesser extent a slight mis-prediction (~1.0%) in the local power distribution. The observed difference in the worth for Control Bank B in cycle 6 is also the result of a slight mis-prediction in the local power distribution and possibly measurement error. However, the observed differences are well within the test acceptance criteria for individual bank worths of +/-30% or 200 pcm, whichever is greater.



**Attachment 2  
(Non-Proprietary)**



***Response 2c:***

A 10CFR 50.59 evaluation was performed to determine if any Unreviewed Safety Questions (USQs) exists when the current methodology is applied to a fuel design that differs from those previously benchmarked and documented in topical report DPC-NE-1004A. For the Duke Power Westinghouse designed nuclear plants, DPC-NE-1004A is considered applicable to Westinghouse OFA, Standard, and FCF Mark-BW (similar to Westinghouse Standard) fuel. The November 1992 SER to this topical stipulated that *“the application of CASMO-3 and SIMULATE-3P to fuel designs that differ significantly from those included in the topical data base should be supported by additional code validation to ensure that the DPC-NE-1004 methodology and uncertainties apply.”* The fuel type evaluated in this 10CFR 50.59 evaluation was the Westinghouse

**Attachment 2**  
**(Non-Proprietary)**

Performance Plus fuel type (similar to Westinghouse Standard and RFA fuel) with integral fuel burnable absorber (IFBA). The integral fuel burnable absorber consists of a thin coating of  $ZrB_2$  applied directly to fuel pellets of selected fuel rods. The analysis is applicable to the Westinghouse RFA fuel design as discussed in the answer to question 2b.

The results of the evaluation concluded that the methodology described in DPC-NE-1004A is applicable to fuel containing IFBA coated fuel pins. This conclusion is based on the results of benchmark calculations that showed code performance commensurate with that described in DPC-NE-1004A. Power distribution uncertainty factors calculated for fuel containing IFBA coated fuel rods, based on a 95% probability and confidence level, were bounded by uncertainty factors approved by the NRC in DPC-NE-1004A. Consequently, the introduction of IFBA fuel will not change the power peaking uncertainties assumed in the analysis of Updated Final Safety Analysis Report (UFSAR) Chapter 15 accidents. Therefore, it can be concluded from a nuclear design perspective that the consequences of UFSAR accidents previously evaluated are not increased and the margin to safety as defined in the bases to Technical Specifications is not decreased. In addition, safety margin will be maintained in future analyses through the application of a conservative combination of uncertainty factors. There are no USQs associated with this change.

**Attachment 2  
(Non-Proprietary)**

3. Section 3.2 states that (1) in all nuclear design analysis, both the RFA and the Mark-BW fuel are explicitly modeled in the transition cores, and (2) when establishing Operating and reactor protection system limits (i.e., loss of coolant accident (LOCA) kw/ft, departure from nucleate boiling (DNB), centerline fuel melt (CFM), transient strain), the fuel specific limits or a conservative overlay of the limits are used. Please elaborate on the mixed core model for nuclear design analyses, and how fuel-specific limits are used.

**Response:**

The mixed core model used in the evaluation of transition cores containing RFA and Mark-BW fuel is based on the same methodology that is used to setup a nuclear model for a reactor cores containing a single fuel type. A SIMULATE-3 model is developed for each reload core design in accordance with the methodology described in DPC-NE-1004A. For mixed cores, this model contains cross sections and few group constants for each unique combination of fuel type (ie. RFA or Mark-BW), enrichment and burnable poison loading and geometry. Cross sections and few group constants are derived from [ ] CASMO-3 calculations. The SIMULATE-3 model is used to confirm the acceptability of key physics parameters assumed in UFSAR Chapter 15 accident analyses and to develop core power distributions used in the evaluation of LOCA, DNB, transient strain and centerline fuel melt limits.

The generation of core power distributions for the development of core operational axial flux difference (AFD) limits and the  $f(\Delta I)$  portion of the over-power delta-T and over-temperature delta-T trip functions (i.e. RPS limits) are conservatively performed using SIMULATE-3 based on the methodology described in DPC-NE-2011PA. The power distributions developed during this process are compared against fuel specific Mark-BW and RFA LOCA, DNB, CFM and transient strain limits by assigning specific Mark-BW and RFA limits to each fuel type. Mark-BW and RFA fuel limits are developed using NRC approved methodologies. If positive margin exists to all limits, then no changes are made to operational AFD, or the RPS limits used in the development of the  $f(\Delta I)$  trip functions. If any of the limits are exceeded, then either (1) the AFD or RPS limits are reduced to produce positive margin to all limits, (2) a specific analysis is performed on the out-of-limit parameter, or (3) the core is redesigned.

In some instances it may be desirable to develop a single composite set of limits that can be used to evaluate both fuel types. For this scenario, a conservative overlay of Mark-BW and RFA limits would be performed to develop a single set of limits that would be applicable to both Mark-BW and RFA fuel. Either of the above mentioned approaches is equally valid.

**Attachment 2  
(Non-Proprietary)**

4. Section 5.2 states that in using the VIPRE-01 code for the reactor core thermal-hydraulic analysis, the reference power distribution based on a 1.60 peak pin from DPC-NE-2004P-A, Rev.1, was used.
- (a) The report states that this reference pin power distribution “was” used. Will it be used for future RFA reload analyses?
  - (b) Does the reference pin power distribution used in the core thermal-hydraulic analyses bound all power distribution for the RFA cores for future reload cycles?

***Response 4a:***

The reference power distribution given in DPC-NE-2004P-A, Rev. 1 will be used in all future RFA analyses. This radial pin power distribution (the relationship of the peak pin to the remaining fuel pins in the highest power fuel assembly) used in DPC-NE-2009P and previous topical reports will not be modified. This maintains the relative radial power distribution the same as previously approved. There are no plans to change this distribution.

The peak pin value, however, could be increased in the future to utilize the increased thermal performance available in the RFA design. For DNB analyses using the Maximum Allowable Peaking (MAP) methodology described in DPC-NE-2004P-A, Rev. 1, the key DNB parameter is the reference power distribution, not the peak pin power. The peak pin power is only meaningful when all other DNB parameters are specified (axial peak location and magnitude, core power level, RCS pressure, flow rate, and temperature). The reference power distribution is used to create the Maximum Allowable Peaking (MAP) limits that ensure the required level of DNBR protection is provided. The MAP limits define the maximum allowable peak pin as a function of axial peak. The reference power distribution is used consistently in all DNB analyses (core DNB limit lines, transient analyses, SCD statepoint determinations, etc.). Any change in the peak pin value will be evaluated in all DNB analyses and will be reflected in the Maximum Allowable Peaking limits provided in the COLR for each reload cycle.

The ability to increase the peak pin value is a result of a new fuel design, additional design features, a new or modified CHF correlation, or changes to the analysis conditions. If the performance improvement is related to fuel hardware or correlation change, a submittal is made to the NRC and approval required prior to use. If the change is to the analysis conditions and no methodology is modified, the change can be implemented through the 10CFR50.59 process. In either case, any increase in the peak pin value is not made unless all analyses and related licensing limits are verified to be conservatively satisfied.

***Response 4b:***

The reference power distribution used to create the Maximum Allowable Peaking (MAP) limits is used in all steady state generic analyses. This distribution is verified each reload by performing DNB calculations with cycle specific predicted radial pin power distributions. This specific pin distribution comparison between what is predicted for a particular cycle and the generic analysis reference power distribution verifies the conservatism of the reference distribution.

**Attachment 2  
(Non-Proprietary)**

5. Section 5.2 states that in the thermal-hydraulic analysis of the RFA design using VIPRE-01, the two-phase flow correlations will be changed from the Levy subcooled void correlation and the Zuber-Findlay bulk void correlation to the EPRI subcooled and bulk void correlations, respectively. While the sensitivity study provided in the report shows a minimal difference of 0.1% between the minimum DNBRs of 51 RFA CHF test data points calculated with both set of correlations, it was stated in DPC-NE-2004 that the Levy/Zuber-Findlay combination compared most favorably with the Mark-BW test results as the DNBRs of the tests calculated with this combination yielded conservative results relative to the EPRI correlations.
- (a) Discuss whether the EPRI correlations will be used for the RFA design only, or they will also be used for the Mark-BW design.
  - (b) If the EPRI correlations will also be used for Mark-BW design, provide justification for their use.
  - (c) If the Levy/Zuber-Findlay correlations will continue to be used for Mark-BW fuel design, discuss how the VIPRE-01 code will be used to analyze transient mixed cores having both Mark-BW and RFA fuel designs.

***Response 5a:***

The EPRI correlations will only be used in the RFA models in VIPRE-01. The Levy/ Zuber-Findlay combination will be used when modeling Mark-BW fuel.

Duke considers the selection of the two-phase flow correlations to be a very minor effect on DNBR analyses. Mark-BW CHF test data was analyzed with both Levy/Zuber-Findlay and EPRI/EPRI in the same manner as the RFA with comparable results.

***Response 5b:***

See 5(a) above.

***Response 5c:***

The transition core models use the simplified (8 Channel) models to maximize the impact of different fuel types. In the transition core model, the limiting assembly is modeled as an RFA and the rest of the core is modeled as Mark-BW fuel. Since the MDNBR occurs in the limiting assembly, the void correlations are input for the fuel type modeled as the limiting assembly. For the RFA/Mark-BW transition core model, the RFA design is the limiting assembly; thus the EPRI set of correlations are used.

The transition analyses covered a wide range of statepoint fluid conditions and 3-dimensional core power distributions. This matrix of conditions were analyzed using both the EPRI and Levy/Zuber-Findlay correlations with minimal difference in transition core results using either set of void correlations (average difference of <1% in peaking).

**Attachment 2  
(Non-Proprietary)**

6. Section 5.7 describes the use of a transition 8-channel RFA/Mark-BW core model to determine the impact of the geometric and hydraulic differences between the resident Mark-BW fuel and the RFA design, and determine a conservative DNBR penalty to be applied for the transition cores. Table 5-4 presented the statistical DNBRs for the 500 and 5000 case runs for various statepoints including the transition core case of the most limiting statepoint 12. The statistical design limit is chosen to bound both the full RFA cores and RFA/Mark-BW transition cores for the 5000 case runs.
- (a) Why is the statistical design limit value proprietary information?
  - (b) With respect to the statistical core design methodology, describe how the uncertainties of the CHF correlation and the VIPRE code/model are propagated with the uncertainties of the selected parameters of each statepoint for the calculation of the statistical DNBR for each statepoint in Table 5-4.
  - (c) With the statistical design limit specified in Section 5.7, is it your intention to use a full core of RFA in the thermal hydraulic analysis for the transition core without the transition core DNBR penalty factor?

***Response 6a:***

The Statistical Design Limit (SDL) will be changed to non-proprietary. This change will be included when the approved versions of the report are issued.

***Response 6b:***

When a statepoint is selected, all key parameters, including CHF correlation and code/model uncertainties, are randomly varied based on the uncertainty distribution and magnitude. The resulting values of power, pressure, temperature, flow, and 3-D power distribution are used to create the VIPRE-01 input for the cases. After the code is executed and the DNBR calculated for each case, the DNBR value is multiplied by the propagated values for the CHF correlation uncertainty and the VIPRE code/model uncertainty. This final DNBR value for each case (500 or 5000 cases are run for each statepoint) is used to determine the statepoint's statistical DNBR value.

***Response 6c:***

The analysis discussed in the last paragraph of Section 5.7 verified that the statistical DNB limit developed with a full core RFA model is valid for transition RFA/Mark-BW cores. The limiting statepoint (12TR) was evaluated using the RFA/Mark-BW transition core model, confirming that the same statistical design limit can be used for transition and full core analyses.

The transition core DNB penalty factor is determined separately using the RFA/Mark-BW transition core model described in Section 5.7. The DNB penalty is determined by evaluating the effect of the transition core hydraulic behavior on the Maximum Allowable Peaking (MAP) limits calculated for a full RFA core. The resulting DNB penalty is then accounted for in all RFA/Mark-BW transition core DNB analyses.

**Attachment 2  
(Non-Proprietary)**

7. Section 2.0 states that the RFA is designed to be mechanically and hydraulically compatible with the Mark-BW fuel. Table 2.1 provides a comparison of the basic design parameters of the two fuel designs, but does not provide a comparison of the hydraulic characteristics of spacer grids. Section 5.2 states that the VIPRE-01 core thermal-hydraulic analyses were performed with applicable form loss coefficients according to the vendor. Table 5.1 provides general RFA fuel specifications and characteristics without the hydraulic characteristics of the spacer grids.
- (a) Provide comparisons for the thickness, height, and form loss coefficients of the RFA and Mark-BW fuel spacer grids, including mixing-vane and non-mixing vane structural grids, and intermediate flow mixing grids.
  - (b) Provide the form loss coefficients of the spacer grids used in the analyses and in the RFA CHF test assemblies if they are different from the values described in item (a).
  - (c) Describe the procedures to ensure that the form loss coefficients of the RFA grids are comparable to those used in the statistical core design analysis and the CHF tests so that both the WRB-2M CHF correlation DNBR limit and the statistical core design limit are valid.

***Response 7a:***

The grid data is shown in the following table:

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**Attachment 2  
(Non-Proprietary)**

***Response 7b:***

The RFA CHF tests used Mixing Vane (MV) and intermediate flow mixing (IFM) grids representative of the production RFA design fuel assembly. The CHF test sections are a 5x5 rod bundle with either all typical (unit) cells or typical cells with a thimble (guide tube) cell in the center. The form loss coefficients for the CHF test section are calculated for these subchannels and are based on the total 5x5 bundle flow area. Likewise, the fuel assembly subchannel form loss coefficients are calculated based on the fuel assembly flow area. The ratio of thimble/typical cell form loss coefficients, to which DNBR is sensitive, is equivalent for the CHF test section and the production grid (for both MV and IFM grids). Therefore, the CHF test section and production RFA grids are identical with respect to DNBR analyses.

In comparing the test versus production geometry, the vanes and strap features of the respective grid types are consistent. There is one slight difference between one of the CHF test sections and the production fuel assemblies. The thimble OD was 0.474 inches for the thimble CHF rod bundle section tested. The production assembly will have thimbles with an OD of 0.482 inches. The difference in thimble tube OD has negligible impact on the correlation's predictive capability. This difference was addressed in WCAP-15025 and determined to be acceptable.

***Response 7c:***

The RFA analysis was completed with the form loss coefficients supplied in response to Question 7a. The transition core analysis used the RFA and Mark-BW values listed in the table in the respective model locations to accurately capture the hydraulic differences between the fuel types side-by-side incore.

For each batch of fuel manufactured, critical RFA grid dimensions and form loss coefficients are supplied by the vendor to Duke Power. This data, along with other critical reload analysis parameters, are transmitted to Duke, on a batch basis, in a QA document known as the Databook. Upon receipt of the Databook, the fuel design is frozen and may not be changed without Duke Power concurrence. This design notification process, including the process for changes occurring after the batch is frozen, is described in Duke Power Nuclear Engineering Workplace Procedure XSTP-101. The batch specific design information, transmitted in the Databook, will be used to ensure the validity of the Duke VIPRE-01 RFA models and associated SCD limit.

Any changes in the design data will be evaluated to verify that the generic analyses remain valid or the analyses will be revised using the new design data.

**Attachment 2  
(Non-Proprietary)**

8. Section 6.1.3 states that the thermal-hydraulic methodology described in DPC-NE-3000-PA Revision 1, with a simplified core model will be used for thermal-hydraulic analysis for the Updated Final Safety Analysis Report Chapter 15 non-LOCA transients and accidents for the RFA design. It also states that (1) no transition core transient analyses are performed as the results determined in Chapter 5 also apply for transient analyses, (2) the simplified core model of DPC-NE-3000-PA used for transient analyses was originally developed with additional conservatism over the 8-channel model used for steady-state analyses to specifically minimize the impact of changes in core reload design methods or fuel assembly design, and (3) should it be determined in the future that transition core transient analyses are warranted, they will be performed accordingly.
- (a) Explain what additional conservatism is provided in using the simplified core model of DPC-NE-3000-PA.
- (b) What is the criterion/criteria used to determine if transition core transient analyses are warranted? How would it be determined that the criteria have been exceeded without RFA transition core analyses?

***Response 8a:***

The additional conservatism provided in using the simplified core model of DPC-NE-3000-PA is described in detail in Section 3.3.4 of DPC-NE-3000-PA (Reference 6-1 of DPC-NE-2009-P).

***Response 8b:***

Section 6.1.3 states the following. "No transition core transient analyses are performed as the results determined in Chapter 5 also apply for transient analyses. . . . Should it be determined in the future that transition core transient analyses are warranted, they will be performed accordingly." These statements summarize the results of an evaluation that has concluded that based on current information there is no need for performing transition core analyses for transients. The transition core effects on core thermal-hydraulic analyses for transients are adequately assessed by the steady-state core thermal-hydraulic transition core analysis in Chapter 5. The purpose of the second sentence quoted above was to state Duke's intent to evaluate any emerging issues or information, and, if necessary, to re-evaluate the current conclusion that no transient analyses of transition core effects are necessary. Duke does not expect any emerging information to change this conclusion, but Duke will address any such situations in the future.

**Attachment 2  
(Non-Proprietary)**

9. Regarding rod ejection analysis using SIMULATE-3K, Section 6.6.2.2.1 states that the transient response is made more conservative by increasing the fission cross sections in the ejected rod location and in each assembly and by applying “factors of conservatism” in the moderator temperature coefficient, control rod worths for withdrawal and insertion, Doppler temperature coefficient, effective delayed neutron fraction, and ejected rod worth, etc.

- (a) What are the values of the multiplication factors used for fission cross sections, and how are they determined?
- (b) How are the input multipliers “VAL” in Equations 6.1 and 6.2 determined? Does “VAL” have a different value for different parameters, such as MTC or DTC? What are the values for these VALs?
- (c) In Equation 6.1, the X’s are described as “moderator temperatures.” Should they be moderator temperature coefficients?

*Response 9a:*

An iterative process is used to determine the [

] Note

that these multipliers are [ ]

The methodology used to determine the [ ] adjustments is consistent with the power distribution adjustment methodology described in DPC-NE-3001 with one exception. [

]

The BOC and EOC [ ] multipliers are shown in Figures 9-1 and 9-2.

**Figure 9-1  
BOC [ ] Multipliers**

	H	G	F	E	D	C	B	A
8	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
9	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
10	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
11	1.0	1.0	1.0	1.1450	0.9810	1.0445	1.0	1.0
12	1.0	1.0	1.0	0.9810	1.0904	0.9435	1.0	
13	1.0	1.0	1.0	1.0445	0.9435	1.1358	1.0	
14	1.0	1.0	1.0	1.0	1.0	1.0		
15	1.0	1.0	1.0	1.0				



**Attachment 2  
(Non-Proprietary)**

The multipliers calculated for each of the key physics parameters assumed in each of the four rod ejection accidents (ie. BOC HFP and HZP and EOC HFP and HZP) are shown below. These multipliers were developed to produce bounding key physics parameters to ensure a conservative transient response. The multipliers presented are unique to each of the accidents presented in this report [ ]

Parameter	BOC HFP		BOC HZP	
	Multiplier (VAL)	Target Value	Multiplier (VAL)	Target Value
DTC	0.689	-0.90 pcm/°F	0.555	-0.90 pcm/°F
MTC	-0.005	0.0 pcm/°F	-1.247	0.0 pcm/°F
Ejected Rod Worth	1.168	200 pcm	1.029	720 pcm
Trip Worth	0.510	250 pcm	1.650	250 pcm
Beta-effective	0.882	0.0050	0.878	0.0050

Parameter	EOC HFP		EOC HZP	
	Multiplier (VAL)	Target Value	Multiplier (VAL)	Target Value
DTC	0.810	-1.20 pcm/°F	0.666	-1.20 pcm/°F
MTC	0.283	-10.0 pcm/°F	0.478	-10.0 pcm/°F
Ejected Rod Worth	1.055	200 pcm	0.868	900 pcm
Trip Worth	1.073	250 pcm	1.650	250 pcm
Beta-effective	0.768	0.0040	0.763	0.0040

**Response 9c:**

No. The X's in equation 6.1 are moderator temperature. Refer to answer "9b" for additional information.

**Attachment 2  
(Non-Proprietary)**

**10. Regarding the SIMULATE-3K code, there is an optional “frequency transform” approach, under the “Temporal Integration Models,” that can be chosen to separate the fluxes into exponential time varying and predominately spatial components, thus accelerating convergence of the transient neutronic solution and preserving accuracy on a coarser time mesh (see Page 5, Ref. 6-9).**

- (a) What determines when the “frequency transform” approach should be used?**
- (b) What are the consequences of exercising (or not exercising) this option? Please provide technical justification and comparisons of results.**

***Response 10a:***

The frequency transform method is SIMULATE-3K’s default transient neutronics solution option and was used in all of the transient evaluations presented in DPC-NE-2009. This approach was used because it is computationally more efficient and reproduces the results of finite difference methods, which require smaller time steps to achieve the same accuracy as the frequency transform method.

The method used to solve the transient neutronics equations is determined by the code user and used throughout the transient. There is no switching of solution methods during the transient.

***Response 10b:***

There are no physical consequences from using either the frequency transform or finite difference methods to solve the transient neutronic equations since both methods are equally accurate. From theory, the flux variation from one time step to the next is exponential. The frequency transform method takes credit for this behavior, instead of an assumed linear variation in simple finite difference methods. By taking credit for the exponential flux variation, computational efficiency is increased because larger time steps can be taken without loss in accuracy as is the case in traditional methods. Therefore, the frequency transform method is the preferred solution technique because it produces the same answers as finite difference methods, but with reduced code execution time.

Sensitivity studies performed showed no difference in the peak core power or the time of the peak core power for cases where the frequency transform method was turned on and off. Therefore, it can be concluded that there is no consequence of using the frequency transform approach.

Attachment 2  
(Non-Proprietary)

11. The licensing analyses of reload cores with the RFA design will use the methodologies described in various topical reports and revisions for the analyses of fuel design, core reload design, physics, thermal-hydraulics, and transients and accidents, which were approved by NRC for analyses of current Catawba cores not having the RFA design. For example, DPC-NE-1004A, DPC-NE-2011-PA, DPC-NF-2010A and DPE-NE-3001-PA are used for the nuclear design calculations. DPC-NE-2004-PA, DPC-NE-2005-PA, and the VIPRE-01 code are used for the core thermal-hydraulic analyses and statistical core design. DPC-NE-3000-PA, DPC-NE-3001-PA, DPC-NE-3002-A, and RETRAN-02 code are used for non-LOCA transient and accident analyses. Westinghouse small- and large-break LOCA evaluation model described in WCAP-10054-P-A and WCAP-10266-P-A, and related topical reports, are used for the small- and large-break LOCA analyses. Some of these methodologies have inherent limitations, and some have conditions or limitations imposed by the NRC safety evaluation reports in their applications. Provide a list of the inherent limitations, conditions, or restrictions applicable to the RFA core design from all the methodologies to be used for the RFA reload design analyses, and describe the resolutions of these limitations, conditions and restrictions in the applications to the RFA cores and the transitional RFA/Mark-BW cores.

*Response:*

**DPC-NE-1004A, Duke Power Company Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, Rev. 1, April 1998.**

The SER states that this methodology is acceptable for performing reload analyses for B&W 177 and Westinghouse 193 assembly reactor cores, subject to the following restrictions:

- a. The application of CASMO-3 and SIMULATE-3 to fuel designs that differ significantly from those included in the topical data base should be supported by additional code validation to ensure that the DPC-NE-1004A methodology and uncertainties apply.

*Resolution:* While Duke does not consider the introduction of the Integral Fuel Burnable Absorber (IFBA) in the Westinghouse RFA design to be a significant fuel design change, a conservative approach was adopted to confirm the acceptability of current nuclear uncertainty factors because of the availability of IFBA benchmark data. The uncertainty analysis (described in Section 3.2 and in the answer to question 2) confirmed the acceptability of the currently licensed nuclear uncertainty factors for  $F\Delta H$ ,  $Fq$  and  $FZ$  for Westinghouse fuel containing IFBA and WABA burnable absorbers.

- b. The system of codes represented in the topical report must be protected with appropriate quality assurance procedures, subject to auditing by the NRC staff.

*Resolution:* The codes represented in the topical report DPC-NE-1004A are procedurally controlled and are in compliance with the Duke Energy Corporation Quality Assurance Topical Report which is in compliance with the requirements of 10CFR 50, Appendix B and other approved industry standards such as ANSI N45.2-1971 and ANSI N18.7-1976.

**Attachment 2  
(Non-Proprietary)**

**DPC-NE-2011PA, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," March 1990.**

The SER for this methodology imposes the following restrictions:

- a. The application of this methodology is limited to the McGuire and Catawba Nuclear Stations.

*Resolution:* Duke is only using this methodology for McGuire and Catawba

- b. The application of this methodology to other Westinghouse plants would be acceptable provided that plant-specific differences be considered and justified.

*Resolution:* Duke is only using this methodology for McGuire and Catawba. The use of this methodology for application to another Westinghouse unit (or units) would require the submittal of license amendments and NRC approval.

- c. Application of this methodology is contingent upon NRC approval of the Reload Design Thermal-Hydraulic Methodology DPC-NE-2004P-A using the VIPRE-01 code. (Topical Approved)

*Resolution:* The Thermal-Hydraulic Methodology described in Topical Report DPC-NE-2004P-A has been approved.

- d. Calculation of power and xenon distributions are limited to the use of the EPRI-NODE-P and the PDQ-07 codes

*Resolution:* The approval of the Topical Report DPC-NE-1004A allowed the substitution of either the CASMO-3/SIMULATE-3P or the CASMO-3/NODE-P codes in place of the EPRI-NODE-P and PDQ-07.

**DPC-NE-2010A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985.**

The SER for this methodology imposes the following restrictions:

None, with the exception that the methodology in sections 6.3, 7.1, 7.2, 7.3, 7.4 and 7.4.1 were excluded from this report. The replacement methodology is described below:

- a. Section 6.3: Comparison of Cycle Specific Safety Related Physics Parameters

*Resolution:* This methodology was replaced by the methodology described in Topical Report DPC-NE-3001PA.

- b. Section 7.1 – 7.4.1: Three-dimensional peaking analysis

*Resolution:* This methodology was replaced by the methodology described in Topical Report DPC-NE-2011PA.

**Attachment 2  
(Non-Proprietary)**

**DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," Revision 1, February 1997.**

The limitations, conditions or restrictions identified in the SER and TER for DPC-NE-2004, Revision 1, are:

- a. The DPC developed statistical core design methodology, as described in the submittal, is a generic methodology and is conceptually acceptable and generally applicable to other PWR plants; however, the approval we recommend at this time is only for McGuire and Catawba Nuclear Stations due to DPC's use of the specific uncertainties and distributions based upon plant data and its selection of statepoints used for generating the statistical design limit.

*Resolution:* The RFA SCD analysis presented in DPC-NE-2009 is only for McGuire and Catawba.

- b. Either the response surface model (RSM) must be re-evaluated or the "simplified method" for determining an SDL using VIPRE-01 directly must be used whenever any of the following occur:
- a significant change is made in the fuel assembly design
  - a new or revised CHF correlation is developed
  - operating conditions outside the range of conditions considered in the development of the RSM

The licensee is further required to make a submittal to the NRC for review if a new SDL is calculated as a result of conditions outside the range of conditions considered in the development of the RSM.

*Resolution:* The RSM was not used to calculate the SDL in DPC-NE-2009. The RFA analysis presented in DPC-NE-2009 calculates the SDL for the RFA fuel with the WRB-2M CHF correlation as per the "simplified method" referenced in DPC-NE-2004, Rev 1 which is the SCD calculation methodology subsequently approved in DPC-NE-2005, Rev 1.

- c. Whenever DPC intends to use other CHF correlations, power distribution, fuel pin conduction model, or any other input parameters and default options which were not part of the original review of the VIPRE-01 code, DPC must submit its justification for NRC review and approval.

*Resolution:* DPC-NE-2009 identifies the VIPRE-01 modeling requirements as well as the CHF correlation and statistical analysis limit.

- d. Core bypass flow is cycle dependent. DPC will verify, in future applications, that its use of a particular core flowrate resulting from a bypass flowrate for that cycle is bounded by the range of values used in the subject topical report. Otherwise DPC will reassess the need for regeneration of a new response surface model.

**Attachment 2  
(Non-Proprietary)**

*Resolution:* The value of core bypass flow used in the generic RFA SCD analysis is expected to bound the values for all reload cores. The core bypass flow will be verified on a cycle by cycle basis to ensure conservatism.

