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SUBJECT **RESPONSE SUBMITTAL 1 TO THE REQUEST FOR ADDITIONAL INFORMATION FOR TECHINCAL SPECIFICATION AMENDMENT 19 – USE OF 6% ENRICHED FUEL**
License No. R-120
Docket No. 50-297

Please find enclosed a partial response to the Request for Additional Information for the License Amendment Request related to the use of six percent enriched fuel (TAC NO. MF6088). Responses to the Request for Additional Information questions 1 through 16, except question 4, are being submitted. The response to question 4 will be submitted separately.

If you have any questions regarding this amendment or require additional information, please contact Andrew Cook at (919) 515-4602 or atcook@ncsu.edu.

I declare under penalty of perjury that the forgoing is true and correct. Executed on 18 December 2015.

Sincerely,

Ayman Hawari

Ayman I. Hawari, Ph.D.
Director, Nuclear Reactor Program
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Enclosures: Submittal 1 – Response to Request for Additional Information
Attachment 1 – Safety Analysis Report Updates Related to the Request for Additional Information

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION FOR LICENSE AMENDMENT REQUEST FOR THE USE OF SIX PERCENT ENRICHED FUEL (TAC NO. MF6088)

SUBMITTAL 1

NORTH CAROLINA STATE UNIVERSITY
LICENSE NO. R-120; DOCKET NO. 50-297

DECEMBER 18TH, 2015

1. Appendix A, Section 1.2, of the NCSU LAR (ADAMS Accession No. ML15076A020) states, in part, "[t]he reactor has been operated under eight core configurations for 1541 MWd with a core average burn-up of less than 5 GWd/MTU and a corresponding maximum fuel burn-up and assembly average burn-up of no more than 15 GWd/MTU and 10 GWd/MTU, respectively." NCSU Technical Specification (TS) 5.1.b., states, "Total burn-up on the reactor fuel is limited to 20,000 MWD/MTU." Explain how the maximum burn-up of the six weight percent fuel will be managed and where and how this will be documented.

RESPONSE

Technical Specification 5.1(b) [1] specifies the limit for burnup for the PULSTAR Fuel as the NRC accepted burnup limit of 20,000 MWd/MTU given in NUREG 1537, Part 1, Chapter 14, Section 3.1(6)(d) [2] and ANSI/ANS-15.1-2007 Section 5.3 [3].

To satisfy TS 6.4 [1], SAR Chapter 12 [4] and NUREG 1537, Part 1, Section 12.3 [2], the PULSTAR maintains surveillance procedures that detail the processes for checks, calibrations and inspections including the burnup of the reactor fuel.

Burnup is tracked to the fuel assembly level and is given by the following equation:

$$Burnup_{assembly} = \frac{MWdays_{assembly} \times Flux\ Factor_{assembly}}{MTU_{assembly}}$$

where;

$MWdays_{assembly} =$	the actual MW-days of operation for each fuel assembly
$Flux\ Factor =$	the measured average flux of the fuel assembly divided by the measured average flux of the entire core. Data for each fuel assembly is measured using flux mapping procedure PS-8-03-1 *Core Flux Mapping* or may be obtained from the computer models discussed in Appendix A [5] of the LAR [6].
$MTU_{assembly} =$	the metric tons of uranium in each assembly calculated using the actual mass of the UO_2 from the fuel pin quality assurance documentation provided by the manufacturer.

$$U_{assembly} = UO_{2_{assembly}} \times \frac{[(238 \times (100 - e)\%) + (235 \times e\%)]}{[(238 \times (100 - e)\%) + (235 \times e\%)] + (16 \times 2)}$$

where;

$U_{assembly} =$	the calculated mass in kilograms of uranium in each fuel assembly
$UO_{2\,assembly} =$	the mass in kilograms of UO_2 listed in PULSTAR fuel quality control documentation for each fuel assembly
$e =$	U-235 enrichment, either 4% or 6%

Therefore;

$$U^{4\%}_{assembly} = UO_{2\,assembly} \times 0.88143$$

$$U^{6\%}_{assembly} = UO_{2\,assembly} \times 0.88140$$

$$MTU_{assembly} = \frac{U_{assembly}}{1000}$$

Alternatively, burnup may be calculated using the validated computer models discussed in Appendix A [5] of the LAR [6].

The burnup of the reactor fuel is currently tracked by the PULSTAR Surveillance procedure PS-4-07-2 – *Reactor Fuel Burnup*. This procedure tracks and documents the burnup for each fuel assembly. This surveillance procedure has been updated to incorporate tracking of the six percent enriched assemblies.

2. NCSU TSs 3.1.e., 3.1.f., and 3.2.b., express limits on maximum worth of a single fuel assembly, total nuclear peaking factor in any fuel assembly, and excess reactivity, respectively, such that the reactor can be shutdown at all times and that the Safety Limits will not be exceeded. Explain how the proposed allowable core configuration loadings will be controlled to prevent the insertion of a six weight percent fuel assembly into a grid plate location that is not allowed.

RESPONSE

It is important to note that this license amendment does not seek to change or alter any Technical Specification other than TS 5.1(a) [1] which will permit the use of six percent fuel. Therefore all new core configurations must satisfy all current Technical Specification [1] requirements, including but not limited to:

3.1.e maximum worth of a single fuel assembly shall not exceed 1590 pcm

3.1.f, total nuclear peaking factor in any fuel assembly shall not exceed 2.92

3.2.a minimum shutdown margin of 400 pcm

3.2.b maximum excess reactivity of 3970 pcm

3.2.d maximum reactivity insertion rate of control rods not greater than 100 pcm/second (critical region only).

New core configurations will be simulated using the verified and validated computer model described in Appendix A [5] of the LAR [6] to confirm that the core configurations meet all requirements. As discussed in Section 4.3 and shown in Table 4.2 of Appendix A [5] of the LAR [6], the computer model can reliably predict core behavior and identify core configurations that will satisfy all Technical Specification requirements. Only after the computer model described in Appendix A [5] verifies that all Technical Specifications are satisfied will the new core configuration be loaded and commissioned as detailed in RAI #13.

Consistent with NUREG 1537, Part 1 Chapter 14 Section 6.4 [2]. ANSI/ANS-15.1-2007 Section 6.1.3 [3] and 6.4 [3], 10 CFR 50.36 and 50.54, and TS 6.1.4 [1] and SAR [4], all fuel movement activities are performed under the supervision of the Designated Senior Reactor Operator using approved operational procedures.

Fuel handling procedure NRP-OP-301 *Reactor Fuel Handling* details the controls and safeguards that are in place to ensure that fuel assemblies are loaded into correct grid plate locations for each approved core configuration. The movement of fuel assemblies follows the exact order as detailed in NRP-OP-301 *Reactor Fuel Handling*, Appendix B - *Fuel Movement Schedule*. Each fuel assembly number and grid location is confirmed by the fuel handling team prior to, during and after the movement of each individual fuel assembly. The movement of each assembly is certified as complete and accurate by the reactor operator and documented on Appendix B- *Fuel Movement Schedule* of the fuel handling procedure NRP-OP-301 Reactor Fuel Handling.

Refer to RAI #3 for details on fuel assembly identification.

3. The “fuel comparison” provided in the NCSU LAR states that the NCSU four weight percent fuel is physically and visually identical to the six weight percent fuel assemblies obtained from Buffalo Materials Research Center (BMRC) at the State University of New York at Buffalo. NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” Part 2, Chapter 14 states, “All conditions that provide reasonable assurance that the facility will function as analyzed in the [safety analysis report] SAR should be in the technical specifications.” Please provide a TS that will ensure proper placement of the fuel assemblies or explain how positive identification will be provided so the correct fuel assembly is handled and positioned within an allowable grid plate location.

RESPONSE

The PULSTAR fuel assemblies are positively identified as detailed in Section 4.3.6 of Specifications for PULSTAR Fuel Assemblies APR-1 Revision 5 [7]:

> “Each fuel assembly shall have a serial number engraved on two of the four box sides in letters at least 2 inches high. The lettering shall be 0.01 inches deep. The method of marking shall not alter the requirements of the specification. The mark will still be legible after use in the reactor.”

To distinguish between the four percent fuel assemblies and six percent fuel assemblies the fuel assemblies are numbered sequentially but are of different number sequences. In addition, each fuel assembly has markings that make it possible to identify the assembly without removing it from the core grid plate.

In accordance with facility procedure *NRP-OP-301 Reactor Fuel Handling,* fuel assemblies are positively identified when they are removed from the core grid plate and prior to inserting into the core grid plate by all members of the fuel handling team consisting, at a minimum, of a Senior Reactor Operator, Reactor Operator and Certified Fuel Handler. All fuel movements are thoroughly documented throughout the fuel movement process.

4. Appendix A, Section 3.1, of the NCSU LAR describes the general purpose Monte Carlo N-Particle code (MCNP6) model used to evaluate the steady-state neutronic characteristics of the PULSTAR reactor. The guidance in NUREG-1537, Part 2, Section 4.2, “Reactor Core,” and Section 4.5, “Nuclear Design,” states, in part, that the applicant should present all design information and analyses necessary to demonstrate that the core can be safely operated. Please provide the following information:

   a) Discuss the details/assumptions made on how the fuel assembly was modeled. Explain how the fuel assembly cells (material/geometry) were treated and tracked during the depletion calculations.
   b) Discuss whether the effects of manufacturing tolerances for the fuel assembly were considered in the analysis.
   c) Address how the statistical variations were addressed in the calculations. (For example, was the average value of 10 runs with varying random number generator utilized?)
   d) Discuss how the model statistics (e.g., minimizing the calculation uncertainty) were addressed and verification of fission occurring were checked for in each cell.
   e) Explain how the MCNP6 calculated statistical uncertainty was applied to the core reactivity parameter and power peaking results.
   f) Provide justification for why the 15 percent margin added to the calculated peaking factor for mixed enrichment cores provides sufficient confidence that the observed peaking factor will not exceed the limit in TS 3.1.f., given that the experimental margin was determined from uniform enrichment experiments only.
   g) The uncertainty margin is the difference between the experimental measured results and the MCNP6 calculated result.
      1) Explain how the MCNP6 code utilizing the ENDF/B-VII cross-sections libraries was validated and the associated uncertainty margin was addressed.
      2) Explain whether other benchmark cores were utilized, other than the initial fresh fuel core, to validate the core model and to calculate the uncertainty margin. If so, was the use of other benchmark cores documented in a separate report as part of the Verification and Validation of the MCNP6 code?
   h) In the summary of parameter comparison of measured vs. calculated historic core configurations, the measured vs calculated parameters are relatively in good agreement, except for core configuration 8. Explain whether there are any unaccounted measurement or calculation errors that could be affecting the results for core configuration 8.
   i) Describe the predicted core reactivity parameters and power peaking behaviors for the core configuration using six weight percent fuel assemblies for the end of core burn-up calculations.
   j) Discuss the verifications of model geometry definition and input data as well as configuration changes for specific core analyses (i.e., fuel shuffling locations) that were performed.
   k) Discuss the extent computer scripts were used to shuffle fuel assembly locations vs. manual data entry.

