REED COLLEGE
RESEARCH REACTOR
LICENSE No. R-112
DOCKET NO. 50-288

RESPONSES TO NRC
REQUEST FOR ADDITIONAL INFORMATION
AND CLARIFICATIONS TO THE
SAFETY ANALYSIS REPORT

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

***REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS**

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Dec 12, 2011

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Docket: 50-288
License No: R-112
RE: Clarifications to the Safety Analysis Report submitted in support of in the relicensing of the Reed Research Reactor - Response to phone calls on July 7, 2011 and July 15, 2011

Enclosed is the response to the request for clarifications in phone calls on July 7, 2011 and July 15, 2011. Per your request, the 10 CFR 50.59 screening for replacing the remaining aluminum clad fuel with stainless steel clad fuel is attached.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on 12/12/11

Melinda P. Krahenbuhl, Ph.D.
Director, Reed Research Reactor.

A020
NRR

Reed Research Reactor Response

RAI 14. Please identify the Limiting Core Configuration as per NUREG 1537 and ANSI/ANSI 15.1.

The March 8, 2011, "Analysis of the Neutronic Behavior of the Reed Research Reactor" written by Oregon State University describes the neutronic evaluation of the core loading at the time of application. Additionally, Oregon State University also completed a thermal hydraulic analysis titled "Analysis of the Thermal Hydraulic Behavior of the Reed Research Reactor". The core loading consisted of TRIGA fuel clad in either stainless steel or aluminum. These reports were attached to the response dated May 20, 2011.

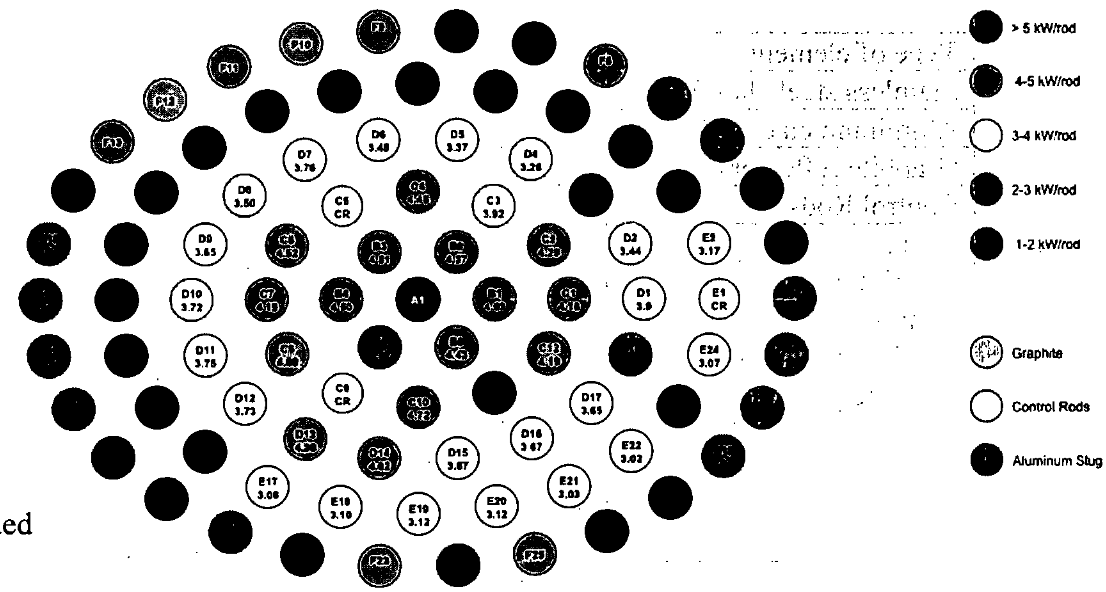
Clarifications for these submitted reports are:

1. The inlet temperature used for the analysis is 49 °C and a corresponding DNBR of 6.3.
2. The flow channel area used is 3.28E-4 m^2, which corresponds to a full hexagonal flow channel.
3. The negative temperature coefficient, and calculations using it, are not used.
4. Equation 1 in the thermal hydraulic report is inconsistent with the discussion in the text.