**DPC-NE-2005P-A, Thermal-Hydraulic Statistical Core Design Methodology”, Revision 1, November 1996.**

The limitations, conditions or restrictions identified in the SER and TER for DPC-NE-2005P-A are:

- a. The statistical core design (SCD) methodology developed by DPC, as described in the submittal (DPC-NE-2005), is direct and general enough to be widely applicable to any pressurized-water reactor (PWR) fuel or reactor, provided that the VIPRE-01 methodology is approved with the use of the core model and correlations including the critical heat flux (CHF) correlation subject to the conditions in the VIPRE safety evaluation report (SER). DPC committed in their topical report that its use of specific uncertainties and distributions will be justified on a plant specific basis, and also that its selection of statepoints used for generating the statistical design limit will be justified to be appropriate. The methodology is approved only for use in DPC plants.

*Resolution:* Addressed in Chapter 5 of DPC-NE-2009. The RFA analysis presented in DPC-NE-2009 is only for McGuire and Catawba.

- b. Of the two DNBR limits, only the use of the single, most-conservative DNBR limit is approved.

*Resolution:* Use of two DNBR limits was not requested in this submittal. The single DNBR limit stated for use for RFA fuel in full cores or transition cores will be used for all statepoints within the conditions listed in Table 5-5.

**WCAP-15025, “Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids”.**

The limitations, conditions or restrictions identified in the SER and for WCAP-15025 are:

- a. Since WRB-2M was developed from test assemblies designed to simulate Modified 17x17 Vantage 5H fuel the correlation may only be used to perform evaluations for fuel of that type without further justification. Modified Vantage 5H fuel with or without modified intermediate flow mixer grids may be evaluated with WRB-2M.

*Resolution:* The SCD analysis presented in DPC-NE-2009 is for the RFA design, which includes the modified low pressure drop structural mid-grids and modified intermediate flow mixing grids (see Chapter 2 of DPC-NE-2009). The WRB-2M CHF correlation is used for all DNBR calculations on the RFA.

**Attachment 2  
(Non-Proprietary)**

- b. Since WRB-2M is dependent on calculated local fluid properties these should be calculated by a computer code that has been reviewed and approved by the NRC staff for that purpose. Currently WRB-2M with a DNBR limit of 1.14 may be used with the THINC-IV computer code. The use of VIPRE-01 by Westinghouse with WRB-2M is currently under separate review.

*Resolution:* The DNB analyses in DPC-NE-2009 are performed with VIPRE-01. As stated in Section 2.3 and 3.3 of WCAP-15025, both VIPRE-01 and THINC-IV were used to analyze the CHF test data. Tables A-1 to A-4 in the Appendix to WCAP-15025 show the local fluid conditions calculated with VIPRE-01. Also, the SER for WCAP-15025 states that the “results of the THINC-IV analyses agreed with those from VIPRE-01.” Additionally, VIPRE-01 was approved for use in thermal/ hydraulic analyses at McGuire and Catawba in DPC-NE-2004, Revision 1. Based on this, Duke has used and will continue to use VIPRE-01 to perform all RFA analyses.

- c. WRB-2M may be used for PWR plant analyses of steady state and reactor transients other than loss of coolant accidents. Use of WRB-2M for loss of coolant accident analysis will require additional justification that the applicable NRC regulations are met and the computer code used to calculate local fuel element thermal/hydraulic properties has been approved for that purpose.

*Resolution:* The RFA LOCA analysis is not described in DPC-NE-2009. The LOCA analysis is performed by the fuel vendor with the approved correlations specified by the vendor’s methodology. The CHF correlation used in the LOCA analysis is listed in WCAP-8301.

- d. The correlation should not be used outside the range of applicability defined by the range of the test data from which it was developed. This range is listed in Table 1.

*Resolution:* Table 1 is listed below for reference.

Parameter	Range
Pressure (psia)	$1495 \leq P \leq 2425$
Local Mass Velocity (Mlbm/hr-ft <sup>2</sup> )	$0.97 \leq G \leq 3.1$
Local Quality	$-0.1 \leq X \leq 0.29$
Heated length, inlet to CHF location (ft)	$L_h \leq 14$
Grid Spacing (in)	$10 \leq gsp \leq 20.6$
Equivalent hydraulic diameter (in)	$0.37 \leq D_e \leq 0.46$
Equivalent heated diameter (in)	$0.46 \leq D_h \leq 0.54$

The WRB-2M CHF correlation was used for all RFA DNBR calculations. The fluid parameter ranges (first two items on the parameter list) are confirmed by the statepoint selection listed in Chapter 5 of DPC-NE-2009. The fuel design related parameters (last four items on the parameter list) are confirmed implicitly by the fuel model. The local quality limit is verified for each analysis.

**Attachment 2  
(Non-Proprietary)**

**DPC-NE-3000-PA, "Thermal-Hydraulic Transient Analysis Methodology," Revision 2, December 1997.**

The original SER dated November 15, 1991 lists the following conditions (Section 3.0). The SER for Revision 1 dated August 8, 1994 does not have any limitations or conditions for McGuire and Catawba. The SER for Revision 2 dated October 14, 1998 does not have any new limitations or conditions.

- a. With respect to analyzing transients which result in a reduction in steam generator secondary water inventory, use of the RETRAN-02 steam generator modeling is acceptable, only for transients in that category for which the secondary side inventory for the effective steam generator(s) relied upon for heat removal never decreases below an amount which would cover enough tube height to remove decay heat.

*Resolution:* By letter dated September 25, 1998 (G. R. Peterson to NRC Document Control Desk), Duke notified the NRC of a new RETRAN-02 steam generator model which addresses this SER condition for the Catawba Unit 2 UFSAR Section 15.2.7 loss of normal feedwater analysis. This submittal is currently under NRC review. The subject of this condition is not applicable for all other UFSAR Chapter 15 transients and accidents for McGuire and Catawba.

- b. All generic limitations specified in the RETRAN-02 SER.

*Resolution:* By letter dated June 3, 1991 (M.S. Tuckman to NRC Document Control Desk), Duke responded to the generic limitations specified in the RETRAN-02 SER in the response to Question #29. This response along with subsequent methodology revisions (including the revisions in DPC-NE-2009-P) have all been submitted to the NRC. Later RETRAN-02 SERs were reviewed and it was determined that three new SER conditions exist for the RETRAN-02 MOD005.0 code version (SER dated November 1, 1991). The responses to these conditions are as follows:

- 1) The user must justify, for each transient in which the general transport model, the selected degree of mixing with considerations as discussed in Section 2.1 of this SER.

Response: Topical report DPC-NE-3001-P described the application of the general transport model in the Duke methodology. The topical report was reviewed and approved by the NRC.

- 2) The user must justify, for each use of the ANS 1979 standard decay heat model, the associated parameter inputs, as discussed in Section 2.2 of this SER.

Response: The Duke modeling of decay heat as described in topical report DPC-NE-3002-A is based on the ANSI/ANS-5.1-1979 standard plus a two-sigma uncertainty. This decay heat modeling approach is standard in the industry for non-LOCA analyses, and meets the intent of this condition. The NRC has reviewed and approved DPC-NE-3002-A.

- 3) Because of the inexactness of the new reactivity edit feature, use of values in the edit either directly or as constituent factors in calculations of parameters for comparison to formal performance criteria must be justified.

**Attachment 2  
(Non-Proprietary)**

Response: The Duke methodology does not use the reactivity edit feature in the manner that is the subject of this condition. Therefore this condition is not applicable.

- c. Determination of acceptability is based upon review of selection of models/correlations for transients involving symmetric core neutronic and thermal-hydraulic conditions only. Thus, VIPRE-01 models are approved for use in analyzing symmetric transients only.

*Resolution:* The DPC-NE-3001-PA topical report submitted VIPRE-01 models for transients involving asymmetric core neutronic and thermal-hydraulic conditions. NRC approval of DPC-NE-3001-PA closed out this condition.

- d. When using the DPC developed SCD method, the licensee must satisfy the conditions set forth in the staff's safety evaluation of DPC-NE-2004.

*Resolution:* Duke responded to this question by letter dated August 29, 1991 (M. S. Tuckman to NRC Document Control Desk). Attachment 2 to this letter addresses the applicable conditions and how these conditions are met, and is summarized as follows. The first condition requiring submittal of models for asymmetric transients was met with the submittal of DPC-NE-3001. The second condition required a transition core penalty to be applied. The details of the transition core penalty modeling were presented in the response. The third condition required modeling to avoid errors related to the use of the subcooled boiling models. A commitment to properly apply this model was made. The fourth condition required submittal of the BWCMV correlation prior to use, which was done. The fifth condition required future submittal of any methodology changes to important inputs and models such as different CHF correlations, power distributions, input options, etc. Duke observes this condition and has and will submit such methodology changes prior to implementation. The sixth condition requires that the core bypass flow be determined and justified on a cycle-by-cycle basis. Duke commits to confirming that the core bypass flow for each reload cycle will be bounded by the core bypass flow assumed in the analyses.

- e. Whenever DPC intends to use other CHF correlations, power distribution, fuel pin conduction model or any other input parameters and default options which were not part of the original review of the VIPRE-01 code, DPC must submit its justification for NRC review and approval.

*Resolution:* Duke recognizes the requirements of this condition and continues to meet this condition. For example, Revision 2 to DPC-NE-3000-P (Letter, M. S. Tuckman to NRC Document Control Desk, December 23, 1997) submitted revised VIPRE-01 methodology to include the Mk-B11 fuel assembly design and the BWU-Z CHF correlation, along with other minor changes. Future revisions to Duke topical reports will be submitted as necessary per the requirements of this condition.

**Attachment 2**  
**(Non-Proprietary)**

**DPC-NE-3001P-A, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology," November 1991.**

The SER dated November 15, 1991 lists the following limitations (Section 4.0).

- a. The licensing application of the SIMULATE-3P static methods for determining the key safety parameters requires NRC approval of the reference topical report, DPC-NE-1004 (Section 3.1)

*Resolution:* Topical report DPC-NE-1004-A was reviewed and approved by the NRC. The latest NRC SER for DPC-NE-1004-A, Revision 1, is dated April 26, 1996

- b. The licensing application of the DPC-NE-3001-P transient analysis methods requires NRC approval of MOD005 of RETRAN-02 for boron transport calculations (Section 3.5)

*Resolution:* The NRC SER for RETRAN-02 MOD005.0 was dated November 1, 1991.

- c. The licensing application of the DPC-NE-3001-P transient analysis methods requires NRC approval of the thermal-hydraulics topical report DPC-NE-3000 (Section 3.5)

*Resolution:* The NRC SER (McGuire/Catawba scope) for DPC-NE-3000-PA was dated November 15, 1991.

**DPC-NE-3002-A, "UFSAR Chapter 15 System Transient Analysis Methodology," Revision 2, December 1997.**

The original SER dated November 15, 1991 lists the following conditions (Section 3.0).

- a. DPC's Statistical Core Design methodology treats seven state variables as key parameters. Four of these variables were accounted for in this topical report. Of the remaining parameters, the power factors are also input items for systems analysis, which was not presented in the topical report. Similarly, reactivity feedback was not discussed in this report. Both of these parameters can significantly influence the course of the transient. Therefore, when application of the philosophical approach reported in this topical report is made and submitted for NRC review and approval, review should be made of the modeling of power and reactivity feedback, and to assure that such modeling has no adverse impact on the other modeling described herein.

*Resolution:* The power factors used in the models are described in the DPC-NE-3000 topical report, which has been reviewed and approved by the NRC. The reactivity feedback modeling is described in the DPC-NE-3001 topical report, which has been reviewed and approved by the NRC. The application of the integrated methodologies, including UFSAR Chapter 15 revisions, was submitted on June 26, 1991 for the McGuire 1 Cycle 8 reload license amendment application. The SER for this submittal was dated November 27, 1991. The above DPC-NE-3002-A SER condition appears to be directed at the NRC review of the other topical reports and to the application of the methodology. It is inferred via NRC review and approval of all of the related topical reports and of the McGuire 1 Cycle 8 reload that the intent of this condition has been met.

**Attachment 2**  
**(Non-Proprietary)**

- b. Validity of DPC's assumption of 120% of design pressure as part of the acceptance criteria for Reactor Coolant Pump Locked Rotor should be determined by the NRC staff.

*Resolution:* Duke has adopted an acceptance criterion of 110% of design pressure for the locked rotor accident analysis as stated in Section 4.3 of DPC-NE-3002, Revision 2.

- c. No justification was presented for trip and actuation times assumed in the Feedwater System Pipe Break event analysis. Such justifications must be presented when this methodology is applied.

*Resolution:* The NRC SER, Section 2.2.1, dated November 15, 1991 specifically states that this TER condition is outside of the scope of DPC-NE-3002 and this review. Therefore, Duke has not prepared a response to this TER condition.

- d. DPC documented intent to perform parametric studies in order to select conservative scenarios or assumptions throughout the subject topical report. Therefore, such parametric studies must be presented when this methodology is applied.

*Resolution:* The DPC-NE-3002-A topical report states that parametric studies are necessary to determine the conservative modeling approach for a limited number of assumptions for some of the transients. These parametric studies were performed and are documented in the engineering calculations. The results of the analyses using the conservative modeling approach and assumptions were submitted for NRC review with the McGuire 1 Cycle 8 reload license amendment request dated June 26, 1991. These results were in the form of UFSAR revisions. Since it is not typical to include results of parametric studies in the UFSAR, only the results of the limiting cases were presented in the submittal of the application of the methodology. The engineering calculations which document the parametric studies are available for audit. It is concluded that this condition has been adequately addressed.

The SER for Revision 1 dated December 28, 1995 has the following conditions in Section 4.0. The SER for Revision 2 dated April 26, 1996 does not have any new limitations or conditions.

- a. The acceptability of the use of DPC's approach to FSAR analysis is subject to the conditions of SERs on all aspects of transient analysis and methodologies (DPC-NE-3000, DPC-NE-3001, DPC-NE-3002, DPC-NE-2004, DPC-NE-2005) as well as the SERs on RETRAN and VIPRE computer codes.

*Resolution:* This condition has been addressed in this submittal.

- b. There are scenarios in which an SGTR event may result in loss of subcooling and the consequent two-phase flow conditions in the primary system. In such instances, the use of RETRAN is not acceptable without a detailed review of the analysis.

*Resolution:* The McGuire and Catawba UFSAR SGTR analyses do not result in a loss of subcooling.

**Attachment 2  
(Non-Proprietary)**

- c. In the future if hardware or methodology changes, selection of limiting transients needs to be reconsidered, and DPC is required to perform sensitivity studies to identify the initial conditions in such a way to avoid conflict between transient objective, such as DNB and worst primary pressure.

*Resolution:* Duke's methodology, as described in DPC-NE-3002-A, does select initial and boundary conditions with consideration of the possibility that different selections and possibly separate analyses may be necessary depending on the acceptance criteria and the margin to the acceptance criteria. This approach will be continued for future re-analyses due to hardware or methodology changes.

- d. It is emphasized that, when using the SCD methodology to determine DNBR, the range of applicability of the selected CHF correlation must not be violated.

*Resolution:* Duke recognizes the need to restrict the use of CHF correlations to within their ranges of applicability. Any deviations from this approach will be submitted for review and approval.

- e. DPC's assumption of 120% of design pressure as part of the acceptance criteria for Reactor Coolant Pump Locked Rotor is not acceptable. DPC is required to use 110% of design pressure for that limit.

*Resolution:* Duke has revised DPC-NE-3002-A to use 110% of design pressure as an acceptance criterion for the locked rotor accident (See Section 4.3 of DPC-NE-3002-A, Revision 2).

**Westinghouse LOCA Topical Reports**

Westinghouse will provide the requested information regarding SER limitations, conditions, and restrictions for the LOCA-related topical reports referenced by DPC-NE-2009-P and to be used for the RFA and transitional RFA/Mark-BW cores. This information will be submitted to the NRC by April 1, 1999.

**Attachment 2  
(Non-Proprietary)**

12. Section 8.0 states that TS Figure 2.1.1-1 for the reactor core safety limits will be modified by deleting the 2455 psia safety limit line and making the 2400 psia safety limit line as the upper bound pressure allowed for power operation. Since the upper range of applicability of the WRB-2M CHF correlation for the RFA design is 2425 psia, the 2400 psia safety limit line is within the range of the CHF correlations for the Mark-BW and RFA fuel designs.

However, the safety limit lines in Figure 2.1.1-1 were based on the CHF correlation for the Mark-BW fuel design, in addition to the hot leg boiling limit. Has an analysis been performed to ensure these safety limit lines bound the safety limit for the DNBR limit of the WRB-2M correlation for the RFA design?

*Response :*

Yes. As stated, the 2400 psia line was selected since it was already defined for the Mark-BW fuel. Using the reference power distribution and the reactor inlet conditions defined by the hot leg boiling and DNB portions of the 2400 psia Safety Limit Line, the MDNBR was calculated using the full RFA core VIPRE-01 model and the WRB-2M CHF correlation. Additionally, the transition RFA/Mark-BW cores were also evaluated to ensure the established limits were conservative. The MDNBR values were greater than the design DNBR limit for all of the cases in both evaluations.

Attachment 2  
(Non-Proprietary)

13. TS Surveillance Requirements (SRs) 3.2.1.2, 3.2.1.3, and 3.2.2.2, respectively, require the heat flux hot channel factor  $F_q(x,y,z)$  and the enthalpy rise hot channel factor  $F_{\Delta H}(x,y)$  to be measured periodically using the incore detector system to ensure the values of the total peaking factor and the enthalpy rise factor assumed in the accident analyses and the reactor protection system limits are not violated. To avoid the possibility that these hot channel factors may increase beyond their allowable limits between surveillances, these SRs currently specify a penalty factor of 1.02 for the heat flux and enthalpy rise hot channel factors if the margin to the  $F_q(x,y,z)$  or  $F_{\Delta H}(x,y)$  has decreased since the previous surveillance. For the reactor core containing the RFA fuel design with integral burnable absorbers, a larger penalty may be required over certain burnup ranges early in the cycle due to the rate of burnout of this poison. Section 8.1 proposes to remove the 2% penalty value from these surveillance requirements and replace them with tables of penalty values as functions of burnup in the Core Operating Limits Report (COLR) to facilitate cycle-specific updates. Tables 8-1 and 8-2, respectively, provide "typical values" for the burnup-dependent margin-decrease penalty factors for the heat flux and enthalpy rise hot channel factors.

- (a) Provide the actual values of the margin-decrease penalty factors, as well as the bases for these values.
- (b) Provide references for the approved methodologies used to calculate these values, and to be included in TS 5.6.5 as a part of acceptability for COLR.

*Response 13a:*

Margin decrease penalty factors will be calculated for each reload core. The actual margin decrease penalty factors for the initial transition core can not be provided until the final design for this core is complete. The cycle-specific factors for each core design will be included in each units' cycle-specific Core Operating Limits Report (COLR).

The methodology used to calculate the  $F_q(x,y,z)$  and  $F_{\Delta H}(x,y)$  margin decrease penalty factors is described below. The peaking factors used to calculate the margin decrease penalty factors are obtained from the analysis performed to establish operational axial flux difference limits as described in DPC-NE-2011PA ("Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors).

*Nuclear Heat Flux Hot Channel Factor,  $F_q$ :*

$F_q(x,y,z)$  is measured periodically using the incore detector system to ensure that the value of the total peaking factor,  $F_q$ -RTP, assumed in the accident analysis is bounding. The frequency requirement for this measurement is 31 effective full power days (EFPD). In order to account for the possibility that  $F_q(x,y,z)$  may increase between surveillances, a trend of the measurement is performed to determine the point where peaking would exceed allowable limits if the current trend continues. If extrapolation of the measurement indicates that the  $F_q(x,y,z)$  measurement would exceed the  $F_q(x,y,z)$  limit prior to 31 EFPD beyond the most recent measurement, then either the surveillance interval would be decreased based on the available margin, or the  $F_q(x,y,z)$  measurement would be increased by an appropriate penalty and compared against the  $F_q(x,y,z)$  operational and RPS surveillance limits to ensure allowable total peaking limits are not exceeded.

**Attachment 2  
(Non-Proprietary)**

The  $F_q(x,y,z)$  penalty factor is calculated by projecting the change in the [

]

[

[

] The  $F_q$  margin decrease factor may be applied directly to the measured  $F_q$  or may be incorporated into the  $M_q(x,y,z)$  and  $M_c(x,y,z)$  margin factors as described in DPC-NE-2011PA. For burnup ranges where the  $F_q$  margin decrease factor is less than 1.02, a value of 1.02 will be maintained.

*Nuclear Enthalpy Rise Hot Channel Factor,  $F_{\Delta H}$ :*

The nuclear enthalpy rise hot channel factor,  $F_{\Delta H}(x,y)$ , is measured periodically using the incore detector system to ensure that fuel design criteria are not violated and accident analysis assumptions are not violated. The frequency requirement for this measurement is 31 effective full power days (EFPD). In order to account for the possibility that  $F_{\Delta H}(x,y)$  may increase between surveillances, a trend of the measurement is performed to determine the point where peaking would exceed allowable limits if the current trend continues. If extrapolation of the measurement indicates that the  $F_{\Delta H}(x,y)$  measurement would exceed the  $F_{\Delta H}(x,y)$  surveillance limit prior to 31 EFPD beyond the most recent measurement, then either the surveillance interval would be decreased based on the available margin, or the  $F_{\Delta H}(x,y)$  measurement would be increased by an appropriate penalty and compared against the  $F_{\Delta H}(x,y)$  surveillance limit to ensure allowable peaking limits are not exceeded.

The  $F_{\Delta H}(x,y)$  penalty factor is calculated by projecting the change in the [

]

[

[

] The  $F_{\Delta H}$  margin decrease factor may be applied directly to the measured  $F_{\Delta H}$  or may be incorporated into the  $M_{\Delta H}(x,y)$  margin factors. For burnup ranges where the  $F_{\Delta H}$  margin decrease factor is less than 1.02, a value of 1.02 will be maintained.

**Attachment 2**  
**(Non-Proprietary)**

***Response 13b:***

The methodology used to calculate the  $F_{\Delta H}(x,y)$  and  $F_q(x,y,z)$  margin-decrease peaking penalty factors was described in answer 13a. Duke intends to reference this topical report (DPC-NE-2009) in Technical Specification 5.6.5 for the approved methodology used to calculate these parameters.



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April 7, 1999

U. S. Nuclear Regulatory Commission  
Washington, D. C. 20555-0001

ATTENTION: Document Control Desk

Subject: Duke Energy Corporation

McGuire Nuclear Station Units 1 & 2  
Docket Nos. 50-369, 50-370

Catawba Nuclear Station Units 1 & 2  
Docket Nos. 50-413, 50-414

Response to NRC Requests for Additional Information  
on License Amendment Requests for McGuire and  
Catawba Nuclear Stations

By letters dated December 9, 1998 and January 5, 1999 the NRC requested additional information on Duke Energy Corporation's July 22, 1998 license amendment requests (LARs) for the McGuire Nuclear Station, Units 1 & 2; and the Catawba Nuclear Station, Units 1 & 2 Technical Specifications. These LARs would permit use of Westinghouse fuel at McGuire and Catawba. Topical Report DPC-NE-2009P/DPC-NE-2009 was also included in the July 22, 1998 Duke submittal.

By letter dated January 28, 1999, Duke Energy Corporation responded to the thirteen questions contained in the December 9, 1998 and January 5, 1999 NRC letters. However, the response to Question No. 11 was incomplete, pending Duke's receipt of additional information from Westinghouse Electric Company. Duke has now received this information from Westinghouse and hereby submits this to the NRC. This information is contained in a Westinghouse letter dated March 31, 1999 which is included as the attachment to this letter.

U. S. Nuclear Regulatory Commission  
April 7, 1999  
Page 2

Please address any comments or questions regarding this matter  
to J. S. Warren at (704) 382-4986.

Very truly yours,

*M. S. Tuckman*

M. S. Tuckman

Attachments

xc (w/Attachment):

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U. S. Nuclear Regulatory Commission  
April 7, 1999  
Page 3

bxc (w/Attachment):

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Catawba Document Control File (T. K. Pasour)

Catawba RGC File 801.01 (T. K. Pasour)

ELL



Westinghouse Electric Company

Box 355  
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March 31, 1999

DPC-99-016

Mr. G. Swindlehurst, Section Manager  
Safety Analysis, Nuclear Generation  
Duke Power Company  
P. O. Box 1006  
Charlotte, NC 28201-1006

**DUKE POWER COMPANY**  
**SER Restriction Evaluations for SB and LB LOCA**

Dear Mr. Swindlehurst:

In response to your request for assistance in responding to NRC Question 11 (included as Attachment 1), Westinghouse has prepared a response which addresses Westinghouse's compliance with restrictions imposed by NRC Safety Evaluation Reports (SERs) related to the Westinghouse 1985 SBLOCA Evaluation Model with NOTRUMP (References 1-4) and the 1981 Evaluation Model with BASH (References 5-13). Attachment 2 provides the NOTRUMP SER Restriction Compliance Summary. Attachment 3 contains the large break LOCA SER compliance information.

References:

1. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", N. Lee, et al., August 1985.
2. WCAP-1054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model", C. M. Thompson, et al., July 1997.
3. WCAP-11145-P-A, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code", S. D. Rupprecht, et al., 1986.
4. WCAP-14710-P-A, "1-D Heat conduction Model for Annular Fuel Pellets", D. J. Shimeck, May 1988.
5. WCAP-10484-P-A, "Spacer Grid Heat Transfer Effects During Reflood", J. S. Chiou, et al., March 1991.
6. WCAP-10484-P-A, Addendum 1, "Spacer Grid Heat Transfer Effects During Reflood", D. J. Shimeck, December 1992.
7. WCAP-10266-P-A, Revision 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using BASH", M. Y. Young, et al., March 1987.

8. WCAP-10266-P-A, Addendum 1, Revision 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code Addendum 1: Power Shape Sensitivity Studies", M. Y. Young, et al., January 1987.
9. NTD-NRC-95-4518, "Withdrawal of WCAP-12909-P on Power Shape Sensitivity Model (PSSM)", August 1995. [copy attached to DPC-95-224, "LOCA Axial Power Shape Sensitivity Model", K. B. Hanahan, August 1995.]
10. WCAP-9220-P-A, Revision 1, "Westinghouse ECCS Evaluation Model - 1981 Version", February 1982.
11. WCAP-8471-P-A, "The Westinghouse ECCS Evaluation Model: Supplementary Information", April 1975.
12. WCAP-8354-P-A, Supplement 1, "Long-Term Ice Condenser containment LOTIC Code Supplement 1", T. Hsieh, et al., July 1974.
13. ET-NRC-92-3746, "Extension of NUREG-0630 Fuel Rod Burst Strain and Assembly Blockage Models to High Fuel Rod Burst Temperatures", N. J. Liparulo, September 1992.

If you have any questions, please call Mr. John Besspiata at 412-374-4524 or me at 412-374-5651.

Sincerely,

Dwain W. Alexander  
Customer Projects Manager

cc: J. J. Besspiata, W  
S. P. Shaver, W Charlotte

Mr. G. Swindlehurst

- 3 -

DPC-99-016

bcc: J. M. Leonelli  
D. W. Alexander  
M. J. Boyles

**NRC Question 11, as received:**

11. The licensing analyses of reload cores with the RFA design will use the methodologies described in various topical reports and revisions for the analyses of fuel design, core reload design, physics, thermal-hydraulics, and transients and accidents, which were approved by NRC for analyses of current McGuire/Catawba cores not having the RFA design. For example, DPC-NE-1004A, PDC-NE-2011-PA, DPC-NE-2010A, and DPC-NE-3001-PA are used for the nuclear design calculations. DPC-NE-2004-PA, DPC-NE-2005-PA, and the VIPRE-01 code are used for the core thermal-hydraulic analyses and statistical core design. DPC-NE-3000-PA, DPC-NE-3001-PA, DPC-NE-3002A, and RETRAN-02 code are used for non-LOCA transient and accident analyses. Westinghouse small- and large-break LOCA evaluation models described in WCAP-10054-P-A and WCAP-10266-P-A, and related topical reports, are used for the small- and large-break LOCA analyses. Some of these methodologies have inherent limitations, and some have conditions or limitations imposed by the NRC SERs in their applications. Provide a list of the inherent limitations, conditions, or restrictions applicable to the RFA core design from all the methodologies to be used for the RFA reload design analyses, and describe the resolutions of these limitations, conditions and restrictions in the applications to the RFA cores and the transitional RFA/Mark-BW cores.

### NOTRUMP SER Restriction Compliance Summary

The following document contains a synopsis of the NRC imposed Safety Evaluation Report (SER) restrictions/requirements and the Westinghouse compliance status related to these issues. Not all the items identified are clearly SER restrictions, but sometimes state the NRC's interpretation of the Westinghouse Evaluation Methodology utilized for a particular aspect of the Small Break Loss Of Coolant (LOCA) Evaluation Model.

#### WCAP-10054-P-A

WCAP-10054-P-A is titled "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," and is dated August, 1985. The following summarizes the SER restrictions and requirements associated with this WCAP:

#### SER Wording (Page 6)

"The use of a single momentum equation implies that the inertias of the separate phases can not be treated. The model therefore would not be appropriate for situations when separate inertial effects are significant. For the small break transients, these effects are not significant."

