PURDUE UNIVERSITY
RESEARCH REACTOR
LICENSE NO. R-87
DOCKET NO. 50-182

RESPONSE TO
REQUEST FOR ADDITIONAL INFORMATION
FOR
THE CONVERSION OF THE PURDUE
UNIVERSITY RESEARCH
REACTOR FROM HEU TO LEU FUEL

REDACTED VERSION

SECURITY-RELATED INFORMATION REMOVED

REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS

PURDUE
UNIVERSITY

SCHOOL OF NUCLEAR ENGINEERING

3 May 2007

Mr. Alexander Adams, Senior Project Manager
Research and Test Reactors Branch A
US Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD, 20852-2738

SUBJECT: Docket No. 050-0182, Facility License R-87
Response to RAIs for Conversion of PUR-1, ML 070680273

Dear Mr. Adams:

Enclosed please find the response to the Request for Additional Information (ML 070680273). Included in this package are one unbound and four bound copies of the response, and one unbound and four bound copies of the specifications for the new reactor fuel. Digital copies of these documents are also included on an enclosed CD.

Should you have any questions, or require further information, please contact me at 765.496.3573, or jere@purdue.edu.

I declare of penalty of perjury that the foregoing and the original submission of the conversion proposal are true. Executed on this day of 3 May 2007.

Respectfully submitted,

Jere Jenkins
Director of Radiation Laboratories
Purdue University Nuclear Engineering

Enclosures: As stated

A020

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RESPONSES TO
REQUEST FOR ADDITIONAL INFORMATION
FOR
THE CONVERSION OF THE PURDUE UNIVERSITY RESEARCH REACTOR FROM HEU TO LEU FUEL

DOCKET NUMBER 50-182

FACILITY LICENSE NO. R-87

3 May 2007

Submitted By:
Jere Jenkins, Director of Radiation Laboratories
Ed Merritt, Asst. Director of Radiation Laboratories/Reactor Supervisor

Purdue University Research Reactor
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REQUEST FOR ADDITIONAL INFORMATION
PURDUE UNIVERSITY RESEARCH REACTOR
DOCKET NO. 50-182

3 May 2007

Question 1

1. The regulations in 10 CFR 50.30(b) require applications, such as your application for conversion, to be made under oath or affirmation. Please provide a statement that your application of August 13, 2006, is made under oath or affirmation.

Response:

A statement of oath and affirmation will be submitted with these answers that will cover both this document and the original conversion proposal (CP).

Question 2

2. Table 4-1 and Figure 4-35. The figure shows the plate locations with a single dummy plate and 13 fueled plates. What is the distribution within the fuel assembly when two dummy plates are used? The figure shows nine standard fuel assemblies with 13 plates and four with 12 plates. This brings the total of plates to 189, not 190 as stated in Table 4-1 and on page 22. Please explain.

Response:

This is a typographical error. Fig 4-35 for position E2 should show 13 fuel plates instead of 12. Since the fuel assembly design has been modified (ref. Q5), the new model will be discussed in the answer to that question.

Question 3

3. Table 4-3. What is the basis for picking the width of the fuel meat for the LEU fuel plates (59.6 mm with a range between 58.9 mm and 62.7 mm)?

Response:

It is not half way in between 58.9 mm and 62.7 mm, but this was the value used for the University of Florida case and was carried forward. This should have no effect on these analyses, since the critical factor is the U-235 mass. If the average fuel meat width of 60.8 mm were used, the power density would be slightly lower and all of the safety margins slightly increased from those shown in the conversion proposal.

Question 4

4. Section 4.2. Proposed Technical Specification (TS) 5.2.2 says that the LEU assembly will have up to 185 g of U-235. Please explain why that is different than 12.5 g/plate times a maximum of 14 plates/assembly as specified in Section 4.2.

Response:

The specifications for the U-235 loading of the LEU fuel plates is 12.5±0.35 g. With 14 plates per assembly, the nominal loading is 175 g with an allowed range of 170.1 to 179.9 g. The proposed TS 5.2.2 should be 180 g of U-235.

Question 5

5.

A. Section 4.2.1. Please provide a copy of Reference 5.

B. Our understanding is that the final design of the Purdue fuel elements was still underway when the conversion Safety Analysis Report (SAR) was submitted to NRC. Please verify that the description of the fuel in the SAR is accurate or discuss any fuel design changes made since the SAR was written.

C. Our understanding is that the fuel plates and fuel boxes are being fabricated by different vendors and that final assembly of the fuel elements will be performed at Purdue. Please describe the quality assurance requirements the Department of Energy employs at the fuel component vendors to ensure that the fuel element components are consistent with the SAR.

D. Please verify that approved procedures will be used at Purdue to assemble the fuel elements.

E. Please describe the quality assurance requirements to be employed at Purdue to ensure that the assembled fuel elements are consistent with the SAR.

F. Please discuss how the fresh fuel plates will be stored and handled during fuel element assembly such that the requirements of TS 5.3 are met.

Response:

A. A copy of the fuel specification will be provided with the response to these questions.

B. The old LEU fuel element and fuel assembly designs are described in Sections 4.2.1.2 and 4.2.1.3, respectively, of the August 2006 Conversion Proposal (CP). This LEU design used fuel plates bolted at the top and bottom to form a fuel element. The fuel element in this design was inserted into a smooth-walled aluminum box to form a fuel assembly.

The changes that occurred after the submission of the conversion proposal were to the assembly cans, and the method of inserting the plates into the cans. Specifically, the plates in the new LEU fuel assembly design slide into wall spacers affixed to two sides of the fuel can, as shown in Figures Q5B-1 to Q5B-7. The nominal plate-to-plate spacing was slightly reduced, and the nominal plate-to-wall spacing was slightly increased, as shown in Table Q5B-1. An updated version of Table 4-1 from the August 2006 Conversion Proposal that reflects the changes to the assembly and core design is also included below. It should be noted that the design of the fuel plates and dummy plates was not changed as a result of the design change.

Table Q5B-1: Channel Types and Thickness in PUR-1 Assemblies (Ref: Table 4-23 in CP)

	Plate-to-plate (mils)		**Plate-to-wall (mils)**	
	Standard	**Control**	**Standard**	**Control**
HEU	207	207	160[1]	160[1]
OLD LEU	147	197	79	79
NEW LEU	144±15	181±15	127±8	127±8

[1] This is the smaller of the two bolt heads on the HEU elements.

A new Table 4-1 from the CP is shown below, reflecting the changes in the fuel spacing.

Table 4-1: Summary of Key Nominal Design Parameters of HEU (current) and LEU (Expected) Cores (Ref: Table 4-1 in CP).

DESIGN DATA	**HEU**	**OLD LEU Design**	**NEW LEU Design**
Fuel Type	MTR Plate	MTR Plate	MTR Plate
Fuel "Meat" Composition	U-Al Alloy	U_3Si_2-Al	U_3Si_2-Al
Fuel Enrichment U-235 (nominal)	93%	19.75%	19.75%
Mass of U-235 per plate (g, nominal)		12.5	12.5
Fuel Meat Dimensions			
Width (mm)	62.7	59.6	59.6
Thickness (mm)	0.508	0.508	0.508
Height (mm)	600.1	600.1	600.1
Fuel Plate Dimensions			
Width (mm)	70.2	70.2	70.2
Thickness (mm)	1.52	1.27	1.27
Height (mm)	638.6	638.6	638.6
Cladding Composition	1100 Al	6061 Al	6061 Al
Cladding Thickness (mm)	0.508	0.381	0.381
Dummy Plate Composition	1100 Al	6061 Al	6061 Al
Dummy Plate Dimensions	Same as Fuel	Same as Fuel	Same as Fuel
Standard Fuel Assemblies			
Number of standard assemblies	13	13	13
Number of plates per standard assembly	10	14	14
Control Fuel Assemblies			
Number of control assemblies	3	3	3
Number of plates per control assembly	6	8	8
Total plates in core (fuel and dummy)	148	206	206
Fuel plates in core (current, expected)	124	190	191
Dummy plates in core (current, expected)	24	16	15
Plate spacing in standard assemblies (mm)	5.26	3.71	3.66
Plate spacing in control assemblies (mm)	5.26	5.00	4.60

The calculated reactivity change as a result of the plate spacing change was -0.05±0.07 %Δk/k. The fuel orientation in the standard assemblies was also modified by a 90° rotation (making the plates parallel to the control assembly plates) in order to change the fuel direction relative to the handle such that that the plates would remain caged in the fuel box, as shown in Fig. Q5B-8. The HEU plates, and the original LEU plates, were bolted together. The new handle orientation is now normal to the plates to restrict their possible movement. The calculated reactivity change as a result of the fuel orientation change was -0.19±0.03 %Δk/k. These two changes together resulted in the replacement of one dummy plate with a fuel plate in the model. Overall we believe that this design is superior to the HEU and previous LEU designs, since the additional spacers will provide a smaller uncertainty in the spacing of the plates over their entire length and result in less possible variation in the channel width.

Figure Q5B-1: New Standard LEU fuel assembly. (Replaces Figure 4-5 in CP)

Figure Q5B-2: New Control LEU fuel assembly. (Ref Fig. 4-7 in CP)

Figure Q5B-3: New Standard LEU assembly can detail, wall spacers.

Figure Q5B-4: New Control LEU assembly can detail, wall spacers.

Figure Q5B-5: New standard LEU assembly can spacer detail.

Figure Q5B-6: New control LEU assembly can spacer detail.

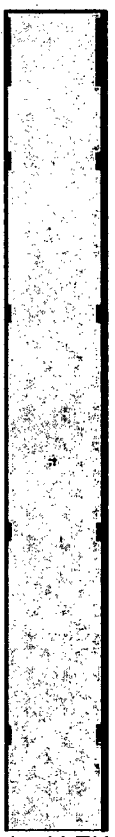

Figure Q5B-7: Model representation of standard LEU assembly plate spacing detail, showing wall spacers.

Figure Q5B-8: LEU core model. (Replaces Figure 4-20 in CP)

C. The fuel and fuel assembly cans are being purchased by the Idaho National Laboratory, and must meet INL qualification and certification. A summary of the INL procurement qualifications is provided below. Close communication between Purdue University and the Idaho National Laboratory has been maintained through the design process. This communication ensures that the appropriate technical and functional requirements from the reactor safety basis are carried forward to the fabrication drawings and specifications for the new reactor components. The drawings and fabrication specifications denote the dimensions and other design parameters that must be met for the item to be in compliance with the reactors safety basis. These documents are used to convey the requirements to the INL procurement personnel, the vendor, and QA Engineers. The Project then relies on the well established QA processes at the INL to ensure that the final product meets the requirements per the drawings and specifications.

The INL meets or exceeds the requirements for procuring items and services as established by the Department of Energy (DOE). These requirements are contained in 10 CFR 830 Subpart A, *Quality Assurance Requirements*; DOE Order 414.1C, *Quality Assurance*; and NQA-1-2000, *Quality Assurance Requirements for Nuclear Facility Applications.* These requirements establish the methods that the INL must use to procure items and services. More specifically, the process requirements relevant to the procurement of items for the Purdue reactor are:

- Items and services shall be procured to meet established requirements and perform as specified. [DOE Order 414.1C, Attachment 2, 3.g. (1)] [10 CFR 830.122 (g) (1)]
- Prospective suppliers shall be evaluated and selected on the basis of specified criteria. [DOE Order 414.1C, Attachment 2, 3.g. (2)] [10 CFR 830.122 (g) (2)]
- Processes shall be established and implemented to ensure that approved suppliers continue to provide acceptable items and services. [DOE Order 414.1C, Attachment 2, 3.g. (3)] [10 CFR 830.122 (g) (3)]

To complete the process, final inspection of the items will be performed by qualified INL Inspectors at the vendor before shipment. Inspections of the items will also be performed by Purdue upon receipt at the reactor.

D. Procedures for assembly of the fuel elements will be written and certified by the Facility Director, the Reactor Supervisor, and the safeguards and oversight committee (CORO) for PUR-1. All fuel handling will be done under the supervision of USNRC licensed Senior Reactor Operators. Accurate records of fuel disposition will be maintained.

Figure Q5D-1: Fuel plate	Figure Q5D-2: Dummy plate,

Loading of the fuel assemblies prior to the initial assembly of the core and the initial approach to critical will be according to the preliminary loading plan. LEU plate serial numbers will be recorded and verified by two SROs independently, and matched to the uniquely identified fuel assembly cans. All placements of assemblies will again be independently verified by two SROs,

and accurate records maintained. During all of these operations, compliance with Technical Specifications for the PUR-1 reactor will be verified.

E. There are 13 standard elements and 3 control elements in the LEU core. A standard element can contain up to 14 fuel plates, and the control elements can contain up to 8 fuel plates. The configuration analyzed in the August 2006 Conversion Proposal had 10 standard elements with 13 fuel plates, 3 standard elements with 12 plates, and 3 control elements with 8 fuel plates – or a total of 190 fuel plates. The new LEU core design is modeled with 11 standard fuel elements with 13 fuel plates, 2 standard elements with 12 plates, and 3 control elements with 8 fuel plates – or a total of 191 fuel plates, This is the configuration that is planned at startup, and adjustments to the number of plates per assembly will be made at startup to build the initial critical core and the working core.

Fuel and dummy plate locations will be tracked through each step of the initial loading and approach to critical following approved procedures as discussed in the answer to Q5D. During the insertion of fuel and dummy plates into the assembly cans, the fuel plates will be identified by serial number on forms associated with each fuel element. The location of individual plates will be specifically tracked, and verified by two SROs on the form. As the approach to critical process continues and fuel plates and dummies are relocated, new locations of plates and dummies will be tracked on forms for each assembly.

The location of fuel and dummy plates in the initial critical core will likely change as plates are added to obtain a final core load that satisfies the TS requirements and the operation parameters of the conversion proposal. At each step of this process, fuel assembly forms will be modified to reflect the new plate locations, and will again be independently verified by two SROs. These records will be maintained for the duration of that core load, or the lifetime of PUR-1.

F. Upon receipt of the LEU fuel, it will be stored in the 6M drums in a secure location. One bundle of fuel plates will be removed at one time from the 6M shipping containers and examined for quality assurance with appropriate procedures. Upon completion of the inspection of the bundle of plates, they will be moved to a secure storage area within the facility.

Racks specifically constructed for the purpose of storage of the plates will contain each bundle. A criticality analysis of the storage facility was performed with MCNP. The model is shown in Figure Q5F-1.

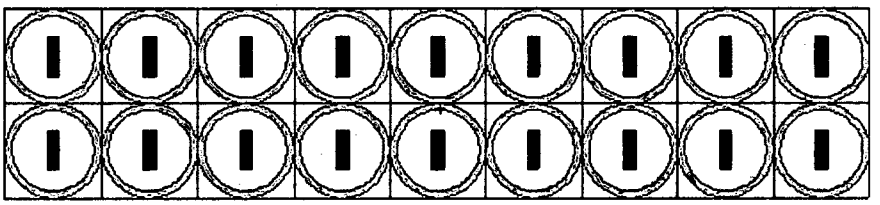

Figure Q5F-1: MCNP model of dry storage facility.

With all of the plates moved to that new location, the calculated eigenvalue of the storage rack is 0.015±0.0003. Should the storage flood with water, the eigenvalue becomes 0.26±0.004. In both of these conditions, the TS limit of k_{eff}<0.8 is met. Parametric analyses were performed with spacing of the plates within the tubes, and the above listed cases are the worst case. All movement of fuel will be performed under the direction of licensed senior reactor operators.

Question 6

6. Section 4.2.5. What is the effect on reactivity and power distribution of replacing graphite in the reflector with an aluminum sample holder?

Response:

The aluminum sample tubes in the irradiation facility are filled with graphite unless they are being used. Replacing the graphite in the six sample tubes with air, as would be expected if samples were inserted, reduces the calculated eigenvalue of the LEU core from 1.00731±0.0002 to 0.996±0.0002 without any reactivity bias, but with the estimated bias of 0.32%, these values would be 1.0041 and 0.993, respectively.

The effect of replacing graphite in the irradiation facility with an aluminum sample holder on the reactor power distribution used in the thermal-hydraulic analyses is well covered by the global hot channel factor of 1.5 for the reactor power measurement uncertainty used in all the reported analyses. The reactor power hot channel factor was used in addition to (1) the radial power factor (used to account for plate-to-plate power variation), (2) the factor used to account for power density variation along the width of the hot plate, and (3) the axial power profile. While replacing the graphite in the irradiation facility with an aluminum sample holder will cause a change in the reactor power distribution, this change will be small compared with the hot channel factors already applied in the thermal-hydraulic analysis. Therefore, no additional thermal-hydraulic analysis is needed.

Question 7

7.

A. Table 4-6. The measured to calculated eigenvalues for the excess reactivity of a fresh HEU core are compared. This difference establishes the claim that the model introduces a 0.32% bias to the calculation of the eigenvalue. What are the uncertainties for the measured and calculated eigenvalues?

B. What is the justification to use this 0.32% bias for the LEU core? Note that if the bias for the LEU core is not used (or is significantly less than 0.32%) the excess reactivity exceeds the TS limit of 0.6%.

Response:

A.

There is no estimate available for the uncertainty of the measured value.

Table 4-6: Calculated and Measured excess reactivity for fresh HEU core.
(Replaces Table 4-6 in the CP)

Measured Eigenvalue	Calculated HEU Eigenvalue	Bias (% Δk/k)
1.00431	1.00753±0.00015	0.32±0.015*

* The uncertainty on the bias is based on the uncertainty calculated by MCNP.

B.

This is the best available estimate for what the bias will be in the LEU core model. Until the core is assembled, the actual bias cannot be known. A careful approach to criticality will be used to ensure that

all Technical Specifications are met. Upon achieving initial criticality, adjustments to the final plate loading will be performed while maintaining sufficient margin to the TS limit of 0.6% excess.

Question 8

8. Table 4-7. The title of Table 4-7 does not reflect the contents of the table. The table is a tabulation of calculated eigenvalues at five measured critical rod positions. It does not contain any calculated critical rod positions. Please correct.

Response:

A corrected table 4-7 is shown below.

Table 4-7: Calculated eigenvalues at measured critical rod positions for HEU core (Replaces Table 4-7 in CP)

Case	RR Position (cm)	SS-2 Position (cm)	SS-1 Position (cm)	Eigenvalue	Bias %Δk/k
1	64.12	43.60	64.12	1.00305±0.00015	0.30±0.015
2	64.12	64.12	49.68	1.00285±0.00015	0.29±0.015
3	51.93	51.92	53.19	1.00327±0.00013	0.33±0.013
4	1.89	54.40	64.12	1.00340±0.00014	0.34±0.014
5	31.44	48.47	64.12	1.00327±0.00014	0.33±0.014

Question 9

9. Table 4-8. In order to understand the differences between calculation and measurement quoted in the table, provide the uncertainties for both the measured and calculated control rod worths, if available. This will also clarify whether the last column in the table is really an "error" or more appropriately a "relative difference." Provide uncertainties for all values, and correct header of last column to relative difference.

Response:

A corrected Table 4-8 is shown below.

Table 4-8: Comparison of measured and calculated control rod worths for HEU core (Replaces Table 4-8 in CP)

	Measured Value (Δk/k)*	Calculated Value (Δk/k)	Relative Difference
Shim Safety 1	0.0450	0.0436±0.0002	-3.11%
Shim Safety 2	0.0252	0.0235±0.0002	-6.74%
Regulating Rod	0.0028	0.0027±0.0002	-3.57%

* No uncertainties are available for measured quantities.

Question 10

10. Section 4.5, page 17. It is stated that in the LEU model there was an addition of 20 ppm of boron-equivalent to the 6061 cladding material. This section also states that the 6061 aluminum assembly cans had a 10 ppm boron content. Why were different boron contents assumed for the same aluminum alloy? Does this difference impact the calculated results, i.e., what is the reactivity worth of 10 ppm boron?

Response:

The natural boron equivalent content of the aluminum used to manufacture the cladding for MTR fuel assemblies has increased since the 1960s due to increased use of boron carbide crucibles in the aluminum industry. The PUR-1 HEU fuel was manufactured in 1960. The 1100-Al cladding was assumed to have an impurity content of 10 ppm natural boron equivalent content because no specific data was available. Analysis of a 2005 representative sample of 6061-Al supplied to BWXT by Kaiser Aluminum had a natural boron equivalent content (excluding boron itself) of 12 ppm. Normally, analyses of boron content provide a result of < 10 ppm. Values < 10 ppm are difficult to measure. Consequently, the natural boron equivalent content of the 6061-Al used in the conversion analysis for the cladding of the PUR-1 LEU fuel and for the new 6061-Al cans was estimated to be 20 ppm.

A full core calculation was run assuming 10 ppm instead of 20 ppm natural boron equivalent in the LEU core. The increase in reactivity was 0.40% ±0.03 $\Delta k/k$. This possible reactivity change will be compensated by adjusting the number of LEU fuel plates in the startup core.

Question 11

11. Section 4.5. What is the nominal thermal neutron flux for the PUR-1 core either on average or at experimental locations?

Response:

Maximum thermal neutron flux in the fuel region of the HEU core is 2.1E10 n/(cm^2*s), and the average thermal flux in the fuel region is 1.2E10 n/(cm^2*s) at 1kW according to the 1988 conversion proposal. The MCNP calculated values for the LEU core are 2.01E10 n/(cm^2*s) peak in the fuel region, and 1.38E10 n/(cm^2*s) average in the fuel region.

Question 12

12. What is the expected reactivity lifetime of the shim-safety and regulating rods?

Response:

The critical position of the control rods has not changed considerably over the lifetime of the HEU core, as shown in Fig. Q12-1. The rod positions shown in the figure reflect a normalized value for the upper limit measured for SS-1.

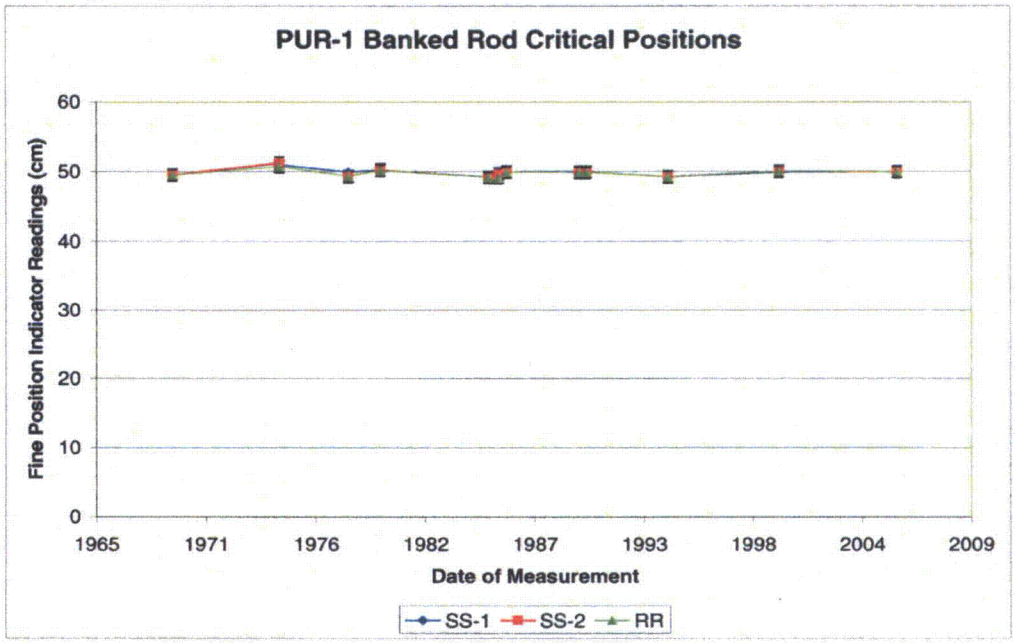

Figure Q12-1: Normalized critical rod positions for banked rods.

Question 13

13. Figures 4-21, 4-22, and 4-23. These figures show calculated and measured calibration curves for SS-1, SS-2, and RR, respectively, with an HEU core, but only for rod position where there are measurements. Figures 4-24, 4-25 and 4-26 show the calculated curves for an LEU core for the full range of control rod travel. Please provide calculated curves for the HEU core with a full range of travel so they can be compared with the full-range curves for the LEU core.

Response:

Updated Figures 4-21, 4-22, and 4-23 are provided below.

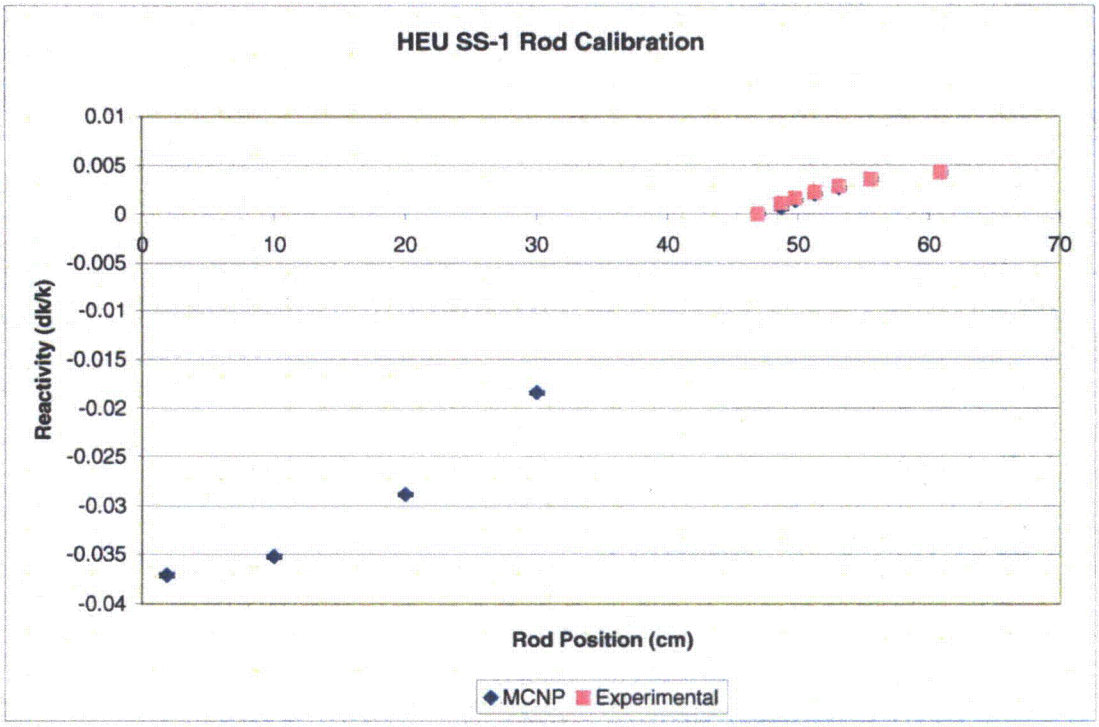

Figure 4-21: HEU control rod calibration for SS-1. (Replaces Fig. 4-21 in CP)

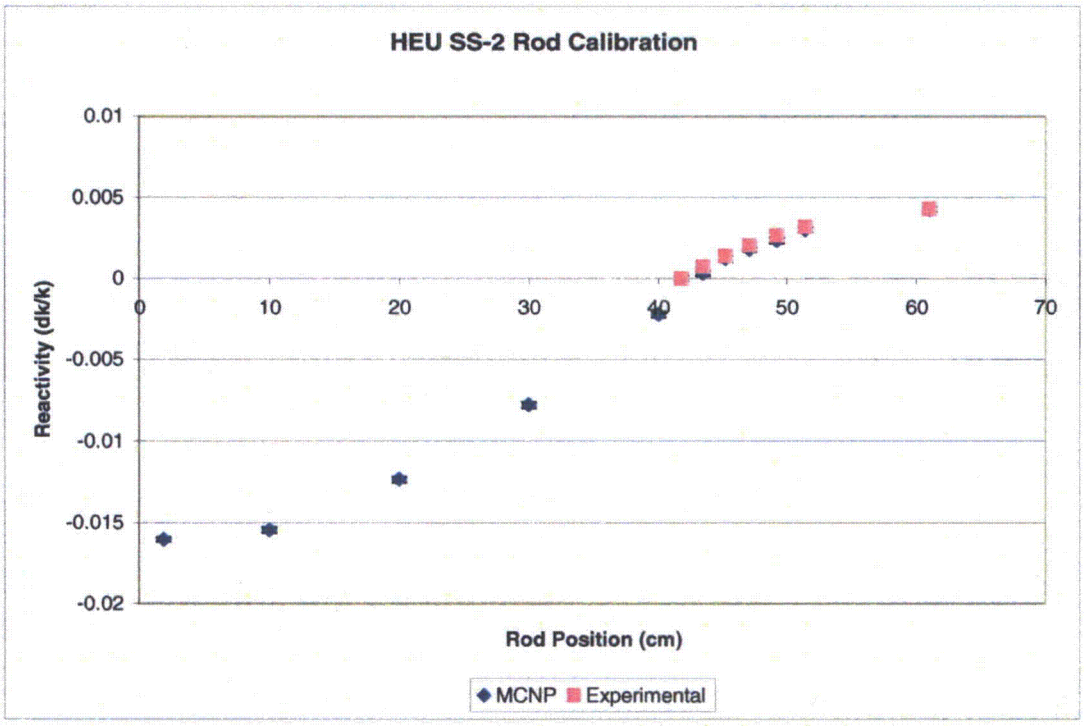

Figure 4-22: HEU control rod calibration for SS-2. (Replaces Fig. 4-22 in CP)

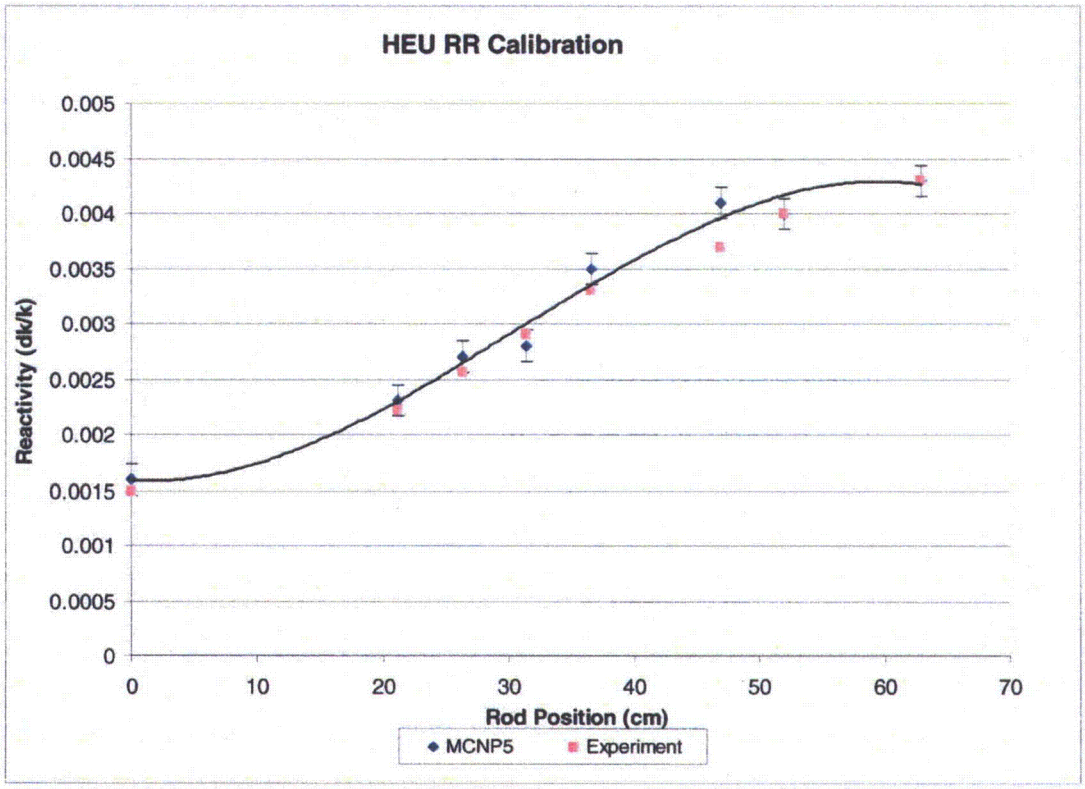

Figure 4-23: HEU control rod calibration for RR. (Replaces Fig. 4-23 in CP)

Question 14

14. Table 4-10. The text in Section 4.5.1 indicates that Table 4-10 contains calculated and measured control rod worths while the headings in the table only refer to calculated values. Please clarify.