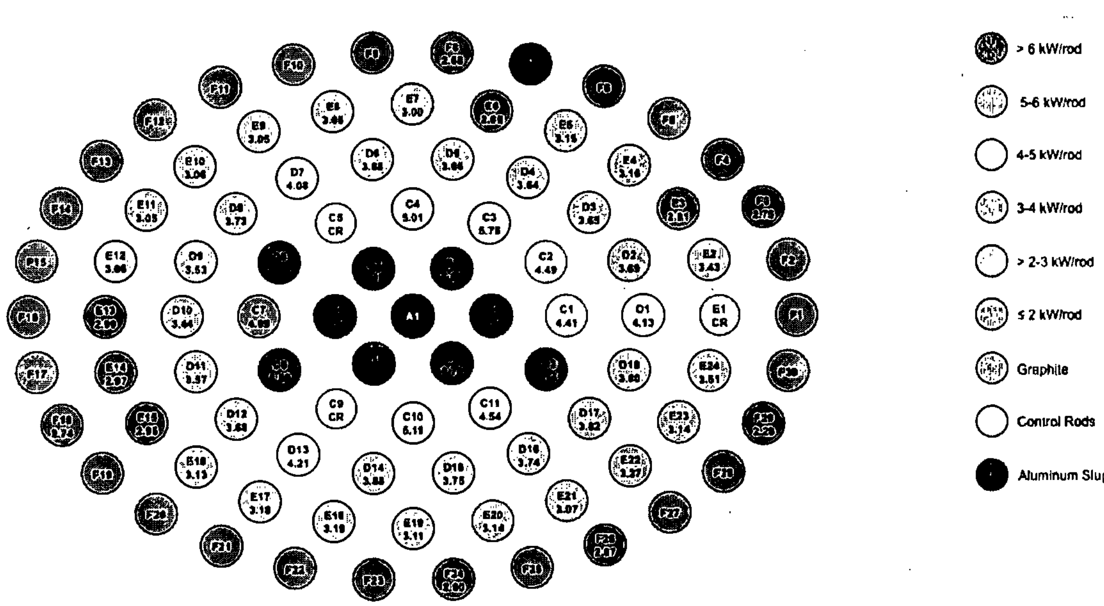
The mixed clad fuel core analyzed in the aforementioned reports has since been replaced with an all stainless-steel clad fuel core. The neutronics for the all stainless-steel clad fueled core was evaluated using the same methodology as outlined in the report. Table 1 contains a summary of the core characteristics for comparison. The SS clad fuel was received as slightly irradiated fuel with an average of 3.8 MW-D on the fuel. This burn-up resulted in an increased number of fuel elements loaded into the core to achieve similar operational characteristics: core excess, shutdown margin and control rod worth. A second neutronic evaluation was completed for the stainless steel loaded core. The results of this analysis are included in Table 1. Although the core loading had changed from 64 to 79 elements, the hot channel identified remains in the same location. The higher burned stainless steel clad fuel has a lower power output per element as expected. The dominating parameter for determining the hot rod channel position is pitch. The core structure, upper grid and lower grid plates were visually inspected for wear and were determined to be usable for the requested license period. Since the pitch is the critical parameter and the all stainless steel core is slightly more burned, the fuel temperatures predicted for the mixed core are higher and has a higher maximum to average rod power output. The mixed core will be used as the limiting core configuration (LCC). The power distribution of the RRR for both core loadings are illustrated in Figure 1.

Table 1 is a comparison of the two cores.