#### SER Compliance

Inherent compliance due to the use of a single momentum equation.

#### SER Wording (Page 8)

"To assure the validity of this application, the bubble diameter should be on the order of  $10^{-1}$ -2 cm. As long as steam generator tube uncover (concurrent with a severe depressurization rate) does not occur, this option is acceptable."

#### SER Compliance

Westinghouse complies with this restriction for all Appendix-K licensing basis calculations. Typical Appendix-K calculations do not undergo a significant secondary side system depressurization in conjunction with steam generator tube uncover due to the modeling methodology utilized.

#### SER Wording (Page 14)

"The two phase multiplier used is the Thom modification of the Martinelli-Nelson correlation. This model is acceptable per 10 CFR Part 50 Appendix K for LOCA analysis at pressure above 250 psia"

#### SER Compliance

The original NOTRUMP model was limited to no less than 250 psia since the model, as contained in the NOTRUMP code, did not contain information below this range. Westinghouse

extended the model to below 250 psia, as allowed by Appendix K paragraph I-C-2, and reported these modifications to the NRC via the 1995 annual reporting period (NSD-NRC-96-4639).

**SER Wording (Page 16)**

“Westinghouse, however, has stated that the separator models are not used in their SBLOCA analyses.”

**SER Compliance**

Westinghouse does not model the separators in the secondary side of the steam generators for Appendix-K Small Break LOCA analyses; therefore, compliance exists.

**SER Wording (Pages 16-17)**

“Axial heat conduction is not modeled.” and “Deletion of clad axial heat conduction maximizes the peak clad temperature.”

**SER Compliance**

The Westinghouse Small Break LOCA is comprised of two computer codes, the NOTRUMP code which performs the detailed system wide thermal hydraulic calculations and the LOCTA code which performs the detailed fuel rod heatup calculations. The NOTRUMP code does not model axial conduction in the fuel rod and therefore complies. The LOCTA code has always accounted for axial conduction as is clearly stated in WCAP-14710-P-A which supplements the original NOTRUMP documentation.

**SER Wording (Page 17)**

“...; critical heat flux, W-2, W-3, or Macbeth, or GE transient CHF (the W-2 and W-3 correlations are used for licensing evaluations);...”

**SER Compliance**

The information presented here indicates that the NRC apparently misstated that Westinghouse was utilizing the W-2,W-3 correlations for Critical Heat Flux (CHF) in the fuel rod heat transfer model. A review of the analyses performed by Westinghouse, including those in WCAP-11145-P- A, indicates that the Macbeth CHF correlation has been utilized for all Appendix-K analyses performed by Westinghouse. This is consistent with the slab heat transfer map as described in WCAP-10054-P-A. In addition, the Macbeth correlation is specifically called out in Appendix K I-C-4-4 as an acceptable CHF model.

In a supplemental response to NRC questions (Specifically question 440.1 found in Appendix-A of WCAP-10054-P-A, Page A-10), a description of the core model describes the Macbeth as being utilized as the CHF correlation in the NOTRUMP Small Break LOCA model.

**SER Wording (Page 21)**

“The standard continuous contact model is not appropriate for vertical flow,...”

### **SER Compliance**

The standard continuous contact flow links are not utilized when modeling vertical flow in the Appendix-K NOTRUMP Evaluation Model analyses; therefore, compliance is demonstrated.

### **SER Wording (Page 27)**

"..., the hardwired choice of one fuel pin time step per coolant time step should result in sufficient accuracy."

### **SER Compliance**

The NOTRUMP code continues to utilize only one fuel pin time step per coolant time step and therefore complies with this requirement.

### **SER Wording (Page 47)**

"The code options available to the user but not applied in licensing evaluations were not reviewed."

### **SER Compliance**

Westinghouse complies with this requirement.

### **SER Wording (Page 53)**

"4. Steam Interaction with ECCS Water, a. Zero Steam Flow in the Intact Loops While Accumulators Discharge Water."

### **SER Compliance**

Per paragraph I-D-4 Appendix-K, the following is stated:

"During refill and reflood, the calculated steam flow in unbroken reactor coolant pipes shall be taken to be zero during the time that accumulators are discharging water into those pipes unless experimental evidence is available regarding the realistic thermal-hydraulic interaction between the steam and the liquid. In this case, the experimental data may be used to support an alternate assumption."

As can be seen, the specific Appendix-K wording can be considered applicable to Large Break LOCAs only since Small Break LOCAs do not undergo a true refill/reflood period. However, the Westinghouse Small Break LOCA Evaluation Model methodology is such that for break sizes in which the intact loop seal restriction is not removed (WCAP-11145-P-A Page 2-11), steam flow through the intact loop(s) is automatically (artificially) restricted via the loop seal model. While not specifically limited to zero, the flow is drastically reduced via the application of the artificial loop seal restriction model.

For breaks sizes above which the loop seal restriction is removed (typically  $\geq 6$  inch diameter breaks), this criterion is not explicitly adhered to. The implementation of the COSI condensation model into NOTRUMP (As approved by the NRC in WCAP-10054-P-A, Addendum 2, Revision 1), which is based on additional experimental documentation and improved modeling

techniques, more accurately models the interaction of steam with Emergency Core Cooling Water in the cold leg region. This experimental documentation supports the more accurate modeling of steam/water interaction in the cold leg region as allowed by Appendix-K. Note however that even with the COSI condensation model active, the accumulator injection condensation model still utilizes the conservative model as originally licensed in the NOTRUMP code.

**SER Wording (Page 7 of enclosure 2)**

“Per generic letter 83-35, compliance with Action Item II.K.3.31 may be submitted generically. We require that the generic submittal include validation that the limiting break location has not shifted away from the cold legs to the hot or pump suction legs.”

### **SER Compliance**

Westinghouse submitted WCAP-11145-P-A in support of generic letter 83-35 Action Item II.K.3.31. As part of this effort, verification was provided which documented that the cold leg break location remains limiting.

### **WCAP-10054-P-A, Addendum 2, Revision 1**

WCAP-10054-P-A, Addendum 2, Revision 1 is titled "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," and is dated July 1997. The following summarizes the SER restrictions and requirements associated with this WCAP:

### **SER Wording (Page 3)**

"It is stated in Ref. 5 that the range of injection jet velocities used in the experiments brackets the corresponding rates in small break LOCAs for Westinghouse plants and that the model will be used within the experimental range. Also in References 1 and 5 Westinghouse submitted analyses demonstrating that the condensation efficiency is virtually independent of RCS pressure and state that the COSI model will be applied within the pressure range of 550 to 1200 psia."

### **SER Compliance**

The coding implementation of the COSI model correlation in the NOTRUMP model restricts the application of the COSI condensation model to a default pressure range of 550 to 1200 psia and limits the injection flow rate to a default value of 40 lbm/sec-loop. The value of 40 lbm/sec-loop corresponds to the 30 ft./sec velocity utilized in the COSI experiments. As such, the default NOTRUMP implementation of the COSI condensation model complies with the applicable SER restrictions.

### **WCAP-11145-P-A**

WCAP-11145-P-A, is titled "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With The NOTRUMP Code," and is dated 1986. No specific SER restrictions were provided by the NRC as part of this WCAP review; however, the SER contains verification that the requirements of Item II.K.3.31 have been satisfied (i.e. break location study).

### **SER Wording (Page 5)**

"We therefore, find that the requirements of NUREG-0737, Item II.K3.31, as clarified by Generic Letter 83-35, have been satisfied.

We find that a condition of the safety evaluation for NOTRUMP as applied to Item II.K.3.30 has been satisfied. The limiting cold leg break size for a 4-loop plant was reanalyzed at pump suction and at hot leg locations. The results confirmed that the cold leg break was limiting."

### **WCAP-14710-P-A**

WCAP-14710-P-A, is titled "1-D Heat Conduction Model for Annular Fuel Pellets," and is dated May 1998. No specific SER restrictions are provided by the NRC in this document; however, a

conclusion was reached regarding the modeling of annular pellets during Small Break LOCA event.

**SER Wording**

“Based on its conclusions that the explicit modeling of annular pellets, as described in WCAP-14710(P), provides a more realistic representation in W Appendix K ECCS evaluation models of the annular pellets, while retaining conservatism in those evaluation models, the staff finds that the explicit modeling of annular pellets, as described in WCAP-14710(P), in W Appendix K LOCA evaluation models permits those models to continue to satisfy the regulations to which they were approved, and is, therefore, acceptable for incorporation into those models.”

**SER Compliance**

Westinghouse performs sensitivity studies to assess the impact of modeling annular pellets on plant specific analyses.

## LARGE BREAK LOCA SER COMPLIANCE

Over the years a number of SERs have been issued with specified restrictions on model applications. The individual WCAP titles which have been published with an SER included that have conditions or limitations relevant to the BASH Evaluation Model have been reviewed in the context of the Catawba 2 Cycle 11 large break LOCA analysis. The relevant restrictions, limitations, and conditions specific to the BASH Evaluation Model as imposed by the NRC are listed, together with the means by which they are resolved in the Westinghouse Catawba Unit 2 Cycle 11 large break LOCA analysis.

### WCAP-10484-P-A

WCAP-10484-P-A is titled "Spacer Grid Heat Transfer Effects during Reflood" and is dated March, 1991. The following summarizes the SER restrictions and requirements associated with this WCAP:

SER Wording - "Acceptance of the droplet breakup model is premature due to the limited..information..." (page 16 of the WCAP-10484 SER)

SER Compliance - The droplet breakup model has been deleted from the LOCBART computer code and is not used in any BASH Evaluation Model (EM) analysis.

SER Wording - "The length average heat transfer coefficient  $h(Z/L)$  in the node should be used in applying the Y-H-L (Yao-Hochreiter-Leech) correlation." (page 16 of the WCAP-10484 SER)

SER Compliance - A review of the LOCBART computer code, Version 17.0 logic to compute grid single phase heat transfer enhancement identified the Y-H-L correlation value was not being averaged over the length of the node in some situations. This discovery led to the preparation of a Nonconformance Report concerning this error. LOCBART Version 18.0, which corrects the subject error, has been created and documented for use in the Catawba 2 Cycle 11 large break LOCA BASH EM analysis.

SER Wording - "The use of BART with grid rewet models should be restricted to the range of conditions consistent with the data base tested as indicated in Table 1 of this SER." (page 16 of the WCAP-10484 SER)

SER Compliance - WCAP-10484-P-A Addendum 1 supersedes the "BART with grid rewet" simulations presented here and is the reference for validation of these LOCBART models within the current BASH EM. The Catawba BASH EM analyses are "under the limitations delineated in (that) report," quoting the SER letter for the Addendum 1 WCAP.

## **WCAP-10266-P-A, Revision 2**

WCAP-10266-P-A is titled "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code " and is dated March, 1987. The following is a review of the SER restrictions and requirements found in this WCAP:

SER Wording - "The EM has no downward quench capability and therefore cannot be used for the analysis of either upper head injection plants or upper plenum injection plants." (page 10 of the WCAP-10266 SER)

SER Compliance - The BASH EM has not been applied in the large break LOCA analysis of any plant equipped with upper head injection or upper plenum injection. The upper head injection system has been removed from service at the Catawba Units.

SER Wording - "Westinghouse has committed to continue to analyze the large break LOCA with both minimum and maximum safety injection to confirm which produces the limiting large break scenario for each plant." (page 11 of the WCAP-10266 SER)

SER Compliance - The maximum SI scenario will be analyzed for the limiting discharge coefficient DECLG break as identified in the Catawba Unit 2 Cycle 11 BASH EM analysis.

SER Wording - "Westinghouse has committed to submit confirmatory analyses with the first BASH plant calculation of each type (2, 3 and 4 loop) to demonstrate that the cosine power shape is limiting and is the appropriate power shape to use for licensing calculations." (page 11 of the WCAP-10266 SER)

SER Compliance - This SER requirement was originally fulfilled for 3 and 4 loop plants via sensitivity studies presented in WCAP-10266-P-A, Addendum 1, Revision 2. However, the current BASH EM power shape methodology is to use an explicit approach introduced in 1995 for top-skewed power distributions, which was noticed to the NRC in NTD-NRC-95-4518. The disposition of this issue for the Catawba Units has been included in the PCT Margin Utilization Sheets, beginning with the 1995 10CFR50.46 Annual Reporting. The Catawba Unit 2 Cycle 11 BASH EM analysis will consider power shape effects consistent with the current methodology.

## **WCAP-9220-P-A, Revision 1**

WCAP-9220-P-A, Revision 1 is titled "Westinghouse ECCS Evaluation Model - 1981 Version" and is dated February, 1982. This WCAP documents a number of changes to the existing large break LOCA EM which were implemented to correct errors and/or to obtain more favorable PCT results. Two separate Safety Evaluation Reports, dated August 29, 1978 and December 1, 1981 are included in the WCAP. The following is a review of the SER restrictions and requirements found in this WCAP which remain applicable to the BASH Evaluation Model. First, from the August 29, 1978 SER:

SER Wording - "Westinghouse has recently decided to cancel requests for using (Dougall-Rohsenow post-CHF heat transfer) correlation in place of the Westinghouse transition boiling correlation." (page xiv of WCAP-9220 Revision 1)

SER Compliance - The Westinghouse transition boiling correlation continues to be used in BASH EM calculations.

The restrictions and requirements identified in the December 1981 SER are discussed below:

SER Wording - "Based on the data and analyses contained in NUREG-0630....we find the (algorithm for computing heatup rates; rupture, strain, and blockage models; the prerupture strain model and the artificial limit on the degree of swelling), and their proposed applications to be acceptable." (page B-15 of WCAP-9220 Revision 1)

SER Compliance - The NUREG-0630 fuel rod burst and blockage models are programmed into the SATAN and LOCBART codes and are applied in BASH EM computations, including Catawba Unit 2 Cycle 11. Note that the SER restricted the usage of the NUREG-0630 model to calculated burst temperatures less than 950°C. In letter ET-NRC-92-3746 dated September 16, 1992 Westinghouse described the extension of the NUREG-0630 modeling to conservatively consider burst temperatures greater than 950°C.

#### **WCAP-8471-P-A**

WCAP-8471-P-A is titled "The Westinghouse ECCS Evaluation Model: Supplementary Information" and is dated April, 1975. The SER for this WCAP covers the entire set of WCAP reports that documented the Westinghouse model originally created and submitted when 10CFR50 and its Appendix K first became effective in 1974; it contains no restrictions and requirements as such other than for LOTIC as noted below.

SER Wording - "Until such time that LOTIC is modified to resolve the staff concerns.... a conservative minimum containment pressure of zero psig must be assumed in ECCS analyses of plants using an ice condenser containment." (page 3 of the WCAP-8471 SER)

SER Compliance - Approval of the LOTIC-2 computer code for ECCS minimum containment pressure analysis of ice condenser containments was obtained in the SER of WCAP-8354-P-A, Supplement 1. This code version has the capability of modeling additional ice condenser phenomena as required by the NRC. LOTIC-2 will be employed in the Catawba Unit 2 Cycle 11 BASH EM analysis.

ATTACHMENT 7a

Detailed Listing of Changes to DPC-NF-2010-A

## Attachment 7a - Detailed Listing of Changes to DPC-NF-2010A

This attachment provides a detailed list of proposed changes to the topical report DPC-NF-2010. Changes are listed according to the location in DPC-NF-2010A. Cited references are listed at the end of this attachment.

1. Cover, Table of Contents, List of Figures

Description: Editorial changes to correspond to changes made throughout the report.

2. Section 1.1, Section Heading

Description: Changed the heading to "General Nuclear Design Description" from "Introduction" for clarity.

3. Section 1.1, First Paragraph

Description: Changed the third sentence to give examples of intervals between refueling outages.

Justification: The original sentence implies a maximum fuel cycle length of 18 months, and possible fuel cycle lengths are not limited to 18 months. This change is made to avoid difficulties with the literal interpretation of the original description.

4. Section 1.1, List of Design Criteria

Description: Changed Item 1 to include consideration for any planned power coastdown at the end of cycle.

Justification: The original sentence implies full power operation for the entire cycle length, and operating strategy is not limited to full power operation (i.e. coastdowns). This change is made to avoid difficulties with the literal interpretation of the original description.

5. Section 1.1, List of Design Criteria

Description: For completeness, changed Item 3 to include allowance for core redesign if important core parameters are not conservative relative to safety analysis assumed values.

6. Section 1.1, New Paragraph

Description: Added a new paragraph addressing the applicability of the QA Program.

7. Section 1.2, First Paragraph

Description: Removed the first sentence for clarity.

Justification: Although terms used by DPC may be the same as or similar to Westinghouse nomenclature, these terms and definitions are subject to change.

8. Section 1.2,  $F_Q^N$  and  $F_Q^E$

Description: Added the word "Factor" to the names of these terms to avoid confusion.

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9. Section 1.2, Definition of  $F_Q^E$ , Engineering Heat Flux Hot Channel Factor

Description: Added the word “typically” for clarification since.

Justification: The value of 1.03 is subject to change depending on the fuel design.

10. Section 1.2

Description: Deleted the discussion pertaining to the subfactors of  $F_Q^T$ .

Justification: This discussion is not important to the methods discussed in this report.

11. Section 2

Description: Replaced the entire section with basic descriptive information.

Justification: Specific fuel design details are subject to change and are available in the UFSAR.

12. Section 3.1

Description: Updated this section to include CASMO-3/SIMULATE-3.

Justification: This change makes the report consistent with current NRC approved methods (Reference 1).

13. Section 3.2, Second Paragraph

Description: changed “FSAR” to “UFSAR” to make the report consistent with current terminology.

14. Section 3.2, Second Paragraph

Description: Clarified the third sentence.

Justification: The original statement applies to COMETHE-IIK and TACO-2. DPC may use other fuel performance codes approved by the NRC (e.g. Reference 2), therefore, this change makes the report consistent with current NRC approved methods.

15. Section 3.2, New Paragraph

Description: Added new paragraph describing cross section development for SIMULATE.

Justification: This change makes the report consistent with current NRC approved methods (Reference 1).

16. Section 3.3

Description: Updated this section to include CASMO-3/SIMULATE-3.

Justification: This change makes the report consistent with current NRC approved methods (Reference 1).

17. Section 3.3, First Paragraph

Description: Moved the reference to Table 3.1 to the end of the second paragraph for clarification.

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18. Section 3.4, First Paragraph

Description: Clarified the second sentence.

Justification: This sentence implies that PDQ is currently used. Since DPC may use other nuclear design methods approved by the NRC (e.g. Reference 1), this change is made to avoid difficulties with the literal interpretation of the original description.

19. Section 3.4.2, Fifth Paragraph

Description: Changed the first sentence to use more commonly understood terminology.

20. Section 3.4.2, Fifth Paragraph

Description: Clarified the second sentence.

Justification: This sentence implies full power operation for the entire cycle length; however, it is common to perform a power coastdown at the end of cycle. This change is made to avoid difficulties with the literal interpretation of the original description.

21. New Section

Description: Added a section pertaining to SIMULATE for completeness.

22. Section 4.1, First Paragraph

Description: Clarified the first three sentences.

Justification: Operation and system requirements are needed to initiate a reload design. Although guidelines are used to perform a reload design, there is no "initialization procedure". This change is made to avoid difficulties with the literal interpretation of the original description.

23. Sections 4.1.1, 4.1.2, and Table 4.1

Description: Replaced "design basis" or "design bases", with "design" or "design considerations" for clarity.

Justification: The expression "design basis" has a safety specific meaning, which is not intended in these sections. This change is made to avoid difficulties with the literal interpretation of the original description.

24. Section 4.1.3

Description: Clarified the second sentence of the first paragraph and removed the second and third paragraphs.

Justification: Computing efficiency has eliminated the need to estimate the cycle lifetime, and modeling the depletion of a reload core confirms that a reload design will meet system requirements. Also, this change removes specific reference to PDQ and NODE to avoid confusion.

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25. Section 4.2, First Paragraph

Description: Changed the acronym "PFCD" to "preliminary fuel cycle design" for clarity.

26. Section 4.2, List of design criteria

Description: Reworded the definitions for  $F_{\Delta H}^N$ , MTC, SDM, and linear rod power to make the report consistent with current Technical Specifications.

Justification: Technical Specifications cover applicable operating modes; mention of operating modes is unnecessary.

27. Section 4.2, List of design criteria

Description: Reworded the statement for fuel burnup to make it current.

Justification: Burnup limits are subject to change with the fuel design and may include limits on assembly and rod burnup.

28. Section 4.2, Second Paragraph

Description: Updated the description of design calculations for clarification.

29. Section 4.2.1, First Paragraph

Description: Deleted the first sentence and reworded the second sentence to clarify fuel shuffle optimization.

Justification: The original statement applies to SHUFFLE and PDQ, and DPC may use other nuclear code methods approved by the NRC (e.g. Reference 1).

30. Section 4.2.1, Second Paragraph

Description: Clarified the first sentence.

Justification: There are no "preliminary" safety criteria, and safety criteria may be limiting at times in core life other than BOC.

31. Section 4.2.1, Second Paragraph

Description: Removed the remainder of this paragraph to make the discussion current,

Justification: The preliminary burnup window assessment performed in the final fuel cycle design calculation does not include the calculations described here. These calculations are performed later in the reload design process (see References 2-5).

32. Section 4.2.1, Third Paragraph

Description: Clarified the first sentence.

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Justification: Depletion model statepoints may be specified in MWD/MTU or EFPD and may be different than those listed. This change is made to avoid difficulties with the literal interpretation of the original description.

33. Section 4.2.1, New Paragraph

Description: Added a new paragraph after the third paragraph to clarify the connection of the reload design model to the entire reload design process.

34. Section 4.2.1, Fourth Paragraph

Description: Clarified this paragraph, since the shuffle pattern may be optimized on a number of factors besides power distribution. This change also reflects that the FFCD calculation develops the shuffle pattern.

35. Section 4.2.2, Second Paragraph

Description: Added the word 'Control' when referring to bank D and banks B and C for clarification.

36. Section 4.2.2.1, First Paragraph

Description: Clarified the second sentence to accommodate other applicable core models.

Justification: The original statement applies to quarter core PDQ/NODE models. Since DPC may use other nuclear code methods approved by the NRC (e.g. Reference 1), this change is a clarification.

37. Section 4.2.2.2, First Paragraph

Description: Reworded the first sentence to make the report consistent with other NRC approved methods (e.g. Reference 1).

38. Section 4.2.2.2, Third Paragraph

Description: Clarified the second, fourth, seventh, and eighth sentences.

Justification: The original statements apply to PDQ and/or NODE, and DPC may use other nuclear code methods approved by the NRC (e.g. Reference 1). Also, the conditions when the flux redistribution effect is to be included are clarified.

39. Section 4.2.2.2, New Paragraph

Description: Added a paragraph explaining the use of penalties to account for xenon redistribution.

40. Section 4.2.2.2, Last Paragraph

Description: Changed the last sentence to make the report consistent with the current Technical Specifications.

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41. Section 4.2.2.3, First Paragraph

Description: Corrected a typographical error in the reference section number.

42. Section 4.2.2.4

Description: Revised parts of this section to make the report consistent with methods approved by the NRC (References 1, 3-5) and the UFSAR, and to clarify the description of the REA calculations.

43. Section 4.2.2.5

Description: Revised parts of this section to make the report consistent with methods approved by the NRC (Reference 4) and the UFSAR, and to clarify the description of the DRA calculations.

44. Section 4.2.3, First Paragraph

Description: Revised most of this section for clarity.

Justification: Some of the original sentences apply to PDQ/NODE models, and DPC may use other nuclear code methods approved by the NRC (e.g. Reference 1). These changes also remove specific burnup limits, since burnup limits are subject to change with the fuel design and may include limits on assembly and rod burnup. Finally, information not important to the methods discussed in this report is removed.

45. Section 4.2.4, Second Paragraph

Description: Changed the first sentence to make the report consistent with other NRC approved methods (e.g. Reference 1).

46. Section 4.2.4. 1, First Paragraph

Description: Removed the acronym for Doppler Coefficient "(DC)". It is not used elsewhere in the report.

47. Section 4.2.4. 1, Fourth Paragraph

Description: Clarified the last sentence.

Justification: These calculations do not have to be performed in the final fuel cycle design phase, but may be performed in other supporting calculations. Also, the original statement applies to PDQ/NODE, and DPC may use other nuclear code methods approved by the NRC (e.g. Reference 1). Finally, conditions in addition to HZP and HFP may be used.

48. Section 4.2.4.2, First Paragraph

Description: Removed the third sentence, since it is not important to the methods discussed in this report.

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49. Section 4.2.4.2, Second Paragraph

Description: Changed the first sentence to make the report consistent with other NRC approved methods (e.g. Reference 1).

50. Section 4.2.4.2, Second Paragraph

Description: Clarified the second sentence.

Justification: Temperature coefficient calculations need not be limited to 5 °F temperature difference. This change is made to avoid difficulties with the literal interpretation of the original description.

51. Section 4.2.4.2, Third Paragraph

Description: Clarified the verification of MTC when considering the burnup window of the previous cycle.

52. Section 4.2.4.3

Description: Clarified the second sentence.

Justification: Temperature coefficient calculations may be based on an increase or decrease in temperature. This change is made to avoid difficulties with the literal interpretation of the original description.

53. Section 4.2.4.3

Description: Removed the last sentence for clarity.

Justification: The original statement applies to PDQ/NODE, and DPC may use other nuclear code methods approved by the NRC (e.g. Reference 1).

54. Section 4.2.4.4, Fifth Paragraph

Description: Reworded the first and second sentences to make the report consistent with other NRC approved methods (e.g. Reference 1).

55. Section 4.2.4.4, Fifth Paragraph

Description: Clarified the second sentence.

Justification: Power coefficient calculations need not be limited to a 5% power change. This change is made to avoid difficulties with the literal interpretation of the original description. Also, "everything else" is defined.

56. Section 4.2.4.4, Sixth Paragraph

Description: Corrected a typographical error in a section number reference in the first sentence (4.2.2.2 instead of 4.3.2.2).

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57. Section 4.2.4.4, Seventh Paragraph

Description: Removed the first two sentences and combined the last sentence with the sixth paragraph for clarity.

Justification: The original statement applies to PDQ/NODE, and DPC may use other nuclear code methods approved by the NRC (e.g. Reference 1). Also, removed redundant information and moved the last sentence.

58. Section 4.2.4.5

Description: Clarified the first sentence to specify coefficients other than those previously discussed.

59. Section 4.2.4.5

Description: Changed the beginning of the third sentence to clarify that coefficients discussed in this section are not required to be calculated.

Justification: This change is made to avoid difficulties with the literal interpretation of the original description.

60. Section 4.2.4.6

Description: Changed the first sentence to make the report consistent with other NRC approved methods (e.g. Reference 1).

61. Section 4.2.4.7

Description: Revised this section to make the report consistent with other methods approved by the NRC (e.g. Reference 1) and to clarify the descriptions of the xenon calculations.

62. Section 4.2.4.8, First Paragraph

Description: Changed the word "inhour" to "kinetics" in the first sentence for clarity.

Justification: The inhour equation is specifically related to a reactor with a stable reactor period. Accident and transient analyses may calculate kinetic data for a non-stable period.

63. Section 4.2.4.8, Second Paragraph

Description: Combined the first two sentences for clarity.

Justification: The original statement implies PDQ and DELAY are the only codes used to perform kinetics calculations, and DPC may use other nuclear code methods approved by the NRC (e.g. Reference 1).

64. Section 4.2.4.8, Second Paragraph

Description: Removed the fourth sentence, since it is not important to the methods discussed in this report.

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65. Section 4.2.4.8, Second Paragraph

Description: Combined the last two sentences into a new paragraph for clarity, and changed “FSAR” to “UFSAR” to make the report consistent with current terminology.

66. Section 4.2.4.8, Second Paragraph

Description: Added a sentence pertaining to CASMO-3/ SIMULATE-3 to the end of the paragraph to make the report consistent with current NRC approved methods (Reference 1).

67. Section 4.2.5

Description: Revised most of this section to include consideration of the fuel design limits and to relocate the Quality Assurance aspect to Section 1.1.

Justification: The original statement did not relate to consideration of fuel design limits. Paragraphs 2 and 4 are covered by the DPC QA program referenced in Section 1.1. Section 4.2 lists the design criteria that are summarized in the third paragraph. Finally, the methods that address calculations of parameters needed for safety analyses and core operation are summarized in Sections 5-7.

68. Table 4-2

Description: Revised Item 8 consistent with the change described for Section 4.2.2.2, and revised the NOTE to make the report consistent with the current Technical Specifications.

69. Table 4-3

Description: Removed this table, since it is not important to the methods discussed in this report.

Justification: This table contains fuel assembly type specific power peaking requirements, which are covered in the Technical Specifications. Also, Section 4.2 addresses the power peaking limits. As a result of this change, remaining tables are renumbered, and the table number reference in Section 4.2.4.6 is changed.

70. Section 5.1

Description: Revised and split the fourth paragraph and added a new paragraph pertaining to SIMULATE-3 to make the report consistent with current NRC approved methods (References 1, 6).

