TEXAS A&M UNIVERSITY
NUCLEAR SCIENCE CENTER REACTOR
LICENSE NO R-83
DOCKET NO. 50-128

SUBMITTAL OF RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION FOR THE CONVERSION FROM HIGH-ENRICHED URANIUM TO LOW-ENRICHED URANIUM FUEL

REDACTED VERSION

SECURITY RELATED INFORMATION REMOVED

Redacted text and figures blacked out or denoted by brakets

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July 17, 2006

U.S. Nuclear Regulatory Commission
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2006-0058

Subject: Texas A&M University System Texas Engineering and Experiment Station Nuclear Science Center (TAMU NSC), Docket 50-128, License R-83. Submittal of Responses to NRC Request for Additional Information and of Revised Tech Specs (RE: High-Enriched to Low Enriched Uranium Conversion for TAMU NSC)

On June 1, 2006 Marvin Mendonca submitted a letter requesting additional information (RAI). The RAI was a part of the NRC review of Texas A&M University System Texas Engineering and Experiment Station Nuclear Science Center's (TAMU NSC) request for conversion of our TRIGA reactor from high-enriched uranium (HEU) to low-enriched uranium (LEU) fuel.

Attached are two documents submitted in support of the TAMU NSC request for conversion. The first document is TAMU NSC's responses to each of the 35 individual questions which made up the June 1, 2006 RAI. The second is a copy of the necessary technical specification changes with error bars required to make this conversion.

If you have any questions, please call me at 979-845-7551.

I certify under penalty of perjury that the foregoing is true and correct. Executed on July 17, 2006.

Sincerely,

W. D. Reece
Director, NSC

Attachments

xc: 2.11/central file
M. Mendonca, NRC Project Manager
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A020

REQUEST FOR ADDITIONAL INFORMATION
TEXAS A&M UNIVERSITY NUCLEAR SCIENCE CENTER REACTOR
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1. **License conditions. Please propose and justify license possession limits for the new low-enriched uranium (LEU) fuel if they are different than your current approved possession limits. Will any additional changes be needed to the facility license to allow for conversion from high-enriched uranium (HEU) to LEU fuel?**

   Response: The TAMU NSC has already sent a letter to the NRC requesting a license to possess the new LEU core prior to conversion.

2. **Section 4.1. What were the bases for validating the neutronic and thermal-hydraulic codes against the Puerto Rico Nuclear Center (PRNC) reactor instead of the Texas A&M reactor? While comparing the results of calculated parameters against measured parameters for the PRNC would serve to validate the use of the codes in general and the ability to develop inputs for the codes, it does not provide validation of the specific input model for the Texas A&M reactor. Provide an assessment of the effects of the design differences between the PRNC and TAMU TRIGA and demonstrate that the differences notwithstanding, the performance of the PRNC FLIP core effectively represents the performance expected of the TAMU FLIP core. Please provide a discussion of the steps taken to ensure the validity of the Texas A&M reactor model and code inputs.**

   Response: The basis for validating the neutronic and thermal hydraulic codes against Puerto Rico Nuclear Center (PRNC) reactor instead of the Texas A&M reactor is twofold:

   1. NUREG-1538 Chapter 18 requires a performance review of the fresh initial fuel. Fresh HEU fuel was never loaded into TAMU but was transferred as partially depleted fuel from PRNC to TAMU. A complete startup with fresh HEU fuel was conducted for PRNC.
   2. The two cores (PRNC and TAMU) were the same essentially identical reactors, using grid plate layouts, control rods, etc. in parallel configurations. The only essential design difference lies in the reflector region. Furthermore, both cores were evaluated also for full water reflector, in order to eliminate any essential difference between the cores.

Both the PRNC and Texas A&M models were developed by the same modeler using the same level of modeling detail thus assuring model fidelity between the two models. The MCNP model of the Texas A&M model was independently verified.

The TAMU core uses fuel-followed control rods unlike the PRNC core. Fuel-followed control rods have a higher reactivity worth which was noted in the calculation. The total control rod worth for PRNC was calculated as $12.00 for five control rods. For TAMU, the total worth is $16.34 for four fuel-followed control rods, a transient rod and a regulating rod. The TAMU core has a transient rod with an air follower while the PRNC transient rod has no follower so that water replaces the transient rod when it is pulsed or removed. Peak power density during pulsing occurs next to the transient rod in PRNC but next to water-filled experiment location in TAMU. The intra-rod peaking factor for PRNC is 1.967 and for TAMU, it was calculated to be 2.297.

As shown below, both cores are similar and in fact the core arrangement presently used at TAMU is not the same as the core arrangement when FLIP fuel was first added. FLIP fuel was added to TAMU to form a mixed core with 35 FLIP fuel elements. Neutronically, a mixed core of FLIP and standard fuel is different from a core with uniform loading of either FLIP or LEU fuel. The PRNC core is neutronically more similar to a uniform core of LEU fuel than a mixed core of FLIP and standard TRIGA fuel.

TAMU Core PRNC Core

A comparison of the FLIP and LEU (30/20) fuel is presented in Table 4-2 which shows a large increase in U-238 due to the decreased enrichment but all other parameters are similar or identical between the two fuel types.

3. **Section 4.5.6. No references, or information on its validity, are given for the BURP code. Please provide such in order to help establish confidence in its veracity.**

Response: The BURP computer code and its validation were documented for the Thailand TRIGA project (Sherman, 2000). Calculations for the Thailand TRIGA project using BURP were independently verified by Argonne National Laboratory (Hanan, 2000).

Sherman, R., "PCBURP Code Validation Report," TRIGA Technologies report 21C024, Rev. 0, April 2000.

Excerpts Comparing ANL and GA Burnup Calculations from *Hanan, N. A., M. M. Bretscher, A. F. DiMeglio, and J. E. Matos, "Preliminary Safety Analysis Report Review addressing Action Sheet 1 between OAEP of Thailand and US DOE," Argonne National Laboratory, June 2000.*

4. **Sections 4.5.1 and 4.5.3. For the PRNC TRIGA core the MCNP5 model calculated criticality with 62 fuel elements in the core whereas the actual criticality was with 59 fuel elements. A calculation of the unrodded PRNC core provided a total reactivity of \$6.26 compared to a measured value of \$7.12. In light of the fact that the TAMU calculational model under-predicts core reactivity significantly, provide verification that the shutdown margin is sufficient.**

   Response: Possible reasons for under-predicting the PRNC full core reactivity were discussed in the conversion analysis. The manufacture of the TAMU fuel will have tighter controls on uranium and erbium content which will reduce the uncertainty and possible deviation of calculated and actual shutdown margin. The discrepancy of \$0.86 in calculated and measured reactivity at PRNC for full core loading is similar to the discrepancy of \$0.64 on calculated and measured reactivity at cold critical. The approach to critical will provide a first indication of calculational accuracy. As noted in the startup plan in section A.1.4, both the approach to initial criticality and full core loading have acceptance criteria. Failure to satisfy an acceptance criterion will require resolution by the startup engineer and appropriate modifications to the startup plan. Full core loading will be performed with two control rods removed which provides assurance that adequate shutdown margin is always maintained. Frequent control rod calibrations during the gradual approach to full core loading will provide further evidence of adequate shutdown margin. The total number of fuel elements loaded into the core may be adjusted from the calculated 90 fuel elements in order to provide adequate shutdown margin. Final control rod calibration after full core loading will verify adequate shutdown margin during startup of the LEU conversion core.

5. **Section 4.5.3. Only integral control rod worths and no differential worths are provided. How is the maximum reactivity insertion rate, when control rods are removed, determined and integrated into the safety analysis?**

   Response: Differential rod worths are not needed to determine maximum reactivity insertion rates for control rod removal accidents since the accident analysis has included pulsing of the transient rod for both cold conditions and full power conditions. Pulsing will bound any credible rod withdrawal event.

6. **Section 4.5.4. Two of the conditions for analyzing shutdown margin are identical. Should one of them be "The highest worth non-secured experiment in its most reactive state."?**

   Response: Agreed. One of the conditions shall be corrected to state "The highest worth non-secured experiment in the most reactive state."

7. **Section 4.5.5. The calculation of β_{eff} utilizes a formula that is based on a core with ^{235}U. What is the origin of this formula and what is its error, in particular when applied to an LEU core?**

Response: In a thermal reactor, delayed neutrons are usually more effective than prompt neutrons in sustaining the chain reaction since their birth at lower energies reduces their probability of leakage before causing a fission. As shown by Henry (1955) and Henry (1958), the effectiveness of delayed neutrons may be easily calculated from a static diffusion calculation in which the fission spectrum, which normally sums to 1.0, is increased by the amount β_o, the physically measured total yield fraction. The calculational model must have enough energy groups so that the different energy distributions of the prompt neutrons and the delayed neutrons may be properly represented. The derivation and comparison with GA methods is given below.