RESPONSE

This question will be addressed in a separate submittal.

5. The bases for NCSU TS 3.1.e., indicates that the single fuel assembly worth limit is set such that the safety limit is not exceeded during a postulated fuel loading accident as presented in Section 13.2.2.1, of the NCSU SAR, dated September 4, 1995. Provide an evaluation showing that the assumptions leading to this limit remain valid for a mixed core using six weight percent fuel assemblies.

RESPONSE

The calculations in Section 3.2.3.5.4, Section 3.2.3.5.5 and Section 13.2.2.1 of the SAR [4] with regard to pulsing and step reactivity insertions are based on the results of the 6% fuel tests completed at Buffalo. Refer to PULSTAR Pulse Tests WNY-017 [8] and WNY-023 [9]. The results of the Buffalo fuel tests were completed with full six percent core configurations and are the bounding assumptions for the analysis of the four percent enriched NCSU core configurations.

The following equations, detailed in Section 3.2.3.5.4 of the SAR [4], can be used to calculate the total energy release of the core for a reactivity transient. The equations are based on a step input of reactivity which would occur during the postulated fuel loading accident.

$$E_{total} = \frac{F_S \times \rho_0}{b}$$

Where,

$E_{total} = total\ energy\ released\ in\ a\ pulse\ (MW \cdot sec)$

$\rho_0 = excess\ reactivity\ above\ prompt\ critical\ (\$)$

$b = constant\ of\ proportionality$

$F_S = Symmetry\ factor$

The constant b equals the compensating reactivity. In the PULSTAR core this compensating or shutdown mechanism is provided by the doppler effect in the UO_2 fuel.

$$b = \frac{\alpha_F}{C}$$

Where,

$\alpha_F = doppler\ coefficient$

$C = \ heat\ capacity\ of\ the\ UO_2\ fuel$

The expression for E_{total} can be rewritten as:

$$E_{total} = \frac{F_S \times \rho_0 \times C}{\alpha_F}$$

From the In-hour equation

$$\tau = \frac{l^*}{\rho_0}$$

Where,

$l^* = \ neutron\ generation\ time\ (sec)$

$\tau = stable\ reactor\ period\ (sec)$

During the initial startup testing for the Standard Core Startup Test 4.4, (15 March 1973), the ratio of β/l was determined and using a β_{eff} value of 0.0073, the value of l^* was measured to be 26.8 microseconds. This is consistent with the β_{eff} value of 0.0073 and l^* value of 30 microseconds that was generated by the computer model described in Appendix A [5] of the LAR [6] for mixed core configurations.

The peak power for a pulse can be obtained from the following equation:

$$P_{max} = P_0 + \frac{\rho_0^2}{F_S \times b \times l^*} = P_0 + \frac{E_{total}}{F_S \times \tau}$$

If the pulse is initiated from a low power level then P_0 is negligible compared to P_{max} so

$$P_{max} = \frac{E_{total}}{F_S^2 \times \tau}$$

The above relationships were used to calculate the pulse characteristics for the six percent core configurations of the BMRC PULSTAR. The calculated results compared well with the measured data as detailed in the PULSTAR Pulse Test Reports WNY-017 [8] and WNY-023 [9].

Technical Specification 3.1.e [1] limits the maximum fuel assembly worth to 1590 pcm. This limit remains unchanged for this license amendment. Figure 4.6 of Appendix A [5] of the LAR [6] lists the maximum worth of a six percent fuel assembly for an acceptable core configuration as the fuel assembly located in grid location F4 of Reflected Core No. 9-2 and has a worth of 1051 pcm. Shown below is the calculation for the total energy resulting from the step reactivity insertion of this fuel assembly during the postulated fuel loading accident scenario described in SAR 13.2.2.1 [4]

$$\rho_0 = \rho - \beta_{eff} = 0.01051 - 0.00733 = 0.00318\, {}^{\Delta k}/_{k}$$

The mass of UO_2 for mixed enrichment core configurations remains unchanged at 359 kg.

$$C_{core} = C_{UO_2}(@100°\mathrm{F}) \times M_{core} = 250\,\frac{watt \cdot sec}{kg \cdot °\mathrm{C}} \times 359kg = 8.98 \times 10^4\,\frac{watt \cdot sec}{°\mathrm{C}}$$

The doppler coefficient for Reflected Core No. 9-2 has been calculated to be -1.64 pcm/°F (-2.95 pcm/°C) by the computer model as shown in Table 4.3 of Appendix A [5] of the LAR [6].

$$\alpha_F = 2.95 \times 10^{-5}\,\frac{\Delta k}{k°\mathrm{C}}$$

The symmetry factor F_S is a function of the pulse shape. The BMRC PULSTAR used a value of $F_S = 2.0$ for a complete six percent enrichment core while the NCSU PULSTAR has used a value of $F_S = 2.1$ for a complete four percent enrichment core. For the mixed enrichment core calculations the more conservative value of 2.1 will be used.

$$E_{total} = \frac{F_S \times \rho_0 \times C}{\alpha} \cong \frac{2.1 \times 0.00318\, {}^{\Delta k}/_{k} \times \left(8.98 \times 10^4\,\frac{watt \cdot sec}{°\mathrm{C}}\right)}{2.95 \times 10^{-5}\,\frac{\Delta k}{k°\mathrm{C}}}$$

$$E_{total} \cong 20.3\ MW \cdot sec$$

If the transient is initiated from a low power then,

$$P_{max} = \frac{E_{total}}{F_S^2 \times \tau} \cong \frac{20.3\ MW \cdot sec}{(2.1)^2 \times \frac{26.8 \times 10^{-6}\ sec}{0.00318\, {}^{\Delta k}/_{k}}} \cong 544\ MW$$

Figure 3-20 of the SAR [4] depicts the measured E_{total} and as can be seen this is below the $E_{total} = 58\ MW \cdot sec$ original pulse safety limit (based on DBNR) and is consistent with previous measured pulse data in the SAR [4]. Figure 3-20 also shows the originally predicted total energy release which deviates from the measured values due to originally overestimating the doppler coefficient for the four percent enrichment core. The measured doppler coefficient of -1.6 pcm/°F for the four percent core configurations are nearly identical to the -1.5 pcm/°F for six percent core configurations measured at the BMRC PULSTAR [10].

The total energy release from a step insertion is largely dependent on the compensating reactivity (i.e. the shutdown mechanism) which for the PULSTAR is provided by the doppler effect of the UO_2 fuel. The calculated doppler coefficient (ranging from -1.58 pcm/°F to -1.68 pcm/°F) for the mixed enrichment core configurations are nearly identical to the -1.66 pcm/°F doppler coefficient calculated for the four percent enrichment core configurations and consistent with the measured value of -1.6 pcm/°F. It is also consistent with the doppler coefficient of -1.5 pcm/°F for the six percent enrichment core configurations at the BMRC PULSTAR [10]. Therefore the mixed enrichment core configurations are consistent with analysis in Section 13.2.2.1 of the SAR [4] for step reactivity insertions therefore the analysis remains valid.

6. The bases for TS 3.2.d., indicates that the limit on rate of reactivity insertion of the control rods is set such that the energy pulse from a postulated startup accident (as described in NCSU SAR, Section 13.2.2.2) is significantly less than the nominal original design pulse for the reactor core. Provide an evaluation showing that the assumptions leading to this limit remain valid for a mixed core using six weight percent fuel assemblies.

RESPONSE

As detailed in Section 13.2.2.2 of the SAR [4], a start-up accident provides the following assumptions as bases:

1. Console operator has failed to return the Linear Power Channel range selector to the most sensitive or mid-scale position. This error makes it possible for the Over-Power SCRAM to occur only once power reaches 120% of the 1 MW range.
2. Three controls rods are withdrawn in gang and the gang rate at which reactivity is inserted in the core is at the maximum TS 3.2.d [1] limit of 100 pcm/sec.
3. The power level is assumed to be 1 milliwatt at the time the reactor reaches critically.

The first and third assumptions are unaffected by core enrichment. Regarding the second assumption above, Table 4.2 of Appendix A [5] of the LAR [6] shows that the reactivity insertion rate [62 pcm/sec to 68 pcm/second] of the control rods for acceptable mixed enrichment core configurations are identical to the four percent enriched cores and are well below the limit of 100 pcm/sec, therefore this assumption is still valid for mixed enrichment core configurations.

The Section 13.2.2.2 of the SAR [4] states that the shortest period which could result from a ramp rate of reactivity insertion of 100 pcm/s was calculated to be no less than 29 milliseconds. As discussed in Section 4.3 of Appendix A [5] of the LAR [6] and in RAI #5, the core kinetic parameters used in this calculation, including β_{eff} and l^*, of the four percent core configurations and the mixed enrichment core configurations are identical therefore this calculation for the shortest period remains unchanged.

Given that the above assumption remain unchanged the results of the start-up accident do not change due to mixed core configurations and the analysis in Section 13.2.2.2 of the SAR [4] is still valid.

7. Provide updates to NCSU SAR Table 1-1, “Comparison of PULSTAR reactors,” (NCSU and BMRC) to reflect the current NCSU core configuration and proposed six weight percent fuel.

RESPONSE

SAR Table 1-1 [4] has been revised as requested. See attachment.

8. Section 3 of the NCSU SAR provides the current NCSU core configuration. Provide the proposed six weight percent fuel mixed core configuration information that:

a) Updates the core history including the use of beryllium reflector elements and the planned introduction of six weight percent fuel in a mixed core configuration.
b) Updates the operational reactivity requirements for the PULSTAR as shown in SAR Table 3-2.
c) Updates NCSU SAR, Section 3.2.2.4, “Fuel,” data for the six weight percent fuel elements planned for used in the proposed mix core configurations.
d) Updates NCSU SAR, Section 3.2.3.3, “Core Configuration,” to reflect use of beryllium reflectors.

RESPONSE

a) SAR Section 3.2.3.3 [4] has been revised as requested. See attachment.

b) SAR Table 3-2 [4] has been revised as requested. See attachment.

c) SAR Section 3.2.2.4 [4] has been revised as requested. See attachment.

d) SAR Section 3.2.3.3 [4] has been revised as requested. See attachment.

9. NUREG-1537, Part 2, Chapter 13, “Accident Analyses,” states, in part, the applicant should provide information and analyses demonstrating that all potential accidents at the reactor facility have been considered and their consequences adequately evaluated. Provide justification that the NCSU SAR analyses of transients, etc. are not significantly impacted by six weight percent fuel mixed core configuration.

RESPONSE

The Section 13 of the SAR [4] classifies potential accidents into two categories: non-excursion and excursion.

The non-excursion accidents are:

- Loss of Primary Flow

  The consequences of this accident are dependent on the hot channel fuel centerline temperature which is a function of thermal power and peaking factors in the core. The maximum thermal power of 1000 kW [11] as specified by the license and the maximum peaking factor, F_Q , of 2.92 as specified by TS 3.1.(f) [1] remain unchanged.