	As Loaded	LCC
Type of element	Number of elements	
Stainless steel clad fuel	79	10
Aluminum clad fuel	0	54
Graphite reflectors	6	21
Control Rods	3	3
Source	1	1
Rabbit terminus	1	1
Central Thimble	1	1
Hot Rod position	B5	B5
Average predicted power per element at 250 kW	3.16 kW	3.91 kW
Maximum predicted power output per element at 250 kW	5.07 kW	7.20 kW
	As measured (2/7/11)	As measured (8/13/10)
Shutdown margin	$2.27	$2.82
Core excess	$1.63	$1.54
Safety rod worth	$2.64	$3.10
Shim rod worth	$3.00	$3.05
Reg rod worth	$1.25	$1.31



As loaded



LCC

Figure 1 Power distribution for the RRR

These analyses indicate that the parameters of the RRR core loaded in a close packed array of LEU fuel operated under the conditions as specified will have a hot rod channel power output less than the LCC which is 7.20 kW and a corresponding maximum centerline fuel temperature of 264 °C at 250 kW. The fuel temperature is significantly lower than the safety limit for stainless steel clad fuel of 1000 °C.

The maximum predicted fuel temperature as a function of thermal power is found in Figure 2 and Table 2. These predicted maximum fuel temperatures were determined using RELAP5-3D as described in the "Analysis of the Thermal Hydraulic Behavior of the Reed Research Reactor." The predicted fuel temperatures compare favorably with measured fuel temperatures at IPR-R1. The measured fuel temperatures in the IPR-R1 loaded with 64 TRIGA elements (5 stainless steel clad and 59 aluminum clad) for a B ring position was 300 °C at 265 kW and 250 °C at 108 kW (Mesquita A, 2007).

Mesquita A.Z."Experimental Heat Transfer Analysis of the IPR-R1 TRIGA Reactor",http://wwwpub.iaea.org/MTCD/publications/PDF/P1360_ICRR_2007_CD/Papers/A.%20Mesquita.pdf.

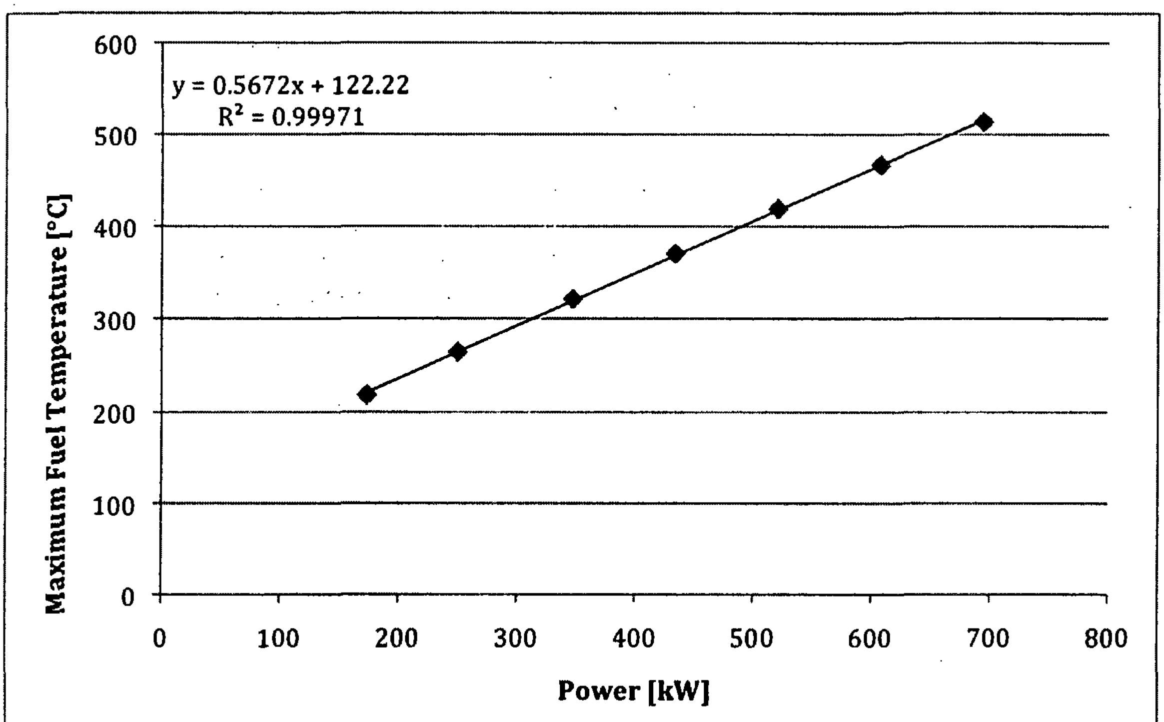


Figure 2 Maximum predicted fuel temperatures for the Reed Research Reactor as a function of thermal power

