6-19-78
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MICHIGAN MEMORIAL–PHOENIX PROJECT

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Docket No. 50-2
License R-28

June 12, 1978

United States Nuclear Regulatory Commission
ATTN: Mr. Robert W. Reid
Operating Reactors Branch #4
Division of Operating Reactors
Washington, D. C. 20555

Gentlemen:

This letter provides additional information concerning the University of Michigan's reactor license amendment request to utilize aluminide and oxide fuels.

1. The answers to 13 questions posed by the Nuclear Regulatory Commission concerning aluminide and oxide fuel utilization are enclosed as RESPONSES TO QUESTIONS CONTAINED IN INFORMATION ON THE UNIVERSITY OF MICHIGAN REQUEST THAT THE FORD NUCLEAR REACTOR LICENSE NO. R-28 BE AMENDED TO UTILIZE ALUMINIDE AND OXIDE FUELS.

2. An amended SAFETY ANALYSIS is enclosed. Revisions are dated in the margins and underlined. None of the revisions are significant. Those dated 10/77 were made after a telephone discussion of the analysis with your consultant to correct minor mathematical errors. Revisions dated 4/77 were inserted after measurements were made and a peak/average flux ratio of 1.86 was calculated. A value of 1.5 had been used in the original analysis.

3. Insufficient data about UO_2 was provided in the SAFETY ANALYSIS. Since its use is not anticipated, the request to utilize UO_2 is withdrawn. The proposed revised section 5.2 of Technical Specifications should read as shown below. Changes to the present section 5.2 are underlined.

   5.2 REACTOR FUEL

   a. The fuel elements shall be of the MTR type, consisting of plates containing enriched uranium - aluminum alloy, uranium aluminide (UAl_4, UAl_3, UAl_2), or uranium oxide (U_3O_8) fuel clad with aluminum. There shall be 18 fuel plates containing 140 (± 2%) grams of uranium-235 in standard fuel elements and nine fuel plates containing 70 (± 2%) grams of uranium-235 in control rod fuel elements. Partially loaded fuel elements in which some plates do not contain uranium may be used. Elements containing uranium-aluminum alloy, uranium aluminide, and uranium oxide may be intermixed within the core.

4. A fission density limit has not been set for FNR fuel in Technical Specifications.

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If a fission density limit is required, it could be added as section 5.2.c. The limits on fission densities requested in the amendment have been reduced to a single value of 1.8×10^{21} fissions/cc for all types of fuel. This value is equal to or below operational fission density limits already in use at other reactors.

Fuel Type	Reactor	Operational Fission Density (Fissions/cc)
Uranium-Aluminum Alloy	General Electric Test Reactor (GETR)	2.0×10^{21}
Uranium Aluminide (UAl_2, UAl_3, UAl_4)	Advance Test Reactor (ATR)	1.8×10^{21}
Uranium Oxide (U_3O_8)	High Flux Isotope Reactor (HFIR)	1.9×10^{21}

The reduction has been made to expedite approval of the amendment and because it is not anticipated that a level above 1.8×10^{21} fissions/cc will ever be required at the Ford Nuclear Reactor.

c. The fission density limit shall be 1.8×10^{21} fissions/cc.

Your prompt attention to the amendment request is requested in order that we may continue reactor operation. The reactor has seven fuel elements remaining, enough to last through September, 1978. The original amendment was submitted in June, 1977.

The amendment is requested to utilize fuel types which were developed to provide an improvement over uranium - aluminum alloy fuels and which have been operationally proven in reactors which operate at much higher thermal power densities, fission densities, heat fluxes, flow rates, and temperatures than exist in the Ford Nuclear Reactor.