  The Section 4.3 and Table 4.2 of Appendix A [5] of the LAR [6] shows that all technical specifications including maximum peaking factors will be satisfied with acceptable mixed enrichment core configurations, therefore the analysis for loss of primary flow accidents remains valid.

- Waterlogging

  Pulsing is no longer permitted for the NCSU PULSTAR Reactor therefore failure of a waterlogged fuel pin during pulsing is not applicable. It should be noted, however, that if a defect in the cladding should occur, then the fuel pin clad failure scenario applies and is discussed in RAI#14.

- Loss of Pool Water

  This accident is dependent on the hot channel fuel centerline temperature which is a function of thermal power and peaking factors in the core. The maximum thermal power of 1000 kW as specified by the license and the maximum peaking factor, F_Q , of 2.92 as specified by TS 3.1.(f) [1] remain unchanged.

  The Section 4.3 and Table 4.2 of Appendix A [5] of the LAR [6] shows that all technical specifications including maximum peaking factors will be satisfied with acceptable mixed enrichment core configurations, therefore the analysis for loss of pool water accidents remains valid.

- Fuel Pin Clad Failure

  This accident is discussed in RAI #14.

- Heat Exchanger Pressure Boundary Breach

  This accident is a function of the radionuclide inventory of the pool water. The radionuclide inventory is dependent on the neutron flux and pool water chemistry both of with remain unchanged for mixed enrichment core configurations. Therefore the analysis for heat exchanger pressure boundary breach remains valid.

- Failed Fueled Experiment

  All experiments must satisfy Technical Specifications 3.2(a), 3.2(e), 3.2(f), 3.7 and 3.8 [1] which details specific requirements for fueled experiments. The analysis in the SAR [4] assumes complete failure of the entire permissible fission product inventory of a fueled experiment. The fission product inventory of the fueled experiment is a function of limitation specified in TS 3.8 [1] and not a function of the enrichment of the reactor fuel and therefore is not affected by mixed enrichment core configurations. Therefore the analysis for failed fueled experiment accidents remains valid.

The excursion accidents are:

- Fuel Loading Accident

  This accident is discussed in RAI #5.

- Start-up Accident

  This accident is discussed in RAI #6.

- Pulse Rod Fails to Return

  Pulsing is no longer permitted for the NCSU PULSTAR Reactor therefore this scenario no longer applies.

- Experiment Failure

  This accident is discussed in RAI #15.

- Control Rod Failure

  Failure of a single control rod to drop after a SCRAM will not result in a hazard to reactor operations since positive shutdown of the core is assured by:

  TS 3.2.a [1] The shutdown margin, with the highest worth scrammable control rod fully withdrawn with the shim rod fully withdrawn, and with experiments at their most reactive condition, relative to the cold clean critical condition, is greater than 400 pcm.

  Table 4.2 of Appendix A [5] of the LAR [6] shows that for acceptable mixed enrichment core configurations all Facility Operating License [11] and Technical Specification [1] requirements are satisfied including shutdown margin requirements. Therefore the analysis for control rod failure remains valid.

- Cold Primary Coolant Slug

  This accident is discussed in RAI #16.

- Fuel Storage

  This accident is discussed in RAI #10 and #11.

All credible accident scenarios are addressed in the SAR [4] and remain valid for mixed enrichment core configurations as detailed in Appendix A [5] of the LAR [6] and in the responses to specific RAIs listed in this document. New or different types of accidents are not created with the use of mixed enrichment core configurations.

10. NCSU currently has a license to possess, but not use the six weight percent fuel assemblies. NUREG-1537, Part 1, Section 9.2, "Handling and Storage of Reactor Fuel," states, in part, "[t]he applicant should discuss briefly the methods that ensure the prudent control of fuel." Describe the location and conditions under which six weight percent fuel assemblies will be handled and stored. Include discussions of fuel handling and storage, the bases of related technical specifications, including inspections, testing, and surveillance and applicable administrative controls and procedures. [*The discussion should not contain safeguards information or a separate response should be submitted in accordance with 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements."]*

RESPONSE

Consistent with NUREG 1537, Part 1, Chapter 9.2 [2] and ANSI/ANS-15.1-2007 Section 5.4 [3], ANSI/ANS-15.21 [12], 10 CFR 50.68, the Technical Specification TS 5.3 [1] requires that fuel be stored such that k_{eff} is not greater than 0.9 for all conditions of moderation and reflection.

The Out-of-Pool Storage Rack is currently used to store un-irradiated fuel six percent fuel. The rack was reviewed and approved in 1998 as a design change made in accordance with 10 CFR 50.59 and the facility license [11], technical specifications [1] and procedures for storage of six percent enriched fuel. A calculation for storage of six percent enriched fuel in the rack was performed in support of the design change which concluded that k_{eff} is not greater than 0.9 for all conditions of moderation and reflection. Only new, un-irradiated fuel is stored in the Out-of-Pool Storage Rack.

The In-Pool storage locations as detailed in Section 6.1.5 of the SAR [4] have been analyzed for the storage of six percent fuel and the results are discussed in RAI #11.

Consistent with NUREG 1537, Part 1 Chapter 9.2 [2]. ANSI/ANS-15.1-2007 Section 6.1.3 [3] and 6.4 [3], 10 CFR 50.36 and 50.54, and TS 6.1.4 [1] and SAR [4], all fuel movement activities are performed under the supervision of the Designated Senior Reactor Operator using approved operational procedures and is discussed in detail in RAI #2 and RAI #3.

Fuel shipments are performed in accordance with facility procedures that meet 10 CFR 20 and 10 CFR 71. A Quality Assurance Program for Shipment of Radioactive Material required under 10 CFR 71 is on file with the NRC. Security requirements for shipment are also in place as required by 49 CFR and 10 CFR.

Consistent with 10 CFR 20 and 10 CFR 51.22(c)(9), the use of 6% enriched fuel does not present any of the following:

- A significant hazard;

  The six percent enriched fuel is identical to the four percent enriched fuel already in use other than the enrichment, is off low strategic significance, and is approved for possession by the current reactor license. The six percent enriched fuel has been in storage at the facility since 1998. All fuel is monitored and protected as required by the facility physical security plan. There are no new or different hazards introduced by the storage or use of six percent enriched fuel.

- A significant change in the types or significant increases in the amounts of effluents released offsite;

  The storage of un-irradiated 6% enriched fuel does not create a new type of accidental release or increase the significance of an accidental release since the fuel is stored in a secured, monitored location inside the reactor facility.

- A significant increase in individual or cumulative occupational radiation exposure;

  - Radiation dose is monitored for compliance with 10 CFR 20 limits using accredited personnel dosimetry. Both individual and cumulative occupational radiation exposure is tracked as specified in facility procedures. In addition, the radiation monitoring system is in continuous operation and has alarm set points to warn personnel of abnormal radiation levels. Also, radiation surveys and air sampling analysis are performed as specified in the facility Health Physics procedures.

- Six percent enriched fuel has been stored in the facility since 1998. Irradiated four percent enriched fuel is currently stored in the reactor pool. Six percent fuel may be stored in the reactor pool if the pool storage facilities are determined to meet license requirements and are approved as required by the facility license and procedures.
- The Out-of-Pool Storage Rack is monitored as required by 10 CFR 70.24 for criticality accidents. Monitoring was evaluated and in place prior to receipt of the six percent enriched fuel in 1998.
- Fuel storage areas will remain unchanged inside the reactor pool. The reactor shield and reactor pool reduce radiation exposure to levels well within 10 CFR 20 limits.
- All fuel storage locations are below 2 mrem/h and air samples historically have not had detectable activity.

Radiation exposure to occupational workers and the public from storage and handling of all fuel meets limits in 10 CFR 20 and is consistent with ALARA.

11. NUREG-1537, Part 1, Section 9.2, states, in part, "[t]he applicant should provide an analysis and discuss how subcriticality is ensured." Provide an analysis that demonstrates the calculated effective neutron multiplication factor (k_{eff}) for the in-pool storage racks located in the PULSTAR pool containing un-irradiated and irradiated 6 weight percent fuel assemblies will remain subcritical (i.e., k_{eff} not to exceed 0.90).

RESPONSE

Consistent with NUREG 1537 Part 1 Chapter 9.2 [2] and ANSI/ANS-15.1-2007 Section 5.4 [3], ANSI/ANS-15.21 [12], 10 CFR 50.68, the TS 5.3 [1] requires that fuel be stored such that k_{eff} is not greater than 0.9 for all conditions of moderation and reflection. The In-Pool Storage Racks and In-Pool Storage Pits in the pool have been previously analyzed for four percent fuel and is discussed in Section 6.1.5 of the SAR [4]. The In-Pool Storage Racks have been analyzed for six percent fuel assembly storage using a MCNP model and the results indicate that the maximum K_{eff} remains below 0.6.

Operationally, the k_{eff} of the reactor fuel in the In-Pool Storage Racks will be measured using Startup Test 2.12 Revision A – *Installation of Neutron Source and Fuel in the Reactor Pool* to verify that k_{eff} is less than 0.9 as required by TS 5.3 [1].

Six percent fuel assemblies will only be authorized to be stored in the Large and Small In-Pool Storage Racks and will not be authorized to be stored in the In-Pool Storage Pits. Storage of six percent fuel assemblies in the In-Pool Storage Pits may be authorized at a later date through the 10 CFR 50.59 process as long as TS 5.3 [1] requirements are satisfied. Storage of fuel in allowed locations is controlled using the fuel handling procedure NRP-OP-301 *Reactor Fuel Handling* and is described in detail in RAI# 2, RAI #3 and RAI #10.

12. NUREG-1537, Part 2, Section 7.4, “Reactor Protection System,” states, in part, that the Reactor Protection System should place the reactor in a subcritical, safe shutdown condition when any of the monitored parameters exceeds the limit as defined in the SAR. Describe any changes to the instrumentation and control system that affect the reactor protection system for the PULSTAR reactor necessitated by the use of the mixed core using six weight percent fuel assemblies or explain why they were not required.

RESPONSE

The Reactor Protection System is comprised of multiple systems. These systems include; Nuclear Instrumentation System, Non-Nuclear Instrumentation System and the Scram Logic Unit.

The Nuclear Instrumentation System which is a portion of the Reactor Protection System at the PULSTAR includes instrumentations channels such as the ion chamber based Log/Linear Channel, Linear Channel, and Safety Channel and the fission chamber based Source Range Channel which are described in detail in Section 7.2 of the SAR [4]. These channels function by measuring and responding to the neutron leakage flux from the reactor core. As discussed in Section 4 of Appendix A [5] of the LAR [6] the core kinetic parameters such as moderator feedback coefficient, reactivity insertion rate, void coefficient, doppler feedback coefficient, power coefficient, and β_{eff} all remain essentially unchanged when comparing four percent enrichment core configurations to the mixed enrichment core configurations. Since the kinetic parameters remain unchanged along with all other core parameters, e.g. core size, power, temperature, etc., the Nuclear Instrumentation System will respond exactly the same for the mixed enrichment core configurations compared to the four percent enrichment core configurations, therefore for this LAR [6] no changes are required nor were any changes made to the Nuclear Instrumentation System.

