University of California – Irvine (UCI)
License No. R-116
Docket No. 05000326

Response to NRC Request for Additional Information (RAI)
Dated May 26, 2010

Redacted Version*

Security-Related Information Removed

*Redacted text and figures blacked out or denoted by brackets

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June 7, 2011

US Nuclear Regulatory Commission
Document Control Desk
Washington DC 20555
Attention: Linh Tran, Senior Project Manager
Francis DiMeglio, Senior Project Manager

Re: **Docket 50-326 Relicense, RAI dated May 26th 2010, TAC ME1579**

Dear Ms Tran, Mr DiMeglio:

Please find attached the final response regarding remaining items from the earliest RAI that were formerly transmitted to you in draft form. Included are copies of two reports from a contractor. THESE CONTAIN MATERIAL OF A SENSITIVE NATURE AND SHOULD BE PROTECTED from public disclosure.

I declare under penalty of perjury that the foregoing and the attached are true and correct to my knowledge.

Executed on June 7, 2011

G. E. Miller

Dr. George E. Miller

A020
NRR

University of California, Irvine Reactor License R-116, Docket 50-326
Response to NRC Request for Additional Information (RAI) dated May 26th 2010
(TAC NO. ME1579)
Dated February28th 2011.

Please note that any references to Technical Specifications apply only to PROPOSED Technical Specifications (not yet implemented), a complete copy of which is attached to this response (may be separately transmitted).
Response to three issues raised were deferred pending contract work to analyze the core.
This response addresses RAI issues3, 5, and 7 (e). A previous response dated May 26th 2010 dealt with the other RAI issues.

3. NUREG-1537 states, a thermal hydraulic analysis should be performed for the reactor. In your response dated January 27, 2010 to NRCs request for additional information (RAI) dated December 3, 2009, the analysis is provided through a reference for analyses made at two research reactors. However, the information is incomplete in that no information is provided on the similarity of the research reactors involved.
Please provide a comparison between the thermal hydraulic parameters (i.e., channel dimensions and geometry, linear power, etc.) and characteristics of the UCINRF core vs. the referenced research reactors core so as to provide validity to using this information for the UCINRF.

A contract between DOE and General Atomic has recently been completed in this regard. A copy of the full report GA 911201 will be provided for information. This will be incorporated into the facility SAR.
As anticipated and demonstrated through prior operations over many years without incident, this core design provides adequate heat removal for the standard TRIGA fuel employed. During either steady state operations, peak power dissipation and temperatures will remain considerably lower than those that have been routinely experienced safely with this fuel type elsewhere. Analyses were performed usingRELAP5 code up to 300kw (20% above licensed power). Convection cooling using available channels between elements is clearly more than adequate to maintain fuel temperatures well below any safety limits and avoid any near approach to DNB conditions.
Key parameters determined are contained in Table 2-1 which is reproduced below.

DESIGN DATA	
Number of Fuel Rods	[redacted]
Fuel Type	$UZrH_x$
Uranium Enrichment, %	19.79
Zirconium Rod Outer Diameter, mm	[redacted]
Fuel Meat Outer Diameter, mm	[redacted]
Fuel Meat Length, mm	[redacted]
Clad Thickness, mm	[redacted]
Clad Material	[redacted]
THERMAL-HYDRAULIC REACTOR PARAMETERS	
Reactor Steady State Operation, kW	250
Limited Safety System Setting, kW	275
Number of fuel elements	[redacted]
Diameter, mm (in.)	[redacted]
Length (heated), mm (in.)	[redacted]
Core total flow area, mm^2 (ft^2)	36,362 (0.3914)
Core total wetted perimeter, mm (ft.)	9416 (30.89)
Flow channel hydraulic diameter, mm (ft.)	15.45 (0.05068)
Core total heat transfer surface, m^2 (ft^2)	3.587 (38.61)
"Radial" peaking factor (rpf)	1.446
Axial peaking factor (apf)	1.352
Hot rod factor (rpf x apf)	1.955
Inlet coolant temperature, °C (°F)	25. (77)
Coolant saturation temperature, °C (°F)	114. (237)
Peak fuel temperature in average fuel element, °C (°F) [1]	214 (418)
Maximum wall temperature in hottest element, °C (°F) [1]	123 (254)
Peak fuel temperature in hottest fuel element, °C (°F) [1]	253 (488)
Core average fuel temperature, °C (°F) [1]	164 (327)
Minimum DNB ratio at 0.275 MW	7.27
Minimum DNB ratio at 0.30 MW	6.67

[1] The thermal-hydraulic parameters shown are for the reactor assumed operating at 300 kW.