Sincerely,

William Kerr

William Kerr
Director

Enclosures

WK/RRB/at

Subscribed to and sworn to before me this 13th day of June 1978
at Ann Arbor, Michigan, County of Washtenaw

Donna M. Zeeb
Notary Public

DONNA M. ZEEB
Notary Public, Washtenaw County, Mich.
My Commission Expires on Oct. 21, 1981

RESPONSES TO QUESTIONS CONTAINED

IN

INFORMATION ON THE UNIVERSITY OF MICHIGAN REQUEST THAT THE FORD NUCLEAR REACTOR LICENSE NO. R-28 BE AMENDED TO UTILIZE ALUMINIDE AND OXIDE FUELS

DOCKET NO.: 50-2

June, 1978

1. What maximum stress level will the FNR elements be operated at? Compare this with the aluminide and oxide core stress limit at which plastic deformation/failure occurs and show that the strength reduction of the aluminide and oxide core is not a serious analytical consideration.

RESPONSE

Maximum stress levels in the FNR fuel elements are not known. However, FNR elements are operated under minimal thermal and hydraulic stress levels. Peak fuel plate temperature is approximately 156°F. Coolant velocity is approximately 2.5 ft/sec which is almost laminar.

Fission product and thermal swelling do not cause adjacent elements to come into contact so stresses due to these phenomena are not produced. The minimum radial clearance available to each element in the reactor core grid configuration is .029 inches which represents 0.9% of the corresponding radial dimension. Tests of 19 Advance Test Reactor (ATR) aluminide fuel element plates at fission densities up to 11.0×10^{20} fissions/cc resulted in average radial swellings of 0.13%.[1] This represents only 14% of the minimum clearance available in the FNR core. FNR elements are unrestricted in the axial direction.

The ATR observed buckling in fuel plates which had been operated to fission densities of 2.0×10^{21} fissions/cc and corresponding core swelling levels of 13%$\Delta V/V$, measured as an increase in plate thickness. The plate length change corresponding to 13% $\Delta V/V$ is approximately 0.3%.[1] Figure 1 in the amendment SAFETY ANALYSIS shows that core swelling is less than 13% $\Delta V/V$ for alloy fuel up to the requested fission density limit of 1.8×10^{21} fissions/cc. Since alloy fuel has no inherent voids, it provides the worst fission density-swelling combination possible. It is our contention that any voids will tend to reduce swelling.

In the ATR, the buckling which was observed was in the axial direction along the plate length. It occurred at a fission density of 2.0×10^{21} fissions/cc. Up to the point of buckling, no plate failures were observed.

Thus, even though strength is somewhat reduced in aluminide and oxide cores, it is not a serious analytical consideration at fission densities up to the limit requested, 1.8×10^{21} fissions/cc, and under FNR thermal/hydraulic conditions.

2. Does the phrase ". . . the same temperature . . ." in the FNR statement, "In any case, it is the disruption of the fuel clad which could result in the release of fission products, which would occur at about the same temperature for any type of fuel.", refer to 1200°F which is the melting point of the Al clad (as well as the Al in the matrix mixture)?

RESPONSE

Yes, "the same temperature" refers to the 1200 °F melting point of aluminum.

3. Provide the basis for concluding that the "curved geometry" of the FNR fuel is more stable, and; therefore, a 7% $\Delta V/V$ would not result in any failure mode.

RESPONSE

In flat plates, when buckling occurs, it is possible for the clad and the fuel core to separate and for the clad to bow in one direction while the core buckles in the opposite direction.

In curved plates, the clad and fuel core will tend to bend in the direction of curvature. The probability of clad-core separation and subsequent warping due to fission product swelling is minimized. In this sense, curved geometry is more stable than flat geometry.

The conclusion concerning curved geometry refers to section 5.1 of the amendment safety analysis. A 7% $\Delta V/V$ swelling limit was set for flat plate Engineering Test Reactor (ETR) fuel because some warping, though not failure, was periodically observed in flat test plates irradiated to the 7% $\Delta V/V$ swelling level. ATR, which has curved fuel plates, has operated up to 13% $\Delta V/V$ swelling before buckling was observed. The buckling observed was along the plate length rather than along the plate width, which is the direction of curvature. This operational experience seems to confirm the increased stability of curved geometry.

4. Provide a comparison of the thermal-hydraulic parameters for the cases with and without a 20% change in the fuel thickness (.004 inches) and show that there is no operationally significant change.

