Proposed Changes to the Safety Analysis Report for the NBSR

4.5.1.1.3 Alternative Fuel Management Schemes (AFMS)

Alternative Fuel Management Schemes (AFMS) may be implemented based on NBSR inventory or other operational needs. These AFMS core loadings do deviate from the Original Fuel Management Scheme (OFMS) as described in 4.5.1.1.2. For instance the number of fresh fuel elements, their locations, the types of used elements (7th cycle, 6th cycle, etc.) and their locations may be changing amongst other core configuration characteristics. The analysis scenarios in the updated FSAR assume a prescribed core loading pattern, namely OFMS. Since nuclear core characteristics may vary for such AFMS's, further evaluations are required to ensure safe operation. Henceforth, the "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure" details the bounding (limiting) conditions, analysis of methodology, neutronics and thermal-hydraulics models, relevant codes and scripts, correlations, accident scenarios to be evaluated, quality assurance, version control, and verification and validation necessary to ensure the safety of any AFMS that may be proposed for NBSR operations.

Section 1 of NBSR-0018-DOC-00 provides a description of the OFMS and AFMS, Section 2 of the document provides a basis for the analysis providing limitations to evaluate potential AFMS. Section 3 provides detailed safety analysis for a demonstration AFMS, namely the Demonstration Core Loading (DCL), along with discussion of results and conclusions. Appendix-A provides detailed analysis results for demonstration AFMS core loading such as burnup for each fuel element, kinetic parameters and prompt neutron lifetime. Appendix-B provides detailed method for fuel element composition calculations. Appendix C provides descriptions and modifications for accident scenarios. Some scenario conditions are updated based on facility changes and available new information. Appendix D describes the power input development process for the Relap5 model.

No Significant Hazard Consideration Determination

In a license amendment request dated February 1, 2023, the NIST Center for Neutron Research (NCNR) requested an amendment to the facility license involving a SAR change to describe engineering analysis procedures for any Alternative Fuel Management Schemes (AFMS) where the NBSR core is loaded with a different core loading pattern than as described in the updated FSAR. An AFMS is any core loading pattern that deviates from the Original Fuel Management Scheme (OFMS) in a manner such that the number of the specific type of fuel elements, such as fresh or used, is different than usual and/or their locations in the core are modified. Any such AFMS is a modification in how the NBSR core performs its design function of producing 20 MW and therefore requires a License Amendment Request (LAR). Furthermore, Technical Specifications Section 5.3 Basis bullet (1) in part states that "Significant changes in core loading patterns would require a recalculation of the power distribution to ensure that the CHFR would be within acceptable limits.". As required by 10 CFR 50.91(a), the following revised analysis is presented to show the proposed amendment does not create a significant hazard using the criteria of 10 CFR 50.92(c).

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

AFMS loadings deviate from the core loading scheme as described in the FSAR 4.5.1.1.2 "Fuel Management Scheme". This amendment introduces a new section to the updated FSAR, "4.5.1.1.3 Alternative Fuel Management Schemes (AFMS)", which describes bounding conditions and analysis requirements for any AFMS. The amendment also introduces an engineering procedure, namely "NBSR-0018-DOC-00 NBSR Alternative Core Loading Schemes Analysis Procedure" which describes the OFMS and AFMS, a basis for the analysis, providing limitations to evaluate potential AFMS, detailed safety review for a demonstration AFMS, along with a discussion of results and conclusions to be included in subsequent ECN's dealing with similia AFMS core loadings. The procedure provides a basis to analyze core loading so that none of the Technical Specifications (TS) are exceeded.

Several accident scenarios and therefore consequences may be affected by AFMS core loading deviations. Particularly, all accidents shown in Table 1 are required to be re-evaluated for any AFMS core loading. Other accident scenarios given in the FSAR, including "Loss of Primary Coolant" (a major rupture in the cold leg of the primary system is assumed, which leads to draining the reactor core), "Maximum Hypothetical Accident (MHA)", "Experiment Malfunction" and "External Event" are independent of core loading changes and therefore remain unchanged. Additionally, Natural Circulation Cooling at Low Power Operation must be analyzed for each AFMS to show compliance with Technical Specification 2.2. Natural Circulation Cooling at Low Power Operation is not an accident scenario but an analysis to show natural circulation at low power operations.

Note that all of the accident scenarios and Natural Circulation Cooling at Low Power Operation conditions are analyzed using the RELAP5 model as described in the "NBSR-0018-DOC-00 NBSR Alternative Core Loading Schemes Analysis Procedure". The misloading accident is the only one that will require unique power distributions from corresponding MCNP simulations with the misloaded fuel configuration. Some scenario conditions are updated based on facility changes and available new information. "NBSR-0018-DOC-00 Appendix C" provides descriptions and modifications for accident scenarios.

Table 1. The accident sequences to be re-analyzed, and their references in the SAR.

	Accident Sequence	**Section in the SAR**
#1	Startup Accident	13.1.2.2.2.1
#2	Maximum Reactivity Insertion Accident	13.2.2
#3	Loss of Offsite Power	13.1.4.1
#4	Loss of Offsite Power with Shutdown Pump failure	13.1.4.5
#5	Seizure of One Primary Pump	13.1.4.2
#6	Throttling of Coolant Flow to the Outer Plenum	13.1.4.4
#7	Throttling of Coolant Flow to the Inner Plenum	13.1.4.3
#8	Misloading of Fuel	13.1.5

Based on detailed analysis provided in the technical report, and because these AFMS accident scenarios are specifically analyzed for probability and consequences, there are, by definition, no changes in the probability of occurrences or the consequences of previously analyzed accidents. Therefore, the proposed FSAR amendment allowing analysis of AFMS does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The requested amendment to the facility license involves a SAR change to describe engineering analysis procedures for any Alternative Fuel Management Schemes (AFMS) wherein which the NBSR core is loaded with a different core loading pattern than as described in the updated FSAR. An AFMS is any core loading pattern that deviates from the Original Fuel Management Scheme (OFMS) in a manner such that the number of the specific type of fuel elements, such as fresh, or used is different than usual and/or their locations in the core are modified. As there are no other changes besides that of fuel loading, changes in the core loading pattern do not initiate a different kind of accident. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The requested amendment to the facility license involves a SAR change to describe engineering analysis procedures for any Alternative Fuel Management Schemes (AFMS) wherein which the NBSR core is loaded with a different core loading pattern than as described in the updated FSAR. An AFMS is any core loading pattern that deviates from the Original Fuel Management Scheme (OFMS) in a manner such that the number of the specific type of fuel elements, such as fresh, or used is different than usual and/or their locations in the core are modified. The AFMS can be deemed acceptable as long as the proposed AFMS is analyzed according to the “NBSR-0018-DOC-00” and found to be within the updated FSAR, Technical Specifications limitations and boundary conditions listed therein. The boundary conditions are based on the Technical Specifications and updated FSAR requirements. Because these alternate fuel management schemes are specifically analyzed for a reduction in margin of safety, there is, by definition, no significant reduction in margin of safety. The proposed amendment contains no changes in the Technical Specification or other safety limitations as described in the updated FSAR. Therefore, the proposed amendment of the SAR in allowing this operation does not involve a significant reduction in a margin of safety.

Conclusion

Based on the above determinations, the NCNR concludes that the proposed amendment to the NBSR SAR does not involve a significant hazards consideration under the standards set forth in 10CFR50.92(c), and accordingly, a finding of no significant hazards is justified.

ENGINEERING CHANGE NOTICE (ECN)			ECR No.:	1269
ECR Title:	Alternative Fuel Management Schemes			
System:	Reactor Structures, Core, etc.		Date:	1/30/2023

Design Description

After the incident in February 2021, debris was found on several fuel elements and all of the fuel elements in core loading 654 were deemed unusable. At this point, there are only the 7th cycle and fresh fuel elements that could be loaded to the core. Hence, a series of Alternative Fuel Management Schemes (AFMS) needs to be developed for the NBSR to reach the equilibrium core as described in the updated FSAR (UFSAR), so that the Original Fuel Management Scheme (OFMS) can be used. These AFMS core loadings do deviate from the OFMS, for instance, the number of fresh fuel elements, their locations, and the number of used elements and/or their locations, etc. may be changing, which is a modification in how the NBSR core performs its design function of producing 20 MW and therefore requires a License Amendment Request (LAR).) Furthermore, Technical Specifications Section 5.3 Basis bullet (1) in part states that "Significant changes in core loading patterns would require a recalculation of the power distribution to ensure that the CHFR would be within acceptable limits.".

To that end, a new section will be inserted in the UFSAR, "4.5.1.1.3 Alternative Fuel Management Schemes (AFMS)", which describes bounding conditions and analysis requirements for any AFMS. The "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure", attached to this ECN, details the bounding (limiting) conditions, analysis of methodology, neutronics, and thermal-hydraulics models, relevant codes and scripts, correlations, accident scenarios to be evaluated, quality assurance, version control, and verification and validation necessary to ensure the safety of any AFMS that may be proposed for NBSR operations.

Attachments

1- "SARBinder.pdf" updated FSAR, January 2023
2- "2009 SER.pdf" license update 2009 SER
3- "SAR2009.pdf" updated FSAR 2009
4- "TS3.1.4_Elaboration.pdf" elaborated evaluation of Tech Spec 3.1.4
5- "FEER_Report_v3.pdf" Fuel element self-protection evaluations
6- "DCL50.59.docx" sample 50.59 analysis for the Demonstration Core Loading (DCL)
7- "Calc Notebook - non-LOCA Models 4-21-15.pdf"
8- "TR5 Technical Specification with Amend 11 Dec 19 2017.pdf"
9- "BNLAccidentMemo.docx" BNL Thermal Hydraulic analysis report
10- "BNLNeutronicMemo.docx" BNL independent neutronic analysis report
11- "NBSR-0018-DOC-00.pdf" NBSR Alternative Fuel Management Schemes Analysis Procedure
12- **Refs.zip**: Compressed folder containing all reference documents used for the ECN

There are no Technical Specifications, or procedural or technical drawing changes associated with this ECN.

The License Amendment proposes following changes when compared to updated SAR as of 2009, license renewal

- New section in the SAR, "4.5.1.1.3 Alternative Fuel Management Schemes (AFMS)", which describes bounding conditions and analysis requirements for any the Alternative Fuel Management Schemes (AFMS).
- "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure", which:

  - describes the OFMS and AFMS, a basis for the analysis; detailed safety analysis for a demonstration AFMS, along with a discussion of results and conclusions to be included in subsequent ECN's dealing with AFMS core loadings.
  - provides descriptions for accident analyses as corrected and updated to better reflect current plant characteristics
  - provides a detailed summary of the differences between the neutronic and thermal hydraulic models and the UFSAR model (2009).
  - the MCNP model with increased discretization in the fuel elements and geometrical changes adapted to match current core characteristics.
  - the design-basis accidents for the AFMS can be analyzed with the system thermal-hydraulics code RELAP5/MOD3.3. (updated versions of the ones used in the UFSAR-2009).
  - Additional CHFR correlations (Sudo-Kaminaga in addition to Mirshak, Costa in addition to Saha-Zuber)
- Burnup based linear approximation method to calculate fuel element compositions based on equilibrium core characteristics (linearly related to the ^{235}U content).

Safety Considerations, Identification, and/or Analysis

Safety considerations and analysis are described in the "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure", "NSH SAR" and "50.59" documents attached to this ECN.

Required Tests, Retests, Surveillances, or Measurements

Required tests and surveillances are provided in the "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure", attached to this ECN.

Safety Evaluation and Conclusion

This ECN introduces an engineering procedure, namely "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure" which describes the OFMS and AFMS, a basis for the analysis providing limitations to evaluate potential AFMS, detailed safety analysis for a demonstration AFMS, along with a discussion of results and conclusions to be included in subsequent ECN's dealing with AFMS core loadings. The procedure provides a basis to analyze core loading so that none of the technical specifications (TS) are exceeded.

Based on the safety considerations and analysis described in the "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure", "NSH SAR" and "50.59" documents, the NCNR concludes that the proposed amendment to the NBSR SAR is safe for the operation of the NBSR.

10 CFR 50.59 EVALUATION			ECR No.:	1269
ECR/Experiment Title:	Alternative Fuel Management Schemes			
System:	Reactor Structures, Core, etc.		**Date:**	**1/30/2023**

Does the proposed ECN:	YES	NO
A. **Require** a change to the Technical Specifications (Enter justification below).	☐	☒

After the incident in February 2021, debris was found on several fuel elements and all of the fuel elements in core loading 654 were deemed unusable. At this point, there are only the 7th cycle and fresh fuel elements are available in the NBSR inventory. Hence, a series of Alternative Fuel Management Schemes (AFMS) need to be developed for the NBSR to reach the equilibrium core as described in the updated FSAR (UFSAR), so that the Original Fuel Management Scheme (OFMS) can be used. These AFMS core loadings do deviate from the OFMS, for instance, the number of fresh fuel elements, their locations, the types of used elements (7th cycle, 6th cycle, etc.), and their locations may be changing amongst other core configuration characteristics. Any such AFMS is a modification in how the NBSR core performs its design function of producing 20 MW and therefore requires a License Amendment Request (LAR).

This ECN introduces an engineering procedure, namely "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure" which describes the OFMS and AFMS, a basis for the analysis providing limitations to evaluate potential AFMS, detailed safety analysis for a demonstration AFMS, along with a discussion of results and conclusions to be included in subsequent ECN's dealing with AFMS core loadings. The procedure provides a basis to analyze core loading so that none of the technical specifications (TS) are exceeded. Hence, there are no changes necessary to the existing Technical Specifications.

Does the proposed ECN:	YES	NO
B. Result in more than a minimal increase in the ***frequency*** of occurrence of an accident previously evaluated in the updated FSAR (Enter justification below).	☐	☒

This ECN introduces an engineering procedure, namely "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure" which describes the OFMS and AFMS, a basis for the analysis providing limitations to evaluate potential AFMS, detailed safety analysis for a demonstration AFMS, along with a discussion of results and conclusions to be included in subsequent ECN's dealing with AFMS core loadings. The procedure provides a basis to analyze core loading so that none of the technical specifications (TS) are exceeded. The engineering procedure "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure" and the requested LAR does not result in more than a minimal increase in the frequency of an accident previously evaluated in the UFSAR.

Does the proposed ECN:	YES	NO
C. Result in more than a minimal increase in the ***likelihood*** of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the updated FSAR (Enter justification below).	☐	☒

This ECN introduces an engineering procedure, namely "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure" which describes the OFMS and AFMS, a basis for the analysis providing limitations to evaluate potential AFMS, detailed safety analysis for a demonstration AFMS, along with a discussion of results and conclusions to be included in subsequent ECN's dealing with AFMS core loadings. The procedure provides a basis to analyze core loading so that none of the technical specifications (TS) are exceeded. There are no changes to structures, systems, or components important to safety previously evaluated in the UFSAR. The proposed change does not result in more than a minimal increase in the likelihood of the occurrence of a malfunction of a structure, system, or component important to safety.

Does the proposed ECN:	YES	NO
D. Result in more than a minimal increase in the ***consequences*** of an accident previously evaluated in the updated FSAR (Enter justification below).	☐	☒

AFMS loadings deviate from the core loading scheme as described in the FSAR 4.5.1.1.2 "Fuel Management Scheme". Several accident scenario analysis results may be affected by such core loading deviation from the OFMS. Namely, the "Startup Accident", "Maximum Reactivity Insertion Accident", "Loss of Offsite Power", "Loss of Offsite Power with Shutdown Pump failure", "Seizure of One Primary Pump", "Throttling of Coolant Flow to the Outer Plenum", "Throttling of Coolant Flow to the Inner Plenum" and "Misloading of Fuel", are required to be re-evaluated for any AFMS core loading. Additionally, Natural Circulation Cooling at Low Power Operation must be analyzed for each AFMS to show compliance with Technical Specification 2.2. Natural Circulation Cooling at Low Power Operation is not an accident scenario but an analysis to show natural circulation at low power operations.

Other accident scenarios, including "Loss of Primary Coolant" (a major rupture in the cold leg of the primary system is assumed, which leads to draining the reactor core), "Maximum Hypothetical Accident (MHA)", "Experiment Malfunction" and "External Event" are independent of core loading changes and are therefore unchanged.

This ECN introduces an engineering procedure, namely "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure" which describes the OFMS and AFMS, a basis for the analysis providing limitations to evaluate potential AFMS. This is only a methodology change, therefore, the proposed activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

Does the proposed ECN:	YES	NO
E. Result in more than a minimal increase in the ***consequences*** of a malfunction of a structure, system, or component important to safety previously evaluated in the updated FSAR (Enter justification below).	☐	☒

This ECN introduces an engineering procedure, namely "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure" which describes the OFMS and AFMS, a basis for the analysis providing limitations to evaluate potential AFMS, detailed safety analysis for a demonstration AFMS, along with a discussion of results and conclusions to be included in subsequent ECN's dealing with AFMS core loadings. The procedure provides a basis to analyze core loading so that none of the technical specifications (TS) are exceeded. There are no changes to the structures, systems, or components important to safety previously evaluated in the UFSAR. There is no more than a minimal increase in the challenge to the integrity of the fuel cladding. So, the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR remain the same.