Response:

Remove "and measured" from text. The correct text should read: "A comparison of the HEU and LEU calculated control rod worth values are shown in Table 4-10."

The uncertainties of the values in Table 4-10 were added for convenience. Also, Table 4-10 has been updated to include calculated control rod worths for the new LEU core design described in response to Question 5.B.

Table 4-10: Comparison of HEU and LEU calculated control rod worths. (Replaces Table 4-10 in CP)

	HEU Calculated (Δk/k)	Old LEU Design Calculated (Δk/k)	New LEU Design Calculated (Δk/k)
Shim Safety 1	0.0436±0.0002	0.0391±0.0002	0.0377±0.0003
Shim Safety 2	0.0235±0.0002	0.0200±0.0002	0.0189±0.0003
Regulating Rod	0.0027±0.0002	0.0029±0.0002	0.0023±0.0003

For consistency, a comparison of the Rod Worth Curves for the three control rods are shown in Figures Q14-1, Q14-2 and Q14-3. These figures are related to Figures 4-24, 4-25 and 4-26 in the August 2006 Conversion Proposal, respectively. The comparison shows that the rod worth curves have been only slightly changed by the new LEU core design.

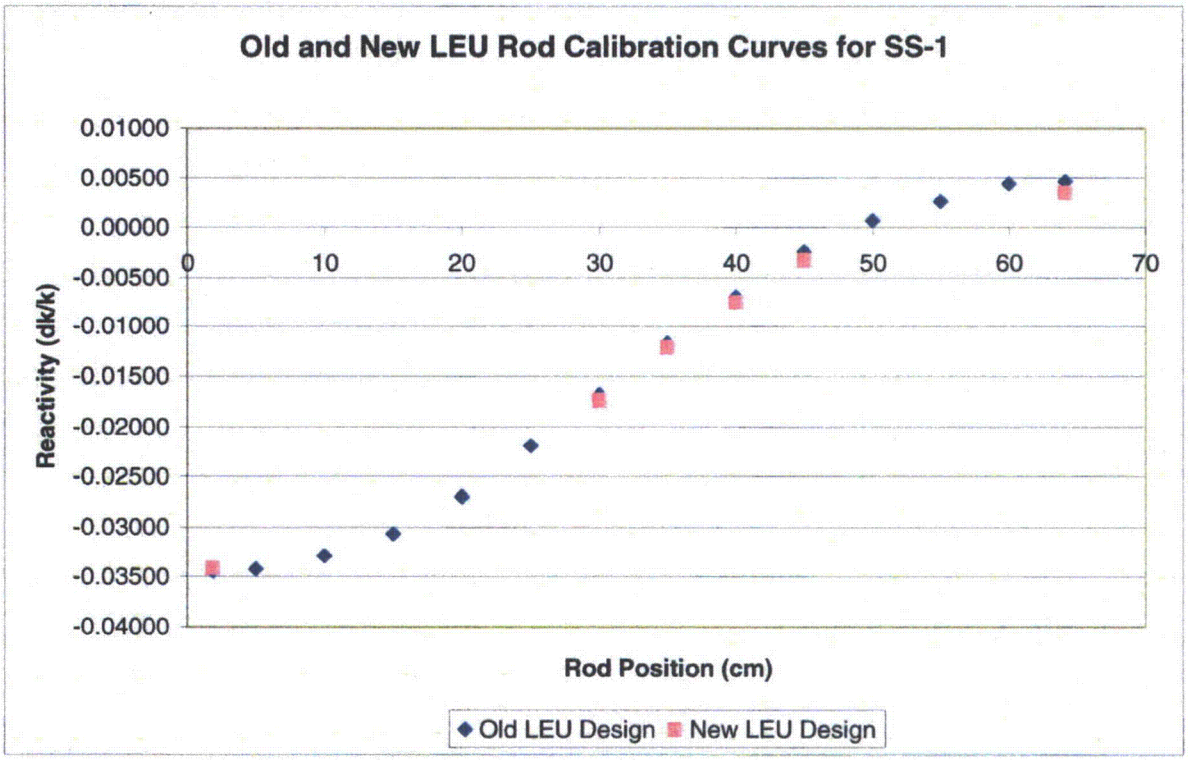

Fig Q14-1: Comparison of Old and New LEU Shim Safety 1 Rod Worth Curves (Ref: Fig 4-24 in CP)

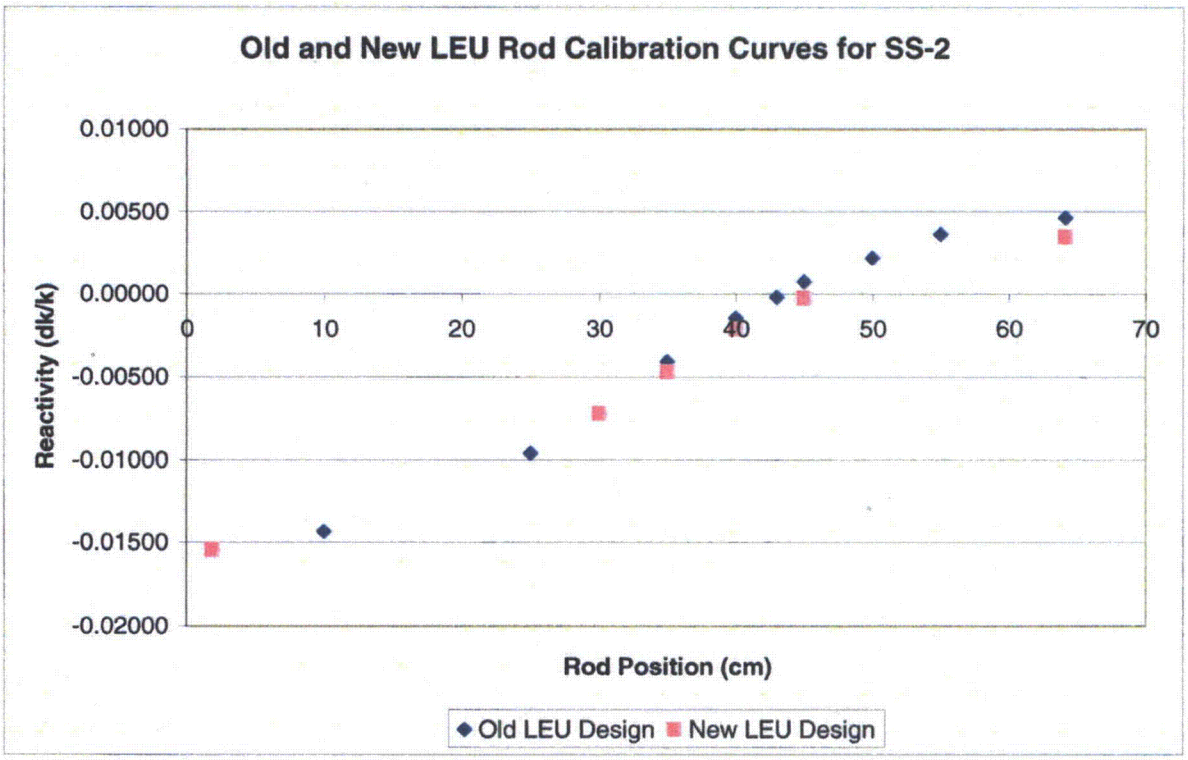

Fig Q14-2: Comparison of Old and New LEU Shim Safety 2 Rod Worth Curves (Ref: Fig 4-25 in CP)

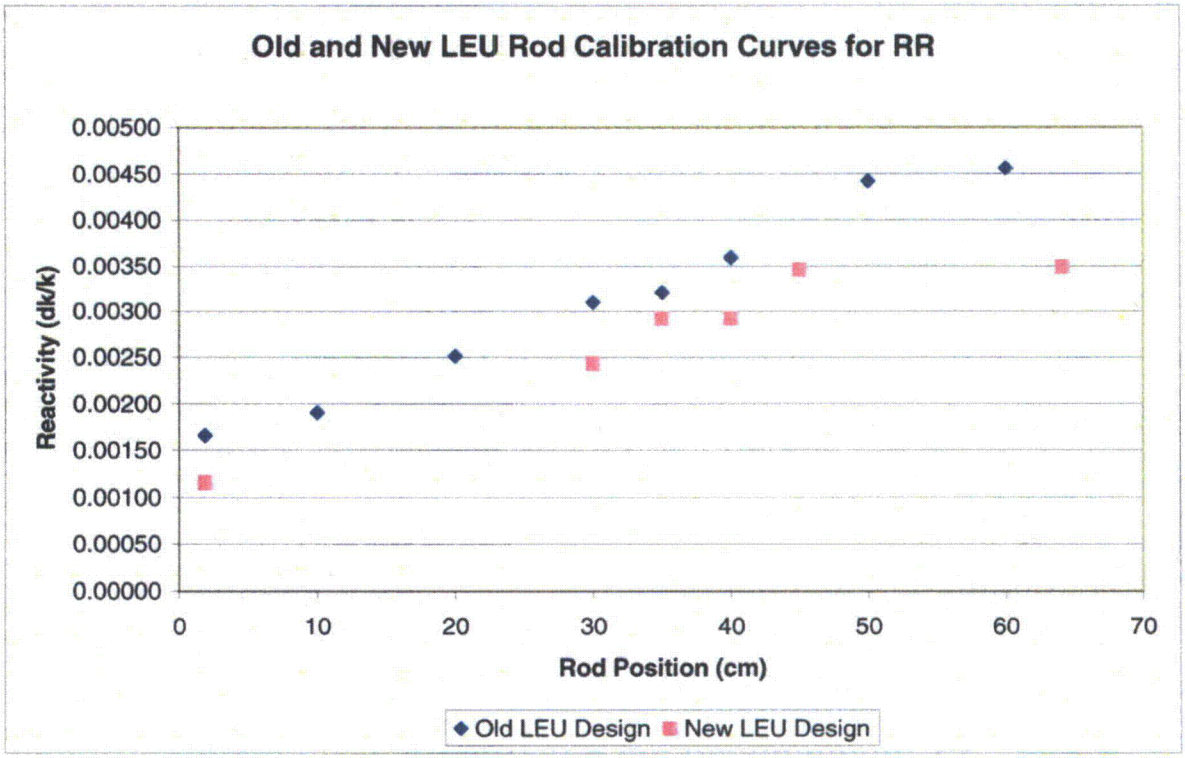

Fig Q14-3: Comparison of Old and New LEU Regulating Rod Worth Curves (Ref: Fig 4-26 in CP)

Question 15

15. Table 4-11. Comparing Tables 4-2 and 4-11 there is a discrepancy in the maximum insertion rate for the shim-safety rods and the regulating rod. The insertion rates for the LEU core are exactly the same while for the HEU core there is a factor of 100 in the SS-1 and SS-2 values and a factor of 1000 for the RR. What are the correct values?

Response:

Typographical and calculation errors were present in both tables. Figure Q15-1 shows calculated regulating rod worth curves for the calculated and measured HEU, and the new LEU cores. The maximum speed of the regulating rod is 0.75 cm/s. Corrected tables are shown below.

As a result of changes to the design of the LEU assembly cans subsequent to the submission of the Conversion Proposal (see response to Question 5.B), control rod reactivity insertion rates and other reactor parameters have been recalculated. The updated values are included in Tables 4-11 and 4-2.

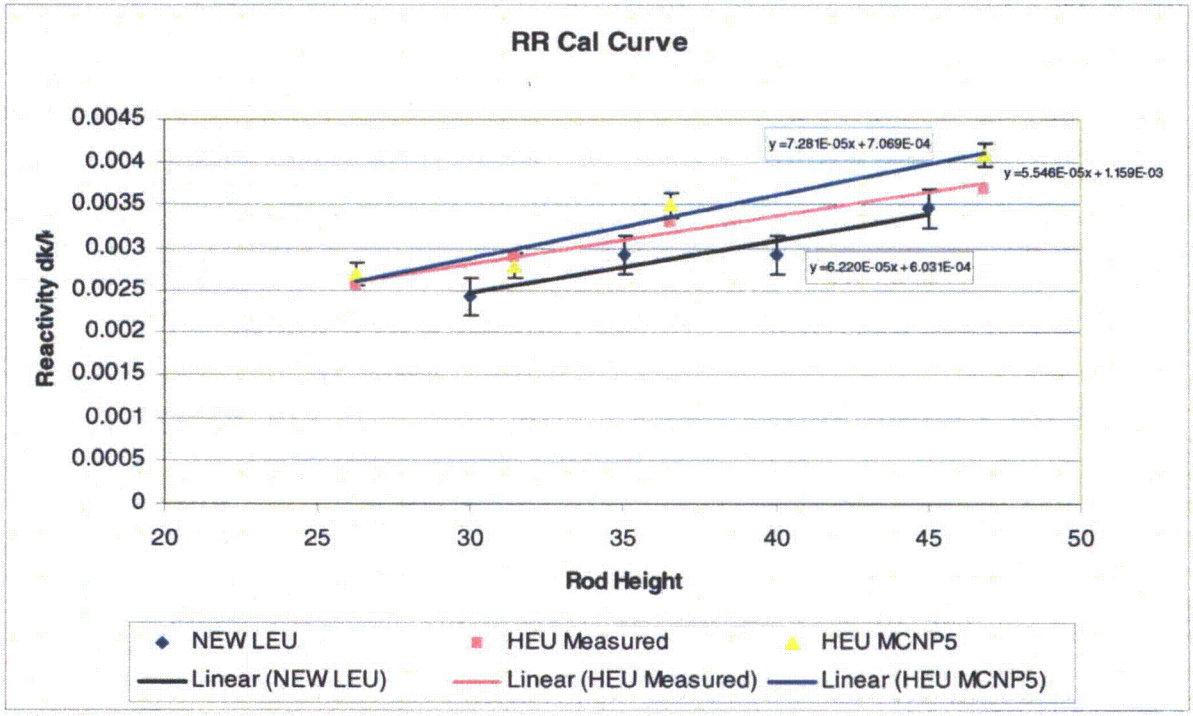

Figure Q15-1: Calculated and Measured HEU and Calculated LEU Regulating Rod Curves

Table 4-11: Comparison of maximum reactivity insertion rates for HEU and LEU cores
(Replaces Table 4-11 in CP)

	Maximum Reactivity Insertion Rates for Control Rods $\left(\frac{\Delta k}{k \cdot s}\right)$			
	HEU Measured	**HEU Calculated**	**Old LEU Design Calculated**	**New LEU Design Calculated**
Shim-safety 1	1.12E-04	9.31E-05	1.84E-04	1.75E-04
Shim-safety 2	7.45E-05	7.45E-05	1.07E-04	8.75E-05
Regulating Rod	4.16E-05	5.46E-05	4.07E-05	4.66E-05

Table 4-2: Summary of Key Reactor Physics and Safety Parameters for the HEU (current) and LEU (Expected) Cores (Replaces Table 4-2 in CP)				
REACTOR PARAMETERS	HEU Measured	HEU Calculated	Old LEU Design Calculated	New LEU Design Calculated
Fresh core excess reactivity (%Δk/k)	0.43	0.43[1]	0.47[2]	0.35[3]
Shutdown margin (%Δk/k)	-2.07	-1.93[1]	-1.53[2]	-1.58[3]
Control rod worth (%Δk/k)				
Shim-safety 1	4.50	4.36	3.91	3.78
Shim-safety 2	2.53	2.35	2.00	1.90
Regulating Rod	0.28	0.27	0.29	0.23
Maximum reactivity insertion rate $\left(\frac{\%\Delta k}{k \cdot s}\right)$				
Shim-safety 1	1.12E-02	9.31E-03	1.84E-02	1.75E-02
Shim-safety 2	7.45E-03	7.45E-03	1.07E-02	8.75E-03
Regulating Rod	4.16E-03	5.46E-03	4.07E-03	4.66E-03
Avg. coolant void coefficient $\left(\frac{\%\Delta k}{k \cdot \%void}\right)$[4]		-1.54E-1±8%	-1.90E-1±8%	-1.93E-1±7%
Coolant temperature coefficient $\left(\frac{\%\Delta k}{k \cdot {}^{\circ}C}\right)$[5]		-1.05E-2±7%	-9.74E-3±8%	-9.05E-3±9%
Fuel temperature coefficient $\left(\frac{\%\Delta k}{k \cdot {}^{\circ}C}\right)$[6]		0	-7.01E-4±12%	-8.05E-4±10%
Effective delayed neutron fraction (%)		0.795	0.787	0.784
Neutron lifetime (μs)		76.7	81.3	81.3[7]

[1] Includes a bias of 0.32%Δk/k for the HEU core, see Table 4-7: Calculated eigenvalues at measured critical rod positions for HEU core. The calculated excess reactivity is 0.75% Δk/k. The calculated shutdown margin is -1.61% Δk/k.

[2] Assumes the same bias as the HEU core. The calculated excess reactivity is 0.79% Δk/k. And the calculated shutdown margin is -1.21% Δk/k.

[3] The calculated excess reactivity is 0.67%Δk/k. And the calculated shutdown margin -1.26%Δk/k.

[4] Calculated for the representative range of 0-0.6% void.

[5] Calculated for the representative range of 20-30°C.

[6] Calculated for the representative range of 20-127°C.

[7] This value was not recalculated for the new LEU design. It is assumed to be the same as the old LEU design value.

Question 16

16. Table 4-11. According to Figure 4-23 the slope of the calculated and measured reactivity curve as a function of rod position for the regulating rod are approximately the same. Why is there a factor of ten difference between the measured and calculated maximum reactivity insertion rate for the regulating rod in the HEU core in Table 4-11?

Response:

The difference noted in Table 4-11 was a typographical error. The values should be as shown in the updated Table 4-11 provided in the response to Question 15.

Question 17

17. Table 4-12. Some of the values in Table 4-2 do not agree with values in Table 4-12. Agreement: k_{excess}; SS-2 worth calculations; shutdown margin calculations Disagreement: Measured SS-2 worth; measured shutdown margin Please verify all values in these two tables.

Response:

Table 4-12 should be as shown here:

Table 4-12: Comparison of shutdown margins for HEU and LEU cores
(Replaces Table 4-12 in CP)

	HEU Measured	HEU Calculated	Old LEU Design Calculated	New LEU Design Calculated
SS-2 Worth (Δk/k)	-2.53%	-2.35%	-2.00%	-1.89%
k_{excess} (Δk/k)	0.431%	0.433%	0.468%	0.351%
Shutdown Margin (Δk/k)	-2.07%	-1.93%	-1.53%	-1.58%

Question 18

18. Table 4-12. In Table 4-12 and Table 4-2 the calculated shutdown margin for the LEU core, taking into account the -0.32% systematic bias, is -1.53%. A footnote for Table 4-2 gives the calculated shutdown margin (without bias) as -1.31%. Should the value of the calculated unbiased shutdown margin be -1.21% (-1.53%+0.32% = 0.79-2.00 = -1.21%)?

Response:

Second footnote in 4.2 should be "Assumes the same bias as the HEU core. The calculated excess reactivity is 0.79% Δk/k. And the calculated shutdown margin is -1.21% Δk/k." The value of 1.31 is a typo.

Question 19

19. Section 4.5.3. In calculating βeff from keff with and without prompt neutrons, was any bias applied to the eigenvalues?

Response:

The delayed neutron fraction was calculated using the formulation:

$$\beta_{eff} = 1 - \frac{k_{eff}^{prompt}}{k_{eff}} = \frac{k_{eff} - k_{eff}^{prompt}}{k_{eff}}.$$

The eigenvalue calculations were performed using MCNP5. In the conversion submittal, the bias of the PUR-1 MCNP model was not taken into account when calculating β_{eff}. Thus, the unbiased β_{eff} values for the HEU and LEU cores are calculated as follows:

HEU Core:
$k_{eff} = 1.00369 \pm 0.00012$
$k_{eff}^{prompt} = 0.99571 \pm 0.00017$
$\beta_{eff} = 1 - 1.00369/0.99571 = 7.951\text{x}10^{-3} \pm 2.07\text{x}10^{-4}$ (2.6%)

LEU Core:
$k_{eff} = 1.00379 \pm 0.00010$
$k_{eff}^{prompt} = 0.99589 \pm 0.00015$
$\beta_{eff} = 1 - 1.00379/0.99589 = 7.870\text{x}10^{-3} \pm 1.79\text{x}10^{-4}$ (2.3%)

The bias of the PUR-1 MCNP model is $\Delta\rho_{bias}$= 0.32% $\Delta k/k$. This was determined by comparing the results of eigenvalue calculations for several cases with the control rods at measured critical positions. Accounting for the bias in the core model, the β_{eff} is calculated as:

$$\beta_{eff}^{biased} = \frac{k_{eff} - \Delta\rho_{bias} - (k_{eff}^{prompt} - \Delta\rho_{bias})}{k_{eff} - \Delta\rho_{bias}} = \frac{k_{eff} - k_{eff}^{prompt}}{k_{eff} - \Delta\rho_{bias}}.$$

HEU Core:
β_{eff}^{biased} = (1.00369 – 0.99571)/(1.00369 – 0.0032) = $7.976\text{x}10^{-3} \pm 2.07\text{x}10^{-4}$ (2.6%)

LEU Core:
β_{eff}^{biased} = (1.00379 – 0.99589)/(1.00379 – 0.0032) = $7.895\text{x}10^{-3} \pm 1.79\text{x}10^{-4}$ (2.3%)

The reactivity insertion accident analyses were performed using the *unbiased* β_{eff} values, which are slightly smaller (0.3%) than the effective delayed neutron fraction determined by accounting for the bias in the MCNP model. Consequently, the reactivity insertion accident analyses were performed with a value of β_{eff} that gives more conservative results.

Question 20

20. Please clarify the derivation of the temperature and void coefficients of reactivity by providing the following information:

A. Is the water density perturbation used in the water temperature coefficient and for the void coefficient?
B. What is the meaning of the void coefficient expressed in the unit of $\Delta\rho$/°C? Why is the void coefficient expressed in $\Delta\rho$/°C in Tables 4-14 and 4-15?
C. How was the temperature coefficient calculated and what components were included?
D. What scattering kernel data were used for these calculations?
E. What are the uncertainties of these calculations?
F. For both the HEU and LEU cores, the temperature coefficient is negative if the temperature of the water between the fuel plates and between the fuel cans is increased but is slightly positive if there is a simultaneous increase in the temperature of the water outside the core, that is if the entire inventory of water in the system is included. This might imply that if heating of the water occurs external to the core, there would be a positive change in reactivity. Are there any scenarios where the temperature of the water can be increased external to the core without increasing the water temperature within the core? Is there any physical scenario

where temperature of the water exterior to the core can be simultaneously increased along with the temperature of the water in the interior of the core?

Response:

A.
No. The effect of heating the reactor coolant on water temperature and water density were treated as separable effects for the purpose of calculating reactivity feedback coefficients. This question is addressed in more detail in the response to Question 20.C below.

B.
The different units given for the void coefficient in Tables 4-14 and 4-15 in the Conversion Proposal are simply different expressions of the same quantity. The unit $\Delta\rho$/% void is the typical unit for expressing the void coefficient, and the unit required by the PARET code which was used for the accident analyses. The methodology for calculating the void coefficient is described in the text of the Conversion Proposal and in the response to Question 20.C below.

The void coefficient was also converted to the unit of $\Delta\rho/^{o}C$. The reason for expressing the void coefficient in the alternative units was to facilitate a comparison of the water void and temperature reactivity feedback effects, which were treated as separable. This comparison is illustrated in Figures 4-28 and 4-31. When the reactor coolant within the fuel element (i.e., between the fuel plates, which is the region of interest for accident analyses in the PUR-1) is heated, the reactivity feedback from the water temperature increase is slightly larger.

The unit conversion to $\Delta\rho/^{o}C$ was accomplished by equating the void (water density) perturbation to a corresponding water temperature perturbation at 1.5 atmospheres. This is the water pressure in the PUR-1 core, which is at the bottom of a 15 foot tank of water. The density of water at 1.5 atmospheres as a function of temperature is shown in Figure 4-29.

C.
Temperature coefficients of reactivity were calculated assuming separability of the reactivity feedback effects due to water temperature, water density (void), and fuel temperature. This is a common practice and makes it easy to understand the inherent shutdown mechanisms that are responsible for affecting reactivity-induced transients. Not only are the feedback coefficient treated as separable, the reactivity coefficients were also calculated under non-isothermal conditions because different regions of the reactor will heat up at different rates during a power increase.

Three distinct regions containing water in the PUR-1 were considered.

1. The water between the fuel plates; this is called the "fuel assembly" water.
2. The water between the fuel element cans and also the water in the control elements between the control rod guard plates; this is called the "inter-assembly" water.
3. The water in the reactor tank; this is called the "reflector" water.

Water temperature coefficients were calculated by adjusting the temperature of the water in each of these regions and calculating the impact on the core reactivity. Reactivity feedback coefficients due to the perturbation of the fuel assembly water temperature, the fuel assembly plus inter-assembly water temperature, and fuel assembly, inter-assembly, and reflector water temperature were calculated. It is important to note that only the feedback effect due to heating of the fuel assembly water (between the fuel plates) was considered in the accident analyses, because only this water would experience an immediate heating due to power increases during the transient. The other regions containing water would take much longer to heat up.

Heating the water increases the thermal motion of the hydrogen atoms in the water. The result is to increase the energy of neutrons which are in "thermal equilibrium" with the hydrogen moderator, thus hardening the neutron spectrum in the reactor. At higher neutron energies, the U-235 fission cross

section is reduced. Thus, there is a negative reactivity feedback effect due to heating of the water between the fuel plates.

Increasing the temperature of the reflector (tank) water was found to have a positive reactivity feedback effect over a small temperature range from nominal conditions. This is due to a decrease in the neutron absorption in the reflector as the temperature is increased and the neutron spectrum hardens. For the HEU core, the feedback coefficient due to heating all water in the reactor tank from 20 to 30 °C was calculated to be $2.38 \times 10^{-3} \pm 2.11 \times 10^{-3}$ % $\Delta k/k/°C$. However, it would take a long time for any transient to heat the reflector water, so it is judged that there are no significant safety issues related to the positive reactivity feedback coefficient when all the water is heated. It should be noted that for larger increases in the temperature of the reflector water, the reactivity feedback effect is negative.