Table 2 Maximum fuel temperature for selected core powers

Core power	max fuel temp
[kW]	[C]
174	218
250	264
275	278
300	292
347	321
434	370
500	406
521	419
607	466

RAI 16. The rate of reactivity withdrawal is inconsistent between the TS and the SAR (9 or 16 cents/sec). Additionally the time delay for the electronic delay is also inconsistent (0.5 or 1 second).

RRR response
The negative temperature coefficient suggested in NUREG-1282 is -9.5 x 10^{-5} delta k/k °C. GA-7882 contains data with negative temperature coefficients as a function of fuel temperature, ranging from -6 x 10^{-5} delta k/k °C for an aluminum-clad element in a graphite-reflected core to -11 x 10^{-5} delta k/k °C for a stainless-steel-clad, water-reflected core operating at 250 °C. A constant negative temperature coefficient of -7.72E-5 delta k/k °C was used. The constant used is the average of the MCNP predicted values and is between the negative temperature coefficients in GA-7882 for fuel temperatures under 200°C. Ramp rate analysis used the method outlined in attachment 1 of the May 20, 2011 response to RAI#8. Additionally the following parameters were used: Beta = 0.0075, lifetime = 90 microseconds. Table 3 contains a summary of the analysis.

Table 3 Results of the ramp rate analysis

Ramp Rate	Alpha T	Peak Power	time to Peak	Point Reactor ΔT	Max RRR hot spot temp
[cent/sec]	delta k/k °C	[MW]	[sec]	[degrees C]	[degrees C]
16	-7.72E-5	3.9	6.91	26.1	127.0

TS 3.2.1c is consistent with the insertion rate of $0.16 per second. Additionally the maximum scram time is defined as 1 second total from scram initiation to all rods down. A $0.16 ramp withdraw and subsequent electronic delay results in a

peak power of 3.9 MW and a maximum fuel hot spot temperature of 127 °C. The analysis was completed using the LCC. This temperature is well below the safety limit for stainless steel clad fuel. Water temperature was set at 50 °C, ten degrees higher than the technical specification limit placed on the pool water temperature (TS 3.2.3).

RAI 42. Make sure to include the $3.00 insertion analysis for experiment limitation.

RRR response

A $2.00 insertion was evaluated and the value in the basis section was changed to be consistent with the results of the analysis. The following pulse insertions were evaluated using the LCC. Water temperature was set at 50 °C. A prompt negative temperature coefficient of -7.72E-5 delta k/k °C was used for these estimates. The maximum temperatures are below the requested safety limit of 1000 °C with the exception of the $2.00 insertion. However this maximum temperature is below the 1150 °C fuel temperature safety limit proposed by GA and accepted by the NRC (NUREG 1537). Table 4 contains a summary of the analysis.

Table 4 Maximum hot spot fuel temperatures for pulse insertions of reactivity

Pulse insertion [$]	Peak Power [MW]	time to Peak [msec]	Point Reactor ΔT [degrees C]	Max RRR hot spot temp [degrees C]
1.00	1.6	1773	140.2	463.9
1.33	27.2	433	207.1	661.4
1.67	102.4	253	268.2	841.7
2.00	231.5	182	327.3	1016.2

RIA 36. Demonstrate that there is no room-to-room leakage that results in an inhalation exposure to an individual in the psychology building. Confirm that the assumptions are sufficiently conservative. Link the evacuation time in the Emergency Plan to the postulated doses.