The Non-Nuclear Instrumentation System (i.e. process instrumentation) is another portion of the Reactor Protection System and includes channels such as Flow Measuring Channel, Temperature Measuring Channel, and Pool Level Measuring Channel and is described in detail in Section 7.3 of the SAR. All process instrumentation parameters remain unchanged and are listed in TS Table 3.3-1 [1]. These process variables are independent of core enrichment therefore there for this LAR [6] no changes are required nor were any changes made to the Non-Nuclear Instrumentation System.

The SCRAM Logic Unit as described in Section 7.4.2 of the SAR [4] controls the supply of electrical current to the magnets of the control rods by responding to the outputs coming from the channels described above. Since the channels suppling the input signals to the SCRAM Logic Unit remain unaffected by mixed enrichment core configurations so does the SCRAM Logic Unit itself, therefore no changes are required nor were any changes made to the SCRAM Logic Unit.

This LAR does not change reactor power levels, the Technical Specifications [1] (other than 5.1.a), SAR limits [4], or any operating conditions, therefore there have been no instrumentation or equipment changes to the PULSTAR due to this LAR [6].

13. NUREG-1537, Part 2, Section 12.11, “Startup Plans,” states, in part, that the applicant should submit a startup plan whenever significant core modifications are being made. Describe the NCSU start-up procedure for the reactor core utilizing the mixed enrichment core configuration loadings that will provide confirmation of analysis predictions for the mixed enrichment core.

RESPONSE

Consistent with NUREG 1537 Part 1 Chapter 12 Section 12.3 [2], ANSI/ANS-15.1-2007 Section 6.4 [3], ANSI/ANS-15.21 Section 2.12 [12], Technical Specification 6.4 [1] and SAR Section 12 [4], the NCSU PULSTAR Reactor has approved procedures for activities related to the operation of the reactor. This includes procedures for loading, commissioning and operating with new core configurations as discussed in NUREG 1537 Part 1 Chapter 12 Section 12.11 [2].

The procedure for evaluating and loading mixed cores will be the same as that for any new core configuration.

1.) An acceptable core configuration that satisfies all license criteria will be selected using the computer models.
2.) The core will then be loaded. Using established approved procedures, parameters such as peaking factors, fuel assembly worths, excess reactivity, shutdown margin and reactivity coefficients will be verified to be within all Technical Specifications and SAR limits.
3.) Upon completion of successful verification measurements, the core will be certified for operation.

The procedures used for this process include:

- NRP-OP-301 *Reactor Fuel Handling*
- NRP-OP-301 *Reactor Fuel Handling – Section 5 – Fuel Assembly Reactivity Worth Determination*
- PS-8-03-1 *Core Flux Mapping*
- PS-8-02-1 *Excess Reactivity and Shutdown Margin Calculation*
- PS-8-04-1 *Reactivity Coefficients*
- PS-4-01(02)(03)(04)(08)-2 *Control Rod Calibration*

14. In regards to NCSU SAR, Section 13.2.1.4, "Fuel Pin Clad Failure," explain whether the use of six weight percent fuel impacts the fission products released, and provide updates to Tables 13.1, and Tables 13.2, as appropriate.

RESPONSE

Accidental release for PULSTAR fuel is analyzed in Section 13.2.1.4 of the SAR [4]. Fission products in the fuel cladding gap from three fuel pins from one assembly for a reactor core with 25 assemblies with a burnup of 20,000 MWd/MTU is used as the source term. With the exception of enrichment the six percent fuel is identical to the four percent fuel. The TS 5.1.b [1] burnup limit of 20,000 MWd/MTU remains unchanged, therefore the source term is solely a function of the MTU per pin. The average MTU per pin provided by the quality assurance documentation from the manufacturer is listed below.

The MTU for the fuel pins are as follows:

- 4% Fuel Pin 5.068 E-4 MTU
- 6% Fuel Pin 5.136 E-4 MTU
- %Difference 1.32%

Burnup is limited to 20,000 MWd per MTU for both four percent and six percent enriched fuel. At the burnup limit, this gives:

- 4% pin 10.136 MWd
- 6% pin 10.272 MWd
- %Difference 1.32%

As a result of the difference in enrichment the gap activity increases by 1.32% at maximum burnup for six percent enriched fuel. This change would increase the radiation dose from an accident by 1.32% in both restricted and unrestricted areas.

Section 4.3.3 of NEI 96-07 Rev 1 [13] considers an increase of ten percent or more as being more than a minimal increase. Because the increase is less than 10 percent, it would be classified as a minimal increase, therefore the analysis in SAR Section 13.2.1.4 [4] is not affected by the planned use of six percent enriched fuel. The results remain well below public dose limits. An update to SAR Tables 13-1 and 13-2 [4] is not required.

15. NUREG-1537, Part 2, Chapter 10, "Experimental Facilities and Utilization," and Section 13.1.6, "Experiment Malfunction," states, in part, that the applicant should provide an analysis to demonstrate that the reactor and experimental facilities can be operated safely during normal and abnormal events. NCSU SAR, Section 13.2.2.4, analyzes the impacts of an experiment failure, but does not discuss the possibility of a beam tube flooding.

a) Explain any impact the experimental facility imposes on the mixed core configuration and any impact the mixed core configuration imposes on the experimental facility.

b) Describe the impact of a neutron beam tube flooding event on a mixed core utilizing six weight percent fuel.

RESPONSE

a) This is discussed in SAR Section 3.2.2.4 [4] and is based upon the restriction that non-secured experiments shall not exceed a worth of 1000 pcm. This excursion accident is bounded by the Fuel Loading Accident detailed in SAR section 13.2.2.1 [4] and discussed in RAI #5 but is less severe, falling within the step reactivity insertion limit of 1590 pcm.

Therefore since the mixed enrichment core configurations are consistent with the analysis in Section 13.2.2.1 of the SAR [4] for step reactivity insertions as detailed in RAI #5, and since experiment limits in SAR Section 13.2.2.4 [4] are bounded by that analysis, the analysis remains valid.

Classification	Experiment Reactivity Limit [pcm]
Movable	300 pcm or 100 pcm/second
Non-secured	1000 pcm
Secured	1590 pcm

b) The flooding of one of the beamtubes would be classified as a reactivity insertion event. The following table lists the measured reactivity for these experimental facilities. As can be seen, Beamtube No. 6 has the highest reactivity value of 750 pcm which is below both the step reactivity insertion event of 1590 pcm as discussed in RAI #5 and the ramp reactivity insertion event of 850 pcm as discussed in RAI #6.

Experimental Facility	Reactivity Worth [pcm]
Rotating Exposure Port	≤20
Dry Exposure Port	≤10
Beamtube No. 1	150

Beamtube No. 2	75
Beamtube No. 3	25
Beamtube No. 4	75
Beamtube No. 5	75
Beamtube No. 6	750
Pneumatic Transfer System	≤10
Noseport	30

The reactivity worth of the beamtube facilities calculated by the computer model detailed in the Appendix A [5] of the LAR [6] are consistent with what has been measured and shows that the use of the six percent fuel assemblies in the mixed enrichment core configurations has negligible impact on the reactivity worths of the beamtubes compared to four percent enrichment core configurations. Therefore the consequences of a beamtube flooding are the same for a four percent core configuration and mixed enrichment core configurations, both of which are bounded by the step reactivity insertion described in SAR Section 13.3.3.1 [4] and RAI #5 and the ramp reactivity insertion event described in SAR Section 13.2.2.2 [4] and RAI #6.

16. NCSU SAR, Section 3.2.2.6, “Cold Primary Coolant Slug,” provides an analysis for the cold primary coolant slug event. The facility modifications made in 2013 to the primary and secondary system to increase cooling flow rates/capacities combined with the mixed core use of six weight percent enriched fuel may make the cold primary coolant slug event more severe. Discuss the potential impact of the use of six weight percent fuel on the cold primary coolant slug event described in the NCSU SAR.

RESPONSE

The flow rates remain unchanged following the facility modifications made in 2013 to the primary and secondary coolant systems. An automatic reactor SCRAM will occur if flow rates fall outside of the nominal operating band of 500 gpm ± 25 gpm.

In the SAR Section 3.2.2.6 [4] accident analysis it is assumed that the reactor is operating in natural convection mode at 150 kW when the flapper is closed and the primary pump is started. Closing the flapper will start forced convection mode such that the 27°F temperature rise across the core will no longer exist. The analysis assumes that the temperature rise across the core with drop to zero, which it will not do, but the assumption is conservative. This decrease in temperature will result in a positive reactivity affect as summarized in the table below.

$$\rho = \alpha_T \times \Delta T_{moderator}$$

	SAR Value	Core No 8 Calculated	Core No. 9-1 Calculated	Core No. 9-2 Calculated
α_T	-3.9 pcm/°F	-3.44 pcm/°F	-2.96 pcm/°F	-3.44 pcm/°F
ΔT	27°F	27°F	27°F	27°F
ρ	105.3 pcm	92.9 pcm	79.9 pcm	92.9 pcm
P_{final}	318 kW	282 kW	242 kW	282 kW

The use of six percent fuel assemblies would have a minimal impact on the outcome of a cold primary coolant slug event. As listed in Table 4.3 of Appendix A [5] of the LAR [6], the calculated moderator temperature coefficients (α_T) for the mixed enrichment cores range from -2.96 pcm/°F to -3.44 pcm/°F. These values are less than or equal to the calculated value for four percent Reflected Core No. 8 of -3.44 pcm/°F, and the historically measured value used in the SAR [4] of -3.9 pcm/°F. Having a lower value for α_T would have the effect of mitigating the cold primary slug event, rather than making it more severe.

Therefore, the final reactor power due to the reactivity insertion from a cold slug event occurring in a mixed enriched core would be equal to or less than the 320 kW listed in the SAR. Where P_{final} is:

$$P_{final} = 1\ MW \times \frac{\rho}{\rho_{1MW}}$$

$$\rho_{1MW} = 330\ pcm$$

In addition, the moderator temperature coefficient is measured and verified for all new mixed enrichment core configurations as discussed in RAI #13.

References

[1] North Carolina State University, *Facility License - Appendix A - Technical Specifications for the North Carolina State Univeristy PULSTAR Reactor,* 2008.

[2] *NUREG 1537 Part 1 - Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors - Format and Content,* Nuclear Regulatory Commission, 1996.

[3] *ANSI/ANS-15.1-2007 - Development of Technical Specifications for Research Reactors,* American Nuclear Society, 2007.

[4] North Carolina State University, *Safety Analysis Report,* 1995.

[5] A. I. Hawari and J. L. Wormald, *License Amendment - Appendix A - Examination of Mixed Enrichment Core Loading for the NCSU PULSTAR Reactor,* 2015.

[6] North Carolina State University, *License Amendment for the Use of 6% Enriched Fuel,* 2015.