D.

MCNP allows for two adjustments on neutron scattering reactions based on the temperature of the medium. For neutron energies above 4 eV, the code adjusts the elastic scattering cross sections of nuclei in the medium using a free gas thermal treatment. The code user can specify the temperature of each cell within the model and the cross sections are adjusted if the specified temperature differs from the temperature of the processed nuclear data in the cross section library.

For modeling elastic and inelastic scattering events for neutrons below 4 eV, an $S(\alpha,\beta)$ treatment is employed. These data are available in the MCNP libraries for certain materials; light water and graphite are of interest for the PUR-1 analyses. $S(\alpha,\beta)$ data for graphite at 20 °C were employed. Furthermore, $S(\alpha,\beta)$ which had previously been evaluated for light water at 20, 30, 60, 100, and 150 °C were also used.

E.

Eigenvalue calculations were performed with MCNP, typically using from 25 to 50 million neutron histories. These calculations yielded a reactor k_{eff} with a 1-σ uncertainty of ±17 pcm (0. 017% $\Delta k/k$) for 25 million histories; the uncertainty was reduced to ±12 pcm (0. 012% $\Delta k/k$) when 50 million histories were employed. The 1-σ uncertainty of the eigenvalue calculations reduces by the square root of the number of histories. The uncertainty of the reactivity feedback coefficients can be calculated as the square root of the sum of the squares of the eigenvalue calculations. Based on the eigenvalue calculations with 25 to 50 million histories, the uncertainties of the water temperature and water void coefficients were found to be on the order of 15% to 30% when calculated over the expected 10 to 20 °C perturbations of water temperature.

It was decided that the calculational uncertainty should be reduced by extending the number of histories in the MCNP eigenvalue calculations. Eigenvalue calculations for the nominal core state and certain other cases which exhibit only small perturbations to the core k_{eff} were performed with 200 to 300 million neutron histories. This reduced the 1-σ uncertainty of the eigenvalue calculations to around 5 pcm (0.005% $\Delta k/k$). The new results for the water temperature, water void, and fuel temperature coefficients, along with the corresponding uncertainties, are presented in Table Q20E-1.

Furthermore, a subset of the LEU-core kinetics parameters and reactivity coefficients were recalculated for the new LEU core design (see response to Question 5.B). This was done in order to quantify the impact of the changes to the assembly can design and the orientation and number of fuel plates in the core on the reactivity coefficients. The recalculated values are also included in Table Q20E-1. The impact of these changes on the accident analyses are discussed in the response to Question 39.

An updated Table 4-13 from the August 2006 Conversion Proposal is also provided to reflect the newly calculated values.

Table Q20E-1: Water Temperature and Void Coefficients for PUR-1 HEU and LEU Cores.
(Ref: Tables 4-14 and 4-15 in CP)

PUR-1 HEU Core Water Temperature Coefficient (α_{water})				
20 to 30 °C	-1.054E-04	Δρ/°C	±	7%
30 to 60 °C	-1.156E-04	Δρ/°C	±	5%
60 to 100 °C	-1.384E-04	Δρ/°C	±	4%
PUR-1 Old LEU Core Design Water Temperature Coefficient (α_{water})				
20 to 30 °C	-9.743E-05	Δρ/°C	±	8%
30 to 60 °C	-1.155E-04	Δρ/°C	±	5%
60 to 100 °C	-1.229E-04	Δρ/°C	±	5%
PUR-1 New LEU Core Design Water Temperature Coefficient (α_{water})				
20 to 30 °C	-9.051E-05	Δρ/°C	±	9%
PUR-1 HEU Core Water Void Coefficient (α_{void})				
0 to 0.60% void	-1.535E-03	Δρ/% void	±	8%
0.60% to 1.50% void	-1.460E-03	Δρ/% void	±	6%
1.50% to 3.99% void	-1.683E-03	Δρ/% void	±	4%
PUR-1 Old LEU Core Design Water Void Coefficient (α_{void})				
0 to 0.60% void	-1.898E-03	Δρ/% void	±	8%
0.60% to 1.50% void	-1.933E-03	Δρ/% void	±	7%
1.50% to 3.99% void	-1.980E-03	Δρ/% void	±	4%
PUR-1 New LEU Core Design Water Void Coefficient (α_{void})				
0 to 0.60% void	-1.933E-03	Δρ/% void	±	7%
PUR-1 Old LEU Core Design Fuel Temperature Coefficient (α_{fuel})				
20 to 127 °C	-7.014E-06	Δρ/°C	±	12%
PUR-1 New LEU Core Design Fuel Temperature Coefficient (α_{fuel})				
20 to 127 °C	-8.053E-06	Δρ/°C	±	10%
PUR-1 Old LEU Core Design Effective Delayed Neutron Fraction (β_{eff})				
	0.00787		±	0.00018
PUR-1 New LEU Core Design Effective Delayed Neutron Fraction (β_{eff})				
	0.00784		±	0.00008

Table 4-13: Comparison of other core physics parameters for HEU and LEU cores.
(Replaces Table 4-13 in CP)

	HEU (calculated)	**Old LEU Design (calculated)**	**New LEU Design (calculated)**
α_{fuel} (%Δk/k °C)	0	-7.01E-04	-8.05E-04
$\alpha_{moderator}$ (%Δk/k °C)	-1.05E-02	-9.74E-03	-9.05E-03
α_{void} (%Δk/k %-void)	-1.54E-01	-1.90E-01	-1.93E-01
β_{eff}	0.795%	0.787%	0.784%
ℓ(μs)	76.7	81.3	81.3[8]

F.

Unless additional fixtures were to be placed in and around the core to physically separate the core and the pool, there are no scenarios where the water external to the core could be heated without heating of the water within the core. The temperature of the entire pool and the water in the core could be reduced with operation of the chiller system. In addition, heaters and stirrers could be added to the pool to heat the water uniformly.

[8] This value was not recalculated for the new LEU design. It is assumed to be the same as the old LEU design value.

Question 21

21. Section 4.5.3. What is the expected lifetime of the proposed LEU core and how does that translate into burnup? Explain if burnup over core life will have any significant impact on the reactor neutronic parameters.

Response:

The reported burnup of the PUR-1 reactor as of the 2005 annual report was 0.00099% of the original HEU fuel load, which was first put into service in 1962 (43 years of service). The expected lifetime of the LEU fuel is at least as long as the HEU fuel. At a licensed power of 1 kW, there is no expectation that the LEU fuel burnup will exceed what the HEU core experienced, therefore, no impact on the reactor neutronic parameters is expected. As is the present practice, the control rods will still be examined for material degradation annually, as part of the regular inspection cycle.

Question 22

22. Section 4.5.3. Once the LEU core is loaded, is there any anticipated reloading or rearrangement of the fuel? If yes, how would this change the nuclear design parameters?

Response:

It is possible that fuel may be rearranged within the 16 fuel assemblies (i.e. dummies and fuel plates relocated) during the lifetime of the PUR-1 LEU core to allow for experimental or instructional needs. Approved standard operating procedures 8, 9A and 9B were used previously for this evolution, with an standard approach to critical performed each time. Verification of control rod worths and core excess reactivity will ensure that TS requirements are maintained.

Changes to the fuel configuration are addressed in Technical Specifications 4.1 and 6.1.10.

Question 23

23. Table 4-13. The fuel temperature reactivity coefficient shown in Table 4-13 for the LEU fuel is not the most conservative value as compared to the values shown in Table 4-15. Explain the basis for selecting the representative value for the fuel temperature coefficient.

Response:

Values of the fuel temperature coefficient were calculated for fuel temperatures ranging in three intervals: from 20 to 127 °C, from 127 to 227 °C, and from 227 to 327 °C. It is unlikely that the fuel temperature in the PUR-1 will increase much above 27 °C for the hypothetical accidents considered in the Conversion Proposal. Furthermore, the analysis of the rapid insertion of 0.6% $\Delta k/k$ (see response to Question 39) revealed that the LEU fuel temperature increased to no higher than 120 °C. Consequently, it is appropriate to use the fuel temperature coefficient calculated over the lower range of temperature increase.

Question 24

24. Section 4.5.4.1. In Section 4.5.4.1 it says that the peak power in plate 89 of bundle 3-3 is about 13% lower than that of plate 262. Please explain how the 13% value was calculated.

Response:

In Table 4-17 it is observed that the power in plate 262 is 7.3% higher than the power in plate 89. The power given in this table is for the entire plate. Taking into account the axial power profile in these plates, however, it is found that the difference in the peak power densities between plates 262 and 89 is 13%.

Question 25

25. Section 4.5.4.2. The hottest fuel plate for the proposed LEU core was calculated considering two different critical control rod configurations while the hottest fuel plate for the HEU core was calculated considering five different critical control rod configurations. The critical control rod configuration used to determine the hottest fuel plate in the LEU and HEU cores differ. Why were different critical control rod configurations used for the two cores? Which configuration is closest to the control rod configuration used during reactor operation? Does the calculation for the HEU core represent the limiting case? If not, what is the limiting case for the hottest fuel plate?

Response:

The five measured configurations that were available were calculated for the HEU core:

1. SS-2 controlling with SS-1 and RR fully withdrawn,
2. SS-1 controlling with SS-2 and RR fully withdrawn,
3. all three rods at the same height (banked)
4. RR fully inserted, with SS-2 controlling and SS-1 fully withdrawn
5. RR at 30 (half inserted), with SS-2 controlling and SS-1 fully withdrawn.

The case with all three rods banked (Case 3, above), was found to be the limiting case for the HEU core power distribution, although the peak power densities for all five cases were found to be very similar (see Figure 4-34 of the Conversion Proposal). Calculations using any of these five power densities would result in peak cladding temperatures that are different by less than 1°C, and probably by tenths of 1°C. For the LEU core design evaluated in the Conversion Proposal, only the first two configurations (with SS-2 and SS-1 controlling, respectively) were calculated. As shown in Figure 4-36 of the August 2006 document, the peak powers in these configurations are virtually identical. We expect a peak power that is slightly higher if the rods are banked, but that this slightly higher peak power will lead to a maximum cladding temperature that is higher by a few tenths of 1°C, and have no effect whatsoever on the safety of the facility.

However, for the sake of consistency with the HEU core calculations, the power distribution for the new LEU core design (see response to Question 5.B) was evaluated for a "banked rod" critical configuration. The results of the new power distribution analysis are discussed in the response to Question 27.

For the HEU core, all of the configurations listed are used for the operation of the reactor, and it is anticipated that all of these configurations will be used for the LEU core when it is complete.

The banked rod calculation for the HEU core represents the limiting case.

Question 26

26. Figure 4-37. Figure 4-37 shows the radial power profile in LEU plates 1215 and 1348. According to Figure 4-35, the "bottom" edge of plate 1348 is closer to the core centerline and hence is expected to have a higher power density than the "upper" edge. However the text of the conversion SAR says that the radial segments (as shown in Figure 4-37) for plate 1348 are numbered from top-to-bottom. Figure 4-27

has radial segment 1 showing a higher power density than radial segment 11. Should the radial segments be numbered from bottom-to-top instead?

Response:

There is a typographical error in the conversion proposal. The radial segments for plate 1348 in the LEU core and shown in Fig. 4-37 and Table 4-21 are numbered from bottom to top.

Question 27

27. Section 4.5.4.2. In Section 4.5.4.2 it states that radial power profiles shown in Tables 4-21 and 4-22 for plates 1348 and 1215 are used for the thermal-hydraulic analysis of the plates. Where is the radial power profile incorporated in the thermal-hydraulic analysis? If the radial power profile is not explicitly included in the analysis, explain the impact on the thermal-hydraulic analysis.

Response:

Old LEU Core Design

The NATCON code input RPEAK on the input card 4 (described in Appendix 1) was used in the conversion proposal to account for *only the average* (over the width of the specific fuel plate) of the radial power profiles shown in Tables 4-21 and 4-22 of the August 2006 Conversion Proposal. RPEAK (referred to as the radial power factor of a specific plate) is the ratio of power generated by a specific plate to the average power per plate for the core. The impact of the power density variation along the width of the plate itself was not incorporated in the analysis presented in the conversion proposal. It has been incorporated now as described below, and the impact is that the most limiting ONB power for the old LEU core design becomes 88.6 kW (down from 96.1 kW in the conversion proposal). In addition to the improvement discussed in this question, the 88.6 kW value also includes the effect of two additional improvements suggested questions 30 and 33, and addressed in the answers to these questions.

For the most limiting LEU plate 1348 (located plate-to-plate with 20 mil uncertainty on 197 mil thick channel), the radial power factor RPEAK already incorporated in the analysis presented in the conversion proposal is 1.6404. Over the 59.563 mm width of the fuel meat, the power density varies as shown in Table 4-21 of the August 2006 Conversion Proposal, and the maximum-to-average ratio of this power density variation is 1.0857. To account for this variation, the hot channel factor (FFILM) for temperature drop across the film is increased by a factor of 1.0857, i.e., set equal to 1.248x1.0857=1.355. The radial power factor RPEAK of the plate is kept unchanged at 1.6404, and the hot channel factor (FBULK) for bulk coolant temperature rise is also kept unchanged at 1.321. The ONB power calculated using this FFILM (= 1.355), along with changes in hot channel factors FW and FBULK (in response to Question numbers 30 and 33), is 88.6 kW. This result is shown in Table Q27-1 (column H) for comparison with the results of changes in FW and FBULK. The effect of the change in FFILM alone is a reduction in ONB power from 95.8 kW to 88.6 kW (column G versus column H in Table Q27-1). Compared to this, the proposed operating power of PUR-1 is only 1 kW.

New LEU Core Design

The following calculations were done to determine the core power distribution and the ONB power for the new design of the PUR-1 LEU fuel assemblies, as described in the response to Question 5.B. The power distribution analyses discussed below shows that the LEU plate 1348 is the most limiting. The ONB power calculation given below includes the impact of the power density variation along the width of the plate itself (shown in Figure Q27-3 and Table Q27-4), and also the effect of two additional improvements suggested questions 30 and 33.

For the most limiting LEU plate, 1348 (located plate-to-plate in control assembly 4-4 as shown in Fig. Q27-1), the ONB power is found to be 94.2 kW, using a 20 mil uncertainty on the 181 mil thick channel. The 20 mil uncertainty used is conservative compared to the 15 mil uncertainty given in Table Q5B-1 for

the new LEU design. This ONB power is shown in Table Q27-1 (column I) for comparison with the HEU core and the old LEU core design. The ONB power for the new LEU design is higher than that for the old LEU design, mainly due to a smaller radial power factor in plate 1348 and due to a smaller channel thickness in the new LEU design. Reducing the channel thickness increases the ONB power because the new LEU design is approaching the optimum channel thickness (≈ 100 mil) which gives the highest ONB power.

Power Distribution in New LEU Core

Table Q27-2 compares the heating by assembly and in non-fueled components for the LEU core with the rods banked at 53.5 cm. The banked rod critical configuration was found to have the highest peak power density in the analysis of the HEU core. Elements 3-4, 4-3, and 4-4 are noted as having the highest average plate powers, and are therefore of interest for thermal-hydraulics analysis. Figure Q27-1 provides a schematic of the core layout and the plate orientations.

Table Q27-3 compares the power in individual plates in elements 4-4, 3-4, and 4-3 for the reactor with the banked rods critical configuration. The tallies were summed over the fuel meat in each fuel plate, all clad, coolant, and the bundle can. The plates are numbered from left-to-right in these elements (see the element drawings in Figure Q27-1). It can be seen that plate 1348 in assembly or bundle 4-4 has the highest power (8.07 W). This plate is adjacent to the large water hole that the SS1 rod falls into, and nearer the center of the reactor than plate 1355 on the other side of the water hole. Plates 1228 (bundle 3-4) and 1315 (bundle 4-3) face the center-line of the core, and have roughly equal power (6.51 and 6.41 W).

Figure Q27-2 compares the local-to-average axial power density profiles for fuel plates 1348, 1228, and 1315 for the banked rods critical configuration. Plate 1348 has the highest peak power density of all the plates in the LEU core and was evaluated in the thermal-hydraulics analyses. The slight "pinching" of the axial power profile due to the insertion of the SS1 shim rod is evident.

The local-to-average axial power density profiles in plates 1228 and 1315 are nearly identical due to their symmetric positioning across the core center-line. For these two plates, the highest peak/average density (1.712) occurs in plate 1228. The peak power density in plate 1228 of standard element 3-4 is about 22% lower than that of plate 1348. Furthermore, the coolant channel thickness in the standard fuel assembly is narrower than that in the control assembly (144 vs. 181 mils). This reduction in channel thickness brings the channel thickness closer to the optimum value (≈ 100 mil), which will result in a higher ONB power with natural convection cooling. Therefore, plate 1348 is the most limiting of all fuel plates.

A higher water-to-fuel ratio at the edges of the fuel plates induces a power density profile along the width of the fuel plates. This is shown in Figure Q27-3 for plates 1348, 1228, and 1315 with the rods in the banked critical position. The radial segments are numbered from bottom-to-top in the MCNP model (see the plate orientations indicated in Figure Q27-1). It is expected that the power density will be higher at the edge of the plate closest to the core center-line.

Lastly, Tables Q27-4, Q27-5, and Q27-6 provide the axial and radial power profiles for plates 1348, 1228, and 1315, respectively. These data were used for thermal-hydraulic analysis of the LEU-fueled PUR-1.

Table Q27-2 is related to Table 4-19 in the August 2006 Conversion Proposal for the old LEU core design. Table Q27-3 is related to Table 4-20 in the Conversion Proposal. Tables Q27-4, Q27-5, and Q27-6 are related to Tables 4-21 and 4-22 in the Conversion Proposal.

Figure Q27-1 is related to Figure 4-35 in the August 2006 Conversion Proposal for the old LEU core design. Figures Q27-2 and Q27-3 are related to Figures 4-36 and 4-37 in the Conversion Proposal.

Table Q27-1. ONB Powers of HEU and LEU Cores With Hot Channel Factor FW to Account for Hydrodynamically Developing Flow, and the Effect of Viscosity Temperature Dependence on FBULK (Most Limiting Fuel Plate of Each Core)

	HEU Plate 262[1]				LEU Plate 1348[2]				
Channel Type	Plate-to-Plate 20 mil Uncertainty on 207 mil Channel				Plate-to-Plate 20 mil Uncertainty on 197 mil Channel (Old Design)				Plate-to-Plate 20 mil Uncertainty on 181 mil Channel (New Design)
Column Identification	A	B	C	D	E	F	G	H	I
Hot Channel Factor FW	1.000	1.054[3]	1.054		1.000	1.048[4]	1.048	1.048	1.048
Hot Channel Factor FBULK	1.312[5]	1.312	1.301[5]		1.321[6]	1.321	1.308[6]	1.308	1.322[8]
Hot Channel Factor FFILM	1.154	1.154	1.154		1.248	1.248	1.248	1.355[7]	1.357[8]
ONB Power (kW)	76.3	75.8	75.9		96.1	95.7	95.8	88.6	94.2
ΔP Due to Wall Shear, Pa	11.88	12.53	12.54		12.60	13.21	13.21	12.74	14.88
Inlet + Outlet Loss ΔP, Pa	2.29	2.32	2.32		2.13	2.15	2.16	1.98	2.01
Total Friction ΔP, Pa	14.17	14.85	14.86		14.73	15.36	15.37	14.72	16.89
ΔP Due to Buoyancy, Pa	14.17	14.85	14.86		14.73	15.36	15.37	14.72	16.89
Flow rate in hot channel, g/s	20.83	19.89	19.90		19.12	18.34	18.36	17.61	16.28
Reynolds Number at Channel Outlet	844	816	816		792	769	770	730	698

1. Other hot channel factors used for HEU plate 262 are taken from Table 4-25 of the conversion proposal: F_Q = 1.500, F_H = 1.200, FFLUX = 1.120

2. Other hot channel factors used for LEU plate 1348 are taken from Table 4-26 of the conversion proposal: F_Q = 1.500, F_H = 1.200, FFLUX = 1.226

3. The ratio of apparent friction parameter C_{app} for *developing* laminar flow to the friction parameter C for *fully developed* laminar flow, C_{app}/C is 1.1105 for the 207 mil thick channel in the HEU core. This ratio was calculated using Eqs. (341) and (576) of Shah and London [Ref. 2]. FW in the NATCON code input was set to $1.1105^{0.5}$ because the frictional pressure drop is multiplied by a factor FW^2 by the code.

4. The ratio of apparent friction parameter C_{app} for *developing* laminar flow to the friction parameter C for *fully developed* laminar flow, C_{app}/C is 1.0985 for the 197 mil thick channel in the LEU core. This ratio was calculated using Eqs. (341) and (576) of Shah and London [Ref. 2]. FW in the NATCON code input was set to $1.0985^{0.5}$ because the frictional pressure drop is multiplied by a factor FW^2 by the code.

5. For the HEU core (20 mil uncertainty on 207 mil thick channel), the hot channel factor for bulk coolant temperature rise, FBULK, calculated ignoring the temperature-dependence of water viscosity is 1.312 (as reported in Table 4-25 of the conversion proposal). The revised value including the temperature-dependence of water viscosity is 1.301 (calculated using the formula in the revised Appendix 1).

6. For the LEU core (20 mil uncertainty on 197 mil thick channel), the hot channel factor for bulk coolant temperature rise, FBULK, calculated ignoring the temperature-dependence of water viscosity is 1.321 (as reported in Table 4-26 of the conversion proposal). The revised value including the temperature-dependence of water viscosity is 1.308 (calculated using the formula in the revised Appendix 1).

7. The value of FFILM that accounts for the power density variation over the plate width, as explained in response to Question number 27.

8. FBULK has changed to 1.322 due to 20 mil uncertainty on 181 mil channel thickness in the new LEU core design as shown in Table Q27-7. The FFILM in Table Q27-7 (1.251) was increased by a factor of 1.085 to account for the power density variation along the width of plate 1348 (shown in Table Q27-4). This results in FFILM = 1.357 (=1.0850x1.251) that is used here.

Table Q27-2. PUR-1 LEU Rotated Core Power Distribution for Banked Rod Critical Configuration.

	Power (W)	Average Plate Power (W)
2-2 (RR)	38.19	4.77
2-3	61.85	4.76
2-4 (SS2)	45.65	5.71
2-5	45.93	3.83
3-2	62.56	4.81
3-3	78.09	6.01
3-4	**78.42**	**6.03**
3-5	61.20	4.71
4-2	63.52	4.89
4-3	**80.78**	**6.21**
4-4 (SS1)	**60.09**	**7.51**
4-5	62.29	4.79
5-2	52.32	4.03
5-3	63.97	4.92
5-4	63.49	4.88
5-5	47.64	3.97
Inter-assembly water	3.78	
Graphite reflector	9.52	
Grid plate	2.27	
Water reflector (pool)	18.00	
SS1	0.23	
SS2	0.17	
RR	0.03	
Total	**1000.0**	

Purdue LEU Core Layout (191 plates)

Figure Q27-1. PUR-1 New LEU Core Design Layout. Numbers in parentheses reflect number of fueled plates in element.

Table Q27-3. Plate Power in Bundles 4-4, 3-4, and 4-3 in New Design of PUR-1 LEU Core with 191 Fuel Plates for Banked Rod Critical Configuration.

Bundle 4-4	**Power (W)**
Plate 1345 Meat	7.00
Plate 1346 Meat	7.09
Plate 1347 Meat	7.40
Plate 1348 Meat	**8.07**
Plate 1355 Meat	7.74
Plate 1356 Meat	6.98
Plate 1357 Meat	6.59
Plate 1358 Meat	6.43
Clad	0.36
Water	2.19
Can	0.24
Total	**60.09**
Bundle 3-4	**Power (W)**
Plate 1215 Meat	5.80
Plate 1216 Meat	5.56
Plate 1217 Meat	5.49
Plate 1218 Meat	5.50
Plate 1219 Meat	5.61
Plate 1220 Meat	5.84
Plate 1222 Meat	5.91
Plate 1223 Meat	5.76
Plate 1224 Meat	5.73
Plate 1225 Meat	5.76
Plate 1226 Meat	5.92
Plate 1227 Meat	6.12
Plate 1228 Meat	**6.51**
Clad	0.48
Water	2.19
Can	0.24
Total	**78.42**
Bundle 4-3	**Power (W)**
Plate 1315 Meat	**6.41**
Plate 1316 Meat	6.16
Plate 1317 Meat	6.07
Plate 1318 Meat	6.04
Plate 1319 Meat	6.09
Plate 1320 Meat	6.28
Plate 1322 Meat	6.21
Plate 1323 Meat	5.92
Plate 1324 Meat	5.76
Plate 1325 Meat	5.68
Plate 1326 Meat	5.65
Plate 1327 Meat	5.68
Plate 1328 Meat	5.84
Clad	0.49
Water	2.19
Can	0.24
Total	**80.71**

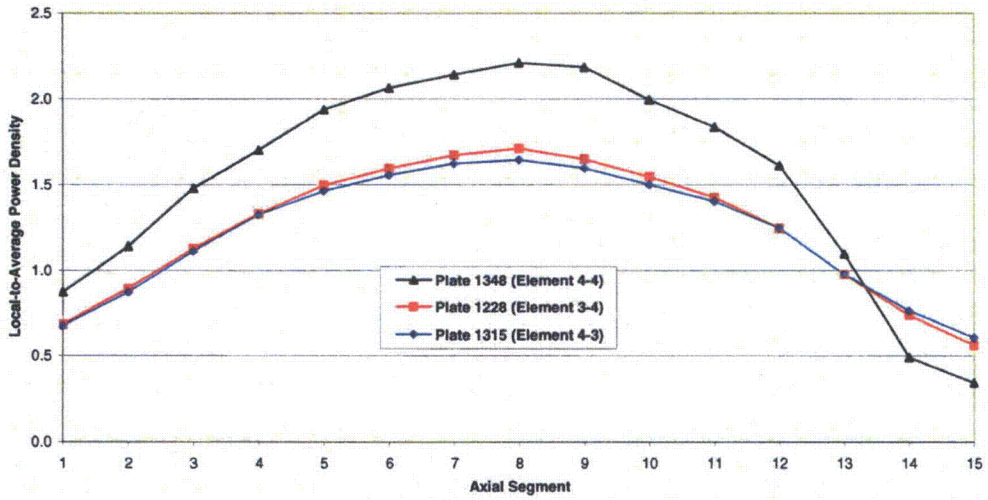

Figure Q27-2. Axial Power Profiles in LEU Plates 1348, 128, and 1315 for Banked Rod Critical Configuration in New LEU Core Design.

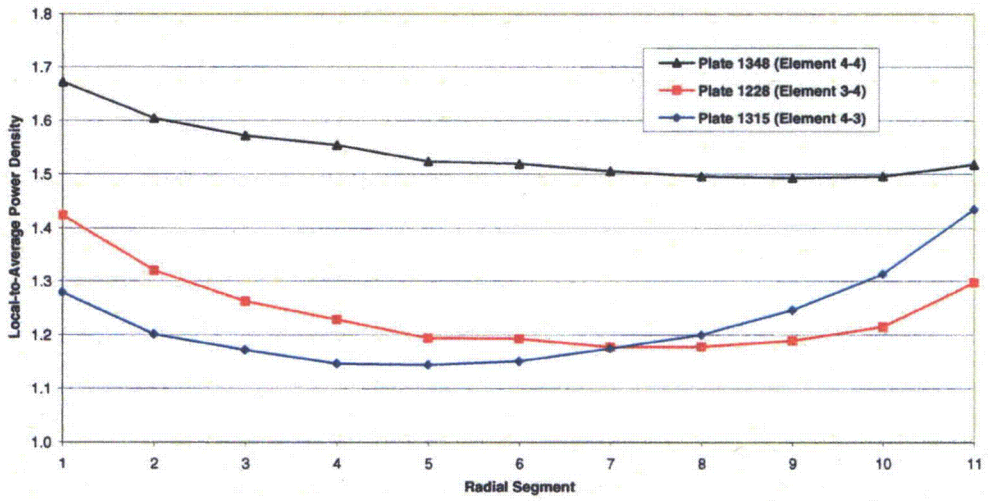

Figure Q27-3. Radial Power Profiles in LEU Plates 1348, 128, and 1315 for Banked Rod Critical Configuration in New LEU Core Design

Table Q27-4. Axial and Radial Heating Profile for LEU Plate 1348 of Bundle 4-4 for Banked Critical Configuration in New LEU Core Design.

Axial Segment	z-low[1] (cm)	z-high[1] (cm)	Power (W)	□
1	1.88595	5.88645	0.304	0.99%
2	5.88645	9.88695	0.400	0.86%
3	9.88695	13.88745	0.517	0.76%
4	13.88745	17.88795	0.595	0.70%
5	17.88795	21.88845	0.676	0.66%
6	21.88845	25.88895	0.720	0.64%
7	25.88895	29.88945	0.747	0.63%
8	29.88945	33.88995	0.771	0.62%
9	33.88995	37.89045	0.762	0.63%
10	37.89045	41.89095	0.697	0.65%
11	41.89095	45.89145	0.641	0.68%
12	45.89145	49.89195	0.563	0.74%
13	49.89195	53.89245	0.385	0.86%
14	53.89245	57.89295	0.172	1.25%
15	57.89295	61.89345	0.120	1.48%
Total			**8.070**	**0.21%**
Radial Segment	y-low[1] (cm)	y-high[1] (cm)	Local/Average Power Density	□
1	24.0838	24.6253	1.672	0.48%
2	24.6253	25.1668	1.604	0.48%
3	25.1668	25.7082	1.572	0.48%
4	25.7082	26.2497	1.554	0.49%
5	26.2497	26.7912	1.523	0.49%
6	26.7912	27.3327	1.519	0.49%
7	27.3327	27.8742	1.506	0.49%
8	27.8742	28.4157	1.496	0.50%
9	28.4157	28.9571	1.493	0.50%
10	28.9571	29.4986	1.496	0.50%
11	29.4986	30.0401	1.518	0.50%

[1]Positions correspond to MCNP model of PUR-1.