RRR response:

The reactor facility is connected to the psychology building via a single door. Additionally, there are 4 intervening spaces and a flight of stairs prior to the connecting door. Figure 3 is a floor plan of the reactor facility. The space depicted in Figure 3 is under the control of the facility director. The leak rate from the psychology building **into** the reactor facility was measured using TSI™ VelociCalc while the ventilation system was operated in normal, isolation and completely shut down. Figure 4 and Figure 5 are schematics of the reactor facility's ventilation system operating in normal and isolation modes. The ventilation system is an isolated and dedicated to the reactor facility. The measured values are summarized in Table 5.

Table 5. Summary of measured leak rate from the psychology building into the reactor facility.

Ventilation mode	Leak rate at the connecting door into the reactor facility m^3/sec
Normal	0.001-0.142
Isolation	0.093-0.033
Off	0.001- 0.124

Based on the tortuous pathway to the connecting door and the measured leak rates, the most probable dose and highest estimated dose to an individual in an unrestricted area would not occur as the result of room-to-room leakage. The highest dose estimates for an member of the population is a total effective dose equivalent (TEDE) of 0.6 mrem resulting from the fuel element failure in air, with the loss of the entire north wall as described in the attachment 1 of the May 20, 2011 response to RAI#36.

[redacted]

Abbreviation	Location	Free volume m^3
CD	Connecting door	-
B	Reactor Bay	300
CR	Control room	59
H	Exit hallway	43
C	Classroom	100
BR	Breakroom+counting lab+sump	67

[redacted]