We define:

ν_d ≡ avg. number of delayed neutrons emitted per fission;

ν_d^{eff} ≡ effective avg. number of delayed neutrons emitted per fission;

≡ value of ν_d that would give the real k if delayed neutrons had same spectrum as prompt neutrons;

ν_p ≡ avg. number of prompt neutrons emitted per fission;

r ≡ ν_d / ν_p;

r_{eff} ≡ ν_d^{eff} / ν_p;

$$\beta_0 \equiv \frac{\nu_d}{\nu_p + \nu_d} = \frac{r}{1+r}; \quad \beta_{eff} \equiv \frac{\nu_d^{eff}}{\nu_p + \nu_d^{eff}} = \frac{r_{eff}}{1+r_{eff}} = 1 - \frac{1}{1+r_{eff}}.$$

Also note that:

$$\frac{k_t}{k_p} = \frac{\nu_p + \nu_d^{eff}}{\nu_p + \nu_d} = \frac{1+r_{eff}}{1+r}.$$

It follows that:

$$1+r = \frac{1}{1-\beta_0}; \; 1+r_{eff} = \frac{k_t}{k_p}(1+r).$$

Solve for β_{eff}:

$$\beta_{eff} = 1 - \frac{1}{1+r_{eff}} = 1 - \frac{1}{\frac{k_t}{k_p}(1+r)} = 1 - \frac{k_p}{k_t}\frac{1}{\left(\frac{1}{1-\beta_0}\right)} = \boxed{1 - \frac{k_p}{k_t}(1-\beta_0)}.$$

This differs from the GA formula above, but the difference is small:

$$
\begin{aligned}
dif = \beta_{eff}^{GA} - \beta_{eff}^{MLA} &= \left[\frac{k_t}{k_p}(1+\beta_0)-1\right]-\left[1-\frac{k_p}{k_t}(1-\beta_0)\right] \\
&= \left[\frac{k_p+\delta k}{k_p}(1+\beta_0)-1\right]-\left[1-\frac{k_p}{k_p+\delta k}(1-\beta_0)\right] \qquad (\text{where } \delta k \equiv k_t - k_p) \\
&= \left[(1+\varepsilon)(1+\beta_0)-1\right]-\left[1-\frac{1}{1+\varepsilon}(1-\beta_0)\right] \qquad (\text{where } \varepsilon \equiv \delta k / k_p) \\
&= \left[\cancel{1}+\varepsilon+\beta_0+\varepsilon\beta_0\cancel{-1}\right]-\left[1-\left(1-\varepsilon+O\left(\varepsilon^2\right)\right)(1-\beta_0)\right] \\
&= \left[\cancel{\varepsilon}+\cancel{\beta_0}+\varepsilon\beta_0\right]-\left[\cancel{1-1}+\cancel{\varepsilon}+\cancel{\beta_0}-\varepsilon\beta_0+O\left(\varepsilon^2\right)\right] \\
&= 2\varepsilon\beta_0 + O\left(\varepsilon^2\right).
\end{aligned}
$$

Given the following numbers:

β_0 = *0.0065,*
k_t = *1.0452063,*
k_p = *1.0445676,*

the formulas give:

$\beta_{eff}^{MLA} = 0.00710710$, $\qquad \beta_{eff}^{GA} = 0.00711542$, $\qquad dif = 8.3\times10^{-6}$,

$2\varepsilon\beta_0 = 8.0\times10^{-6}$ = *very close to "dif".*

I am not sure where the GA formula came from; maybe a different definition for β-eff. I am not sure how a different definition would be justified. Maybe the GA formula came from simple approximations, but the formula derived without approximation is just as simple and suffers no more roundoff, so I don't know why one would bother with approximation.

At any rate, the difference is very much in the noise—not of practical concern.

- Marvin Adams, Prof of Nuclear Engineering, Texas A&M University

Henry, A. F., "Computation of Parameters Appearing in the Reactor Kinetics Equations," Westinghouse Bettis Laboratory report WAPD-142, December 1955.

Henry, A. F., "The Application of Reactor Kinetics to the Analysis of Experiments," Nuclear Science and Engineering, 3, 1958, p. 52.

8. **Section 4.5.5. The numbers quoted for the prompt temperature coefficient on page 33 are inconsistent with those given on Figure 4.14. Please discuss.**

Response: In Fig. 4.14, it can be seen that the prompt negative temperature coefficient (a) for LEU (30/20) fuel has only a modest decrease in values at 2000 MWD burnup (EOL) (e.g., 13.1×10^{-5} to 9.9×10^{-5} $\Delta k/°C$ at 700 to 1000°C. As illustrated in the TAMU SAR, the corresponding decrease for FLIP fuel is much larger, as an example, 17×10^{-5} to 4×10^{-5}

$\Delta k/^{\circ}C$ at 800°C for 2000 MWD burnup. The exponents in the prompt temperature coefficients on page 33 should be 10^{-5} instead of 10^{-4}.

9. **Section 4.5.6. The conversion SAR states that the placement of fresh fuel into the center of a high burnup core raises concerns for excessive power peaking and that shuffling lower burnup fuel from the perimeter of the core to the core center and adding fresh fuel to the core perimeter addresses this concern. How will this movement be controlled? Should a TS requirement be added? Also, it is possible that thermocouple failures in an instrumented fuel element (IFE) could require replacement of an IFE. Because IFEs tendto be placed in the center of the core, there may not be a low burnup IFE at the perimeter of the core to move to the center. Discuss how IFE replacement will be accomplished without raising a concern about power peaking.**

   Response: Excessive power peaking is controlled by two Tech Specs. Tech Spec 2.2 controls excessive power peaking during normal operation by limiting the maximum fuel temperature to less than the LSSS of 525°C. Tech Spec 3.1.2 controls excessive power peaking during pulsing by limiting the reactivity insertion so that the resulting pulse will not produce a peak fuel temperature exceeding 830°C. The addition of a fresh IFE in a high burnup core is addressed in Section 12.6.

10. **Sections 4.5.9 and 4.5.11. The conversion SAR discusses the desirability of placing the IFE at certain locations within the core. It does not appear that a conclusion is reached. What are the proposed locations of IFEs in the LEU core and why are these locations used?**

    Response: The proposed locations of the IFEs are 5E4 and 6D4 which satisfy the Tech Spec requirement for IFE placement and are the closest possible IFE locations to the fuel element with the highest power factor. While it would be scientifically interesting to place an IFE in 4D4, Tech Spec 3.1.4 precludes placing an IFE in that location.

11. **Section 4.6. This section discusses the addition of an electro-mechanical interlock on the transient rod to limit reactivity additions. How does this differ from the system required by existing TS 3.1.2.a, which appears to be a mechanical system? If the electronic interlock is a new feature, discuss the need for a TS LCO requirement with an associated surveillance requirement.**

    Response: Upon further review, the TAMU NSC management determined that the above addition of an electrical-mechanical interlock for the transient rod during reactor pulsing is not a conversion issue. This interlock will be discussed in the application for re-licensing of the facility. Current administrative precautions and mechanical interlocks currently installed are sufficient to preclude accidental pulses (Reactivity insertion greater than $2.95) from exceeding the Safety Limit.

    Current procedures for pulsing require the insertion of a mechanical pulse-stop prior to pulsing to preclude pulsing in excess of $2.00 or lower administrative limits. Procedurally, the transient rod cylinder is restricted from being raised above that rod height which would cause a reactor pulse in excess of $2.00 or the administrative limit. Current Technical Specification Surveillance Requirements (4.2.3) ensure that the reactivity worth of control

rods is determined on an annual basis. This aids in verifying that the reactivity insertion during a pulse will not be greater than $2.95.

By procedure for a startup to steady-state power the transient rod is raised to fully withdrawn before other rod movement. Also, a current interlock is in place to prevent pulsing from an initial reactor power greater than one (1) kilowatt, although it has been satisfactorily demonstrated that a pulse from full power (1 MW) would not cause a violation of the safety limit.

Due to the above mentioned controls, an accidental pulse leading to the violation of the Safety Limit is highly improbable. These controls, along with a proven track record, indicate that an electro-mechanical interlock is not required. Exceeding the Safety Limit during a pulse would require intentional disregard and violation of current procedures and interlocks. The intent is to remove statements from the Conversion SAR referring to the electro-mechanical interlock from sections 4.5.11, 4.5.12, 4.6, 6.0, 7.0, and 13.5.

