PURDUE
UNIVERSITY

SCHOOL OF NUCLEAR ENGINEERING

31 January 2012

Document Control Desk
US Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Attn: Ms. Cindy Montgomery, Research & Test Reactors (NRR/DPR/PRLB), Mailstop O12 D20

SUBJECT: PURDUE UNIVERSITY - REQUEST FOR ADDITIONAL INFORMATION REGARDING THE PURDUE UNIVERSITY REACTOR LICENSE RENEWAL (TAC NO. ME 1594), RESPONSES TO RAIs (ML103400115 and ML103400250)

Dear Ms. Montgomery:

Enclosed please find the responses to the Request for Additional Information regarding the Purdue University Reactor License Renewal dated 6 July 2011. Included with this submission are responses to questions 45, 55, 62, 65, 66, 67, 68, 70, and 73. Should you have any questions or require further information, please don't hesitate to call me at 765.496.3573, or e-mail at jere@purdue.edu.

I hereby certify under penalty of perjury with my signature below that the information contained in this submission is true and correct to the best of my knowledge.

Very respectfully,

/SA

Jere H. Jenkins
Director of Radiation Laboratories

Attachments: As described.

Cc: Duane Hardesty, USNRC Project Manager for PUR-1
Leah Jamieson; Purdue University College of Engineering
Jim Schweitzer, Purdue University REM, CORO Chair
Ahmed Hassanein, Purdue NE

REQUESTED ADDITIONAL INFORMATION IN RESPONSE TO RAIs

REGARDING THE PURDUE UNIVERSITY REACTOR LICENSE RENEWAL (TAC NO. ME 1594)

45 Pursuant to 10 CFR Part 55.59(a)(2), each licensee shall: "Pass a comprehensive requalification written examination and an annual operating test." In your Requalification Plan, Section B you state that "Completion of the biennial requalification program will consist of a written examination and a demonstration of operator proficiency in reactor operation.

A. Please explain how the facility ensures that operator proficiency examinations are performed annually during the biennial requalification cycle in compliance with 10 CFR 55.59(a)(2) or update your plan accordingly.

B. According to 10 CFR 55.53(h), licensees are required to complete a requalification program as described by 10 CFR 55.59. Regulation 10 CFR 55.59(a) states that each licensee shall:

(1) Successfully complete a requalification program developed by the facility licensee that has been approved by the Commission. This program shall be conducted for a continuous period not to exceed 24 months in duration.

(2) Pass a comprehensive requalification written examination and an annual operating test. Section F. of the PUR-1 Requalification plan states that:

"During intervals when the licensed operations crew consists only of senior operators who are instructors for topics in part a.1.b., the requalification program will be modified to exempt those senior operators from parts A and B.1. Parts B.2, C, D, and E will remain in effect. When the licensed operations crew increases to include those who do not instruct in the program, the program will revert to its initial content. Operators may place a statement into the file stating that they have done a literature review and/or instructed the topics in Section A and B.1 in lieu of meetings and exams. During intervals when the licensed operations crew consists of only one senior operator this operator will be exempt from parts A and B, part C would be documented in the console logbook and as stated in C.3, parts D and E will remain in effect. In any of the requalification activities, exclusive of operations, additional methods may be used to accomplish the training requirement. These may include mail, electronic classroom or other methods may be used for training, meetings, testing or other required communication(s)."

Response:

The number of staff members at PUR-1 changes from time to time, and there have been periods where there was only one staff member with on operating license. PUR-1 previously had an exemption to allow for this, and will request one again. It will be prepared and submitted under separate cover.

55. Radiation exposure for reactor staff are provided in SAR Section 11.1.5 to show exposures are maintained below 10 CFR 20 and as low as reasonably achievable (ALARA). Please provide analysis data to support that the maximum exposed member of the public, at the closest residence to the reactor and at any other points of special interest (e.g., adjacent classrooms and offices, dormitories), as applicable, are maintained below regulatory limits and ALARA.

Response:

PUR-1 maintains perimeter dosimetry via thermo-luminescent dosimeters at four different locations, including the closest classroom to the reactor room. These dosimeters are read every two months, and the records are maintained by the university's department of Radiological and Environmental Management. PUR-1 has continuously met regulator limits and ALARA principles with respect to public dose.