Figure 3 Floor plan of the reactor facility

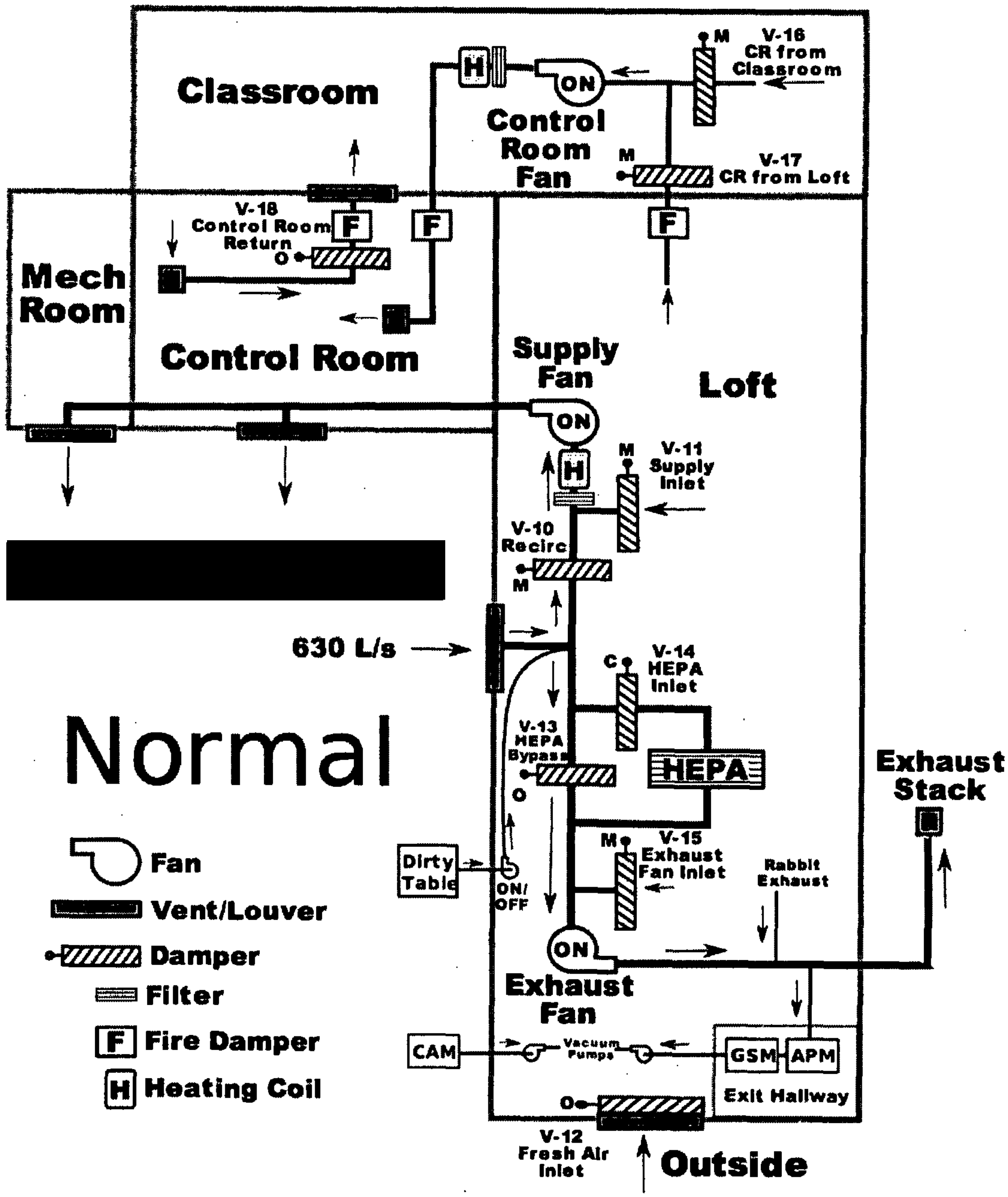


Figure 4 Ventilation System in Normal Operating Mode

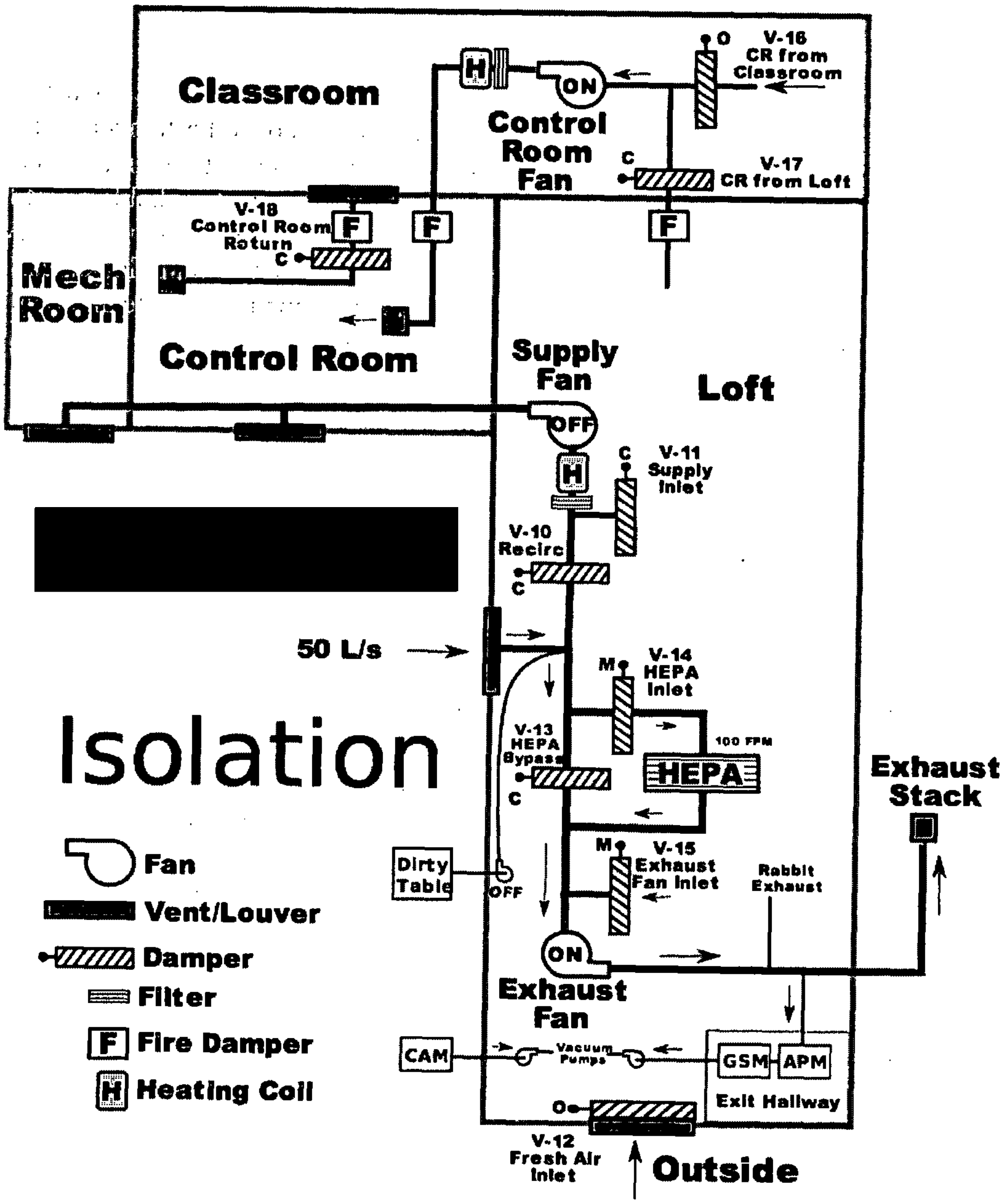


Figure 5 Ventilation System in Isolation Mode

RRR analysis Loss of coolant flow

The RRR is located in tank surrounded by 25,000 gallons of water and is cooled by natural convection. At 250 kW steady state power, the bulk pool water temperature will increase adiabatically at rate of 0.037 °C/min. The RRR is designed to operate without additional cooling capacity, specifically a heat exchanger. The most probable event for loss of coolant flow is complete or partial blockage of flow through the core. However, the core does have cross flow between channels that mitigates any small blockages. The slow rate of bulk water temperature increase creates ample time to evaluate and begin a corrective action.

10 CFR 50.59 Screen Form

62A

Screen Number: 11-01 Date: 4/26/11

Description: Replace the remaining aluminum clad fuel in the reactor with stainless steel clad fuel.

Yes	No		
___	√	1.	Does the proposed change, test, or experiment (activity) require a change to the Technical Specifications?
___	√	2.	Does the proposed activity involve a change to a structure, system, or component (SSC) that adversely affects an Updated Final Safety Analysis Report (UFSAR) described design function?
___	√	3.	Does the proposed activity involve a change to a procedure that adversely affects how an UFSAR design function is performed or controlled?
___	√	4.	Does the proposed activity involve revising or replacing an UFSAR described evaluation methodology that is used in establishing the design basis or used in the safety analysis?
___	√	5.	