71. Section 5.2, First Paragraph

Description: Reworded the second and third sentences.

Justification: This makes the report consistent with current NRC approved methods (Reference 1). Replacing the specific normalization statepoint with a wider range of possibilities and removing specific burnup statepoints avoids difficulties with the literal interpretation of the original descriptions.

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72. Section 5.2, Second Paragraph

Description: Clarified this paragraph.

Justification: The term "history file" is NODE specific and is not used for data files created by other codes.

73. Section 5.2, Third Paragraph

Description: Removed Item 5, since it is implied by the second paragraph.

74. Section 5.3.1, First Paragraph

Description: Reworded the third sentence to make the report consistent with other NRC approved methods (e.g. Reference 1).

75. Section 5.3.2, First Paragraph

Description: Reworded the second sentence to make the report consistent with other NRC approved methods (e.g. Reference 1).

76. Section 5.4.1, First Paragraph

Description: Clarified and combined the first two sentences.

Justification: The original statement applies to NODE, and DPC may use other nuclear code methods approved by the NRC (e.g. Reference 1).

77. Section 5.4.1, Second Paragraph (including numbered list)

Description: Revised this paragraph to make the report consistent with other NRC approved methods (e.g. Reference 1).

78. Section 5.4.1, New Paragraph

Description: Added a paragraph explaining the use of penalties to account for xenon redistribution to make the report consistent with methods approved by the NRC (Reference 4).

79. Section 5.4.2, First Paragraph

Description: Changed the words "EPRI-NODE-P" to "nodal code" in the second sentence for clarity.

Justification: The original statement applies NODE, and DPC may use other nuclear code methods approved by the NRC (e.g. Reference 1).

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80. Section 5.4.2, Second Paragraph

Description: Changed the first and second sentences to make the report consistent with other NRC approved methods (e.g. Reference 1).

81. Section 5.4.2, Second Paragraph

Description: Changed the last sentence for completeness.

82. Section 5.5, First Paragraph

Description: Clarified the second sentence to use more appropriate terminology.

83. Section 5.5

Description: Removed the Section 5.5.1 and 5.5.2 sub-section headings to make the format of this sub-section similar to the others in Section 5.

84. Section 5.5.1

Description: Clarified the limiting criteria in Items 1 and 2 to make the report consistent with the current Technical Specifications.

85. Section 5.5.1

Description: Clarified Item 3 to accommodate UFSAR Chapter 15 accident analyses.

Justification: RILs may impact more than the three accidents and the two safety significant physics parameters originally listed. This is a literal compliance related change made to avoid confusion.

86. Section 5.5.2, Second Paragraph

Description: Changed the last sentence for completeness.

87. Section 5.5.2, Third and Fourth Paragraphs

Description: Changed all occurrences of " $F_{\Delta H}^N$ " to "power peaking" or "peaking factor".

Justification: Peaking requirements may also include FQ. This change is made to avoid difficulties with the literal interpretation of the original description.

88. Section 5.5.2, Third Paragraph

Description: Revised this paragraph to make the report consistent with other NRC approved methods (e.g. Reference 1).

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89. Section 5.5.2, Fourth Paragraph

Description: Removed the last sentence to avoid confusion with NRC approved safety analysis methods.

90. Section 5.6, First Paragraph

Description: Changed the last sentence for completeness.

Justification: Trip reactivity parameters are used in more than the accidents listed. This change makes the methods consistent with current NRC approved methods (Reference 4).

91. Section 5.6.1, Second Paragraph

Description: The minimum trip reactivity description is updated to be consistent with safety analysis methods approved by NRC (Reference 4).

92. Section 5.6.1, Second Paragraph and Third Paragraphs

Description: Editorial change moving the first sentence of the third paragraph to the beginning of the second paragraph and moving the last sentence of the third paragraph to the end of the second paragraph.

93. Section 5.7

Description: Removed everything except the first paragraph, since it is redundant with the information contained in Section 4.2.5.

94. Table 5-1

Description: Updated the example to be consistent with safety analysis methods approved by the NRC (Reference 4).

95. Section 6

Description: Complete rewrite of this section.

Justification: Subsequent to the initial NRC approval of this topical report, methods for performing safety related calculations were approved by the NRC in References 3-5. Note that Section 6.3 was excluded from the NRC evaluation of the original version of this report. The rewrite of this section references safety analysis methods approved by the NRC (References 3, 4) and provides a brief outline of the physics parameters and power peaking analyses performed, including uncertainty factors. These changes make the methods consistent with current NRC approved methods.

96. Section 7

Description: Complete rewrite of this section.

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Justification: This section was excluded from the NRC evaluation of the original version of this report. Methods for performing the power peaking analyses have been reviewed and approved by the NRC in Reference 3. The rewrite of this section references these methods and provides a brief outline of the peaking analyses performed. This change makes the methods consistent with current NRC approved methods.

97. Section 8.1, First Paragraph

Description: Added a sentence and equation discussing the calculation of pin powers using the radial local factor for clarity.

98. Section 8.1, Second Paragraph

Description: Changed the first sentence to make the report consistent with other NRC approved methods (e.g. Reference 1).

99. Section 8.2, Second Paragraph

Description: Removed the third sentence, since it is not important to the methods discussed in this report.

100. New Section

Description: Added a section discussing the PDQ pin power uncertainty as determined from the BNL benchmark problem for completeness.

101. Section 8.4 (original section number)

Description: Complete rewrite of this section.

Justification: This change makes the conclusions consistent with a DPC response (dated December 19, 1984) to a NRC Request for Additional Information (dated November 5, 1984) and with the SER (Reference 6) issued with the original version of this report.

102. Section 9, First Paragraph

Description: Changed the first sentence to make the report consistent with other NRC approved methods (e.g. Reference 1).

103. Section 9.1.1, First Paragraph

Description: Revised parts of this paragraph to make the report consistent with other NRC approved methods (e.g. Reference 1). These changes also clarify the methods for determining the critical boron concentration.

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104. Section 9.1.1

Description: Clarified the third sentence of the third paragraph and the last sentence of the sixth paragraph.

Justification: These changes are made to avoid difficulties with the literal interpretation of the original descriptions. Boron letdown and differential boron worth data may be shown in plot or table format.

105. Section 9.1.2, First Paragraph

Description: Changed the first sentence to make the report consistent with other NRC approved methods (e.g. Reference 1).

106. Section 9.1.2, First Paragraph

Description: Changed the last sentence for clarity.

Justification: This change is made to avoid difficulties with the literal interpretation of the original description. Equilibrium xenon worth data may be shown in plot or table format.

107. Section 9.1.2, Second Paragraph

Description: Clarified the last sentence by inserting the word "typical".

Justification: This change is made to avoid difficulties with the literal interpretation of the original description.

108. Section 9.1.3.1, Section Heading

Description: Editorial change from the word "Group" to "Bank".

109. Section 9.1.3.1, Second Paragraph

Description: Changed the second sentence to make the report consistent with other NRC approved methods (e.g. Reference 1).

110. Section 9.1.3.1, Second Paragraph

Description: Changed the last sentence for clarity.

Justification: This change is made to avoid difficulties with the literal interpretation of the original description. Integral rod worth data may be shown in plot or table format.

111. Sections 9.1.3.2, 9.1.3.3, and 9.1.3.4

Description: Clarified the first sentence in each section.

Justification: The calculations discussed are not performed in the final fuel cycle design calculation, but later in the reload design process. Also, the section number references are not required.

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112. Sections 9.1.4.1

Description: Clarified the first and second paragraphs.

Justification: These changes are made to avoid difficulties with the literal interpretation of the original descriptions pertaining to temperatures. Also, the original statements apply to PDQ/NODE, and DPC may use other nuclear code methods approved by the NRC (e.g. Reference 1) to perform these calculations.

113. Section 9.1.4.2

Description: Revised parts of this section.

Justification: Temperature coefficient calculations need not be limited to a 5 OF temperature difference and power Doppler coefficient calculations need not be limited to a 5% power change. These changes are made to avoid difficulties with the literal interpretation of the original description.

114. Section 9.1.5, First Paragraph

Description: Clarified the beginning of the second sentence.

Justification: The original statement applies NODE, and DPC may use other nuclear code methods approved by the NRC (e.g. Reference 1).

115. Section 9.1.6, First Paragraph

Description: Corrected a typographical error in a section number reference in the first sentence (4.2,4.8 instead of 4.3.3.8).

116. Section 9.1.6, First Paragraph

Description: Clarified the second and last sentences.

Justification: Reactivity versus doubling time is not directly provided for startup testing, since current reactivity computers used in startup testing perform equivalent checks of kinetic data using inhour results during stable periods. Also, EOC data need not be developed for startup testing.

117. Section 9.2

Description: Revised parts of this section for clarification and to be consistent with current terminology.

118. Table 9-3

Description: Changed the title.

Attachment 7a - Detailed Listing of Changes to DPC-NF-2010A

119. Section 10, New Paragraph

Description: Added a new paragraph to provide a description of the startup tests and the specific methodology for the test predictions.

120. Table 10-2

Description: Added the equation for calculating the difference.

121. Section 11, New Paragraph

Description: Added a new paragraph providing clarification of the methods associated with the predicted data.

122. Section 11.1.1, First Paragraph

Description: Changed the first sentence and added a new sentence to clarify the contents of this section.

123. Sections 11.1.2 and 11.5.3

Description: Revised these sub-sections to include the assembly normalized axial peak information.

Justification: This information was not provided in the original version of this report, but included in Reference 3. It is more appropriately located in this report.

124. Section 11.3.1, First Paragraph

Description: Clarified the first sentence to avoid confusion about the methods used in this section.

125. Section 11.5.6, First Paragraph

Description: Clarified the assembly ONRF table to include the assembly normalized axial peak information.

126. Section 11.5.6, New Paragraphs

Description: Added new paragraphs to provide the information on the statistically combined peak pin power uncertainties for completeness.

127. Section 12

Description: The following editorial changes are made to the references due to changes previously described:

(1) References 6, 7, 22, 23, and 24 are no longer used, (2) References 18 and 19 are now the Updated Final Safety Analysis Report, and (3) Added References 28-32.

Attachment 7a - Detailed Listing of Changes to DPC-NF-2010A

128. Appendix A

Description: The following editorial changes are made to the references due to changes previously described:

(1) The descriptions for COMETHE-III-J, NODE UTILITY CODE (NUC), TAC02, VIPRE, and TACO are no longer referenced, (2) Code descriptions for CASMO-3 and SIMULATE-3P are added, and (3) Methods references are provided for EPRI-NODE and PDQ07.

References:

1. "Duke Power Company, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P", DPC-NE-1004A, Revision 1, SER Dated April 26, 1996.
2. "Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, Core Thermal-Hydraulic Methodology using VIPRE-01", DPC-NE-2004P-A, Revision 1, SER Dated February 20, 1997.
3. "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", DPC-NE-2011P-A, March 1990.
4. "Multidimensional Reactor Transient's and Safety Analysis Physics Parameter Methodology", DPC-NE-3001P-A, November 1991.
5. "FSAR Chapter 15 System Transient Analysis Methodology", DPC-NE-3002-A, Revision 3, SER Dated February 5, 1999.
6. "Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design", DPC-NF-2010A, June 1985.

ATTACHMENT 7b

DPC-NF-2010, Revision 1



*A Duke Energy Company*

ATTACHMENT 7b

DPC-NF-2010, Revision 1

DUKE POWER COMPANY  
MCGUIRE NUCLEAR STATION  
CATAWBA NUCLEAR STATION

Nuclear Physics Methodology  
for  
Reload Design

DPC-NF-2010  
Revision 1

August 2001

Nuclear Generation Department  
Nuclear Engineering

Revision History

Revision	Description
DPC-NF-2010, Original Issue	Originally submitted to the NRC for approval in July 1984. Additional information was submitted to the NRC supplying responses to a request for additional information.
DPC-NF-2010-A, Original Issue	NRC approved version issued in May 1985.
DPC-NF-2010, Revision 1	<p>Submitted to the NRC for approval in August 2001.</p> <p>This revision updates the report for completeness to indicate the use of NRC approved methods approved subsequent to the implementation of the original issue including the use of CASMO-3/SIMLUATE-3 reactor physics methods.</p> <p>This revision also removes unnecessary data including items not reviewed by the NRC in the original version and general fuel data not necessary for these methods.</p> <p>Also various editorial changes are made, including updating the Table of Contents.</p> <p>Changes associated with this revision are denoted by revision bars, except format changes.</p>

STATEMENT OF DISCLAIMER

This report was prepared by Duke Power Company ("Duke Power") for filing with the United States Nuclear Regulatory Commission ("USNRC") for the sole purpose of obtaining approval of Duke Power's PWR nuclear design methods at McGuire and Catawba. Duke Power makes no warranty or representation and assumes no obligation, responsibility, or liability with respect to the contents of this report or its accuracy or completeness. Any use of or reliance on the report or the information contained in this report is at the sole risk of the party using or relying on it.

## ABSTRACT

This Technical Report describes Duke Power Company's Nuclear Design Methodology for the McGuire and Catawba Nuclear Station. The nuclear design process consists of mechanical properties used as nuclear design input, the nuclear code system and methodology Duke Power intends to use to perform design calculations and to provide operational support, and the development of statistical reliability factors.

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## 1. INTRODUCTION

### 1.1 General Nuclear Design Description

A commercial Pressurized Water Reactor (PWR) is designed to hold a constant number of nuclear fuel assemblies which are generally identical mechanically, but differ in the amount of fissile material content. During cycles subsequent to the initial cycle, fuel assemblies differ in burnup as well. Refueling occurs at intervals appropriate for the power production needed, for example 12, 18, or 24 months. At refueling, a predetermined number of irradiated fuel assemblies are discharged and the same number are loaded as fresh (reload region) or possibly irradiated assemblies. The fuel management scheme determines the locations of all fresh and irradiated assemblies.

This report describes some of the various aspects of nuclear design with principal emphasis placed upon development of a core loading pattern and nuclear calculations performed to evaluate safety and operational parameters. The following sections provide detailed discussion, including descriptions, of design methods, analytical formulations, and calculational procedures involved in the various nuclear design tasks for the McGuire and Catawba Nuclear Stations. The nuclear design is essentially a series of analytical calculations with the objective of designing the reload core in such a manner that the reactor can be operated up to a specified power level for a specified number of days within acceptable safety and operating limits. It consists of the development of the basic specifications of the reload region (fuel enrichment, number of assemblies, uranium loading, etc.); it sets forth the number and identity of each residual fuel assembly, selects the location of each fuel assembly in the core for the new fuel cycle, and establishes the core characteristics. The nuclear design used in conjunction with the thermal hydraulic and safety analyses establishes the operating limits, control rod limits, and protection system setpoints.

In arriving at the final nuclear design, the designer tries to meet the requirements imposed by the operational considerations, fuel economics

considerations, and safety considerations. These requirements are called nuclear design criteria and are as follows:

1. Initial core excess reactivity will be sufficient to enable full power operation for the desired length of the cycle, with appropriate allowance for any planned coastdown.
2. The fuel assemblies to be discharged at the end of the fuel cycle will attain maximum permissible burnup so that maximum energy extraction consistent with the fuel mechanical integrity criteria is achieved.
3. Values of important core parameters (moderator temperature coefficient, Doppler coefficient, ejected rod worth, boron worth, control rod worth, maximum linear heat rate of the fuel pin at various elevations in the core, and shutdown margin) predicted for the cycle are conservative with respect to the values assumed in the safety analysis of various postulated accidents. If they are not conservative, acceptable reevaluation or reanalysis of applicable accidents is performed, or the core is redesigned.
4. The power distributions within the reactor core for all possible (or permissible) core conditions that could exist during the operation of the cycle will not lead to exceeding the thermal design criteria of the fuel or exceeding the LOCA-limited peaking factors.
5. Fuel management will produce fuel rod power and burnup consistent with the mechanical integrity analysis of the fuel rod.

The nuclear design process described in this report consists of mechanical properties used as nuclear design input, the nuclear code system and methodology Duke Power intends to use to perform design calculations and to provide operational support, and the development of statistical reliability factors.

The nuclear design calculations described in this report are covered by the Duke Power Quality Assurance program (Reference 21).

## 1.2 Definition of Terms

Presented below are terms which will be needed throughout the text of the report:

a/o	atom percent
ARI	all rods in
ARO	all rods out
axial offset (AO)	$\frac{P_T - P_B}{P_T + P_B}$ , where $P_T$ is the integrated power in the top half of the core, and $P_B$ is the integrated power in the bottom half of the core
$\beta_i$	delayed neutron fraction for group i
$\beta_{eff}$	effective delayed neutron fraction in core
BOL	beginning of life
BP	burnable poison
BU	fuel burnup
$C_B$	Chemical shim boron concentration in the main coolant
CZP	cold zero power
EOL	end of cycle life
EQXE	equilibrium xenon condition
GWD/MTU	Gigawatt days per metric ton of initial uranium metal, 1 GWD/MTU is 1000 MWD/MTU
HFP	hot full power
HZP	hot zero power

$\bar{I}$	delayed neutron importance factor
$\Delta I$ or Axial Flux Difference (AFD)	flux difference between the top and bottom halves of the core; in this report, $\Delta I$ is a calculated value, rather than a difference between measured signals from the excore detectors
$K(z)$	$F_Q^T$ normalized to the maximum value allowed at any core height
$t^*$	prompt neutron lifetime
MOL	middle of cycle life
MWD/MTU	measure of energy extracted per unit weight of initial uranium metal fuel; is equal to 1 megawatt times 1 day, divided by 1 metric ton of uranium
pcm	percent mille (a reactivity change that equals $10^{-5} \Delta\rho$ )
ppm	parts per million by weight; which specifies the amount of chemical shim boron present by weight in the main coolant system
radial local	ratio of assembly maximum rod to assembly average x-y power
RCCA	rod cluster control assembly; the type of control rod assembly used in McGuire and Catawba. (All RCCA are full length absorbers for both plants.)
$\rho$	reactivity
$\Delta\rho$	$\frac{K_1 - K_2}{K_1 \times K_2}$ , where $K_1$ and $K_2$ are eigenvalues obtained from two calculations where only one parameter was varied
shutdown margin	amount of negative reactivity ( $\rho$ ) by which a reactor core is maintained in a HZP subcritical condition after a control rod trip
step	unit of control rod travel equal to 0.625 inch
TMOD	moderator temperature; defined as the temperature corresponding to the average water enthalpy of the core
$T_{res}$	resonance temperature of the fuel
w/o	weight percent

Power distributions will be quantified in terms of hot channel factors. These factors are a measure of the peak pellet power and the energy produced in the coolant. The factors are:

<u>Power density</u>	thermal power produced per unit volume of the core (KW/liter)
<u>Linear Power Density</u>	thermal power produced per unit length of active fuel (KW/ft)
<u>Average Linear Power Density</u>	total thermal power produced in the core divided by the total active fuel length of all fuel rods in the core
<u>Local Heat Flux</u>	local heat flux on the cladding surface (BTU/ft <sup>2</sup> /hr)
<u>Rod Power or Integral Power</u>	is the length integrated linear power density in one rod (KW)

Various hot channel factors are:

$F_Q^T$ , Heat Flux Hot Channel Factor, the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux, including conservatisms for fuel pellet and rod dimensional uncertainties.

$F_Q^N$ , Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average linear power density, assuming nominal fuel rod and pellet dimensions.

$F_Q^E$ , Engineering Heat Flux Hot Channel Factor, is the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter. Combined statistically the net effect is typically a factor of 1.03 to be applied to calculated KW/ft.

$F_{AH}^N$ , Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

## 2. FUEL DESCRIPTION

The reactor cores for McGuire and Catawba contain 193 fuel assemblies. Each fuel assembly consists of 264 fuel rods, 24 guide thimble tubes and 1 instrumentation thimble tube assembled in a square 17x17 lattice. The assembly structure consists of top and bottom nozzles and grid assemblies positioned axially along the fuel assembly. Each fuel rod contains a column of stacked fuel pellets.

Detailed design data for the fuel pellets, fuel rods, fuel assembly, and reactivity control components can be found in the Updated Final Safety Analysis Report<sup>18,19</sup> (UFSAR).

### 3. NUCLEAR CODE SYSTEM

#### 3.1 Introduction

Nuclear design calculations performed for Westinghouse reactors employ the EPRI-ARMP code system<sup>1</sup> and the CASMO-2 code<sup>2</sup> or the CASMO-3/SIMULATE-3P code system. A summary description of each code is given in Appendix A. The ARMP/CASMO-2 and the CASMO-3/SIMULATE-3 code sequences have been reviewed and approved by the NRC for use in the design of reload cores for the McGuire and Catawba Nuclear Stations by Duke Power<sup>28,32</sup>.

Presented in this section will be a description of the sequence, cross section preparation and parameterization, and reload design modeling procedures.

The nuclear calculational system enables the nuclear engineer to numerically model and simulate the reactor core. The ARMP/PDQ code system sequence used by Duke Power for McGuire and Catawba is outlined in Figure 3-1. The CASMO-3/SIMULATE-3 code system sequence is outlined in Reference 28.

#### 3.2 Sources of Input Data

The determination of nuclear fuel loading patterns and core physics characteristics requires an accurate database consisting of:

1. Core operating conditions
2. Dimensional characteristics
3. Composite materials and mechanical properties
4. Nuclear cross sections

The UFSAR, supplemented by vendor reports and open literature, is the primary source of data for Items 1 to 3. These data are used as input to the cross section generators and core simulators. A secondary data source for the core simulators are estimates of fuel pellet volume-averaged temperatures which are calculated by fuel performance codes as a function of power and burnup.

The cross section generators CASMO-2 and EPRI-CELL<sup>8</sup> use processed ENDF/B libraries unique to each code.

EPRI-CELL is a unit cell lattice code which is used to calculate few-group cross sections for fuel and non-fuel compositions as shown in Table 3-1.

CASMO-2 uses a processed version<sup>9</sup> of the ENDF/B-3 library. Group cross sections of  $\sigma_a$ ,  $\sigma_f$ ,  $v\sigma_f$ ,  $\sigma_{tr}$ , scattering kernels, resonance integrals, and fission product data are among the data contained in this library. The 69 group library is divided into 14 fast, 13 resonance, and 42 thermal energy range groups. A 25 group version of this library is also used.

The EPRI-CELL library is derived from the ENDF/B-4 library<sup>10</sup>. The 97 energy groups are divided into 62 fast groups and 35 thermal groups.

CASMO-3 cross section development for SIMULATE-3 is described in Reference 28.

### 3.3 Cross Section Preparation

In order to model the neutronics of a reload core, it is necessary to generate a set of cross sections for use in a diffusion theory code. CASMO-3, CASMO-2, and EPRI-CELL are the cross section generators that may be used.

Inputs which are provided to these codes are: lattice materials and geometry, temperatures for fuel, clad, and moderator, effective resonance temperature, fuel enrichment, soluble boron concentration, number of depletion steps, length of depletion steps, etc. Table 3.1 shows the core materials or compositions which are parameterized by CASMO-2 and EPRI-CELL.

PDQ requires pin cell cross sections calculated as described Sections 3.3.1 and 3.3.2.

Reference 28 describes the cross section and nuclear data requirements for SIMULATE-3.

### 3.3.1 Fuel Calculations

Calculations for fuel regions employ fixed fuel and moderator temperatures for the cell depletion. Restart calculations are performed at various burnups to parameterize fuel cell cross sections at varying moderator and fuel temperatures.

The output of EPRI-CELL and CASMO-2 consists of sets of broad group cross sections which characterize the regions of interest. Cross sections are then formatted into PDQ07 tableset structure using either NUPUNCHER<sup>11</sup> (1-dimensional parameterization), or MULTIFIT<sup>12</sup> (2 and 3 dimensional parameterization or g-factors). Cross sections from CASMO-2 are similarly formatted using CHART<sup>13</sup>.

### 3.3.2 Non-Fuel Calculations

Cross sections for empty control rod guide tubes, reflector, instrument thimble, and the water gap are calculated with either EPRI-CELL or CASMO-2. Separate cross section sets are generated for various moderator temperatures.

Strong absorbers such as RCCA and BP require reaction rate matching to obtain diffusion theory equivalent cross sections. Calculations using CASMO-2 are performed for these strong absorbers where first a transport theory method determines absorption rates, and then a series of diffusion theory iterations are performed to calculate a g-factor such that the absorption rates agree between both types of flux solutions. These g-factors are then incorporated in the tabulated cross sections.

RCCA cross sections are evaluated at BOL HFP conditions, while BP cross sections are evaluated with an HFP depletion calculation.

In both types of g-factor calculations, lattices with expected core average enrichments are used. Core baffle cross sections are also calculated with CASMO-2. A lattice geometry is employed, with the baffle material density

modified to reflect real versus modeled thickness in the quarter core PDQ07 discrete pin model.

### 3.4 PDQ07 Models

The PDQ07 few-group diffusion-depletion code is employed for core modeling. Two different models are used.

The first is the assembly colorset model, which is used for calculating  $k_{\infty}$  and  $M^2$  data for EPRI-NODE-P 3-D simulations. The second model is the quarter-core model, which is used for X-Y power distribution calculations and for normalization of EPRI-NODE-P radial power distributions.

Aspects which are common to both PDQ07 models are:

1. Discrete pin representation
2. Two-group cross sections
3. Mixed Number Density thermal group constants
4. Improved Removal Treatment removal cross sections
5. Microscopic cross section parameterization for uranium, plutonium, burnable absorber, soluble boron, xenon, samarium, and lumped fission products
6. Thermally expanded geometry - pin pitch and assembly pitch

#### 3.4.1 Colorset PDQ07 Modeling

The colorset PDQ07 model consists typically of four quarter assemblies arranged such that a representative neutron spectrum is obtained. Figure 3-2 shows a typical colorset geometry.

To accommodate asymmetric burnable poison rod loadings, full or half assembly geometries are used. The EPRI-ARMP PWR Procedures<sup>14</sup> are used for modeling, and most of the conventions and guidelines are employed.

Fuel types are determined according to enrichment and BPRA loading.  $k_{\infty}$  and  $M^2$  data for each fuel type are calculated by performing the following operations:

- I. BOL Cases - 0 MWD/MTU
  - A.  $k_{\infty}$  and  $M^2$  - unrodded vs.  $T_{mod}$  (Inlet, Average, Outlet)
  - B.  $k_{\infty}$  and  $M^2$  - rodded vs.  $T_{mod}$  (Inlet, Average, Outlet)
  - C. Boron worth
  - D. Doppler worth
  
- II. Depletion Data - Exposure dependent data
  - A. Nominal HFP depletion at constant  $T_{mod}$ ,  $T_{fuel}$
  - B. Branch cases from depletion
    - 1. Boron worth
    - 2. Control rod worth
    - 3. Equilibrium Xenon worth
    - 4. Doppler worth
    - 5. Moderator temperature worth

In the above PDQ07 branch calculations, only one parameter is varied, allowing a partial derivative of reactivity with respect to that parameter to be calculated.

The parameterization procedure involves approximately 150-200 cases, depending on the number of depletion steps.

The output from the PDQ07 colorset cases is written to PDQ07 integral files which in turn are processed by the linking codes EPRI-FIT<sup>15</sup> and SUPERLINX<sup>16</sup> to yield B-constant data for EPRI-NODE-P.

#### 3.4.2 Quarter Core PDQ07 Model

Two-dimensional X-Y core simulations are performed with a discrete pin PDQ07 model. Assembly average and maximum pin powers are calculated, along with

critical boron concentrations and other reactivity parameters. Moderator and Doppler feedbacks are incorporated in this model.

The geometry employed utilizes thermally expanded dimensions. Figure 3-3 shows a geometry of a fuel assembly and water gaps. Figure 3-4 shows the complete quarter core mesh layout.

The plane of solution used in quarter core analyses is the axial midplane or the six foot level of the active fuel. Moderator and Doppler feedbacks are employed as described in Reference 17.

The depletion calculation is used to determine burnup dependent parameters. The soluble boron concentration is modified at each timestep such that the reactor is approximately critical.

Timesteps are taken using point depletion so that the core average exposure advances by: 150, 500, 1000, 2000, ...,  $N * 2000$  MWD/MTU until the end of cycle is reached.

PDQ07 depletion calculations are used to determine the following parameters:

1. Assembly average and maximum pin powers
2. Core reactivity
3. Nuclide reaction rates: Fission and absorption
4. Nuclide inventories
5. Neutron flux distributions

Other calculations performed with the quarter core model may include:

1. RCCA bank worths
2. Boron and xenon worths
3. Power deficits
4. Moderator and Doppler temperature coefficients

Cases 2, 3, and 4 are usually performed with a nodal code; however, these are shown to demonstrate the quarter core model's flexibility.

### 3.5 EPRI-NODE-P Model

EPRI-NODE-P is the nodal code employed for three-dimensional analyses and reactivity studies. A summary description of EPRI-NODE-P is given in Appendix A. Typical calculations which are performed with the Duke Power EPRI-NODE-P model are:

1. Full core ejected rod worths
2. Power deficits
3. Differential rod worths
4. Axial xenon transients
5. Three-dimensional power distributions, etc.