Does the proposed ECN:	YES	NO
F. Create a possibility for an accident of a different type than any previously evaluated in the updated FSAR (Enter justification below).	☐	☒

AFMS core loadings may involve the introduction of a different number of fresh fuel elements, compared to 4 fresh fuel elements within a usual core loading. Each of these loadings may therefore require a separate analysis of whether a new type of accident is introduced or not. This ECN introduces an engineering procedure, namely "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure" which describes the OFMS and AFMS, a basis for the analysis providing limitations to evaluate potential AFMS, detailed safety analysis for a demonstration AFMS, along with a discussion of results and conclusions to be included in subsequent ECN's dealing with AFMS core loadings. The procedure provides a basis to analyze core loading so that none of the technical specifications (TS) are exceeded. Hence, there are no new accidents due to the introduction of a procedural analysis methodology for AFMS core loadings for the NBSR operations.

Does the proposed ECN:	YES	NO
G. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the updated FSAR (Enter justification below).	☐	☒

This ECN introduces an engineering procedure, namely "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure" which describes the OFMS and AFMS, a basis for the analysis providing limitations to evaluate potential AFMS, detailed safety analysis for a demonstration AFMS, along with a discussion of results and conclusions to be included in subsequent ECN's dealing with AFMS core loadings. The procedure provides a basis to analyze core loading so that none of the technical specifications (TS) are exceeded. There are no changes to structures, systems, or components important to safety previously evaluated in the UFSAR. The proposed change does not create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the UFSAR.

Does the proposed ECN:	YES	NO
H. Result in ***exceeding or altering*** a design basis limit for a fission product barrier as described in the updated FSAR (Enter justification below).	☐	☒

This ECN introduces an engineering procedure, namely "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure" which describes the OFMS and AFMS, a basis for the analysis providing limitations to evaluate potential AFMS, detailed safety analysis for a demonstration AFMS, along with a discussion of results and conclusions to be included in subsequent ECN's dealing with AFMS core loadings. The procedure provides a basis to analyze core loading so that none of the technical specifications (TS) are exceeded. There are no structural, dimensional, or material changes to the fission product barriers as described in the UFSAR and the accident analysis for the proposed analysis procedure for AFMS. Therefore, the proposed change does not result in exceeding or altering a design basis limit for a fission product barrier as described in the UFSAR.

Does the proposed ECN:	YES	NO
I. Result in a departure from a method of evaluation described in the updated FSAR used in establishing the design bases or in the safety analysis (Enter justification below).	☒	☐

This ECN introduces an engineering procedure, namely "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure" which describes the OFMS and AFMS, a basis for the analysis providing limitations to evaluate potential AFMS, detailed safety analysis for a demonstration AFMS, along with a discussion of results and conclusions to be included in subsequent ECN's dealing with AFMS core loadings. There are deviations/updates from the UFSAR in the thermal-hydraulic modeling and analysis, neutronic model, and accident scenario descriptions. Details of these changes are described and evaluated in the "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure", specifically in Section 2 and Appendix C.

Monte Carlo N-Particle code (MCNP) shall be used for all neutronics analyses. The MCNP model used for AFMS analysis implements the same model as the UFSAR with increased discretization in the fuel elements and geometrical changes adapted to match current core characteristics. "NBSR-0018-DOC-00" Section 2 provides a more detailed summary of the differences between this updated neutronics model and the UFSAR model.

The design-basis accidents for the AFMS can be analyzed with the system thermal-hydraulics code RELAP5/MOD3.3. It is important to note that the RELAP models used here are updated versions of the ones used in the UFSAR. The updated model uses a newer version of RELAP5, which comes with improved stability and more features, and it relies on more detailed heat distribution inputs from MCNP (i.e., increased discretization in the fuel elements). This allows for more detailed tracking of the heat transfer behavior throughout the fuel, and it offers a better understanding of the thermal safety margins, namely the local critical heat flux ratio (CHFR) and the onset of flow instability ratio (OFIR). More specifically, the UFSAR-2009 uses the Mirshak correlation for calculating the CHFR. As an additional verification, the Sudo-Kaminaga correlations are also used for calculating the CHFR. The UFSAR-2009 uses Saha-Zuber correlations for OFIR. The new RELAP models uses additionally the Costa correlation along with Saha-Zuber correlations. "NBSR-0018-DOC-00 " Section 2.3 provides a more detailed summary of the differences between this updated model and the UFSAR model.

Conclusion (Check one)	
☐	Based on the evaluation conducted in the above table, it is concluded that the proposed action ***does not meet*** any of the 10 CFR 50.59 criteria; therefore, the activity **does not require** a license amendment or prior NRC approval to perform the proposed action.
☒	Based on the evaluation conducted on the above table, it is concluded that the proposed action ***does meet one or more*** of the 10 CFR 50.59 criteria; therefore, the activity **does require** a license amendment to be obtained from the NRC under 10 CFR 50.90 to perform the proposed action.

NOTE
Consistent with the intent of 10 CFR 50.59, the justifications should be complete in the sense that another knowledgeable reviewer could draw the same conclusion. Restatement of the criteria in a negative sense or making simple statements of conclusion is **NOT** sufficient and should be avoided. The basis and logic used for engineering judgment and the determination should be documented to the extent practicable and to a degree commensurate with the safety significance and complexity of the activity.

NBSR Engineering Procedure NBSR-0018-DOC-00

NBSR Alternative Fuel Management Schemes Analysis Procedure

Initial Release

Dağistan Şahin, Osman Şahin Çelikten, Anil Gurgen, Abdullah G. Weiss

Lap-Yan Cheng, Peter Kohut, Cihang Lu, Athi Varuttamaseni

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Record of Revisions

Revision	Description of Revision	ECN No. (if applicable)	Date
Initial	0	1269	1/30/23

Contents

1 Introduction

After the incident in February 2021, debris was found on several fuel elements and all of the fuel elements in core loading 654 were deemed unusable. At this point, only the 7th cycle and fresh fuel elements are available in the NBSR inventory. Hence, there is a need to develop a series of Alternative Fuel Management Schemes (AFMS) to enable the NBSR to reach the equilibrium core as described in the updated FSAR (UFSAR), so that the Original Fuel Management Scheme (OFMS) can be used. These AFMS core loadings do deviate from the OFMS, for instance, the number of fresh fuel elements, their locations, the types of used elements (7th cycle, 6th cycle, etc.), and their locations may be changing amongst other core configuration characteristics. Any such AFMS is a modification in how the NBSR core performs its design function of producing 20 MW and therefore requires a License Amendment Request (LAR). Furthermore, Technical Specifications Section 5.3 Basis bullet (1) in part states that "Significant changes in core loading patterns would require a recalculation of the power distribution to ensure that the CHFR would be within acceptable limits. ". To that end, a new section will be inserted in the UFSAR, "4.5.1.1.3 Alternative Fuel Management Schemes (AFMS)", which describes bounding conditions for any AFMS. This technical report details the bounding (limiting) conditions, methodology od analytical approach, neutronics and thermal-hydraulics models, relevant codes and scripts, correlations, accident scenarios to be evaluated, quality assurance, version control, and verification and validation necessary to ensure the safety of any AFMS that may be proposed for NBSR operations.

Hence, Section 1 of this document describes the OFMS and AFMS, and Section 2 of this document provides a basis for the analysis identifying limitations to evaluate potential AFMS. Section 3 provides a detailed safety analysis for a demonstration AFMS, namely the Demonstration Core Loading (DCL), along with a discussion of results and conclusions. Section 4 provides a list of references. Appendix-A provides detailed analysis results for the demonstration of AFMS core loadings such as burnup for each fuel element, kinetic parameters, and prompt neutron lifetime. Appendix B provides a detailed method for fuel element composition calculations. Appendix C provides descriptions and modifications for accident scenarios. Some scenario conditions are updated based on facility changes and available new information. Appendix D describes the power input development process for the RELAP5 model.

1.1 Original Fuel Management Scheme (OFMS)

The equilibrium fuel management scheme will be denoted as the Original Fuel Management Scheme (OFMS), which is the scheme in use during normal operations. In the OFMS, shown in Figure 1, the fuel elements are denoted with two numbers and one letter. he letters are either E or W for the east or west side of the core, noting that a fuel element always stays on the east side or the west side of the core. The core position of a fuel element is given by a column letter (A through M) and a row number (1 through 7).

The updated NBSR FSAR [1], except new figure (see Figure 1) with color and arrows, in part, states that:

"Normally, fuel elements reside in the core for either seven or eight reactor cycles, a cycle including thirty-eight days of operation at full power. After each cycle, four fuel elements are removed to the spent-fuel storage pool, and the remaining twenty-six elements are moved following the pattern shown

in Figure 1. *In this figure, each position has two numbers associated with it. The first number indicates how many cycles the FE will be irradiated. The second number indicates the FE irradiation history. The *-1 representation is a FE in its first cycle, the *-2 representation is a FE in its second cycle, etc. As such, the 7-7 and the 8-8 fuel elements are in their seventh and eighth and hence final fuel cycles, respectively.*

At the end of a cycle, the fuel elements in positions E-4 and I-4, which are the 7-7 fuel elements, and in positions F-5 and H-5, the 8-8 fuel elements, are removed from the reactor. A FE in the 8-1 position is moved to an 8-2 position. A FE in the 8-2 is moved to an 8-3 position, and so forth. All remaining elements are moved accordingly with fresh fuel elements loaded into positions D-1, J-1, A-4, and M-4. Fuel migration is generally from the outside ring of elements toward the center, with fresher fuel elements located on the north side (rows 1 through 3), benefiting the cold neutron source. It will be shown later in this section, that since the thermal neutron flux peaks in the core center, the elements with the highest burnup have about the same power production as the freshest ones on the perimeter; that is, the radial power distribution is very flat, ± 15%, even though the thermal neutron flux may vary by nearly a factor of three.

While the fuel loading has increased over the years from 170 g to 350 g of high enrichment uranium (HEU) per element, the fuel management scheme shown in Figure 1 has been in use for most of the life of the facility.”

Out of the thirty fuel elements, 16 stays in the core for eight cycles, and the other 14 stay in the core for seven [1]. Per Figure 1, the red arrows follow the movement of the 7-cycle elements throughout their irradiated cycles, while the blue arrows follow the movement of the 8-cycle elements throughout their irradiated cycles. Note that the east and west sides have mirrored movement patterns for the fuel elements.

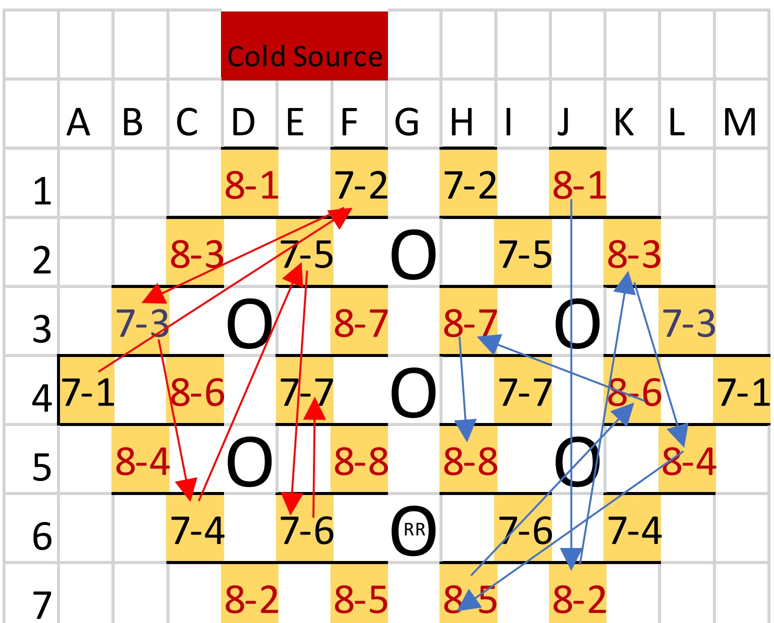

Figure 1. OFMS Fuel Element Movements

1.2 Alternative Fuel Management Schemes (AFMS)

An AFMS is any core loading pattern that deviates from the OFMS in a manner such that the number of the specific type of fuel elements, such as fresh, dummy or used is different than usual and/or their locations in the core are modified. This variation in core loading pattern will result in power peaking factors, both radial and axial, that are different from those assumed for the accident scenarios in the updated SAR. This document describes in Section 2 an evaluation methodology for the determination of the power peaking factors for cores using an AFMS and the corresponding safety limits for any acceptable AFMS for the NBSR. Illustration of the AFMS methodology is by way of applying it to a demonstration core loading (DCL), described in Section 3 of this report.

2 Analysis Methodology for Alternative Fuel Management Schemes

The methodologies adopted in this work rely heavily on neutronics and thermal-hydraulics computations that rigorously check for compliance with Technical Specifications and UFSAR requirements, and they are set up in such a way as to ensure complete transparency in the results. This includes verification, validation, and quality assurance efforts that conservatively assess the safety of the AFMS.

2.1 Boundary Conditions and Limitations

To ensure compliance with accepted operation conditions, a set of boundary conditions and limitations are considered for assessing the safety of the AFMS. This section provides the list of limitations, boundary conditions, and assumptions applicable to the analysis of any AFMS.

The list of limitations listed below are based on the TS [2] and UFSAR [1] requirements.

- At a given moment, up to 45.0 kilograms of contained Uranium-235 of any enrichment, provided that less than 5.0 kilograms of this amount be unirradiated can be in the NCNR inventory.
- NBSR is authorized to operate the reactor at steady-state power levels up to a maximum of 20 megawatts (thermal).
- The reactor fuel cladding temperature shall not exceed 842°F (450°C) for any operating conditions of power and flow.
- The maximum available excess reactivity for reference core conditions shall not exceed 15% $\Delta\rho$.
- The reactor shall not be operated unless the shutdown margin provided by the shim arms is greater than 0.757% $\Delta\rho$ ($1.00) with:
    - the reactor in any core condition,
    - all movable experiments in their most reactive condition,
    - the regulating rod fully withdrawn, and
    - the most reactive shim arm fully in the most reactive position.
- The reactor shall not operate unless all grid positions are filled with full-length fuel elements or thimbles, except during subcritical and critical startup testing with natural convection flow.
- The average fission density of a fuel element shall not exceed 2 x 10^{27} fissions/m^3 (less than 77.52% burnup [1]).

- The reactivity insertion rate for the four shim arms shall not exceed 5 x 10^{-4} $\Delta\rho/sec$.
- Steady state thermal hydraulic analysis shows that operation at less than 500 kW with natural circulation results in a CHFR and OFI ratio greater than 2.
- The minimum CHFR is dictated by the 95/95 statistic, which is a minimum CHFR of 1.21 for a 95% probability of no departure from nucleate boiling (DNB) and 1.4 for a 99.9% probability of no DNB when using Mirshak correlation [1]. For reference, the lowest CHFR using Mirshak correlation in the UFSAR-2009 is listed for the "Maximum Reactivity Insertion Accident" as 1.28 and 1.18 at BOC and EOC, respectively.
- Reactor power, with natural circulation cooling flow, shall not exceed 10 kW. Operation in this mode shall only be made with a core that has been previously analyzed and shown to be within the envelope of conditions described in the UFSAR.

2.2 Neutronics Model

Monte Carlo N-Particle code (MCNP) [3] shall be used for all neutronics analyses. The MCNP model used for AFMS analysis implements the same model as the UFSAR with increased discretization in the fuel elements and geometrical changes adapted to match current core characteristics. The MCNP model used for AFMS analysis includes geometrical changes to match current facility conditions and implements increased discretization in the fuel elements. The MCNP model provided in the repository, the 720-composition model, has been verified against the original MCNP 60-composition model that was used in the UFSAR. Keeping the material compositions, regulating rod position and shim arm angles identical in both models, the reactor's effective multiplication factors (k_{eff}) are calculated as 1.01721 ± 0.00045 and 1.01695 ± 0.00044, for 60-composition model and the 720-composition model respectively. There is negligible difference between the two models (results agree to within 0.03%, which is within the uncertainty of the calculations), ensuring that the deviations in the neutronic analysis method do not result in a departure from the neutronic evaluation method described in the UFSAR, which is used in establishing the design bases or in the safety analysis. Note that this is consistent with the NRC-endorsed NEI guidelines for interpreting 50.59 evaluations [4], particularly the adopted definition for a departure in a methodology where a departure is defined as any change in a method adopted by the UFSAR unless it yields results that are conservative or "essentially the same". Given the 0.03% difference between the effective multiplication factors of the 720-composition model and the 60-composition model, the results are deemed "essentially the same", and the input model updates are not considered a departure in the methodology. See **ModelComparison** folder in the SVN repository (see 2.12) for MCNP input files and scripts used for data processing.