Does the proposed activity involve a test or experiment not described in the UFSAR where an SSC is utilized or controlled in a manner that is outside the reference bounds of the design for that SSC or is inconsistent with the analysis of descriptions in the UFSAR?
___	√	6.	Does the proposed activity present a significant hazard of another sort (fire, safety, ALARA, etc.)?

Results

[___] License amendment from NRC required ("yes" to Question 1)
[___] 50.59 Evaluation required ("yes" to Question 2, 3, 4, or 5)
[___] ROC or RSC approval required ("yes" to Question 6 or other reason)
[_√_] No further approval necessary

Attach a written justification for the conclusion.

[signature] 4/28/2011
Associate Director Signature Date

[signature] 4-29-11
Director Signature Date

Committee approval for activity (if applicable):

NA NA
ROC/RSC Chair Date

The original aluminum clad fuel in the reactor is subject to damage and leaks. Over the years we have had to remove four of the elements due to physical damage or leaks. We have occasionally received replacement stainless steel clad fuel to replace our aluminum clad fuel. As of January 2011, there were only 54 remaining aluminum clad elements in the core; the remaining 10 were newer stainless steel clad elements. The recent shipment of fuel from the University of Arizona (UA) allows us to complete the change.

The technical specifications allow either aluminum clad or stainless steel clad fuel in the reactor, without respect to location or number. The new fuel is much more durable due to the better cladding material. Each element was inspected using US DOE equipment before it was shipped to Reed, and all the elements which Reed will use passed the inspection.
Attached is the information on the UA fuel.