The quarter core model uses one radial node per assembly and eighteen axial nodes.

Each unique combination of enrichment and BPRA loading comprises a separate fuel type. The fuel type is parameterized by sets of fitting coefficients which determine reactivity due to control rods, exposure, soluble boron, xenon, etc. Doppler and moderator feedbacks are explicitly treated.

EPRI-NODE-P radial power distributions are normalized near the beginning of cycle. Assembly average powers are adjusted to match quarter core PDQ07 calculations with radial albedoes -  $\alpha_H$  and an internal leakage factor -  $g_H$ . The axial power distribution, is adjusted using vertical leakage factors  $\alpha_V$  determined from comparisons of calculated and measured axial power distributions from benchmark core follow calculations.

Sections 5, 6, 7, and 9 discuss in depth calculational procedures of EPRI-NODE-P. Sections 10 and 11 address benchmarking of EPRI-NODE-P and PDQ07 calculations to measured power and reactivity data.

### 3.6 SIMULATE-3 Model

The SIMULATE-3 methodology used by Duke Power Company is described in Reference 28.

TABLE 3-1

EPRI-CELL/CASMO-2 Cross Sections  
Calculation by Composition

1. EPRI-CELL

- a. Uranium Fuel
- b. Empty Control Rod Guide Tube/Instrument Tube
- c. Reflector
- d. Water Gap

2. CASMO-2

- a. Burnable Poison Rod Assembly
- b. Gadolinia doped Uranium Fuel
- c. Control Rod - AgInCd or B<sub>4</sub>C
- d. Baffle

FIGURE 3-1

Nuclear Flow Chart for EPRI-ARMP

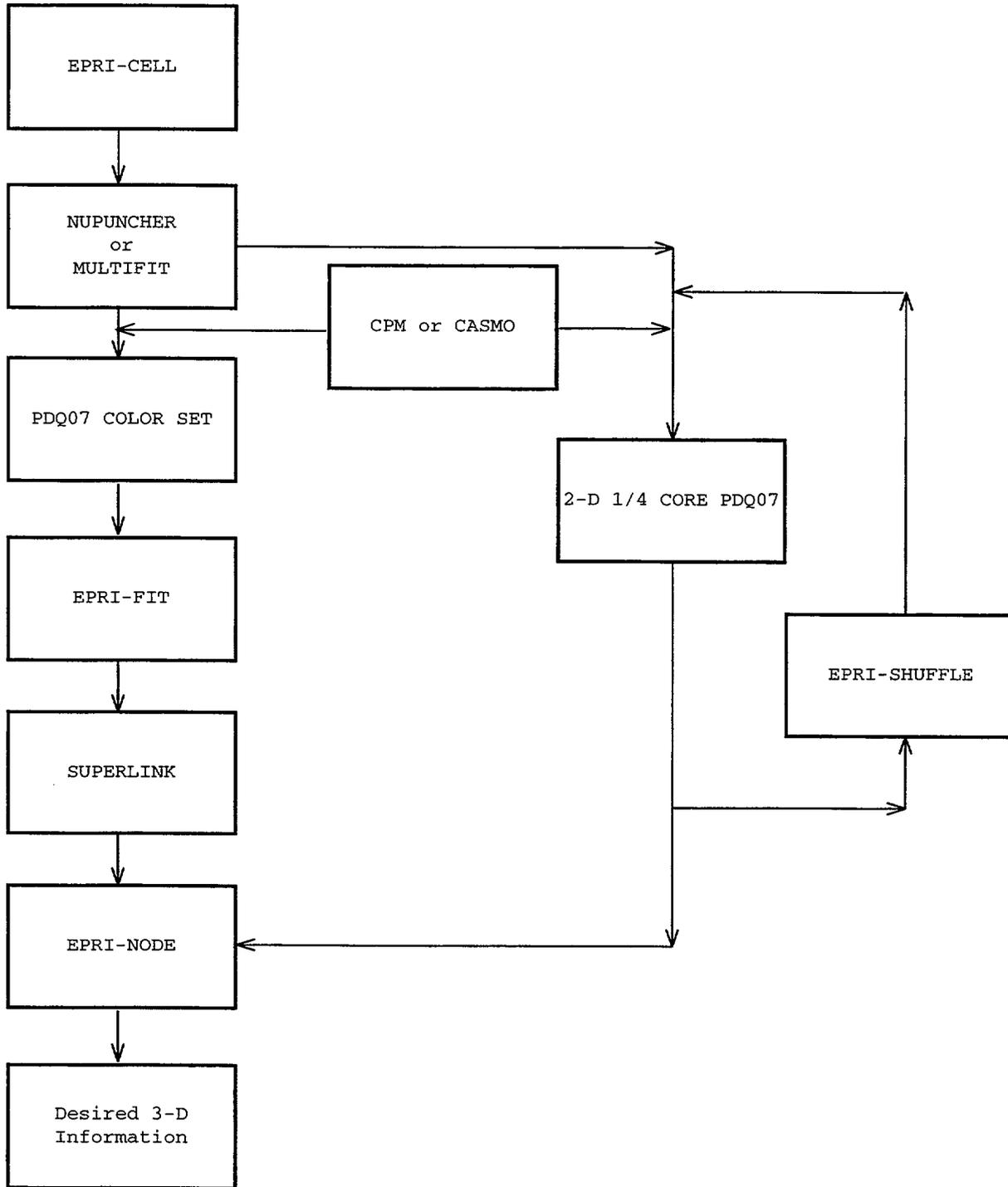
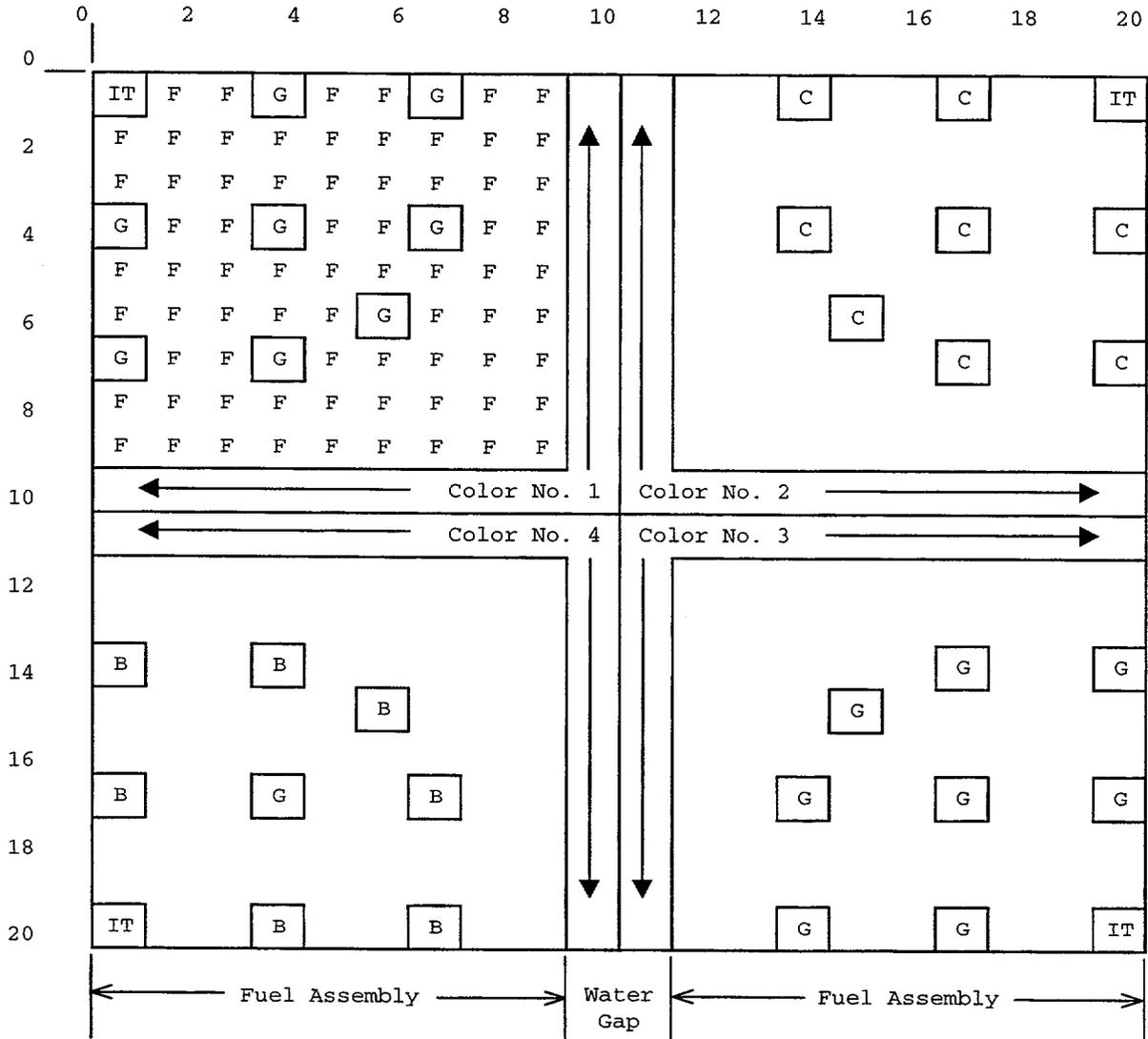


FIGURE 3-2

17 x 17 Assembly PDQ07 Colorset Geometry

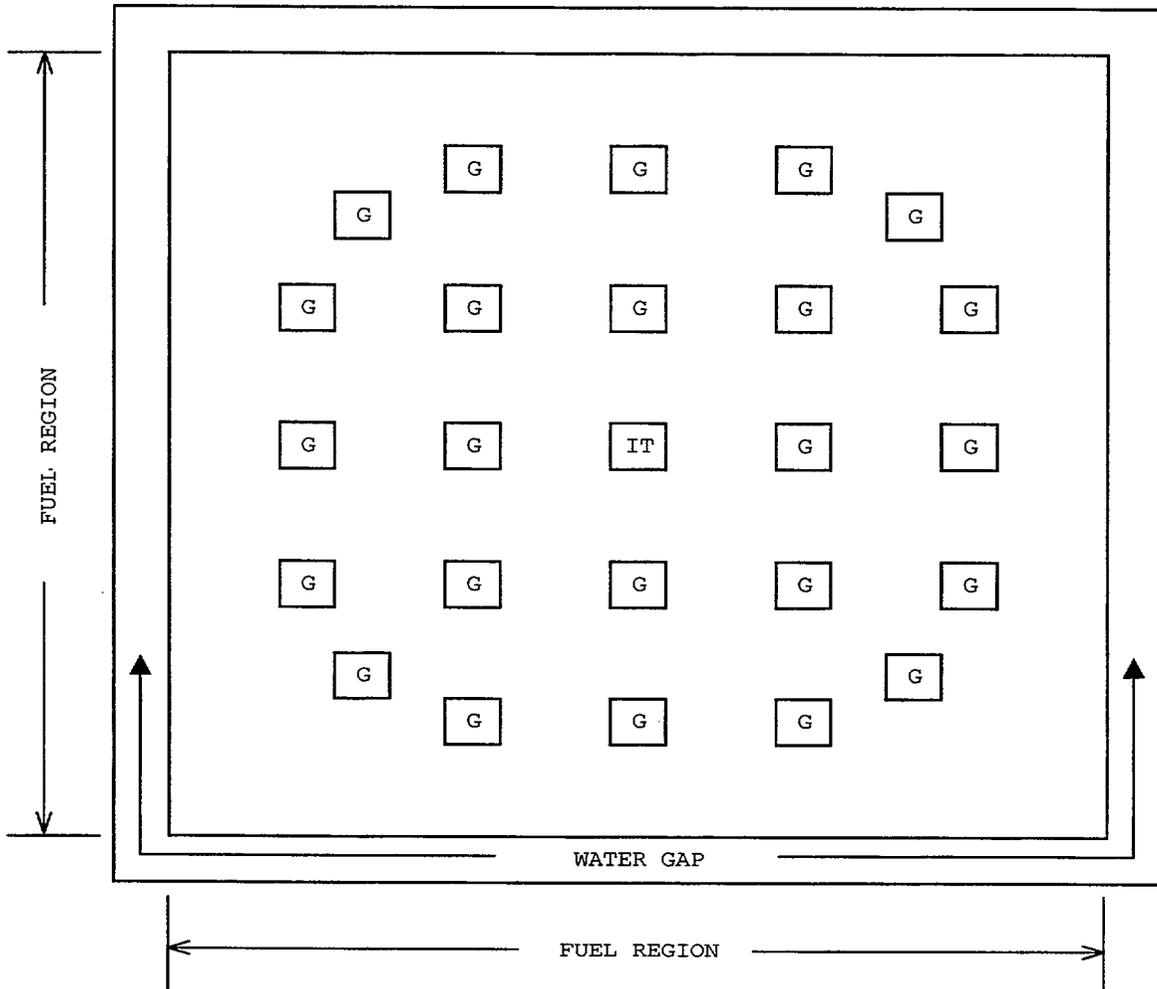


LEGEND:

- IT Instrument Tube
- C Guide Tube With Control Rod
- B Guide Tube With Burnable Poison Rod
- G Empty Guide Tube
- F Fuel Rod

FIGURE 3-3

PDO07 Quarter Core Model Assembly Geometry

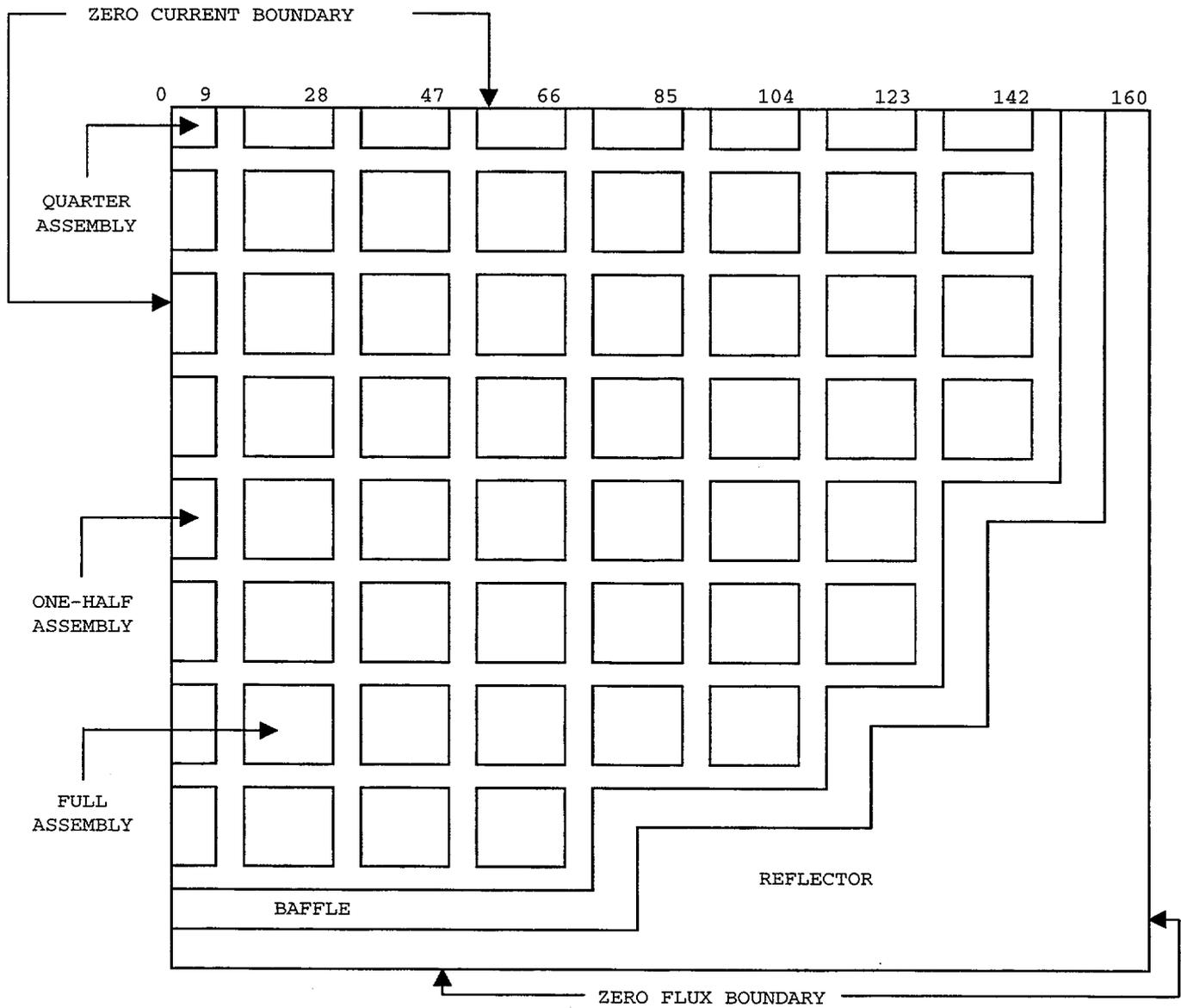


LEGEND:

- IT Instrument Tube
- G Control Rod Guide Tube

FIGURE 3-4

PDQ07 Quarter Core 17 x 17 Geometry



## 4. FUEL CYCLE DESIGN

### 4.1 Preliminary Fuel Cycle Design - Initialization

To commence the design of a reload, core operation requirements along with planned changes in reactor primary or secondary systems are assembled. A preliminary loading pattern is designed which meets operational requirements. Physics data from the preliminary design are compared with core operating requirements to determine the adequacy of the reload design. Likewise, physics data are compared to Technical Specifications to verify that the preliminary design will conform to existing limits.

#### 4.1.1 Review of Design Information

The preliminary design procedure requires assembly of design information which in turn will determine the cycle's operational capabilities. Typical design data are shown on Table 4-1.

Table 4-1 and other pertinent nuclear design data are assembled and reviewed for consistency with previous sets of design data.

#### 4.1.2 Determination of Cycle-Specific Operating Requirements

Design data from Table 4-1 uniquely determines expected operating requirements and capabilities. For instance, a longer than annual cycle may require a low leakage loading pattern and the use of burnable absorber rods. A larger energy requirement than can be provided by normal operation with a given reload enrichment may require a planned power coastdown at end of cycle. Similarly, other design considerations will govern the rest of the cycle-specific operational characteristics.

#### 4.1.3 Preliminary Loading Pattern and Reload Region Determination

The purpose of a preliminary loading pattern analysis is to determine the uranium and separative work requirements to meet a desired cycle lifetime. The cycle lifetime is confirmed by modeling the depletion of a reload core. If the number of new fuel assemblies and the enrichment are known, this analysis will yield an estimate of the cycle lifetime.

#### 4.2 Final Fuel Cycle Design

Having determined the number and enrichment of the fuel assemblies during the preliminary fuel cycle design, the final fuel cycle design (FFCD) concentrates on optimizing the placement of fresh and burned assemblies and burnable poison assemblies (if any) to result in an acceptable fuel cycle design. It must meet the following design criteria with appropriate reductions to account for calculational uncertainties:

1.  $F_{\Delta H}^N$  must meet the limits specified in the Technical Specifications.
2. Moderator Temperature Coefficient must meet the limits specified in the Technical Specifications.
3. Maximum fuel burnup must be less than the limits applicable for the type of fuel being used.
4. Shutdown Margin must meet limits specified in the Technical Specifications.
5. Maximum linear rod power must meet the limit specified in the Technical Specifications.
6. UFSAR Chapter 15 related physics parameters must be validated.

A preliminary verification of the above is made in the FFCD. Final verification of the above is made in fuel mechanical performance analyses, thermal and thermal-hydraulic analyses, safety-analysis physics parameters analyses, and maneuvering analyses.

#### 4.2.1 Fuel Shuffle Optimization and Cycle Depletion

The preliminary fuel shuffle scheme is modified to minimize power peaking. This is accomplished by a trial and error type search until an acceptable BOC power distribution results.

The reload design's "burnup window" is assessed to ensure that applicable safety criteria are met.

The cycle is then depleted to various times in the cycle to verify that power peaking versus burnup remains acceptable. The shuffling variations include rearranging the location of the burned or fresh fuel assemblies, BP placement, and rotation of the spent fuel assemblies. These calculations are typically performed assuming quarter core symmetry.

The core neutronic model resulting from the FFCD is the core model used for other nuclear design calculations.

The shuffle pattern determined in the FFCD may later need to be modified based upon results obtained in the remaining nuclear calculations.

#### 4.2.2 Rod Worth Calculations

Control rods serve several functions in the McGuire and Catawba reactors. The primary function is to provide adequate shutdown capability during normal and accident conditions. They are also used to maintain criticality during power maneuvers and to maintain the Axial Flux Difference (AFD) within Technical Specification limits. Since the presence of control rods influences both power distributions and criticality, it is necessary in many calculations to evaluate not only the reactivity effect but also the perturbation that a given rod configuration has on the power distribution.

McGuire and Catawba are typically operated in the ARO or feed and bleed mode. All RCCA have full length absorber rods. During full power operation, Control

Bank D is typically inserted about six inches (215 steps withdrawn) in the active core. Control Bank D is used to control power during load follow maneuvers, and in conjunction with Control Banks B and C, to achieve criticality during startup.

Calculations of control rod worth and power peaking ( $F_0$ ) are used in the safety analysis of the reload core. The calculations discussed in subsequent sections include the following:

1. Control Rod Worths
2. Shutdown Margin
3. Ejected Rod Analysis
4. Dropped Rod Analysis

#### 4.2.2.1 Control Rod Worths

RCCA bank locations in McGuire and Catawba usually are fixed and do not change from cycle to cycle. The worth of each control bank (A, B, C, D) is calculated at BOC and EOC, at HFP and HZP. The total rod worth (ARI) is calculated at BOC, EOC, and any limiting burnup at HZP only for use in the shutdown margin calculation.

#### 4.2.2.2 Shutdown Margin

Searches for the highest worth stuck rod are performed at BOC, EOC, or any limiting burnup for HZP conditions using full core calculations.

Table 4-2 summarizes the results of a shutdown margin calculation. The total rod worth described in section 4.2.2.1 is shown as Item 1. Item 2 is the worth of the highest stuck rod. The total worth reduced by the stuck rod worth is shown as the net worth (Item 3). A calculational uncertainty of 10% is subtracted off in Step 4, and Step 5 shows the available rod worth.

The required rod worth is calculated next in Steps 6-9. The power deficit obtained by modeling the core at HFP and HZP (using constant boron and xenon) and subtracting the reactivities is shown as Item 6. This reactivity insertion accounts for Doppler and Moderator deficits. The maximum allowable inserted rod worth, Item 7, is obtained from the allowable rod insertion and the integral rod worth curve for that insertion. This accounts for the maximum allowed rod insertion at HFP. An axial flux redistribution occurs when the power level is reduced from HFP to HZP. This redistribution causes an increase in reactivity. If Item 6 is calculated using a 3-D model, no additional penalty is required. If Item 6 was calculated using a 2-D model, where redistribution effects are not modeled, an additional reactivity penalty is assessed as Item 8. The sum of these required worths (Item 9) is the total required worth.

Additional reactivity penalties are applied to both the power defect and the rod insertion allowance to account for xenon redistribution effects.

The shutdown margin is shown as Item 10 and is defined as the total available worth minus the total required worth. Shutdown margin requirements are specified in the Technical Specifications.

#### 4.2.2.3 Rod Insertion Limit Verification

As part of the reload design procedure, the Rod Insertion Limits are verified for applicability in the reload core (see Section 5.5).

#### 4.2.2.4 Ejected Rod Analysis

The UFSAR<sup>18,19</sup> presents the limiting criteria for the ejected rod accident. The accident has been analyzed at HFP and HZP conditions at BOL and EOL.

Ejected rod calculations are performed on a cycle-specific basis to verify that UFSAR accident analysis values are not exceeded.

Calculational limits are established using the methodology described in References 30 and 31.

To verify that the ejected rod parameters are within calculational limits, ejected rod calculations are performed at BOC and EOC or at other limiting times in cycle life at both HFP and HZP.

The HZP ejected rod calculations are performed using full core geometry with Control Banks B and C at their insertion limits in the core and with Control Bank D fully inserted. Single rods in Control Banks D, C, and B are removed in subsequent cases and the worth of the ejected rod is calculated by subtracting the reactivities of the cases before and after the rod was removed. The fuel and moderator temperature is held constant and equal to the HZP moderator temperature for these calculations. The highest worth calculated by the above procedure is the worst ejected rod at HZP. If the ejected rod worth exceeds the calculational limit, one of the following is performed: an evaluation, a revision to the rod insertion limits, a reanalysis of the Chapter 15 REA analysis, or a redesign of the core loading pattern.

The HFP ejected rod calculations are performed in a similar manner to the HZP calculations with the exceptions that only Control Bank D is inserted at the HFP insertion limit and that the fuel temperature and moderator temperatures correspond to those of HFP conditions. The HFP ejected rod worths are determined without thermal feedback to be conservative. If the ejected rod worth exceeds the calculational limit, one of the following is performed: an evaluation, a revision to the rod insertion limits, a reanalysis of the Chapter 15 REA analysis, or a redesign of the core loading pattern.

A parallel analysis, addressing core peaking, is performed at the same time as the rod worth analyses. Additional discussion of the rod ejection accident analysis methodology is contained in Reference 30.

#### 4.2.2.5 Dropped Rod Analysis

The UFSAR<sup>18,19</sup> presents the limiting criteria for the dropped rod accident. The calculational limits are established using the methodology described in Reference 30.

A core model is used that evaluates pre-drop and post-drop physics parameters for possible dropped rod combinations. The physics parameters important to the dropped rod analysis are presented in Reference 30.

#### 4.2.3 Fuel Burnup Calculations

The reload design must meet fuel burnup limits. This is confirmed during the final fuel cycle design. Depletion calculations yield core, assembly average, single fuel rod burnups, and peak local burnups which can be compared to the design limits.

#### 4.2.4 Reactivity Coefficients and Defects

Reactivity coefficients define the reactivity insertion for small changes in reactor parameters such as moderator temperature, fuel temperature, and power level. These parameters are input to the safety analysis and used in modeling the reactor response during accidents and transients. Whereas reactivity coefficients represent reactivity effects over small changes in reactor parameters, reactivity defects usually apply to reactivity inserted from larger changes typical of HFP to HZP. An example of a reactivity deficit is the power defect from HFP to HZP used in the shutdown margin calculation. A different way of looking at the terms is that the coefficient when integrated over a given range yields the defect, or the coefficient is the partial derivative of reactivity with respect to one specific parameter.

Coefficients of reactivity are calculated using the core model. First a nominal case is established at some reference conditions. 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 is calculated as the reactivity coefficient.

#### 4.2.4.1 Doppler Coefficient

The Doppler Coefficient 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, the resonances broaden increasing the chance that a neutron will be absorbed and thus decreasing the core reactivity.

If Case 1 represents the reference case with an effective fuel temperature  $T_1$  (and  $K^1$  effective) and Case 2 represents a second case where the fuel temperature has been increased or decreased by approximately 50°F and is  $T_2$  (and  $K^2$  effective), the Doppler Coefficient is mathematically calculated from the following equation:

$$\alpha_D = \frac{\left( \frac{K_{eff}^1 - K_{eff}^2}{K_{eff}^1 \times K_{eff}^2} \right)}{(T_1 - T_2)} \times 10^5 = \Delta\rho \text{ (pcm/°F)}$$

Doppler Coefficients are calculated at various core conditions to validate safety analysis assumptions.

#### 4.2.4.2 Moderator Temperature Coefficient

The Moderator Temperature Coefficient (MTC) is the change in reactivity produced by a small change in moderator temperature. In McGuire and Catawba, the average core moderator temperature increases linearly as power is escalated from 0 to 100% HFP. Therefore, for accident and transient analyses it is necessary to know the moderator temperature coefficient over a range of moderator temperatures from CZP to HFP.

These analyses are performed by modeling changes in the core average moderator temperature. Cases are run changing the moderator temperature from the reference temperature. If the cases and resulting  $K_{\text{effective}}$ 's are identified as Case 1 ( $T_{\text{MOD1}}, K_{\text{eff}}^1$ ) and Case 2 ( $T_{\text{MOD2}}, K_{\text{eff}}^2$ ) the moderator temperature coefficient is calculated from the following equation:

$$\alpha_{T_{\text{MOD}}} = \frac{\left( \frac{K_{\text{eff}}^1 - K_{\text{eff}}^2}{K_{\text{eff}}^1 \times K_{\text{eff}}^2} \right)}{(T_{\text{MOD1}} - T_{\text{MOD2}})} \times 10^5 = \Delta\rho \text{ (pcm/}^\circ\text{F)}$$

Since the reload core is designed with a predetermined flexibility (burnup window), the MTC is verified to be within its design limit for the current cycle considering the burnup window of the previous cycle.