Texas A&M University Nuclear Science Center intends to continue with the installation of an electro-mechanical interlock which prevents pulsing without the pulse-stop installed. This change will be incorporated in the application for re-licensing. TAMU intends to submit a report and record of this change in accordance with 10 CFR 50.59. The change will incorporate revisions to the Technical Specification *Limiting Conditions for Operations* requiring the electro-mechanical interlock to be active and operable during pulse-mode operations. A revision will also be made to the *Surveillance Requirements* to verify proper operation of the interlock prior to pulsing.

12. Section 4.7.2. Provide a discussion on the verification and validation of the STAT code for thermal-hydraulic conditions applicable to the PRNC and the TAMU reactors.

Response: The STAT code was developed by J. F. Petersen and uses the methodology that he used to perform the thermal analysis of using TRIGA fuel elements in MTR reactors (Petersen, 1965). Although the available data did not in all cases correspond to the conditions being analyzed, the data were applied in such a way as to make the method conservative. The method was applied to the Torrey Pines TRIGA reactor to calculate the flow rate in the coolant channel between the innermost (B and C) rings of fuel elements. The calculated flow rate was 35% lower than that deduced from the measured channel exit temperatures. The STAT code and its associated methodology have been applied to the licensing of TRIGA reactors for the past 40 years.

Petersen, J. F., "Steady State Thermal Analysis for the Proposed Use of TRIGA Fuel Elements in MTR Reactors," General Atomic report GA-5708, April 1965.

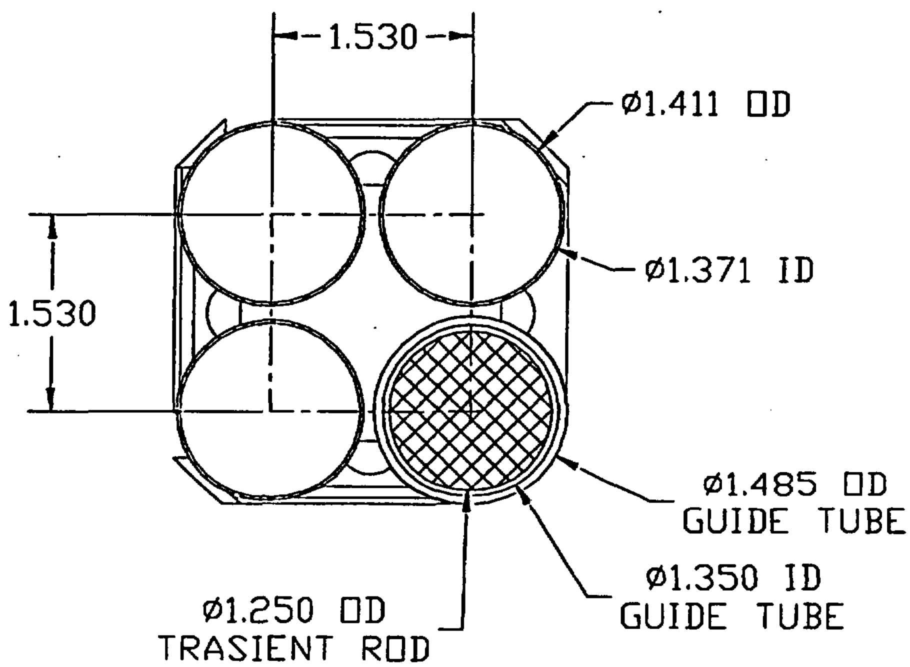
13. Section 4.7.2. There were four fuel elements associated with each coolant channel modeled by STAT. Were all fuel elements in each problem assumed to have the same rod power? By symmetry each fuel element is bounded by four coolant channels. How was the maximum powered channel determined?

Response: The STAT model examines a single fuel element and a representative flow area associated with that single fuel element. This approach is conservative since the flow

area is not shared by high powered and lower powered fuel elements. The maximum powered fuel element was determined by the DIF3D diffusion code.

14. Section 4.7.2. There are at least three different channel flow areas in the TRIGA core and each is associated with different bundle geometry, the four-element bundle, the three-element bundle (with fuel-followed control rod), and the three-element bundle with guide tube (for regulating rod or transient rod). Did the flow area used in the STAT analysis take into account the location of the fuel element in the core? Was a limiting flow area used in the analysis?

Response: The response to question 13 also applies to this question. For both the average and maximum powered fuel rod, a representative flow area was used. As shown in the figure below, the guide tube for the transient rod has an OD of 1.485 in. which is slightly larger than the fuel element OD of 1.411 in. The guide tube reduces the flow area of the maximum powered fuel rod by about 3.5%. The flow rate for the maximum rod reduces from 0.089 kg/s to 0.087 kg/s. The effect on the MDNBR is a reduction from 2.42 to 2.34.



15. Section 4.7.2. Provide justification for ignoring diversion of coolant to neighboring flow channels.

Response: Cross flow from one flow channel to another is ignored by restricting the flow modeling to a single fuel element. Cross flow would allow flow mixing from cooler, lower powered flow channels into the maximum powered flow channel. The total pressure drop in the average and maximum fuel rod channel is balanced by the pressure gain due to buoyant forces so that the pressures at the exit of the average and hot channels are equal. This is described more fully in Petersen, 1965. The total pressure drop reported by STAT does not include the pressure change due to elevation and density differences between

the hot fluid in the core and the cold fluid outside of the core which determines the pressure of the water at the core inlet. The hydrostatic head is not counted as part of the pressure loss but as part of the buoyancy pressure gain. The buoyancy pressure gain is simply as follows:

$$\Delta P = (\rho c - \rho h)\,\Delta z$$

STAT integrates the density changes as the coolant passes up through the core.

Pressures at different elevations of the average and hot channels are also essentially equal due to the balance of pressure losses and buoyancy pressure gains. As noted in a Sandia report (Rao, 1994), various experiments have revealed that this cross flow is negligible for tightly packed geometries such as PRNC and TAMU. The references for these experiments are Becker (1969), Silvestri (1966), and Gaspari (1974). The ultimate effect of cross flow mixing is to increase CHR, so that its neglect through the use of subchannel approach is expected to result in conservative estimate of CHF.

Rao, D. V., and M. S. El-Genk, "Critical Heat Flux Predictions for the Sandia Annular Core Research Reactor," Sandia report SAND 90-7089, August 1994.

Becker, K. M., G. Hernborn, M. Brodl, and G. Erikson, "Burnout Data for Flow of Boiling Water in Vertical Round Ducts, Annuli and Rod Clusters, AE-177, AB Atomenergi, Sweden, 1969.

Silvestri, M., "On the Burnout Equation and on the Location of Burnout Points," Energia Nucleare, Vol. 13, No. 9, pp. 469-479, 1966.

Gaspari, G. P., A. Hassid, and F. Lucchini, "A Rod Centered Subchannel Analysis with Turbulent Mixing for Critical Heat Flux Prediction in Rod Clusters Cooled by Boiling Water," Paper B6.12, Proc. Fifth Int. Heat Transfer Conf., 1974.

16. Section 4.7.2. Provide the units and reference for the boiling heat transfer correlation in Section 4.7.2 of the SAR. Is the correlation for pool boiling or flow boiling? If it is for flow boiling, provide justification for not using a pool boiling correlation that is more applicable to natural convection cooling.

Response: T_w and T_{sat} are in degrees F. $Q_{boiling}$ is in Btu/hr-ft^2. The reference for the correlation is McAdams. The correlation is for subcooled boiling in a narrow vertical annulus at pressures of 2-6 atm and flow velocity of 1 to 12 ft/sec. The natural circulation flow loop established due to buoyancy difference between the coolant heated in the core and the cold pool water behaves like a low-head pump generating a low velocity flow through the core. It does not behave like free convection from either a horizontal or vertical surface. Under normal operating conditions of 1 MW, the flow velocity in the maximum powered channel is 0.5 ft/s but increases to 1 ft/s for DNB conditions as determined in the thermal hydraulic calculations using STAT. A similar correlation is found in the Handbook of Heat Transfer, eds. Roshenow and Bartnett and was used in the Sandia report referenced in question 15 (Rao, 1994).

McAdams, W. H., "Heat Transmission," 3rd Edition, McGraw-Hill Books, New York (1954).

17. Section 4.7.2. The single-phase heat transfer correlation shown in Section 4.7.2 is for turbulent forced flow. Provide justification for using a forced convection heat transfer correlation in a problem dominated by natural convection.

Response: As noted in the response to question 16, a natural circulation flow loop is established through the compact TRIGA core. The Reynolds number is much higher than laminar. Depending on power level, the Reynolds number would either be in the turbulent regime or at the high end of the transition regime. Typically, the TRIGA reactor operates in a mixed convection regime. The subcooled boiling correlation is used when boiling heat transfer is occurring. The single-phase heat transfer correlation has no safety significance since it is only used where a single phase exists. Further validation of the convective boundary condition on the fuel is provided by comparison of predicted versus measured fuel temperatures. As part of a UNM-SNL-GA joint research program (Kim, 1988), both turbulent flow and natural convection correlations were found to give similar predictions of the heat transfer coefficient for the ACRR conditions which are similar to the TAMU conditions.