Following OFMS, as described in section 1.1, the 720-composition model was used in MCNP 6.2 with the BURN card to calculate the equilibrium inventory of all elements after nine 38-day cycles with a 10-day cooling period in between, following a similar methodology in the UFSAR. Eight 38-day cycles were required for all the elements to have inventories that were completely determined by the code system. Once this inventory was determined, another 38-day cycle was performed to ensure consistency between the sets of calculations. Calculations of inventories were made for four cores: the startup (SU) core, the beginning of cycle (BOC) core, the middle of cycle (MOC) core, and the end of cycle (EOC) core. For the SU cores, the fresh fuel elements contain only ^{235}U, ^{238}U, pure aluminum, and ^{16}O. At the end of nine cycles, the inventory in each element then represents an EOC core. To increase the prediction of cycle and inventories at EOC, quarter-2 (Q2) and quarter-4 (Q4) core were also added between BOC&MOC and MOC&EOC cores, respectively.

2.2.1 Fuel Element Compositions

Fuel elements in an AFMS will have compositions that deviate from nominal equilibrium core loadings and therefore it may be necessary to compute material compositions in some used fuel elements. Specific used fuel elements, such as 6-cycle and 7^{-}cycle fuel elements in an AFMS may have experienced atypical operational histories and hence various burnup levels and/or different compositions.

The approach for the composition calculations for previously used fuel elements (before February 2021) is as follows. For conservatism, the equilibrium concentrations calculated in the MCNP equilibrium model cannot be directly used for calculations. Burnup is linearly related to the ^{235}U content as shown in Figure 2 which can be seen in Figure 3-6 from UFSAR Appendix A. Moreover, for each cycle, approximately 30 grams of ^{235}U fuel is consumed per element (UFSAR 3.2). Ignoring minor decay time or operational differences for simplicity, the material compositions, including the estimated ^{235}U in each used fuel element (in grams), can be calculated by linear interpolation using the decay-corrected equilibrium core concentrations as explained in Appendix B: Fuel Element Composition Estimation Method. This linear approximation will not significantly affect the results since the interpolations are performed within about 38-day intervals, where the composition is linearly varying as shown in Figure 2.

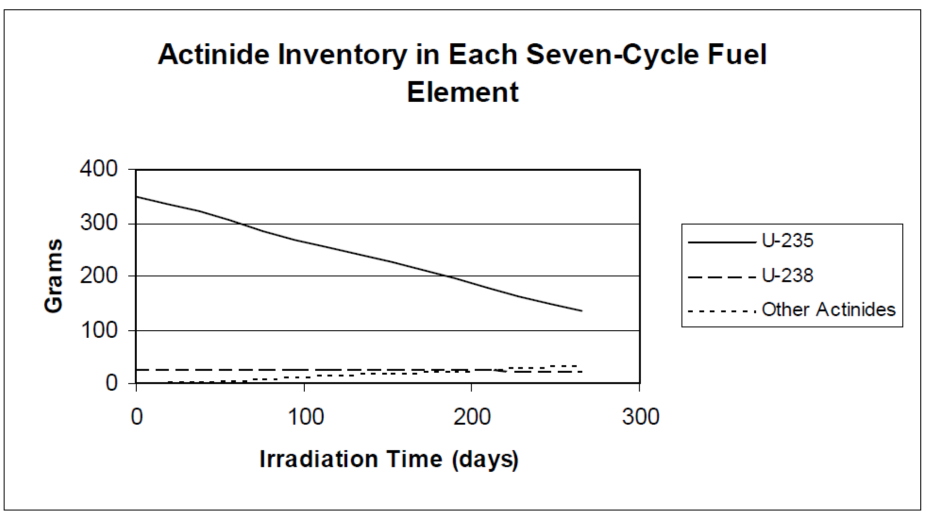

Figure 3-6. Actinide Inventory per Fuel Element as a Function of Irradiation Time for the 7-Cycle Fuel Elements.

Figure 2. The actinide inventory variations in a 7-cycle fuel element as a function of irradiation time. This figure can be found in the UFSAR [1].

2.3 Thermal-hydraulics Model

The design-basis accidents for the AFMS can be analyzed with the system thermal-hydraulics code RELAP5/MOD3.3 [5]. It is important to note that the RELAP models used here are updated versions of the ones used in the UFSAR [1]. The updated model uses a newer version of RELAP5, which comes with improved stability and more features, and it relies on more detailed heat distribution inputs from MCNP (i.e., increased discretization in the fuel elements). This allows for more detailed tracking of the heat transfer behavior throughout the fuel, and it offers a better understanding of the thermal safety margins, namely the local critical heat flux ratio (CHFR) and the onset of flow instability ratio (OFIR). Below is a more detailed summary of the differences between this updated model and the UFSAR model [6]. For convenience, the updated model will be referred to as UM, and the UFSAR model is referred to as SM. Note that the thermal-hydraulics computations are typically performed by Brookhaven National Laboratory (BNL), who developed both the UM and SM.

2.3.1 Heat Structures

Both the UM and SM use hot channels to represent limiting conditions in the NBSR fuel element, but the hot channels are defined differently in the models.

The SM only has one type of hot channel, which is a hot stripe that accounts for the power peaking across the entire width of the plate. The SM assumed a hot stripe multiplier of 1.27, which is then used as a multiplier for the power delivered to the hottest coolant channel. Each coolant channel in the SM is assumed to take ½ of the power from each of the two bounding fuel plates on each side of the channel. The hot stripe channel is a contagious coolant channel that includes the lower core, mid-gap, and upper core.

The UM refines the representation of the hot channels. It accounts for two types of hot channels including the SM's hot stripe, and the hot cell channel. The hot channels are used to capture the CHFR, and OFIR, biasing on the local heat flux and biasing on the integral power along a flow channel. The hot channel represents the fuel stripe that has the maximum local power peaking. The power distributions in the UM are derived from the more-detailed MCNP outputs provided by the 720-composition model used in this work. For each fuel plate, the power at each axial location is given for three equally spaced transverse segments, which yields three vertical stripes within the fuel meat. These stripes span the upper and lower fuel plates separately. The fuel stripe with the maximum power is used for the hottest stripe, and the fuel stripe with the maximum local peaking becomes the hottest cell stripe. With this setup, a hot channel is then assumed to have a fuel plate consisting of three hot stripes for either type of hot channel (hottest stripe channel and hottest cell channel). More details on the heat structures setup can be found in section 8 (Appendix D).

2.3.2 Thermal Safety Margins

It is noted that the UFSAR, the SM and the UM all adopted the same correlations for the calculation of the thermal safety limits. The thermal limit calculation for the DCL will again use the same correlations. More specifically, the UFSAR-2009 uses the Mirshak correlation [7] for calculating the CHFR. As an additional verification, the Sudo-Kaminaga correlations [8] are also used for calculating the CHFR with version updates as discussed in 2.5. The thermal-hydraulics analysis used in this ECN also adopts those two correlations for CHFR calculations. As for the onset of flow instability ratio (OFIR), the RELAP models uses 2 correlations as well, namely the Costa correlation [9] and the Saha-Zuber correlations [10]. The UFSAR-2009 uses Saha-Zuber correlations for OFIR. The new RELAP models uses additionally the Costa correlation along with Saha-Zuber correlations.

The SM computes the thermal safety margins with control directives within the RELAP5 code system. The UM computes the thermal safety margins in post-processing using a utility program built in FORTRAN to extract outputs from the restart-plot file of the RELAP5 output and together with the RELAP5 heavy water library to compute the safety margins as a function of time. Note that this post-processing can be done in any code/utility as long as it uses the same heavy water thermophysical properties as RELAP5. The post-processing approach provides the flexibility to survey the RELAP5 result for all heat structures and locate the most limiting condition in the core at any given time.

2.3.3 Other Updates

The UM contains updated parts of the reactor vessel geometry to better represent the cross-sectional area at different elevations. The closure of control valves (valves in the primary piping leading to the inner and outer plenums inlets) was updated with a physical model that modulates valve flow area as a function of time. The discharge valve at the outlet of the primary pumps was also added to the model alongside a shutdown pump flow loop in the primary system.

The updates in the UM offer significant improvements in the fidelity of the results when compared to the SM. The more accurate representation of the NBSR with the more detailed hot channel groupings provide a more accurate representation of the power distribution in the fuel, and also provides a more conservative approach to capturing the CHFR and OFIR. The introduction of Sudo-Kaminaga correlations adds redundancy to the analysis, while the addition of OFIR correlations enables the detection of the onset of flow excursion during an accident. The thermal safety margins correlations are discussed in more detail in section 2.5.

2.4 Power Distributions

The power distribution shall be computed in MCNP using radiation heating tallies on each fuel meat in the core, where each fuel element is composed of 17 plates with a top and bottom meat section. Each of the meat sections is discretized into 6 axial zones to allow for tracking of the axial heating profile in each of the respective fuel meats. The neutron heating tallies ($F7$) are track length estimates of the fission energy deposited in each of the discretized fuel meat axial zones. The MCNP-provided results for $F7$ are normalized using the full core power ($\dot{Q}_{core}$ in W), the volume and density of the discretized fuel meat section (V_{cell} and ρ_{cell} in units of cm^3 and g/cm^3, respectively), the average number of neutrons produced per fission ($\bar{\nu}_f$, directly obtained from MCNP output), the average energy released per fission ($\bar{E}_f$ in MeV, obtained from table 98 in the MCNP output), and the effective multiplication factor (k_{eff}, from MCNP output). This normalization is demonstrated in the equation below, where the total power ($\dot{Q}$) of each axial zone of the respective fuel meats is computed.

$$\dot{Q} = \frac{(F7)\, V_{cell}\, \rho_{cell}\, \dot{Q}_{core}\, \bar{\nu}_f}{\bar{E}_f\, k_{eff}}$$

The total power of each fuel element in the core is computed by summing the $\dot{Q}$ from each fuel meat section in that element. For this work, MCNP is allowed to sum the contributions from each of the 17 plates in any one fuel element and report the sum for each axial zone, enabling quicker computations of the axial and radial power distributions across the core. To enable comparisons with the OFMS from previous analyses, the power is normalized to the core-wide average fuel half-element power$\dot{Q}_{avg}$, $\left(\frac{20\ MW}{30\ elements \times 2\ sections}\right)$. The half-element power is used because the power is tracked for the top and bottom halves of the fuel elements independently, allowing for separate power distributions for the top and bottom halves of the core.

$$PPF = \frac{\dot{Q}}{\dot{Q}_{avg}}, \qquad \dot{Q}_{avg} = \frac{20\ \text{MW}}{30\ \text{Fuel Elements x 2 sections}}$$

Note that each fuel meat is discretized into 6 axial segments, allowing for detailed $\dot{Q}$ distributions throughout any core loading. Those detailed distributions are then used to set-up accident analysis models in RELAP5 as discussed.

2.5 CHFR and OFIR

The OFIR correlations used in the UFSAR, namely the Costa correlation and the Saha-Zuber correlations [9], [10] are to be used as noted in the UFSAR for evaluating the OFIR for an AFMS. The CHFR computations will also rely on two correlations, namely the Mirshak correlation [7] (1959), and the Sudo-Kaminaga correlations [8] (1998). Note that for the AFMS, the updated 1998 form of the Sudo-Kaminaga correlations should be used instead of the UFSAR's 1993 version, where the most notable update consists of a variation in the 3rd dimensionless critical heat flux term ($q^*_{CHF,3}$) as shown below. Note the inclusion of the $(1 + 3 \times \Delta T^*_{sub,in})$ term in $q^*_{CHF,3}$, which is absent in the 1993 format adopted by the UFSAR. This update also changes the 2nd and 3rd dimensionless mass fluxes (G^*_2 and G^*_3, respectively), which are shown below. This update is relevant because it maintains consistency with current approaches for computing CHFRs in plate-fueled test reactors. For more information on the variables and their expected units, refer to UFSAR. For more information on the Sudo-Kaminaga correlations, refer to the source paper [8].

$$q^*_{CHF,3} = 0.7\,\frac{A}{A_H}\,\frac{\sqrt{W/\lambda}}{\left(1 + {\rho_g/\rho_l}^{1/4}\right)^2}\left(1 + 3.0 \times \Delta T^*_{sub,in}\right)$$

$$G_2^* = \left[140\frac{A}{A_H}\frac{\sqrt{W/\lambda}}{\left(1+{\rho_g/\rho_l}^{1/4}\right)^2}\left(1+3.0\times\Delta T_{sub,in}^*\right)\right]^{\frac{1}{0.611}}$$

$$G_3^* = 0.7\frac{\sqrt{W/\lambda}}{\left(1+{\rho_g/\rho_l}^{1/4}\right)^2}\frac{1+3.0\times\Delta T_{sub,in}^*}{\Delta T_{sub,in}^*}$$

2.6 Core Excess Reactivity

The core excess reactivity is the reactivity feedback due to a complete withdrawal of all shim arms and the regulating rods from a position where the core was critical. This excess reactivity is computed as shown in the equation below.

$$\rho_{excess,out} = \frac{k_{all\ out} - k_{critical}}{k_{all\ out} \times k_{critical}}$$

Per the equation above, $\rho_{excess,out}$ is the excess reactivity of the core with all shim arms withdrawn with the regulating rod, $k_{critical}$ is the effective neutron multiplication factor when shim arms and regulating rod are at the estimated critical positions, and $k_{all\ out}$ is the effective neutron multiplication factor when all shim arms and the regulating rod are fully withdrawn. Note that although it may be intuitive to have $k_{critical} = 1$, the methodology should adopt the same number of decimal places for $k_{critical}$ as adopted for $k_{all,out}$, which will yield a $k_{critical} \approx 1$, but rarely equal to a perfect unity. The analyst should adopt the MCNP output-generated $k_{critical}$ value from the nominal simulation case.

2.7 Shut Down Margin

The current NBSR TS 3.1 [2] limits core loadings that yield excess reactivity greater than 15% $\Delta\rho$, or core loadings that prevent shutdown with the most reactive blade fully withdrawn ("stuck blade criterion"), all movable experiments in their most reactive condition, and the regulating rod fully withdrawn. In other words, the core excess reactivity must be less than 15% $\Delta\rho$ (~$20) and the Shutdown Margin (SDM) must be greater than 0.757% $\Delta\rho$ (~$1) when the most reactive blade is assumed to be stuck (fully withdrawn) and the regulating rod fully withdrawn. Note that the most reactive blade is the one with the highest worth.

To determine if these conditions are met, k_{eff} is calculated under the following conditions: all shims inserted (Shutdown Reactivity), all shim arms withdrawn (excess reactivity), and three of the four shim arms inserted with the other withdrawn (Shutdown Margin, SDM). Note that SDM is computed for each shim arm playing the role of the "stuck blade", which yields 4 SDMs: one for each shim arm being completely withdrawn or stuck. The calculations should be performed at the most reactive state point in the cycle, which is typically (but not necessarily) at startup (SU).

2.8 Shutdown Reactivity at SU

The shutdown reactivity at SU is a relatively straightforward calculation where k_{eff} is determined for all shim arms and the regulating rod inserted and then comparing those values to the nominal criticality value, which yields the reactivity feedback in the equation below.

$$\rho_{excess,in} = \frac{k_{critical} - k_{all\ in}}{k_{critical} \times k_{all\ in}}$$

Per the above equation, $\rho_{excess,in}$ is the excess reactivity of the core at a shutdown state, $k_{all\ in}$ is the effective multiplication factor when all shim is fully inserted, and $k_{critical}$ is the effective multiplication factor when all shim arms are at critical positions.

2.9 Integral Fission Density and Burnup

The fission density provides the number of fissions per unit volume and is therefore directly linked to local power generation and burnup. The integral fission density is the total number of fissions that occurred within a set volume. The NBSR TS specifies that the average fission density for a fuel element shall not exceed 2x 10^{27} fissions/m^3 [2]. The Safety Analysis states that with a burnup of 73% in the 8-cycle fuel elements, the typical fission density is 1.9 x 10^{27} fissions/m^3 [1]. This alludes to the fact that as long as the fuel elements in the OFMS do not exceed 73% burnup, the fission density specification will be met. A more accurate characterization of the fission density limit's correlation to percent burnup is presented in a separate memo [11], which further elaborates based on Technical Specification 3.1.4. The memo states that the maximum burnup allowed is 77.52%, which is considered the set burnup limit for any NBSR fuel element to not exceed technical specification limits.