#### 4.2.4.3 Isothermal Temperature Coefficient

The fractional change in reactivity due to a small change in core temperature is defined as the isothermal temperature coefficient (ITC) of reactivity. This is equal to the sum of the moderator and Doppler temperature coefficients and may be explicitly calculated at HZP for isothermal conditions ( $T_{\text{FUEL}}=T_{\text{MOD}}$ ) by changing both the fuel and moderator temperatures from the reference HZP moderator temperature.

#### 4.2.4.4 Power Coefficient and Power Defect

The power coefficient of reactivity is the core reactivity change resulting from an incremental change in core power level. The power defect is usually the total reactivity change associated with a power level change from HZP to HFP.

The power coefficient is defined by the following equation:

$$\alpha_P = \frac{\left( \frac{K_{\text{eff}}^1 - K_{\text{eff}}^2}{K_{\text{eff}}^1 \times K_{\text{eff}}^2} \right)}{(P_1 - P_2)} \times 10^5 = \Delta\rho \text{ (pcm/\%FP)}$$

where:  $K_{\text{eff}}^1$  is K-effective for the core at power  $P_1$  (%)  
 $K_{\text{eff}}^2$  is K-effective for the core at power  $P_2$  (%)

Neglecting second order effects this equation is equivalent to the following:

$$\alpha_P = \alpha_{\text{TMOD}} \frac{\Delta \text{TMOD}}{\Delta P} + \alpha_D \frac{\Delta \text{TFUEL}}{\Delta P}$$

where:  $\alpha_{\text{TMOD}}$  is the moderator temperature coefficient and  $\alpha_D$  is the Doppler temperature coefficient.

Since the power coefficient should include flux redistribution effects resulting from axial variations in burnup and isotopics as well as non-uniform fuel temperature distributions, it should be performed using a 3-D simulator with thermal hydraulic feedback. If the calculation is performed using a 2-D model then it should be corrected for the 3-D effects.

A typical power coefficient calculation for HFP would proceed in the following manner: The HFP case is run, and the core  $K_{\text{eff}}$  is calculated ( $K_{\text{eff}}^1$ ). Then a second case is run with the core power level reduced while holding control rods, boron, and xenon constant. The  $K_{\text{eff}}$  from this case,  $K_{\text{eff}}^2$ , is used along with the results from the reference case to calculate the power coefficient:

$$\alpha_P = \frac{\left( \frac{K_{\text{eff}}^1 - K_{\text{eff}}^2}{K_{\text{eff}}^1 \times K_{\text{eff}}^2} \right)}{(P_1 - P_2)} \times 10^5 = \Delta p \text{ (pcm/\%FP)}$$

The power defect is calculated for use in the shutdown margin calculation (see Section 4.2.2.2) and is the reactivity change from HZP to HFP. This calculation should be performed in three dimensions to satisfactorily model the axial flux redistribution, however, a two dimensional calculation may be performed and corrected for this flux redistribution phenomenon. These calculations are usually performed at BOC and EOC.

Both HFP and the HZP cases should have the equilibrium xenon concentration corresponding to HFP. The power defect is calculated from the following equation:

$$\text{Power Defect (HFP to HZP)} = \left( \frac{K_{\text{eff}}^1 - K_{\text{eff}}^2}{K_{\text{eff}}^1 \times K_{\text{eff}}^2} \right) \times 10^5 = \Delta p \text{ (pcm)}$$

where:  $K_{\text{eff}}^1$  is K-effective at HZP and  
 $K_{\text{eff}}^2$  is K-effective at HFP

#### 4.2.4.5 Miscellaneous Coefficients

For reload design, certain other coefficients of reactivity are not routinely calculated. These include moderator density coefficient, moderator pressure coefficient, and moderator void coefficient. These coefficients can be calculated in an analogous manner by varying the appropriate core reactivity parameters.

#### 4.2.4.6 Boron Related Parameters

Critical boron concentrations for various core conditions during cycle lifetime are calculated using the core model. Table 4-3 lists typical conditions that critical boron concentrations and boron worths are calculated. In addition to these, an ARO critical boron letdown curve is generated for HFP EQXE.

#### 4.2.4.7 Xenon Worth

Xenon worth calculations are performed to support plant operation (e.g. startup after trip), rather than as a safety parameter. Xenon worth is calculated as a function of burnup. The equilibrium xenon worth is calculated as the difference in reactivities between the equilibrium and no xenon cases. The peak xenon worth is calculated as the difference between the peak and no xenon cases. The peak xenon worth is determined at approximately 8 hours following a reactor shutdown from HFP, equilibrium xenon conditions.

#### 4.2.4.8 Kinetics Parameters

The kinetics behavior of the nuclear reactor is often described in terms of solutions to the kinetics 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, the six group precursor decay constants, and the prompt neutron lifetime are known.

PDQ07 and DELAY may be used to calculate these parameters<sup>20</sup>. PDQ07 is used to obtain spatially averaged isotopic fission rates as a function of burnup and DELAY calculates kinetics parameters and then uses these parameters to solve the Inhour equation and thereby relate the stable reactor period to the reactivity insertion. CASMO-3 data libraries contain delayed neutron data, and SIMULATE-3 is capable of calculating the core averaged kinetics parameters of interest.

Calculations are performed at BOL and EOL. The sum of the six group  $\beta_i$  effective,  $\beta$  effective, for the new reload cycle is compared to those values used in the UFSAR.

#### 4.2.5 Assessment of the Fuel Cycle Design

Once the FFCD calculations are performed, the resultant data are assessed for validity and consistency with core operation requirements as well as fuel design and safety analysis limits.

Design criteria for a reload design are outlined in Section 4.2. A preliminary verification of these criteria or parameters important to these criteria is made in the FFCD. Additional calculations that validate a reload design are described in Sections 5 - 7 of this report.

TABLE 4-1

Nuclear Design Data  
For Reload Design

1. Power operation mode: load follow or base load.
2. Vessel internal or core component modifications.
3. Expected minimum and maximum cycle burnups.
4. Feed enrichment (if already contracted for).
5. Number and design of feed assemblies.
6. RCS hydraulic conditions.

TABLE 4-2

Shutdown Margin Calculation

	<u>BOC, % <math>\Delta\rho</math></u>
Available Rod Worth	
1. Total rod worth, HZP	6.46
2. Maximum stuck rod, HZP	<u>-1.39</u>
3. Net Worth	5.07
4. Less 10% uncertainty	<u>.51</u>
5. Total available worth	4.56
Required Rod Worth	
6. Power defect, HFP to HZP	.88
7. Max allowable inserted rod worth	1.36
8. Flux/Xenon redistribution	<u>.63</u>
9. Total required worth	1.87
10. Shutdown Margin (total avail. worth minus total required worth)	2.69

NOTE: Required shutdown margin is specified in the Technical Specifications.

TABLE 4-3

Typical Boron Parameters

Critical Boron - ppm

HZP, ARO, BOC, No Xenon

HZP, Bank D inserted, BOC, No Xenon

HZP, Bank D + C inserted, BOC, No Xenon

HFP, ARO, EQXE vs exposure

Boron Worth - ppm/% $\Delta\rho$

HFP, EQXE, ARO vs. exposure

HZP, NOXE, ARO vs. exposure

Boron Worth Versus Boron Concentration - HZP, NOXE

BOC

EOC

## 5. NODAL ANALYSIS METHODOLOGY

### 5.1 Purpose and Introduction

Nodal analysis allows for modeling of the reactor core in three-dimensions and for performing calculations which because of either code restraints or economic restraints cannot be performed by any other means. Examples of nodal code capabilities include:

1. Calculations which need a three-dimensional geometry such as differential rod worths, axial xenon transients and three-dimensional power distributions.
2. Calculations which need a full-core geometry such as stuck and ejected rod worths.

This section addresses the role of a nodal code in performing cycle depletions, generating rod worth data, determining shutdown margins and shutdown boron concentrations, setting control rod insertion limits, and determining trip reactivity worths and shapes.

A nodal code is also used to calculate many of the startup test parameters and core physics parameters described in Section 9 of this report.

The nodal codes used for McGuire and Catawba analyses are EPRI-NODE-P and SIMULATE-3. (See descriptions in Section 3 and Appendix A).

EPRI-NODE-P can be run with either a quarter-core or a full-core geometry. The McGuire and Catawba models utilize one radial node per assembly and twelve to eighteen axial nodes. EPRI-NODE-P radial powers are normalized to the two-dimensional PDQ07 assembly powers near the beginning of each cycle.

The SIMULATE-3 model is described in Reference 28.

## 5.2 Fuel Cycle Depletion - Nodal Code

A fuel cycle depletion is performed for each cycle using nodal analysis. For EPRI-NODE-P, the nodal radial powers are normalized to the two-dimensional quarter core PDQ07 or SIMULATE-3 powers at various conditions. SIMULATE-3 does not require normalization. The nodal core model is then depleted from BOC to EOC at appropriate burnup intervals. This depletion is typically performed in the critical boron search mode, with nominal rod insertion (usually 215 SWD) and equilibrium xenon.

Data files may be saved at each burnup step throughout the cycle depletion. These files contain records of the power, exposure, and xenon concentration for each node in the core.

As a result of the nodal core depletion, the following data is obtained:

1. Two and three-dimensional power distributions at each burnup step.
2. A boron letdown curve, i.e., critical boron concentrations as a function of burnup.
3. Axially-dependent parameters such as offset or axial flux difference as a function of burnup.
4. Assembly exposures as a function of core-averaged burnup.

## 5.3 Rod Worth Analysis

Nodal analysis is used to calculate various rod worths which require three-dimensional capabilities. These calculations include differential rod worths and integral rod worths.

### 5.3.1 Differential Rod Worth Analysis

Differential rod worths are calculated as a function of rod insertion. The differential rod worth is defined as the change in reactivity associated with a small change in rod position. This rod worth is determined by running two cases at different rod insertions with all other parameters held constant (power, burnup, xenon, boron) and then by dividing the reactivity difference by bank height difference.

Differential rod worths for the control banks are calculated at HZP and HFP, at BOC and EOC, and at no xenon, equilibrium xenon, and peak xenon conditions. The rod banks are inserted both sequentially and in 50% overlap.

### 5.3.2 Integral Rod Worth Analysis

Integral rod worths are defined as the integral of the differential rod worth data. Integral rod worths are determined by summing up the reactivities resulting from the differential rod worth analysis. Total integral rod worths for a rod bank can be calculated either with a two-dimensional or three-dimensional code by subtracting the reactivities resulting from cases where the rod bank is out and then in (other parameters held constant). However, in order to get the integral rod worth as a function of rod position, i.e., the shape of the rod worth curve, the three-dimensional nodal code is used.

Integral rod worth calculations for the control banks are performed at HZP and HFP, at BOC and EOC, and at no xenon, equilibrium xenon, and peak xenon conditions. The rod banks are inserted sequentially with 50% overlap. The total rod worth (ARI) is calculated at BOC, EOC, and any limiting burnup at HZP for use in the shutdown margin calculation.

## 5.4 Shutdown Margin Analysis

### 5.4.1 Shutdown Margin

Shutdown margin calculations are described in Section 4.2.2.2. Table 4-2 summarizes the results of a shutdown margin calculation.

The calculation consists of:

1. The total rod worth (ARI) at HZP, BOC, and EOC (Item 1 in Table 4-2). This worth is determined by running cases at ARO and ARI (with constant boron and xenon) and subtracting the reactivities.
2. The maximum stuck rod worth at HZP, BOC, and EOC (Item 2 in Table 4-2). Utilizing full-core capabilities, the worth of the worst stuck rod is determined by subtracting the reactivities between two cases, one with ARI, the other with ARI and the stuck rod out.
3. The power deficit from HFP to HZP, at BOC and EOC (Item 6 in Table 4-2). This deficit is determined by running cases at HFP and HZP (with constant boron and xenon) and subtracting the reactivities. This reactivity insertion accounts for Doppler and Moderator deficits, and for axial flux redistribution.
4. The maximum allowable inserted rod worth at HFP, BOC, and EOC (Item 7 in Table 4-2). This worth is obtained by reading the integral rod worth curve at the rod insertion limits (See Section 5.3.2).

Additional reactivity penalties are applied to both the power defect and the rod insertion allowance to account for xenon redistribution effects.

#### 5.4.2 Shutdown Boron Concentration

The shutdown boron concentration is another parameter that is determined using three-dimensional nodal analysis. Since the shutdown margin is determined based on the worst case stuck rod out of the core with all other rods in, the full-core capability of the nodal code is needed.

The nodal code is first used to determine the worst case stuck rod by calculating the worth of various rods in the core. After the worse case stuck rod is determined, a boron search case is performed at the ARI-stuck rod out conditions. This boron concentration is adjusted based on boron worth results until the core reactivity reflects the appropriate margin (1.3%  $\Delta\rho$  for temperatures greater than 200°F, 1.0%  $\Delta\rho$  for temperatures less than or equal to 200°F). The resulting boron concentration is the shutdown boron concentration required for the conditions modeled in the nodal code. This calculated boron concentration is conservatively increased by a boron equivalent of 10% of the ARI-stuck rod out worth and by at least an additional 100 ppm.

A shutdown boron concentration can be determined for any moderator temperature provided the input cross sections remain valid. Typical average moderator temperatures for which shutdown boron concentrations are provided are 68°F, 200°F, 500°F, and the HZP average moderator temperature (approximately 557°F).

#### 5.5 Rod Insertion Limit Assessment

Control rod insertion limits define how deep the control rods may be inserted into the core during normal operation as a function of the power level. It is a Technical Specification requirement that the rods not be inserted deeper than the established limits. This analysis is usually a verification that the Rod Insertion Limits from cycle N-1 are adequate for cycle N.

The control rod insertion limits are determined based on:

1. Maintaining the required minimum shutdown margin, as specified in the Technical Specifications, throughout the cycle life.
2. Maintaining the maximum calculated power peaking factors within the limit specified in the Technical Specifications.
3. The acceptability of the UFSAR Chapter 15 accident analyses.

Determining control rod insertion limits involves an iterative process based on satisfying the above criteria. This process begins with insertion limits from the previous cycle.

The first requirement for insertion limits is that of satisfying the reactivity constraints, i.e., maintaining the required shutdown margin. The insertion limits from the previous cycle, along with integral rod worth curves for control banks in ~50% overlap for the current cycle, are used to calculate the maximum allowable inserted rod worth for input into the shutdown margin calculation. The shutdown margin is calculated at BOC, EOC, and any limiting burnup in order to determine if the control rod insertion limits are acceptable. If the shutdown margin criteria is not satisfied, the insertion limits are adjusted until satisfactory margin is obtained or the core is redesigned.

The insertion limits also have to satisfy the peaking factor constraints. For ARMP methods, the nodal powers are synthesized with discrete pin PDQ07 pin powers to give values of power peaking at various power levels from HZP to HFP. For SIMULATE-3, power peaking factors are calculated directly. The power peaking values are then compared to the Technical Specification limits. If the Technical Specification limits are not satisfied, the control rod insertion limits are adjusted until satisfactory power peaking values are obtained, or the core is redesigned.

In addition to satisfying reactivity and peaking factor constraints during normal operation, the control rod insertion limits may need to be modified based on the worst case consequences of an ejected RCCA, a dropped RCCA or a statically misaligned RCCA. Evaluations are performed with the nodal code to identify the worst case rod configuration during a withdrawal or misalignment event, that is, to identify the single RCCA which produces the maximum peaking factor (control rods held at insertion limits). The results of the three-dimensional nodal analysis with these worst case rod configurations are compared to the design criteria associated with each event. The acceptability of the control rod insertion limits is dependent on the criteria being satisfied.

## 5.6 Trip Reactivity Analysis

The minimum trip reactivity and the shape of the trip reactivity insertion curve (inserted rod worth as a function of rod position) are both generated using nodal analysis. These parameters are needed to perform the safety analysis for various UFSAR Chapter 15 accidents/transients.

### 5.6.1 Minimum Trip Reactivity

The minimum trip reactivity is the minimum amount of reactivity available to be inserted into the core in the event of a reactor trip. It is evaluated for each reload core to ensure that the previously set limits are still valid.

The minimum trip reactivity is calculated at BOC and EOC at HFP and HZP conditions. The minimum trip reactivity is the total rod worth reduced by (1) the most reactive stuck rod worth, (2) a 10% uncertainty on available rod worth, and (3) the rod insertion allowance including applicable penalties to account for xenon redistribution. The rod insertion allowance is the amount of reactivity associated with the control rod insertion limits. It is the difference in reactivity between an ARO case and one with control rods at their insertion limits. A sample BOC calculation is shown in Table 5-1.

### 5.6.2 Trip Reactivity Shape

The shape of the trip reactivity insertion curve defines the inserted rod worth as a function of rod position. The most limiting shape is the one which defines the minimum inserted rod worth as a function of rod position. This most limiting shape is evaluated each reload cycle to ensure that the values for the minimum inserted rod worth vs. rod position used in the safety analysis are still applicable.

The most limiting trip reactivity shape typically corresponds to the most bottom-skewed axial power shape. HFP axial power distributions are examined from BOC to EOC, with control rods at the full power rod insertion limits and the most reactive rod stuck out of the core. After the most limiting power shape is found, the N-1 control rods are inserted into the core in a stepwise manner. The results of this insertion yield the minimum inserted rod worth vs. position curve.

### 5.7 Assessment of Nodal Analyses

Once the nodal calculations are performed, the resultant data are assessed for validity and consistency with core operation requirements and safety limits.

TABLE 5-1

Example BOC Trip Reactivity Calculation

<u>Trip Reactivity</u>	<u>HFP % <math>\Delta\rho</math></u>
Minimum Available N-1 Rod Worth	6.18
10% Rod Worth Uncertainty	<u>-0.62</u>
Total Available Rod Worth	5.56
Rod Insertion Allowance	-1.13
Xenon Redistribution Penalty	<u>-0.08</u>
Minimum Trip Reactivity	4.35

## 6. CALCULATION OF SAFETY RELATED PHYSICS PARAMETERS

### 6.1 Safety Analysis Physics Parameters

With a reload of fresh fuel, a reactor core's physics characteristics are altered in three major areas:

1. Power distribution
2. Control rod worths
3. Kinetics

Each of the above has its own subset of specific parameters. These core physics parameters are considered in the UFSAR Chapter 15 safety analyses. The core physics parameters whose values have an important influence on the course or the consequences of the accidents analyzed in the UFSAR are designated as safety analysis physics parameters. Reference 30 identifies these parameters, describes the approach for calculating the values of these parameters, and discusses the manner in which the reload values are evaluated for acceptability with respect to the accident analysis assumptions.

### 6.2 Core Power Distributions

As part of the reload design, detailed analyses of the core power distributions are performed for core conditions of normal operation and anticipated transient conditions. These analyses are performed:

1. to confirm that the power peaking factors assumed as initial conditions for certain accidents remain valid,
2. to verify that certain transient induced power peaks will be acceptable for the fuel design thermal limits, and
3. to facilitate the selection of the operating limits and protection system setpoints.

The methodology for performing these analyses is presented in Reference 29.

### 6.3 Power Peaking Factors and Reliability Factors

Power peaking factors to be compared to a design limit are conservatively increased by total peaking reliability (or uncertainty) factors. The peaking reliability factor is determined by statistically combining manufacturing and calculation uncertainties. Additional potential power peaking uncertainties may be included for things such as rod bowing, etc.

The general formulation for the peaking reliability factor is:

$$\text{Peaking Reliability Factor} = 1 + \text{BIAS} + \sqrt{(\text{UC}^2 + \text{Ux1}^2 + \text{Ux2}^2 + \dots)} \quad (6-1)$$

Where:

UC - Calculation Uncertainty

For the Pin Total Peak ( $F_Q$ ):  $\text{UC}^2 = \text{UT}^2 + \text{URL}^2$

For the Pin Radial Peak ( $F_{\Delta H}$ ):  $\text{UC}^2 = \text{UR}^2 + \text{URL}^2$

For the Assembly Axial Peak ( $F_Z$ ):  $\text{UC}^2 = \text{UA}^2$

UT - Total Peaking Uncertainty

URL - Assembly Radial Local (or Pin) Power Peaking Uncertainty

UR - Assembly Radial Power Peaking Uncertainty

UA - Assembly Axial Power Peaking Uncertainty

Uxi - Additional Uncertainties, e.g. manufacturing tolerance, rod bow, etc.

BIAS - Calculation Bias

For the PDQ07 code methodology, Section 8 contains the calculation of the radial local uncertainty factor. For the EPRI-NODE-P code based on ARMP methodology, Section 11 contains the calculation of the assembly and pin total peak and assembly and pin radial peak uncertainty factors. When the EPRI-NODE-P code is based on the CASMO-3/SIMULATE-3 methodology, Reference 28 presents the uncertainty factors. For the CASMO-3/SIMULATE-3 code methodology, Reference 28 contains the calculation of the uncertainty factors.

## 7. 3D POWER PEAKING ANALYSIS

As part of the reload design, detailed analysis of the core power distribution for normal operation and anticipated transient conditions are made. These analyses are performed (1) to confirm that the initial condition power peaking factors for certain accidents remain valid, (2) to verify that certain transient induced power peaks will satisfy the fuel design limits, and (3) to facilitate the selection of operating limits and RPS setpoints. The methods for performing these analyses are outlined in Reference 29.

## 8. RADIAL LOCAL ANALYSIS

### 8.1 Background

The radial local is an important factor in fuel cycle design because of its significant influence on LOCA and DNB analysis. The premise for performing this analysis is to evaluate the ability of PDQ07 to predict the radial local. The radial local is defined as the ratio of the maximum pin power, to the assembly average planar (x-y) power. It is used to calculate pin power by combining assembly power ( $F_Q$  or  $F_{\Delta H}$ ) from the nodal analysis with the radial local factor by the equation shown below.

$$\text{Pin Power} = \text{Nodal Power} \times \text{Radial Local Factor}$$

In the ARMP methodology, PDQ07 and CASMO-2 may be used to calculate radial local factors. PDQ07 is a 1, 2, or 3 dimensional two neutron energy group diffusion theory code, whereas CASMO-2 is a 2-dimensional multigroup transport theory code, which utilizes transport probabilities in the solution of the transport equation. The 2-dimensional PDQ07 code is the primary calculational tool used to model reactor cores (for additional information concerning the use of this code, refer to Section 3.4). Energy and burnup dependent Mixed Number Density (MND) cross sections used by PDQ07 are developed in accordance with ARMP<sup>14</sup> procedures. CASMO-2 is used primarily to generate multigroup constants (i.e., control rod and burnable absorber cross sections), and as a benchmark code.

SIMULATE-3 is capable of calculating pin power peaking directly and does not require the use of radial local factors (Reference 28).

### 8.2 Comparison of PDQ07 to CASMO-2 at Hot Full Power Condition

The predictive capability of PDQ07 was assessed by performing a series of eighth assembly calculations using both PDQ07 and CASMO-2. A typical Westinghouse 17x17, 3.2 w/o Uranium-235 optimized fuel assembly was modeled

using these codes.

All simulations were performed at beginning of life (BOL), hot full power (HFP), no xenon conditions, for at this time severe pin power peaking is most prominent. Simulations were performed for a variety of burnable absorber loadings and soluble boron concentrations. Table 8-1 contains a summary of the cases that were investigated.

Figures 8-1 through 8-10 contain 1/8 assembly pinwise power comparisons between PDQ07 and CASMO-2. Results from these comparisons indicate that PDQ07 conservatively overpredicts the maximum CASMO-2 pin power. This overprediction ranges from 0.86% to 2.26%. PDQ07 also correctly identifies the location of the CASMO-2 maximum pin power. Comparisons between PDQ07 and CASMO-2 maximum pin powers for each case are tabulated in Table 8-2.

The predictive capability of PDQ07 was assured by performing a statistical analysis over all pins in the problem and for pins with powers greater than or equal to 1.000. The average and average absolute differences and respective standard deviations are presented in Table 8-3 for all cases investigated.

### 8.3 Comparisons of PDQ07 to Cold Criticals

The ability of PDQ07 to predict pin powers at cold conditions was assessed by performing a series of simulations based on the B&W uranium criticals. In all simulations, PDQ07 conservatively and accurately predicted the maximum pin power. For additional specifics concerning the comparisons of PDQ07 to the B&W uranium criticals, refer to Reference 3.

### 8.4 BNL Benchmark Assembly Problem

The BNL evaluation of the Duke solution to the BNL benchmark assembly problem determined that PDQ07 methods overpredict the peak pin power by just over 1% at BOC and underpredict the peak pin power by approximately 1% at 40,000 MWD/MTU with the cross over occurring at approximately 15,000 MWD/MTU.

## 8.5 Conclusion

Comparisons between PDQ07 and CASMO-2 at HFP conditions indicate that PDQ07 conservatively predicts maximum pin powers. PDQ07 comparisons to B&W cold criticals also indicate that PDQ07 conservatively predicts maximum pin powers. However, the solution to the BNL benchmark problem shows an underprediction at high burnups. Therefore, an uncertainty of 2% is applied to the predicted pin peaking factors (see Appendix B).