Table Q27-5. Axial and Radial Heating Profile for LEU Plate 1228 of Bundle 3-4 for Banked Critical Configuration in New LEU Core Design.

Axial Segment	z-low[1] (cm)	z-high[1] (cm)	Power (W)	□
1	1.88595	5.88645	0.236	1.07%
2	5.88645	9.88695	0.311	0.94%
3	9.88695	13.88745	0.395	0.83%
4	13.88745	17.88795	0.465	0.77%
5	17.88795	21.88845	0.523	0.73%
6	21.88845	25.88895	0.557	0.70%
7	25.88895	29.88945	0.584	0.70%
8	29.88945	33.88995	0.598	0.69%
9	33.88995	37.89045	0.576	0.69%
10	37.89045	41.89095	0.540	0.71%
11	41.89095	45.89145	0.498	0.74%
12	45.89145	49.89195	0.436	0.80%
13	49.89195	53.89245	0.342	0.89%
14	53.89245	57.89295	0.255	1.05%
15	57.89295	61.89345	0.195	1.17%
Total			**6.511**	**0.22%**
Radial Segment	y-low[1] (cm)	y-high[1] (cm)	Local/Average Power Density	□
1	24.0838	24.6253	1.424	0.52%
2	24.6253	25.1668	1.320	0.52%
3	25.1668	25.7082	1.262	0.53%
4	25.7082	26.2497	1.229	0.54%
5	26.2497	26.7912	1.194	0.54%
6	26.7912	27.3327	1.193	0.55%
7	27.3327	27.8742	1.178	0.54%
8	27.8742	28.4157	1.178	0.55%
9	28.4157	28.9571	1.189	0.54%
10	28.9571	29.4986	1.215	0.54%
11	29.4986	30.0401	1.297	0.54%

[1]Positions correspond to MCNP model of PUR-1.

Table Q27-6. Axial and Radial Heating Profile for LEU Plate 1315 of Bundle 4-3 for Banked Critical Configuration in New LEU Core Design.

Axial Segment	z-low[1] (cm)	z-high[1] (cm)	Power (W)	□
1	1.88595	5.88645	0.233	1.07%
2	5.88645	9.88695	0.303	0.96%
3	9.88695	13.88745	0.389	0.85%
4	13.88745	17.88795	0.463	0.79%
5	17.88795	21.88845	0.511	0.75%
6	21.88845	25.88895	0.543	0.72%
7	25.88895	29.88945	0.567	0.70%
8	29.88945	33.88995	0.574	0.70%
9	33.88995	37.89045	0.558	0.70%
10	37.89045	41.89095	0.523	0.73%
11	41.89095	45.89145	0.490	0.76%
12	45.89145	49.89195	0.436	0.81%
13	49.89195	53.89245	0.342	0.90%
14	53.89245	57.89295	0.265	1.02%
15	57.89295	61.89345	0.210	1.13%
Total			**6.407**	**0.23%**
Radial Segment	y-low[1] (cm)	y-high[1] (cm)	Local/Average Power Density	□
1	16.3894	16.9309	1.279	0.54%
2	16.9309	17.4724	1.201	0.55%
3	17.4724	18.0138	1.172	0.56%
4	18.0138	18.5553	1.147	0.56%
5	18.5553	19.0968	1.144	0.57%
6	19.0968	19.6383	1.151	0.56%
7	19.6383	20.1798	1.174	0.55%
8	20.1798	20.7213	1.199	0.55%
9	20.7213	21.2627	1.246	0.54%
10	21.2627	21.8042	1.313	0.52%
11	21.8042	22.3457	1.435	0.51%

[1]Positions correspond to MCNP model of PUR-1.

ONB Power for New LEU Core

Table Q27-7 shows the hot channel factors for the new LEU core design, which were calculated using the equations in the revised Appendix 1 that includes the effect of temperature-dependent water viscosity. A conservative uncertainty of 20 mil (rather than 15 mil shown in Table Q5B-1) on the 181 mil channel thickness (the most limiting fuel plate 1348 in control assembly 4-4) was used in finding the hot channel factors. To calculate the ONB power, the NATCON code was run using

(i) the hot channel factors shown in Table Q27-7,
(ii) a total of 191 fuel plates,
(iii) a channel thickness of 181 mil,
(iv) a radial power factor of 1.5414 (=the ratio of 8.07 kW power in plate 1348 to 5.2356 kW per average plate), and
(v) the axial power shape for plate 1348 shown in Table Q27-4.

Table Q27-7. Hot Channel Factors for the New LEU Core [1]					
			Hot Channel Factors		
Uncertainty	**Type of Tolerance**	**Tolerance Fraction**	**FBULK Coolant Temp. Rise**	**FFILM Film Temp Rise**	**FFLUX Heat Flux**
Fuel meat thickness	Random	0.000	1.000	1.000	1.000
U-235 Homogeneity	Random	0.200	1.000	1.200	1.200
U-235 Mass per plate	Random	0.030	1.014	1.030	1.030
Power Density	Random	0.100	1.046	1.100	1.100
Channel Thickness	Random	0.110	1.180	1.111	1.000
Flow Distribution	Random	0.200	1.200	1.000	1.000
Random Uncertainties	**Combined**		**1.322**	**1.251**	**1.226**
Power Measurement F_Q	Systemic	0.500	**1.500**		
Flow friction factor F_W	Systemic	0.048	**1.048**		
Heat Transfer Coeff. F_H	Systemic	0.200	**1.200**		

1. Plate-to-plate channel, 20 mil uncertainty on 181 mil channel thickness. The revised Appendix 1 was used to include the effect of temperature-dependent water viscosity.

To account for the power density variation along the width of plate 1348, the FFILM in Table Q27-7 (1.251) was increased by a factor of 1.085 (= 1.672/1.5412 = the maximum-to-average power density ratio variation over the width of plate 1348, shown in Table Q27-4). This results in FFILM =1.085x1.251 = 1.357 that was used in NATCON to calculate the ONB power.

The NATCON code calculates the Darcy-Weisbach friction factor f = C/Re for fully developed laminar flow, using a built-in table of the parameter C for different aspect ratios of the rectangular channel cross section (values of parameter C are given in the response to Question number 29). To account for the increased pressure drop due to hydrodynamically developing laminar flow in the channel, an apparent value of the parameter C averaged over the channel length, called C_{app}, was calculated using Eq. (576) of Shah and London [Ref. 2]. The ratio C_{app}/C was found to be 1.0897 at a Reynolds number of 800 at the exit of the 181 mil channel in the new LEU design. Since the NATCON code multiplies the fully developed friction factor by FW^2, the hot channel factor FW equals 1.044 (= $1.0897^{0.5}$). A higher value of 1.048 for FW was used in the NATCON calculation to be conservative.

The ONB power is found to be 94.2 kW which is shown in Table Q27-1 (column I) for comparison.

The step-by-step change in the ONB power of 88.6 kW for the old LEU design, due to each design change, to the ONB power of 94.2 kW for the new LEU design, was determined by running NATCON with one input change at time. These results are summarized in Table Q27-8 below.

Table Q27-8. Effect of LEU Assembly Design Changes on ONB Power.

	Design Parameter Change From the Old LEU Design to the New LEU design		**ONB power Change, kW**
1	Channel thickness plate-to-plate, mil	197 → 181	88.6 → 95.1
2	Radial power factor of plate 1348	1.6404 → 1.5414	95.1 → 101.2
3	Axial power shape (peak-to-average ratio)	1.345 → 1.436	101.2 → 94.0
4	All hot channel factors changed: FBULK FFILM	1.308 → 1.322 1.355 → 1.357	94.0 → 93.7
5	Number of fuel plates	190 → 191	93.7 → 94.2

Question 28

28. Appendix 1. From the information in Appendix 1 it is not clear how insignificant are the channel inlet and outlet losses when compared to the wall shear. Please clarify.

Response:

The information in Appendix 1 was used only to obtain hot channel factors for input to a more detailed thermal-hydraulic calculation using the NATCON code [Ref. 8 of the conversion proposal]. Therefore, Appendix 1 is a simplified modeling of what is calculated in detail in NATCON, and it is used only for the purpose of obtaining closed-form equations from which hot channel factors could be found. Appendix 1 does not include the minor losses. The minor losses calculated by NATCON are reported below, and found to be about 16% of the total frictional pressure drop in the HEU core, and 14% of the total frictional pressure drop in the LEU core (see Table Q27-1).

The pressure drop due to inlet and outlet losses were calculated (by the NATCON code) using loss coefficients of 0.5 and 1.0 respectively. The pressure drop due to wall shear along the channel length is found by summing the pressure drop for each axial mesh which is calculated using temperature-dependent coolant viscosity and density for the axial mesh (14 mesh intervals were used over the channel length in all calculations). The pressure drops are calculated by NATCON assuming fully developed laminar flow in a rectangular cross-section channel, and then multiplied by a factor FW^2 (FW squared) where FW is an input which may be used to account for the increased pressure drop due to hydrodynamically developing laminar flow. In the calculations presented in the conversion proposal, FW was always set to 1.0, and thus the increased pressure drop due to developing laminar flow was not included. It is included in the calculations presented here (Table Q27-1). The method used is described below.

For the most limiting fuel plate in Table 4-27 of the conversion proposal for each core (HEU and LEU), a comparison of the pressure drops due to inlet plus outlet loss and wall shear, with and without the effect of developing laminar flow, are tabulated in Table Q27-1.

NATCON calculates the Darcy-Weisbach friction factor f = C/Re for laminar flow, using a built-in table of the parameter C for different aspect ratios of the rectangular channel cross section (values of parameter C are given in the response to Question number 29). An apparent value of the parameter C averaged over the channel length, called C_{app}, was calculated using Eq. (576) of Shah and London [Ref. 2 listed at the end of all responses] to account for the increased pressure drop due to hydrodynamically developing laminar flow in the channel. The ratio C_{app}/C was found to be 1.1105 for the 207 mil HEU channel, 1.0985 for the 197 mil LEU channel. Since the NATCON code multiplies the fully developed friction factor by FW^2 as mentioned above, the input FW equals 1.054 and 1.048 for the HEU and LEU channels respectively. NATCON calculations were done using these values of FW, and the pressure drops due to inlet plus outlet loss and wall shear are compared in Table Q27-1 (column B for the HEU channel, and column F for the LEU channel).

Table Q27-1 shows that the pressure drops due to wall shear and minor losses are 84% and 16%, respectively, of the total pressure drop in the HEU channel at its ONB power; and the pressure drops due to wall shear and inlet plus outlet loss are 86% and 14%, respectively, of the total pressure drop in the LEU channel at its ONB power.

Question 29

29. Appendix 1. From the information in Appendix 1 it is not clear what is the functional dependency of the laminar friction parameter C to the channel cross-section dimensions. Provide a reference for the evaluation of C.

Response:

The following values (rows 1 and 2 of Table Q29-1) of the parameter C for fully developed laminar flow in a channel of *rectangular cross section* versus the width-to-thickness aspect ratio (w_c/t_c) of the channel are used in the NATCON code that was used in the thermal-hydraulics calculations. The table starts from the square cross section (aspect ratio = 1.0) and goes to the infinite value of the aspect ratio (parallel plates). In order to find the parameter C for the aspect ratio of the PUR-1 reactor, the NATCON code simply interpolates between the tabulated values. The original author of the code obtained these values from an old Reference [E. R. G. Eckert and T. F. Irvine, Heat Transfer Laboratory, University of Minnesota (1957)] but these values are also given in a textbook by Frank Incropera [Ref. 3]. These values are obtained from the closed-form analytical solution for the fully developed laminar velocity distribution in a rectangular channel summarized by R. K. Shah and A. L. London [Ref. 2]. Equation (341) in [Ref. 2] is a fitted equation to easily find the parameter C. It should be noted that the aspect ratio used in [Ref. 2] is channel thickness-to-width ratio (the reciprocal of that used in NATCON and shown below in Table Q29-1), and the friction factor in [Ref. 2] should be multiplied by 4 to get the Darcy-Weisbach friction factor that is used in NATCON and tabulated below.

Table Q29-1. Friction Parameter C Used in the NATCON Code

w_c/t_c	1.0	2.0	3.0	4.0	5.0	6.3	8.0	11.0	15.0	18.0	100.0
C in NATCON	58.0	63.0	69.0	72.5	77.0	80.0	83.0	85.0	88.0	89.0	96.0
C in Ref. 3	57.0	62.0	69.0	73.0			82.0				96.0
C in Ref. 2	56.9	62.2	68.4	72.9	76.3	79.5	82.4	85.6	88.1	89.3	94.7

Question 30

30. Appendix 1. From the information in Appendix 1 in both the calculation of the channel flow and the calculation of the bulk coolant temperature rise the ratio of the coolant kinematic viscosity to density (μ/ρ) was assumed to be insensitive to temperature. Please demonstrate the validity of this assumption.

Response:

The information in Appendix 1 was used only to obtain hot channel factors for input to a more detailed thermal-hydraulic calculation using the NATCON code [Ref. 8 of the conversion proposal]. NATCON does account for the temperature dependence of coolant viscosity and density in the calculation of the channel flow and the calculation of the bulk coolant temperature rise. Therefore, Appendix 1 is a simplified modeling of what is calculated in detail in NATCON, for the purpose of obtaining closed-form equations from which hot channel factors could be found.

As suggested in the question, water viscosity is temperature-dependent, i.e., it decreases with rising temperature. Appendix 1 was revised to account for the effect of temperature dependence of viscosity on hot channel factors, and the revised Appendix 1 is enclosed herewith. The temperature dependence of the dynamic viscosity of water over the temperature range 27 °C ≤ T ≤ 50 °C (adequate for the PUR-1 reactor) can be written as follows.

$$\mu(T) = \mu(T_0)\,(1+T-T_0)^{-\alpha} \qquad \text{(A1)}$$

where $\alpha = 0.12$
$T_0 = 27$ °C = Pool temperature of PUR-1

$\mu(T_0) = 0.875 \times 10^{-3}$ N-s/m ^{2}
$\mu(T)$ = Temperature-dependent dynamic viscosity of water, N-s/m^2

As shown in the revised Appendix 1, the revised relationship between the flow rate W in a channel and the friction parameter C is given by Eq. (A2). The revised formula for hot channel factor FBULK for bulk coolant temperature rise is given by Eq. (A3).

$$W \propto \left(\frac{1}{C}\right)^{\frac{1}{2+\alpha}} \tag{A2}$$

$$\text{FBULK} = 1 + \sqrt{\left\{ (1+u_1)^{\frac{1}{2+\alpha}} (1+u_2)^{\frac{1}{2+\alpha}} \left(\frac{1}{1-u_5}\right)^{\frac{3}{2+\alpha}} - 1 \right\}^2 + u_6^{\,2}} \tag{A3}$$

The exponent on the right hand side of Eq. (A2) changed from 0.5 (in the conversion proposal ignoring temperature dependence of μ) to the revised value 1/2.12 = 0.4717. There exponents in Eq. (A3) for FBULK also changed, e.g., from 3/2 to 3/2.12 = 1.415. As a result of this revision, the hot channel factor FBULK decreased from 1.312 (in the conversion proposal) to 1.301 for the most limiting fuel plate 262 in the HEU core. Similarly, FBULK decreased from 1.321 (in the conversion proposal) to 1.308 for the most limiting fuel plate 1348 in the LEU core. The effect of ignoring the temperature dependence of viscosity is conservative.

NATCON calculations were done with these revised values of FBULK along with a value of FW > 1.0 to account for the increased friction due to developing laminar flow (in response to Question number 33). The results are shown in Table Q27-1 (column C for the HEU core, and column G for the LEU core).

As a consequence of the two effects (i.e., increased friction due to developing laminar flow and the temperature dependence of viscosity) on hot channel factors FW and FBULK, the ONB power of the HEU core changes from 76.3 kW (reported in the conversion proposal) to 75.9 kW, and the ONB power of the LEU core changes from 96.1 kW (reported in the conversion proposal) to 95.8 kW. The effect is small for the PUR-1 reactor.

Question 31

31. Table 4-24. A pool temperature (coolant inlet temperature) of 27°C is used in the LEU thermal-hydraulic analyses. The TSs contain no limits on pool temperature. Please explain why a limit on pool temperature should not be added to the TSs or propose a limit on pool temperature.

Response:

The average pool temperature in recent PUR-1 operating history is 26°C. An estimated value of 27°C was used in the calculations. The figure below shows measured pool temperatures from 1993 to 2006.

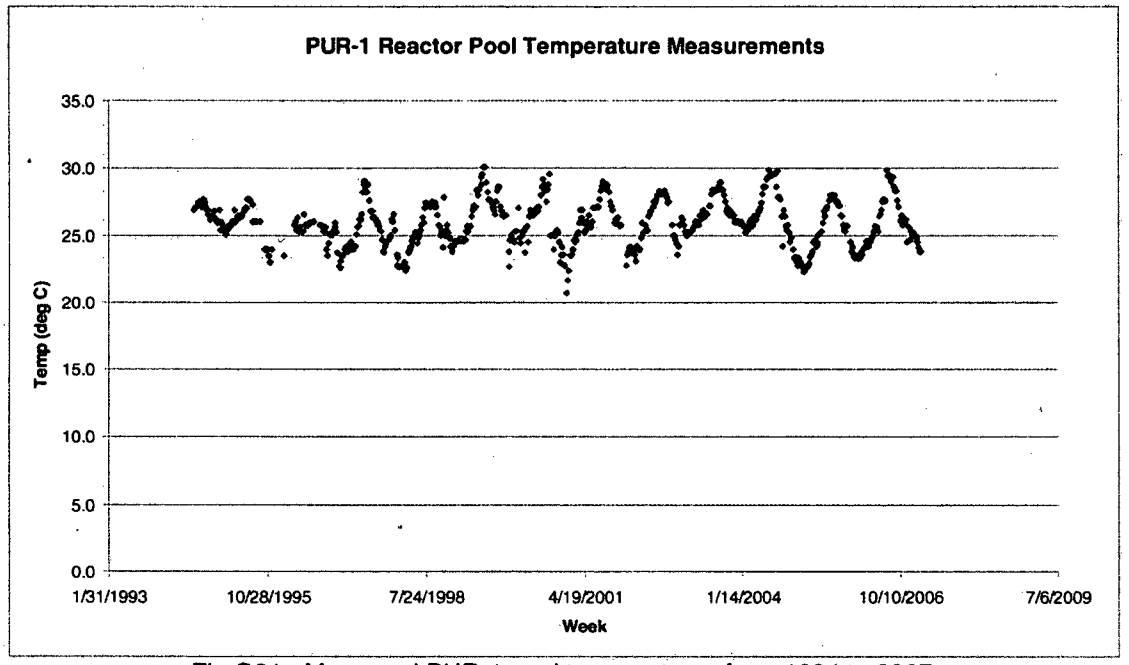

Fig Q31: Measured PUR-1 pool temperatures from 1994 to 2007.

Pool temperatures up to 30°C will have no effect on reactor safety. Peak temperatures in the thermal-hydraulic and transient analyses would be about 3°C higher than those reported in the conversion proposal. All safety margins would be far below their limiting values. However, PUR-1 Staff believes that the maximum pool temperature is not a HEU to LEU conversion issue, since there is currently no technical specification on maximum pool temperature for the HEU core.

However, a NATCON thermal-hydraulic calculation for the LEU plate 1348 (for case H reported in Table Q27-1 included in response to Question 27) was performed assuming a hypothetical pool temperature of 35 °C, and a hypothetical inlet loss coefficient of 10.0 (increased from 0.5), while applying all six hot channel factors of the case H. The ONB power was found to be 79.3 kW, indicating a large margin compared to the PUR-1 operating power of 1 kW.

Question 32

32. Appendix 1. Equation (30) has two terms and the conversion proposal states that the expression within the parenthesis on the right hand side of the equation varies slowly compared to the heat flux t_{fuel} q'''/2. Demonstrate the validity of the statement with reference to the PUR-1 fuel plate.

Response:

Equation (30) of Appendix 1 is for finding a hot channel factor for the temperature drop from the meat mid-plane to cladding surface (ΔT_{metal}). This temperature drop is very small compared to the temperature drop from the cladding surface to bulk coolant (ΔT_{film}). For example, in the PUR-1 HEU fuel plate 262 without hot channel factors, ΔT_{metal} is 0.07 °C and ΔT_{film} is 46.98 °C (at meat mid-height) at a high power of 100 kW. Similarly, in the PUR-1 LEU fuel plate 1348 without hot channel factors, ΔT_{metal} is 0.05 °C and ΔT_{film} is 34.5 °C at a power of 100 kW. Therefore, the hot channel factor for ΔT_{metal} is not important for PUR-1. The important hot channel factor is the factor FFILM for ΔT_{film}. In the case of PUR-1, ΔT_{film} is the bigger component (bigger than the bulk coolant temperature rise) in the total temperature rise from the inlet temperature to the cladding surface temperature at the axial level experiencing the onset of nucleate boiling. The hot channel factor FFILM found by Eq. (29) of Appendix 1 in the conversion proposal remains unchanged. It depends on the uncertainties in $q'''t_{fuel}$ and channel thickness (as shown in Eq. 28), but not on the uncertainty in $[t_{fuel}/(4K_{fuel}) + t_{clad}/K_{clad}]$.

In short, PUR-1 is not limited by the fuel peak temperature, but by the onset of nucleate boiling, and the uncertainty in $[t_{fuel}/(4K_{fuel}) + t_{clad}/K_{clad}]$ is not important for PUR-1. We believe that the hot channel factor FFILM has been determined accurately.

Question 33

33. Section 4.7.2. According to Appendix 1 the systematic uncertainty in flow rate is accounted for by applying the hot channel factor Fw to the laminar friction factor C. Explain the reason for the value of the flow friction factor Fw being unity in Tables 4-25 and 4-26.

Response:

As suggested in the question, a value of FW (hot channel factor for flow) greater than 1.0 should be used to account for the increased frictional pressure drop due to the hydrodynamically developing laminar flow in the entrance region of the coolant channel, otherwise the code (NATCON) accounts only for the fully developed frictional pressure drop. This has been done now and the results are presented in Table Q27-1. Since each coolant channel creates its own buoyancy to drive its own coolant flow, there is no uncertainty due to *redistribution* of a total reactor flow rate. The loss coefficients of 0.5 and 1.0 at channel inlet and outlet are used in the calculations. To account for the reduction in flow rate due to the hydrodynamically developing laminar flow in the channel, the values of FW were calculated for the most limiting channels in the HEU and LEU cores as follows.

NATCON calculates the Darcy-Weisbach friction factor f = C/Re using a built-in table of the parameter C for different aspect ratios of the rectangular channel cross section (values of parameter C are given in the answer to Question number 29). These values of parameter C are for the fully developed laminar flow in a rectangular cross-section channel. An apparent value of the parameter C averaged over the channel length, called C_{app}, was calculated using Eq. (576) of Shah and London [Ref. 2] to account for the increased pressure drop due to hydrodynamically developing laminar flow in the channel. The ratio C_{app}/C was found to be 1.1105 for the 207 mil HEU channel, and 1.0985 for the 197 mil LEU channel. Since the NATCON code multiplies the fully developed frictional factor by FW^2, the input FW equals 1.054 and 1.048 for the HEU and LEU channels respectively. The flow reduction factor is input factor FW or more accurately $FW^{2/(2+\alpha)} = FW^{0.9434}$ (noting that $\alpha = 0.12$ for the PUR-1 reactor as mentioned in the revised Appendix 1 enclosed herewith).

The results of using these values of FW in NATCON calculations (excluding the effect of temperature dependence of μ on hot channel factors) are shown in Table Q27-1. The ONB power of the HEU core changes to 75.8 kW from 76.3 kW reported in the conversion proposal. The ONB power of the LEU core changes to 95.7 kW from 96.1 kW reported in the conversion proposal.

The channel flow indeed gets reduced by the factor $FW^{0.9434}$ as expected. For the HEU plate 262, the flow reduces from 0.02083 kg/s to 0.01989 kg/s (see Table Q27-1) when the input hot channel factor FW is changed from 1.0 to 1.054. The expected reduced flow should be $0.02083/(1.054)^{0.9434} = 0.01982$ kg/s which is close to the NATCON-calculated value of 0.01989 kg/s. For the LEU plate 1348, the flow reduces from 0.01912 kg/s to 0.01834 kg/s (see Table Q27-1) when the input FW is changed from 1.0 to 1.048. The expected reduced flow should be $0.01912/(1.048)^{0.9434} = 0.01829$ kg/s which is close to the NATCON-calculated value of 0.01834 kg/s.

Question 34

34. Section 4.7.2. In Table 4-25 a random tolerance fraction of 0.05 was specified for the fuel meat of the HEU core. Explain why the corresponding tolerance fraction was given as 0.0 in Table 4-26 for the LEU core.

Response:

The reason is that the hot channel factors for the HEU fuel in Table 4-25 are appropriate for uranium-aluminum alloy fuel and the hot channel for the LEU fuel in Table 4-26 are appropriate for uranium-aluminum dispersion fuel.

U-Al alloy fuel plates have a smooth interface between the fuel meat and the cladding. The 0.05 tolerance fraction is appropriate to account for potential variation in the U-235 distribution due to variation of the thickness of the fuel meat. The homogeneity of the U-235 in U-Al alloy fuel is quite high, resulting in a tolerance fraction of 0.03.

Uranium-aluminum dispersion fuel is different from the U-Al alloy fuel. The U_3Si_2 particles in the fuel meat have a jagged profile, as shown in fuel micrographs. For dispersion fuel, the variation in both the "thickness" and homogeneity is combined in the tolerance on homogeneity, which is taken from the LEU silicide fuel manufacturing specifications. The manufacturer measures the areal density of U-235 along each fuel plate to quantify the axial variation of U-235. In this context, a tolerance fraction on fuel meat thickness is not appropriate for dispersion fuel, so its value in Table 4-26 was set to 0.0.

Question 35

35. Section 4.7.3.1. It is not clear if the hot channel factors were used in the determination of the ONB power. Please clarify.

Response:

The hot channel factors were indeed applied in the determination of the ONB powers reported in the conversion proposal. The ONB power is defined as that *nominal* power level at which the cladding surface temperature, with the five hot channel factors applied (F_Q, F_W, F_H, FBULK, and FFILM) at every axial location in the coolant channel, will equal the onset-of-nucleate-boiling temperature at any axial location in the channel. As an example, the ONB power of the HEU core calculated without applying any hot channel factors to the most limiting fuel plate 262 (plate-to-plate case) is 162.0 kW compared to 76.3 kW (as reported in the conversion proposal) with the hot channel factors applied. The ONB power of the LEU core calculated without applying any hot channel factors to the most limiting fuel plate 1348 (plate-to-plate case) is 220.4 kW compared to 96.1 kW (as reported in the conversion proposal) with the hot channel factors applied. In these calculations *with or without* applying hot channel factors, the radial power factor of the specified fuel plate (power of a specified plate divided by average power per plate) and the axial power distribution shape in the plate are always accounted for.

Question 36

36. Table 4-28. Define the parameter "margin to incipient boiling."

Response:

The margin to incipient boiling shown in Table 4-28 was calculated at the nominal operating power of PUR-1 (i.e., 1 kW), and it is the smallest value of the temperature difference (T_{ONB} - T_w) over the coolant channel length in the hottest channel where T_W is cladding surface temperature with all hot channel factors applied, and T_{ONB} is the local onset-of-nucleate-boiling temperature. This basically gives an idea of how far below the onset of nucleate boiling condition the reactor is operating. This definition can be written as an equation as follows:

Margin to ONB = Minimum $T_{incp}(p, q''(z)F_{flux}) - \left[F_{bulk}\left\{T(z) - T_0\right\} + F_{film}\left\{T_{wall}(z) - T(z)\right\}\right]$

where

$T(z)$	= Bulk coolant temperature at axial position z in the channel heated by the plate power of $P_{op}F_r F_Q/N$ and applying the global hot channel factors for flow and Nusselt number of F_w and F_h
$T_{wall}(z)$	= Cladding surface temperature at axial position z in the channel heated by a plate power of $P_{op}F_r F_Q/N$ and applying the global hot channel factors for flow and Nusselt number of F_w and F_h
$q''(z)$	= Heat flux at position z for the plate power of $P_{op}F_r F_Q/N$ and applying the global hot channel factors for flow and Nusselt number of F_w and F_h
$p(z)$	= Absolute pressure in the channel at axial position z
$T_{incp}(p(z), q''(z)F_{flux})$	= Onset of nucleate boiling temperature at absolute pressure p(z) and heat flux $q''(z)F_{flux}$
P_{op}	= Operating power of the reactor (e.g., 1 kW for PUR-1)
N	= Number of fuel plates in the core (e.g., 190 for PUR-1 LEU core)
T_0	= Coolant temperature at the channel inlet
F_r	= RPEAK = Radial power factor of the plate cooled by the channel
F_W	= Hot channel factor for flow in the channel
F_Q	= Hot channel factor for reactor power
F_h	= Hot channel factor for Nusselt number
F_{film}	= FFILM = Hot channel factor for temperature drop across the coolant film on cladding surface
F_{flux}	= FFLUX = Hot channel factor for heat flux
F_{bulk}	= FBULK = Hot channel factor for bulk coolant temperature rise in the channel

Question 37

37. Section 9.4. Please provide an analysis of storage of LEU fuel in the in-pool storage racks that shows that TS 5.3 will be met.