2.10 Fuel Management Scheme Analysis Method

For each cycle, the uranium consumption, power distributions, core excess reactivity, shutdown margins, shutdown reactivity at SU, integral fission density for each fuel element, and the excess reactivity at the end of cycle (EOC) are calculated and compared to values from the OFMS and the limitations as listed in section 2.1. Figure 3 shows the flow chart of the program that performs the neutronic analysis. The program initially reads a core configuration file to load given core layout and fuel compositions and prepares fuel assemblies, and then it verifies that the total unirradiated uranium and total uranium in the core are less than the limits specified in TS.

Once all fuel assemblies are positioned appropriately in the core per the AFMS, a preliminary run is used to check whether the excess reactivity and shutdown margin values are acceptable. If the AFMS is deemed acceptable, then the critical shim arm angles are determined by a series of MCNP simulations. Upon finding the critical positions, a model is run to simulate the SU (with 1.5 days of burnup at full power). At this point, power distributions, core excess reactivity, shutdown margins, shutdown reactivity at SU, and the integral fission density (particularly at EOC) are computed and compared against Section 2.1 limitations and the OFMS values. The program then repeats the (1) shim arm critical angle determination, (2) burnup (length is determined based on the determined cycle state), (3) data extraction, and (4) comparison to the OFMS steps. Eventually, the routine reaches the EOC, followed by an appropriate cooling period. At the end of the cooling period, the repository of data is processed for the cycle.

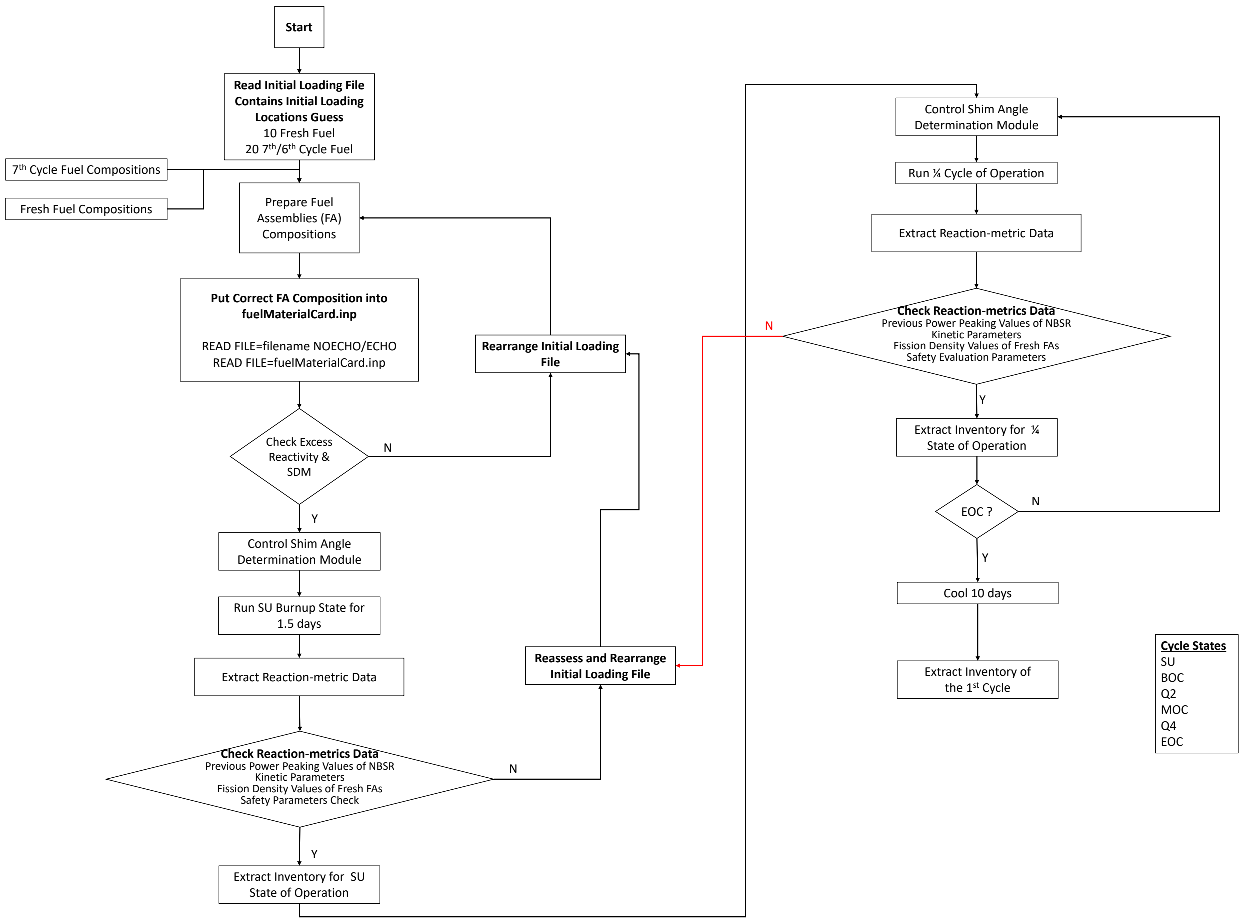

Figure 3. The fuel management scheme analysis flow chart.

2.11 Analysis of Accident Scenarios and Natural Circulation Cooling at Low Power Operation

AFMS loadings deviate from the core loading scheme as described in the UFSAR 4.5.1.1.2 “Fuel Management Scheme”. Several accident scenarios and therefore consequences may be affected by AFMS core loading deviations. Particularly, all accidents shown in Table 1 are required to be re-evaluated for any AFMS core loading. Other accident scenarios given in the FSAR, including “Loss of Primary Coolant” (a major rupture in the cold leg of the primary system is assumed, which leads to draining the reactor core), “Maximum Hypothetical Accident (MHA)”, “Experiment Malfunction” and “External Event” are independent of core loading changes and therefore remain unchanged.

Additionally, Natural Circulation Cooling at Low Power Operation must be analyzed for each AFMS to show compliance with Technical Specification 2.2. Natural Circulation Cooling at Low Power Operation is not an accident scenario but an analysis to show natural circulation at low power operations.

Note that all of the accident scenarios and Natural Circulation Cooling at Low Power Operation conditions are analyzed using the RELAP5 updated model. The misloading accident is the only one that require unique power distributions from corresponding MCNP simulations with the misloaded fuel configuration.

Table 1. The accident sequences to be re-analyzed, and their references in the UFSAR.

	Accident Sequence	Section in the UFSAR
#1	Startup Accident	13.1.2.2.2.1
#2	Maximum Reactivity Insertion Accident	13.2.2
#3	Loss of Offsite Power	13.1.4.1
#4	Loss of Offsite Power with Shutdown Pump failure	13.1.4.5
#5	Seizure of One Primary Pump	13.1.4.2
#6	Throttling of Coolant Flow to the Outer Plenum	13.1.4.4
#7	Throttling of Coolant Flow to the Inner Plenum	13.1.4.3
#8	Misloading of Fuel	13.1.5

Accidents #1 through #7 require a transient run while accident #8 require only a steady-state analysis. The only changes were for the power inputs to the 13 heat structures representing the fuel assemblies. There was no change to the point kinetics parameters. The power inputs are derived from the neutronics analyses (i.e., MCNP simulations: section 2.4). Description of the accident scenarios as corrected and updated to better reflect current plant characteristics can be found in Appendix C.

2.12 Version Control and Quality Assurance

A Visual SVN (www.visualsvn.com) software was previously installed on a virtual windows server which is locally hosted on NCNR servers. The virtual server is backed-up weekly to a physical drive. The SVN server keeps versioned backup copies of the source code and configuration files for critical NBSR systems.

The configuration repository can only be accessed by authorized ROE users using Windows authentication within NIST Active Directory, inheriting all enterprise security features of the NIST network. More details on the NBSR SVN system, user manual etc. can be found in ECN 910.

A new repository is created in the NBSR SVN system for AFMS analysis. Within the repository, there are the following folders

- ***"MCNPComparison"*** folder with MCNP input and output files comparing MCNP input model as used in the UFSAR, namely the 60-composition model, with respect to the 720-composition model used in this ECN.
- ***"MCNPEquilibrium"*** folder contains all files for equilibrium calculations: HEU NBSR MCNP6.2 720-composition model inputs, output files, relevant scripts and a README file explaining script usage
- ***"BNL"*** folder contains "***ThermalHydraulics" folder*** which contain all files relevant for the RELAP code Thermal Hydraulics and accident analysis performed independently by BNL

- ***"FEER"*** folder contains MATLAB scripts for generating the transient dose-rate assessments of NBSR fuel elements, and for post-processing the results from the simulations. The MCNP input files for each run used in generating the self-protection assessments in this document and post-processed results files for each run are also saved in (.mat) format.
- ***"Demonstration"*** folder contains input and output files for the demonstration core loading as described in Section 3 of this report.

2.13 Verification and Validation

During the startup, at low power, core excess reactivity, shim arm worth, shutdown margin, and regulating rod worth will be measured and compared with Technical Specification limitations and bounding conditions listed in Section 2.1.

Required Test #1: The Procedure MP 5.55 - 1/M Approach to Criticality, will be used to experimentally determine an estimated critical position.

Required Test #2: TSP 4.1.2 measurement of Regulating Rod Worth

Required Test #3: TSP 4.1.2 measurement of Bank Shim Worth (at critical)

Required Test #4: TSP 4.1.2 measurement of Individual Shim Worth

Required Test #5: TSP 4.1.2 measurement of Core Excess Reactivity (at beginning of cycle)

Required Test #6: TSP 4.1.2 measurement of Shutdown Margin (at beginning of cycle)

Required Tests number 2 through 6 will be completed during TSP 4.1.2, Core Excess Reactivity/Shutdown Margin Reactivity Worth of each Shim Arm and Reg Rod. This procedure experimentally determines the values for: individual shim arm worth, regulating rod worth, shim bank integral worth, shim bank differential worth, excess core reactivity, shutdown margin, and maximum reactivity insertion speed, meeting the requirements of the above tests.

2.14 Fuel Elements Self-Protection

Due to the AFMS, fuel elements (particularly fresh ones) may have atypical burnup histories, which can create self-protection concerns. Per 10 CFR 73.6, a licensee is exempt from additional security measures if the radioactive material (the fuel elements in this case) has radiation levels more than 100 Rad/hour at a distance of 1 meter from any accessible surface without intervening shielding. A fuel element exposure rate model was developed in MCNP and validated with measurements from 1st and 2nd cycle elements. The model allowed for a study that is discussed with significantly greater detail in a separate report [12], where the results demonstrated that any NBSR HEU fuel element can remain self-protected for more than 42 days (more than 100 days with an extrapolation) if it experienced at least 3 days of burnup in the core at full power. If the AFMS can last around 15 days, the study demonstrates that the freshest fuel elements can remain self-protecting for around 200 days. The analysis also adopted a factor of safety of 2 for all dose rates computed, and therefore the results are considered conservative. The following figure is taken directly from the referenced report [12], and it summarizes the findings of the fuel element self-protection analysis.

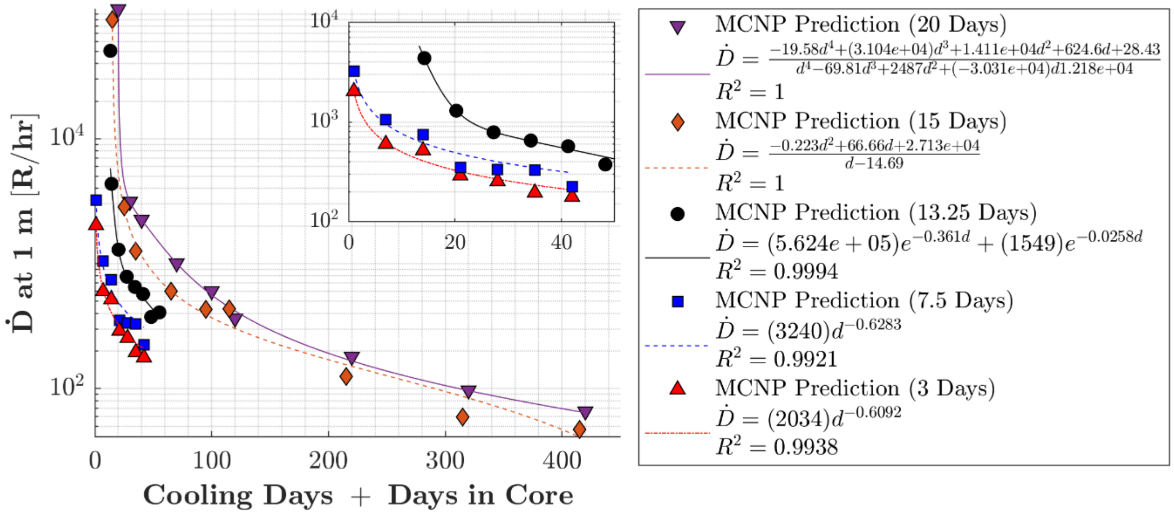

Figure 4. The expected dose rate of a fuel element after full-power burnup for 20, 15, 13.25, 7.5 and 3 days. Note that the prescribed 100 R/hr limit for self-protection aligns with the "10^2" on the y-axis, which is plotted in log-scale. All fits are only expected to be valid within the range of their data points (i.e., within 400 days for the 20-day and 15-day elements, and within 42 days for the other elements).

3 Analysis & Review– Demonstration AFMS

Each AFMS would implement a new ECN to evaluate proposed core loadings following this procedure (**NBSR-0018-DOC-00**) before operational implementation. An example 50-59 evaluation for the hypothetical DCL core loading is also provided in the attachments (DCL50.59.docx). This section provides necessary discussions and results to be included in any such ECN documentation.

For this purpose, a demonstration AFMS core loading is analyzed which introduces ten fresh fuel elements, compared to four fresh fuel elements within a usual core loading. There are twenty 7th cycle fuel elements in the Demonstration Core Loading (DCL) as shown in Figure 5. The 7th cycle elements are numbered as **7001** through **7020** for simplicity. Hence, DCL results in Section 3 are provided for demonstration purposes only and cannot be used for core loading decisions in further ECN's.

	A	B	C	D	E	F	G	H	I	J	K	L	M
1				7005		7009		7011		7012			
2			7017		FF		O		FF		7013		
3		FF		O		7006		7014		O		FF	
4	7001		FF		7007		O		7018		FF		7010
5		FF		O		7004		7008		O		FF	
6			7019		FF		O		FF		7015		
7				7003		7016		7002		7020			

Figure 5: Sample AFMS, Demonstration Core Loading (DCL) scheme

Hypothetical 7th cycle fuel elements numbered from 7001 through 7020, total burnup levels in MWH, without any correction for location-dependent power peaking, decay time as of January 1st, 2023, and the core discharge position for the DCL are listed in the Table 2.

Table 2. The ^{235}U burnup level of the spent fuel elements in the proposed NBSR core for DCL.

Fuel Element Number [a]	Total Burnup (MWh) [b]	Total Decay Time (day)	Position Discharged from the Core
7001	129789.8	1996	E4
7002	129038.4	1939	E4
7003	129054.1	1781	E4
7004	127945.3	1730	E4
7005	126768.3	1681	E4
7006	110241.7	1681	I6
7007	128029.4	1622	E4
7008	127314.8	1622	I4
7009	127083.6	1556	E4
7010	128281.7	1556	I4
7011	127330.5	1503	E4
7012	127335.8	1503	I4
7013	128239.6	1402	E4
7014	120762.0	1402	I4
7015	128917.5	1349	E4
7016	129285.3	1349	I4
7017	130935.4	1225	E4
7018	121261.2	1056	E4
7019	129479.8	743	E4
7020	128717.8	743	I4

[a] Fuel element numbers are arbitrary, and they do not reflect actual fuel data

[b] Total core burnup is listed. Values are for demonstration purposes, and they do not reflect actual fuel data

3.1 Demonstration of Results

Based on the DCL configuration, using MCNP, the core loading power peaking values, core excess reactivity, shutdown margin, uranium consumption, and burnup are computed and compared with UFSAR values. The results are presented in the following subsections.

3.1.1 Power Distributions

Since an NBSR's fuel element contains two separate parts, the bottom section, and the top section, these parts' power generation values for each fuel element section are obtained by using MCNP. The relative power is normalized so that unity represents the average power in a half-fuel element, i.e., 1/60 of the total core power (=1/3 MW). The bold numbers given in Figure 6 and Figure 7 represent the SU and EOC cycle states assembly section's power peaking factors while the normally indicated small numbers shown under the peaking are the UFSAR peaking values.

At the SU cycle state, since the shim arms are surrounded by the top section of fuel elements, the bottom part of the core power peaking values compared to the top part of the core are higher, and the highest peaking locations are determined in Figure 6.

Detailed analyses are performed at the C4 location. Each plate located in the top and bottom parts are divided into 3 longitudinal and 14 axial meshes, and the power generation is calculated via MCNP. As a result of the highest local peaking determination, the highest peaking is found as 3.52 in 1st stripe of plate 17 at the C4 location and the detailed axial relative powers of plate 17's stripes at C4 are also given in Figure 8 as additional information.