[7] American Machine and Foundry Co., *Specifications for PULSTAR Fuel Assemblies APR-1 Revision 5,* 1970.

[8] Western New York Nuclear Research Center, Inc., *WNY-017 - PULSTAR Pulse Test,* 1964.

[9] Western New York Nuclear Research Center, Inc., *WNY-023 - PULSTAR Pulse Tests III and IV - Comparison of Natural and Forced Convection Cooling,* 1966.

[10] Western New York Research Center, Inc., *Hazards Summary Report Revision II,* 1963.

[11] North Carolina State University, *Facility License R-120,* 2006.

[12] *ANSI/ANS-15.21-1996 - Format and Content for Safety Analysis Reports for Research Reactors,* American Nuclear Society, 1996.

[13] Nuclear Energy Institute, *NEI 96-07 - Revision 1 - Guidlines for 10 CFR 50.59 Implementation,* 2000.

[14] *NUREG 1537 Part 2 - Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors - Standard Review Plan and Acceptance Criteria,* Nuclear Regulatory Commission, 1996.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

FOR LICENSE AMENDMENT REQUEST FOR

THE USE OF SIX PERCENT ENRICHED FUEL (TAC NO. MF6088)

ATTACHMENT 1 – SAFETY ANALYSIS UPDATES

NORTH CAROLINA STATE UNIVERSITY

LICENSE NO. R-120; DOCKET NO. 50-297

DECEMBER 18TH, 2015

TABLE 1-1
COMPARISON OF PULSTAR REACTORS

	NCSU	BMRC
Fuel		
Material	UO_2	
Form	Sintered Pellets	
Enrichment (w% ^{235}U)	4% and/or 6%	6%
Design Inventory Core (kg UO_2)	359	285
Density (gm/cm^3)	10.5 – 10.76	10.3
^{235}U per Fuel Pin (gm)	20.2/30.7	30.7
Fuel Pin		
Clad Material	Zr-2	
Pellet Diameter (in)	0.423	
Diametrical Gap (in)	0.0085	
Clad thickness (in)	0.4725	
Rectangular Spacing (in)	0.606 × 0.524	
Clearance, pin to pin (in)	0.051 × 0.133	
Clearance, pin to box (in)	0.025 × 0.066	
Height of Pellet Stack (in)	24	
Pins per Core	625	500
Height of Pellet (in)	0.60	
Fuel Box		
Material	Zr-2 or Zr-4	
Inside Dimensions (in)	2.620 × 3.030	
Wall Thickness (in)	0.060	
Clearance between Assemblies (in)	0.040	
Clearance between Control Rod Guide and Assemblies (in)	0.060	

Fuel Pins per Assembly	25	
Weight (lbs)	44	
Moderator – Coolant		
Material	Light Water	
Nominal Inlet Temp (ºF)	105	100
Nominal Outlet Temp (ºF)	118.8	111.8
Primary Flow Rate (gpm)	500	1150
Secondary Flow Rate (gpm)	700	800
Reflector	Light Water, Graphite, Beryllium	
Standard Core	Light Water	
Reflected Core No. 1	Light Water, Graphite	
Reflected Core No. 2	Light Water, Graphite	
Reflected Core No. 3	Light Water, Graphite	
Reflected Core No. 4	Light Water, Graphite, Beryllium	
Reflected Core No. 5	Light Water, Graphite, Beryllium	
Reflected Core No. 6	Light Water, Graphite, Beryllium	
Reflected Core No. 7	Light Water, Graphite, Beryllium	
Reflected Core No. 8 (Current)	Light Water, Beryllium	
Neutron Source	Pu-Be	Sb-Be
Control Rods		
Absorber Material	Ag-In-Cd (80-15-5)	
Guide Material	Aluminum	
Shape	Rectangular	
Transverse Dimensions (in)		
Guide	6.30 × 0.43	
Absorber	4.85 × 0.18	
Clearance Absorber to Guide (in)	0.00625	
Clad Material	Sn/Ni	Ni
Number of Control Rods	3	5
Number of non-scrammable rods	1	
Core Dimensions		
Overall (in)	$15^{7}/_{8} \times 15 \times 24$	$15^{7}/_{8} \times 12^{1}/_{8} \times 24$
Volume Fractions		
UO_2	0.3823	0.3789
Gap (Helium)	0.0155	0.0154
Cladding (includes warts)	0.0803	0.0795
Lattice Water	0.3858	0.3824
Assemblies (Zr-2)	0.0753	0.0747
Water between Assemblies	0.0255	0.0252
Control Rod Guides (Al)	0.0128	0.0158
Water Inside Guides (Rods out)	0.0226	0.0280
Total Volume Fractions		
UO_2	0.3823	0.3789

Helium Gap	0.0155		0.0154
H_2O	0.4339		0.4356
Zr-2	0.1555		0.1542
Aluminum	0.0128		0.0158
Core Volume Fractions			
H_2O/UO_2	1.135		1.150
H_2O/U	2.06		2.11
Physics Parameters			
Effective Neutron Temperature (ev)	0.0509		0.0587
Disadvantage Factors			
Moderator to Fuel	1.243		1.309
Clad to Fuel	1.125		1.163
Neutrons per Thermal Fissions (v)		2.43	
Capture to Fission Rate (α)		0.18	
Thermal Utilization	0.951		0.964
Fast Fission Number	1.397		1.050
Age (cm^2)	43.8		30.7
Resonance Escape (p)	0.558		0.767
Thermal Diffusion Area (L^2)	2.215		1.178
Reflector Savings (cm)	7.81		7.77
Bucklings (cm^2)			
Width	0.00342		0.00320
Length	0.00316		0.00260
Height		0.00168	
k_{eff} (cold, clean w/ H_2O Reflector)	1.0178		1.045
Minimum Critical No. of Assemblies (4%)	20		16
Doppler Coefficient (pcm/ºF)	-1.60		-2.15
Void Coefficient (pcm/cm^3)	-1.60		-1.1
Moderator Temperature Coefficient (pcm/ºF)	-3.9		-8.0
Beamtube Worths, air to water filled (pcm)			
6” diameter	75		100
8” diameter	45		-
12” × 12”	300		-
Neutron Lifetime (sec)	2.6×10^{-5}		2.9×10^{-5}
β_{eff}	0.0073		0.0076
Steady State Power Level (MW)	1		2
Design Pulse Peak Power (MW)	2200		2000
Design Pulse Total Energy Release	38		40
Design Step Input (pcm)	1590		1740
Maximum Rate of Reactivity Insertion by Control Rods (pcm/sec)	100		100
Reactor Bay			
Dimensions (ft)			
Height	55		52
Width	37		-

Diameter	-	70
Length	94	
Free Air Volume (ft^3)	86,500	186,000
Ventilation (cfm)		
Normal	1870	12,000
Emergency	600	3,200
Type Building	Confinement	Containment
Exhaust Stack Height (ft)	100	167
Pool Volume (gal)	15,600	14,592

TABLE 3-2 – OPERATIONAL REACTIVITY REQUIREMENTS	
PARAMETER	REACTIVITY
Xenon (typical operations)	600 pcm
Moderator Coefficient	140 pcm
Power Defect	330 pcm
TOTAL	1070 pcm

3.2.2.4 Fuel

The reactor core is comprised of twenty-five fuel assemblies. Each assembly as shown in Figure 3-4, contains twenty-five fuel pins. Each pin consists of a zircaloy-2 tube of 0.02 inch wall thickness, filled with sintered UO_2 pellets and sealed at the top and bottom. The uranium is enriched to 4 percent or 6 percent by weight in the U-235 isotope. Each pellet measures about 0.42 inches in diameter and 0.6 inches in length. The finished fuel pin is 0.47 inches in diameter and 26 inches long. Approximately 20.2 grams of U-235 is contained in each of the 4% fuel pins while 30.7 grams of U-235 are contained in each 6% fuel pin.

Twenty-five pins are fastened mechanically into bundles and are placed in a zircaloy-2 box open at the top and bottom with a cross section measuring about 2.7 inches by 3.2 inches. The upper end fittings and the lower end fitting (nosepiece) are attached, bringing the overall length of the assembly to about 38 inches. A bail is inserted between side plates at the top of the assembly to serves as a handle for moving the assembly. There are also two alignment holes in the shoulder of the lower end fitting which mate with pins on the grid plate to prevent misalignment of the assemblies in the core. Openings are provided at the sides of each fuel assembly box to allow coolant flow if the top of the fuel assembly should become blocked by a foreign object. The fabrication of the PULSTAR fuel assemblies is in accordance with AMF Specification APR-1.

3.2.3.2 Comparison with Buffalo PULSTAR

The Buffalo PULSTAR has a 20 fuel assembly core of six percent enrichment, and was licensed to operate at a steady-state power level of 2 MW and to pulse routinely with a total energy release of 40 MW·sec, which is equivalent to a maximum specific energy release of 490 MW·sec/gram. During the BMRC PULSTAR license renewal in 1983, the facility elected to remove pulsing from their license. Hence the use of experience at BMRC in pulsing is based on their pulsing prior to this licensing change. The North Carolina State University PULSTAR, a modified version of the BMRC PULSTAR, is designed to operate with a twenty five assembly core of 4 percent and/or 6 percent enrichment, at a steady-state power level of 1 MW, and to a pulse with an originally estimated total energy release of 38 MW·sec, which, because of the larger core, is equivalent to specific energy release of 310 MW·sec/gram, based on a hot spot factor of 2.92 and a 25 fuel assembly core.

A comparison of the significant operating parameters for the BMRC and NCSU design is as follows:

TABLE 3-4A – COMPARISON OF OPERATING PARAMETERS
FOR THE BMRC AND NCSU PULSTAR REACTORS

OPERATING PARAMETER	BMRC	NCSU
Design Steady-state Power (MW)	2	1
Mass UO_2 per core (kg)	287	359
Uranium Enrichment	6	4 and/or 6%
Maximum Specific Energy Release (MW·sec/gram)	490	310
Original Design Pulse Peak Power (MW)	2000	2200
Original Design Pulse Energy Release (MW·sec)	40	38

As with the BMRC PULSTAR, the NCSU PULSTAR Facility has elected to discontinue pulsing as of License Amendment No. 9.

As part of the BMRC test program, a test pin was located at a point where the flux was a factor of 1.8 higher than the peak flux in the fuel region. The test pin thereby served to lead the core in peak power and to provide a warning of any abnormality as the test program pulses were progressively increased to the design level. The core physics design for the NCSU PULSTAR is analyzed from the viewpoint that the actual test results obtained from the very similar BMRC PULSTAR core served as the most accurate basis for predicting the NCSU PULSTAR core performance. The applicable results of the BMRC test program are therefore referenced frequently in the following sections and serve as a point of departure in providing design modifications for the NCSU requirements.

3.2.3.3 Core Configurations

The major components of the NCSU PULSTAR core have been described in the previous sections. This section contains a description of normal reactor operations, including the power distribution, excess reactivity, effects of burnup and resulting reactivity changes. A model of the core, used for MCNP6 simulations of core performance, is presented, along with benchmarks needed for validation of the model.