Kim, S.-H., M. S. El-Genk, R. A. Rubio, J. W. Bryson, and F. C. Foushee, "Heat Transfer Experiments and Correlations for Natural and Forced Circulations of Water in Rod Bundles at Low Reynolds Numbers," 11th TRIGA User-Owner Conference, April 10-13, 1988, Bethesda, MD.

18. Section 4.7.2. Confirm that the flow is in the turbulent regime and not in the transition or laminar flow regime such that a turbulent friction factor and heat transfer correlation are applicable.

Response: For the maximum powered fuel element at 1 MW operation, the Reynolds number at the peak flux location is 7,000 which would be considered at the high end of the transition regime. Under DNB conditions, the Reynolds number at the same location would be 17,000 which is in the turbulent regime. The pitch-to-diameter ratio is typically about 1.08 which has been shown in a UNM study (Kim, 1988) to transition to the turbulent regime at Reynolds numbers less than 2000. This study also showed that turbulent flow and natural convection correlations were found to give similar predictions of the heat transfer coefficient.

19. Section 4.7.2. Demonstrate that both the McAdams and the Bernath correlation are applicable to natural convection boiling and the predicted CHF is conservative as compared with the one using a pool boiling CHF correlation.

Response: Pool boiling does not describe the flow and heat transfer in the closely packed fuel elements of a TRIGA reactor. Both McAdams and Bernath CHF correlations have been used in TRIGA licensing for decades. The McAdams burnout heat flux is applicable to a narrow vertical annuli at low flow velocities near atmospheric pressure. The Bernath correlation encompasses a wider range of variables and takes into account the effect of different flow geometries. The Bernath correlation generally gives a lower value for CHF. The references below suggest that the McAdams and Bernath models agree to within 10% for 1-12 ft/s flow in narrow annuli.

Bernath, Louis, "Predictions of Heat Transfer Burnout", Chemical Engineering Progress Symposium Series No. 18 Vol. 52, pp. 1-6, 1956.

Bernath, Louis, "Theory of Local-Boiling Burnout and it Application to Existing Data", Chemical Engineering Progress Symposium Series No. 30 Vol. 56, pp. 95-116, 1960.

20. **Section 4.7.3. In a transient, such as in pulsed operation, the rate of energy generation in the fuel rod is not equal to the rate of heat transfer to the coolant. How is the coupling between the TAC2D calculation and the thermal-hydraulic calculation (coolant flow rate, coolant temperature, and surface heat transfer coefficient) handled in a transient?**

    Response: There is no coupling between the TAC2D calculation and the thermal-hydraulic calculation during a pulse transient. TAC2D was only used for steady-state heat transfer calculations. The short time-scale of the pulsing transient is such that heat transfer to the coolant is not significant but is accounted for in the BLOOST calculation.

21. **Section 4.8. Calculations were performed for the LEU core at a power level of 1 MW. However, TS 3.1.1 allows the reactor to be operated at power levels up to 1.3 MW. The TS does not appear to have a constraint on the amount of time the reactor can be operated at this power level. Please provide steady-state thermal-hydraulic calculation results for a 1.3 MW power level.**

    Response: Table 4-11 provides steady-state thermal-hydraulic results for operation at 1.3 MW power level. In addition, Section 4.8.4 show that the reactor does not reach DNB until the reactor power reaches 2.42. The results show that TS 3.1.1 does not require a time constraint.

22. **Section 4.8.3. It is noted on page 39 of the SAR that based on the under prediction of the reactivity loss with increasing power it is suggested that the actual average core temperature is under predicted by 30-60 C at 1.0 and 1.4 MW for the PRNC core. This assertion appears to contradict the good agreement between the measured and predicted temperature for the instrumented fuel element. Quantify the uncertainty in the predicted peak fuel temperature if the predicted average fuel temperature is attributed to have an error in the range of 30-60 C.**

    Response: Several factors contribute to the calculation of an effective average core temperature and a simple average temperature may not truly represent the core-wide reactivity change. The good agreement in measured and predicted temperature as shown in Table 4-8 provides assurance that peak fuel temperature would have similar good agreement since the IFEs are located in the best possible positions to represent the peak fuel temperature.

23. **Section 4.8.4. In the TAMU LEU (30/20) core, the peak power density does not occur in the fuel element with the maximum rod power. Which rod has the peak fuel temperature and the MDNBR, the rod with the maximum rod power or the rod with the peak power density? Is the fuel element in position 4D3 the one experiencing the peak power density at 1.0 MW steady state and in a pulse operation?**

    Response: As stated on page 43 (Ref. Fig. 4.2), the fuel element (5D3) immediately adjacent to the transient rod (5D4) produces the greatest power (17.4 kW/element) and

peak fuel temperature (Table 4-11). The MDNBR is calculated for this fuel element. A three rod locking plate on the transient rod cluster precludes placing an IFE within this cluster. The IFE is located at the next nearest location, 5E4 where the power generated is 15.4 kW/element.

Water in position 3D leads to power peaking especially in 4D3 and 4D4. The peak power density in 4D3 is significant only to pulsing operation. The rod power factor at this location is 1.446 which is less than the rod power factor of 1.565 in location 5D3.

24. **The references in Section 13 of the SAR need to be corrected. What are referred to in the text as References 13, 14, 15, and 16 should be References 2, 13, 14, and 15, respectively. In addition there is an unknown Reference 8-3 cited in the text.**

    Response: Agreed. The references shall be corrected to References 2, 13, 14, and 15. The unknown reference listed as Ref. 8-3 should not be included because it was calculated for the LEU conversion. Starting with 2000 MWD burnup for the LEU core and 90 fuel elements and peak power of 1.55, it was determined that the burnup capability based on 50% of the U-235 gives 57 MWD/fuel element.

25. **Section 13.3. The conversion SAR discusses the safety margin to account for accuracy of fuel temperature measurements and overshoot in reactor power resulting from a reactor transient during steady state mode operation. However, there does not appear to be any justification for this statement. Please show that sufficient safety margin exists.**

    Response: The conversion SAR calculates the overshoot in reactor power from a reactor transient during steady state mode operation. The calculation shows that the peak fuel temperature has a margin of 550°C, which is clearly sufficient.

    The basis for Tech Spec 2.2 discusses various errors in measuring temperature in the core including any overshoot in reactor power resulting from a reactor transient during steady state mode of operation. A minimum safety margin of 10% was applied on an absolute temperature basis. The final LSSS of 525°C, when viewed on an absolute temperature scale represents a 37% safety margin. In the pulse mode of operation, the LSSS on temperature will have no effect on limiting peak powers generated because of its relatively long time constant as compared with the width of the pulse. A temperature trip will act to reduce the amount of energy generated in the entire pulse transient by cutting the tail off the energy transient.

26. **The basis for Technical Specification 14.3.1.2, Pulse Mode Operation, discusses the effect on fuel of temperatures exceeding 874 C. In light of this fuel behavior, discuss whether a lower temperature should be the safety limit which currently is 1150 C for any conditions of operation.**

    Response: Tech. Spec. 14.3.1.2, Pulse Mode Operation discusses the effect on fuel of temperatures exceeding 874°C. Two different effects are involved. The Safety Limit has been evaluated at about 1300°C from which GA initially chose 1150°C with already a large safety margin as accepted in NUREG-1282.

The 874°C limit addresses repetitive pulsing of a heavily burned, high power density core. These repeated pulses can slowly deform a fuel element. A single accidental pulse exceeding 874°C does not damage such a fuel element and leaves the 1150°C Safety Limit unchanged. The basis for the 874°C limit is from the fuel damage that occurred at Texas A&M in 1976 after the pulse limit was increased from $2.00 to $2.70 the previous year. The analysis of this damage is discussed in NUREG/CR-2387 and described in detail in a General Atomics document (Simnad, 1981).

Simnad, M. T., G. B. West, J. D. Randall, W. J. Richards, and D. Stahl, "Interpretation of Damage to the FLIP Fuel During Operation of the Nuclear Science Center Reactor at Texas A&M University," General Atomics document GA-A16613, December 1981.

27. **Analysis of pulse operation shown in Sections 4.5.11 and 4.5.12 of the conversion SAR showed that peak fuel temperature occurred in core location 4D3, not an instrumented fuel element. Provide an uncertainty analysis for the prediction of the peak fuel temperature in the pulse mode of operation. The analysis should also account for uncertainties in calculated rod power and axial power profile. Do the uncertainties reduce the safety factor significantly?**

    Response: The axial power distribution of core location 4D3 is essentially the same as the axial power distribution in the IFE locations. In general, calculational errors in the flux distribution (axial or radial) will be observed in the temperature measurements at the IFE location 5D4 due to its proximity to core location 4D3.