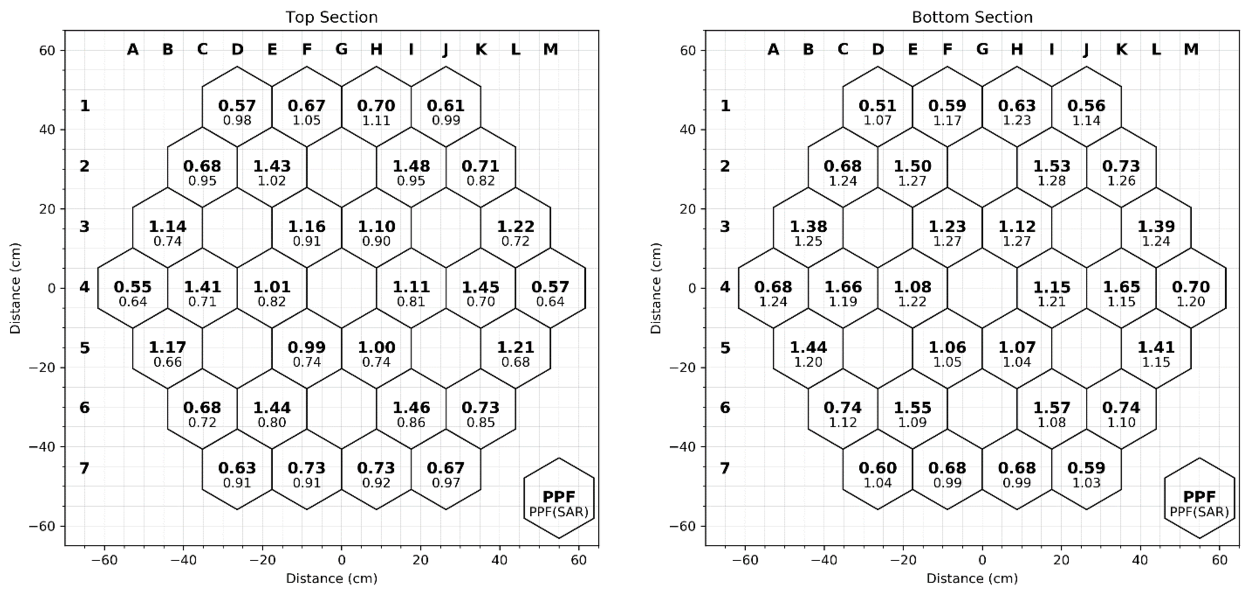

Figure 6. SU Power Peaking Factors in the Fuel Elements.

Similar to the SU cycle state analysis, the highest radial and axial power generation locations are determined in the EOC cycle state. As seen in Figure 8, the highest radial peaking is determined at the K4 location's top fuel section as 1.63. At the same time, the highest local peaking location at the K4 fuel elements is calculated in the 1st plate's bottom stripe as 3.28 from data obtained from MCNP.

Finally, to investigate the fuel misloading accident, the scheme in Figure 9 is simulated to obtain the SU power peaking distributions per Figure 10. Finally, to investigate the fuel misloading accident, twenty different cases were investigated where a fresh fuel element is incorrectly placed in a 7th cycle element location. The loading scheme shown in Figure 9 leads to the highest power peaking and is further evaluated for thermal-hydraulic safety. The misloaded case SU power peaking distributions are shown in Figure 10. Also note that the ECN has the following excel files attached, and they contain the detailed stripe-by-stripe relative powers for DCL and its misloaded cases.

- GeneratedPowerPeakings_SU_DCL.xlsx
- GeneratedPowerPeakings_EOC_DCL.xlsx
- GeneratedPowerPeakings_SU_DCLMisloaded.xlsx

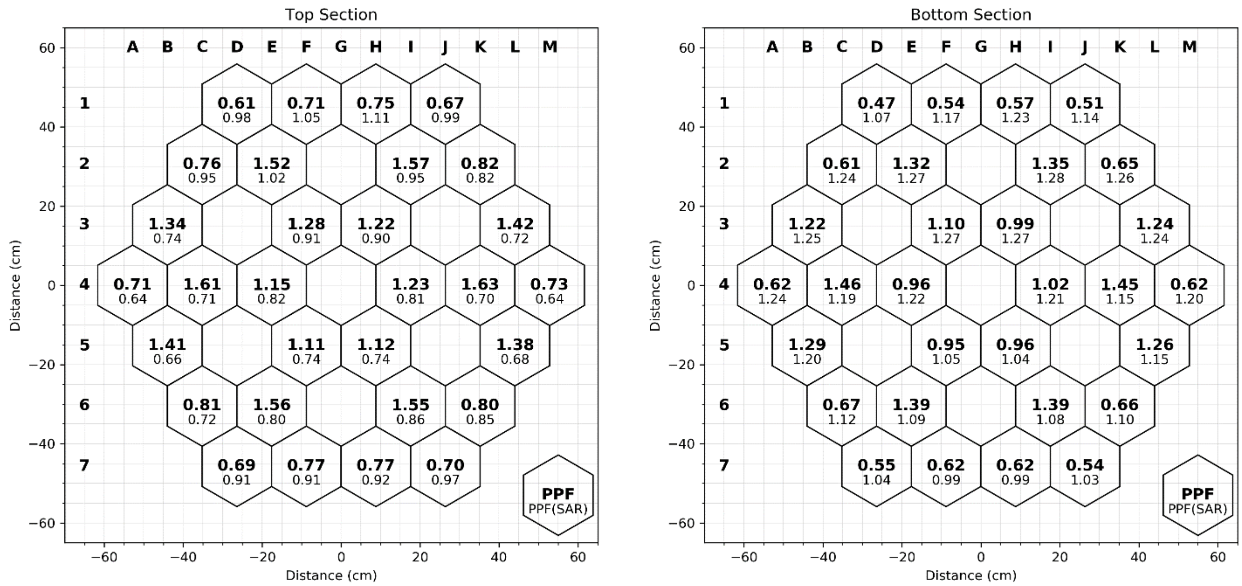

Figure 7. EOC Power Peaking Factors in the Fuel Elements.

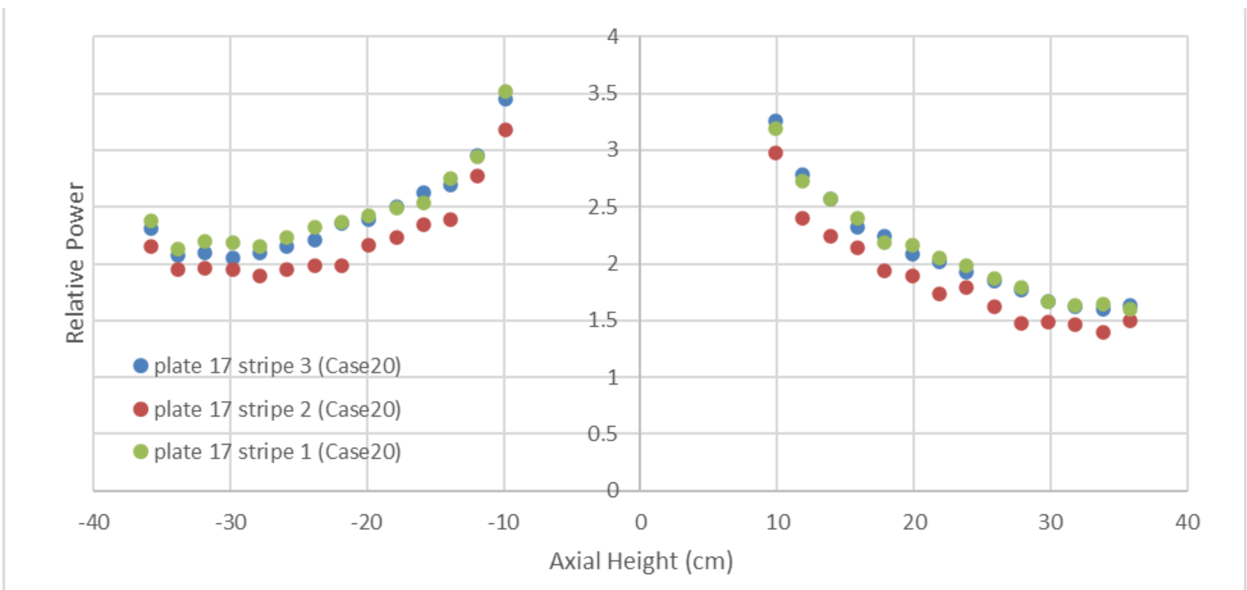

Figure 8. Axial relative power of the hottest stripes in the fuel element at C4 (hottest element).

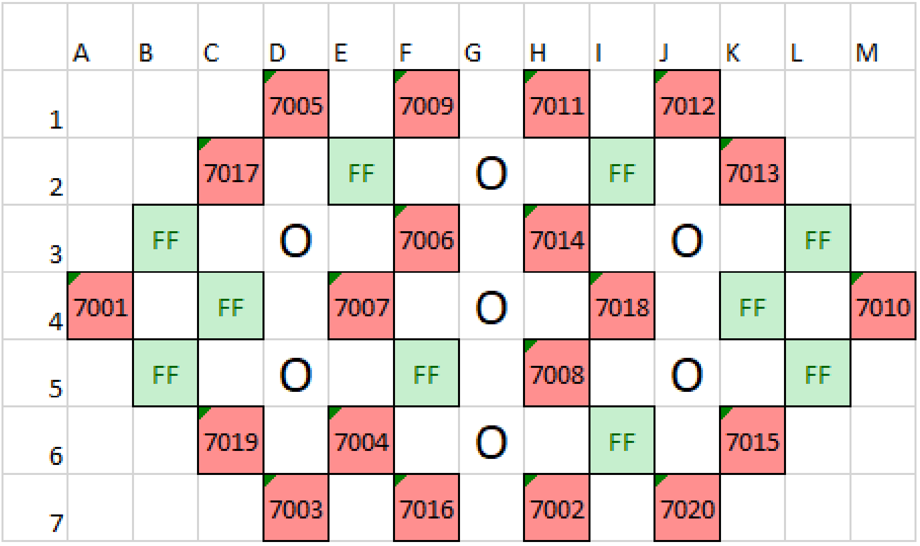

Figure 9. DCL with a misloaded fuel element in F5.

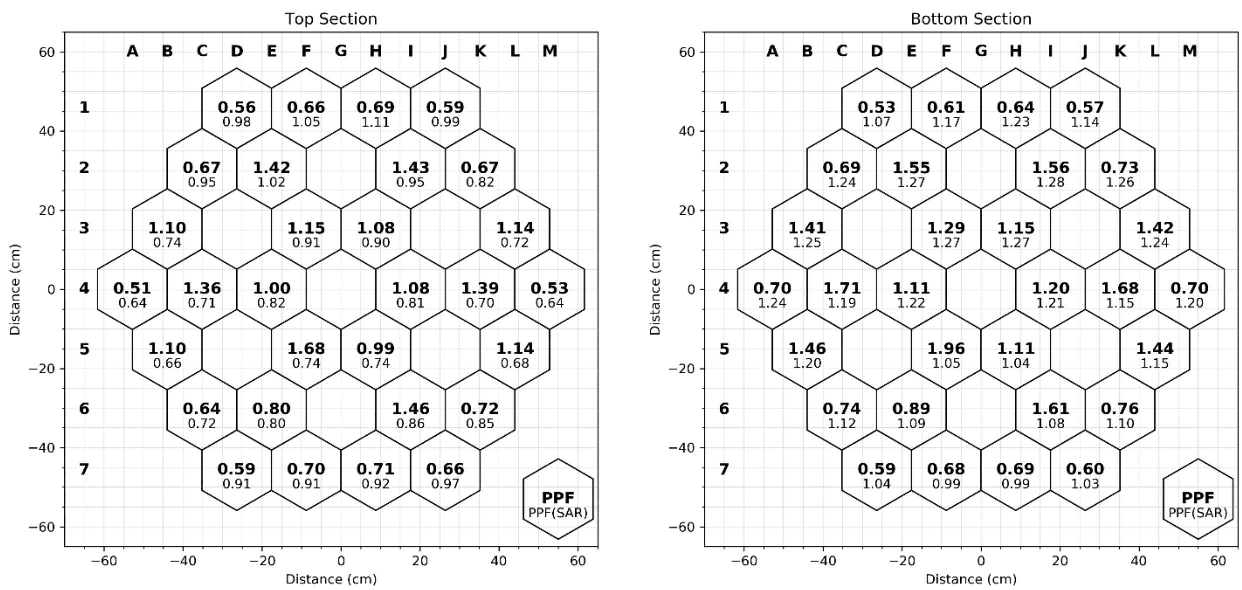

Figure 10. SU Power Peaking Factors in Fuel Element Sections of DCL Misloaded Fuel Element.

3.1.2 Uranium Consumption and Burnup

As mentioned in the NBSR's Technical Specifications (TS), the fuel element's maximum fuel density at the end of the cycle shall not exceed 2.0 x 10^{27} fissions/m^3 for 38 days of cycle length operation for the 8-cycle fuel elements. Because the proposed core is arranged with 7^{th}-cycle fuel elements and fresh fuel elements, the burnup, and maximum fission density must be determined for safe operation. As stated in the previous sentence, the 8th cycle fuel element that completes an operation of 38 days shall not exceed the determined fission density limit given in the TS. However, the proposed fuel cycle (8th cycle of 7th-cycle fuel elements) length is much smaller than the normal operating cycle length of an 8th-cycle fuel element.

Table 3 provides Uranium-235 consumption through DCL for individual fuel elements. The total Uranium-235 consumed within the cycle is about 177.7 grams, and the maximum burnup for an element is 66.33% (in location C2), which is well within TS limitations.

3.1.3 Reactivity Analysis

The NBSR Technical Specification pointed out that the core cannot be loaded such that the excess reactivity will not exceed 15% Δk/k and the NBSR shall not be operated if it cannot be kept shutdown with the most effective shim arm fully withdrawn out of the core. To specify the abovementioned conditions are met, k_{eff} values are calculated under the following conditions:

- All shims and regulating rod inserted (shutdown reactivity),
- All shim arms and regulating rod withdrawn (excess reactivity),
- Three of the four shim arms are inserted with the most reactive arm withdrawn and all moveable experiments in the most reactive condition, and regulating rod is fully out (shutdown margin, SDM)

The calculations were performed at the most reactive state point in the cycle, which is SU with ten fresh fuel elements and no ^{135}Xe poison, and the least reactive state point in the cycle, which is EOC without taken into consideration of ^{135}Xe poison concentration.

3.1.3.1 Core Excess Reactivity at SU

To determine the core excess reactivity limit, which is defined in NBSR Technical Specification, all shims are completely withdrawn from the reactor core to their final position. K_{eff} values are determined from the MCNP model and reactivity values are calculated with k_{eff} values. These results given in Table 4 show that neither the DCL nor the DCL-misloaded core loadings exceed the excess reactivity limit of 15% Δk/k.

In DCL, 4.04% excess reactivity is smaller enough than NBSR's equilibrium excess core reactivity value. Because of that reason, the cycle length will be shorter than the normal operational cycle time. Additionally, 20 7th cycle fuel elements and 10 fresh fuel composition creates more reactivity drop in the time interval between SU and BOC states, that additional reactivity drop also causes more shortened operational time for the proposed core configuration.

The DCL core loading yields a cycle length of 6.8 days. The EOC criticality is determined as 1.00018 with 0.0 reactivity when all the shim arms are fully withdrawn from the core.

Table 3. Uranium consumption for DCL core Loading.

Assembly Location	Initial U-235 Mass (g) [a]	Final U-235 Mass (g)	Burned U-235 Mass (g)	Burnup (%)
A4	123.82	120.01	3.81	65.71
C2	123.71	117.83	5.88	66.33
C6	124.88	120.6	4.28	65.54
F7	125.03	120.97	4.06	65.44
D7	124.99	121.41	3.58	65.31
H7	125.51	121.44	4.07	65.30
K6	125.94	121.67	4.27	65.24
J7	125.7	122.05	3.65	65.13
M4	126.77	122.88	3.89	64.89
K2	127.22	122.96	4.26	64.87
E4	129.53	123.36	6.17	64.75
F5	129.59	123.52	6.07	64.71
H1	128.55	124.69	3.86	64.37
J1	128.11	124.68	3.43	64.38
H5	130.8	124.72	6.08	64.37
F1	128.82	125.16	3.66	64.24
D1	128.9	125.76	3.14	64.07
I4	142.84	136.24	6.6	61.07
H3	143.72	137.19	6.53	60.80
F3	164.25	157.21	7.04	55.08
C4	350	340.97	9.03	2.58
K4	350	340.95	9.05	2.59
I6	350	341.21	8.79	2.51
E6	350	341.25	8.75	2.50
I2	350	341.36	8.64	2.47
E2	350	341.57	8.43	2.41
B5	350	342.14	7.86	2.25
L3	350	342.25	7.75	2.21
L5	350	342.33	7.67	2.19
B3	350	342.6	7.4	2.11
	Total Consumed (g)	177.7	**Max burnup (%)**	66.33%

[a] Initial U-235 amounts listed are set for demonstration purposes only and they do not reflect correct U-235 content for actual fuel elements. Use Appendix-B as a guide for calculating the fuel element compositions as necessary for each fuel element.