3.2.3.3.1 Historical Core Configurations

The NCSU PULSTAR Reactor has a flexible core in which the geometric arrangement of fuel assemblies can be modified to satisfy various experiment requirements. The reactor shall not be operated unless the following conditions exist:

a. A maximum of twenty-five fuel assemblies.

b. A maximum of ten reflector assemblies of either graphite or beryllium or a combination of these located on the core periphery.

c. Unoccupied grid plate penetrations plugged.

d. A minimum of four control rod guides are in place.

e. The maximum worth of a single fuel assembly shall not exceed 1590 pcm.

f. The total nuclear peaking factor in any fuel assembly shall not exceed 2.92.

Specification *a* through *f* allow for flexibility when designing the reactor core but assures that there is no bypass cooling flow around the fuel assemblies. Specification *e* provides assurances that a fuel loading accident will not result in a Safety Limit to be exceeded as discussed in SAR Section 13.2.2.1. Specification *f* provides assurance that core hot channel power is bounded by the assumptions in this SAR analysis.

The initial core loading for start-up of the PULSTAR Reactor was the Standard Core, depicted in Figure 3-8A. This core was comprised of 25 fuel assemblies in a five by five array with four control rods, one of which was a pulsing rod not normally used to control the reactor. As the fuel depleted over the first few years of operation, the core was changed to the Reflected Core No. 1, by incorporating five graphite reflectors on the north side of the five by five array of fuel. Figure 3-8B depicts the fuel, graphite and control rod arrangement for Reflected Core No. 1. The core was then rearranged to increase excess reactivity by adding five more graphite reflectors to Reflected Core No. 3, depicted in Figure 3-8D. Five beryllium reflectors replaced the graphite reflectors on the west side on the fuel array to make Reflected Core No. 4, depicted in Figure 3-8E. The last of the five graphite reflectors on the north side where replaced by beryllium to make Reflected Core No. 8, depicted in Figure 3-8I. The fuel and reflector positions for each core configuration is listed in Table 3-4B and are illustrated in chronological order in Figures 3-8A through 3-8I.

Table 3-4B – Summary of Core Configurations of the NCSU PULSTAR Reactor			
Core Name	Operational Hours	Dates of Operation	Configuration
Standard Core	0 – 95 MWd	August 1972 – February 1977	5×5 array of 4% assemblies
Reflected Core No. 1	95 – 162 MWd	April 1972 – June 1979	Addition of 5 graphite reflectors.
Reflected Core No. 2	Low Power Testing Only		
Reflected Core No. 3	162 – 861 MWd	June 1979 – March 1999	Addition of 5 graphite reflector in row A. Core reconfigured.
Reflected Core No. 4	861 – 969 MWd	March 1999 – June 2005	5 Graphite reflectors replaced with beryllium.
Reflected Core No. 5	969 – 1186 MWd	June 2005 – June 2009	Fresh fuel added. And core reconfigured.
Reflected Core No. 6	1186 – 1241 MWd	June 2009 – October 2010	Core reconfigured.
Reflected Core No. 7	1241 – 1425 MWd	October 2010 – November 2011	Fresh fuel added. And core reconfigured.
Reflected Core No. 8	1425 – Present	November 2011 – Present	5 Graphite reflectors replaced with beryllium.

Standard Core

Figure 3-8A – Standard Core

The Standard Core was the initial core configuration. This core was used until April of 1977 for a total of 2284 MW·hr, giving an average burn-up of 0.29 pcm/MW·hr.

The peak-to-average ratio of the Standard Core was determined by the lucite wand method to be 2.80. Lucite wands were constructed to fit into coolant channels of the PULSTAR fuel assemblies. The wands were notched across the bottom and four shallow grooves were cut into them approximately halfway up from the bottom. The notches and grooves positioned copper wires 90° from each other around the circumference of the wands. The wands were manually inserted into specific fuel element coolant channels while the element was out of the pool. Each wand was oriented so that the copper wires were adjacent to the fuel pins. After irradiation the wands were removed and the copper wires cut off and counted. The β^- decay of ^{64}Cu was counted using thin end window flow proportional counters. After appropriate normalization, the peak to average ratio was computed.

Reflected Core No. 1

Figure 3-8B – Reflected Core No. 1.

Reflected Core No. 1 was loaded in April of 1977, and consisted of:

- Adding 5 graphite reflectors in the 6-column.

This resulted in an increase of core excess reactivity of 920 pcm. This core was used until June of 1979 for a total of 1623 MW·hr, giving an average burn-up of 0.07 pcm/MW·hr.

Reflected Core No.1 peak-to-average ratio was determined by the lucite wand method to be 2.5 and 2.07 by the fission probe method.

The fission probe method uses a miniature fission chamber in a drywell that is then inserted into various coolant channels. The output of the fission chamber is then recorded at various heights throughout the reactor core. This can be done either by continuous withdrawal or at discrete positions.

Reflected Core No. 2

Figure 3-8C – Reflected Core No. 2.

Reflected Core No. 2 was used for testing purposes only, and involved the loading of graphite plugs into the beamtubes to measure the net change in reactivity associated with the increase in graphite reflection. Loading graphite plugs into all of the radial beamtube resulted in a net gain of 380 pcm.

Reflected Core No. 3

Figure 3-8D – Reflected Core No. 3.

Reflected Core No. 3 was loaded in June of 1979, and consisted of:

- Relocating the REP's to the core extension grid plate.
- Moving the A-row of fuel to the F-row.
- Adding 5 graphite reflectors in the A-row.

This core was used until March of 1999 for a total of 16778 MW·hr, giving an average burn-up of 0.12 pcm/MW·hr.

Reflected Core No.3 peak-to-average ratio was determined by the fission probe method to be 2.12.

Reflected Core No. 4

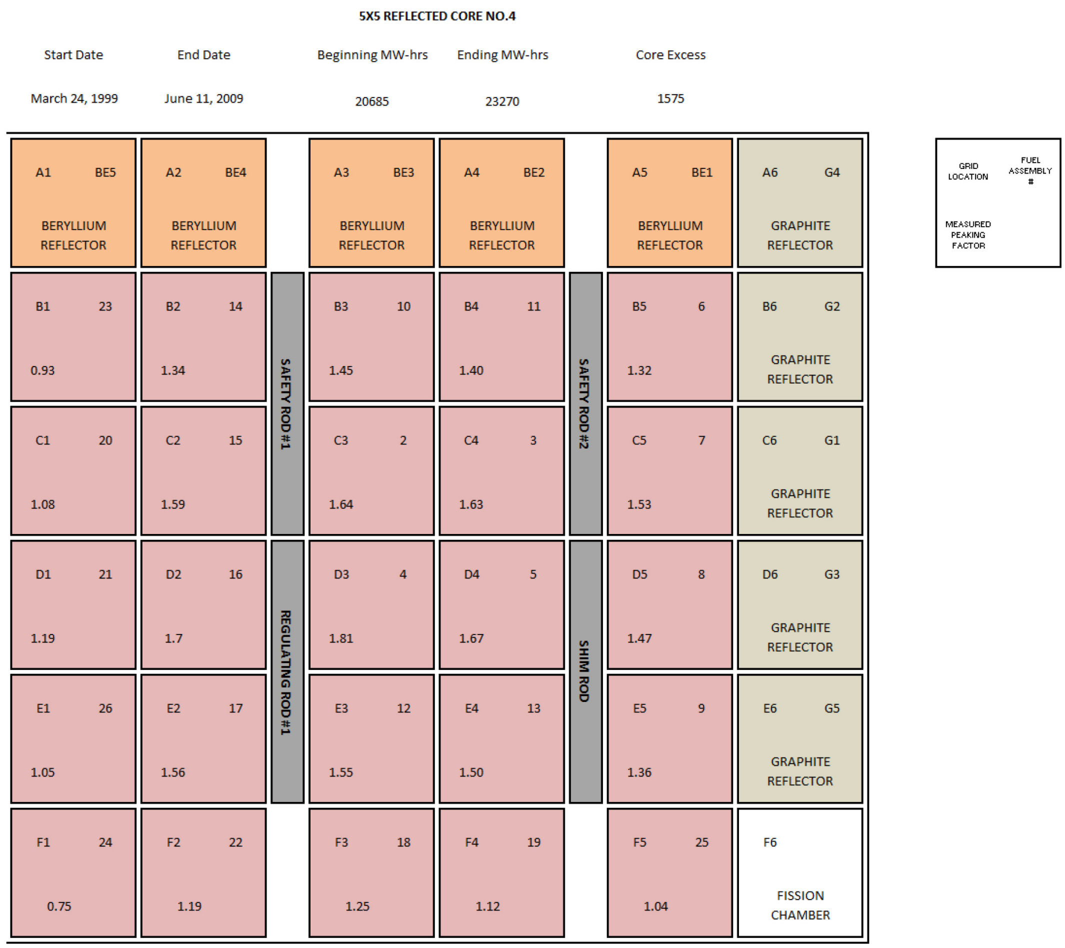

Figure 3-8E – Reflected Core No. 4.

Reflected Core No. 4 was loaded in March of 1999, and consisted of:

- Replacing the A-row graphite reflectors with beryllium reflectors.

This resulted in an increase of core excess reactivity of 740 pcm. This core was used until June of 2005 for a total of 2585 MW·hr, giving an average burn-up of 0.17 pcm/MW·hr.

Reflected Core No.4 peak-to-average ratio was determined by the fission probe method to be 1.80.