Response:

Purdue is scheduled to receive the following:

- 12 standard (14 plates max) assemblies and 2 spares
- 3 control (8 plates max) assemblies and 1 spare
- 1 fission chamber (14 plates max)
- 288 LEU plates.

An MCNP model of the in-pool storage racks was constructed. Two cases for the LEU fuel were examined. One case was run with standard LEU assemblies in all of the 18 positions (which is not possible with the anticipated LEU inventory, but was run as a limiting case), and no credit was taken for the ¼" BORAL plate between the two rows. This first case had a calculated eigenvalue of 0.7660±0.0046. The second case was modeled with the ¼" BORAL plate with a boron density of 23.8E21 $^{atoms}/_{cc}$, and 16 standard assemblies with 14 fuel plates in each. The eigenvalue for this calculation was determined to be 0.3319 ±0.00178. Both of these bracketing cases are below the TS 5.3 limit of 0.8, thus TS 5.3 will be met. The geometry of the model is shown in Figure Q37-1 below.

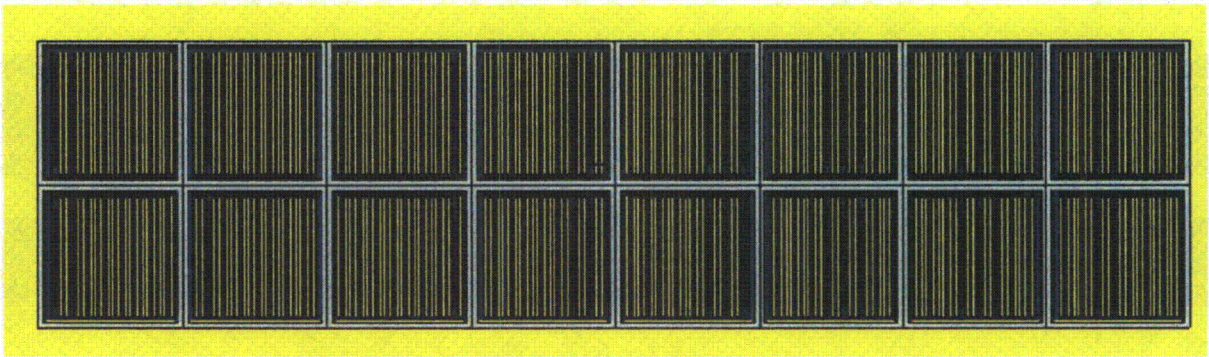

Figure Q37-1: Model of the in-pool fuel storage racks.

Question 38

38. Please propose license possession limits for the reactor conversion. Please provide the following possession limits and any other changes needed:

- amount of U-235 of enrichment less than 20 % possessed in the form of LEU fuel elements;
- amount of U-235 of any enrichment needed in connection with operation of the facility and the form of the material (e.g., fission chambers, flux foils, fueled experiments);
- amount of U-235 of any enrichment in the form of the existing HEU fuel elements until that core is removed from site.

Response:

Special Nuclear Material	**Chemical or Physical Form**	**Maximum Amount Possessed**
Uranium enriched in the U-235 Isotope	Al Clad MTR U-Al alloy Reactor Plates of any enrichment	3000 grams U-235
Uranium enriched in the U-235 Isotope	Al Clad MTR Reactor Plates enriched to ≤20 $^{w}/_{o}$.	3800 grams U-235
Uranium enriched in the U-235 Isotope	Flux foils, fission chambers, or fueled experiments of any enrichment	100 grams U-235

Question 39

39. Sections 13.2 and 13.4. What is the maximum amount of positive rapid reactivity and slow insertion (up to a maximum of the allowed excess reactivity) that can be added to the LEU fueled reactor without exceeding the proposed safety limit?

Response:

Previous Accident Analyses

The rapid insertion of the maximum worth of moveable and unsecured experiments (0.3% Δk/k in 0.1 seconds) was evaluated in the Conversion Proposal. In this accident analysis, it was assumed that the 7 second period trip failed, but scram was initiated due to the 1.2 kW over-power trip. Because of an assumed power measurement uncertainty of 50% in the PUR-1, the calculations were performed using a core power trip setting of 1.8 kW.

Another accident analyzed in the Conversion Proposal was the ramp insertion of 0.04% Δk/k/second. Again, it was assumed that the 7 second period trip failed, but scram was initiated due to the 1.2 kW over-

power trip. A core power trip setting of 1.8 kW was used in the calculations to take into account power measurement uncertainties in the PUR-1.

The analyses of both of these accidents demonstrated the protection of the safety limit by the limiting safety system setting.

New Accident Analyses Based on RAI

In order to address the question raised by the reviewers, additional accident analyses have been performed for the rapid and slow insertion of 0.6% Δk/k, the maximum excess reactivity allowed by the PUR-1 Technical Specifications. In these calculations, it was assumed that all trips failed.

In response to Question 20, the water temperature, water void, and fuel temperature coefficients were recalculated. The coefficients presented in Table Q20E-1 indicate that the reactivity coefficients are dependent on the temperature or void increase. The previous accident calculations were performed using the calculated reactivity coefficients with the smallest magnitude in order to provide the most conservative results. Since then, the PARET code has been modified to allow input of the reactivity coefficients as a function of temperature or void. Since the reactivity coefficients have been recalculated and the PARET code now has better modeling capabilities, the previous accident analyses (0.3% Δk/k rapid insertion and 0.04% Δk/k/second slow insertion) have been repeated, as well.

Table Q39.1 presents the results of 0.3% Δk/k rapid insertion and 0.04% Δk/k/second slow insertion accident analyses for the HEU and LEU cores. As discussed above, a core power trip of 1.8 kW was applied in the analyses of these accidents. These results indicate that the rapid insertion of the maximum worth of moveable and unsecured experiments in the PUR-1 results in a cladding temperature change of less than 1 °C from the steady-state peak clad temperature of 29 °C. The slow insertion of reactivity from a "runaway rod" accident also results in a less than 1 °C change in cladding temperature from the steady-state conditions. The temperatures reached in these accidents are well below the safety limit of 530 °C.

It should be noted that in the rapid insertion accident analyzed in the Conversion Proposal (Section 13.2), the total amount of reactivity inserted was inadvertently set to $0.30 instead of the desired 0.3% Δk/k. Consequently, the new results indicate that the peak power (and reactor scram) is reached sooner than the 2.5 seconds previously reported (see Table 13-1 of the Conversion Proposal).

Table Q39.1. Results for Rapid and Slow Reactivity Insertion Accidents With Scram. (Ref: Tables 13-1 and 13-2 in CP)

Fuel	Reactivity Inserted	P_o (kW)	P_{max} (kW)	Time of Peak Power (s)	$T_{clad,max}$ (°C) at t=0	at Peak Power	Maximum
HEU	0.3% rapid insertion	1.0	1.83	0.81	29	29	29
LEU	0.3% rapid insertion	1.0	1.83	0.75	30	30	30
HEU	0.04%/s slow insertion	1.0	1.83	5.92	29	29	29
LEU	0.04%/s slow insertion	1.0	1.83	5.85	30	30	30

Table Q39.2 presents the results of rapid and slow reactivity insertion accidents up to the maximum allowed excess reactivity in the PUR-1. These accidents were analyzed assuming failure of the reactor protection system and without operator intervention.

For the rapid insertion of the maximum core excess reactivity of 0.6% Δk/k in the HEU-fueled PUR-1, the reactor power increases to 1.63 MW. The clad temperature increases by about 100 °C due to this transient. For the LEU core, the reactor power increases to a peak of 1.55 MW. The total reactor power increase is less in this case because of the negative Doppler feedback effect in the LEU fuel. For the slow reactivity insertion accident (an insertion rate of 0.04% Δk/k/second was assumed) the results for

the peak power and final clad temperature are nearly the same as for the rapid insertion case, although the power and clad temperature profiles as a function of time are shifted due to the slower insertion of reactivity.

Table Q39.2. Results for Rapid and Slow Reactivity Insertion Accidents Without Scram.

Fuel	Reactivity Inserted	P_o (kW)	P_{max} (MW)	Time of Peak Power (s)	$T_{clad,max}$ (°C)		
					at t=0	at Peak Power	Maximum
HEU	0.6% rapid insertion	1	1.63	124	29	134	134
LEU	0.6% rapid insertion	1	1.55	400	30	120	120
LEU	0.6% slow insertion	1	1.55	400	30	120	120

Figures Q39.1 and Q39.2 plot the reactor power and maximum clad temperature as a function of time for the insertion of the maximum excess reactivity into the PUR-1 HEU and LEU cores. The results of the PARET analyses indicate that after the reactivity is inserted, the core reaches a new steady-state power level. The shift in the power response when the reactivity is inserted slowly is evident in the curves. At the new, higher power level, the reactivity feedback effects from coolant density, coolant temperature, and fuel temperature balance the reactivity inserted in the accident. As a result of this accident, the maximum cladding temperature is 134 °C in the HEU core. The maximum cladding temperature in the LEU core is 120 °C. These temperatures are well below the PUR-1 safety limit of 530 °C.

Included on Figures Q39.1 and Q39.2 are results from the SPERT-IV B-1 Test. In this test, 0.6% $\Delta k/k$ was rapidly inserted into an HEU-fueled MTR-type core designated as D12/25 (25 fuel elements with 12 plates/element). This core was similar to the PUR-1 HEU core. Prior to the reactivity insertion, the reactor was operating at low power with zero flow. The SPERT test was terminated by an operator-initiated scram after 30 seconds. The comparison of the SPERT data and the accident analyses with PARET for the PUR-1 with HEU fuel indicate good agreement for the first power peak and corresponding maximum clad temperature. The PARET code predicts the reactor power and maximum clad temperature over an extended period.

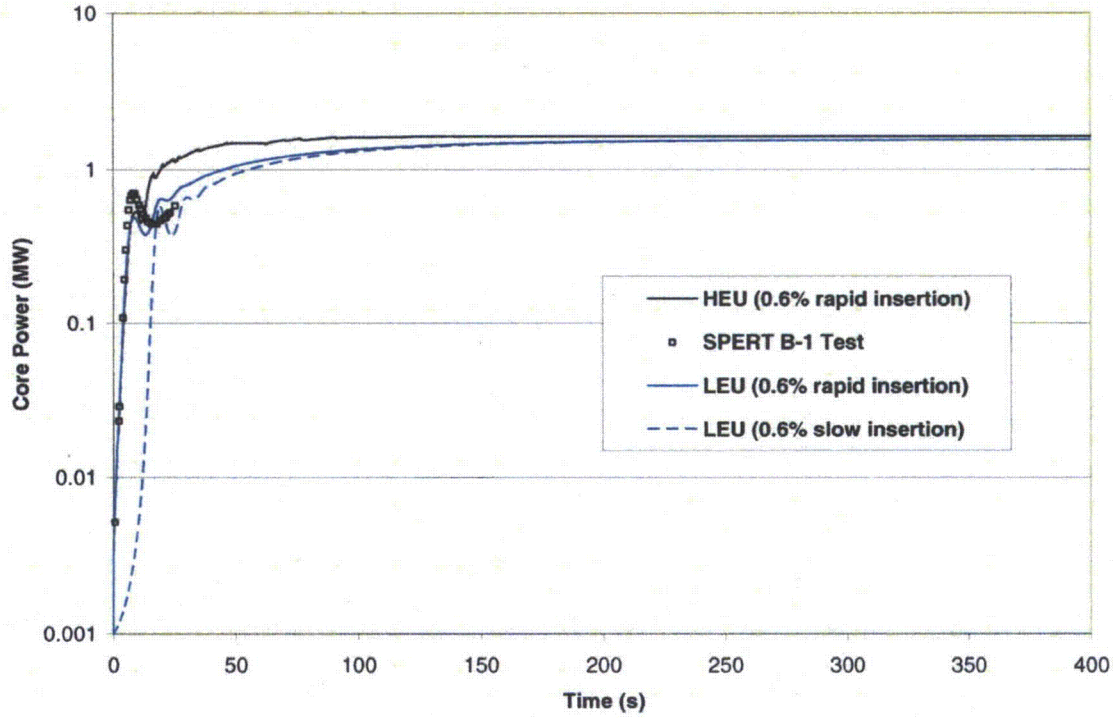

Figure Q39.1. PUR-1 Reactor Power in Reactivity Insertion Accidents without Scram.

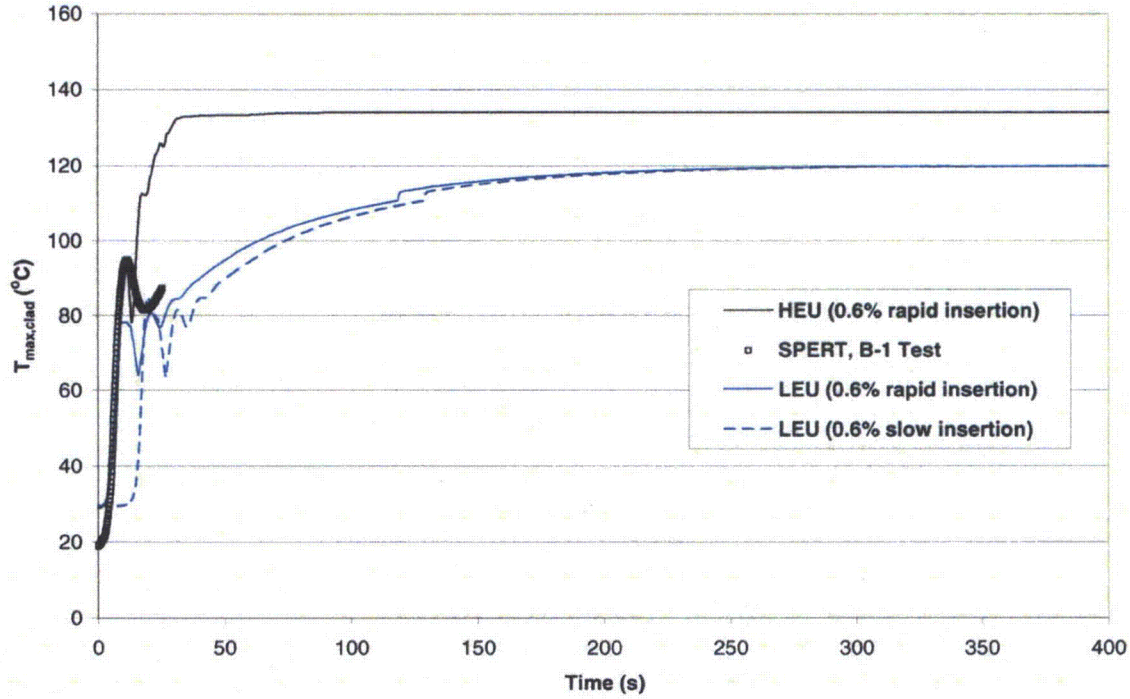

Figure Q39.2. PUR-1 Peak Clad Temperature in Reactivity Insertion Accidents without Scram.

Accidents in the New LEU Core Design

The redesign of the LEU assembly can, and changes to the orientation and number of fuel plates in the LEU core model, have a small effect on the calculated reactivity coefficients and kinetic parameters. The recalculated values are summarized in Table Q20E-1 in response to Question 20.E. It is noted that the calculated water void (or water density) coefficient and fuel temperature coefficient are slightly more negative for the new LEU core design; this change will lessen the impact of the reactivity insertion accidents. Also, the water temperature coefficient and the effective delayed neutron fraction are slightly smaller for the new LEU core design. All changes to the calculated reactivity coefficients and kinetics parameters are within 1-σ of the statistical uncertainty of the MCNP results.

The changes to the reactivity coefficients are small. Furthermore, the analyses for the accidents with reactor scram showed less than 1 °C change in the clad temperature. Therefore, it is not necessary to repeat these accident calculations using the new feedback coefficients because the changes will not affect the safety of the facility.

Larger temperature increases are observed for the rapid and slow insertion of 0.6% Δk/k without scram. The analysis for the old LEU core design found that in these accidents, the PUR-1 reaches a steady-state condition where the reactivity inserted in the accident is balanced by the water temperature, water void, and fuel temperature feedbacks. In the steady-state condition (see results at 400 seconds in Figures Q39-1 and Q39-2), the negative reactivity feedbacks are as follows.

$\Delta\rho_{\text{water temperature}}$	=	-0.32% Δk/k
$\Delta\rho_{\text{water void}}$	=	-0.24% Δk/k
$\Delta\rho_{\text{fuel temperature}}$	=	-0.04% Δk/k
$\Delta\rho_{\text{Feedback, Total}}$	=	**-0.60% Δk/k**

Thus, the net reactivity in the core is 0.0 (0.6% Δk/k inserted by the accident; -0.6% Δk/k in reactivity feedbacks).

In this case, the water temperature feedback effect accounts for roughly half of the total reactivity feedback, while the remainder of the feedback effect is due to the decrease in the coolant density between the fuel plates and the increased temperature of the fuel. It is noted above that the water temperature feedback may be smaller in the new LEU core design, while the water void and fuel temperature feedbacks are stronger. Consequently, the weaker water temperature feedback coefficient in the new LEU core design will be compensated for by the stronger feedback coefficients for the water void and fuel temperature. As such, the new LEU core design will have very little impact on the results of the 0.6% Δk/k insertion accident without scram already analyzed for the old LEU core design. Therefore, it is not necessary to repeat this accident analysis.

Question 40

40. Section 13.4. Was reactivity introduced by the flooding or voiding of an experiment located in the irradiation facility considered? Is that a viable accident (it was evaluated in NUREG-1283 for the 1988 license renewal)? If not, please explain. If so, please provide an analysis.

Response:

This was not considered as a viable accident. However, calculations show that the sudden flooding of an irradiation facility (F6) with water when it is initially full of air results in a reactivity addition of 0.175% Δk/k, which is lower than the other accidents analyzed. Voiding of a flooded irradiation facility results in a negative reactivity insertion (see Q6).

The flooding of the 12.7 cm drop tube when placed in position G6 as described in the SER NUREG-1283 resulted in a reactivity insertion of 0.246% Δk/k, which is below the insertion resulting from the failure of a

moveable experiment examined in the conversion proposal. This accident scenario is within the envelope of the examined accident cases, and therefore was not analyzed.

Question 41

41. TS 4.4.d. There are proposed modifications to TS 4.4.d to change fuel inspection requirements from periodic inspection of representative fuel plates to representative fuel assemblies.

A. What will the fuel assembly inspection consist of and how will this inspection help ensure fuel cladding integrity?

B. TS 4.3.d requires monthly analysis of primary coolant. Is this analysis used as an indicator of fuel cladding integrity? If not, can the analysis be adapted for such purpose?

Response:

A. Representative fuel assembly inspection will still be performed annually, without disassembly of the element. Visual inspection will be performed on the representative sample elements for signs of degradation such as corrosion, channel blockage, warped or bloated plates, Surveillance of the primary water system for contamination will complement this inspection process, please see part B of this question.

A representative HEU plate has been inspected for the 44 years that PUR-1 has been in service, and no degradation has been observed. The particular plate that has been examined had a blemish from the original machining of the plates. The blemish has not changed per visual inspection.

B. TS 4.3d requires monthly testing of the primary coolant for gross alpha and beta contamination in order to assure against undetected leaking fuel assemblies, as stated in the bases for the technical specification. Since the intention of the TS requirement is to detect fuel cladding failures, it is intended to be used for this purpose, and supports the deletion of the fuel element disassembly as part of the inspection. No further changes to this TS should be required.

Question 42

42. Section 14, TS 4.4. Section 4.1 footnotes and Table 4-1 indicate a change in the alloy of aluminum to 6061. Please clarify the following sentence: "No new alloys will be introduced into the reactor as a result of conversion from HEU to LEU fuel."

Response:

The aluminum alloy 6061 composes the reactor deck, and various hardware parts of the control rod assemblies (e.g. the spacer plates) and the assembly can hardware. These were included in the conversion proposal model, therefore the statement can be made that no new alloys will be introduced into the reactor as a result of the conversion.

REFERENCES

[1]. Idaho National Laboratory (INL), "Specification for Purdue University Standard and Control Fuel Elements—Assembled for the Purdue University Reactor," Document ID SPC-382, May 2006.

[2]. R. K. Shah and A. L. London, "Laminar Flow Forced Convection in Ducts," Supplement 1 to Advances in Heat Transfer, Academic Press, New York, page 199 (1978).

[3]. F. P. Incropera, D. P. Dewitt, T. L. Bergman, and A. S. Lavine, "Introduction to Heat Transfer," 5th. Edition, page 489 (see Table 8.1), John Wiley & Sons (2007).

APPENDIX 1: REVISED TO INCLUDE THE EFFECT OF TEMPERATURE DEPENDENCE OF VISCOSITY ON HOT CHANNEL FACTORS (Revised by MK 4/5/07)

1. Hot Channel Factors in the NATCON Code Version 1.0

The NATCON code version 1.0 [Ref. ANL/RERTR/TM-12] uses three hot channel factors (FQ, FW, FH). Using the source code and documentation, the factor FH used in NATCON is found to be the same as the factor FNUSLT used by E. E. Feldman. Table 1 shows the tolerances and uncertainties included in each of the six hot channel factors used by E. E. Feldman. The correspondence between the NATCON hot channel factors and E. E. Feldman's six hot channel factors is as follows.

Feldman's Hot Channel Factor		NCATCON Input Variable
System-wide Factors:		
FFLOW	a factor to account for the uncertainty in total reactor flow	FW (approximately)
FPOWER	a factor to account for the uncertainty in total reactor power	FQ
FNUSLT	a factor to account for the uncertainty in Nu number correlation	FH
Local Factors:		
FBULK	a hot channel factor for local bulk coolant temperature rise	FBULK (new input)
FFILM	a hot channel factor for local temperature rise across the coolant film	FFILM (new input)
FFLUX	a hot channel factor for local heat flux from cladding surface	FFLUX (new input)

2. Hot Channel Factors in the NATCON Code Version 2.0

Sections 2.1 and 2.2 develop, for laminar natural convection, two thermal-hydraulic relationships that are used in section 2.3 to obtain formulas for the hot channel factors from user-supplied manufacturing tolerances and measurement uncertainties. The results of section 2.3 are summarized here for convenience. The first three are local/random hot channel factors, and the last three are system-wide. An example of the use of these hot channel factors is given in section 4, with NATCON running instructions in section 3, and the new input description in section 5.

$$\text{FBULK} = 1 + \sqrt{\left[\left\{(1+u_1)^{\frac{1}{2+\alpha}}(1+u_2)^{\frac{1}{2+\alpha}}\left(\frac{1}{1-u_5}\right)^{\frac{3}{2+\alpha}} - 1\right\}\right]^2 + u_6^{\,2}}$$

FBULK is higher (conservative) if the temperature dependence of water viscosity is ignored.

$$\text{FFILM} = 1 + \sqrt{u_1^{\,2} + u_2^{\,2} + u_3^{\,2} + u_4^{\,2} + u_5^{\,2}}$$

$$\text{FFLUX} = 1 + \sqrt{u_1^{\,2} + u_2^{\,2} + u_3^{\,2} + u_4^{\,2}}$$

$FQ = 1 + u_7$
$FW = 1 + u_8$
$FH = 1 + u_9$

where

u_1 = Fractional uncertainty in neutronics calculation of power in a plate

u_2 = Fractional uncertainty in U-235 mass per plate = $\Delta m / M$

u_3 = Fractional uncertainty in local (at an axial position) fuel meat thickness
u_4 = Fractional uncertainty in U-235 local (at an axial position) homogeneity
u_5 = Fractional uncertainty in coolant channel thickness = $(t_{nc} - t_{hc}) / t_{nc}$
u_6 = Fractional uncertainty in flow distribution among channels
u_7 = Fractional uncertainty in reactor power measurement
u_8 = Fractional uncertainty in flow due to uncertainty in friction factor
u_9 = Fractional uncertainty in convective heat transfer coefficient, or in the Nu number correlation
M = Nominal mass of U-235 per plate, gram
Δm = Tolerance allowed in U-235 mass per plate, gram

The code obtains, for an input nominal reactor power CPWR, a thermal-hydraulic solution using the three systematic hot channel factors FW, FQ and FH. If the user-input reactor power is zero, then the code itself chooses the nominal power from a series of power levels (10 kW, 100 kW, 200 kW, and so on increasing in steps of 100 kW). This thermal-hydraulic calculation is done for a hot plate power of CPWR*FQ*(Radial power peaking factor RPEAK)/(Total number of fuel plates in standard and control assemblies). Also, the frictional resistance to flow is multiplied by FW^2, and the convective heat transfer coefficient found for laminar flow in a rectangular channel is divided by FH. The random hot channel factors FBULK, FFILM and FFLUX are not used in this solution.

Having obtained the above solution, the random hot channel factors FBULK, FFILM and FFLUX are applied to the temperatures obtained, using the following equations. The temperatures calculated with all six hot channel factors are printed after the above solution. The onset of nucleate boiling ratio, ONBR, is computed using the temperatures with all six hot channel factors applied (using the equation below). If the user-input nominal power is zero, then the last nominal power for which the code prints a solution is that at which the ONBR is 1.0.

$$T_{i,6hcf} = T_0 + (T_i - T_0)*FBULK$$

$$T_{wall,i,6hcf} = T_{i,6hcf} + (T_{wall,i} - T_i)*FFILM$$

$$T_{max,i,6hcf} = T_{wall,i,6hcf} + (T_{max,i} - T_{wall,i})*FFLUX$$

$$ONBR = \frac{\left(T_{incp,i} - T_0\right)}{\left(T_{wall,i,6hcf} - T_0\right)}$$

where

T_0 = Bulk water temperature at the coolant channel inlet, i.e., the pool temperature, °C
T_i = Bulk water temperature in node i of the channel with only systematic hot channel factors applied, °C
$T_{wall,i}$ = Cladding surface temperature in node i with only systematic hot channel factors applied, °C
$T_{max,i}$ = Fuel meat centerline temperature in node i with only systematic hot channel factors applied, °C
$T_{i,6hcf}$ = Bulk water temperature in node i of the channel with all six hot channel factors, °C
$T_{wall,i,6hcf}$ = Cladding surface temperature in node i with all six hot channel factors, °C
$T_{max,i,6hcf}$ = Fuel meat centerline temperature in node i with all six hot channel factors, C
$T_{incp,i}$ = Incipient boiling temperature in node i with only systematic hot channel factors applied, C

2.1 Flow Rate in a Coolant Channel versus Power of a Fuel Plate

NATCON is a laminar natural circulation code. The flow rate is calculated in the code by balancing the buoyancy pressure force to the laminar friction pressure drop. Following this concept, an analytical relationship is developed here (with some approximation) for the coolant flow rate in a single coolant

channel in terms of the power generated in a fuel plate and the channel geometrical dimensions. The analytical relationship is needed for obtaining hot channel factors.

The hot channel factor FW used in the code to account for the uncertainty in coolant flow rate is actually applied to the laminar friction factor in the code, that is, the laminar friction factor is multiplied by FW^2. It is not applied directly to the flow rate. The relationship developed here explains how this technique works.

ρ_1 , T_1 at channel outlet

|
|
|
| L = Channel height containing hot coolant (hotter than pool), m
| P = Power in a single fuel plate or the two half plates, W
| W=Upward flow rate in a single channel, kg/s
|
|

ρ_0 , T_0 at channel inlet

Schematic of what the code analyses, that is, a single rectangular coolant channel heated by a half of a fuel plate on each side (right and left sides).