Table 4. Core Excess Reactivity for DCL

	Δρ	% Δρ
DCL	0.040372	4.04
DCL SU Misloaded	0.046329	4.63

3.1.3.2 Shutdown Margins (SDMs) at SU and EOC

Table 5 summarizes all SDM, total shim arm worth, regulating rod worth, and excess reactivity for both the SU and EOC cycle states, the core can be maintained in a shutdown condition with the most reactive shim arm (shim arm #2) withdrawn. Together with all shim arms' SDM margin is nearly the same in the reactor core, the total SDM of four shims is -24.82 which is also greater than the maximum allowable excess reactivity of the NBSR. Meanwhile, the regulating rod has around 0.45% additional shutdown margin.

Table 5. Shutdown Margins and Excess Reactivity at SU and EOC (%Δk/k)

Reactivity Parameter		%Δk/k at SU	%Δk/k at EOC	%Δk/k at SU for UFSAR [1]
Shutdown reactivity (all shim arms in)		-24.82	-29.35	-22.6
SDM	Shim 1 out	-15.82	-20.18	-15.1
SDM	Shim 2 out	-15.16	-19.62	-12.4
SDM	Shim 3 out	-16.56	-21.00	-12.8
SDM	Shim 4 out	-15.87	-20.30	-14.6
Total shim arm worth		28.66	29.32	23.7
Excess reactivity (all shim arms out)		4.04	0.02	6.91
Regulating Rod worth		0.45	0.45	0.61

3.1.4 Analysis of Low Power Natural Circulation Cooling and Accident Scenarios

In general, the progression of the DCL accident sequences is similar to the trends presented in the UFSAR. The results of the RELAP5 analyses are summarized in the following tables. Table 6 through Table 9 provide the two thermal limits, the minimum CHFR, and the minimum OFIR, for each accident transient and the Natural Circulation at Lower Power at the SU and EOC conditions, respectively. The Tables also include minimum values of CHFR and OFIR calculated using the Mirshak correlation and the Costa correlation respectively. These two correlations were used in the license renewal UFSAR [1] and they are included in this study to illustrate that using different correlations does not change the conclusions of the accident analysis for DCL. The dominant factor dictating the min CHFR is the reactor power before scram. For the min OFIR, the dominant factor is the power-to-flow ratio (decreasing core flow before reactor scram). For both cycle conditions, the highest power assembly is in the outer plenum. The outer plenum assembly with the highest local peaking is in position C4 and K4 respectively for the SU and the EOC condition. Table 10 and Table 11 provide the thermal limits for the case of the misloading of fuel. In DCL the core is to contain 10 fresh fuel elements in the outer plenum and twenty 7th cycle spent fuel in the rest of the core positions. The scenario for the misloading of fuel is the loading of fresh fuel in the inner plenum core location F5 and the loading of a 7-cycle fuel in the original fresh fuel location of E6. Table 12 shows the Mirshak correlation CHFR Values Required for Given Probability of no DNB (taken from updated FSAR).

An observation from the results in Table 10 and Table 11 is that for DCL, the thermal limits are not worse for the misloading of fuel incidents when compared to operating the cycle normally. This trend in the result is due to the location of the most limiting hot channel. For the CHFR, based on the RELAP5 results, the limiting hot channel is in the outer plenum for normal operation while the hot channel is in the inner plenum for the misloaded fuel incident. Since fuel assemblies in the inner plenum receive more flow than their counterparts in the outer plenum it is then understandable why the minimum CHFR is higher when the reactor is operating with a misloaded fuel in the inner plenum. In the case of the OFIR, the limiting hot channel is in the outer plenum for both normal operation and the misloaded fuel scenario. With a misloaded fresh fuel in the inner plenum the relative amount of power generated in the outer plenum became lower. With a slightly better power-to-flow ratio the hot channel in the misloaded fuel scenario results in a higher OFIR than in the case of normal operation. Similar reasoning can be made to explain the results for the Throttling of Coolant Flow to the Inner Plenum at the SU condition when the Mirshak and the Costa correlations were used to determine the thermal limits. Though flow was reduced to the inner plenum these two correlations determined that the minimum CHFR and OFIR were located in the outer plenum. The result indicates that initial values of the CHFR and the OFIR numbers for coolant channels in the inner plenum are high enough that the reduction in flow does not cause them to decrease below the minimum values calculated for the outer plenum coolant channels. The outcome of the accident analyzed by using the Mirshak and the Costa correlation is thus a manifestation of the combined effect of higher power and lower flow to fuel assemblies in the outer plenum.

Comparing the CHFR and OFIR numbers, the EOC condition is more limiting than the SU condition. This is partly due to the lower rate of scram reactivity when the reactor is at the EOC with all shim arms in the withdrawn position. Also, accidents for DCL are more limiting thermally than the nominal HEU cycles. Another FOM to indicate the margin to safety is the peak cladding temperature. By searching the heat structure nodes for the hot channels over all time steps the node with the peak cladding temperature is identified. The transient cladding temperature for the hot channel fuel node with the peak cladding temperature (PCT) is plotted in Figures 13 and 14 for the startup accident and the loss of offsite power with shutdown pump failure accident, at both the SU and the EOC conditions. In both accidents, the PCT occurs at a location near the exit of the lower core, i.e. at an elevation just below the mid-gap of a fuel assembly. In both accidents, the PCT is much lower than the NBSR technical specification limit of 450°C.

Table 6. The minimum CHFR for DCL at SU conditions.

Case	Minimum CHFR	Time (s)	Hydraulic Node No.
1. Startup Accident			
Sudo-Kaminaga	1.50	16.1	407-02
Mirshak	1.22	16.1	503-15
2. Maximum Reactivity Insertion			
Sudo-Kaminaga	1.55	0.41	407-02
Mirshak	1.32	0.40	503-15
3. Loss of Offsite Power			
Sudo-Kaminaga	1.75	1.40	407-02
Mirshak	1.77	1.39	503-15
4. Loss of Offsite Power with Shutdown Pump Failure			
Sudo-Kaminaga	1.75	1.39	407-02
Mirshak	1.77	1.38	503-15
5. Seizure of One Primary Pump			
Sudo-Kaminaga	2.00	0.81	407-02
Mirshak	1.85	0.81	503-15
6. Throttling of Flow to Outer Plenum			
Sudo-Kaminaga	2.17	59.9	503-15
Mirshak	1.90	21.8	503-15
7. Throttling of Flow to Inner Plenum			
Sudo-Kaminaga	1.93	44.7	203-15
Mirshak	2.20	0.0	503-15
8. Natural Circulation			
Sudo-Kaminaga	>20	0 – 500	403-15
Mirshak	>200	0 – 500	503-15

Table 7. The minimum OFIR for DCL at SU conditions.

Case	**Minimum OFIR**	**Time (s)**	**Hydraulic Node No.**
1. Startup Accident			
Saha-Zuber	2.24	16.1	503-15
Costa	2.74	16.1	503-15
2. Maximum Reactivity Insertion			
Saha-Zuber	2.05	0.41	503-15
Costa	2.54	0.41	503-15
3. Loss of Offsite Power			
Saha-Zuber	1.86	1.40	503-15
Costa	2.96	1.39	503-15
4. Loss of Offsite Power with Shutdown Pump Failure			
Saha-Zuber	1.86	1.39	503-15
Costa	2.96	1.39	503-15
5. Seizure of One Primary Pump			
Saha-Zuber	2.15	0.81	503-15
Costa	3.24	0.81	503-15
6. Throttling of Flow to Outer Plenum			
Saha-Zuber	2.37	21.8	503-15
Costa	3.45	21.8	503-15
7. Throttling of Flow to Inner Plenum			
Saha-Zuber	3.28	15.6	203-15
Costa	4.55	0.0	503-15
8. Natural Circulation – N/A			

Table 8. The minimum CHFR for DCL at EOC conditions.

Case	Minimum CHFR	Time (s)	Hydraulic Node No.
1. Startup Accident			
Sudo-Kaminaga	1.27	16.5	417-02
Mirshak	1.18	16.4	407-02
2. Maximum Reactivity Insertion			
Sudo-Kaminaga	1.38	0.48	417-02
Mirshak	1.32	0.48	407-02
3. Loss of Offsite Power			
Sudo-Kaminaga	1.79	1.42	407-02
Mirshak	1.90	1.40	407-02
4. Loss of Offsite Power with Shutdown Pump Failure			
Sudo-Kaminaga	1.78	1.42	407-02
Mirshak	1.90	1.40	407-02
5. Seizure of One Primary Pump			
Sudo-Kaminaga	2.04	0.82	407-02
Mirshak	1.98	0.81	407-02
6. Throttling of Flow to Outer Plenum			
Sudo-Kaminaga	2.14	21.8	417-02
Mirshak	2.03	21.8	407-02
7. Throttling of Flow to Inner Plenum			
Sudo-Kaminaga	2.05	45.3	207-02
Mirshak	2.34	0.01	407-02
8. Natural Circulation			
Sudo-Kaminaga	>20	0 – 500	507-02
Mirshak	>300	0 – 500	507-02

Table 9. The minimum OFIR for DCL at EOC conditions.

Case	Minimum OFIR	Time (s)	Hydraulic Node No.
1. Startup Accident			
Saha-Zuber	1.86	16.5	417-15
Costa	2.29	16.5	417-15
2. Maximum Reactivity Insertion			
Saha-Zuber	1.61	0.50	417-15
Costa	2.00	0.5	417-15
3. Loss of Offsite Power			
Saha-Zuber	1.60	1.43	417-15
Costa	2.57	1.42	417-15
4. Loss of Offsite Power with Shutdown Pump Failure			
Saha-Zuber	1.59	1.44	417-15
Costa	2.56	1.43	417-15
5. Seizure of One Primary Pump			
Saha-Zuber	1.92	0.83	417-15
Costa	2.91	0.82	417-15
6. Throttling of Flow to Outer Plenum			
Saha-Zuber	2.19	21.8	417-15
Costa	3.21	21.8	417-15
7. Throttling of Flow to Inner Plenum			
Saha-Zuber	3.44	15.6	117-15
Costa	4.72	0.0	417-15
8. Natural Circulation – N/A			

Table 10. The minimum CHFR for DCL during a misloading of fuel accident.

		Start Up	**End of Cycle**
DCL (nominal conditions)	Sudo-Kaminaga	2.63 (407-02)	2.57 (417-02)
	Mirshak	2.20 (503-15)	2.34 (407-02)
DCL (misloaded fuel)	Sudo-Kaminaga	2.71 (107-02)	Not Analyzed
	Mirshak	2.40 (503-15)	Not Analyzed

The number inside the parenthesis indicates the hydraulic node number in the RELAP5 model

Table 11. The minimum OFIR for DCL during a misloading of fuel accident.

		Start Up	**End of Cycle**
DCL (nominal conditions)	Saha-Zuber	3.69 (503-15)	3.82 (417-15)
	Costa	4.55 (503-15)	4.72 (417-15)
DCL (misloaded fuel)	Saha-Zuber	4.07 (503-15)	Not Analyzed
	Costa	5.02 (503-15)	Not Analyzed

The number inside the parenthesis indicates the hydraulic node number in the RELAP5 model

Table 12: CHFR Values Required for Given Probability of no DNB with Mirshak correlation (from UFSAR).

Probability (%)	95	99.5	99.9	99.999
CHFR	1.21	1.33	1.40	1.62

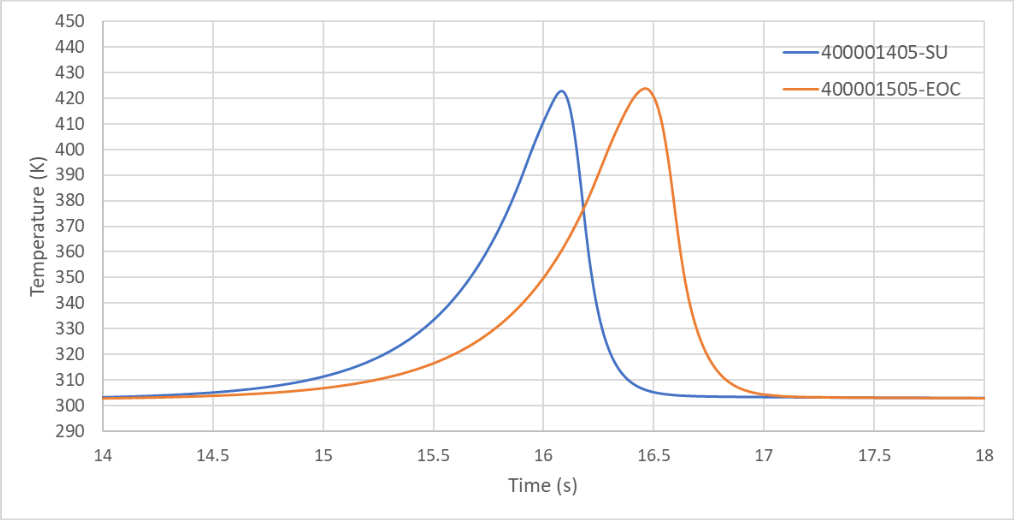

Figure 11. The cladding temperature transient evolution during a startup accident for DCL.

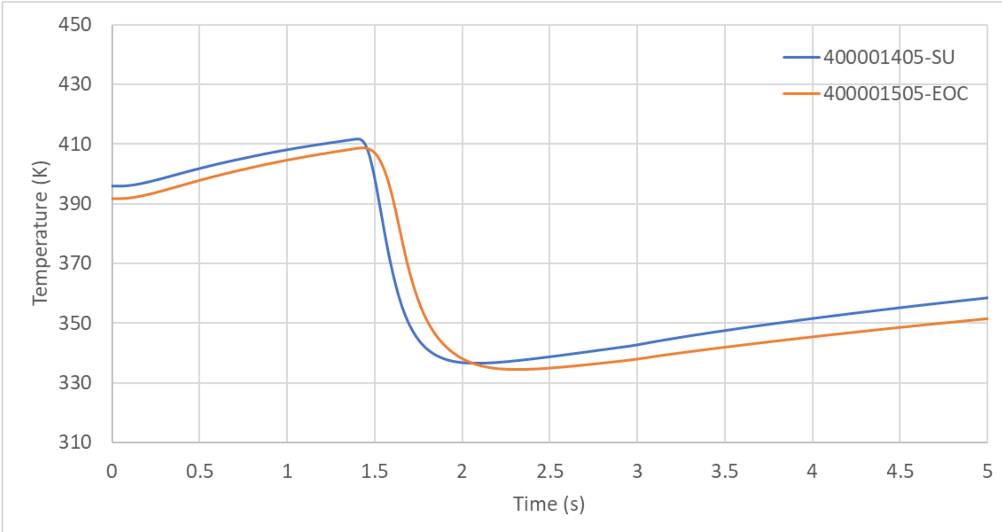

Figure 12. The cladding temperature transient evolution for DCL during a loss of offsite power with shutdown pump failure accident.

3.1.5 Summary of Core Nuclear Analysis

This section provides a summary description as well as results for significant parameters. Results are provided for both the OFMS equilibrium core and proposed core loadings to evaluate the effect of fuel management changes. The results are primarily at the SU condition since this is the most reactive point in the cycle and hence, bounding for most analyses. For some parameters, results are given for both SU and EOC conditions. The latter state point is bounding in transients for which the rate at which the reactor shuts down is important since differential shim arm worth is lowest when the shim arms are withdrawn at EOC. The calculations show that the core can be maintained in a shutdown condition with the most reactive shim arm withdrawn, at any core condition and all movable experiments in their most reactive condition. The kinetic parameters of DCL presented in

Table A– 1 are directly obtained by MCNP by using ENDF/B-VII.1 data [13]. The obtained results are compared with UFSAR [1]. The table shows that there are no significant deviations observed in the total delayed neutron fraction and the prompt neutron lifetime. Therefore, neutron kinetics during normal operation and accident conditions would remain similar to the OFMS. The maximum theoretical burnup is determined as 66.33% (less than the 77.52% burnup limit) which means the proposed cycle does not violate any fission density limit in a fuel element.