    Possible variation in uranium and erbium loadings is the most probable source of uncertainty. They can cause an error in calculated power of around 2% between the peak location in 4D3 and the IFE in 5D4. The temperature rise at 4D3 due to $1.95 pulse is 728°C. A 2% error would result in a temperature rise of 743°C or an increase of 15°C. An uncertainty of that magnitude does not significantly reduce the safety factor.

    Measured and calculated temperatures of pulses in PRNC are shown in Table 4-12. The results show the effect of conservative assumptions on the calculated temperatures. The temperature calculations assume that the fuel is essentially adiabatic without any allowance for increased heat transfer. In particular, the power distribution within a fuel element results in peak temperatures near the surface of the fuel. Increases in heat transfer to the clad and coolant are conservatively ignored which maximizes calculated fuel temperatures.

28. **TS 2.1. Does the peak fuel temperature in a reactivity pulse transient depend on the initial power level? If yes, was the analysis performed with a conservative initial power? The proposed Technical Specification 14.3.2.2 has an interlock included that limits pulsing from any power greater than 1kW. How was this value obtained?**

    Response: Forty years ago, GA chose 1 kW as the maximum steady state power to initiate a pulse. This limit was chosen so that the core fuel elements would have only ambient fuel temperature. In other words, the fuel and cooling water will be the same temperature. The analysis was performed with the above limited initial power. Below 1 kW, there is no change in peak fuel temperatures during a reactivity pulse.

29. **Section 13.5. A safety limit of 950 C is cited in NUREG-1282 when the clad temperature equals the fuel temperature. This is the likely condition for the air cooling of fuel stipulated for the LOCA event by TAMU. What is the fuel temperature limit used in the determination of the power/rod limit for the LOCA event? Explain if the fuel temperature limit for the LOCA is consistent with the technical basis for the safety limit of 1150 C.**

    Response: During a hypothetical LOCA with fuel heatup and air cooling, the integrity of the fuel is maintained for all fuel temperatures up to 950°C. For the LEU 30/20 core operating at 1 MW, the largest power per element is 17.4 kW/element. This is well below the 21 kW/element noted in Foushee as not requiring any delay time between reactor scram and complete loss of coolant. It should be noted that it is incredible to drain over 100,000 gallons of coolant water from the reactor tank in zero time. In conclusion, the LOCA for the 1 MW LEU core is conservative.

    Safety limits are limits on process variables to protect barriers that guard against uncontrolled release of radioactivity. The limiting safety system settings (LSSS) are selected to provide automatic protective action to prevent the measured value of any of these Safety Limit process variables from reaching the Safety Limit. During a hypothetical LOCA which is an incredible event, there are no protective actions required to keep the fuel below 950°C so that there is no need to formally identify it as a Safety Limit in the Technical Specifications.

    Foushee, F. C., "TRIGA Four-Rod Cluster Loss of Coolant Accident Analysis," GA report no. E-117-196 (October 1972).

30. **Section 13.5. Explain the relation of the pulsing accident analyzed in Section 13.5 of the conversion SAR and the pulse operation for BOL and EOL conditions discussed in Sections 4.5.11 and 4.5.12 respectively. The accident analysis used a reactivity limit of $2.95 and a safety limit of 1150 C while the pulse operation limited the reactivity insertion to $2.1 and a technical basis that relies on a temperature limit of 830 C. In view of the LCO imposed on pulse operation explain the basis for using the 1150 C safety limit in a pulsing accident.**

    Response: GA has developed a pulsing analysis in which the redistribution of heat in the pulsed fuel element occurs. The redistribution of heat and hydrogen inside the fuel takes time. GA was able to show that the true Safety Limit is likely nearer 1300°C. However, for conservatism, GA has taken 1150°C as the Safety Limit for all 65 TRIGA reactors.

    The pulse limit of 830°C is aimed at protecting the heavily burned fuel from damage to slowly occurring fuel changes during burnup. Accidental pulsing to high temperature (≤ 1150°C) will not rupture the fuel cladding. Therefore, 1150°C remains as the Safety Limit in a pulsing accident.

    The reactor can be repeatedly pulsed to $2.1 without causing any fuel deformation as has been observed. However, 830°C is not a safety limit and exceeding 830°C in either a lightly burned core or a few times with a heavily burned core will not cause fuel bending.

31. **Section 13.5. In both the 2003 NSCR SAR submittal and the HEU/LEU fuel conversion submittal, a release fraction of 2.6×10^{-5} is used, based on a General**

Atomics experiment. Please provide the technical bases for how this release fraction was determined and why/how it is applicable to the NSCR. If analytical and experimental results are used, as they were in this section of the HEU/LEU fuel conversion document and in the 2003 SAR submittal, they should be cited by reference and adequate technical justification provided for the assumptions used in any analysis.

Response: A summary report of these studies (Foushee, 1971) indicates that release from the $UZrH_{1.6}$ fuel meat at the steady-state operating temperatures is principally through recoil into the fuel-clad gap. At high temperatures (above 400 to 500°C), the release mechanism is through a diffusion process and is temperature-dependent, unlike recoil.

In steady-state operation, the peak fuel temperature establishes the fission products in the fuel-clad gap. Since the axial fuel temperature determines the fission product release, there is no further latitude in fission product release. For 1 MW operation, the peak integrated fuel temperature corresponds to a release fraction of 2.6x10-5.

Foushee, F. C., and R. H. Peters, "Summary of TRIGA Fuel Fission Product Release Experiments," Gulf Energy & Environmental Systems report, Gulf EES-A 10801, 1971.

32. Section 13.5. In this section (analysis of the MHA) it is stated that in the TAMU SAR for re-licensing submitted to the NRC in 2003 (2003 SAR), an individual lingering in the Reactor Hall for one hour would receive a 49 Rem thyroid dose. It is then stated that "It might be better to indicate that workers in the Reactor Hall when the fuel element ruptures would promptly leave this hall in less than an estimated 5 minutes (not 1 hour), thus receiving a thyroid dose of 4.1 Rem." On this basis, the HEU-LEU conversion SAR concludes that the prior analysis submitted in the 2003 SAR for one TRIGA FLIP fuel element bounds the results expected for the LEU (30/20) core.

a. Provide a justification for changing the stay time for a worker in the Reactor Hall from 1 hour to 5 minutes.

Response: For an MHA consisting of fission products leaking from a damaged fuel element in air and with the radiation alarm sounding, there never was a justification to stay in the reactor hall for 1 hour. In all reactor facilities all workers are trained to leave the reactor hall as quickly as possible when the radiation alarm sounds. All workers will leave the reactor hall in less than 5 minutes in accordance with training and confirmed by emergency drills.

b. Since the fission product inventory for LEU fuel is different than for HEU fuel, please provide the technical basis and/or references for the radiological inventory used to calculate the doses in Table 13-1 of the 2003 SAR, as well as the technical justification or rationale for the assumptions and boundary conditions applied in the calculation of those doses. Discuss why the use of the HEU fuel analysis is valid for the LEU fuel or provide and justify a radiological inventory based on LEU fuel.

Response: On a per MWD basis, the fission product inventory for TRIGA HEU fuel and LEU fuel are negligible different. For TRIGA HEU fuel, less than 1% of the fission power

at EOL has come from Pu-239. However, for TRIGA LEU fuel, the percentage of fission power from Pu-239 increases to less than 4%. In both fuels, almost all of the fission product inventory is due to U-235 fission. Any inventory difference between HEU and LEU fuel for a given burnup is minor compared to other uncertainties in the assessment of radiological dose.

The fission product inventories in the below table were calculated using ORIGEN. Both FLIP and LEU (30/20) inventories were calculated for a single fuel element. The element was assumed to have a conservative power density of 28 kW. The burnup calculation was performed for 200 days (5.6 MWD) in order to achieve saturation levels for all of the isotopes except Kr-85. The burnup calculation for Kr-85 was extended to 77 MWD for FLIP fuel and 54 MWD for LEU (30/20) fuel.

Isotopes	FLIP Curies	LEU (30/20) Curies
Br-83		
Br-84		
Br-85		
I-131		
I-132		
I-133		
I-134		
I-135		
I-136		
Kr-83m		
Kr-85m		
Kr-85		
Kr-87		
Kr-88		
Kr-89		
Kr-90		
Kr-91		
Xe-131m		
Xe-133m		
Xe-133		
Xe-135m		
Xe-135		
Xe-137		
Xe-138		
Xe-139		
Xe-140		

c. For the LEU fuel, please provide the thyroid and whole body doses for an individual in the unrestricted area receiving the maximum dose, at the nearest *permanent residence and to an individual remaining in the Reactor Hall.* It can be in a format similar to that of Table 13-1 in the 2003 SAR submittal.