The above schematic shows what the code analyses, that is, a single rectangular coolant channel heated by a half of a fuel plate on each side (right and left sides). See Fig. 1 for details. The buoyancy pressure force is caused by the decrease in water density due to heating in the channel. The temperature dependence of water density can be written as

$$\rho(T) = \rho_0 - \rho_0 \beta (T - T_0) \tag{1}$$

where

- T_1 = Bulk water temperature at channel outlet, C
- ΔT = $T_1 - T_0$ = Temperature rise in channel from inlet to outlet, C
- ρ_0 = Water density at channel inlet, i.e., the water density in the pool, kg/m^3
- β = Volumetric expansion coefficient of water, per C
- $\bar{\rho}$ = Average coolant density in the channel, kg/m^3
- L = Channel height that contains hotter coolant (hotter than pool), m. It is the sum of heat generating length of fuel plate, non-heat generating fuel plate length at top, and the assembly duct length above the top of fuel plate
- g = Acceleration due to gravity, 9.8 m/s^2

The buoyancy pressure force is given by

$$\text{Buoyancy}\,\Delta p = (\rho_0 - \bar{\rho})\, g\, L \tag{2}$$

The average coolant density $\bar{\rho}$ is given by

$$\bar{\rho} = 0.5(\rho_0 + \rho_1) = \rho_0 - 0.5\rho_0 \beta (T_1 - T_0) = \rho_0 - 0.5\rho_0 \beta \Delta T \tag{3}$$

$$\text{Buoyancy}\,\Delta p = 0.5\rho_0 \beta \Delta T\, g\, L \tag{4}$$

The coolant temperature rise ΔT can be written in terms of the input power P generated in a fuel plate, as shown by Eq. (5) below, and then the buoyancy Δp of Eq. (4) can be written in terms of the input power P, as shown by Eq. (6).

$$\Delta T = P/(W C_p) \tag{5}$$

$$\text{Buoyancy } \Delta p = \frac{\rho_0 \beta g L P}{2 W C_p} \tag{6}$$

Ignoring the minor losses at channel inlet and outlet, the laminar frictional pressure drop in the channel is written below as Eq. (9) after using the laminar friction factor given by Eq. (7), and after replacing the coolant velocity by mass flow rate using Eq. (8). The parameter C in Eq. (7) is a constant for a given channel cross section, but it depends upon the channel cross section aspect ratio width/thickness, and varies from 57 for aspect ratio 1.0 (square channel) to 96 for an infinite aspect ratio (infinitely wide channel).

$$f = C / R_e \tag{7}$$

$$W = \bar{\rho} A V \tag{8}$$

$$\text{Frictional } \Delta p = \frac{\bar{\rho} f L_C V^2}{2D} = \frac{C \bar{\mu} L_C W}{2 \bar{\rho} A D^2} \tag{9}$$

where

f	= Moody friction factor for laminar flow in the channel
R_e	= Reynolds number in the channel = $\bar{\rho}$ VD/ $\bar{\mu}$
A	= Flow area of the channel cross section, m^2
D	= Equivalent hydraulic diameter of the channel cross section, m
L_C	= Total coolant channel length causing frictional pressure drop, m.
V	= Coolant velocity averaged over the channel cross section, m/s
W	= Coolant mass flow rate in the channel, kg/s
$\bar{\mu}$	= Average coolant dynamic viscosity in the channel, N-s/m^2
μ (T)	= Temperature-dependent dynamic viscosity of water, N-s/m^2
μ_0	= μ (T_0) = Coolant dynamic viscosity at the channel inlet temperature T_0

For the PUR-1 reactor, the temperature dependence of the dynamic viscosity of water over the temperature range 27 °C ≤ T ≤ 50 °C can be approximated as follows.

$$\mu(T) = \mu(T_0)\,(1+T-T_0)^{-\alpha} \tag{10}$$

where $\alpha = 0.12$, $T_0 = 27$ °C, $\mu(T_0) = 0.875 \times 10^{-3}$ N-s/m^2

The average coolant dynamic viscosity $\bar{\mu}$ used in Eq. (9) can be set equal to the viscosity at the average coolant temperature (T_0 + 0.5ΔT) in the channel. Putting this temperature in Eq. (10), the average viscosity $\bar{\mu}$ is found to be

$$\bar{\mu} = \mu(T_0)\,(1+0.5\Delta T)^{-\alpha}. \tag{11}$$

Equation (11) indicates that the average viscosity $\bar{\mu}$ can be set equal to μ (T_0) if ΔT is just a few °C (this is the case for the PUR-1 reactor at the operating power of 1 kW). If ΔT is greater than a few °C, i.e., 1 << 0.5ΔT (this is the case for the PUR-1 reactor at an ONB power of about 100 kW), then Eq. (11) simplifies to the following.

$$\bar{\mu} = \mu(T_0)\,(0.5\Delta T)^{-\alpha} \qquad \text{if } \Delta T >> 2\ °C \tag{12a}$$

$$\bar{\mu} = \mu(T_0) \qquad \text{if } \Delta T << 2\ °C \tag{12b}$$

Substituting Eq. (12a) into Eq. (9), the frictional Δp becomes

$$\text{Frictional } \Delta p = \frac{C\mu_0(\Delta T)^{-\alpha} L_C W}{2^{1-\alpha}\bar{\rho} A D^2} = \frac{C\mu_0 L_C W}{2^{1-\alpha}\bar{\rho} A D^2}\left(\frac{WC_P}{P}\right)^{\alpha} \tag{13}$$

Equating the frictional Δp of Eq. (13) to the buoyancy Δp of Eq. (6) to find the steady-state coolant flow rate W in the channel, one obtains Eq. (14) below. Equation (14) can be rewritten as Eq. (15).

$$\frac{\rho_0 \beta g L P}{2WC_P} = \frac{C\mu_0 L_C W}{2^{1-\alpha}\bar{\rho} A D^2}\left(\frac{WC_P}{P}\right)^{\alpha} \tag{14}$$

$$W^{2+\alpha} = \frac{\rho_0 \bar{\rho} A D^2 \beta g L P^{1+\alpha}}{2^{\alpha} C \mu_0 L_C C_P^{1+\alpha}} \tag{15}$$

Equation (15) relates the fuel plate power to the channel flow rate in natural circulation. It is used to find the dependence of the flow rate on the parameter C in the laminar friction factor (at constant power). All parameters in this equation are constant ($\bar{\rho}$ is also practically constant) except the parameter C in the laminar friction factor. Based on Eq.(15), the relationship between the flow rate W and the parameter C is given by Eq. (16) below.

$$W \propto \left(\frac{1}{C}\right)^{\frac{1}{2+\alpha}} \tag{16}$$

Equation (16) shows that the friction factor parameter C is multiplied by a factor $(FW)^2$, the coolant flow rate W will be reduced by the factor $(FW)^{\frac{2}{2+\alpha}}$. This has been verified by actually running the NATCON code for the PUR-1 reactor. Since α is small (α = 0.12 for the PUR-1 reactor), 2/(2+α) is nearly 1.0, and the flow rate W is reduced approximately by the factor FW.

2.2 Bulk Coolant Temperature Rise versus Power of a Fuel Plate

Equation (5) expresses, for laminar natural circulation, the bulk coolant temperature rise in terms of fuel plate power, coolant flow rate and specific heat. Putting the value of flow rate obtained in Eq. (15) into Eq. (5), the bulk coolant temperature rise is given by Eq. (17) below, purely in terms of power and the geometrical dimensions of the channel. The right hand side of Eq. (17) is rearranged into two factors in Eq. (18), such that the second factor is sensitive to power and channel geometrical dimensions that

usually have manufacturing tolerances and measurement uncertainties, and the first factor is insensitive to power and channel geometrical dimensions.

$$\Delta T = \left(\frac{2^{\alpha} C \mu_0 L_C P}{C_P \rho_0 \bar{\rho} A D^2 \beta g L} \right)^{\frac{1}{2+\alpha}} \tag{17}$$

$$\Delta T = \left(\frac{2^{\alpha} C \mu_0 L_C}{C_P \rho_0 \bar{\rho} \beta g L} \right)^{\frac{1}{2+\alpha}} \left(\frac{P}{A D^2} \right)^{\frac{1}{2+\alpha}} \tag{18}$$

The nominal flow area and hydraulic diameter of a rectangular coolant channel are given by

$$A = t_{nc}\, w_{nc} \tag{19}$$

$$P_w = 2\,(t_{nc} + w_{nc}) \tag{20}$$

$$D = 4\,A/\,P_w = 2\,t_{nc}\, w_{nc} / (t_{nc} + w_{nc}) \tag{21}$$

where

t_c	= Channel thickness (spacing between fuel plates), m
t_{nc}	= Nominal channel thickness (spacing between fuel plates), m
t_{hc}	= Minimum channel thickness in hot channel (spacing between fuel plates), m
w_c	= Channel width, assumed not to change from its nominal value, m
P_w	= Wetted perimeter of the nominal channel, m
P_{nc}	= Power generated in a fuel plate, without applying manufacturing tolerances, W
P_{hc}	= Power generated in a fuel plate, after applying manufacturing tolerances, W

Because the channel thickness t_c is much smaller than the channel width w_c in most experimental reactors, Eq. (21) reduces to

$$D \approx 2\,t_c \tag{22}$$

Using the channel area and hydraulic diameter given by Eqs. (19) and (22) into Eq. (18), the bulk coolant temperature rise can be written in terms of power, channel thickness, and channel width. This is the desired relationship for use in finding hot channel factors.

$$\Delta T = \left(\frac{2^{\alpha} C \mu_0 L_C}{C_P \rho_0 \bar{\rho} \beta g L} \right)^{\frac{1}{2+\alpha}} \left(\frac{P}{4 w_c t_c^{\,3}} \right)^{\frac{1}{2+\alpha}} \tag{23}$$

2.3 Formulas for Hot Channel Factors

For use in the NATCON version 2.0, six hot channel factors (three global/systemic and three local/random) are obtained from 9 manufacturing tolerances and measurement uncertainties u_1, u_2,.... u_9 that are defined below. These are fractional uncertainties rather than percent. Of these nine uncertainties, those affecting a particular hot channel factor are indicated in Table 1. The systemic hot channel factors are given by Eqs. (24) through (26), and the random hot channel factors are given by Eqs. (27) through (29). A utility Fortran computer program NATCON_HCF and a Microsoft spreadsheet

NATCON.HotChanFactors.xls have also been developed to compute the hot channel factors using these formulas.

$$FQ = 1 + u_7 \tag{24}$$

$$FW = 1 + u_8 \tag{25}$$

$$FH = 1 + u_9 \tag{26}$$

The ratio of the power generated in hot plate to its nominal power, caused by the uncertainties in neutronics-computed power and in U-235 mass per plate, can be written as

$$\frac{P_{hc}}{P_{nc}} = (1+u_1)(1+u_2) \tag{27}$$

The ratio of bulk coolant temperature rise in hot channel to the temperature rise in the nominal channel, caused by the uncertainties in neutronics-computed power, U-235 mass per plate, and channel thickness, is obtained from Eq. (23). Only the quantity in the second parentheses is important here because the quantity in the first parentheses is insensitive to these uncertainties.

$$\frac{\Delta T_{hc}}{\Delta T_{nc}} = \left(\frac{P_{hc}}{P_{nc}}\right)^{\frac{1}{2+\alpha}} \left(\frac{t_{nc}}{t_{hc}}\right)^{\frac{3}{2+\alpha}} = (1+u_1)^{\frac{1}{2+\alpha}} (1+u_2)^{\frac{1}{2+\alpha}} \left(\frac{1}{1-u_5}\right)^{\frac{3}{2+\alpha}} \tag{28}$$

The uncertainty in flow distribution is assumed to reduce the channel flow to (1- u_6) times the flow without this uncertainty, and therefore the bulk coolant temperature rise is increased by the factor (1+ u_6). This uncertainty in bulk coolant temperature rise is statistically combined with that given by Eq. (28) to obtain the following formula for the hot channel factor FBULK for input to the NATCON version 2.0.

$$FBULK = 1 + \sqrt{\left[\left\{(1+u_1)^{\frac{1}{2+\alpha}} (1+u_2)^{\frac{1}{2+\alpha}} \left(\frac{1}{1-u_5}\right)^{\frac{3}{2+\alpha}} - 1\right\}\right]^2 + u_6^{\,2}} \tag{29}$$

The temperature drop across coolant film on the cladding surface at an axial location is given by Eq. (30). Here the heat flux q" (W/m^2) on the cladding surface is replaced by t_f q'''/2 in terms of the volumetric power density q''' (W/m^3) in the fuel meat.

$$\Delta T_{film} = \frac{q''}{h} = \frac{t_f\, q'''}{2h} \tag{30}$$

The convective heat transfer coefficient h (W/m^2-C) is given by Eq. (31). Here the laminar Nusselt number N_u is independent of flow rate, and varies only slowly with the aspect ratio (width/thickness) of coolant channel. The main variation of the heat transfer coefficient with channel thickness is due to the denominator of Eq. (31). The numerator of Eq. (31) is considered to be constant.

$$h = \frac{N_u K_{cool}}{D} = \frac{N_u K_{cool}}{2\, t_c} \tag{31}$$

Using Eq. (31) for the heat transfer coefficient, the temperature drop across coolant film can be written as Eq. (32).

$$\Delta T_{film} = \frac{q'''\ t_f\ t_c}{N_u\ K_{cool}} \tag{32}$$

Equation (32) states that ΔT_{film} is directly proportional to the fuel meat thickness (having uncertainty u_3), the channel thickness (having uncertainty u_5), and the power density in meat. The uncertainty in power density is caused by three uncertainties, that is, u_1, u_2 and u_4. Statistically combining these five uncertainties gives the following formula for the hot channel factor FFILM for input to the NATCON version 2.0.

$$\text{FFILM} = 1 + \sqrt{u_1^2 + u_2^2 + u_3^2 + u_4^2 + u_5^2} \tag{33}$$

The uncertainty in the heat flux at the cladding surface is included in the hot channel factor FFILM given by Eq. (33). A hot channel factor FFLUX for the heat flux alone can be found from Eq. (34) for heat flux in terms of the power density q''' in the fuel meat and the thickness of the meat. The fractional uncertainty in heat flux is the sum of fractional uncertainties in power density and meat thickness, as given by Eq. (35).

$$q'' = \frac{q'''t_f}{2} \tag{34}$$

$$\frac{\delta q''}{q''} = \frac{\delta q'''}{q'''} + \frac{\delta t_{fuel}}{t_{fuel}} \tag{35}$$

In Eq. (35), the uncertainty in power density is caused by three uncertainties, that is, u_1, u_2 and u_4. The uncertainty in the meat thickness is given by u_3. Statistically combining these four uncertainties gives the following formula for the hot channel factor FFLUX for input to the NATCON version 2.0.

$$\text{FFLUX} = 1 + \sqrt{u_1^2 + u_2^2 + u_3^2 + u_4^2} \tag{36}$$

The uncertainty in the temperature drop ΔT_{metal} from fuel meat centerline to cladding surface is not important in the case of the PUR-1 reactor because ΔT_{metal} is very small compared to ΔT_{film}. For example, ΔT_{metal} is 0.05 °C and ΔT_{film} is 34.5 °C at 100 kW without any hot channel factors.

Table 1. Uncertainties Included in the Six Hot Channel Factors Used in NATCON Version 2.0 (X implies that an uncertainty affects a hot channel factor)

	Uncertainty Fraction	FQ	FW	FH	FBULK	FFILM	FFLUX
Local or random uncertainties							
1	Neutronics calculation of power in a plate, u_1				X	X	X
2	U-235 mass per plate, u_2				X	X	X
3	Local fuel meat thickness, u_3					X	X
4	U-235 axial homogeneity, u_4					X	X
5	Coolant channel thickness, u_5				X	X	
6	Flow distribution among channels, u_6				X	X	
System-wide uncertainties							
7	Reactor power measurement uncertainty, u_7	X					
8	Flow uncertainty due to uncertainty in friction factor, u_8		X				
9	Heat transfer coefficient uncertainty due to uncertainty in Nu number correlation, u_9			X			

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FUEL SPECS

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Idaho National Laboratory

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1		All	Revised to add Program Anneal requirements and update Drawing Titles

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APPENDIX A 38

APPENDIX B 40

FIGURES

1. Purdue University Fuel Plate Sampling Procedures For Destructive Tests 38

1. SUMMARY

1.1 General

This *specification* (see def.) defines the materials, components, testing, inspection, certain processes, *quality control* (see def.) requirements and acceptance criteria for the fabrication of standard and control *fuel elements* (see def.) and fuel element containers for the Purdue University Reactor at Purdue University at West Lafayette, Indiana.

2. APPLICABLE CODES, PROCEDURES, AND REFERENCES

2.1 Standards, Specifications, Drawings and Attachments

The applicable portions of the following documents as defined herein, form a part of this specification. Where there is a conflict between the documents cited and the latest revision thereof, the *supplier* (see def.) shall notify the *purchaser* (see def.) of the conflict and use the latest revision in effect at the signing of the contract, unless otherwise directed by the purchaser.

2.1.1 Specifications and Standards

National Codes and Standards

ASTM E 1742-00	Standard Practice for Radiograph Examination
ASTM E 1417-99	Standard Practice for Liquid Penetrant Examination
MIL-C-45662	Calibration System Requirements
RDT F6-2T	Welding of Reactor Core Components, Sections 1,2,3 and 6

American Society for Testing and Materials (ASTM)

ASTM B 209-00	Standard Specification for Aluminum and Aluminum-Alloy Sheet and Plate
ASTM B 210-04	Aluminum and Aluminum-Alloy Drawn Seamless Tubes
ASTM B 211-00	Standard Specification for Aluminum and Aluminum-Alloy Bar, Rod and Wire
ASTM B 214-99	Standard Test Method for Sieve Analysis of Granular Metal Powders

ASTM B 221-00	Standard Specification for Aluminum and Aluminum-Alloy Extruded Bars, Rods, Wires, Profiles and Tubes
ASTM B 241-02	Aluminum and Aluminum-Alloy Seamless Pipe and Seamless Extruded Tube
ASTM E 8-00	Methods of Tension Testing of Metallic Materials
ASTM E 29-93a (1999)	Recommended Practice for Indicating Which Places of Figures are to be Considered Significant in Specified Limiting Values
ASTM E 2016-99	Standard Specification for Industrial Woven Wire Cloth
American Welding Society (AWS)	
AWS A5.10-1995	Aluminum and Aluminum Alloy Welding Rod and Bare Electrodes
American National Standards Institute (ANSI)	
ANSI B46.1-1994	Surface Texture
ANSI Y14.5-1994	Dimensioning and Tolerancing for Engineering Drawings
American Society of Mechanical Engineers (ASME)	
ASME Section V – 2001, without addendum	Boiler and Pressure Vessel Code Section V
ASME NQA-1-1997	Quality Assurance Requirements for Nuclear Facility Applications
Idaho National Laboratory (INL)	
TR TR-11	Specification for Low Enriched Uranium Metal in Test Reactor Fuel
TR TR-14	Specification for Reactor Grade Low Enriched Uranium Silicide Fuel Powder
IN-F-4-TRA	Specification for Aluminum Powder for Matrix Material in Test Reactor Fuel
STD 7022A	Cleanliness Acceptance Levels for Nuclear or Non-Nuclear Service Components

American Society for Nondestructive Test (ASNT)

SNT-TC-1A (1996 or later)	American Society For Nondestructive Testing (ASNT) Recommended Practice

2.1.2 Publications (Idaho National Laboratory)

Appendix A	Requirements for Radiographing Purdue University Fuel Plates
Appendix B	Welding Requirements and Qualification for Purdue University Fuel Elements

2.1.3 Drawings (INL)

635454	Purdue University Test Research and Training Reactor Graphite Reflector Assembly and Source Drive Assembly
635455	Purdue University Test Research and Training Reactor Standard Fuel, Partial, & Dummy Element Assemblies
635456	Purdue University Test Research and Training Reactor Control Fuel Element Assembly and Dummy Control Fuel Element Assembly
635457	Purdue University Test Research and Training Reactor Fission Chamber Fuel Element Assembly
635458	Purdue University Test Research and Training Reactor Standard Fuel Container Assembly
635459	Purdue University Test Research and Training Reactor Control Fuel Container Assembly
635460	Purdue University Test Research and Training Reactor Irradiation Facility Assembly
635461	Purdue University Test Research and Training Reactor Capsule Holder and Capsule Insert Assemblies and Details
635462	Purdue University Test Research and Training Reactor Graphite Container Assembly, and Source Drive Container Assembly
635463	Purdue University Test Research and Training Reactor Fuel Plate Assembly and Dummy Fuel Plate Detail
635464	Purdue University Test Research and Training Reactor Container Tube Assembly and Details

635465	Purdue University Test Research and Training Reactor Graphite Block Detail And Graphite Capsule Holder Detail
635466	Purdue University Test Research and Training Reactor Miscellaneous Details
635467	Purdue University Test Research and Training Reactor Source Drive Nozzle Assembly and Source Drive Top
635468	Purdue University Test Research and Training Reactor Nozzle Preliminary Machined and Fission Chamber Top

3. TECHNICAL REQUIREMENTS

3.1 Production Qualification

The supplier is required to qualify the processes or portions of the process or be exempt from same by written approval of the purchaser. In *qualification* (see def.), only materials that comply with this specification shall be used. Qualification processes, equipment, and operator qualification/training programs shall be identical to those used during *production* (see def.). To qualify, the supplier must demonstrate, to the satisfaction of the purchaser, that the process is capable of producing a product, which satisfies all the requirements of the specifications. Assembly of production fuel elements shall not be initiated until: (1) all required data, to assure compliance with the qualification requirements, has been submitted to the purchaser; (2) data and records required by Section 6.3 have been submitted; and (3) written approval of qualification has been received by the supplier from the purchaser.

3.1.1 Fuel Plate Qualification

Fuel plate (see def.) qualification shall be satisfied by supplier production of a minimum of two consecutively produced plate *lots* (see def.), in lot quantities of 24 *plates* (see def.). The plates shall be made using low enriched uranium in the form of *Silicide* (see def.) powder, which have a yield of no less than 65% acceptable fuel plates meeting all applicable requirements of this specification. The supplier may combine the results of two consecutive lots into a production run in determining the 65% yield requirement provided that there have been no changes in the *manufacturing* (see def.) *procedure* (see def.) between lots which would require *requalification* (see def.) in accordance with Section 3.1.3.

In the event that fuel plate qualification has been performed by the supplier meeting all the requirements of this specification during the past twelve months, and qualified operators are performing the fabrication, fuel plate qualification requirements listed above will be waived.

Fuel plates made in *development* (see def.) (prior to and including qualification runs that fail to meet the 65% yield requirements) will not be used in fabricating production fuel elements without prior approval of the purchaser.

3.1.2 Fuel Element Qualification:

The supplier shall fabricate 1 *dummy standard fuel element assemblies* (see def.) which shall meet the requirements of this specification.

3.1.3 Requalification:

The supplier shall notify the purchaser of any proposed process change. A changed process may not be used in production until the supplier has met all the requirements of Section 3.1.3, submits the results and data of the requalification effort, and receives written approval from the purchaser.

Requalification for any fuel plate or fuel element attribute to the requirements of the specification will be required when the processes, materials, fuel loadings, equipment or equipment operators (welding and rolling) which have been previously qualified are changed, unless the supplier can demonstrate to the satisfaction of the purchaser by engineering explanation or proof test that such changes will have no detrimental effect on the product.

Requalification for compacting, *pack* (see def.) assembly, and rolling mill operators can be less than qualification basis, since the procedure has already been established. Candidate operators who are not qualified for compacting operations, pack assembly operations, and hot/cold rolling mill operations must demonstrate their abilities in performing the individual operations they are assigned.

An operator must qualify by processing two lots of fuel plates with minimum lot size of 24, for the operation he is assigned to qualify, before performing any production operation independently. Each lot of fuel plates shall be processed through final inspection, with a minimum yield rate of 90% acceptable fuel plates required for the operator to be termed qualified.

NOTE: *Failure of an operator to qualify, because of fuel plate deviations, must be based on deviations related to the operation being qualified. The purchaser on a case-by-case basis will determine the quantities and sizes of requalification fuel plates selected to be destructively examined.*

3.1.4 Operator Qualification:

Operator qualification will be accomplished via an approved supplier internal qualification program for the following operations:

A. Arc melting

B. Compacting

C. Pack assembly

D. Hot rolling

E. Cold rolling

F. Final machining.

3.1.4.1 In addition to the operations specified above, the supplier shall also show evidence of the training and competency of those individuals who perform any of the following fuel element fabrication and inspection activities:

A. Powder sieving, weighing, and testing

B. Compact weighing, visual and dimensional inspection

C. Fuel plate/element and component cleaning

D. Fuel plate annealing operations

E. Dimensional inspection of plates, elements, and subcomponents

F. Metallographic sample preparation and inspection

G. Visual inspection of plates, elements, and subcomponents

H. Void volume inspection

I. Fluoroscope inspection of fuel plates

J. Radiography and inspection of fuel plate radiographs

K. Ultrasonic testing and interpretation.

The individuals performing these operations shall have specific requirements imposed on them that will demonstrate their knowledge and ability to perform their respective assignments. Documented evidence of the training of these individuals shall be maintained and shall be made available to the purchaser upon request.

3.2 Materials

The material requirements for the components comprising the fuel element are as specified on Drawings per Section 2.1.3 and requirements of this section.

3.2.1 Fuel Bearing Plates

3.2.1.1 Fuel Cores: The *fuel cores* (see def.) of the fuel plates shall be uranium silicide powder dispersed in aluminum alloy powder which meet the requirements of IN-F-4-TRA and TRTR-14, per Section 2.1.1 of this specification.

3.2.1.2 Frames and Covers: Aluminum for the frames and cover plates shall conform to ASTM B209, Alloy 6061-O. The aluminum plate stock used for frame and cover plates shall be certified by the supplier to contain less than 30 PPM boron, 80 PPM cadmium, and 80 PPM lithium.

The subcontractor shall furnish certified physical properties and chemical analyses of ingots or plates of the 6061 materials to INL.

3.2.2 Aluminum Weld Filler Metal:

All aluminum weld filler metal shall be ER4043 as required by Specification AWS A5.10-1995.

3.2.3 Dummy (Non-Fueled) Plate:

Dummy (non-fueled) plates (see def.) shall be fabricated from aluminum Type 6061-O, that meets the requirements of Section 3.2.1.2.

3.2.4 Material Requirements

All material used or contained in the product shall comply with all the requirements of this specification and Drawings per Section 2.1.3 unless exempted by written document by the purchaser.

3.3 Mechanical Requirements

3.3.1 Fuel Plate Requirements

3.3.1.1 Fabrication: The supplier shall furnish the details of his fuel plate rolling schedule and component cleaning process to the purchaser for approval prior to use in production per 6.3.1.

Compacting details shall include silicide – aluminum compacting pressure and compacting press dwell time.

After hot rolling, each fuel plate shall be blister annealed per Section 4.6.1 and then cold rolled to final thickness at room temperature. After cold rolling operation, the fuel plates shall be subjected to program annealing. The rolling schedule shall contain, at a minimum, the following:

A. Nominal plate reduction

B. Minimum number of hot roll passes

C. Nominal inter-pass reduction and target thickness

D. Hot rolling furnace temperature

E. Preheat time for all hot roll passes

F. Final hot roll plate thickness

G. Type and frequency of roll lubricant utilized

H. Nominal cold roll reduction

I. Final cold roll thickness

Fuel plate *cladding* (see def.) thickness required by Section 3.3.1.4 and fuel core homogeneity requirements of Section 4.4 are independent requirements that must be met.

3.3.1.2 Core Configuration: No fuel particles are allowed within the fuel free zones located at the ends of the plates as shown on Drawing 635463.

The nominally unfueled area of each fuel plate as defined by Drawing 635463 may contain random fuel particles defined as flaking and limited in size, location, and spacing per this Section, as determined by Section 4.5.

The presence of fuel particles detected between the maximum fuel core outline and fuel plate edges and ends is allowed provided they do not violate the following restrictions:

- One or more fuel particles, which fit in a rectangle whose area is not more than 4×10^{-4} in^2 is acceptable

AND

- The fuel particle(s) are no closer than 0.080 in. to any other particle edge to edge

AND

- No particle is closer to the plate edge or end than the major dimension of the particle.

Stray fuel particles (see def.) that violate the above requirements may be removed from fuel plate edges by filing, provided the following:

- The filed out area is no deeper into the edge of the plate than 0.050 in., no longer than 0.250 in.

AND

- Each filed area is at least 1.0 in. apart

Filing of fuel plate ends, for the removal of stray particles, is not allowed, unless previously approved by the purchaser.

3.3.1.3 Internal Defects and Bond Integrity: Metallurgical bond, as determined by Section 4.6 is required at interface areas of the finished fuel plates, specifically fuel core-to-clad and clad-to-frame. The presence of grain growth across the fuel

matrix-cladding interface and across the aluminum frame-cladding interface of at least 50% is required. Fuel core defects in excess of 0.06 in. in any dimension as determined by Section 4.7 are not allowed.

3.3.1.4 Cladding Thickness: During production, all plates will be subjected to UT min-clad inspection. The standard will be calibrated at the nominal 0.008-inch scan depth. The gage will then be adjusted to a 0.010 inch scanning depth and the fuel plates will be scanned at 0.010 inch. Fuel plate UT traces, which display min-clad indications at the 0.010-inch depth, shall be visually compared with the 0.008-inch Standard trace. Fuel plates for which the UT reports show a comparable density of indications, or worse, than the indications displayed on the standard UT report are unacceptable. Fuel plates, which fail the 0.010-inch UT scan, shall be rescanned at 0.008 inch. Only fuel plates which are acceptable when rescanned at 0.008 inch shall be submitted to the Purchaser and User for evaluation.

3.3.2 Non-fueled (dummy) plates:

The supplier shall use a cold rolling method to obtain plate thickness. Non-fueled (dummy) plates shall be subjected to program anneal.

3.3.3 Fuel Element Requirements

3.3.3.1 Welding: All welding operations, procedures and methods shall conform with provisions of Appendix B. All welding procedures, welders, or welding machine operators shall be qualified in accordance with Appendix B. The supplier will provide the purchaser *certification* (see def.) that such is the case prior to production welding.