3.1.6 Summary of Analysis Results

The DCL involves the introduction of 10 fresh fuel elements, compared to 4 fresh fuel elements within a usual core loading. There are 20 7th-cycle fuel elements in the DCL core loading. Several accident analysis scenarios may be affected by such deviation from the nominal core loading scheme as described in the UFSAR 4.5.1.1.2 “Fuel Management Scheme,”. So, the “Startup Accident”, “Maximum Reactivity

Insertion Accident", "Loss of Offsite Power", "Loss of Offsite Power with Shutdown Pump failure", "Seizure of One Primary Pump", " Throttling of Coolant Flow to the Outer Plenum", "Throttling of Coolant Flow to the Inner Plenum",and "Misloading of Fuel", are re-analyzed for DCL core loading and were found that the proposed core would not more than minimally increase the consequences of an accident previously evaluated in the UFSAR. Additionally, Natural Circulation Cooling at Low Power Operation was analyzed at 100 kW to verify compliance with Technical Specification 2.2.

Detailed analyses are performed in the attached reports, but the significant results can be summarized as follows.

- Cycle length is found to be shorter than the nominal cycle length.
  - No more than 7 days of full power operation at the maximum rated core power of 20 MW.
- At SU, the estimated critical position for the shim arms is -13.2°±2° inserted (27.3° withdrawn from fully inserted position)
- Maximum power peaking at SU is 1.66 (554.1 kW), and it occurs in the bottom half of the C4 fuel element.
  - Axially, the highest local power peaking factor can be found in the 1st stripe of plate 17, with a peaking of 3.52.
- Maximum power peaking at EOC is 1.63, and it occurs in the top half of the K4 fuel element.
  - Axially, the highest local power peaking factor is in the bottom stripe of the 1st plate, with a peaking of 3.32.
- The total Uranium-235 consumed within the cycle is about 177.7 grams.
- Minimum CHFR during Natural Circulation at Low Power is greater than 20.
- Nominal conditions CHFR and OFIR are greater than 2.
- In all postulated accident scenario, the hottest temperature reached by the cladding is less than 430 K (<157 °C), which is well within the cladding temperature limit 450 °C set by TS 2.1.
- In all postulated accident scenario analyzed, the proposed loading scheme was found to not experience departure from nucleate boiling, which is deemed safe and comparable with the current UFSAR.
- The maximum theoretical burnup is determined to occur in the C2 fuel element with 66.4% burnup, which is well within TS 3.1.4 restrictions (less than 77.52% burnup).
- The DCL core loading configuration does not exceed the excess reactivity limits of TS 3.1.2.1 of 15% or the shutdown margin limits of 3.1.2.2 of 0.757%.
- Reactor power, with natural circulation cooling flow, shall not exceed 10 kW. The AFMS core loading has been analyzed and shown to be within the envelope of conditions described in the UFSAR.

3.2 Conclusions

This section described the core loading, namely Demonstration Core Loading (DCL), and demonstrated the required analyses prior to its operation in the NBSR. The main difference between the OFMS and the DCL is the fact that the DCL core loading is composed of twenty 7th cycle elements while the rest of the core is loaded with ten fresh fuel elements. The CHFR is computed with both Sudo-Kaminaga and Mirshak correlations. The OFIR is computed using Saha-Zuber and Costa correlations. NBSR Technical Specifications, which establishes as the safety limit that the fuel cladding temperature shall not exceed 450 °C, which is the minimum temperature at which fuel blistering has been observed. For accident analysis, it is sufficient to ensure that neither a Departure from Nucleate Boiling (DNB) nor Onset of Flow Instability (OFI) occurs. DCL configuration is analyzed for previously evaluated accidents, except for MHA and LOCA.

It is found that the CHFR during the postulated accident transients is relatively lower than that of the OFMS core loading. However, the thermal hydraulic results, including CHFR and OFIR for all postulated accident transients were well within the NBSR Technical Specifications and LSSS. The lowest CHFR was found at the EOC state as **1.27** and **1.18** for the “Startup Accident” using Sudo-Kaminaga and Mirshak correlations respectively. The minimum OFIR was found as **1.59** for the “Loss of Offsite Power with Shutdown Pump failure” using Saha-Zuber correlation. When the Costa correlation is used for the OFIR, minimum was 2 for the “Maximum Reactivity Insertion Accident”. For reference, the lowest CHFR in the updated FSAR-2009, Section 5.3, is listed for the “Maximum Reactivity Insertion Accident” as 1.18 at EOC. In all cases, the maximum clad temperature does not exceed 156.9 °C (430 K) as shown in Figure 11 and Figure 12. Henceforth, the maximum clad temperature is well below the TS 2.1 limit of 450 °C (723 K).

The analyses described in this section provide the technical basis for the safe operation of the NBSR under the DCL and all safety parameters are bounded by the NBSR TS and the UFSAR.

4 References

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5 Appendix A: DCL Neutronic Results

This section provides detail neutronic results for the SCL core loading analysis. Table A-1 and Table A-2 provide delayed and prompt neutron parameters, respectively.

Table A– 1: Delayed Neutron Parameters for the DCL compared with the original fuel management scheme from NBSR UFSAR.

	SU				
	DCL			**UFSAR** [1]	
Group	λ_i (1/s)	β_i	σ	λ_i (1/s)	β_i(%)
1	0.01334	0.00025	0.00002	0.0127	0.000276
2	0.03273	0.00117	0.00004	0.0317	0.001546
3	0.12077	0.00115	0.00004	0.115	0.001364
4	0.30278	0.00259	0.00005	0.311	0.002954
5	0.84953	0.00099	0.00003	1.40	0.000929
6	2.85303	0.00044	0.00002	3.87	0.000189
	$\beta^\dagger = \Sigma\, \beta_i$	0.00659	0.00020	$\beta^\dagger = \Sigma\, \beta_i$	0.007258

	EOC				
	DCL			**UFSAR** [1]	
Group	λ_i (1/s)	β_i	σ	λ_i (1/s)	β_i(%)
1	0.01334	0.00027	0.00002	0.0127	0.000276
2	0.03273	0.00128	0.00004	0.0317	0.001546
3	0.12077	0.00113	0.00003	0.115	0.001364
4	0.30278	0.00246	0.00005	0.311	0.002954
5	0.84954	0.00108	0.00003	1.40	0.000929
6	2.85301	0.00044	0.00002	3.87	0.000189
	$\beta^\dagger = \Sigma\, \beta_i$	0.00666	0.00019	$\beta^\dagger = \Sigma\, \beta_i$	0.007258

† Without photo neutron fraction.

Table A-2: Prompt Neutron Lifetime

	SU	**EOC**
UFSAR [1]	780	810
DCL	723	785

6 Appendix B: Fuel Element Composition Estimation Method

6.1 Fuel Element Compositions

Some fuel elements in an AFMS can be 6th or 7th etc. cycle elements while the rest of the core may be completed with fresh fuel elements at nominal compositions as listed in Table 13. It is significant for the neutronic and safety analysis of any proposed core to use conservative isotopic compositions of the used fuel elements in the AFMS. Hence, the equilibrium concentrations may not be directly used for calculations. Considering the equilibrium core burnup using 38-day cycles at 20 MW, the 6th, 7^{th,} and 8th cycle element cumulative total core burnup values are listed in Table 14 [1]. The burnup levels logged for some of the fuel elements in the NBSR inventory can be different due to cycle length changes, operational adjustments on refueling, etc. Hence, some of the spent fuel elements in the NBSR inventory may experience less, and some more than the equilibrium fuel elements at discharge.

Different burnup and cooling times mean different ^{235}U content, which almost linearly would affect the excess reactivity and power peaking estimates. Furthermore, various cooling times of these elements would result in different radioactive decay times which would change isotopic concentrations. The total ^{235}U in each assembly (g) and isotopic compositions can be estimated by an iterative scheme using the decay-corrected equilibrium core concentrations as explained further below.

Table 13. The fresh fuel isotopic composition.

Isotopes	Mass fraction
^{235}U	32.740%
^{238}U	2.459%
^{16}O	6.358%
^{27}Al	58.443%

Table 14. The equilibrium cumulative burnup for the NBSR MCNP model.

Number of Equilibrium Cycles	Cumulative Burnup for Element [a]
6	109440
7	127680
8	145920

[a] Assuming 38-day cycles at 20MWth same as equilibrium calculations performed in the UFSAR

6.2 Zero-power decay calculations

A zero-power decay calculation should be conducted on the MCNP model equilibrium compositions, from the end of the cycle of the equilibrium state, to simulate the duration that the spent fuel elements stayed in the spent fuel pool. This calculation is indispensable because while residing in the spent fuel pool, ^{239}Pu accumulated in the spent fuel elements through the decay chain

$$^{239}_{92}U \xrightarrow{\beta^-(23.5\ min)} {}^{239}_{93}Np \xrightarrow{\beta^-(2.356\ day)} {}^{239}_{94}Pu, \quad (1)$$

whereas ^{135}Xe varied through the decay chain

$$^{135}_{53}I \xrightarrow{\beta^-(6.6\ h)} {}^{135}_{54}Xe \xrightarrow{\beta^-(9.2h)} {}^{135}_{55}Cs. \quad (2)$$

Both isotopes significantly would impact the reactivity at the startup of any AFMS core loading, which emphasizes the importance of performing the zero-power decay calculation. Figure 13 shows the variation of ^{239}Pu and ^{135}Xe relative isotopic mass in both 7-7 fuel elements as a function of the duration in the spent fuel pool. Because the isotopic mass of ^{239}Pu and ^{135}Xe converged well by the end of the 96 days, the fuel meat isotopic compositions calculated with the HEU NBSR MCNP6.2 720-composition model, after the zero-power decay calculation, were further used as the basis for the derivation of the fuel meat isotopic compositions of the twenty spent fuel elements, although the actual time that each spent fuel element resided in the spent fuel pool was different. In addition, some of the spent fuel elements were discharged early in their respective cycle, resulting in different ^{235}U content.

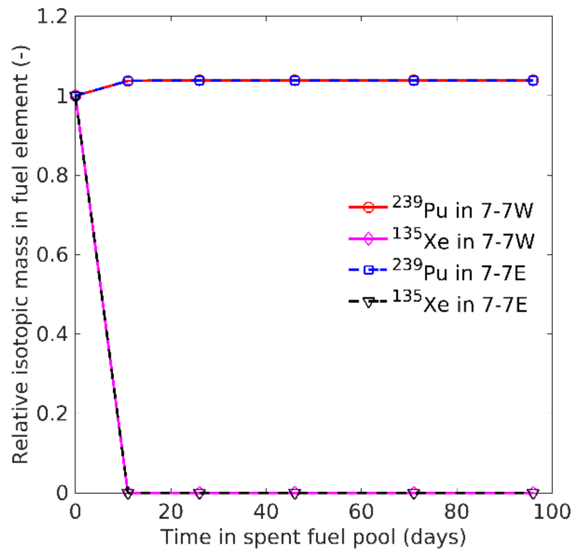

Figure 13: Variation of ^{239}Pu and ^{135}Xe relative isotopic mass in both 7-7 fuel elements as a function of the duration in the spent fuel pool.

6.3 Spent fuel meat isotopic compositions

Due to the fuel shuffling scheme of the NBSR, considering the fuel elements with different burnup is equivalent to considering the fuel elements at different spatial positions. Therefore, the fuel meat isotopic compositions of the spent fuel elements, for instance, a 7^h cycle element, based on cumulative burnup level, can be derived by interpolating those of 7-6 and 7-7 or 7-8 fuel elements. More specifically, the fuel meat isotopic composition of the 7^th cycle spent fuel elements discharged from the west part of the NBSR can be derived from those of 7-6W and 7-7W or 7-8W fuel elements. In contrast, the fuel meat isotopic composition of the 7^th cycle spent fuel elements discharged from the east part of the NBSR was derived from those of 7-6E and 7-7E or 7-8E fuel elements.

The linear interpolation scheme employed is illustrated as follows. The mass of isotope *n* of material *m* of 7^th cycle spent fuel element, X, can be calculated as

$$X_m^n = \text{Burnup}Ratio \cdot \{[7-6W(or\ E)]_m^n - [7-7W(or\ E)]_m^n\} + [7-7W(or\ E)]_m^n, \quad (3)$$

where $[7-6W(or\ E)]_m^n$ means the mass of isotope n of material *m* of 7-6W (or E) fuel element, and

$$BurnupRatio = \frac{Burnup\ X\ -\ Burnup\ of\ (7-7)}{Burnup\ of\ (7-6)\ - Burnup\ of\ (7-7)}. \quad (4)$$

The linear interpolation can be performed for all the isotopes of all the fuel materials of each spent fuel element of the proposed AFMS.

7 Appendix C: Accident Scenario Descriptions and Natural Circulation Cooling at Low Power Operation

7.1 Reactivity Insertion Accidents

7.1.1 Steady-State

The NBSR steady-state operating conditions are summarized in Table 15. This table shows the anticipated range, and the design basis values that are used in the accident and transient analysis. The design basis values generally represent the conservative end of the range. For the thermal-hydraulic analysis, the primary flow is assumed to split between the inner and outer plenums at 145.1 l/s (2,300 gpm) and 403.8 l/s (6,400 gpm), respectively. This flow distribution is based on historic flow measurements that indicated a minimum flow of 6,411 gpm to the outer plenum, where the most limiting fuel element is located. The pressure of the cover gas above the core is only slightly above atmospheric and for simplicity, all analyses are done with the assumption that the pressure in the cover gas region is constant at one atmosphere.

Table 15: Steady-State Operating Conditions

Parameter	Alarm	Normal Operating Range	Design Basis Value
Reactor Power [a]	High	102% (Alarm – servo deviation)	20.0 MW (100%)
	Normal	100% (Normal – servo deviation ±0.5%)	
	Low	98% (Alarm – servo deviation)	
Reactor Water Level [b]	High	164 in (Alarm)	3.81 m (150 in)
	Normal	159 in (Normal)	
	Low	150 in (Alarm)	
Core Inlet Temperature	High	110 °F (Alarm)	316.5 K (110 °F)
	Normal	100 °F (Normal)	
	Low	80 °F (Alarm)	
Primary Flow [c]	High	9,000 gpm [d]	548.9 L/s (8,700 gpm)
	Normal	8,800 gpm	
	Low	8,700 gpm	
Pressure Above Core [e]	High	0.37 psig	101.3 kPa (0.00 psig)
	Normal	0.15 psig	
	Low	0.00 psig	

[a] Rated reactor power is 20 MW. Uncertainty in power is taken into account in the statistical analysis.
[b] Reactor water level is referenced to the bottom of the lower grid plate.
[c] There is no alarm on primary flow. The range of flow is defined by different combinations of main pumps. The ideal and high operating flow is 2,300 gpm to the inner plenum and 6,700 gpm to the outer plenum while the actual total flow with three pumps is 8,800 gpm.
[d] Note that all gpm units refer to U.S. gallons, not imperial gallons.
[e] This is the pressure of the helium cover gas.

The changes in Table 15 compared to UFSAR are:

- The reactor Water Level Normal is 159 in compared to 158 in, and the Design Basis Value for the Reactor Power is listed as 20.0 MW compared to 20.4 MW listed in the UFSAR.

7.1.2 Startup Accident

The analysis of a startup accident uses assumptions that are selected to maximize the reactivity insertion. The reactor is assumed to be initially critical at a power level of 100 W. Contrary to operating procedures and all previous training and experience, the operator is then assumed to withdraw the shim arms steadily without any pause until the reactor is scrammed by a high power level trip. The accident model uses a reactivity insertion rate for the shim arm withdrawal equal to $5x10^{-4}$ Δk/k/s. This rate is greater than the maximum measured and calculated rate at any shim arm initial position (and greater than the rate that moving the regulating rod would produce).

The shim arms are assumed to trip from what would be their initial critical position at full power (19.7º with the HEU fuel at the SU equilibrium core, and 41º with the HEU fuel at EOC, as opposed to in the UFSAR "The shim arms are assumed to remain at the initial critical position of 19º before the reactor trips."). The trip is due to a high power signal. The high power level trip is set to 26 MW (130% of full power). This is conservative because the setting is actually at 125% of power. For conservatism the calculation does not consider any fuel or moderator reactivity feedback and does not consider the period scram which is active below 2 MW.

7.1.3 Maximum Reactivity Insertion Accident

The maximum reactivity insertion accident is analyzed using the RELAP5 point kinetics model. For conservatism the calculation does not consider any fuel or moderator reactivity feedback. For this accident a ramp reactivity insertion of 0.005 Δk/k is assumed to occur in 0.5 s (in the UFSAR "For this accident a ramp reactivity insertion of 0.013 Δk is assumed to occur in 0.5 s."). This amount (0.005 Δk/k) of reactivity is the Technical Specification limit for the reactivity of any experiment.

7.2 Loss-of-Flow Accidents

7.2.1 Loss of Offsite Power

The accident scenario assumes all three primary pumps trip upon loss of offsite power (LOSP) and bounds any loss-of-normal-power accident. The three primary coolant pumps coast down and eventually, the primary coolant flow drops to a value where one or more of the primary coolant flow monitors generates a scram signal. The scram occurs 0.4 s after the flow has reached the trip value taking into account instrumentation delays. The new information compared to UFSAR is "The primary pump discharge valves (DWV-3, 4, and 5) start closing at 1.0 s on the primary pump trip signal. The stroke time of the valves is 2.0 s."