Response: The modern analogs to WBGD and thyroid dose are effective dose equivalent and the committed dose equivalent to the thyroid. Since we cannot change SAR now, we show here that the dose from 30/20 will not increase dramatically.

If we take into account only the differences in the source terms from FLIP to 30/20, then attached will be the new table 13-1.

A worst case scenario "assuming everything fails" and following the methodology outlined in NUREG/CR2387 yields 12 hr EDE (WBGD) outside when ventilation system is shutdown as 0.0396 mrem; the corresponding thyroid dose is 17.8 mrem; exposure to personnel for 1 hr after release is close to 42800 mrem (taking into account factor of 200 dilution given in our SAR).

Using the HOTSPOT code (Lawrence Livermore) that calculates dose from plumes of radioactive material we calculated doses for several scenarios. For example, for Xe-133 for 12 hr release the TEDE is 2.54E-05 rem. The corresponding committed dose equivalent to thyroid is about 21 mrem. Dose to thyroid for personnel in one hr after release is 30,000 mrem (taking into account the 200 dilution factor from outside to inside).

In summary, we can confidently say that the new 30/20 dose will not increase significantly from the existing data.

Table 13-1: Summary of Radiation Exposures Following Cladding Failure of the highest Power Density 30/20 FUEL Element

		WBGD (mrem)	Thyroid Dose (mrem)
A.	**Building Ventilation Operating:**		
1	Maximum Exposure to population Outside Building		
	Pool water remaining	4.00E-03	
	Pool water Drained	1.60E-02	4.25
2	Exposure to operating Personnel in 1 hr after release		
	Pool water remaining	0.966	
	Pool water Drained	2.01	12.1
B.	**Building Ventilation Shut Down**	**WBGD (mrem)**	**Thyroid Dose (mrem)**
1	Maximum Exposure to Population Outside Building (12 hrs)		
	Pool water remaining	4.14E-03	
	Pool water Drained	2.42E-02	20
2	Exposure to operating Personnel in 1 hr after release		
	Pool water remaining	2	
	Pool water Drained	4.8	56 R

33. Please provide a copy of your proposed TS that contains change bars in the margin indicating areas of change. Also, for each proposed change in the TS, please provide a justification for requesting each specific change. To allow NRC to complete its initial review of your proposed TS changes, could you please submit this information as soon as possible.

Response: Agreed. Find attached

34. **TS 14.2.2 If removing the LSSS setting of 125% of 1 MW was intentional, why is it discussed in the Basis for this Technical Specification?**

Response: This is not a conversion issue, but a relicensing issue. Proposed changes to the Technical Specifications have been removed.

35. **TS 14.3.1.2. If the $2.00 limit is still valid, why was it removed from the Technical Specification, and if it has been intentionally removed, why is it still discussed in the <u>Basis</u>?**

Response: This is not a conversion issue, but a relicensing issue. Proposed changes to the Technical Specifications have been removed.

Description of Proposed Changes to Technical Specifications for TAMU NSCR TRIGA Reactor

The following is a list of changes to the Technical Specification submitted with the Safety and Accident Analyses Report for the Texas A&M University Conversion from HEU to LEU Fuel. This list is intended to note and provide justification for differences between the submitted Technical Specifications for the conversion and the Technical Specifications currently in effect dated 9-1-1999. Changes made are annotated in the revised Technical Specifications by use of sidebars in the right margin.

The *Table of Contents* has been modified to incorporate the following changes:

1.9 FLIP Core Definition - REMOVED
1.12 LEU Core Definition - ADDED
1.16 Mixed Core Definition - REMOVED
1.40 Standard Core – REMOVED

As a result of the removal and addition of the above mentioned definitions, items 1.9 – 1.12 and items1.16 – 1.43 have been re-numbered to accommodate these changes. This re-numbering is not reflected below.

Below are the changes as they occur and the associated reason for the changes. **The item number refers to the Technical Specifications dated 9-1-1999.**

Item	Purpose
1.9	The definition for *FLIP Core* has been removed due to the core conversion.
1.11	The definition for *Fuel Element* has been changed to incorporate LEU fuel and remove reference to standard or FLIP fuel.
1.16	The definition for *Mixed Core* has been removed due to the core conversion.
1.20	The definition for *Steady State Operational Core* has been changed to incorporate LEU fuel and remove reference to standard or FLIP fuel.
1.40	The definition for *Standard Core* has been removed due to the core conversion.
2.1	The Specification and Bases for *Safety Limit Fuel Element Temperature* have been changed to reflect conversion to the LEU core. Reference to standard to FLIP fuel has been removed.

2.2 The Basis for *Limiting Safety System* has been changed to reflect analysis of LEU fuel. Reference to standard to FLIP fuel has been removed.

3.1.2 The Specification for *Pulse Mode Operation* item b has been changed to incorporate LEU fuel and remove reference to FLIP fuel.

The Basis for *Pulse Mode Operation* has been changed to reflect the hydrogen-to-zirconium ratio for LEU fuel. The associated values for FLIP and standard fuel have been removed.

3.1.4 The Applicability for *Core Configuration Limitation* has been changed to incorporate the LEU core and removes reference to FLIP, standard and mixed cores.

The Objective for *Core Configuration Limitation* has been changed to incorporate the LEU core and removes reference to FLIP and mixed cores.

The Specifications and Basis for *Core Configuration Limitation* item a has changed to allow for only LEU cores. References to FLIP, standard and mixed cores has been removed.

5.1 The Specifications and Bases for *Reactor Fuel* has been changed to reflect the design features of TRIGA-LEU fuel and remove reference to standard and FLIP fuel.

5.2 The Bases for *Reactor* Core has been changed to reflect the design features of LEU cores and remove reference to standard, FLIP and mixed cores.

5.3 The Bases for *Control Rods* has been changed to reflect design features for fuel-followers in an LEU core region and removes reference to FLIP core regions.

5.7 The Bases for *Reactor Pool Water Systems* item a has been changed to reflect analysis performed for the TAMU TRIGA-LEU core. References to standard, FLIP and mixed cores have been removed.

TECHNICAL SPECIFICATIONS

FOR THE TEXAS ENGINEERING EXPERIMENT STATION

TEXAS A&M UNIVERSITY SYSTEM

NUCLEAR SCIENCE CENTER

REACTOR FACILITY

DOCKET NO. 50-128
LICENSE NO. R-83
MARCH 1983
REVISED THROUGH AMENDMENT NO. 15

TECHNICAL SPECIFICATIONS FOR THE

NUCLEAR SCIENCE CENTER REACTOR

FACILITY LICENSE NO. R-83

March 1983

Revised through Amendment No. 15

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1.5 Core Lattice Position

The core lattice position is that region in the core (approximately 3" x 3") over a grid plug hole. It may be occupied by a fuel bundle, an experiment, or a reflector element.

1.6 Experiment

An operation, hardware, or target (excluding devices such as detectors, foils etc.) which is designed to investigate non-routine reactor characteristics or which its intended for irradiation within the pool, on or in a beam port or irradiation facility and which is not rigidly secured to a core or shield structure so as to be a part of their design.

1.7 Experimental Facilities

Experimental facilities shall mean beam ports, including extension tubes with shields, thermal columns with shields, vertical tubes, through tubes, in-core irradiation baskets, irradiation cell, pneumatic transfer systems and in-pool irradiation facilities.

1.8 Experiment Safety Systems

Experiment safety systems are those systems, including their associated input circuits, which are designed to initiate a scram for the primary purpose of protecting an experiment or to provide information which requires manual protective action to be initiated.

1.9 Fuel Bundle

A fuel bundle is a cluster of two, three or four elements and/or non-fueled elements secured in a square array by a top handle and a bottom grid plate adapter. Non-fueled elements shall be fabricated from stainless steel, aluminum or graphite materials.

1.10 Fuel Element

A fuel element is a single TRIGA fuel rod of LEU type.

1.11 Instrumented Element

An instrumented element is a special fuel element in which a sheathed chromal-alumel or equivalent thermocouple is embedded in the fuel near the horizontal center plane of the fuel element at a point approximately 0.3 inch from the center of the fuel body.

1.12 LEU Core

A LEU core is an arrangement of TRIGA-LEU fuel in a reactor grid plate.

1.13 Limiting Safety System Setting

The limiting safety system setting is the setting for automatic protective devices related to those variables having significant safety functions.

1.14 Measuring Channel

A measuring channel is the combination of sensor, interconnecting cables or lines, amplifiers, and output device which are connected for the purpose of measuring the value of a variable.

1.15 Measured Value

The measured value is the value of a parameter as it appears on the output of a channel.