62. Inconsistencies are noted throughout the SAR for referenced maximum power and requested maximum licensed power under the PUR-1 license renewal. For example, in Section 1.2 of the SAR, PUR-1 is requesting a license "for a power uprate to continuous operation at 10 kW, and maximum short term power of 12.5 kW." Any reactor power excursions, whether short term or long-term must be within the maximum licensed power for the facility. Please clarify the desired maximum licensed power level requested and ensure this power level, including any uncertainty in reactor power, is consistently applied in the safety analyses for the license. Please provide updates to the safety analysis that correctly reflects the desired maximum licensed power under this renewal of the PUR-1 facility license.

Response:

These have been corrected in the SAR. The previously addressed maximum license power requested for PUR-1 is 12 kW. A revised section 4.6 from the SAR is included in this response, reflecting the analysis to 12 kW.

65. SAR section 4.6.2, page 4-42 references the margin to incipient boiling shown in Table 4-28 calculated at an operating power of 1 kW for PUR-1. Table 4-28 is not provided in the SAR. Please update the SAR to provide the correct reference for the margin to incipient boiling and ensure that the parameters in the analysis support the maximum reactor power requested, including any uncertainty for power measurement. Please provide an updated evaluation of a safety analysis that explains all analyses, assumptions and conclusions at the requested maximum licensed power level.

Response:

These have been corrected in the SAR. The previously addressed maximum license power requested for PUR-1 is 12 kW. A revised section 4.6 from the SAR is included in this response, reflecting the analysis to 12 kW.

66. SAR, Section 4.6.3.1 and Table 4-22 provides a summary for the onset of nucleate boiling. The margin to incipient boiling provided in the SAR was calculated using reactor power levels of 1 kW and 10 kW. Using the guidance of NUREG-1537, Section 4.6, please provide an updated evaluation of a safety analysis that explains all thermal margin calculations, including the margin to incipient boiling, at the requested maximum licensed power level with uncertainty, including power measurement uncertainty.

Response:

These have been corrected in the SAR. The previously addressed maximum license power requested for PUR-1 is 12 kW. A revised section 4.6 from the SAR is included in this response, reflecting the analysis to 12 kW.

67. SAR, Section 4.6.3.3 and Table 4-23 has an associated 50% uncertainty with the PUR-1 maximum reactor power level. SAR, Section 13.1.2 has stated that there is a 50% uncertainty in power measurement (p. 13-1). Using the guidance of NUREG-1537, Section 4.5 and Section 4.6, please provide: (1) an updated evaluation of a safety analysis that explains the power level uncertainty in terms of both the reactor power that is read by the reactor operator from the reactor power instrumentation and the limiting safety system settings of the reactor protective system, and (2) an updated evaluation of a safety analysis that explains and justifies the derivation of the estimate of 50% power measurement uncertainty.

Response:

The 50% uncertainty assumption was the standard used in the RERTR program, and was what was used in the safety analysis of the LEU conversion. It is a conservative enveloping assumption which should encompass all possible systematic and measurement errors, and is primarily utilized in the hot channel factors of NATCON. The margin to ONB utilizing the 50% uncertainty assumption in NATCON on 12 kW (for the theoretical max power of 18kW) is actually more conservative than using 0% uncertainty on 18 kW, as shown below. In light of this information, and the fact that the 50% uncertainty has been used and found acceptable previously, we stand by the more conservative assumption.

	12 kW with 50% uncertainty	**18 kW with 0% uncertainty**
NATCON calculated ONB	98.55 kW	147.83 kW

68. SAR, Section 4.6.3.3, p. 4-46 states the Limiting Safety System Setting conditions for PUR-1 thermal-hydraulic calculations were computed using the NATCON code using power levels of 1 kW. Please clarify the desired maximum licensed power level requested and consistently apply this power level, including any uncertainty in reactor power, in the safety analyses for all thermal-hydraulics results. Please provide an updated evaluation of a safety analysis that explains all analyses, assumptions and conclusions at the requested maximum licensed power level.

Response:

These have been corrected in the SAR. The previously addressed maximum license power requested for PUR-1 is 12 kW. A revised section 4.6 from the SAR is included in this response, reflecting the analysis to 12 kW.

70. NUREG-1537, Section 11 provides guidance for radiation protection provisions at the facility. In Section 4.4 of the SAR, it is stated that the radiation level above the reactor pool surface is about 1 mrem/hr and that the radiation level along the outside lateral surface of the concrete

biological shield is about 0.1 mrem/hr, when the core is operating at 1 kW. Please provide an updated evaluation of a safety analysis that explains all analyses, assumptions and conclusions at the requested licensed power level for the maximum potential radiation levels and the potential radiation effects on facility staff. As part of evaluation, please indicate if the radiation levels bound those that would be encountered during fuel handling and maintenance operations. Additionally, include an evaluation of the safety analysis for potential dose to the facility staff and members of the public (i.e., classrooms, hallways, adjacent rooms, nearest dormitories, offices, etc.).