3.4 Physical Properties

Fuel plates shall have a core of U_3Si_2 and aluminum and completed fuel plates and fuel elements shall have fuel loadings per Sections 3.4.1.2, and 3.4.2.1.

3.4.1 Fuel Plate Requirements

3.4.1.1 Fuel Core: The fuel core shall consist of 19.75 ± 0.2 weight % enriched uranium silicide powder dispersed in aluminum powder. The uranium silicide powder shall be -100, +325

U.S. standard mesh particles. However, a blend may contain up to 35 weight percent of -325 U.S. standard mesh particles. Any powder particles greater than 100 mesh particles shall be reground such that they will go thru the 100 mesh sieve. The fuel core shall be fabricated according to standard powder-metallurgical and roll-bonding techniques. The supplier shall provide to the purchaser, a written procedure for pack assembly and the initial rolling step which describes the method used to prevent excessive oxidation that causes non-bond of fuel core to the cladding.

3.4.1.2 Fuel Loading: By using the approved supplier's method of assigning U-235 content, per a detailed description as to the weighing procedure by which the supplier proposes to assign fuel plate U-235 content. Each fuel plate shall contain 12.5 ± 0.35 grams U-235. The weight of each core shall be measured and recorded to within 0.01 gram U-235 based upon weight of the final compact and chemical and isotopic analysis of the constituents.

3.4.1.3 Fuel Homogeneity: Fuel homogeneity requirements are located in section 4.4.

3.4.1.4 Void Volume: In the qualification process, all fuel plates shall be inspected for void volume using the method described in Section 4.2. The percent voids in the fuel cores of all fuel plates shall be determined by the inspection procedure developed by the supplier. The percent voids in the fuel cores shall be at least 3.0% and not more than 11.0%.

3.4.2 Fuel Element Requirements

3.4.2.1 Fuel Loading: Assigned fuel loading for each fuel element shall be 175.00 ± 4.90 grams of U-235. Each Control Fuel Element shall contain 100 ± 2.80 grams of U-235. Control limits for the method used to measure this weight are established at the 95% confidence level for a significant population of measurements of a particular standard. The U-235 enrichment shall be 19.75 ± 0.2 weight % of total uranium per specification TRTR-11.

3.5 Surface Condition

Fuel plates and completed fuel elements must comply with the surface condition requirements of Section 3.5.1, 3.5.2, and 3.5.3 and drawings of Section 2.1.3, per ANSI B46.1. Sanding, or any other finishing procedure that will smear the aluminum surface, will not be allowed on fuel plates unless approved by the purchaser.

3.5.1 Surface Defects

3.5.1.1 Compliance with surface finish and defect requirements shall be established by 100% visual inspection of all fuel plates and elements. The surface of the finished fuel plates shall be smooth and free of gouges, scratches, pits, or removal of metal in excess of 0.005 inch in depth. Dents in the fuel plate shall not exceed 0.012 inch in depth or 0.25 inch in diameter. If there is evidence of dogboning in the plates, surface defects in the *dogbone* (see def.) area shall not exceed 0.003 inch in depth. No degradation of the fuel plates beyond these limits shall be permitted.

3.5.1.2 Fuel Plates shall be free of stringiness, scabs, or cracks. Surface finish shall be as required by Drawing 635463. Compliance with requirements of this section shall be accomplished by visual inspection of all fuel plates and fuel elements.

3.5.1.3 Defects on fuel plate edges or ends are permissible provided they are evaluated and acceptable to the requirements of Paragraph 3.3.1.2.

3.5.1.4 Compliance with surface finish and defect requirements shall be established by 100% visual inspection of all fuel

element containers. Fuel element containers shall be free of surface defects such as pits, dents, or scratches in excess of 0.010 inch in depth and 0.12 inch in diameter or equivalent area.

3.5.2 Cleanliness:

The suppliers fabrication, assembly, and storage areas used for the production of Purdue University fuel elements and/or components shall conform to the requirements of *"controlled work area"* (see def.) as defined in Paragraph 1.3.6 of INL Standard 7022A. Cleanliness shall be in compliance with INL Standard 7022A, Paragraphs 1.1, 1.2.3, 3.1, 3.2-b, d, i, 3.3 - d, e, 4.1.3, 4.2, and 4.3. Freon shall not be used to clean fuel elements or components.

As determined by Section 4.10 of this specification, there shall be no foreign materials on the finished fuel plates or surfaces of the finished fuel elements. All oil, metal chips, turnings, dusts, abrasives and spatter, scale, and other particles shall be removed from the fuel surfaces by procedures which assure that the minimum cladding thickness has not been violated. All components shall be cleaned by a method approved by the purchaser.

3.5.3 Contamination:

The surfaces of each fuel plate shall be counted or smeared and counted for alpha-beta-gamma contamination. The alpha count shall be less than five dpm per 100 cm^2, and the beta-gamma count shall be less than 200 dpm per 100 cm^2.

Each fuel element shall be smeared and counted for radioactive contamination. The alpha count shall be less than five dpm per 100 cm^2, and the beta-gamma count shall be less than 200 dpm per 100 cm^2.

3.6 Marking

NOTE: *All fuel plates, fuel assemblies, and fuel element containers will be marked per this section.*

3.6.1 Fuel Plate Identification:

Each finished fuel plate shall be identified, as shown on Drawing 635463, by a combination of numbers and/or letters that will maintain positive identification relative to the complete traceability to

the supplier fabrication history, including the basic material lots, heat or metal, manufacturing cycle, and quality control phases. The identification number shall be stamped, etched or vibro-peened at the location specified by Drawing 635463. The depth of the identification characters shall not exceed 0.010 in.

3.6.2 Fuel Assembly Identification:

Each fuel assembly shall have an identifying number such as 07-XX (07 signifying year of fabrication). The number shall be placed on the container assembly as shown on Drawings 635455, 635456 and 635457. The identification shall be stamped or entered by a method approved by the purchaser, with two inch block characters not in excess of 0.010 inches in depth. Standard assemblies should be labeled: E2, F2, G2, H2, F3, H3, E4, F4, G4, H4, F5, H5. Control assemblies should be labeled: E3, G3, and E5. The fission chamber assembly should be labeled as G5. The source assembly shall be labeled as C3. The spare Standard Assemblies should be labeled: SP-1, SP-2, SP-3. The spare Control Assembly should be labeled as SP-4.

3.7 Storage

All fuel plates, fuel assemblies, and fuel element containers that have received final cleaning per Section 3.5.2 shall be protected in clean polyethylene containers or other containers approved by the purchaser while (1) awaiting final assembly, (2) being transferred into or being maintained in storage, or (3) being prepared for packaging or shipment. Any material exposed to contamination shall be reinspected to the requirements of Section 3.5.

3.8 Fuel Element Surface Treatment

If boehmite treatment is required during fuel element fabrication, the following shall apply. After fuel elements are assembled and inspected they shall be subjected to an environment that will cause an evenly distributed boehmite layer of 0.00006 to 0.0003 in. thickness (averaged over the surface using eddy current instrumentation) to form on all surfaces of the entire assembly. The treatment process shall be performed under controlled conditions, which shall require the supplier to maintain a record of the thermal history of the autoclave. The records shall include heat charts of recorded time and temperature. Documented evidence of the controls placed on the autoclave shall be maintained by the supplier.

3.8.1 After the boehmite process has been qualified, one fuel element from every 2nd autoclave run shall be inspected for boehmite thickness at three evenly spaced locations along the center of an outer fuel plate. The boehmite film thickness shall be measured over the plate surface using a calibrated eddy current probe. Film thickness measurement, as averaged over the plate surface, less than 0.00006 in. or more than 0.0003 in. thickness, is unacceptable.

3.8.2 Each fuel element shall have a corresponding aluminum plate coupon, made from fuel plate end crops, placed near the fuel element during the boehmite formation process. The aluminum plate coupon shall be subjected to the same environment as the fuel elements and each coupon measured for boehmite thickness.

3.8.3 Fuel elements and aluminum plate coupons subjected to the boehmite formation process must be carefully handled to preclude scratches, dents and gouges that would cause removal of boehmite.

3.9 Graphite Reflectors and Graphite Radiation Baskets

Graphite reflector assemblies (see def.) and *irradiation facility assemblies* (see def.) shall be fabricated as per requirements contained in this section and in drawings 635454, 635460, 635461, and 635465.

3.9.1 Material:

All materials used shall comply with all the requirements of this specification and applicable drawings.

3.9.2 Assembly:

The assembly of the graphite reflector assemblies and irradiation facility assemblies shall be as shown on the applicable drawings.

3.9.3 Welding:

The requirements of welding for the fuel elements by Appendix B are also required for the graphite reflector assemblies and irradiation facility assemblies.

3.9.4 Identification:

The graphite reflector assemblies shall have identifying numbers such as GR-X placed on the side of the assembly as shown drawing 635454.

The graphite reflector shall be labeled as follows: D1, D2, D3, D4, D5, E1, F1, G1, H1, I1, I2, I3, I4, and I5. The irradiation facility assemblies shall have identifying numbers such as IF-X placed on the side of the assembly as shown on drawing 635460. The irradiation facility assemblies shall be labeled as follows: D6, E6, F6, G6, H6, and I6. The identification shall be stamped or entered by a method approved by the purchaser, with two inch block not in excess of 0.010 inches in depth.

3.9.5 Dimensional Inspection:

Verification of all external dimensions of the graphite reflector assemblies and irradiation facility assemblies shall be by 100% inspection, in accordance with drawings 635454 and 635460. All dimensions of this specification shall apply at a temperature of 75°F ± 5°.

3.9.6 Surface Finish and Defects:

The graphite reflector assemblies and irradiation facility assemblies shall be free of surface defects such as pits, dents, scratches in excess of 0.010 inch deep and 0.12 inch diameter or equivalent area.

3.9.7 Storage:

All graphite reflector assemblies and irradiation facility assemblies shall have received final cleaning and shall be protected in clean polyethylene containers or other containers approved by the purchaser while (a) being transferred into storage, (b) being maintained in storage, or (c) being prepared for shipment or packaging.

4. QUALITY ASSURANCE

The supplier shall document, implement, and maintain a quality program in compliance with ASME NQA-1-1997.

The supplier shall permit the purchaser to conduct pre-award and continuing evaluation of the Quality Program.

Personnel performing NDE examinations, specifically radiographic, ultrasonic, liquid penetrant, and visual shall be certified to American Society for Nondestructive Testing (ASNT) Number SNT-TC-1A and certification documentation shall be made available to the purchaser.

Unless otherwise specified, the supplier shall be responsible for the performance of all tests and inspections required prior to submission to the purchaser of any fuel element for acceptance. Provided, however, that the performance of such tests and inspections is in addition to, and does not limit, the right of the purchaser to conduct such other tests and inspections as the purchaser deems necessary to assure that all fuel elements are in conformance with all requirements of this specification. Except as otherwise specified, the supplier may use for inspection purposes either his own or any commercial laboratory acceptable to the purchaser. Records of all tests and examinations shall be kept complete and available to the purchaser. All test and measuring equipment shall be calibrated per the requirements of Standard MIL-C-45662.

The following applies to specified limits for requirements on core density per Section 3.4.1.1 and 4.2 and all dimensional requirements of this specification. For purposes of determining conformance with this specification an observed value or a calculated value shall be rounded off to the nearest unit in the last right hand place of figures used in expressing the limiting value in accordance with the rounding-off method of the Recommended Practices for Designating Significant Places in Specified Limiting Values (ASTM E29).

The supplier shall prepare for his use and the purchaser's approval an integrated manufacturing and inspection test plan. The plan shall include all manufacturing operations, equipment and tooling used, inspection requirements and gaging used, and mandatory hold points established by the purchaser.

Any materials or fuel element components that are fabricated using equipment, personnel, or processes that are not in accordance with approvals as previously granted by the purchaser are subject to *rejection* (see def.). A report of any such incident must be submitted in accordance with Section 6.3.7.

Fuel element inspection for shipment or rejection will be made by the on-site purchaser's representative at the supplier's plant. Final fuel element acceptance will be made by the purchaser at the User's facility.

4.1 Materials

Compliance with the material requirements of Section 3.2 shall be established by supplier certification. A "Certification of Chemical Analysis" or a certified mill test report shall be supplied to the purchaser for each lot of material used in the fabrication of fuel elements. This certificate shall give the results of the chemical analysis for the material. All fuel element materials shall be traceable.

4.2. Core Density

The density of the fuel cores required in Section 3.4.1.3 shall be determined by the Archimedes principle. During qualification of the fuel plate core void density required by Section 3.4.1.3 shall be determined on all qualification fuel plates submitted. After the particular plate type has been qualified, 100% inspection for void density is not required for production lots of fuel plates. For production lots, three randomly selected fuel plates from each lot shall be inspected for void volume density. Should any one of these plates be discrepant, the entire lot must then be inspected for void volume density. If void density discrepancies appear regularly in the process, the purchaser may request 100% inspection.

The actual core volume shall be calculated by the following formula where: weight units are in grams and volumes in cubic centimeters.

$$V_c = V_p - \frac{W_p - W_c}{\rho_{AL}}$$

where:

V_c = immersion volume of fuel plate core

V_p = volume of fuel plate

ρ_{AL} = density of aluminum used for fuel plate cladding

= 2.715 gms/cc

W_p = weight of plate

W_c = deburred weight of fuel plate core compact

The theoretical core volume shall be calculated by the following formulas:

$$V_{ct} = \left(\frac{WU_3Si_2}{\rho U_3Si_2}\right) + \left(\frac{W_{Al}}{\rho_{Al}}\right)$$

where:

V_{ct} = calculated theoretical core volume

WU_3Si_2 = weight of U_3Si_2 powder in core

W_{Al} = weight of aluminum matrix powder in core

ρU_3Si_2 = density of U_3Si_2 powder as measured

ρ_{Al} = density of aluminum powder used for core matrix
= 2.710 gms/cc

The void percent in the core shall be calculated using the following formula:

$$V\% = \frac{V_{cth} - V_{ct}}{V_c} \times (100\%)$$

where:

$V\%$ = percent voids in the fuel plate core

4.3 Fuel Loading

Verification of the fuel loading as specified in Section 3.4.1.2 shall be in conformance to the supplier's procedure required in Section 6.3.1.

In order to determine compliance with the fuel density requirements of Section 4.4, the U-235 loading of the fuel plate, as determined in accordance with the procedures of Section 6.3.1, will be divided by the core volume (Vc) as calculated by the method described in the second paragraph of Section 4.2.

4.4 Fuel Homogeneity

Fuel core homogeneity requirements shall be complied with by a one-piece radiograph of all fuel plates from each fuel plate lot and evaluation of the radiograph by calibrated densitometer measurements. Purchaser approved density standards may be used by the supplier. Fuel plates and density standards shall be

exposed simultaneously. Fuel plate density variations shall be determined by comparison of fuel plate areas to corresponding areas of the standard.

All fuel plates shall be inspected for homogeneity. Homogeneity of the fuel plate core shall be determined by radiograph film density measurements with a densitometer having a 0.080 inch aperture.

When determining fuel core density from plate radiographs, the brighter the image on the radiograph, the more dense is the uranium and the lower the number indicated on the densitometer. The darker the image on the radiograph, the less dense is the uranium and the larger the number indicated on the densitometer. A +30% fuel core density and a +20% fuel core density is indicated by the densitometer readings in the suspect area being 30% or 20% lower than the average densitometer readings for all core locations. A −30% or a −20% fuel core density is indicated by the densitometer readings in the suspect area being 30% or 20% higher than the average densitometer readings for all fuel core locations.

Any one-half inch diameter or greater spot in the plate fuel core area, other than the dogbone area shall not be less in fuel density than −20% of the average fuel density for all fuel core locations. To determine the low density of a one-half inch diameter area, the film is maneuvered under the densitometer in the low-density area until the highest number possible is obtained on the densitometer. This number is recorded. Then four readings are taken one-fourth inch from this spot and symmetrical around it. The average of these five readings is compared to the average densitometer readings for all fuel core locations.

If density standards are used, the average densitometer readings of all fuel core locations will be replaced by the nominal density standard and comparisons will be determined between the suspect spot on the radiograph and the −30% and −20% standards. For the +30% and +20% homogeneity overload inspection compare the nominal density standard to the suspect area. In this case densitometer units from nominal of the fuel plate represent the following percentages: ≤0.15 = +30%; -0.10 = +20%. Fuel plates exceeding these limits are discrepant.

For rectangular shaped, suspected discrepant areas that are evaluated to the one-half inch criteria, orient the four symmetrical readings such that worst case readings will be taken.

Between the minimum and maximum permissible fuel core length boundary, fuel underload condition shall not be evaluated.

Any indication of un-alloyed uranium as determined by radiography shall be cause for rejection.

Any 0.080 inch diameter spot in the fuel plate dogbone area (area within one inch of each fuel core end) shall not be greater in fuel density than +30% of the average fuel density for all core locations. Any one-half inch diameter area in the dogbone area shall not be less in fuel density than -30% of the average fuel density for all fuel core locations. The actual dogbone shall not be more than one-half inch in the longitudinal direction.

Other than the dogbone areas near ends of fuel core, any one-half inch diameter area shall not be greater in fuel density than +20% of the average fuel density for all fuel core locations. To determine the high density of a one-half inch diameter area, the film is maneuvered under the densitometer in the high-density area until the lowest number possible is obtained on the densitometer. This number is recorded. Then four readings are taken one-fourth inch from this spot and symmetrically around it. The average of these five readings is compared to the average densitometer readings for all fuel core locations.

Unless otherwise specified, purchaser approval of all radiographs is required prior to assembly of fuel plates into elements.

4.5 Core Configuration

Each finish-cut flat fuel plate shall be radiographed in accordance with Appendix A and evaluated for compliance with Section 3.3.1.2.

Visual radiograph inspections will be performed without magnification on a light table having a light intensity of 450 to 600 ft-candles at the table surface and the area darkened to give a light range of 5 to 15 ft-candles 18 in. above the light table with radiograph film in place on the table.

4.6 Bond Integrity

4.6.1 Blister Anneal:

After the fuel plate has been hot rolled, it shall be heated to 900°F±13°F, held at that temperature for a period of 2 hours, -15 minutes, +30 minutes, removed from furnace, and allowed to air cool.

Any blisters, in the fuel core region larger than a 0.060 in. diameter or any blister in the frame region of the fuel plate larger than 0.120 in. diameter shall result in rejection of the associated fuel plate. A maximum of two blisters less than 0.060 in. diameter is allowed in the fuel core area, provided they are more than 0.250 in. apart. A maximum of two blisters in any of the four sides of the *picture frame* (see def.)(a maximum of eight) region smaller than 0.120 in. can be tolerated providing that no blister is

any closer to the plate edge or end or to another blister than the major dimension of the blister and no blister is closer to the plate edge or end than 0.050 inch. When there is question as to size or location of the blisters, the acceptance or rejection of the plate shall be determined in the ultrasonic inspection of Section 4.6.2.

4.6.2 Ultrasonic Scanning:

The finished fuel plate area shall be ultrasonically inspected in compliance with ASME Boiler and Pressure Vessel Code, Section V, Article 5, Paragraphs T-110, T-510, T-520, T-521, T-522-a, b, c, e, g, i, j, k, l, o, T-523, T-523-1, and T-534. Any indication of discontinuity in the fuel core region equivalent to that indicated by a 0.060 in. diameter standard or any indication of a discontinuity in the frame region of the fuel plate equivalent to that indicated by a 0.120 in. diameter standard shall result in rejection of the associated fuel plate. Acceptance criteria for number of blisters revealed by ultrasonic scanning are per Section 4.6.1. Any discontinuities, inside the fuel plate, other than blisters and for which acceptance criterion is not already stated, shall be described by the supplier and evaluated by the purchaser.

4.6.3 Metallographic Examination:

During qualification, one fuel plate per lot selected for qualification per Section 3.1.1 will be sectioned per Figure 1, polished and etched, and examined at 50× or above for bond and clad-core-clad dimensions per the requirements of Sections 3.3.1.3 and 3.3.1.4, and Drawing 635463, respectively.

If the fuel plate fails the metallographic examination for grain growth, voids, laminations, core cracking or separation, or foreign particles or materials, then randomly selected another plate in the lot for metallographic examination. If this plate fails the examination, reject the lot.

Fuel plates selected for destruction tests may be rejected fuel plates, providing the attribute to be tested for is not affected by the cause for rejection. Reject fuel plates so used must have purchaser approval before destruct tests are performed.

4.7 Internal Defects

Any internal defect in excess of the requirement of Section 3.3.1.3 in the fuel core, including voids, laminations, U_3Si_2 segregation, clumping, core cracking or

separation, or foreign particles or materials, which is identified by any measurement technique, including radiography per Section 4.4, ultrasonic scanning per Section 4.6.2, or metallography per Section 4.6.3, shall be cause for rejection of the fuel plate.

4.8 Surface Finish and Defects

Compliance with requirements of Section 3.5 shall be established by visual inspection of all fuel plates and fuel elements. Out-of-specification defects shall be measured for size and depth and reported to the purchaser.

4.9 Clad-Core-Clad Dimensions

Fuel Plate Qualification requirements of section 3.1.1 shall be established by ultrasonic techniques using the purchaser-supplied, min-clad inspection gage. All fuel plates will be subjected to ultrasonic min-clad inspection with the fuel core region scanned for each plate. Ultrasonic min-clad inspection shall be accomplished by calibration of the min-clad gage, using the ATR Standard (8E0777) scanned at the normal mode of 0.008 inches. The min-clad gage will then be adjusted and the fuel plates will be scanned at a depth of 0.010 inches. Ultrasonic Test (UT) traces showing fuel at the 0.010 inch depth will be compared to the 0.008 inch standard to determine plate acceptability. If the density of indications from fuel plate exceeds the ATR standard density of indications, the plate is rejectable.

NOTE: *The ATR standard is a small piece of an ATR fuel plate that has fuel particles near the surface. It is used on the UT min-clad machine to indicate min-clad indications, and compare the density of these indications to any indications noted from a fuel plate being inspected by UT.*

During the fuel plate qualification process, compliance with the requirements of Section 3.3.1.4 shall be established by destructive analysis of one fuel plate per lot in accordance with Figure 1.

After fuel plate qualification, all production plates shall be min-clad ultrasonic inspected at a depth of 0.010 inches. Those plates discrepant at 0.010 inches shall be rescanned at 0.008 inches. Plates which are acceptable when re-scanned at 0.008 inches shall be submitted on Information/Change Request (Form 540.33) to the purchaser.

4.10 Cleanliness

Fuel plate, fuel assembly, and fuel element container cleanliness requirements of Section 3.5.2 shall be established by visual inspection without magnification of all fuel plates, fuel assemblies, and fuel element containers.

4.11 Contamination

The surfaces of each fuel plate and fuel assembly shall be counted or smeared and counted for alpha-beta-gamma contamination and meet the requirements of Section 3.5.3.

4.12 Dimensional

It shall be the supplier's responsibility to assure that fabrication is performed in accordance with all dimensions delineated in the Drawings referenced in Section 2.1.3. Noncomplying design dimensions on fuel plates, fuel assemblies, and fuel element containers (actual measurements) shall be submitted to the purchaser for review and approval. Any discrepant component shall not be used in a fuel element assembly unless approved.

The supplier is to certify to compliance with the design dimensional requirements delineated in the Drawings referenced in Section 2.1.3.

All dimensions of finished fuel plates, fuel assemblies and fuel element containers apply at 75°F±5°F.

4.12.1 Final Dimensional Inspection.

Dimensions required by this specification and drawings of Section 2.1.3 shall be inspected using a purchaser approved sample plan and recorded on an inspection sheet with "in specification" dimensions recorded by check mark, "OK," or actual measurements and "out of specification" dimensions recorded as actual measurements.

4.13 Reactor Components and Spare Fuel Element Parts

Reactor components and spare fuel element parts not assembled into fuel element assemblies are required to be certified. The certification shall consist of material certification, fabrication verification, and supplier certificate of compliance to the specification and drawing requirements. The certification documents shall be submitted to the purchaser and user.

5. PACKAGING AND SHIPPING

Packaging and shipping of the fuel elements shall be performed using a Purchaser approved procedure in compliance with this section.

- The purchaser shall provide shipping containers to protect the fuel elements from damage during shipment and which conform to the applicable requirements of the Departments of Energy and Transportation, and other regulatory agencies having jurisdiction of the shipment of radioactive materials. Re-useable shipping containers will be returned to the Supplier by the User at the Purchaser's expense.
- The Supplier is responsible for loading the fuel elements into shipping containers in a sealed polyethylene sleeve in a cleaned dry condition and free of extraneous materials.
- The Supplier shall take necessary precautions during packaging to prevent damage to the fuel elements during shipment. Each container shall be provided with a tamper-proof seal. Loading and shipping documents for the container shall be prepared in accordance with the applicable regulatory requirements.
- The Supplier shall make arrangements for shipment to the User.

6. NOTES

6.1 Definitions

For the purpose of this specification, the following terms are identified:

Batch. The amount of silicide powder mixture which is handled as a unit or traceable to a common step.

Blended. To mix or mingle constituents of a batch.

Certification. The action of determining, verifying and attesting in writing (signed by a qualified party) to the qualifications of personnel and material.

Cladding. The aluminum covers bonded to the fuel core and the picture frame.

Control Fuel Element Assembly. An assembly consisting of the control fuel element container with eight fuel plates.

Controlled Work Area. A work area to which access of personnel, tools, and materials is limited and physically controlled. Temporary enclosures may be used where adjacent activities produce contamination which is detrimental to the job.

Development. A determination of processes, equipment, and parameters required to produce a product in compliance with this specification.

Dogbone Area. Thickening of the fuel core usually in the last 1/2 in. of the core, which may result in clad thinning in those areas.

Dummy Fuel Element Assembly. An assembly consisting of a fuel element container with unfueled simulated dummy fuel plates.

Dummy Fuel Plate. A non-fueled plate made entirely from the aluminum material specified in this document.

Edge Clad. The distance between the edge of the fuel core and the edge of the finished fuel plate, before any stray particles are removed, in the width direction as determined by radiography of a flat fuel plate.

Failure. A condition where the fabrication process appears to be out of control or a breakdown or damage to equipment creates excessive costs and/or schedule delays.

Fuel Compact. A quantity of uranium silicide powder and aluminum powder, cold compacted by pressing into a solid block for assembly into packs for hot roll and cold roll into fuel plates. The compacts are encased in frames and cover plates to form the pack.

Fuel Assembly. An assembly of fuel plates and hardware components. This includes both the standard and control fuel elements.

Fuel Core. The uranium-bearing region of each Fuel Plate.

Fuel Plate. The Fuel Core complete with aluminum frame and cladding.

Graphite Reflector Assemblies. A component consisting of a graphite container assembly with a graphite block inside.

In-Process Controls. Inspections and tests made during production to ensure that the manufacturing processes, equipment, and personnel are producing a product meeting specified requirements.

Irradiation Facility Assemblies. A component consisting of a round tube attached inside a graphite container assembly with graphite blocks filling the annulus between the tube and container. Inserted within the tube is the isotope capsule assemblies.

Lot. A group of pieces handled as a unit or material traceable to a common processing step.

Manufacture(ing). All fabrication, assembly, test, inspection and quality control processes. Fabrication is a synonym for Manufacture.

Pack. The fuel compact, picture frame, and cover plates, assembled together for hot rolling.

Picture Frame. The window shaped aluminum frame, which holds the fuel compact.

Plates. See Fuel Plates.

Procedure. The detailed description of the series of processes during manufacture and inspection, which follow a regular definite order (not to be construed as an outline).

Production. That phase of the program, following Qualification, during which the product is in Manufacture.

Purchaser. Idaho National Laboratory (INL).

Qualification. A demonstration that the Manufacturing process, equipment and personnel can produce a Product in compliance with this Specification.

Quality Control. The sampling plans, inspections, tests and records required and used during Production to assure that the Product is in compliance with this Specification.

Rejection. Materials, parts, components, or assembly products, which will not be accepted as fulfilling the contract requirements because of noncompliance with this Specification.

Requalification. A demonstration that a single or group of manufacturing processes, equipment and personnel can produce a product in compliance with this specification after the original qualification has been completed and becomes invalid.

Silicide. Uranium metal alloyed with silicon and fabricated per the requirements of Specification TRTR-14. The word "fuel" is a synonym for Silicide.

Specification. All parts and appendixes to this document, its references, drawings, and standards, as may be modified from time to time by contractual document.

Standard Fuel Element Assembly. An assembly consisting of the fuel element container with fourteen (14) fuel plates.

Stray Fuel Particles. Isolated fuel particles lying outside the maximum fuel core outline defined on Drawing 635463.

Supplier. The primary vendor selected by INL to manufacture the product.

User. Purdue University, at West Lafayette, Indiana.

6.2 Purchaser Tests

None

6.3 Submittals

The following data and records shall be supplied to the purchaser in the quantities stated. The purchaser's approval, prior to implementation, is required on those marked with an asterisk. All records and data shall be maintained by the supplier for the duration of the Purdue University fuel element contract.

The granting of approval by the purchaser of design, working drawings, specifications, requests, and other technical data submitted by the supplier under the provisions of the subcontract or specification shall not affect or relieve the supplier from such responsibility as the supplier has with respect to adequacy or correctness of the design, working drawings specifications, reports, and other technical data.