Reflected Core No. 5

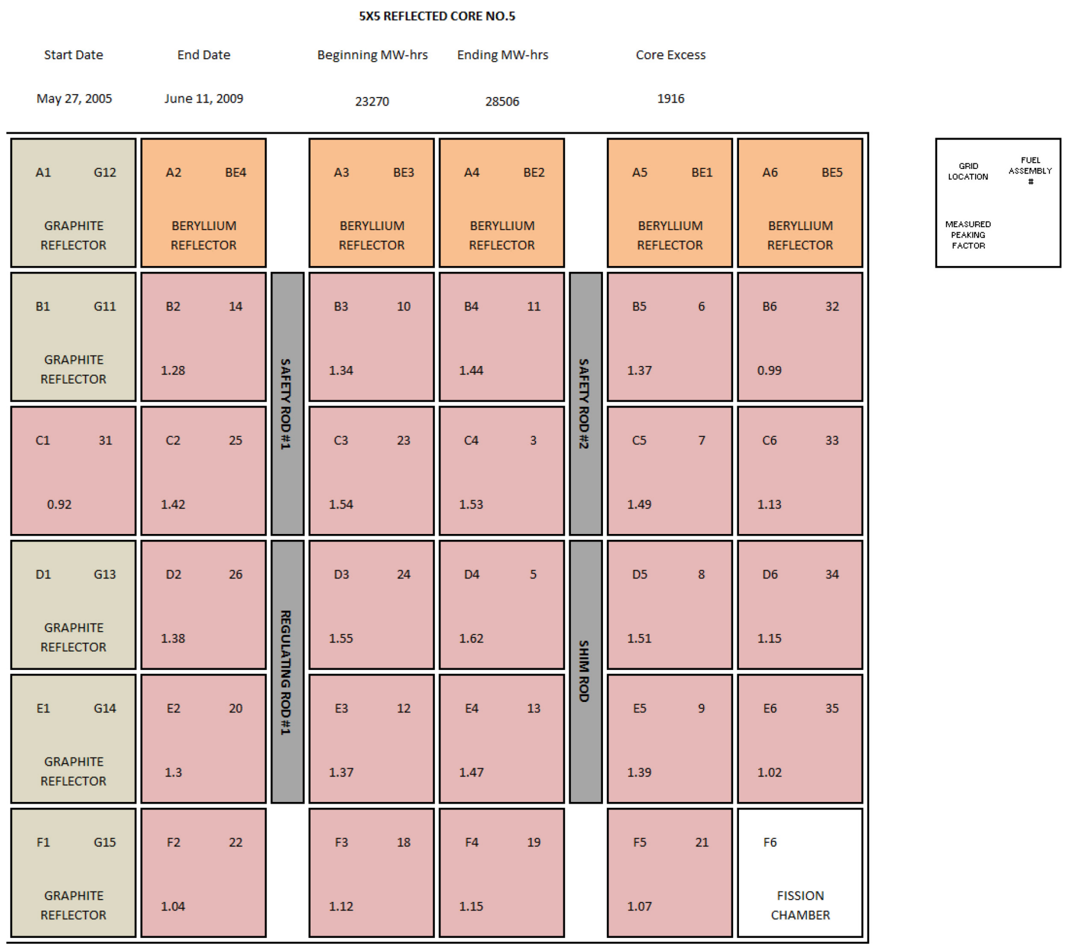

Figure 3-8F – Reflected Core No. 5.

Reflected Core No. 5 was loaded in Jun of 2005, and consisted of:

- Moving the 6-column of graphite reflectors to the 1-column.
- Adding 4 new fuel assemblies to the 6-column.
- Adding 1 new fuel assembly to grid location C1.
- Shuffling select fuel assemblies.

This resulted in an increase of core excess reactivity of 870 pcm. This core was used until June of 2009 for a total of 5193 MW·hr, giving an average burn-up of 0.18 pcm/MW·hr.

Reflected Core No.5 peak-to-average ratio was determined by the fission probe method to be 1.62.

Reflected Core No. 6

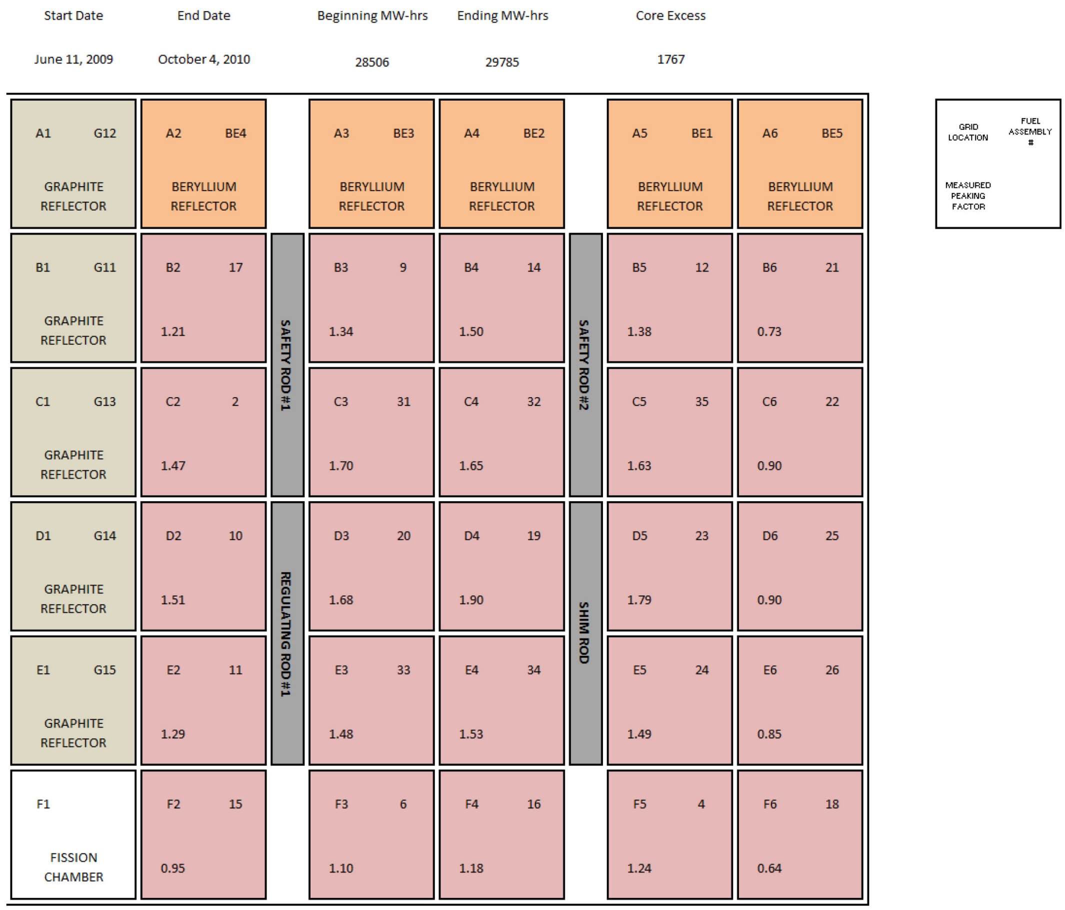

Figure 3-8G – Reflected Core No. 6.

Reflected Core No. 6 was loaded in Jun of 2009, and consisted of:

- Relocating the fission chamber to F1.
- Rearranging the core to a traditional 5x5 array.
- Shuffling select fuel assemblies.

This resulted in an increase of core excess reactivity of 625 pcm. This core was used until October of 2010 for a total of 1279 MW·hr.

Reflected Core No.6 peak-to-average ratio was determined by the fission probe method to be 1.90.

Reflected Core No. 7

Figure 3-8H – Reflected Core No. 7.

The configuration of reflected Core No. 7 was designed to increase core excess reactivity. Four new assemblies (29, 30, 27, 28) were loaded into the core.

This core was used until November of 2011 for a total of 4419 MW·hr.

Reflected Core No.7 peak-to-average ratio was determined by the fission probe method to be 1.86.

3.2.3.3.2 Current Core Configurations

Reflected Core No. 8

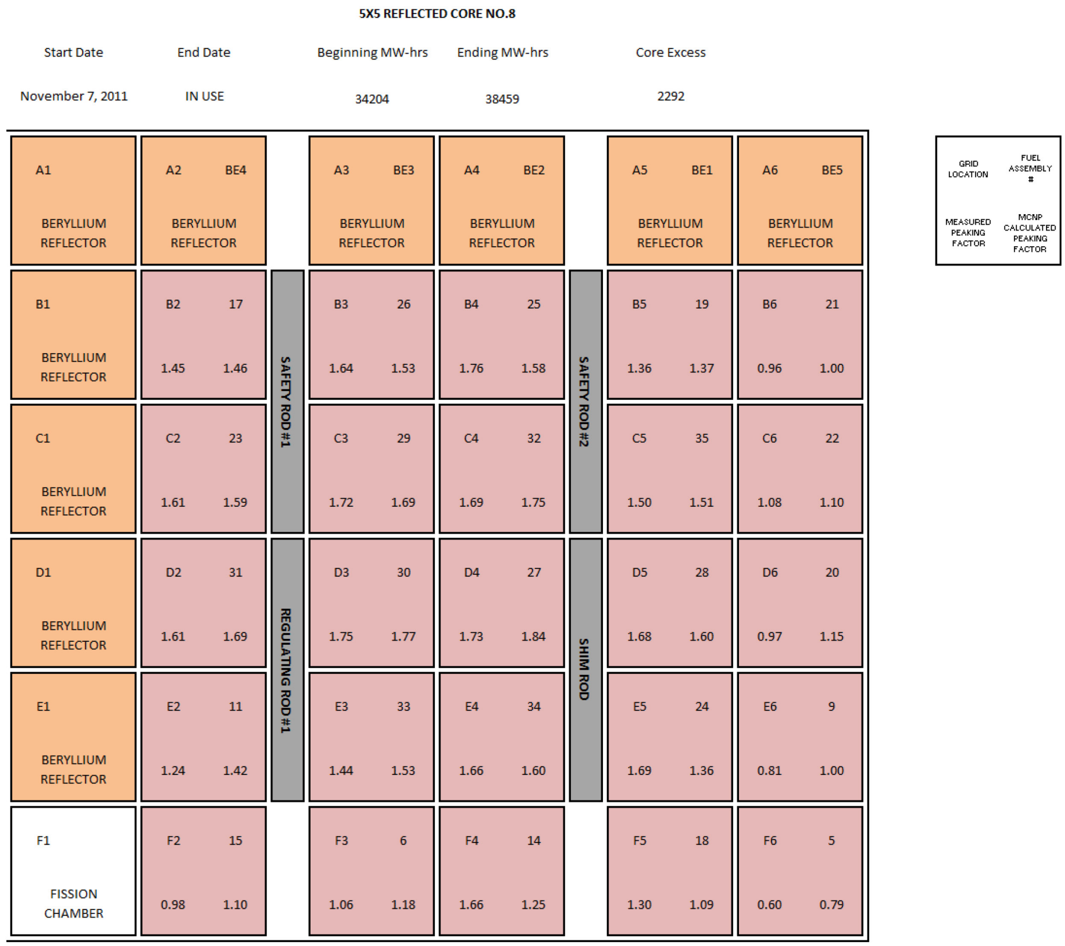

Figure 3-8I – Reflected Core No. 8.

The configuration of reflected Core No. 8 had the last five graphite reflectors replaced with beryllium reflector. The net reactivity gain of the beryllium reflectors was 700 pcm.

This core is currently in use.

Reflected Core No.8 peak-to-average ratio was determined by the fission probe method to be 1.76.

3.2.3.3.3 Future Core Configurations

There is no set of pre-planned core configurations for the PULSTAR Reactor. Rather, any configuration is acceptable provided that all criteria specified in the SAR and Technical Specifications are satisfied. Therefore, the following mixed enrichment test cases are representative examples only. The actual configuration for future cores will be determined based on the conditions and requirements at the time. For these examples, core configurations containing a single six percent enriched fuel assembly were considered. MCNP6 calculations were performed where a single six percent enriched fuel assembly was inserted in a selected position that originally contained a four percent enriched fuel assembly in Reflected Core No. 8 to examine the core limits using such fuel loading patterns. Multiple six percent loading patterns were considered for positions that did not violate safety limits for single six percent fuel assembly loading, and are shown in Figure 3-8J through 3-8M. Each fuel assembly is labeled according to its numeric index in the upper right corner, the core position in the upper left and the fuel assembly power peaking factor in the lower right. The six percent fuel assemblies are shown colored in green and will be numbered sequentially.