Response:

Due to an extended unplanned maintenance outage, we are unable to perform the measurements required to respond to this question. We will forward the results of those tests as soon as they are available.

73. NUREG-1537, Section 11.1.7 provides guidance for environmental monitoring at the facility. SAR section 11.1.7.2 generally discusses perimeter monitoring at the PUR-1 facility. Please provide additional information on the number and types of dosimeters that are deployed within and beyond the site boundary and describe the environmental monitoring program is effectively implemented for the day-to-day operation of the facility, and that any radiological impact on the environment will be accurately assessed.

Response:

There are two TLD based area monitors at the perimeter of the site boundary, and two TLD based area monitors beyond the perimeter of the facility, including in the closest classroom outside the perimeter. These are changed and read bi-monthly.

4.6 Thermal-Hydraulic Design

In this section, the results of the thermal-hydraulic analyses are discussed in order to demonstrate that the PUR-1 core design provides the cooling capacity necessary to ensure fuel integrity under all anticipated reactor operating conditions. Analyses for behavior under hypothetical accident scenarios are presented in Section 13.

4.6.1 NATCON Code Description

Thermal-hydraulic analyses were performed using the computer code NATCON[1,2], which can be used to analyze the steady-state thermal-hydraulics of plate type fuel in a research reactor cooled by natural convection. The reactor core is immersed in a pool of water that is assumed to be at a constant average temperature.

NATCON computes coolant flowrate, axial temperatures in the coolant and fuel plate surface and centerline, and the approach to onset of nucleate boiling (ONB). Other safety related parameters such as the Onset of Nucleate Boiling Ratio (ONBR) and Departure from Nucleate Boiling Ratio (DNBR) are calculated as well. And an automatic search for the power at ONB can be performed.

Flow is driven by density differences in the coolant that are the result of coolant heating by the fuel. Resulting buoyant forces are counter-balanced by viscous forces that result from the flow. Hot channel factors may also be introduced for determining safety margins. NATCON v2.0 documentation is included as Appendix 1 of this document. It includes information on the calculation of hot channel factors, inputs, and use of the code.

4.6.2 Fuel Element and Fuel Assembly Geometry

In PUR-1, each fuel plate element is loaded into its own assembly container, or can, as described in Section. Cross section views of the two different types of assemblies, standard and control, are shown in Figure 4-27 and Figure 4-28.

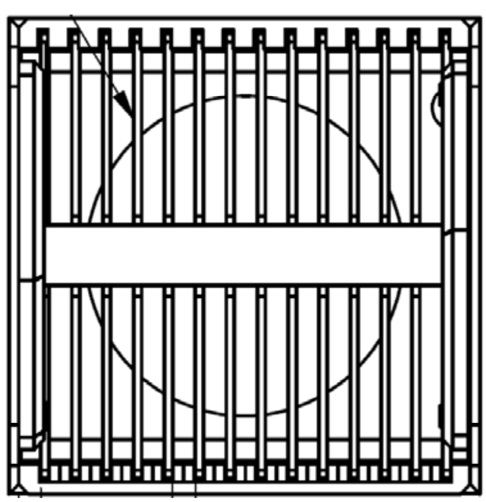

Figure 4-27: Standard LEU fuel assembly.

Figure 4-28: Control LEU fuel assembly.

Two types of channels are encountered in the PUR-1 fuel assemblies. One is the channel between plates, and the other is the channel between the last plate of an element and the assembly can wall. The plate-to-plate and plate-to-assembly wall channel thicknesses are fixed by the spacers on the wall of the assembly. It should be noted that the plate-to-wall channel is heated on only one side, so it can be conservatively assumed that half of the heat from the fuel plate associated with the fuel channel heats the coolant. Table 4-17 summarizes the channel types and thicknesses in PUR-1.