6.3.1 Preproduction:

Documents requiring approval must be submitted prior to production use. The number of copies shall be as specified by the Vendor Data Schedule. These documents include:

*All shop drawings to be used in the fabrication of the fuel elements

*A detailed description as to the weighing procedure by which the supplier proposes to assign Plate U-235 content as required in Section 4.3.

Included in the description must be sampling, analytical, and quality control procedures; a statement as to the established accuracy and precision of the assigned fuel plate and fuel element U-235 content; developmental and production data in support of the accuracy and precision estimate; and data which at the 95% confidence level shows that the method used to assign U-235 values has a bias which is less than 0.2% relative

- *A detailed description as to the manner the supplier will use to verify the fuel Plate U-235 value as required by Section 4.3

- *All fabrication, assembly, cleaning, surface treating, handling, and decontamination procedures (not to be construed as an outline)

- *All production test, inspection, and quality control procedures, including all nondestructive and destructive tests and all standards and sampling section drawings. All data from these tests, including but not limited to: radiographs, metallographic samples, ultrasonic testing traces, and qualification yield rates

- *All packaging, storage and shipping procedures

- *Integrated manufacturing and inspection test plan.

6.3.2 Pre-repair:

*All repair programs and procedures prior to use.

6.3.3 Manufacturing Schedule:

*A schedule using a purchaser approved technique.

6.3.3.1 Reports.

1. Biweekly qualifications phase summary status report. The first such report shall be initiated 1 month after date of contract award.

2. Three (3) copies of a monthly report detailing program progress against a previously submitted schedule shall be supplied by the supplier to the purchaser. Report type, format and submittal schedule shall be as agreed upon between the purchaser and supplier.

6.3.4 Delivery Submittals:

Three copies (except as noted) of the following data and records shall be sent prior to or accompany the shipments. The supplier shall maintain copies of these records for at least 10 years and until the supplier has received written approval from the purchaser for disposition or disposal:

- Certification of product compliance to the requirements of this specification to include any test data pertaining thereto

- Supplier's core compact data sheets with individual fuel plate uranium composition data including:

  - Serial number with *batch* (see def.) identification
  - Uranium content
  - Fuel plate core weight
  - U-235 enrichment
  - Total quantity U-235 content
  - Core void density data

- Individual fuel element composition data, including:

  - Uranium content
  - U-235 content
  - Serial number of each plate in the element

- Radiation count from fuel plate and fuel element exterior as required by Section 3.5.3 and 4.11. The counting period, counter, background, efficiency, and type of counter used shall be reported

- List of all applicable waivers and deviations and related fuel plates or fuel elements.

- If performed, documented evidence of the performance and test results of the boehmite formation from the fuel element surface treatment per Section 3.8.

6.3.5 Fuel Plate Radiographs:

Fuel plate radiographs of all accepted fuel plates required by Sections 4.4 and 4.5 and Appendix A shall be sent to the user.

6.3.6 Core Compact Data Sheets:

Supplier's fuel core compact data sheets shall be supplied to the INL Quality Assurance Representative as they are generated.

6.3.7 Report of Production by Unapproved Process:

Whenever the supplier's previously submitted and approved process control limits are exceeded, or any material or fuel element components are fabricated using equipment, personnel, or processes which are not purchaser approved, the time, nature, description, corrective action to be taken, and proposed further corrective action shall be reported

immediately by the supplier, with a written report to the purchaser to follow within 10 working days.

1. T Samples: Transverse to be taken equally space along Fuel Core length.

2. L Samples: Longitudinal to be taken at center line and to include the Dogbone Area.

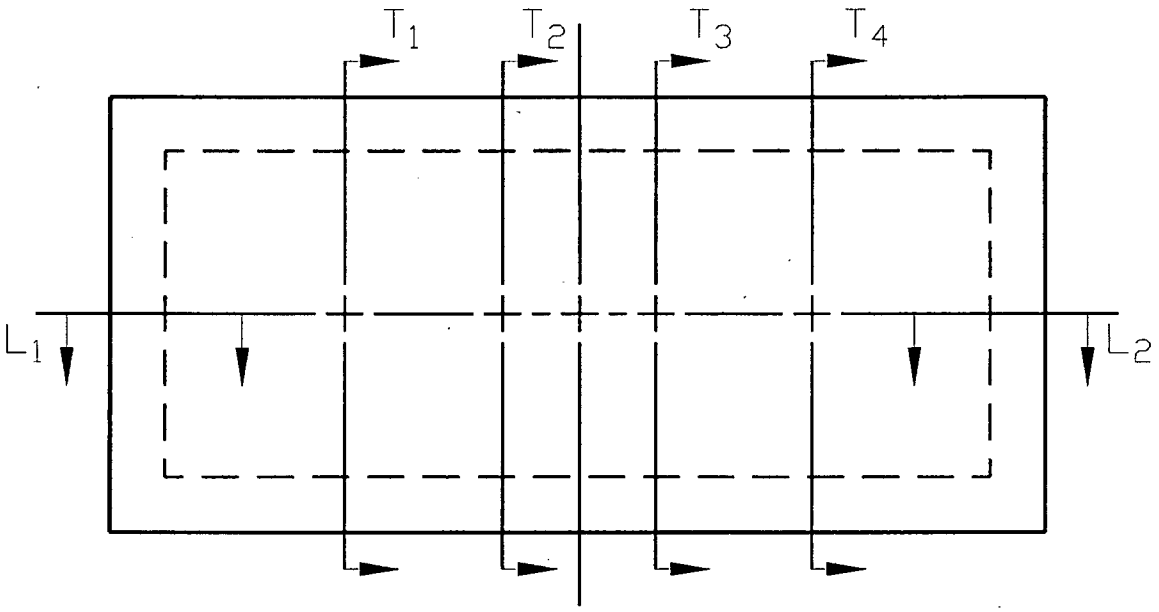

Figure 1. Purdue University Fuel Plate Sampling Procedures For Destructive Tests.

APPENDIX A

Requirements for Radiography of Purdue University Fuel Plates

1. Scope

This specification provides requirements for radiography of Purdue University reactor fuel plates, acceptable film quality and film identification.

2. Requirements

A procedure must be written to specify the details for achieving acceptable fuel plate radiographs. The procedure must include the requirements given in this specification.

2.1 Equipment Setup

The voltage shall be 100 k.v.p. with a focal spot size of 5 mm maximum. The distance between the focal point and the plate shall be at least twice the length of the plate. The focal point shall be centered laterally and longitudinally over the plate or group of plates.

2.2 Film

2.2.1 The image outline shall be clear and sharp; the film shall be free of runs, streaks, scratches, blurs, and cassette defect that will affect the area covered by the fuel plates.

2.2.2 The film density of all points of the radiograph that correspond to the fuel plate border locations outside the plate core shall provide densitometer readings between 1.5 and 2.7. Film density as read over the nominal density standards shall provide densitometer readings between 0.9 and 1.5.

2.2.3 The film shall be extreme sensitivity, extra fine grain, high contrast, double emulsion, industrial x-ray type, (Kodak type M or equal) which is acceptable to the purchaser. Development of the film shall be in accordance with the manufacturer's recommendation.

2.3 Film Identification

A system of identification of the film shall be provided by the supplier, which shall show as a minimum:

A. Plate lot number

B. Plate type and serial number

C. Orientation of density standard

D. Density standard identification

E. Date of radiography.

APPENDIX B

Welding Requirements and Qualification for Purdue University Fuel Elements

1. Scope

The requirements for welding and for the evaluation of welds applicable to the Purdue University Fuel Element Container and components are established by this Appendix.

1.1 Application. This document defines requirements for the following:

1.1.1 Welding procedure qualification.

1.1.2 Performance qualification of welders, welding equipment, and special fixturing.

1.1.3 Information to be included in welding procedure specifications.

1.1.4 Application of qualified procedures to production welding.

1.1.5 Destructive testing and nondestructive examination for qualification and for production welding.

1.2 Special Limitations for Applicability. The requirements contained in this appendix are to some degree based on RDT F6-2T. Those requirements applicable to Manual, GTAW, single pass, welding of Plug Joint welds, Corner Joint welds, and Partial Penetration Butt Joint welds have been included in this appendix. The introduction of a new weld design or weld process requiring a change in these limited parameters would require an appropriate review of RDT F6-2T for requirements applicable to the new parameters.

1.3 Definitions.

Arc Strike. Any localized melting, heat affected zones, or change in the contour of the surface of the finished weld or adjacent base metal resulting from an arc or heat generated by the passage of electrical energy between the weld or base metal and a current source; such as welding electrodes, electron beams, ground clamps, high frequency arc, etc.

Automatic Welding. Welding with equipment which performs the entire welding operation without constant observation and adjustment of controls by an operator. The equipment may or may not perform the loading and unloading of the work.

Face of Weld. The exposed surface of a weld on the side from which welding was done.

Face Reinforcement. Reinforcement of weld at the side of the joint from which welding was done.

Heat. A single homogeneous melt of metal or alloy.

Joint Penetration. The minimum depth a groove or flange weld extends from its face into a joint, exclusive of reinforcement.

Machine Welding. Welding with equipment which performs the welding operations under the constant observation and control of an operator. The equipment may or may not perform the loading and unloading of the work.

Position of Welding. The terms related to positions of welding for joint types and welding processes and the position limits are defined in Section IX, ASME Boiler and Pressure Vessel Code.

Repair. The process of restoring a nonconforming item characteristic to an acceptable condition, although it does not conform to a specified requirement.

Rework. The process by which a nonconforming item is made to conform to specified requirements.

Root of a Joint. That portion of a joint to be welded where the members approach closest to each other. In cross section the root of the joint may be a point, a line or an area.

Root of a Weld. The points, as shown in cross section, at which the back of the weld intersects the base metal surfaces.

Root Penetration. The depth a groove weld extends into the root of a joint measured on the centerline of the root cross section.

Root Reinforcement. Reinforcement of weld at the side opposite that from which welding was done.

Root Surface. The exposed surface of a weld on the side opposite that from which welding was done.

Size of a Groove Weld. The joint penetration (depth of chamfering plus root penetration when specified).

Undercut. A groove melted into the base metal adjacent to the toe or root of a weld and left unfilled by weld metal.

Underfill. A depression on the face of the weld or root surface extending below the surface of the adjacent base metal.

Welder and Welding Operator Performance Qualification. The tests to demonstrate a welder's or welding operator's ability to produce welds meeting prescribed standards.

Welder. One who is capable of performing a manual or semiautomatic welding operation (sometimes erroneously used to denote a welding machine).

Welding Operator. One who operates machine or automatic welding equipment.

Welding Procedure Qualification. The test to demonstrate that welds made by a specified procedure can meet prescribed standards.

Welding Procedure Specification. A written welding procedure which specifies the detailed methods and practices to be used in the production of a weldment and how they shall be carried out. A specification includes all elements of a procedure necessary to produce a satisfactory weldment. Examples of some of the elements included in a specification are: material used, preparation of base materials, preheat and postheat cleaning, assembly method and sequence, fixturing, heat treatments, joint welding procedures, preweld and postweld nondestructive examinations, repair, rework, etc.

Welding Procedure. The detailed methods and practices including all joint welding procedures.

2. Reference Document

The following documents are a part of this appendix to the extent specified herein. The issue of a document in effect on the date of the invitation to bid, including any amendments also in effect on that date, shall apply unless otherwise specified. Where this appendix appears to conflict with the requirements of a reference document, such conflict shall be brought to the attention of the purchaser for resolution.

2.1 American Society for Testing and Materials (ASTM) Standards

2.1.1 ASTM E2, Preparation of Micrographs of Metals and Alloys

2.1.2 ASTM E3, Preparation of Metallographic Specimen

2.2 American Society of Mechanical Engineers (ASME) Codes

2.2.1 ASME Boiler and Pressure Vessel Code, Section IX, Welding Qualifications

2.3 American Welding Society (AWS) Standards

2.3.1 AWS A2.2, Nondestructive Testing Symbols

2.3.2 AWS A3.0, Terms and Definitions

3. Weld Qualification Requirements

3.1 General Requirements

3.1.1 All welding procedures, welders, or welding machine operators shall be qualified in accordance with the provisions identified in this Appendix.

3.1.2 Weld Procedure and Performance Qualification Testing previously qualified to these requirements under other contracts may be used. Existing records to support previously qualified procedures and personnel are subject to review by the purchaser.

3.1.3 Base materials and filler material shall comply with the requirements of the drawings.

3.1.4 Welding processes which satisfy the specified requirements and produce the quality required by this Appendix are permissible. Welding procedures which utilize fluxes and coatings shall not be used.

3.1.5 Fixtures: The capability of fixtures for aligning parts shall be demonstrated before welding of production parts is initiated. If chill bars or blocks are used, the type of material and their location with respect to the joint shall be included in the procedure specification.

3.1.6 Position of Qualification Welds. All procedure and performance qualification test welds shall be made in the same positions as for production welds.

3.1.7 Special Conditions for Qualification Welds: All procedure and performance qualification test welds shall be made under conditions which simulate the actual production welding conditions. These conditions shall include space limitations, joint accessibility, degree of comfort due to heat, position and other handicaps or environmental factors which the welder or welding operator will endure during actual production welding.

3.1.8 Heat Treatment. Weld preheat and postheat treatments shall not be used without prior approval by the purchaser.

3.1.9 Interpass Temperature. For multi-pass weld, the weld interpass temperature shall not be less than 60° F or greater than 350° F without prior approval by the purchaser.

3.1.10 Records. Records of welding, associated processing, and inspection shall be maintained for all welds. Complete records may consist of inspection forms, routings, or reference to Operating Procedures or other documents. These records shall include at least the following:

1. Base Material (Type, material specification, heat or lot number).
2. Filler Material (Type, material specification, heat or lot number).
3. Cleaning procedures.
4. Joint identification and weld maps when applicable.
5. Welding machine type and identification.
6. Welding procedure specification.
7. Welder or welding operator qualification.
8. Procedure and performance qualification.
9. Current-voltage data for machine or automatic welding.
10. Date welds are made.
11. Inert gas analysis, when applicable.
12. Nondestructive examination procedure.
13. Nondestructive examination personnel identification.
14. Examinations and tests (nondestructive and destructive) and the results.
15. Photomacrographs and photomicrographs.
16. Metallographic specimens.
17. If applicable, rework and repair of welds.
18. Disposition of welds.

3.2 Welding Procedure Specification

3.2.1 The welding procedure specification shall meet the requirements of this Appendix, and shall be submitted to the purchaser for approval prior to any production welding.

3.2.2 The welding procedure specification shall include all essential elements and details, as required by this section, to cover each joint to be welded by the supplier. Each joint shall be identified in the welding procedure

specification. The specification shall include a joint design sketch for each joint welding procedure even if the specification references drawing numbers.

3.2.3 The following basic information and essential variables shall be included in sufficient detail to assure that compliance with the requirements of the specification can be verified:

1. Basic Information

   a. Joint Design: (the joint geometry, fit-up, and other required dimensions of the welded joint) tolerances and material thickness.

   b. Method of arc initiation

   c. Electrode size (for gas tungsten arc welding)

   d. Gas type and flow rate (shielding and backing gas)

   e. Welding current range for manual welding

   f. Whether tack welds or fixtures are used for assembly of the joint for welding

   g. Method and frequency of cleaning

   h. Number of weld layers and passes

   i. Whether stringer beads or weave beads are used

2. Essential Variables

   a. General, All Welding Processes.

      i. A change from a base material type or grade (materials of the same nominal chemical analysis and mechanical property range, even though a different product form) to any other base material type or grade. When joints are made between two different types or grades of base material, a procedure qualification shall be made for the applicable combinations of materials, even though procedure qualification tests have been made for each of the two base materials welded to itself.

ii. A change of filler metal type or classification to any other type of classification

iii. A change in welding position.

iv. A change in vertical welding direction, i.e., from upward to downward or vice versa.

v. The addition or omission of integral backing (e.g., "butt-lap" type joint).

vi. The addition or omission of nonfusing metal retainers.

vii. The addition or omission of filler metal to the joint.

viii. Any change in the method by which filler is added, such as preplaced shims, preplaced wire, preplaced consumable inserts, wire feed, or prior weld metal surfacing ("buttering") of one or both joint faces.

ix. The addition or omission or any type of preplaced consumable inserts or joint surfacing.

x. A change in the shape or size of preplaced consumable inserts or joint surfacing.

xi. A change from multiple pass welds to single pass welds.

xii. The omission of inert gas backing during welding, except that requalification is not required where a qualified welding procedure is changed to omit the inert gas backing and then is used only for a single welded butt joint with a backing strip, or a fillet weld. For multiple pass welding, the omission of inert gas backing during welding until three layers or 3/16 of weld metal thickness has been deposited, whichever is greatest.

xiii. A change from one welding process to any other process or combination of welding processes.

b. Manual Welding, All Welding Processes.

i. An increase in the standard size of filler metal from that stated and qualified in the procedure specification.

ii. A change in joint geometry which violates the tolerances given for the joint geometry elements listed below:

Bevel Angle: State in procedure specification. Tolerance: Minus 5%.

Groove Angle: State in procedure specification. Tolerance: Minus 5%

Alignment Tolerance: Assign value in procedure specification. Qualify procedure for single welded joints using maximum permissible misalignment in a portion of the joint.

c. Gas Tungsten Arc Process.

i. A change of electrode material type.

ii. A change in arc starting methods.

iii. A change from a single shielding gas to any other shielded gas or to a mixture of shielding gases or a change in specified composition of gas mixture.

iv. A decrease in shielded gas flow rate of more than ten percent.

3.3 Welding Procedure Qualification

3.3.1 The welding procedure shall be qualified to the requirements of this section.

3.3.2 All welding used in qualifying a welding procedure shall be performed in accordance with a welding procedure specification.

3.3.3 Before any welding is performed on production components, the supplier shall qualify each proposed welding procedure by:

1. Recording all essential elements of the welding procedure in a welding procedure specification (see Section 3.2)

2. Verifying the welding procedure specification by welding test specimens representing each joint to be welded in production and performing nondestructive examination and destructive tests in accordance with the requirements of this Appendix.

3. Submitting to the purchaser, for approval, the welding procedure specification and a certified copy of the detailed results obtained from the tests performed on the test welds. The metallographic sections required by this Appendix shall also be submitted to the purchaser.

3.3.4 Essential Variables. The welding procedure shall be set us as a new welding procedure specification and shall be completely re-qualified when any of the changes listed in Section 3.2.3.2 are made in the procedure.

3.3.5 Chart Recordings. Current-voltage-time charts shall be used for each procedure qualification weld for automatic or machine welding. Calibrated current and voltage indicating meters may be substituted for trace chart type equipment for manual welding. The current and voltage ranges shall be recorded for manual welding.

3.4 Welder Performance Qualification

3.4.1 Performance qualification weld tests shall meet the requirements of this section, except that any welder used to qualify the welding procedure shall also be considered qualified and additional performance weld tests are not required.

3.4.2 General.

1. The performance qualification tests are intended to determine the ability of welders to make sound welds.

2. The performance test may be terminated at any stage of the testing procedure whenever it becomes apparent to the supervisor conducting the tests that the welder does not have skill required to produce satisfactory results. In this event, the welder may be re-tested at the discretion of the supplier in accordance with 3.4.3.

3. Each supplier shall maintain a record of the procedures, including the essential variables, under which welders are examined and the results of the examinations.

3.4.3 Qualification of Welders.

1. Each welder shall pass the tests prescribed for procedure qualification except that tensile tests are not required. The essential variables and the test results obtained by each welder shall be recorded in a Performance Qualification Test Report. Any welder who performs acceptable welding procedure qualification tests shall be considered qualified.

2. Renewal of Qualification. Requalification of a welder is required when:

a. 90 or more days have elapsed since he last produced acceptable welds using the specific welding process, or

b. He has not performed acceptable welds using the production welding procedure.

c. Any time there is a specific reason to question a welder's ability to make welds meeting the requirements of this Appendix, requalification shall be required. Only one test weld shall be required for renewal of qualification. If this test weld fails to meet all of the original requirements, then a complete performance requalification shall be required.

3.4.4 Chart Recordings. Current-voltage-time charts shall be used for each procedure and performance qualification weld for automatic or machine welding. Calibrated current and voltage indicating meters may be substituted for trace chart type equipment for manual welding. The current and voltage ranges shall be recorded for manual welding.

3.5 Welding Machine Qualification

3.5.1 Performance qualification weld tests shall meet the requirements of this section, except that any welding machine used to qualify the welding procedure shall also be considered qualified and additional performance weld tests are not required.

1. The performance qualification tests are intended to determine the ability of welding machines to make sound welds.

2. Any time there is a specific reason to question a welding machine's ability to make welds meeting the requirements of this Appendix, requalification shall be required. Only one test weld shall be required for renewal of qualification. If this test weld fails to meet all of the original requirements, then a complete performance requalification shall be required. Welding machines used for the manual welding of any successful procedure or welder qualification tests shall be considered qualified for manual welding of all core components covered in this Appendix.

3.5.2 Welding machines used for the manual welding of any successful welder performance qualification tests shall be considered qualified for manual welding of all components covers in this session.

3.6 Examination & Tests

3.6.1 Type of Test Required. The following tests shall be used for the qualification of welding procedures and / or welders as applicable:

1. Nondestructive examination by a liquid penetrant method.

2. Nondestructive examination by Visual to test for soundness and surface characteristics of the weld.

3. Destructive examination by sectioning for metallographic examination of weld joints and adjacent areas to test for fusion, weld geometry, weld reinforcement, and soundness of the weld.

4. When the purchaser has reason to believe that the quality of any weldment is doubtful, he may require additional inspection.

5. Nondestructive Examination and Tests

   a. Visual. The test weld shall be examined visually prior to welding and after welding in accordance with Section 5.1

   b. Liquid Penetrant. The test weld shall be examined after the final layer in accordance with Section 3.6.2.2 using a color contrast method.

   c. Unless otherwise specified, inspection of procedure and performance qualification welds shall be performed in the final surface condition.

6. Destructive Examination. Each test weld shall be sectioned transversely to metallographically examine a minimum of:

   a. Three section faces for welds on cylindrical components less than 1¼ inch in diameter or for welds that are one to four inches long on non-cylindrical components.

   b. Four sections faces for welds in cylindrical components that are greater than 1¼ inch in diameter or for welds that are greater than four inches long on non-cylindrical components.

c. One section face for plug welds, arc spot welds, and welds that are less than one inch long on non-cylindrical components.

d. The cross section shall be polished and etched to provide clear definition of the structure in the fusion zone and the heat-affected zones.

e. For welds in (a) and (b) of this paragraph, one cross section shall be made through a weld start and a weld stop area and the remaining sections shall be made at random. For weld described in (c) of this paragraph, the cross section shall be made at the approximate centerline of the weld. Examination of the welds shall be in accordance with Section 3.6.2.3.

3.6.2 Acceptance Criteria for Qualification Test Welds

1. Visual Examination. Visual examination shall be in accordance with Section 5.1.

2. Liquid Penetrant Examination. Unless otherwise specified, final weld surfaces shall be examined using a color contrast method.

a. For welded joints in materials less than 1/8 inch thick the following relevant indications are unacceptable.

i. Any cracks.

ii. Linear indications.

iii. Indications with dimensions exceeding 1/64 inch.

iv. Rounded indication separated by ¼ inch or less edge-to-edge.

v. Five or more rounded indications in any six square inches of weld surface with the major dimension of this area not to exceed six inches with the area being taken in the most unfavorable location relative to the indication being evaluated.

b. For all welds in materials 1/8 inch thick or greater, the following relevant indications are unacceptable. (Only

those indications with major dimensions greater than 1/64 inch are considered relevant for item (iii).)

i. Any cracks.

ii. Any linear indications.

iii. Rounded indications with dimensions exceeding 10 percent of the nominal weld thickness or 1/8 inch, whichever is smaller. Rounded indications separated by 1/16 or less edge-to-edge shall be evaluated as a single indication.

iv. Four or more rounded indications in a line separated by 1/16 inch or less edge-to-edge.

v. Six or more indications in any six square inches of weld surface with the major dimension of this area not to exceed six inches with the area taken in the most unfavorable location relative to the indications being evaluated.

vi. Aligned indications in which the average of the center-to-center distance between any one indication and the two adjacent indications in a straight line is less than 3/16 inch.

3. Metallographic Examination Metallographic examinations shall be performed on qualification test welds at not less than 50X on test welds as required in this Section in accordance with ASTM E.2. Any cross section which is shown by metallographic examination to contain any of the following relevant defects shall be cause for rejection of the test welds.

a. Any cracks.

b. Incomplete fusion, or insufficient joint or root penetration.

c. Any tungsten inclusions, slag inclusions, or porosity having a maximum dimension greater than 20 percent of the weld thickness or 1/32 inch, whichever is smaller.

d. More than four tungsten inclusions or pores which have a maximum dimension less than in (c) above.

e. Any deviation from specified weld geometry or weld reinforcement.

3.6.3 Test Welds.

1. Procedure and / or welder performance qualification shall be made on test welds which duplicate the production weld joint type and which simulate the conditions to be used in production with respect to orientation, the essential variables listed in Section 3.2.3.2, and the dimensions of the parts to be joined to the extent that they affect heat requirements, relative motions, and distortions. All welding used in qualifying a welding procedure and / or welder performance shall be performed in accordance with the procedure specification.

2. For manual welding, two consecutive test welds shall be made when the weld joint is less than six inches in length. Only one test weld shall be required when the weld joint is 6 inches or greater in length.

3. All test welds shall be tested using the required tests listed in Section 3.6.1. To qualify the procedure specification used in making the test welds, each weld shall pass the required tests.

4. Repair of procedure or performance qualification test weld(s) is prohibited.

4. Production Welding

All production welding shall be accomplished using approved welding procedure specifications and qualified welders and/or welding operators.

5. Quality Acceptance of Production Welds

5.1 All completed production welds shall be visually examined in accordance with the following requirements:

5.1.1 General Visual Inspection Requirements. All visual examinations shall comply with the following:

1. Visual examination shall be made under direct daylight-type fluorescent lighting of at least 100 foot-candles at the work examination area.

2. Visual examination shall be performed with the aid of a 5x (minimum) magnifying glass.

3. The inspection required by this standard shall not be performed by the welder who made the welds. However, if the welder is qualified in accordance with this standard, he may visually inspect his own welds prior to the inspections required by this standard.

4. Personnel performing visual inspection shall have 20-20 vision, natural or corrected, stereo acuity, and shall not be color-blind.

5.1.2 Visual Acceptance Criteria (except for porosity). Visual examination of weld joint preparations and welds shall be performed in accordance with the following requirements to verify conformance to the written welding procedure, the design requirements, and the requirements of this standard:

1. Prior to welding, the weld joint edges and adjacent surfaces shall be examined for:

   a. Proper edge preparation, dimensions, and finish.

   b. Alignment and fitup of the pieces being welded.

   c. Verification of correct material by check of records.

   d. Verification of the cleanliness requirements.

2. After welding, the joint shall be examined in the final surface condition for:

   a. Contour, reinforcement and surface finish of welds.

   b. Degree of underfill, undercut, and overlap.

   c. Arc strikes, weld spatter and impression marking.

   d. Burn-through and fuse-through

3. Weld joints and surfaces which are shown by visual examination to have any of the following defects or areas of nonconformance are unacceptable:

   a. Any nonconformance revealed by 5.1.2.1.

b. Any zone of incomplete fusion.

c. Insufficient joint or root penetration.

d. Any undercutting, underfill, or burn through.

e. Any concavity on the face side of groove welds.

f. Any arc strikes, weld spatter, and impression marking.

g. Any visible inclusions, porosity, cracks, and unfilled craters.

4. Machined welds shall meet the drawing requirements.

5. All welds shall be free from surface markings resulting from mishandling, punching, scratching, etc., which exceed the specified surface requirements.

6. All welds shall be free of dross, or slag.

7. All welds shall be free of oxidation due to improper shielding and overheating which produce black or gray spalling or loose particles. Iridescent temper films and the dark metallic vapor deposits which may occur adjacent to the welds are acceptable. These films and deposits shall be removed by approved cleaning procedures when accessible.

5.1.3 Visible unacceptable porosity is as follows:

1. Four or more pores with a major dimension of 0.048 inches or more randomly positioned.

2. A single pore with a major dimension of 0.064 inches or more.

3. Six or more pores with a major dimension of 0.016 inches or greater in one weld.

4. Four or more porosity with a major dimension of 0.016 inches or greater, in line separated by less than 0.063 inches from edge to edge.

5.2 Repair of a defective weld by welding shall be limited to two attempts. Unacceptable defects shall be removed and re-examination made using liquid penetrant color contrast method to assure complete removal of the defect. If the

removal of the defects results in reducing the thickness of the weld metal below the thickness of the base metal, the area shall be rewelded using a welding repair procedure which has been approved by the Purchaser. Whenever a defect is removed and subsequent repair by welding is not required, the excavated area shall be blended into the surrounding surface to remove any sharp notches, crevices or corners. Completed repairs shall be visually re-examined per Section 5.1. Records shall be maintained on all repairs and shall include the following:

5.2.1 Location of joint.

5.2.2 Location of defect.

5.2.3 Description of defect, including type and size.

5.2.4 Reference to approved repair procedure.

5.2.5 Inspections before and after repair and the results thereof.

5.2.6 Identification of repair welders or welding operators.