1.16 Movable Experiment

A movable experiment is one for which it is intended that the entire experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

1.17 Operable

Operable means a component or system is capable of performing its intended function.

1.18 Operating

Operating means a component or system is performing its intended function.

1.19 Steady State Operational Core

A steady state operational core shall be an LEU core for which the core parameters of shutdown margin, fuel temperature and power calibration have been determined.

1.20 Pulse Operational Core

A pulse operational core is a steady state operational core for which the maximum allowable pulse reactivity insertion has been determined.

1.21 Pulse Mode

Pulse mode operation shall mean any operation of the reactor with the mode selector switch in the pulse position.

1.22 Reactivity Worth of an Experiment

The reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter the experiment position or configuration.

1.23 Reactor Console Secured

The reactor console is secured whenever all scrammable rods have been fully inserted and verified down and the console key has been removed from the console.

1.24 Reactor Operating

The reactor is operating whenever it is not secured or shutdown.

1.25 Reactor Safety Systems

Reactor safety systems are those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action. Manual protective action is considered part of the reactor safety system.

1.26 Reactor Secured

A reactor is secured when:

a) It contains insufficient fissile material or moderator present in the reactor and adjacent experiments to attain criticality under optimum available conditions of moderation and reflection, or

b) The reactor console is secured, and

1) No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods, and

2) No experiments in or near the reactor are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value of one dollar.

1.27 Reactor Shutdown

The reactor is shut down when the reactor, at ambient temperature and xenon-free condition and including the reactivity worth of all experiments, is subcritical by at least one dollar.

1.28 Reportable Occurrence

A reportable occurrence is any of the following which occurs during reactor operation:

a) Operation with actual safety system settings for required systems less conservative than the limiting safety-system settings specified in the Technical Specifications 2.2.

b) Operation in violation of limiting conditions for operation established in the technical specifications.

c) A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns. (Note: Where components or systems are provided in addition to those required by the technical specifications, the failure of the extra components or systems is not considered reportable provided that the minimum number of components or systems specified or required perform their intended reactor safety function.)

d) An unanticipated or uncontrolled change in reactivity greater than one dollar.

e) Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or containment boundary (excluding minor leaks) where applicable which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both.

f) An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

1.29 Rod-Control

A control rod is a device fabricated from neutron absorbing material or fuel which is used to establish neutron flux changes and to compensate for routine reactivity losses. A control rod may be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged.

1.30 Rod-Regulating

The regulating rod is a low worth control rod used primarily to maintain an intended power level that need not have scram capability and may have a fueled follower. Its position may be varied manually or by the servo-controller.

1.31 Rod-Shim Safety

A shim-safety rod is a control rod having an electric motor drive and scram capabilities. It may have a fueled follower section.

1.32 Rod-Transient

The transient rod is a control rod with scram capabilities that is capable of providing rapid reactivity insertion to produce a pulse.

1.33 Safety Channel

A safety channel is a measuring channel in the reactor safety system.

1.34 Safety Limit

Safety limits are limits on important process variables which are found to be necessary to reasonably protect the integrity of certain physical barriers which guard against the uncontrolled release of radioactivity.

1.35 Scram Time

Scram time is the time measured from the instant a simulated signal reaches the value of the LSSS to the instant that the slowest scrammable control rods reaches its fully inserted position.

1.36 Secured Experiment

A secured experiment is any experiment, experiment facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions.

1.37 Shall, Should and May

The word "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission, neither a requirement nor a recommendation. In order to conform to this standard, the user shall conform to its requirements but not necessarily to its recommendations.

1.38 Shutdown Margin

Shutdown margin shall mean the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition, if the most reactive rod is stuck in its most reactive position, and that the reactor will remain subcritical without further operator action.

1.39 Steady State Mode

Steady state mode operation shall mean operation of the reactor with the mode selector switch in the steady state position.

1.40 True Value

The true value is the actual value of a parameter.

1.41 Unscheduled Shutdown

An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions which could adversely affect safe operation, not to include shutdowns which occur during testing or check out operations.

2.0 Safety Limit and Limiting Safety System Setting

2.1 Safety Limit Fuel Element Temperature

Applicability

This specification applies to the temperature of the reactor fuel.

Objective

The objective is to define the maximum fuel element temperature that can be permitted with confidence that no damage to the fuel element cladding will result.

Specifications

The temperature in a stainless steel-clad TRIGA LEU fuel element shall not exceed 2100 °F (1150°C) under any conditions of operation.

Bases

The important parameter for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification especially since it can be measured. A loss in the integrity of the fuel element cladding could arise from a buildup of excessive pressure between the fuel-moderator and the cladding if the fuel temperature exceeds the safety limit. The magnitude of this pressure is determined by the fuel-moderator temperature and the ratio of the hydrogen to zirconium in the alloy.

The temperature safety limit for the LEU fuel element is based on data which indicates that the internal stresses within the fuel element due to hydrogen pressure from the dissociation of the zirconium hydride will not result in compromise of the stainless steel cladding if the fuel temperature is not allowed to exceed 2100°F (1150°C) and the fuel element cladding is water cooled.

2.2 Limiting Safety System Setting

Applicability

This specification applies to the scram setting which prevents the safety limit from being reached.

Objective

The objective is to prevent the safety limits from being reached.

Specification

The limiting safety system setting shall be 975°F (525°C) as measured in an instrumented fuel element. The instrumented element shall be located adjacent to the central bundle with the exception of the corner positions.

Basis

The limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated preventing the safety limit from being exceeded. The temperature safety limit for LEU fuel is 2100°F (1150°C). Due to various errors in measuring temperature in the core, it is necessary to arrive at a Limiting Safety System Setting (LSSS) for the fuel element safety limit that takes into account these measurement errors. One category of error between the true temperature value and the measured temperature value is due to the accuracy of the fuel element channel and any overshoot in reactor power resulting from a reactor transient during steady state mode of operation. Although a lesser contributor to error, a minimum safety margin of 10% was applied on an absolute temperature basis. Adjusting the fuel temperature safety limit to degrees Kelvin, °K, and applying a 10% safety margin results in a safety limit reduction of 150°C. Applying this first margin of safety, the safety setting would be 1000°C for LEU. However, to arrive at the final LSSS it is necessary to allow for the difference between the measured temperature value and the peak core temperature, which is a function of the location of the thermocouple within the core. For example, if the thermocouple element were located in the hottest position in the core, the difference between the true and measured temperatures would be only a few degrees since the thermocouple junction is at the mid-plane of the element and close to the anticipated hot spot. However, at the TAMU this core position is not available due to the location of the transient rod. For the TAMU the location of the instrumented elements is therefore restricted to the positions closest to the central element. Calculations indicate that, for this case, the true temperature at the hottest location in the core will differ from the measured temperature by no more than 40%. When applying this 40% worst case measurement scenario and considering the previously mentioned sources of error between the true and measured values, a final LSSS temperature of 975°F (525°C) is imposed on operation. Viewed on an absolute temperature scale, °K, this represents a 37% safety margin in the LEU safety limit.

In the pulse mode of operation, the above temperature limiting safety system setting will apply. However, the temperature channel will have no effect on limiting peak powers generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). In this mode, however, a temperature trip will act to reduce the amount of energy generated in the entire pulse transient by cutting the "tail" off the energy transient in the event the pulse rod remains stuck in the fully withdrawn position.

3.0 Limiting Conditions for Operation

3.1 Reactor Core Parameters

3.1.1 Steady State Operation

Applicability

This specification applies to the energy generated in the reactor during steady state operation.

Objective

The objective is to assure that the fuel temperature safety limit will not be exceeded during steady state operation.

Specifications

The reactor power level shall not exceed 1.3 megawatts under any condition of operation. The normal steady state operating power level of the reactor shall be 1.0 megawatts. However, for purposes of testing and calibration, the reactor may be operated at higher power levels not to exceed 1.3 megawatts during the testing period.

Basis

Thermal and hydraulic calculations indicate the TRIGA fuel may be safely operated up to power levels of at least 2.0 MW with natural convection cooling.

3.1.2 Pulse Mode Operation

Applicability

This specification applies to the peak temperature generated in the fuel as the result of a pulse insertion of reactivity.

Objective

The objective is to assure that respective pulsing will not induce damage tot he reactor fuel.

Specification

a) The reactivity to be inserted for pulse operation shall not exceed that amount which will produce a peak fuel temperature of 1526°F (830°C). In the pulse mode the pulse rod shall be limited by mechanical means or the rod extension physically shortened so that the reactivity insertion will not inadvertently exceed the maximum value.

b) Until the LEU fuel core has been calibrated, maximum pulse shall be limited to $2.00.

Basis

TRIGA fuel is fabricated with a nominal hydrogen to zirconium ratio of 1.6 for LEU fuel.