TABLE 8-1

Characteristics of 1/8th Assembly Simulations

<u>CASE</u>	<u>ENRICHMENT</u> <u>W/O U-235</u>	<u>BURNABLE ABSORBER</u> <u>LOADING</u>	<u>BORON CONCENTRATION</u> <u>(PPMB)</u>
1	3.2	0	0
2	3.2	0	950
3	3.2	4	0
4	3.2	4	950
5	3.2	12	0
6	3.2	12	950
7	3.2	16	0
8	3.2	16	950
9	3.2	20	0
10	3.2	20	950

TABLE 8-2

Peak Pin Power Comparison

<u>CASE</u>	<u>PDQ07</u> <u>PEAK PIN POWER</u>	<u>CASMO</u> <u>PEAK PIN POWER</u>	<u>DIFFERENCE</u> <u>PDQ07-CASMO</u>	<u>% DIFFERENCE</u> <u>(P-C)/C</u>
1	1.053	1.042	0.011	1.056
2	1.051	1.039	0.012	1.155
3	1.055	1.046	0.009	0.860
4	1.053	1.043	0.010	0.959
5	1.152	1.131	0.021	1.857
6	1.137	1.119	0.018	1.609
7	1.188	1.163	0.025	2.150
8	1.170	1.149	0.021	1.828
9	1.178	1.152	0.026	2.257
10	1.164	1.140	0.024	2.105

TABLE 8-3

Statistical Summary of Percent Differences between PDQ07 and CASMO-2  
for Pins in Assemblies with Powers Greater Than or Equal to 1.000

<u>CASE</u>	<u><math>\bar{D}</math> *</u>	<u>ABS (D)</u>	<u>STANDARD DEVIATION (D)</u>	<u>S.D. [ABS (D)]</u>
1	0.4450	0.5566	0.5524	0.4339
2	0.4657	0.5554	0.5627	0.4697
3	0.2151	0.3470	0.4098	0.3010
4	0.2109	0.3595	0.4215	0.2987
5	0.8916	1.0620	0.8705	0.6396
6	0.7936	0.9548	0.7733	0.5503
7	0.9321	1.1509	1.0832	0.8311
8	0.8057	1.0241	0.9635	0.7107
9	0.7130	0.8202	0.8885	0.7851
10	0.6458	0.7530	0.8109	0.7069

Statistical Summary of Percent Differences between PDQ07 and CASMO-2  
For All Pins Within An Assembly

<u>CASE</u>	<u><math>\bar{D}</math> *</u>	<u>ABS (D)</u>	<u>STANDARD DEVIATION (D)</u>	<u>S.D. [ABS (D)]</u>
1	0.0030	0.6395	0.7867	0.4463
2	0.0066	0.6572	0.8119	0.4648
3	-0.0255	0.4606	0.6328	0.4281
4	-0.0066	0.4520	0.6280	0.4298
5	0.0616	1.0682	1.2511	0.6310
6	0.0394	0.9801	1.1449	0.5713
7	0.0585	0.9416	1.2120	0.7499
8	0.0398	0.8436	1.0926	0.6819
9	0.0268	0.8776	1.1696	0.7604
10	0.0293	0.8059	1.0732	0.6972

\* NOTE:  $D = [(PDQ07 - CASMO-2) / CASMO-2] * 100$

$$\bar{D} = \sum_{i=1}^N D_i / N$$

FIGURE 8-1

CASMO-2 AND PDQ-7  
 ROD POWER COMPARISON  
 BOL HFP NO XENON  
 3.2 w/o U-235 OPT 17x17 FA  
 CASE NUMBER 1

0.0									
0.0									
1.020	1.002								
1.024	1.009								
1.021	1.002	1.002							
1.024	1.009	1.010							
0.0	1.024	1.026	0.0						
0.0	1.025	1.027	0.0						
1.020	1.002	1.004	1.033	1.023					
1.023	1.009	1.013	1.037	1.045					
1.018	1.000	1.002	1.033	*	0.0	CASMO-2			
1.021	1.007	1.011	1.038	1.053	0.0	PDQ-7			
0.0	1.018	1.020	0.0	1.030	1.014	0.976			
0.0	1.017	1.019	0.0	1.037	1.010	0.975			
1.008	0.990	0.990	1.011	0.986	0.970	0.959	0.955		
1.006	0.993	0.994	1.006	0.989	0.964	0.947	0.940		
0.986	0.985	0.985	0.985	0.979	0.972	0.966	0.967	0.981	
0.982	0.979	0.979	0.980	0.973	0.961	0.953	0.954	0.971	

	PDQ-7	CASMO-2
PPMB	950	950
NUMBER BA	0	0
K-INFINITY	1.3486	1.3479
*MAX ROD POWER	1.053	1.042

FIGURE 8-2

CASMO-2 AND PDQ-7  
 ROD POWER COMPARISON  
 BOL HFP NO XENON  
 3.2 w/o U-235 OPT 17x17 FA  
 CASE NUMBER 2

0.0									
0.0									
1.017	1.000								
1.021	1.007								
1.018	1.000	1.000							
1.021	1.007	1.008							
0.0	1.021	1.023	0.0						
0.0	1.022	1.024	0.0						
1.018	1.000	1.002	1.030	1.020					
1.021	1.007	1.011	1.035	1.042					
1.016	0.999	1.001	1.030	*	0.0	CASMO-2			
1.020	1.006	1.010	1.035	1.051	0.0	PDQ-7			
0.0	1.017	1.019	0.0	1.029	1.014	0.979			
0.0	1.016	1.018	0.0	1.036	1.011	0.977			
1.007	0.990	0.990	1.011	0.988	0.973	0.963	0.960		
1.006	0.993	0.994	1.007	0.991	0.966	0.950	0.945		
0.987	0.987	0.986	0.987	0.982	0.975	0.971	0.972	0.986	
0.983	0.980	0.980	0.982	0.975	0.964	0.957	0.959	0.976	

	PDQ-7	CASMO-2
PPMB	950	950
NUMBER BA	0	0
K-INFINITY	1.2122	1.2077
*MAX ROD POWER	1.051	1.039

FIGURE 8-3

CASMO-2 AND PDQ-7  
 ROD POWER COMPARISON  
 BOL HFP NO XENON  
 3.2 w/o U-235 OPT 17x17 FA  
 CASE NUMBER 3

0.0									
0.0									
1.011	0.988								
1.013	0.991								
1.006	0.975	0.933							
1.003	0.972	0.915							
0.0	0.989	0.893	0.0						
0.0	0.977	0.890	0.0						
1.011	0.980	0.940	0.905	0.964					
1.009	0.979	0.925	0.908	0.963					
1.022	0.998	0.989	1.011	1.032	0.0	CASMO-2			
1.027	1.006	0.992	1.008	1.040	0.0	PDQ-7			
0.0	1.034	1.033	0.0	*	1.036	1.005			
0.0	1.036	1.034	0.0	1.055	1.039	1.011			
1.033	1.014	1.015	1.037	1.014	1.000	0.992	0.990		
1.037	1.024	1.023	1.036	1.022	1.000	0.987	0.983		
1.015	1.015	1.015	1.016	1.011	1.005	1.002	1.004	1.019	
1.018	1.015	1.014	1.016	1.010	1.001	0.996	0.999	1.018	

	PDQ-7	CASMO-2
PPMB	0	0
NUMBER BA	4	4
K-INFINITY	1.2978	1.2956
*MAX ROD POWER	1.055	1.046

FIGURE 8-4

CASMO-2 AND PDQ-7  
 ROD POWER COMPARISON  
 BOL HFP NO XENON  
 3.2 w/o U-235 OPT 17x17 FA  
 CASE NUMBER 4

0.0									
0.0									
1.012	0.990								
1.014	0.993								
1.007	0.977	0.937							
1.004	0.975	0.919							
0.0	0.991	0.898	0.0						
0.0	0.980	0.896	0.0						
1.011	0.982	0.943	0.909	0.965					
1.010	0.981	0.928	0.912	0.965					
1.021	0.998	0.989	1.010	1.030	0.0	CASMO-2			
1.026	1.006	0.993	1.009	1.039	0.0	PDQ-7			
0.0	1.031	1.031	0.0	*	1.043	1.034	1.005		
0.0	1.035	1.032	0.0	1.053	1.037	1.010			
1.031	1.013	1.013	1.035	1.013	1.000	0.992	0.991		
1.035	1.022	1.021	1.034	1.020	0.999	0.987	0.983		
1.013	1.013	1.013	1.015	1.010	1.005	1.002	1.005	1.020	
1.016	1.013	1.013	1.015	1.009	1.001	0.996	1.000	1.018	

	PDQ-7	CASMO-2
PPMB	950	950
NUMBER BA	4	4
K-INFINITY	1.1757	1.1672
*MAX ROD POWER	1.053	1.043

FIGURE 8-5

CASMO-2 AND PDQ-7  
 ROD POWER COMPARISON  
 BOL HFP NO XENON  
 3.2 w/o U-235 OPT 17x17 FA  
 CASE NUMBER 5

0.0									
0.0									
*									
1.131	1.108								
1.152	1.132								
1.123	1.099	1.088							
1.144	1.123	1.109							
0.0	1.109	1.093	0.0						
0.0	1.123	1.102	0.0						
1.093	1.063	1.036	1.019	0.946					
1.109	1.081	1.045	1.014	0.937					
1.076	1.038	0.980	0.905	0.864	0.0	CASMO-2			
1.085	1.047	0.970	0.906	0.855	0.0	PDQ-7			
0.0	1.043	0.930	0.0	0.860	0.876	0.932			
0.0	1.041	0.934	0.0	0.849	0.867	0.908			
1.055	1.020	0.967	0.907	0.930	0.950	0.973	0.996		
1.062	1.028	0.956	0.906	0.910	0.934	0.959	0.989		
1.035	0.026	1.004	0.984	0.983	0.992	1.007	1.024	1.048	
1.046	1.032	1.000	0.976	0.973	0.984	1.000	1.023	1.054	

	PDQ-7	CASMO-2
PPMB	0	0
NUMBER BA	12	12
K-INFINITY	1.2073	1.2029
*MAX ROD POWER	1.152	1.131



FIGURE 8-7

CASMO-2 AND PDQ-7  
 ROD POWER COMPARISON  
 BOL HFP NO XENON  
 3.2 w/o U-235 OPT 17x17 FA  
 CASE NUMBER 7

0.0									
0.0									
*									
1.163	1.138								
1.188	1.166								
1.150	1.125	1.111							
1.173	1.152	1.137							
0.0	1.123	1.110	0.0						
0.0	1.138	1.124	0.0						
1.072	1.051	1.040	1.035	0.966					
1.080	1.065	1.052	1.034	0.962					
0.950	0.975	0.966	0.915	0.885	0.0	CASMO-2			
0.952	0.962	0.953	0.918	0.879	0.0	PDQ-7			
0.0	0.909	0.904	0.0	0.882	0.905	0.966			
0.0	0.906	0.902	0.0	0.874	0.900	0.946			
0.922	0.949	0.948	0.916	0.955	0.983	1.010	1.036		
0.919	0.934	0.933	0.918	0.937	0.971	1.001	1.035		
0.992	0.993	0.995	0.996	1.009	1.027	1.046	1.066	1.093	
0.989	0.993	0.993	0.993	1.004	1.023	1.046	1.071	1.105	

	PDQ-7	CASMO-2
PPMB	0	0
NUMBER BA	16	16
K-INFINITY	1.1629	1.1576
*MAX ROD POWER	1.188	1.163

FIGURE 8-8

CASMO-2 AND PDQ-7  
 ROD POWER COMPARISON  
 BOL HFP NO XENON  
 3.2 w/o U-235 OPT 17x17 FA  
 CASE NUMBER 8

0.0									
0.0									
*									
1.149	1.126								
1.170	1.150								
1.138	1.114	1.102							
1.157	1.137	1.125							
0.0	1.113	1.102	0.0						
0.0	1.127	1.115	0.0						
1.067	1.047	1.037	1.033	0.968					
1.074	1.059	1.048	1.033	0.965					
0.952	0.976	0.968	0.919	0.890	0.0	CASMO-2			
0.953	0.963	0.955	0.923	0.886	0.0	PDQ-7			
0.0	0.913	0.909	0.0	0.888	0.910	0.969			
0.0	0.911	0.907	0.0	0.881	0.906	0.950			
0.927	0.953	0.953	0.921	0.959	0.986	1.011	1.035		
0.925	0.939	0.938	0.923	0.943	0.974	1.002	1.033		
0.995	0.996	0.997	0.999	1.011	1.027	1.045	1.064	1.088	
0.993	0.996	0.997	0.996	1.006	1.024	1.044	1.068	1.099	

	PDQ-7	CASMO-2
PPMB	950	950
NUMBER BA	16	16
K-INFINITY	1.0655	1.0581
*MAX ROD POWER	1.170	1.149

FIGURE 8-9

CASMO-2 AND PDQ-7  
 ROD POWER COMPARISON  
 BOL HFP NO XENON  
 3.2 w/o U-235 OPT 17x17 FA  
 CASE NUMBER 9

0.0									
0.0									
*									
1.152	1.121								
1.178	1.148								
1.132	1.094	1.036							
1.151	1.112	1.034							
0.0	1.084	0.967	0.0						
0.0	1.087	0.977	0.0						
1.061	1.026	0.971	0.899	0.903					
1.064	1.033	0.958	0.898	0.877					
0.952	0.972	0.951	0.891	0.873	0.0	CASMO-2			
0.956	0.959	0.931	0.887	0.865	0.0	PDQ-7			
0.0	0.923	0.916	0.0	0.898	0.929	0.998			
0.0	0.924	0.915	0.0	0.891	0.928	0.984			
0.948	0.976	0.975	0.943	0.985	1.017	1.048	1.077		
0.950	0.964	0.962	0.948	0.971	1.010	1.046	1.084		
1.025	1.027	1.029	1.031	1.045	1.065	1.087	1.109	1.136	
1.027	1.030	1.031	1.032	1.046	1.069	1.095	1.124	1.160	

	PDQ-7	CASMO-2
PPMB	0	0
NUMBER BA	20	20
K-INFINITY	1.1206	1.1148
*MAX ROD POWER	1.178	1.152

FIGURE 8-10

CASMO-2 AND PDQ-7  
 ROD POWER COMPARISON  
 BOL HFP NO XENON  
 3.2 w/o U-235 OPT 17x17 FA  
 CASE NUMBER 10

0.0									
0.0									
*									
1.140	1.112								
1.164	1.136								
1.123	1.087	1.032							
1.140	1.103	1.029							
0.0	1.078	0.967	0.0						
0.0	1.082	0.977	0.0						
1.058	1.025	0.973	0.905	0.909					
1.062	1.031	0.960	0.904	0.885					
0.954	0.974	0.954	0.897	0.880	0.0	CASMO-2			
0.958	0.962	0.936	0.894	0.873	0.0	PDQ-7			
0.0	0.927	0.921	0.0	0.903	0.932	0.998			
0.0	0.928	0.920	0.0	0.898	0.932	0.984			
0.951	0.978	0.978	0.946	0.986	1.016	1.045	1.072		
0.953	0.967	0.966	0.951	0.973	1.010	1.043	1.077		
1.025	1.027	1.029	1.031	1.044	1.062	1.082	1.102	1.127	
1.028	1.031	1.031	1.031	1.044	1.065	1.089	1.115	1.149	

	PDQ-7	CASMO-2
PPMB	950	950
NUMBER BA	20	20
K-INFINITY	1.0315	1.0238
*MAX ROD POWER	1.164	1.140

## 9. DEVELOPMENT OF CORE PHYSICS PARAMETERS

Upon completion of the Final Fuel Cycle Design, physics parameters such as boron concentrations and worths, power distributions, etc. are calculated 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. The results of these calculations are used for startup test predictions and core physics parameters throughout the cycle.

### 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 and to verify reactor behavior is as predicted by the nuclear simulators which were used in generating the data used in the plant's safety analysis.

#### 9.1.1 Critical Boron Concentrations and Boron Worths

Critical boron concentrations and boron worths at a variety of rod configurations, at HZP and HFP, as a function boron concentration, at different xenon concentrations, and at different times in the fuel cycle are calculated. EPRI-NODE-P and SIMULATE-3 are capable of performing critical boron search calculations. The method used for PDQ07 is to correct the input boron concentration to the critical boron concentration using a calculated boron worth and the calculated reactivity.

Table 9-1 shows some of the critical boron calculations normally performed for startup physics tests. These calculations are performed after the sequential insertion of each control or shutdown bank and are sometimes referred to as boron endpoints.

Critical boron concentrations at HZP and HFP with all rods out are also calculated as a function of cycle burnup. An example of how boron changes

with burnup is shown in Figure 9-1. These curves are referred to as boron letdown curves.

The boron worths are usually calculated by running two identical cases except that the soluble boron concentration is different. The differential boron worth is calculated by subtracting the reactivities and dividing by the boron difference. Differential boron worths are usually quoted in PCM/PPMB. The inverse boron worth is the inverse of the differential boron worth and is usually quoted in PPMB/% $\Delta\rho$ .

Table 9-2 shows the soluble boron worths usually performed for startup physics tests. Similar to critical boron concentrations, these worths are calculated with sequential bank insertions.

Differential boron worth (or inverse boron worth) can also be calculated as a function of boron concentration and as a function of cycle burnup. Figures 9-2 and 9-3 show the results of a typical differential boron worth calculation vs. boron concentration and vs. burnup, respectively.

#### 9.1.2 Xenon Worth and Defect

Xenon worth is calculated as a function of cycle burnup. The nominal HFP depletion cases with equilibrium xenon are used as input to a second set of cases where the xenon concentration is set to zero (or the xenon cross sections are set to zero). The difference in reactivities between the equilibrium xenon and no xenon cases equals the equilibrium xenon worth at HFP. Figure 9-4 shows the results of a typical equilibrium xenon worth calculation.

Xenon worth can also be presented as a function of power level. Worths presented in this manner are usually referred to as the equilibrium xenon reactivity defect and are quoted in either pcm or % $\Delta\rho$ . Figure 9-5 shows the results of a typical xenon defect calculation.

### 9.1.3 Rod Worths

#### 9.1.3.1 Bank Worths

The worth of the shutdown and control banks are calculated at BOC HZP for use in the zero power physics testing. The rod banks are sequentially inserted or withdrawn assuming no control rod overlap. The bank worth is the difference in reactivity between the fully inserted case and the fully withdrawn case.

Integral rod worth curves are calculated at BOC HZP for Control Banks B, C and D. The rod banks are inserted both sequentially and with 50% overlap. Figure 9-6 shows the results of a typical integral rod worth calculation.

Control bank worths with sequential insertion and integral rod worth curves with 50% overlap are calculated at HFP equilibrium xenon both at BOC and EOC.

#### 9.1.3.2 Stuck Rod Worth

The maximum worth of a single control rod stuck out of the reactor core at HZP is calculated. The worth of the stuck rod is used by the site engineers in the reactivity balance procedures to guarantee shutdown margin. If the stuck rod worth is to be measured during the startup test program, then a recalculation of the worth is performed simulating the test conditions. This worth would then be provided as a startup test prediction.

#### 9.1.3.3 Dropped Rod Worth

The maximum worth of a single control rod dropped into the reactor core is calculated. If this parameter is to be measured during the startup test program, then a recalculation of the worth is performed simulating the test conditions. This worth would then be provided as a startup test prediction.

#### 9.1.3.4 Ejected Rod Worth

The maximum ejected control rod worth is calculated. If this parameter is to be measured during the startup test program, then a recalculation of the worth is performed simulating the test conditions. This worth would then be provided as a startup test prediction.

#### 9.1.4 Reactivity Coefficients

##### 9.1.4.1 HZP Coefficients

At HZP the isothermal temperature coefficient is measured by varying the average moderator temperature and determining the corresponding reactivity change. The calculations for predicting the isothermal temperature coefficient should be performed by changing the average moderator temperature in the core model. The resulting reactivity change is then divided by the temperature change to yield the HZP isothermal temperature coefficient.

The Doppler or fuel temperature coefficient at HZP can be calculated by varying the fuel temperature while maintaining the no load moderator temperature. The resulting reactivity change divided by the change in fuel temperature is the Doppler coefficient at HZP.

The predicted moderator coefficient is calculated by subtracting the Doppler coefficient from the isothermal coefficient. It is compared to the (inferred) measured moderator coefficient obtained by subtracting the predicted Doppler coefficient from the measured isothermal coefficient.

Alternately, the moderator temperature coefficient can also be explicitly calculated.

#### 9.1.4.2 HFP Coefficients

Both a temperature coefficient of reactivity and a power Doppler coefficient of reactivity are calculated at HFP. The temperature coefficient is calculated by running one equilibrium HFP case at BOC (4 EFPD) and a second case where the moderator temperature is changed. The difference in reactivity divided by the temperature change is the temperature coefficient. To calculate the power Doppler coefficient, a third case is performed where the power level is reduced. All other parameters are kept at the HFP equilibrium values. The difference in reactivity between the HFP and the reduced power cases divided by change in power is the power Doppler coefficient.

#### 9.1.5 Power Distribution

Power distributions, both assembly radial and total peaking factors, are measured at various power levels as identified in the test procedures for McGuire/Catawba reload startups. Calculations are performed at these power levels and nominal conditions to provide predicted power distributions for comparison.

#### 9.1.6 Kinetics Parameters

Kinetics parameters are calculated using the methodology and codes as discussed in Section 4.2.4.8. These parameters include the six group  $\beta_i$  effective and  $\lambda_i$ , total  $\beta$  effective and  $\ell^*$ . These kinetics parameters are generated for both BOC HZP and BOC HFP conditions with ARO. A second set of delayed neutron data may be generated at EOC.

## 9.2 Startup and Operation Report

The purpose of the Startup and Operation Report is to document the predicted behavior of the reactor core as a function of burnup and power level. It is intended to be used for operator guidance and to aid the site engineer.

This report will include sufficient information to calculate reactivity balance throughout the cycle. Table 9-3 lists items typical of what will be calculated for this report.

TABLE 9-1

Critical Boron Concentrations (ppmB)

HZP, NOXE, 0 EFPD

ARO

Bank D in

Banks D + C in

Banks D + C + B in

Banks D + C + B + A in

Banks D + C + B + A + S<sub>E</sub> in

Banks D + C + B + A + S<sub>E</sub> + S<sub>D</sub> in

Banks D + C + B + A + S<sub>E</sub> + S<sub>D</sub> + S<sub>C</sub> in

Banks D + C + B + A + S<sub>E</sub> + S<sub>D</sub> + S<sub>C</sub> + S<sub>B</sub> in

Banks D + C + B + A + S<sub>E</sub> + S<sub>D</sub> + S<sub>C</sub> + S<sub>B</sub> + S<sub>A</sub> in

HFP, NOXE, 0 EFPD

ARO

HFP, EQXE, 4 EFPD

ARO

Bank D in

HFP, EQXE, EOC

ARO

TABLE 9-2

Boron Worth (pcm/ppmB)

HZP, NOXE, 0 EFPD

ARO

Bank D in\*

Banks D + C in

Banks D + C + B in

Banks D + C + B + A in

ARI

HFP, EQXE, 4 EFPD

ARO

HFP, EQXE, EOC

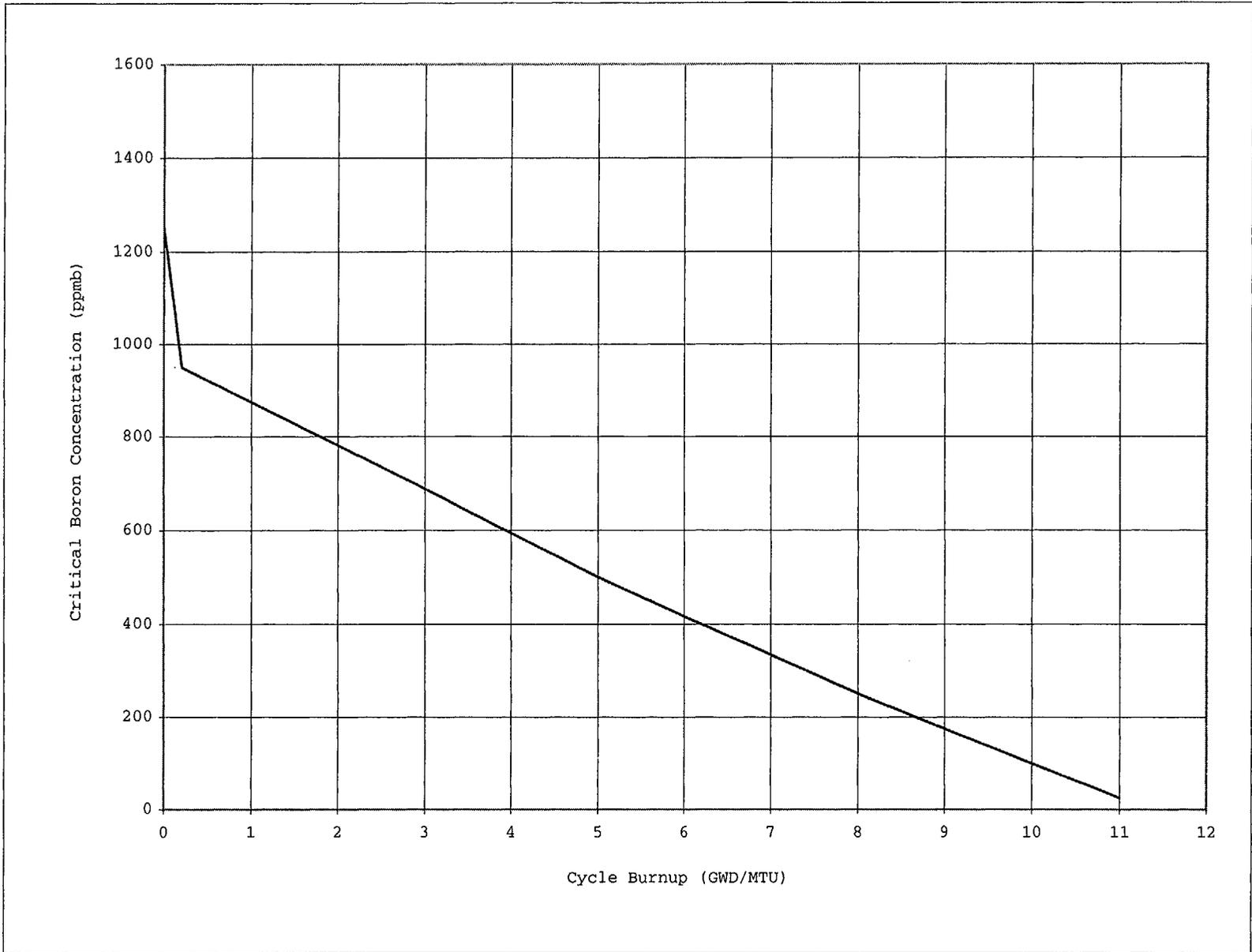
ARO

\* Note: When bank worths are determined using interchange (swap) with the reference control bank, the boron worth with the reference bank only inserted is evaluated in place of sequential insertions.

TABLE 9-3

Typical Core Physics Data

- A. Critical Boron Concentrations
  - 1. ARO HFP Versus Burnup
  - 2. ARO HZP Versus Burnup
  
- B. Shutdown Boron Concentrations Required for Shutdown with Highest Worth Rod Stuck Out (NoXe)
  - 1. HZP Versus Burnup
  - 2. 500°F, 200°F and 68°F Versus Burnup
  