RESPONSE

A comparison of thermal-hydraulic parameters for cases with no swelling and a 20% change in fuel core thickness follows. The format is in the same order as APPENDIX A to the safety analysis. Those parameters which change are denoted by an asterisk. The peak/average flux ratio in item A.3 was changed to a value of 1.86 based on measurements made in April, 1978.

The maximum fuel plate clad temperature actually decreases from 156.2°F to 154.6°F. The reason is that for an assumed constant flow rate of 980 gpm, coolant velocity increases slightly as the net core flow area decreases. The subsequent heat transfer coefficient increases, thermal resistance decreases, and clad temperature decreases.

In any case, clad temperature change and thermal-hydraulic parameter changes are very small for a 20% change in fuel core thickness.

APPENDIX A

THERMAL-HYDRAULIC PARAMETERS

		No Swelling	20% Change In Fuel Core Thickness (.004 in)
A.1	FUEL PLATE PARAMETERS		
A.1.1	Surface Area (ft^2)	345	345
A.1.2	Fuel Meat Volume (ft^3)	.290	.347*
	Core Meat Thickness (ft)	.00167	.00200*
A.1.3	Core Volume (ft^3)	3.125	3.125
A.1.4	Core Flow Area (ft^2)	.884	.848*
	Channel Thickness (ft)	.0098	.0094*
A.2	REACTOR THERMAL POWER DENSITY (MW/l)	.0226	.0226
A.3	REACTOR FISSION DENSITY (fissions/cc)	5.44×10^{20}	4.55×10^{20}*
	Peak/Average Flux Ratio	1.86**	1.86**
A.4	FISSILE MATERIAL DENSITY (gm/cc)	.395	.395 (Fresh Fuel)
A.5	REACTOR HEAT FLUX (BTU/hr ft^2)	3.68×10^4	3.68×10^4
A.6	COOLANT VELOCITY (ft/hr)	8,919	9,298*
A.7	REYNOLDS NUMBER	9.05×10^3	9.03×10^3*
	Characteristic Dimension (ft)	.0196	.0188*
A.8	PRANDTL NUMBER	3.15	3.15
A.9	NUSSELT NUMBER	53	53
A.10	HEAT TRANSFER COEFFICIENT (BTU/hr ft^2 ^oF)	1022	1063*
A.11	THERMAL RESISTANCE (hr-ft^2-^oF/BTU)	9.8×10^{-4}	9.4×10^{-4}
A.12	MAXIMUM FUE PLATE CLAD TEMPERATURE (^oF)	156.0	154.6*

*Parameter changes caused by 20% change in fuel core thickness.

**Measured value, April, 1978.

5. On page 10, FNR stated that 19 ATR irradiated fuel plates had an average length change of .22% caused by fission product swelling. For a 2-foot long fuel plate, this change would amount to an increased plate length of approximately .050". Provide the following information: (a) provide an analysis which would demonstrate the absence of any plate buckling. (b) if this analysis demonstrates there is buckling what is the maximum expected channel closure for a plate length increase of .050"? (c) what would the thermal-hydraulic consequences be in such an eventuality? (d) what would be the significant operational considerations for such a change?

RESPONSE

(a) Experimental test results at ATR showed that buckling did not occur below fission densities of 2.0×10^{21} fissions/cc and corresponding swelling levels of 13% $\Delta V/V$. More recent operating experience is that no buckling has occurred during operation up to fission densities of 2.2×10^{21} fissions/cc which corresponds to a swelling level of approximately 14% $\Delta V/V$.[3]

6. It is noted that the fission density (FD) at which the .22% length change was observed was in the 7 to 11 X 10^{20} fiss/cc range. FNR is proposing a FD limit of 20 X 10^{20} fiss/cc. Provide the following information: (a) what would the expected length change be for this FD? (b) how would this amplify the buckling concern? (c) What would the thermal-hydraulic and operational consequences be from this amplified condition?

RESPONSE

(a) Similar to swelling, length change is a linear function of fission density. Assuming that the average length change of 0.22% corresponds to a fission density of 9.0 X 10^{20}, a fission density of 2.0 X 10^{21} would produce a length change of 0.49%.

(b) Based upon information contained in the response to question 5, buckling would not be expected.

(c) No significant thermal-hydraulic consequences would be expected.