Table 4-17: Channel Types and Thickness in PUR-1 Assemblies

	Plate-to-plate (mils)		**Plate-to-wall (mils)**	
	Standard	**Control**	**Standard**	**Control**
LEU Assemblies	144±15	181±15	127±8	127±8

In the thermal-hydraulic analyses, the peak power plates identified in Section 4.5.2 were analyzed using NATCON. The relative power densities in each fuel plate were obtained from detailed MCNP5 criticality calculations, also described in Section 4.5.2. In the NATCON analysis, the relative axial power profiles of the individual plates were utilized in each respective case. Plate 1348 (see Table 4-9) was the limiting plate.

Hot Channel Factors are used by NATCON to account for dimensional variations inherent in the manufacturing process, as well as variations in other parameters that affect thermal-hydraulic performance. The geometry dimensions used in the NATCON models for the are shown in
Table 4-18: Model Dimensions for the Thermal-Hydraulic Models. And the hot channel factors are listed in

Table 4-19. The hot channel factors were calculated using equations that include the effect of temperature-dependent water viscosity. A conservative uncertainty of 20 mil (rather than 15 mil shown in Table 4-4) 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 4-19,

(ii) a total of [redacted] 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 at 1 kW per average plate), and

(v) the axial power shape for plate 1348 shown in Table 4-9.

To account for the power density variation along the width of plate 1348, the FFILM in

Table 4-19 (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 4-9). This results in FFILM =1.085x1.251 = 1.357 that was used in NATCON to calculate the ONB power.

Table 4-18: Model Dimensions for the Thermal-Hydraulic Models

Number of Axial Nodes	14	
Number of Plates	[redacted]	
Thermal Conductivity (W/m*K) Fuel Meat Clad	 80 180	
Pool Temperature (°C)	30	
	Inches	mm
Fuel Meat Height Width Thickness	 [redacted]	 [redacted]
Channel Height Width	 [redacted]	 [redacted]
Channel Thickness		

Clad Thickness	0.015	0.381
Distance assembly can extends above fuel plate	0.450	11.430

Table 4-19: Hot Channel Factors for the Plate 1348 NATCON Analysis

Uncertainty	Type of Tolerance	Tolerance Fraction	Hot Channel Factors		
			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**		**1.334**	**1.251**	**1.226**
Power Measurement F_Q	Systematic	0.500	**1.500**		
Flow friction factor F_W	Systematic	0.048	**1.048**		
Heat Transfer Coeff. F_H	Systematic	0.200	**1.200**		

Calculations were done to determine the core power distribution and the ONB power for the PUR-1 based on the limiting plate, 1348. The ONB power calculation given below includes the impact of the power density variation along the width of the plate itself. The resulting ONB power was found to be 98.55 kW The 20 mil uncertainty used is conservative compared to the 15 mil uncertainty given in Table 4-17 for the evaluated core design. These results are Table 4-19. 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.

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. 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 margin to incipient boiling shown in Table 4-21 was calculated at the present 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:

$$\text{Margin to ONB} = \text{Minimum } T_{incp}(p, q^{"}(z)F_{flux}) - \left[F_{bulk}\left\{T(z)-T_0\right\} + F_{film}\left\{T_{wall}(z)-T(z)\right\}\right]$$

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_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_r = RPEAK	=	Radial power factor of the plate cooled by the channel
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

4.6.3 Thermal Hydraulic Analysis Results

The NATCON/ANL V2.0 code was used to determine the thermal-hydraulics performance of the PUR-1. First, the code was used to compute the power at which the

ONB is reached for the plates being examined.. This was done to identify the limiting channel. Then the limiting channel was evaluated under nominal operating. The ONB results provide verification that the Safety Limit (SL) and Limiting Safety System Settings (LSSS, trip points) of the Technical Specifications will indeed assure safe operation of PUR-1.

4.6.3.1. NATCON Analyses

The reactor pool temperature varies throughout the year from about 22°C to 30°C depending on the ambient temperature and humidity conditions in the reactor room. In all of the following calculations, the higher value 30°C was used.

The power search function of NATCON was used to determine the power level at the Onset of Nucleate Boiling. **Error! Reference source not found.** provides a summary of the ONB powers for each of the cases analyzed. For the PUR-1 core, the limiting channel/plate was plate 1348 with the plate-to-plate channel, which had an ONB power of 98.6 kW.

Table 4-20: ONB Powers for the high power plates.

	Plate 1348	Plate 1228	Plate 1315
ONB Power (kW)	98.6	153.4	158.9

Using NATCON, and the thermal-hydraulics parameters for the limiting plate (1348), the nominal operating conditions were also calculated. All hot channel factors are included in these calculations. These results are shown in Table 4-21.