The shutdown pumps (SDPs) operate on both AC and DC power so they would be expected to operate normally during the LOSP. In the present analysis, only one of the two SDPs are assumed to operate to remove the decay power. The valves at the outlets of the SDPs begin to open at 0.7 s with the stroke time of 1.5 s. The normal flowrate of one SDP is 800 gpm (~55 kg/s).

7.2.2 Loss of Offsite Power with Shutdown Pump Failure

The accident is not listed in the UFSAR. The new scenario begins with a loss of offsite power and coastdown of the primary pumps. A reactor trip signal is generated on low primary flow. It is assumed that both shutdown pumps and all of the secondary coolant pumps associated with the heat exchanger coast down, since there has been a failure of all backup power sources. The valves at the outlet of the primary pumps start closing at 1 s on the primary pump trip signal due to the loss of offsite power, whereas the valves at the outlet of the shutdown pumps begin opening at 0.7 s. The opening of the valves at the outlet of the shutdown pumps provides the primary system with a path through which the coolant can flow when natural circulation is established. However, significant natural circulation through the entire primary loop is not expected because the elevation of the heat exchangers (heat sink) is lower than that of the core (heat source).

A RELAP5 simulation of this process is followed until the fuel reaches a relatively stable temperature when it is being cooled in a pool boiling mode. The large inventory of water in the core, inner emergency cooling tank, and upper plenum will then be involved in a very gradual warm-up. It will take a much longer time (several hours) for the bulk water temperature to reach the boiling point, allowing time for shutdown cooling to be restored.

7.2.3 Seizure of One Primary Pump

It is assumed that through some failure, such as a faulty bearing, the rotor of one pump suddenly becomes locked. Because of its momentum, coolant flow through the primary loop will decrease over a finite time interval until a one-third flow reduction is achieved. Since the RELAP5 model lumps all three pumps into one effective pump, the seizure of one of the pumps is modeled by an instantaneous step reduction in the pump speed to two-thirds of full speed. This is conservative since the flow with only two pumps operating would actually be more than two-thirds of full flow.

7.2.4 Throttling of Coolant Flow to Outer Plenum

In this accident scenario, the flow control valve DWV-1 is assumed to close in 60 s, reducing the flow through the outer plenum and generating a reactor trip signal 0.4 s after the flow reaches the low flow trip point of 4,700 gpm (297 l/s). The complete closure of the flow control valve isolates the lower plenum of the outer core and cuts off the supply of forced coolant flow. The RELAP5 simulation shows that since all coolant channels in the fuel elements in the outer core share the same inlet and outlet plenums, closed loop recirculation flow paths are established between hotter and cooler coolant channels in the outer core. Buoyancy induces upflow through the hotter coolant channels, while downflow through the cooler channels completes the closed flow loop. The recirculation flow removes heat from the fuel elements by natural convection.

7.2.5 Throttling of Coolant Flow to Inner Plenum

In this accident scenario, the flow control valve DWV-2 is assumed to close, decreasing the flow through the inner plenum and generating a reactor trip signal 0.4 s after the flow reaches the low flow trip point of 1,200 gpm (75.7 l/s). The 8-inch flow control valve has a stroke time of 30 s. The complete closure of the flow control valve isolates the lower plenum of the inner core and at the same time cuts off the supply of forced coolant flow. The RELAP5 calculation shows that since all coolant channels in the fuel elements in the inner core share the same Inlet and outlet plenums, closed loop recirculation flow paths are established between hotter and cooler coolant channels in the inner core. Buoyancy induces upflow through the hotter coolant channels, while downflow through the cooler channels completes the closed flow loop. The recirculation flow removes heat from the fuel elements by natural convection.

7.3 Misloading of Fuel

The fuel for the NBSR is subject to stringent quality control to ensure that there will be no "leaky" elements that could release fission products into the primary cooling system. In addition, if any element were to leak, the fission products would be detected immediately, and the faulty element would be identified and removed. This has only happened once in the operating history of the NBSR, and there were no releases to the atmosphere. The releases to the primary coolant were small, and the normal water treatment system quickly removed all traces of activity once the element was removed.

Four separate scenarios involving mishandling of fuel were extensively analyzed in (NBS, 1980) and shown to present no significant risks. These accidents were: a refueling accident involving a dropped element; dropping of a fuel element into the storage pool; dropping of a heavy object onto the fuel rack in the storage pool; and dropping of the spent fuel cask during a shipping operation.

The fuel misloading accident is analyzed assuming a fresh fuel element (FE) was inserted into an incorrect location. The power level in the misloaded element is analyzed to determine whether thermal limits would be exceeded. To perform this analysis, one fresh fuel element would be placed in each position (where a used fuel element is located) in the core and the fuel element that should have been placed in that location. The radial power distributions were calculated at SU when fresh fuel is available and power peaking has its largest value.

7.4 Natural Circulation Cooling at Low Power Operation

This is not an accident scenario but an analysis to show natural circulation at low power operations. No changes to existing UFSAR evaluations. UFSAR Section 4.6 and the Technical Specifications 2.2-(4) limits reactor power when natural circulation is available. *"Reactor power, with natural circulation cooling flow, shall not exceed 10 kW."* RELAP5 calculations are performed to simulate operation at low power without forced-flow cooling. In order to demonstrate the conservatism in this Technical Specification, the analysis are performed at a power level of 100 kW.

The calculation starts with zero reactor power and zero primary flow in the system. The initial temperature of the primary coolant is set at 43.4°C. The secondary flow in the primary heat exchanger is assumed to be at an arbitrarily low value of 1 kg/s. The reactor power is then ramped linearly from zero to 100 kW in 60 seconds. From that point on the reactor power is maintained constant till the end of the simulation at 500 s.

8 Appendix D: Development of Power Inputs for the RELAP5 Heat Structures

Outputs from the DCL neutronics analysis were used to generate the power input for the RELAP5 model. RELAP5 accepts power for the heat structures representing the fuel plates in the form of power fractions for each fuel node. In the RELAP5 model, each fuel plate is segmented into 14 equal axial nodes. The power fraction of all fuel nodes in the core sums to unity. The neutronics output provided two kinds of output, an assembly-by-assembly power distribution and a detailed plate by plate power distribution for the hot assembly. Each fuel plate is subdivided into three evenly spaced vertical stripes and power distribution normalized to the core average is given for each stripe at 14 axial locations. The following discussion summarizes the processing of the power distribution data into power fractions for the heat structures in the RELAP5 model.

Table D-1 lists the various heat structures (representing fuel plates) in the RELAP5 model and their corresponding hydraulic channels. The arrangement of -heat structures and coolant channels is show in Figure D-1.

Total number of fuel nodes in the RELAP5 model = 14x2x17x30 = 14280. There are six (6) hot fuel plates (single plates), HTSTR- 1000, 1100, 2000, 4000, 4100, and 5000. The remaining heat structures represent multiple fuel plates, and they are HTSTR- 2100, 3000, 5100, 6000, 7000, 8000 and 9000.

The single fuel plate heat structure is used to represent two types of hot plate in the RELAP5 model, hottest cell and hottest stripe. The distinction is to provide conservative conditions for the determination of the minimum CHFR and the minimum OFIR. The hot plates transfer power to their associated coolant channels. The two types of hot plates are determined from the stripe power for each pair of lower and upper fuel plates. The hottest cell is the fuel node with the maximum power fraction in a lower-upper fuel stripe. The hottest stripe is the stripe with the highest power fraction.

For DCL at the start up condition (SU) the fuel assembly (FA) with the highest power is C4 in the outer plenum. In C4 the plate with the highest power (sum of lower and upper) is plate 17. The stripe with the maximum local peaking is stripe 3 of plate 17. The stripe with the highest power (sum of lower and upper) is stripe 1 of plate 17. For the end of cycle condition, the hot FA is in the outer plenum in position K4. It is noted that the upper half of FA K4 has the highest power among all 48 fueled sections of the fuel assembles in the outer plenum. In K4 the plate with the highest power (sum of lower and upper) is plate 1. The stripe with the maximum local peaking is stripe 1 of plate 1, which also happens to be the stripe with the highest power (sum of lower and upper).

Table D-1. Description of Heat Structures

Inner Core		
HTSTR #	**Coolant Channel #**	**Description**
1000(1)	103 & 107	Power distribution along a fuel stripe which includes the hottest cell in the inner core.
1100(1)	113 & 117	Power distribution along a fuel stripe which produces the highest total stripe power in the inner core.
2000(1)	203 & 207	Power distribution along a fuel stripe which includes the hottest cell in the inner core. The power distribution is the same as that for HTSTR 1000.
2100	213 & 217	Power distribution for the remaining 32 fuel plates of the hottest fuel element containing HTSTR 2000. Thirty-two fuel plates (16 lower plates and 16 upper plates) are modeled.
3000	303 & 307	Power distribution for the remaining 5 fuel elements in the inner core. Power of HTSTR 1000 and HTSTR 1100 are subtracted from the power of these remaining 5 fuel elements. One hundred and sixty-six fuel plates (83 lower plates and 83 upper plates) are modeled.
Outer Core		
HTSTR #	**Coolant Channel #**	**Description**
4000(1)	403 & 407	Power distribution along a fuel stripe which includes the hottest cell in the outer core.
4100(1)	413 & 417	Power distribution along a fuel stripe which produces the highest total stripe power in the outer core.
5000(1)	503 & 507	Power distribution along a fuel stripe which includes the hottest cell in the outer core. The power distribution is the same as that for HTSTR 4000.
5100	513 & 517	Power distribution for the remaining 32 fuel plates of the hottest fuel element containing HTSTR 4000. Thirty-two fuel plates (16 lower plates and 16 upper plates) are modeled.
6000	603 & 607	Power distribution for the first five fuel elements among the remaining 23 fuel elements in the outer core. Power of HTSTR 4000 and HTSTR 4100 are subtracted from the power of these five fuel elements. One hundred and sixty-six fuel plates (83 lower plates and 83 upper plates) are modeled.
7000	703 & 707	Power distribution for the next six fuel elements among the remaining 18 fuel elements in the outer core. Two hundred and four fuel plates (102 lower plates and 102 upper plates) are modeled.

8000	803 & 807	Power distribution for the next six fuel elements among the remaining 12 fuel elements in the outer core. Two hundred and four fuel plates (102 lower plates and 102 upper plates) are modeled.
9000	903 & 907	Power distribution for the remaining six fuel elements in the outer core. Two hundred and four fuel plates (102 lower plates and 102 upper plates) are modeled.

(1) Explanation of heat structure power is given only for a single stripe but the same power distribution along the stripe is applied to the two adjoining stripes of the same fuel plate for conservatism.

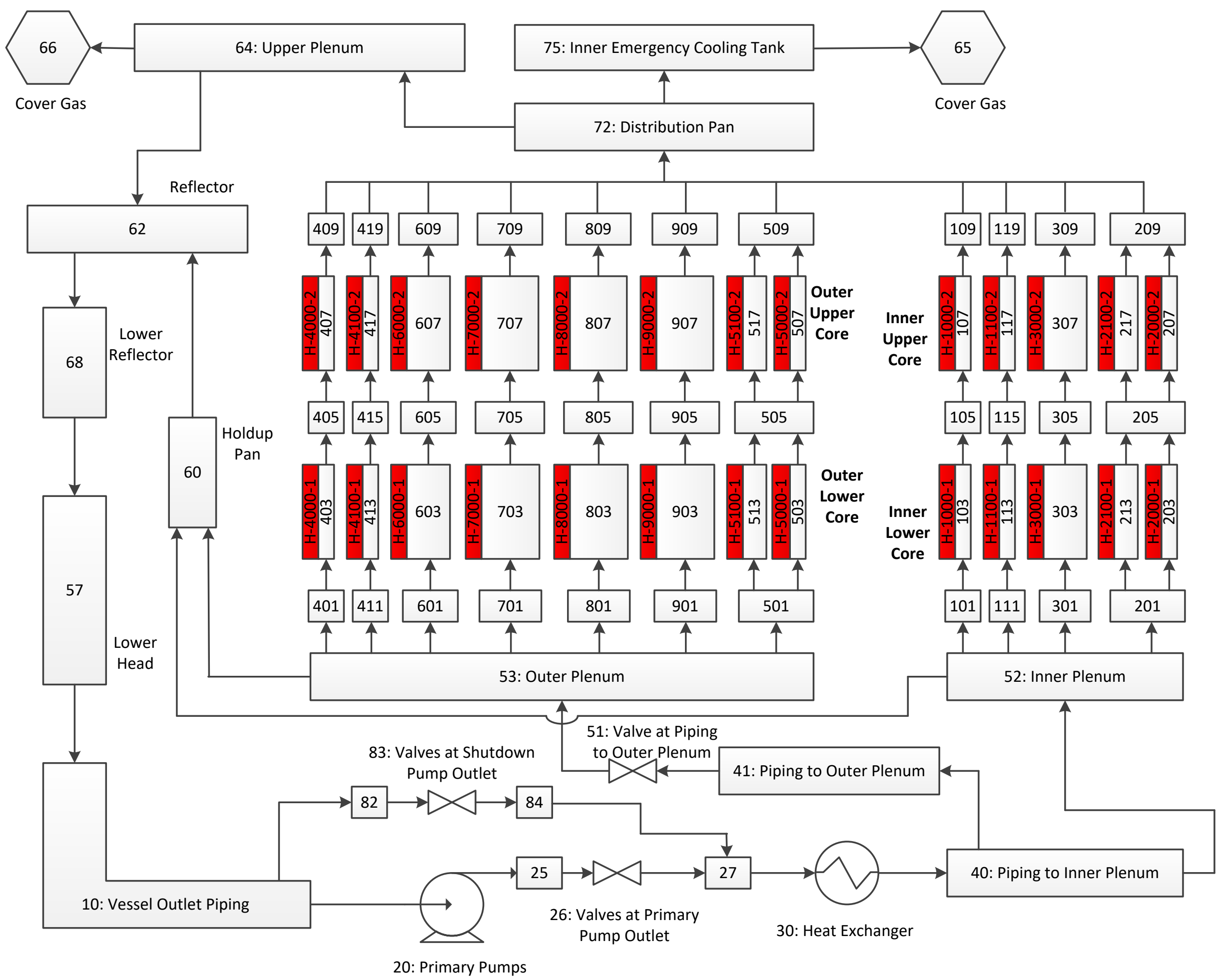

Figure D-1. Node diagram of the RELAP5 model for the NBSR

The DCL power distribution output only provided data for the two hot FAs, C4 for SU and K4 for EOC. There is no data for the power distribution for other FAs. In order to satisfy the node structure of the RELAP5 model shown in Figure 1 there is a need to create the power fractions for the hot plates in the inner plenum. This is done by using the relative power distribution of the 51 (17x3) stripes (lower together with upper) for C4 or K4 and applying them to the hot assembly in the inner plenum. The axial shapes of the hot plates are based on the shapes of their respective stripes. For the other lumped plates, the averaged shape of assembly C4 or K4 is used (averaged across all plates/stripes in C4 or K4 at any given axial location).

Summary of assumptions used to calculate the power fractions for the heat structures:

- C4 (at SU) or K4 (at EOC) is the hot assembly in the outer plenum.
- The hot stripe for local peaking is plate 17 stripe 3 for C4 and plate 1 stripe 1 for K4.
- The hot stripe for maximum stripe power (lower + upper stripe) is plate 17 stripe 1 for C4 and plate 1 stripe 1 for K4.
- The two hot channels will use the axial shape from 1) and 2) respectively.
- The inner plenum hot assembly is determined from the assembly-wise power distribution.
- The hot plates in the inner plenum use the axial shapes of equivalent plates in the outer plenum.

For the SU condition, the power of the hot plates in the inner plenum is based on the relative power of the stripes for plate 17 of assembly C4. For the EOC condition, the power of the hot plates in the inner plenum is based on the relative power of the stripes for plate 1 of assembly K4.

For the rest of the channels in the RELAP5 model, the axial shape is based on the averaged axial shape for C4 (SU) or K4 (EOC) since they are the only assemblies with data to calculate the axial shape. The lumping of multiple FAs to form average channels was done by ranking the FAs in each plenum by power and taking groups of 5 or 6 assemblies to form an average channel. The approach is the same as discussed in the UFSAR[1].

A similar process was used to construct the power fractions for the misloading of fuel accidents. In that case, the hottest FA was in the inner plenum and the data for FA F5 was used to derive the power fractions for the hot FA in the outer plenum. The power distribution in FA indicated that plate 1 stripe 1 is the stripe with the maximum local peaking and also the stripe with the maximum power.