Mixed Core Test Case No. 9-1

Figure 3-8J – Mixed Core Test Case No. 9-1.

Mixed Core Test Case No. 9-2

5X5 REFLECTED CORE NO.9-2

Start Date	End Date	Beginning MW-hrs	Ending MW-hrs	Core Excess
N/A	N/A	N/A	N/A	

A1 BERYLLIUM REFLECTOR	A2 BE4 BERYLLIUM REFLECTOR		A3 BE3 BERYLLIUM REFLECTOR	A4 BE2 BERYLLIUM REFLECTOR		A5 BE1 BERYLLIUM REFLECTOR	A6 BE5 BERYLLIUM REFLECTOR
B1 BERYLLIUM REFLECTOR	B2 17 1.39	SAFETY ROD #1	B3 26 1.46	B4 25 1.51	SAFETY ROD #2	B5 19 1.33	B6 21 0.98
C1 BERYLLIUM REFLECTOR	C2 23 1.51		C3 29 1.62	C4 32 1.69		C5 35 1.49	C6 22 1.10
D1 BERYLLIUM REFLECTOR	D2 31 1.63	REGULATING ROD #1	D3 30 1.72	D4 27 1.82	SHIM ROD	D5 28 1.59	D6 6% 832 1.43
E1 BERYLLIUM REFLECTOR	E2 11 1.37		E3 33 1.51	E4 34 1.61		E5 24 1.39	E6 9 1.02
F1 FISSION CHAMBER	F2 15 1.09		F3 6 1.18	F4 6% 1051 1.57		F5 18 1.14	F6 5 0.83

GRID LOCATION	FUEL ASSEMBLY #
FUEL ASSEMBLY WORTH	ASSEMBLY POWER PEAKING FACTOR

Figure 3-8K – Mixed Core Test Case No. 9-2.

Mixed Core Test Case No. 9-3

5X5 REFLECTED CORE NO.9-3

Start Date	End Date	Beginning MW-hrs	Ending MW-hrs	Core Excess
N/A	N/A	N/A	N/A	

A1 BERYLLIUM REFLECTOR	A2 BE4 BERYLLIUM REFLECTOR		A3 BE3 BERYLLIUM REFLECTOR	A4 BE2 BERYLLIUM REFLECTOR		A5 BE1 BERYLLIUM REFLECTOR	A6 BE5 BERYLLIUM REFLECTOR
B1 BERYLLIUM REFLECTOR	B2 17 1.37	SAFETY ROD #1	B3 26 1.44	B4 25 1.49	SAFETY ROD #2	B5 19 1.30	B6 21 0.95
C1 BERYLLIUM REFLECTOR	C2 23 1.52		C3 29 1.61	C4 32 1.67		C5 35 1.45	C6 22 1.06
D1 BERYLLIUM REFLECTOR	D2 31 1.65	REGULATING ROD #1	D3 30 1.72	D4 27 1.80	SHIM ROD	D5 28 1.57	D6 20 1.14
E1 BERYLLIUM REFLECTOR	E2 11 1.41		E3 33 1.53	E4 34 1.61		E5 24 1.38	E6 9 1.02
F1 FISSION CHAMBER	F2 6% 879 1.36		F3 6 1.22	F4 6% 1117 1.60		F5 18 1.15	F6 5 407 1.02

GRID LOCATION	FUEL ASSEMBLY #
FUEL ASSEMBLY WORTH	ASSEMBLY POWER PEAKING FACTOR

Figure 3-8L – Mixed Core Test Case No. 9-3.

Mixed Core Test Case No. 9-4

5X5 REFLECTED CORE NO.9-4

Start Date	End Date	Beginning MW-hrs	Ending MW-hrs	Core Excess
N/A	N/A	N/A	N/A	

A1 BERYLLIUM REFLECTOR	A2 BE4 BERYLLIUM REFLECTOR		A3 BE3 BERYLLIUM REFLECTOR	A4 BE2 BERYLLIUM REFLECTOR		A5 BE1 BERYLLIUM REFLECTOR	A6 BE5 BERYLLIUM REFLECTOR
B1 BERYLLIUM REFLECTOR	B2 17 1.28	SAFETY ROD #1	B3 26 1.34	B4 25 1.40	SAFETY ROD #2	B5 19 1.22	B6 21 0.91
C1 BERYLLIUM REFLECTOR	C2 23 1.43	SAFETY ROD #1	C3 29 1.52	C4 32 1.59	SAFETY ROD #2	C5 35 1.39	C6 22 1.03
D1 BERYLLIUM REFLECTOR	D2 31 1.59	REGULATING ROD #1	D3 30 1.67	D4 27 1.78	SHIM ROD	D5 28 1.56	D6 20 1.15
E1 BERYLLIUM REFLECTOR	E2 11 1.40	REGULATING ROD #1	E3 33 1.53	E4 34 1.64	SHIM ROD	E5 24 1.41	E6 9 1.04
F1 FISSION CHAMBER	F2 6% 898 1.40		F3 6% 977 1.55	F4 6% 1131 1.68		F5 6% 871 1.48	F6 6% 381 1.08

GRID LOCATION	FUEL ASSEMBLY #
FUEL ASSEMBLY WORTH	ASSEMBLY POWER PEAKING FACTOR

Figure 3-8M – Mixed Core Test Case No. 9-4.

The MCNP6 model was used to characterize the reactivity behavior of the PULSTAR in its current configuration (Reflected Core No. 8), and possible configurations for Reflected Core No. 9 with six percent enriched fuel assemblies. The excess reactivity, six percent fuel assembly worth, and power peaking factors for each single six percent fuel assembly loading is summarized in Table 3-4C. The license limit for excess reactivity, fuel assembly worth, and power peaking factors are 3970 pcm, 1590 pcm, and 2.92 respectively. Permitted and non-permitted single six percent fuel assembly loading positions are illustrated in Figure 3-8N. The core position is designated in the upper left corner of each cell. Each fuel assembly location designates a six percent fuel assembly loaded in that position with the core excess reactivity in the upper right corner, the fuel assembly worth in the lower left corner, and the power peaking factor, F_Q, in the lower right. Positions where single six percent fuel assembly loading is permitted based on Technical Specifications limits are green, whereas positions where single six percent fuel assembly loading does not met the requirements in the Technical Specifications are colored in red. As the core ages, locations that are currently classified as not permitted may become permitted locations. The model is continuously updated to reflect current conditions and burnup.

Table 3-4C – Summary of Single 6% Fuel Assembly Loading

Core Position Loaded	Excess Reactivity (pcm)	6% Assembly Worth (pcm)	F_Q
B2	3010	1812	2.62
B3	3112	1917	2.73
B4	3135	2036	2.87
B5	3008	1564	2.76
B6	2789	714	2.57
C2	3127	2143	2.63
C3	3061	1856	2.70
C4	3147	2030	2.91
C5	2990	1559	2.81
C6	2847	773	2.53
D2	3048	2282	2.82
D3	3092	1963	2.91
D4	3145	2191	3.15
D5	2986	1706	3.07
D6	2858	803	2.56
E2	3036	1673	2.76
E3	3005	1477	2.83
E4	3025	1634	3.09
E5	2997	1302	3.01
E6	2797	590	2.56
F2	2779	825	2.52
F3	2865	936	2.53
F4	2916	1075	2.55
F5	2829	848	2.52
F6	2701	367	2.52

Mixed enrichment core configuration with multiple six percent enriched fuel assemblies were considered where the six percent fuel assemblies were loaded into permitted core positions indicated in Figure 3-8J through 3-8M. The excess reactivity, control rod worth values, SDM, reactivity insertion rate, and core pin peaking factors are tabulated in Table 3-4D for representative mixed enrichment cores with multiple six percent fuel assemblies as illustrated in Figure 3-8J through 3-8M. The control rods are listed as Safety No. 1 (S1), Safety No. 2 (S2), Regulating (Reg), Shim, and Gang. The power peaking factor is the maximum pin power peaking factor in the core. The license limits (as specified in the Technical Specifications) for excess reactivity, SDM, reactivity insertion rate, and power peaking factor are less than 3970 pcm, less than -400 pcm, less than 100 pcm/s, and less than 2.92 respectively. These core parameters are estimated to be within the safety limits for all multiple six percent fuel assembly mixed enrichment core configurations considered. A sensitivity analysis of the power peaking factor to the critical control rod position demonstrated that the power peaking factor predicted by MCNP6 varies by no more than 5% for ganged control rod positions within 50 pcm of the critical position. Fuel assembly averaged power peaking factors maps for Reflected Core No. 8 and representative Reflected Core No. 9 configurations loaded with multiple six percent assemblies are illustrated in Figure 3-8J through 3-8M. Core positions are indicated in the upper left of each cell while the assembly numeric index for four percent enriched fuel assemblies is indicated in the upper right. The six percent enriched fuel assemblies from the SUNY Buffalo PULSTAR have not yet been given a numeric index but will be given serial numbers of a difference sequence than the four percent fuel assemblies. Fuel assembly peaking factors are indicated in the lower right of each cell and the assembly worth of six percent enriched fuel assemblies is indicated in lower left. The MCNP6 calculated peaking factors for all Reflected Core No. 9 cases are within 10% of those calculated for Reflected Core No. 8. The predicted worth of a single six percent enriched fuel assembly in multiple loading Reflected Core No. 9 configurations considered were determined to be below the 1590 pcm license limit for reactivity insertion by a single fuel assembly.

Table 3-4D – Summary of MCNP6 Core Reactivity Parameters for Mixed Enrichment Core Configurations

Parameter	Limit	Reflected Core 8	Reflected Core 9-1	Reflected Core 9-2	Reflected Core 9-3	Reflected Core 9-4
ρ_{excess}	3970	2604	2895	3160	3214	3735
S1	N/A	2713	2621	2534	2532	2357
S2	N/A	3048	2935	2930	2839	2630
Reg	N/A	3135	3171	3043	3162	3133
Shim	N/A	3596	3594	368	3642	3651
Gang	N/A	8895	8728	8507	8533	8120
SDM	-400	-2812	-2661	-2304	-2156	-1253
ρ	100	68	68	65	65	62
F_Q	2.92	2.56	2.51	2.54	2.59	2.78

Based on the above table, representative Reflected Cores No. 9-1 and No. 9-2 are considered acceptable for loading. The ρ_{excess} and SDM are well within the Technical Specifications limits. Reflected Cores No. 9-3 and No. 9-4 would be precluded based on the power peaking factor (F_Q) as its value falls outside a margin of 15% that is set to be greater than the maximum deviation between the calculated and experimentally estimated values. In this case, if F_Q for Reflected Cores No. 9-3 and No. 9-4 is multiplied by 1.15 it would yield a value greater than limit of 2.92.

Mixed Core Permitted Locations

Figure 3-8N – Mixed Core Permitted Location Map