As a note, this analysis was performed on January 11, 2011 prior to replacing the aluminum clad fuel with stainless steel clad fuel but was not documented until now.

RRR Special Experiment 1
Fuel Loading

Submitted to ROC: [signature] Director Signature 1-26-11 Date

Approved by ROC: Juliet W Brosing ROC Chair Signature 3/10/11 Date

Object Of Experiment

This experiment describes how to perform an inverse count rate ratio (ICCR) or 1/M Plot while loading fuel elements. We plan to remove all the aluminum clad fuel elements from the core and replace them with stainless steel clad fuel elements.

Details Of Experiment

1) The actually details of moving fuel elements is covered in SOP 35.
2) The three control rods must remain fully inserted during this experiment.
3) First verify that sufficient fuel storage racks are available to unload the aluminum clad fuel elements. There are 54 aluminum clad fuel elements in the core, so 6 sets of 10-element racks will be necessary.
4) Move the aluminum elements from the core grid locations to the fuel storage racks per SOP 35.
5) Starting with only the 10 original stainless steel elements in the core, start the ICCR. This will be a plot of 1/M versus reactivity, where

   $$M = \text{Subcritical Multiplication Factor} = \frac{(Currently\,Indicated\,Power)}{(Indicated\,Power\,with\,only\,10\,fuel\,elements)}$$

   Reactivity is measured by the number of fuel elements added.
6) Data will be taken in a manner similar to Table 1 and plotted in a manner similar to Figure 1.
7) Five fuel elements will be added between each data point. The data will be used to predict the number of elements necessary to bring the reactor critical ($k_{eff} = 1$) by extrapolating from the two most recent points for the Log Channel and the Linear Channel. Since the purpose of the experiment is to **prevent** the reactor from becoming critical during fuel loading, if the plot predicts that the core will become critical during the next set of 5 elements, fuel loading shall be terminated until the situation is resolved by the director.
8) Continue loading fuel until the number of fuel elements in the core matches the original number before the experiment started, i.e., a total of 64 (10 original and 54 new ones).

RRR Special Experiment #1

Table 1 - ICCR Data

Fuel Elements in Core	Linear Channel (mW)	Linear Channel 1/M	Linear 1/M x- intercept (# FE)	Log Channel (%)	Log Channel 1/M	Log 1/M x- intercept (# FE)	Safe to proceed (SRO initials In Main Log)
10	0.05	1.00	N/A	2.00E-07	1.00	N/A	y
15	0.1	0.50	20.0	1.50E-07	1.33	-5.0	y
17	0.1	0.50	#DIV/0!	1.50E-07	1.33	#DIV/0!	y
22	0.15	0.33	32.0	1.50E-07	1.33	#DIV/0!	y
25	0.175	0.29	43.0	1.50E-07	1.33	#DIV/0!	y
32	0.2	0.25	81.0	1.50E-07	1.33	#DIV/0!	y
37	0.25	0.20	57.0	1.50E-07	1.33	#DIV/0!	y
42	0.25	0.20	#DIV/0!	2.00E-07	1.00	57.0	na
42	0.25	0.20	#DIV/0!	2.00E-07	1.00	#DIV/0!	y
47	0.3	0.17	72.0	2.00E-07	1.00	#DIV/0!	na
47	0.3	0.17	#DIV/0!	2.00E-07	1.00	#DIV/0!	y
52	0.35	0.14	82.0	2.00E-07	1.00	#DIV/0!	y
52	0.325	0.15	52.0	2.00E-07	1.00	#DIV/0!	y
57	0.45	0.11	70.0	2.00E-07	1.00	#DIV/0!	y
57	0.375	0.13	57.0	2.00E-07	1.00	#DIV/0!	na
62	0.475	0.11	80.8	2.00E-07	1.00	#DIV/0!	y
62	0.475	0.11	#DIV/0!	2.00E-07	1.00	#DIV/0!	na
66	0.55	0.09	91.3	3.00E-07	0.6666667	74.0	na -- core loaded