7. What is the void fraction of the aluminide fuel FNR intends to use? How does this compare with the void fraction of fuel whose irradiation test data base FNR uses to support their analysis?

RESPONSE

Fuel specifications were prepared in co-operation with ATR management. Void fraction is not specified. ATR management does not specify void fraction for its fuel. A normal manufacturing void fraction of 3-4% occurs. As fuel loading increases, void fraction increases.

The fuel whose irradiation tests form the basis for the FNR amendment request was manufactured without a specified void content.

It should be pointed out that alloy fuel has no voids so any voids in aluminide or oxide fuels are a bonus.

8. On page 10 pertaining to the XA 003F plate swelling data that is plotted in Figure 6, what is the explanation for sample 3-3's swelling exceeding the theoretically predicted value of 6.38% $\Delta V/V$ per 10^{21} fission/cc?

RESPONSE

The report, reference 16 in the amendment safety analysis, does not provide a specific explanation for the behavior of sample 3-3. It does make the general statements, "It is evident, by comparing the theoretical potential for solid fission product growth with the results, that the accomodation of solid fission products by fabrication voids has not occurred to a great extent. This fact is further evidenced by copious, large voids observed metallographically during post irradiation examination. The small extent of voidage utilization is probably the result of a relatively low operating temperature."[1]

It has been previously stated that FNR has not specified void content.

9. What are the differences in chemical environment between the FNR and those reactors in which the irradiation tests were performed that FNR uses in their analysis? If there are differences, explain why these differences would not enhance conditions that could cause fialures of those plates containing aluminide or oxide fuels under the FNR operating conditions.

RESPONSE

Chemical conditions in the reactors in which the irradiation tests utilized in the safety analysis were performed are more severe than FNR chemical conditions due to higher temperatures and flow rates.

In general, each reactor utilizes demineralized water as coolant. pH is maintained in the 5-7 range to minimize corrosion. No additional chemical controls are maintained. Table 1 of the safety analysis provides a comparison of parameters such as temperatures and flow rates.

10. Since the operation of FNR, have there been any fuel plate failures related to reactor operation? If so, what was the cause of the failures?

RESPONSE

There has been one pinhole leak in one FNR fuel plate. The pinhole was a manufacturing defect.

11. The FNR analysis is primarily based on fuel performance data from irradiation tests and reactor operation. Describe how FNR will assure that the fuel to be used in their reactor will have specifications comparable to the fuel used in the tests and other operating reactors.

RESPONSE

FNR is presently in the process of having aluminide fuel manufactured by Atomics International (AI). Atomics International manufactures ATR fuel. FNR fuel drawings and specifications were prepared by EG and G under DOE authority at the National Reactor Test Site (NRTS), Idaho Falls, Idaho. The specifications were written by engineers and quality assurance personnel associated with ATR.

AI, EG and G, and FNR personnel met for the final approval of the drawings and specifications. All revisions must be approved by EG and G.

If oxide fuel were used at some future time, a similar situation would be set up with personnel from Brookhaven National Laboratory.

12. Provide the basis from which it can be reasonably assumed that powder metallurgy UO_2 would have nearly the same swelling characteristics as U_3O_8.

RESPONSE

Sufficient data on the swelling characteristics of UO_2 was not provided. In addition, the heat transfer characteristics of UO_2 are much poorer than those of UAl_x and U_3O_8.

Since the utilization of UO_2 fuel is not anticipated, the request for its use is withdrawn.

13. What is the fission density limit for alloy (U-Al) at the FNR?

RESPONSE

A fission density limit for alloy fuel at the FNR is not specified.

REFERENCES

1. M. Graber and R. Hobbins, "Irradiation Testing of Sample Fuel Plates to Very High Burnups", INC-16-I, U. S. Atomic Energy Commission Report ANCR-1016, October, 1971.

2. J. H. Crawford and M. C. Wittels, "Radiation Stability of Nonmetals and Ceramics", Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958, Vol. 5, United Nations, Geneva, 1958, pp. 300-310.

3. Telecon with W. C. Francis, E. G. and G., NRTS, Idaho Falls, Idaho, May, 1978.