3.1.4 Core Configuration Limitation

Applicability

This specification applies to a full LEU core.

Objective

The objective is to assure that the fuel temperature safety limit will not be exceeded due to power peaking effects in full LEU cores.

Specifications

a) The TRIGA core assembly shall be LEU.

b) The reactor shall not be taken critical with a core lattice position vacant except for positions on the periphery of the core assembly. Water holes in the inner fuel region shall be limited to single rod positions. Vacant core positions shall contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core.

c) The instrumented element shall be located adjacent to the central bundle with the exception of the corner positions (Reference: 2.2 Limiting Safety System Setting).

Bases

a) Safety and accident analysis were only performed for a TRIGA core using LEU fuel.

b) Vacant core positions containing experiments or an experimental facility will prevent accidental fuel additions to the reactor core. They will be permitted only on the periphery of the core or a single rod position to prevent power peaking in regions of high power density.

c) Reference: 2.2 Limiting Safety System Setting.

3.1.5 Maximum Excess Reactivity

Applicability

This specification applies to the maximum excess reactivity, above cold critical, which may be loaded into the reactor core at any time.

Objective

The objective is to ensure that the core analyzed in the safety analysis report approximates the operational core within reasonable limits.

5.0 Design Features

5.1 Reactor Fuel

Applicability

This specification applies to the fuel elements used in the reactor core.

Objective

The objective is to assure that the fuel elements are of such a design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications

TRIGA-LEU Fuel

The individual unirradiated LEU fuel elements shall have the following characteristics:

1) Uranium content: maximum of 30 wt% enriched to maximum of 19.95% with nominal enrichment of 19.75% Uranium-235.

2) Hydrogen-to-zirconium ratio (in the ZrH_x): nominal 1.6 H atoms to 1.0 Zr atoms with a maximum H to Zr ratio of 1.65.

3) Natural erbium content (homogeneously distributed): nominal 0.90 wt%. (See bases below for contract specifications.)

4) Cladding: 304 stainless steel, nominal 0.020 inch thick.

Bases

The fuel specification permits a maximum uranium enrichment of 19.95%. This is about 1% greater than the design value for 19.75% enrichment. Such an increase in loading would result in an increase in power density of less than 1%. An increase in local power density of 1% reduces the safety margin by less than 2%.

The fuel specification for a single fuel element permits a minimum erbium content of about 5.6% less than the design value of 0.90 wt%. (However, the quantity of erbium in the full core must not deviate from the design value by more than –3.3%). This variation for a single fuel element would result in an increase in fuel element power density of about 1-2%. Such a small increase in local power density would reduce the safety margin by less than two percent.

The maximum hydrogen-to-zirconium ratio of 1.65 could result in a maximum stress under accident conditions in the fuel element clad about a factor of two greater than for a hydrogen-to-zirconium ratio of 1.60. This increase in the clad stress during an accident would not exceed the rupture strength of the clad.

5.2 Reactor Core

Applicability

This specification applies to the configuration of fuel and in core experiments.

Objective

The objective is to assure that provisions are made to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities will not be produced.

Specifications

a) The core shall be an arrangement of TRIGA uranium-zirconium hydride fuel-moderator bundles positioned in the reactor grid plate.

b) The reflector, excluding experiments and experimental facilities, shall be water or a combination of graphite and water or D_2O.

Bases

a) Standard TRIGA cores have been in use for years and their characteristics are well documented. LEU cores including 30/20 fuel have also been operated at General Atomics and their successful operational characteristics are available. General Atomics and Texas A&M have done a series of studies documenting the viability of using LEU fuel in TRIGA reactors.

b) The core will be assembled in the reactor grid plate which is located in a pool of light water. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements.

5.3 Control Rods

Applicability

This specification applies to the control rods used in the reactor core.

Objective

The objective is to assure that the control rods are of such a design as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications

a) The shim-safety control rods shall have scram capability and contain borated graphite, B_4C powder or boron and its compounds in solid form as a poison in aluminum or stainless steel cladding. These rods may incorporate fueled followers which have the same characteristics as the fuel region in which they are used.

b) The regulating control rod need not have scram capability and shall be a stainless rod or contain the materials as specified for shim-safety control rods. This rod may incorporate a fueled follower.

c) The transient control rod shall have scram capability and contain borated graphite or boron and its compounds in solid form as a poison in an aluminum or stainless steel clad. The transient rod shall have an adjustable upper limit to allow a variation of reactivity insertions. This rod may incorporate an aluminum or air follower.

Bases

The poison requirements for the control rods are satisfied by using neutron absorbing borated graphite, B_4C powder or boron and its compounds. Since the regulating rod normally is a low worth rod, its function could be satisfied by using a solid stainless steel rod. These materials must be contained in a suitable clad material, such as aluminum or stainless steel, to insure mechanical stability during movement and to isolate the poison from the pool water environment. Control rods that are fuel followed provide additional reactivity to the core and increase the worth of the control rod. The use of fueled followers in the LEU region has the additional advantage of reducing flux peaking in the water filled regions vacated by the withdrawal of the control rods. Scram capabilities are provided for rapid insertion of the control rods which is the primary safety feature of the reactor. The transient control rod is designed for a reactor pulse. The nuclear behavior of the air or aluminum follower which may be incorporated into the transient rod is similar to a void. A voided follower may be required in certain core loadings to reduce flux peaking values.

5.4 Radiation Monitoring System

Applicability

This specification describes the functions and essential components of the area radiation monitoring equipment and the system for continuously monitoring airborne radioactivity.

Objective

The objective is to describe the radiation monitoring equipment that is available to the operator to assure safe operation of the reactor.

Specification

The radiation monitoring equipment listed in the following table will have these characteristics.

Radiation Monitoring Channel and Function

Area Radiation Monitor (gamma sensitive instruments)
Function: Monitor radiation fields in key locations, alarm and readout at control console and readout in reception room.

Specifications

a) The reactor shall be housed in a facility designed to restrict leakage. The minimum free volume in the facility shall be 180,000 cubic feet.

b) The reactor building shall be equipped with a ventilation system designed to filter and exhaust air or other gases from the reactor building and release them from a stack at a minimum of 85 feet from ground level.

c) Emergency shutdown controls for the ventilation system shall be located in the reception room and the system shall be designed to shut down in the event of a substantial release of fission products.

Bases

The facility is designed such that the ventilation system will normally maintain a negative pressure with respect to the atmosphere so that there will be no uncontrolled leakage to the environment. The free air volume within the reactor building is confined when there is an emergency shutdown of the ventilation system. Controls for startup, emergency filtering, and normal operation of the ventilation system are located in the reception room. Proper handling of airborne radioactive materials (in emergency situations) can be conducted from the reception room with a minimum of exposure to operating personnel.

5.7 Reactor Pool Water Systems

Applicability

This specification applies to the pool containing the reactor and to the cooling of the core by the pool water.

Objective

The objective is to assure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding.

Specifications

a) The reactor core shall be cooled by natural convective water flow.

b) The pool water inlet and outlet pipe to the demineralizer shall not extend more than 15 feet below the top of the reactor pool when fuel is in the core.

c) Diffuser and skimmer pumps shall be located no more than 15 feet below the top of the reactor pool.

d) Pool water inlet and outlet pipes to the heat exchanger shall have emergency covers within the reactor pool for manual shut off in case of pool water loss due to external pipe system failure.

e) A pool level alarm shall indicate loss of coolant if the pool level drops approximately 10% below operating level.

Bases

a) This specification is based on thermal and hydraulic calculations which show that the TAMU TRIGA-LEU core can operate continuously in a safe manner at power levels up to 2,420 kW with natural convection flow and sufficient bulk pool cooling.

b) In the event of accidental siphoning of pool water through inlet and outlet pipes of the demineralizer system, the pool water level will drop no more than 15 feet from the top of the pool.

c) In the event of pipe failure and siphoning of pool water through the skimmer and diffuser water systems, the pool water level will drop no more than 15 feet from the top of the pool.

d) Inlet and outlet coolant lines to the pool heat exchanger terminate at the bottom of the pool. In the event of pipe failure, these lines must be manually sealed from within the reactor pool. Covers for these lines will be stored in the reactor pool. The time required to uncover the reactor core due to failure of a single pool coolant pipe system is 17 minutes.

e) Loss of coolant alarm after 10% loss requires corrective action. This alarm is observed in the reactor control room and the reception room.

5.8 Physical Security

The licensee shall maintain in effect and fully implement all provisions of the NRC staff approved physical security plan, including amendments and changes made pursuant to the authority of 10 CFR 50.54 (p). The approved security plan consists of documents withheld from public disclosure pursuant to 10 CFR 2.70, collectively titled "Texas A&M University System, Nuclear Science Center Reactor Security Plan."