- C. Differential Boron Worth HFP, HZP Versus Burnup
  
- D. Power Distributions from the Cycle Depletion
  
- E. Rod Worths BOC, EOC, HFP and HZP
  
- F. Xenon Worth Versus Power Level
  
- G. Xenon Worth Versus Burnup
  
- H. Reactivity Coefficients Versus Temperature, Power Level and Burnup



Boron Letdown Curve  
HFP, ARO

FIGURE 9-1

FIGURE 9-2

Differential Boron Worth  
HZP, BOL

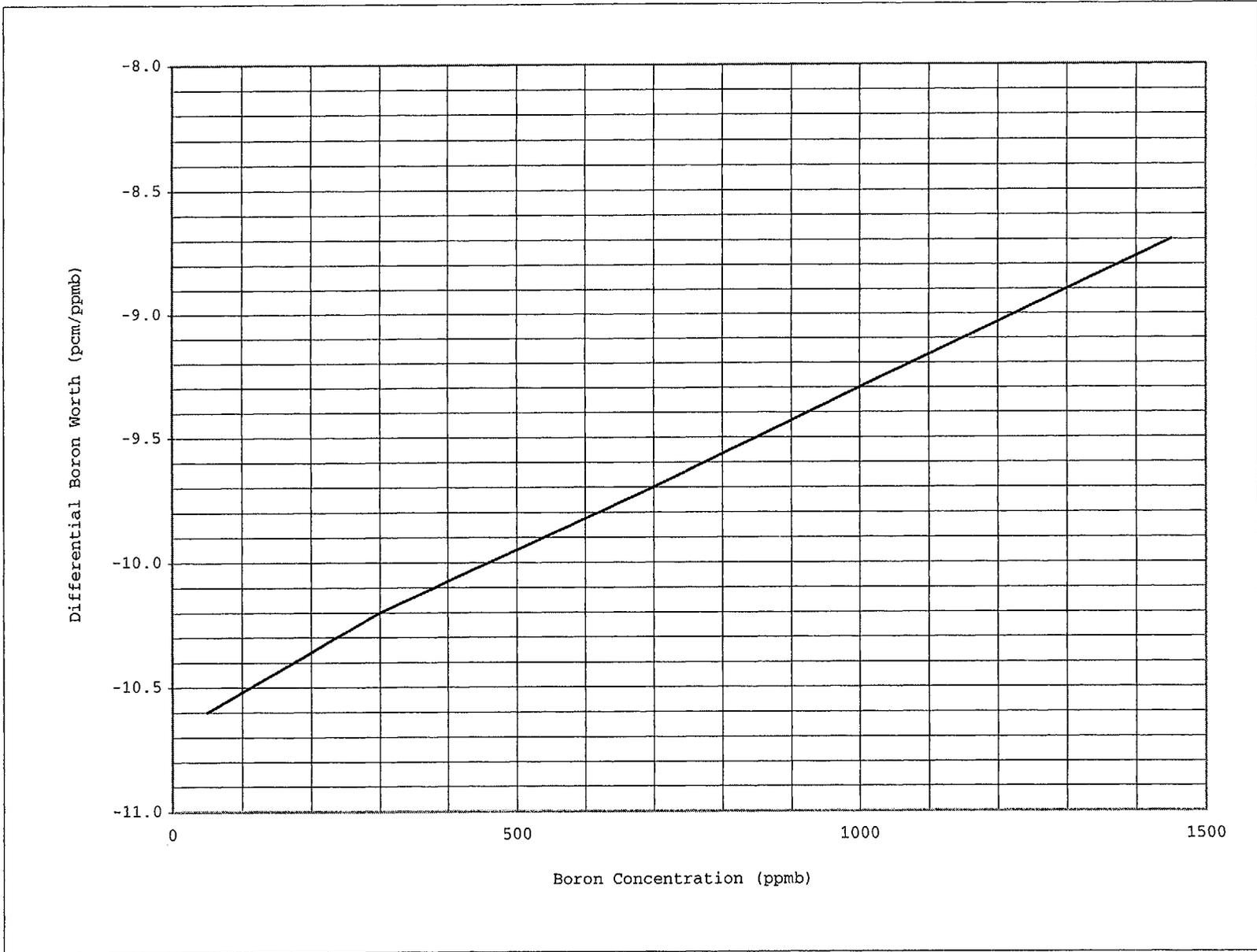
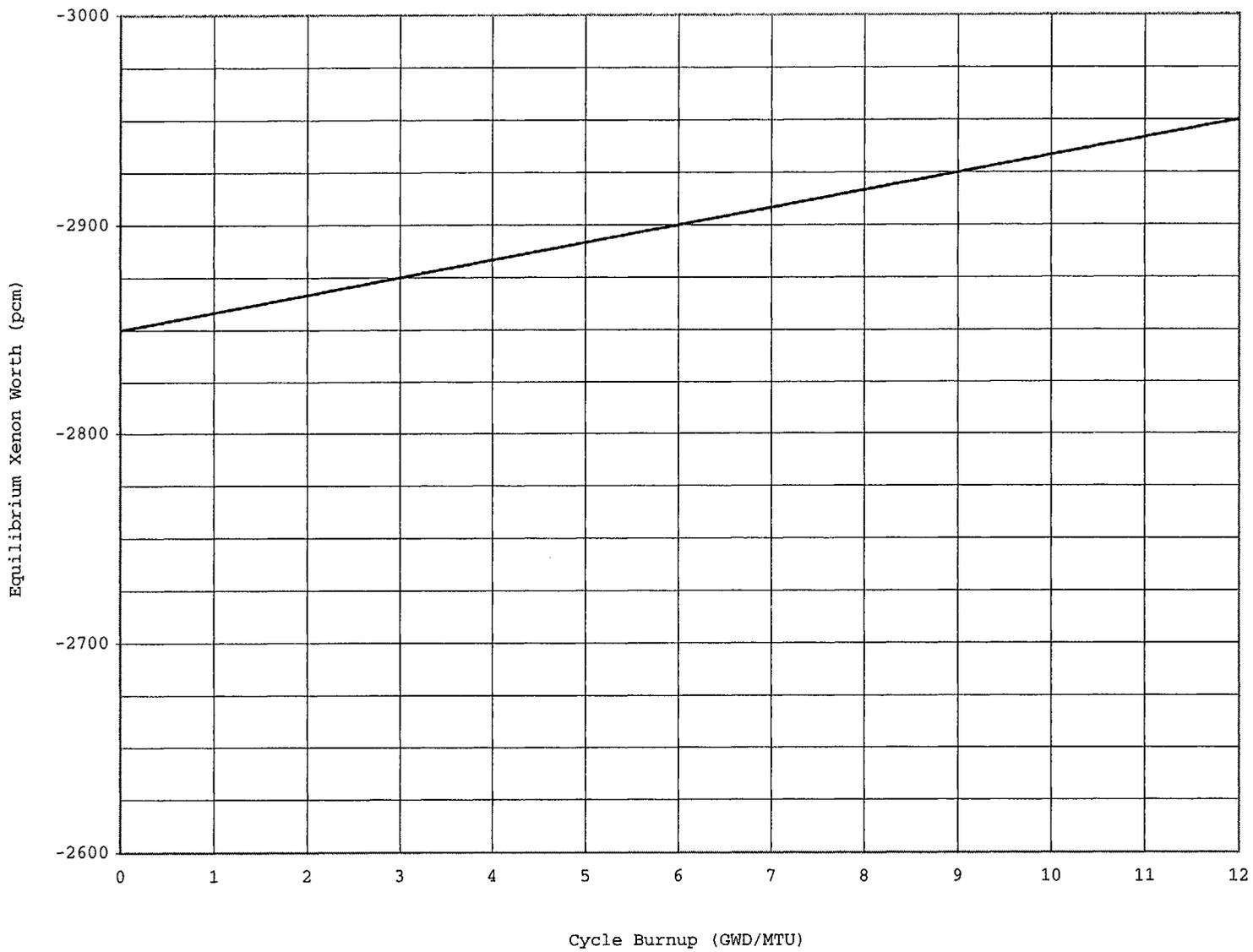
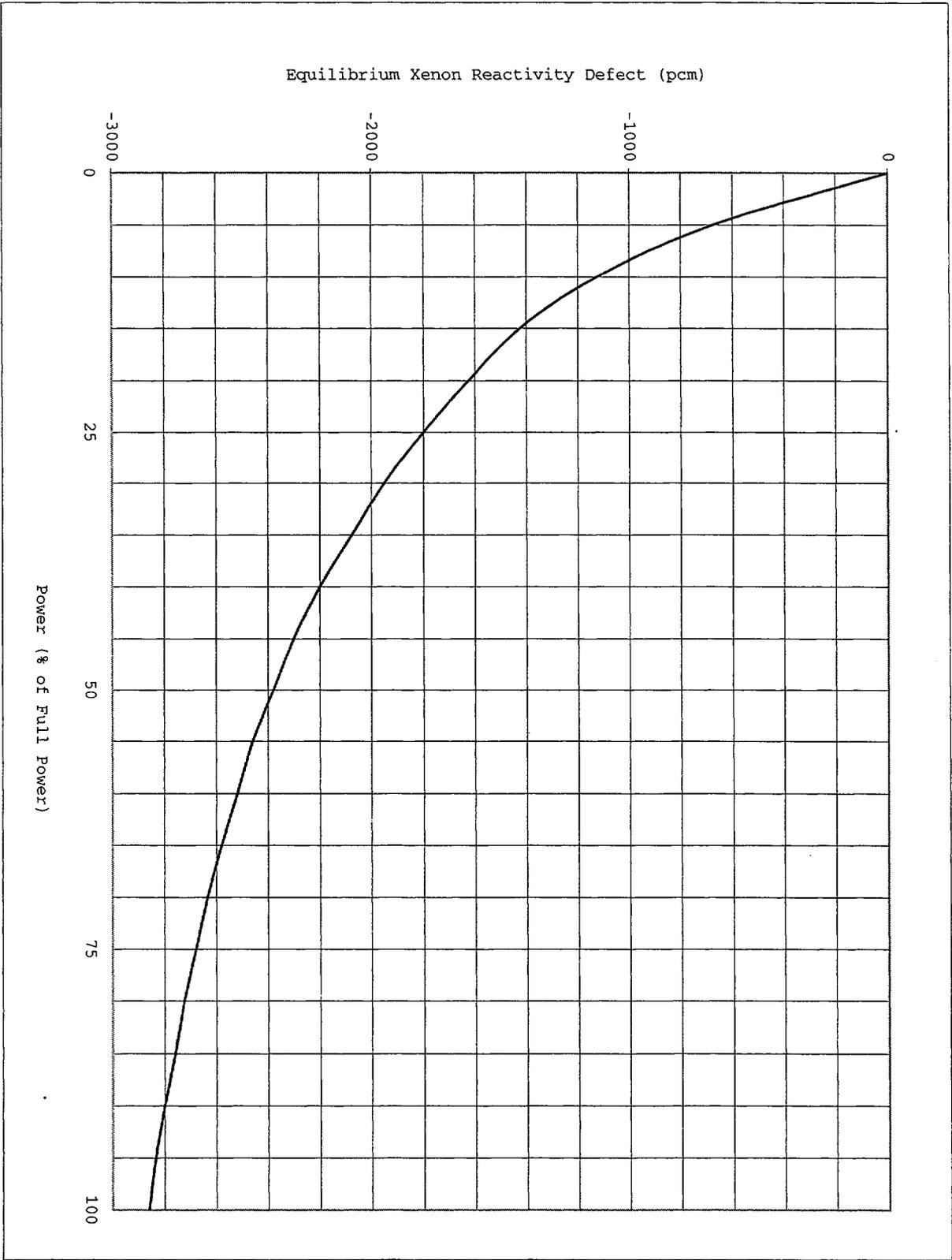


FIGURE 9-3  
Inverse Boron Worth  
HFP\_ARO



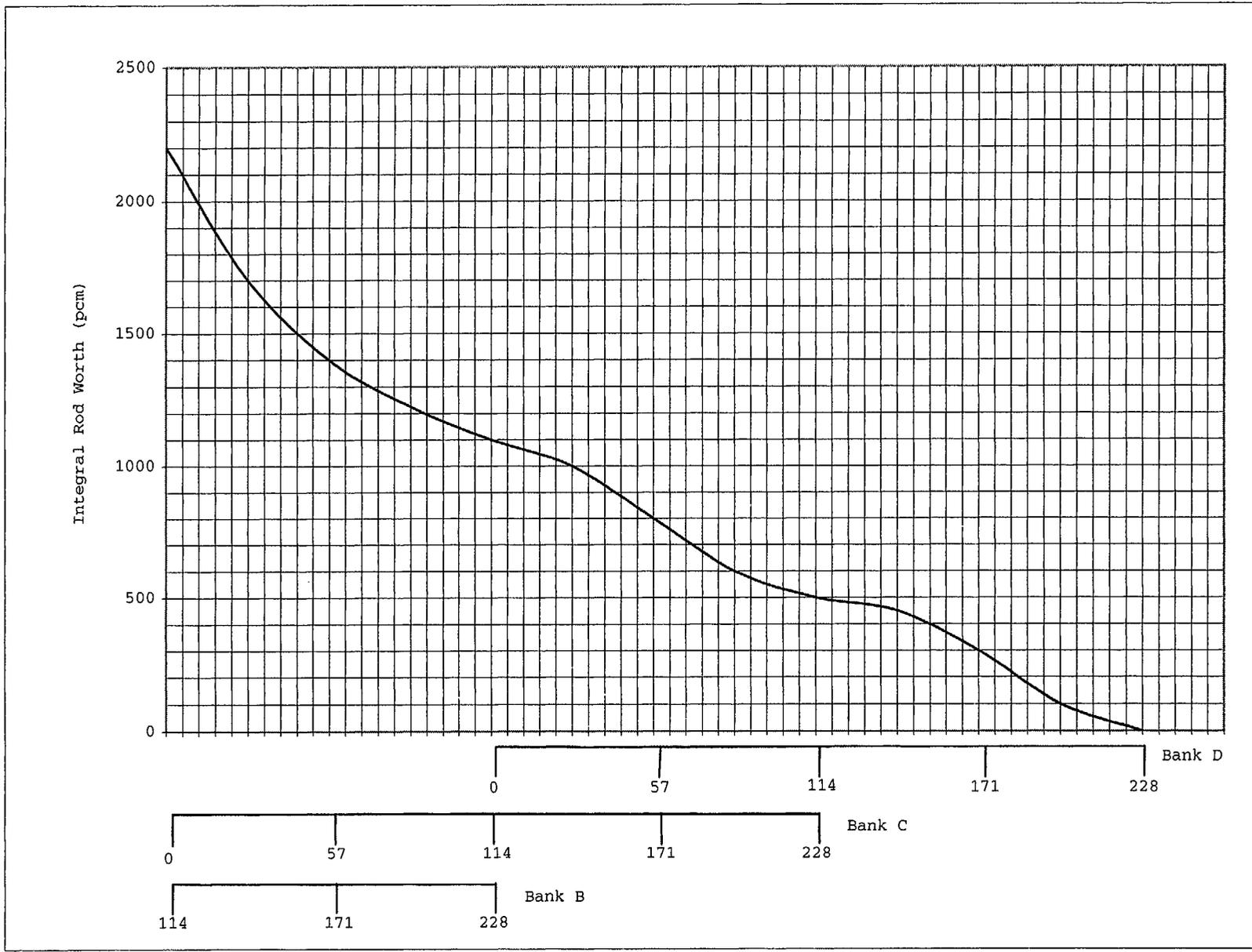
FIGURE 9-4  
Equilibrium Xenon Worth  
HEP





Xenon Reactivity Defect  
BOL

FIGURE 9-5



Integral Rod Worth  
HZP, BOL

FIGURE 9-6

## 10. PHYSICS TEST COMPARISONS

Startup physics testing is described in UFSAR Chapter 14. The startup testing information contained in this section is presented to provide the perspective on the benchmarking of measured to predicted results. The predicted results presented are based on the PDQ07/EPRI-NODE-P models. Reference 28 presents the benchmarking results for the CASMO-3/SIMULATE-3 models.

### 10.1 Introduction

This section presents measurement and calculational techniques and comparisons of calculated and measured results for some key core physics parameters. The physics parameters include hot zero power (HZIP) and hot full power (HFP) critical boron concentrations, HZIP control rod worths and ejected rod worths, and HZIP isothermal temperature coefficients.

The measured data is from the McGuire Nuclear Station, Unit 1 Cycles 1 and 1A, and Unit 2, Cycle 1. (Broken hold down springs on some Burnable Poison rods were found during an outage on McGuire Unit 1 at 191.5 EFPD. During this outage, 94 of 96 Burnable Poison Rod Assemblies were removed from the core. Cycle 1A is the continuation of Cycle 1 but without the Burnable Poison Rods.) The measurement techniques discussed are those currently used at the station. The HZIP measurements were taken at beginning-of-cycle (BOC) during the Zero Power Physics Testing. The HFP boron concentration measurements were taken at various time steps throughout the cycles. All calculations were performed with EPRI-NODE-P.

The comparisons of calculated and measured results present the means of the differences between the measured and calculated data and the corresponding standard deviations. The mean and standard deviation are defined as follows:

$$\text{Mean} = \bar{X} = \frac{\sum X_i}{n}$$

$$\text{Standard Deviation} = S = \sqrt{\frac{\sum(\bar{x} - x_i)^2}{n-1}}$$

where:  $x_i$  = value for the  $i^{\text{th}}$  observation

$n$  = number of observations.

## 10.2 Critical Boron Concentrations

### 10.2.1 Measurement Technique

Critical boron concentrations are measured at HZP and HFP by an acid-base titration of a reactor coolant system sample.

The measurement uncertainty for critical boron concentrations is due to (1) error in the titration method and (2) error due to differences between the sample concentration and the core average concentration. Based on conservative estimates of these errors, the total uncertainty associated with the critical boron concentration measurements is less than 20 ppmb.

### 10.2.2 Calculational Technique

Critical boron concentrations are calculated at HZP and HFP using EPRI-NODE-P in the boron search mode. Since the search does not yield an exactly critical value, fixed boron runs using EPRI-NODE-P are also made to calculate a boron worth, which is then used to correct the calculated boron concentration to exactly critical.

### 10.2.3 Comparison of Calculated and Measured Results

#### 10.2.3.1 Hot Zero Power Comparison

The calculated and measured critical boron concentrations at HZP and BOC for McGuire Unit 1, Cycles 1 and 1A, and Unit 2, Cycle 1 are compared in Table 10-1. Each entry corresponds to a different control rod position. The mean

of the differences for these three cycles was found to be -7 ppmb with a standard deviation of 16 ppmb.

#### 10.2.3.2 Hot Full Power Comparison

The calculated and measured critical boron concentrations at HFP for McGuire Unit 1, Cycles 1 and 1A, are compared in Table 10-2. The mean of the differences for these cycles is -41 ppmb with a standard deviation of 11 ppmb.

The data displayed in Table 10-2 can be visualized better by examining plots of soluble boron concentration as a function of burnup. These boron letdown curves are shown in Figures 10-1 and 10-2.

#### 10.2.4 Summary

The comparison between EPRI-NODE-P and measured critical boron concentrations at HZP and HFP indicate EPRI-NODE-P can adequately predict soluble boron concentrations.

### 10.3 Control Rod Worth

#### 10.3.1 Measurement Techniques

Individual control rod bank worths are measured by the boron swap technique. This technique involves a continuous decrease in boron concentration together with an insertion of the control rods in small, discrete steps. The change in reactivity due to each insertion is determined from reactivity computer readings before and after the insertion. The worth of each rod bank is the sum of all the reactivity changes for that bank. Measured bank worths in ppmb can be determined independent of the reactivity computer by using the measured boron endpoints.

#### 10.3.2 Calculational Techniques

Individual and total controlling rod bank worths in terms of reactivity are

calculated by making two EPRI-NODE-P runs. The first is a boron search run with the rod bank(s) out. The boron concentration found in this run is then used in a fixed boron run with the rod bank(s) in. The difference in reactivity between these two runs with constant boron concentration is the rod bank(s) worth.

Bank worths were also calculated using the calculated Boron endpoints. These bank worths are in terms of ppmB.

### 10.3.3 Comparison of Calculated and Measured Results

A comparison of calculated and measured control rod worths in terms of reactivity is shown in Table 10-3. This table compares the worths of control banks: D, C, B, and A and shutdown banks: E, D, and C at HZP and BOC for McGuire Unit 1, Cycles 1 and 1A, and McGuire Unit 2, Cycle 1. A comparison of calculated and measured control rod worths in terms of ppmB is shown in Table 10-4. This table also compares the worths of control banks: D, C, B, and A and shutdown banks: E, D, and C at HZP and BOC for McGuire Unit 1, Cycles 1 and 1A, and McGuire Unit 2 Cycle 1. Table 10-5 is a comparison of PDQ07 calculated and measured control rod worths.

PDQ07 calculated bank worths agree well to measured with an average difference of 2.7% and a standard deviation of 3.3%. EPRI-NODE-P calculated bank worths similarly agreed well with an average difference of -4.5% and a standard deviation of 5.1%. Rod worths calculated using boron endpoints also agreed well, with an average difference of -2.2% and a standard deviation of 7.9%.

### 10.3.4 Summary

The comparisons between the calculated and measured control rod worths at HZP indicate that EPRI-NODE-P can adequately predict control rod worths. Tables 10-3 and 10-4 indicate consistent agreement using either reactivity or boron endpoint measurement techniques.

## 10.4 Ejected Rod Worths

Ejected rod worth is defined here as the measured worth of the worst case ejected rod. No error adjustments have been included.

### 10.4.1 Measurement Technique

Ejected rod worths are measured by boron swap. The boron swap method is similar to the method used to measure control rod worth. It involves maintaining criticality by varying the boron concentration to compensate for the ejection of the worst case rod. The control rod positions are held constant. As was done for control rod worth, the ejected rod worth is determined from the reactivity computer.

### 10.4.2 Calculational Techniques

Ejected rod worths are calculated using EPRI-NODE-P to simulate boron swap. A boron search run is first performed to determine the critical boron concentration at the rod group position. The boron concentration as calculated in the EPRI-NODE-P run should be corrected for exact criticality. Using this corrected boron concentration and a constant rod group position, the reactivity is determined with the worst case rod first in and then out. The ejected rod worth is the difference in reactivity between the worst case rod in and out.

### 10.4.3 Comparison of Calculated and Measured Results

A comparison of calculated and measured ejected rod worth for McGuire Unit 1, Cycle 1, is given in Table 10-6.

## 10.5 Isothermal Temperature Coefficients

The isothermal temperature coefficient is defined as the change in reactivity per unit change in moderator temperature at hot zero power, i.e.,

$$\alpha_T = \frac{\Delta\rho}{\Delta T}$$

#### 10.5.1 Measurement Techniques

The isothermal temperature coefficient is measured by executing an average moderator temperature ramp to +5°F and then a ramp down to the initial equilibrium critical conditions. During each change, reactivity is measured on the reactivity computer and other pertinent data is measured. After each change, steady state conditions are established. The isothermal temperature coefficient is determined as the change in reactivity between plateaus divided by the change in temperature. Since two different temperature ramps are executed, two coefficients can be determined. The reported isothermal temperature coefficient is an average of these two coefficients.

#### 10.5.2 Calculational Technique

The isothermal temperature coefficient at HZP is calculated using EPRI-NODE-P. Two cases with the same boron concentration and rod positions but different moderator temperatures are run. The isothermal temperature coefficient is the difference in reactivity between the two cases divided by the difference in the moderator temperatures.

#### 10.5.3 Comparison of Calculated and Measured Results

A comparison of calculated and measured isothermal temperature coefficients at HZP and BOC for McGuire Unit 1, Cycles 1 and 1A, and Unit 2, Cycle 1 is presented in Table 10-7. The mean of all the differences was found to be 1.38 pcm/°F with a standard deviation of 1.87 pcm/°F.

#### 10.5.4 Summary

The comparison between calculated and measured isothermal temperature coefficients indicates that EPRI-NODE-P is a good predictor of isothermal temperature coefficients.

TABLE 10-1

McGuireCritical Boron Concentrations at Hot Zero Power, BOC

<u>Unit</u>	<u>Cycle</u>	<u>Critical Boron Conc. PPM</u>		<u>Difference</u>
		<u>Calculated</u>	<u>Measured</u>	
1	1	1301	1310	-9
		1242	1248	-6
		1123	1128	-5
		1033	1029	4
		972	967	5
		888	891	-3
		822	819	3
		728	723	5
1	1A	1269	1310	-41
		1200	1242	-42
		1090	1125	-35
2	1	1280	1295	-15
		1221	1217	4
		1101	1097	4
		1002	997	5
		944	938	6
		861	860	1
		788	791	-3
		691	694	-3
Mean		----	----	-6.6
Standard Deviation		----	----	15.7

Difference = Calculated - Measured

TABLE 10-2

McGuire 1 Cycle 1-1A  
Hot Full Power Critical Boron Concentrations

<u>Unit</u>	<u>EFPD</u>	<u>Critical Boron Conc. PPM</u>		<u>Difference</u>
		<u>Calculated</u>	<u>Measured</u>	
1	24.6	860	880	-20
	34.4	846	865	-19
	39.2	838	864	-26
	49.2	823	862	-39
	82.2	761	801	-40
	90.4	745	790	-45
	99.0	729	771	-42
	101.2	724	762	-38
	126.0	667	724	-57
	154.4	600	650	-50
180.7	531	591	-60	
1A	203.7	782	831	-49
	217.5	713	751	-38
	227.8	673	719	-46
	232.9	653	696	-43
	238.5	631	677	-46
	255.2	566	615	-49
	279.8	473	511	-38
	300.6	395	434	-39
	330.8	281	318	-37
	Mean		----	----
Standard Deviation		----	----	10.5

Difference = Calculated - Measured

TABLE 10-3

McGuire  
Control Rod Worths at Hot Zero Power, BOC

<u>Unit/Cycle</u>	<u>Bank</u>	<u>Rod Worth (PCM)</u>		<u>Difference (PCM)</u>	<u>Difference (%)</u>
		<u>Calculated</u>	<u>Measured</u>		
1/1	CD	606	669	-63	-9.4
	CC	1217	1250	-33	-2.6
	CB	925	996	-71	-7.1
	CA	654	695	-41	-5.9
	SE	884	840	44	5.2
	SD	668	755	-87	-11.5
	SC	961	1011	-50	-4.9
1/1A	CD	685	712	-27	-3.8
	CC	1100	1038	62	6.0
2/1	CD	604	664	-60	-9.0
	CC	1224	1283	-59	-4.6
	CB	1004	1105	-101	-9.1
	CA	618	678	-60	-8.8
	SE	862	853	9	1.1
	SD	738	771	-33	-4.3
	SC	992	1026	-31	-3.0
Mean		----	----	-37.6	-4.5
Standard Deviation		----	----	43.8	5.1

Difference (PCM) = Calculated - Measured

$$\text{Difference(\%)} = \frac{\text{Calculated} - \text{Measured}}{\text{Measured}} \times 100$$

TABLE 10-4

McGuire  
Control Rod Worths at Hot Zero Power, BOC  
Using Boron Endpoints

<u>Unit/Cycle</u>	<u>Bank</u>	<u>Rod Worth (PPM)</u>		<u>Difference (PPM)</u>	<u>Difference (%)</u>
		<u>Calculated</u>	<u>Measured</u>		
1/1	CD	59	62	-3	-4.8
	CC	119	120	-1	-0.8
	CB	90	99	-9	-9.1
	CA	61	62	-1	-1.6
	SE	84	76	8	10.5
	SD	66	72	-6	-8.3
	SC	94	96	-2	-2.1
1/1A	CD	69	68	1	1.5
	CC	110	117	-7	-6.0
2/1	CD	59	78	-19	-24.4
	CC	120	120	0	0.0
	CB	99	100	-1	-1.0
	CA	58	59	-1	-1.7
	SE	83	78	5	6.4
	SD	73	69	4	5.8
	SC	97	97	0	0.0
Mean		----	----	-2.0	-2.2
Standard Deviation		----	----	6.3	7.9

Difference (PCM) = Calculated - Measured

$$\text{Difference(\%)} = \frac{\text{Calculated} - \text{Measured}}{\text{Measured}} \times 100$$

TABLE 10-5

McGuirePDQ07 Calculated Rod Worths vs. Measured Rod Worths at HZP, BOC

<u>Unit/Cycle</u>	<u>Bank</u>	<u>Rod Worth (PCM)</u>		<u>Difference (PCM)</u>	<u>Difference (%)</u>
		<u>Calculated</u>	<u>Measured</u>		
1/1	D	644	669	-25	-3.7
	C	1214	1250	-36	-2.9
	B	962	996	-34	-3.4
1/1A	D	667	712	-45	-6.3
	C	1088	1038	50	4.8
2/1	D	637	664	-27	-4.1
	C	1261	1283	-22	-1.7
	B	1090	1105	-15	-1.4
	A	638	678	-40	-5.9
Mean		----	----	-22	-2.7
Standard Deviation		----	----	28	3.3

Difference (PCM) = Calculated - Measured

$$\text{Difference(\%)} = \frac{\text{Calculated} - \text{Measured}}{\text{Measured}} \times 100$$

TABLE 10-6

McGuire 1 Cycle 1  
Ejected Rod Worths

<u>Cycle</u>	<u>Location</u>	<u>Worth (PCM)</u>		<u>Difference (PCM)</u>
		<u>Calculated</u>	<u>Measured</u>	
1	D-12	406	432	-26

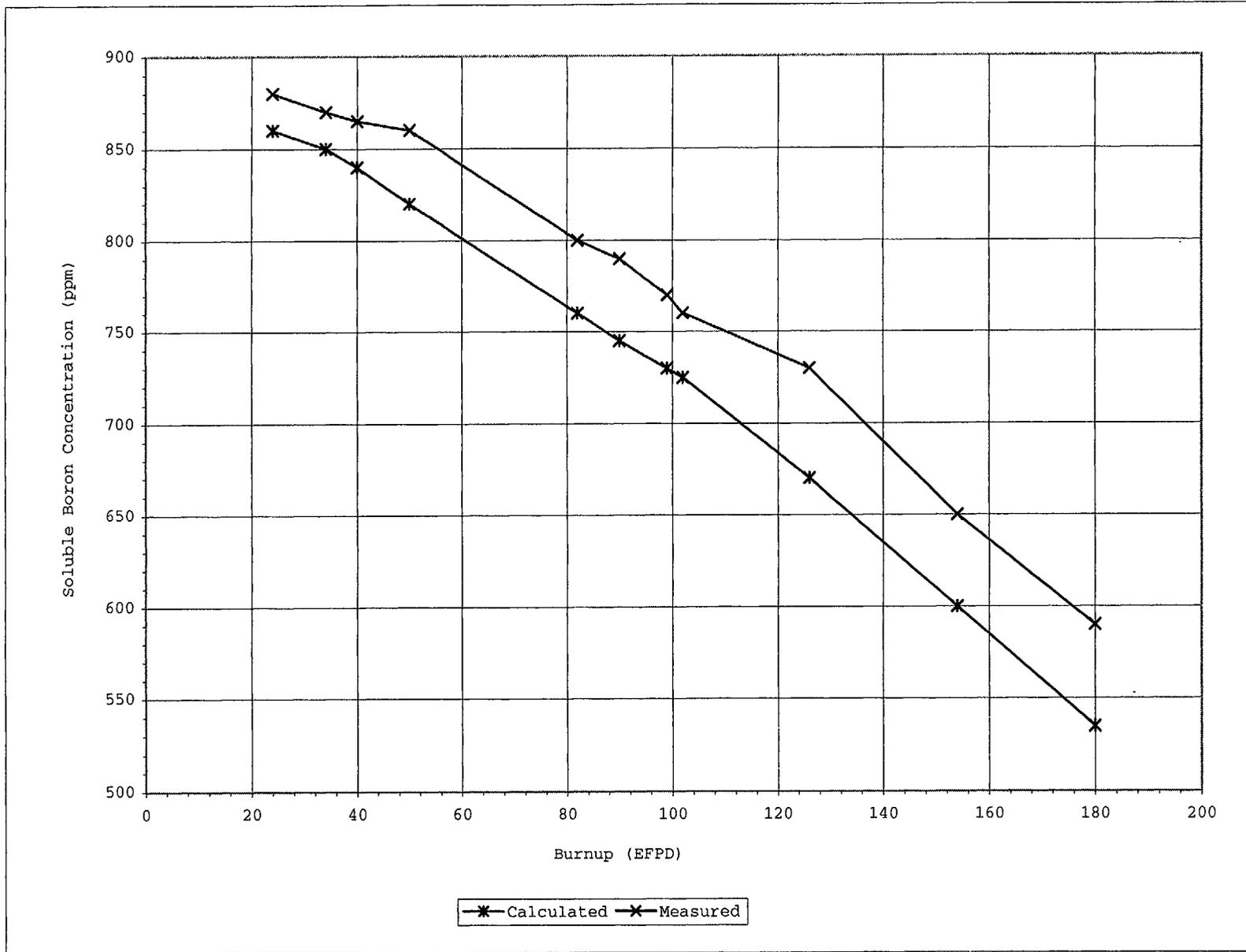
Difference (PCM) = Calculated - Measured

TABLE 10-7

McGuireIsothermal Temperature Coefficients at Hot Zero Power, BOC

<u>Unit/Cycle</u>	<u>Control Rod Configuration</u>	<u>Temp. Coeff., (PCM/°F)</u>		<u>Difference (PCM/°F)</u>
		<u>Calculated</u>	<u>Measured</u>	
1	ARO	-1.03	-0.57	-0.46
	D in	-2.09	-2.02	-0.07
	C & D in	-6.03	-5.86	-0.17
	B, C, & D in	-6.08	-6.83	0.75
	A, B, C, & D in	-9.37	-9.72	0.35
1	ARO	-4.51	-1.13	-3.38
	D in	-5.86	-1.98	-3.88
	C & D in	-9.76	-4.83	-4.93
2	ARO	-2.34	-1.41	-0.93
	D in	-3.54	-2.73	-0.81
	C & D in	-7.70	-6.07	-1.63
Mean		----	----	-1.38
Standard Deviation		----	----	1.87

Difference (PCM/°F) = Calculated - Measured



McGuire 1 Cycle 1  
Boron Letdown Curves

FIGURE 10-1

McGuire 1 Cycle 1A  
Boron Letdown Curves

FIGURE 10-2