Table 4-21: Operating Conditions for PUR-1 as Determined by NATCON for Limiting Plate 1348

	Present Power	**Uprate Power**	**ONB Power**
Power Level	1 kW	12 kW	98.6 kW
Max. Fuel Temp. (°C)	31.92	39.1	112.6
Max. Clad Temp. (°C)	43.42	43.4	112.5
Coolant Inlet Temp. (°C)	30.0	30.0	30.0
Coolant Outlet Temp. (°C)	31.7	35.8	46.3
Margin to incipient boiling (°C)	78.3	67.9	0
Coolant Velocity (mm/s)	5.18	18.5	54.0
Coolant Mass Flow Rate (kg/m^{2}s)	5.16	18.41	53.4

4.6.3.2. Safety Limits for the LEU Core

In PUR-1, the first and principal physical barrier protecting against the release of radioactivity is the cladding of the fuel plates. The 6061 aluminum alloy cladding has an incipient melting temperature of 582 °C. However, measurements (NUREG 1313) on irradiated fuel plates have shown that fission products are first released near the blister temperature (~550 °C) of the cladding. To ensure that the blister temperature is never reached, NUREG-1537 concludes that 530 °C is an acceptable fuel and cladding temperature limit not to be exceeded under any conditions of operation. As a result, PUR-1 has proposed a safety limit in its Technical Specifications requiring that the fuel and cladding temperatures should not exceed 530 °C.

4.6.3.3. Limiting Safety System Settings for the LEU Core

Limiting safety system settings (LSSS) for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. When a limiting safety system setting is specified for a variable on which a safety limit have been placed, the setting must be chosen such that the automatic protective actions will correct the abnormal situation before a safety limit is reached. Table 4-22 shows the maximum power, the LSSS and operating power for PUR-1.

Table 4-22: Key Power Levels for Reactor Operation and LSSS for PUR-1

	1 kW	12 kW Proposed
Maximum Power Level Including 50% Uncertainty	1 kW	12
Limiting Safety System Settings Power Level	1.2 kW	12.0 kW
Operating Power Level	1.0 kW	10.0 kW

During steady-state operation, peak clad temperatures are maintained far below 530°C , as well as below the temperatures required for ONB (see Table 4-21). NATCON was used to determine the minimum power for ONB for core in the limiting channels, as well as the thermal-hydraulic parameters at these calculated powers. The results of these calculations are shown in Table 4-21.

The licensed operating power level of PUR-1 is 1 kW. The LSSS scram setting of 120% power (1.2 kW for 1 kW, 12.0 kW for uprate) is well below the power level of 98.6 kW at which ONB would occur in the respective limiting channels. Thus, the present LSSS on power at 12 kW (120% normal operating power) will easily protect the reactor fuel and cladding from reaching the Safety Limit under steady state operations.

Chapter 13 (Accident Analyses) analyzes two hypothetical transients based on values of the Technical Specifications for the LEU core. These transients are: (1) Rapid insertion of the maximum reactivity worth of 0.3% Δk/k of all moveable and non-secured experiments, and (2) Slow insertion of reactivity at the maximum allowed rate of 0.04% Δk/(k*s) due to control blade withdrawal.

For the case of the rapid insertion, of 0.3% Δk/k, the reactor scram was initiated based on the power level trip, assuming failure of the period trip. For the case of the slow insertion of 0.04% Δk/(k*s), scram was initiated on the second power level trip, assuming the first power level trip failed. The reason for this is that the period trip is never reached for the case of this slow reactivity insertion.

Thus the selected LSSS is a conservative setting which ensures that the maximum fuel and cladding temperatures do not reach the safety limit of 530 °C for the range of accident scenarios that were analyzed. In summary, the selected LSSS will protect the reactor fuel and cladding from reaching the safety limit of 530°C under any condition of operation.

However, a NATCON thermal-hydraulic calculation for the LEU plate 1348 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. The ONB power was found to be 79.3 kW, indicating a large margin compared to the proposed PUR-1 operating power of 12 kW.

4.7 References

1. R. S. Smith and W. L. Woodruff, “A Computer Code, NATCON, for the Analyses of Steady-State Thermal-Hydraulics and Safety Margins in Plate-Type Research Reactors Cooled by Natural Convection,” Argonne National Laboratory (ANL), ANL/RERTR/TM-12, Dec 1988.
2. M. Kalimullah, “NATCON v2.0 Instructions”, Argonne National Laboratory (ANL), ANL/RERTR, July 2006.