5. NUREG 1537, Part 1, Section 4.5, Nuclear Design, states the applicant should discuss normal operating conditions, reactor core physics parameters and operating limits. The discussion should include a discussion of the complete, operable core; control rod worths; kinetic parameters; excess reactivities; shut down margins; and flux distribution for all planned configurations for the life of the core.
Section 4.5 of the 1999 SAR presents a representation of the flux distribution in the core. However, based on 1999 SAR Figure 4-13, this flux plot in 1999 SAR Figure 4-16 appears to not be representative in that it shows no flux peaking in the center flux trap and does not portray the flux near the position of the adjustable transient rod (which is important to the Limiting Safety System Setting (LSSS)). Please provide appropriate flux distribution information including how the distribution will affect the peak to average power ratio.

In response to this request, a contract with General Atomic was placed and fulfilled to provide a computer modeling of the UCI reactor core utilizing MCNPX code. This report, GA 91196, will be provided for information and incorporated into the facility SAR. As anticipated, as a result of burn-up and the addition of fuel elements over time, the report revises calculated values of some of the core parameters proposed at the time of initial construction. However none of the revisions affects the conclusions as to the continued safety of operations for this facility with its current core and fuel designs. Key parameters determined are summarized in Table 2-1 of that report which is reproduced below. Again, the parameters show that the fuel is operating well below any design considerations for peak power concerns, and which are demonstrated daily to be safe at higher power TRIGA facilities using similar, and even higher fuel density, fuels.
The calculated core design predicts negative temperature and void coefficients, as well as a decreasing moderator effectiveness with coolant temperature increases.

DESIGN DATA	
Number of Fuel Rods	[redacted]
Fuel Type	UZrH
Uranium Enrichment, %	19.79
Zirconium Rod Outer Diameter, mm	[redacted]
Fuel Meat Outer Diameter, mm	
Fuel Meat Length, mm	
Clad Thickness, mm	
Clad Material	
REACTOR PARAMETERS	
Reactor Steady State Operation, kW	250
Cold Clean Excess Reactivity, Δk/kβ ($)	2.82
Measured Cold Clean Excess Reactivity, Δk/kβ ($)	2.66
Prompt Fuel Temperature Coefficient of Reactivity (BOL), Δk/k-°C, 23-1000°C (x 10^{-4})	-0.70 to -1.11
Coolant Void Coefficient, Δk/k-% void, 0 - 10%, (x 10^{-4})	-7.40 to -3.68
Moderator Coefficient, Δk/k -°C, 23-1000°C, (x 10^{-4})	0.884 to 0.396
Maximum Rod Power at 250 kW, kW/element	4.519
Average Rod Power at 250 kW, kW/element	3.125
Prompt Neutron Lifetime, μsec	98.5
Effective Delayed Neutron Fraction	0.0079
ARI cold, clean core, Δk/kβ ($)	-5.88
Shutdown Margin, Δk/kβ ($) (with most reactive rod out)	-2.03
Additional Shutdown case, Δk/kβ ($) (with most reactive rod out and next most reactive rod stuck 50% out)	-1.27

In respect to specificity of the fluxes at various locations, presumably the concerns relate to the fuel temperatures that might be attained. This is best shown in the thermal report where the maximum power peaking values (>1.40) are shown in Figure 2-3 reproduced below. These occur at only 5 locations; B2, B4, C5, C6, and C7. The present location of the Instrumented Fuel Element (IFE) at B4 is calculated to be at an ideal location for assessing actual fuel temperature reached. Technical specification revisions will recognize the conclusions of the analysis in specifying locations for the IFE.
It must be emphasized however, that analyses and practice with similar core and fuel design continue to confirm that operations with this fuel type at UCI are well below safety limits and below actual operational levels at other facilities.

[redacted]

7. NUREG 1537, Part 1, Chapter 14, Technical Specifications, states the applicant needs to establish Technical Specifications that will provide reasonable assurance that the facility will function as analyzed in the SAR without endangering the environment or the health and safety of the public and the facility staff.

e) Proposed TS 3.1.5 Fuel Burnup. This statement described why there is no limitation on fuel burnup and is not a TS. Propose appropriate TS wording or justify why a TS is not needed.

While there is no clear case for establishing a limit on burn up for 8.5% by weight, stainless steel clad, 20% enriched TRIGA® fuel, this issue has been revisited with the contractor.

The contractor has directed our attention to NUREG 1282 (1987) which addressed the issue of high uranium content for TRIGA fuels. In that study the results from experiments with various formulations of TRIGA fuel (zirconium hydride alloys) were analyzed. In addition to demonstrating that increasing uranium content from 8.5 wt% up to 45 wt% had little effect, the higher uranium content fuels were irradiated in ORR to over 50% burn-up. In this process fuel swelling and growth, rod bowing, hydrogen migration, and fission product release fraction were measured. Only small percentage changes were observed. Thus for 8.5 % fuel (and others0 the contractor feels confident in stating that fuel burn-up to 50% of uranium content is an acceptable goal for TRIGA fuel. This will be incorporated as a Tech Spec limitation for the UCI reactor.