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1.25% CO-PRODUCER FUEL ELEMENT TEST

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HAZARDS REVIEW - N-REACTOR

1.25% CO-PRODUCER FUEL ELEMENT TEST

by

N. R. Miller
and
W. S. Nechodom

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HAZARDS REVIEW - N-REACTOR
1.25% CO-PRODUCER FUEL ELEMENT TEST

I. INTRODUCTION

The N-Reactor Hazard Summary Report⁽¹⁾ examines the hazard from operating the N-Reactor with a uniform fuel loading enriched to 0.947% U²³⁵.

Incentives have been developed for reactor testing of a block of 49 tubes loaded with co-producer elements, i.e. elements capable of producing both weapons grade plutonium and tritium.⁽²⁾ The element utilizes an outer fuel tube enriched to 1.25% U²³⁵ with an inner target lithium-aluminum rod.⁽²⁾ Criteria have been developed to guide the evaluation of safety aspects of such tests.⁽³⁾ It is the purpose of this document to review the hazards associated with the proposed test and to set forth special precautions which will be necessary to maintain a high level of safety.

II. SUMMARY

The introduction of a 49-tube co-producer block will have no significant impact upon the satisfaction of the total control or speed of control criteria.

A review of the thermal-hydraulic aspects of the special fuel load shows that the test column will operate at about the same tube powers as normal columns but will receive about nine per cent more flow. Hence, the test columns will operate at a net lower outlet temperature than adjacent normal columns. Further, it is expected that the two coolant channels in the test columns will operate at about the same outlet temperature.

Because of these thermal-hydraulic characteristics, insurance of adequate cooling* will not require the imposition of special, stringent limits. Specifically, flow monitor protection will be provided to produce a reactor scram before coolant boiling would initiate as a result of:

- a. an obstruction in either flow annulus,
- b. an obstruction upstream or downstream of the fuel charge, or
- c. a leak upstream of the fuel charge.

This "trip-before-boiling" philosophy should provide entirely adequate assurance of single-tube flow protection and may well represent a more conservative than necessary approach to providing this protection.

* Adequate cooling is defined as a minimum the cooling required to prevent melting.

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Calculations have been made of the burnout margin available when a tube containing the test fuel is operating at "hot channel" tube powers, at coolant flows reduced to the probable flow trip, and with the bulk boiling point suppression at the probable trip point. Results show that the burnout flux under these conditions would, conservatively, be a factor of two higher than the actual heat flux. This is considered a more than adequate burnout margin.

If a "dump" of the primary coolant system should occur, the heat output of the special test columns during the transient would be about the same as normal columns. Inasmuch as the test columns will offer a lower hydraulic resistance than normal columns, the special columns should receive adequate cooling under any conditions where normal columns are adequately cooled.

In the very remote event that an accident would result in fuel melting, the co-product driver elements would receive the same confinement as the standard 0.947% enriched elements, provided only that the melting time of the 1.25% enriched driver is not so short as to permit release of fission products prior to resealing of the steam vents.

Very conservative calculations indicate that the driver element will have an acceptably long melting time upon complete loss of coolant up to tube power levels of at least 4000 KW. We believe that refined calculations, presently underway, will show that the co-product test load will not limit reactor power below design levels.

The consequences of an accident leading to core meltdown would not be significantly altered by the presence of the test load. Fission product release from the driver element would be of the same order as obtained from normal fuel. Tritium in the test load would contribute about 455,000 curies. Of this, most would be expected to react with Zircaloy-2 components and be retained in the reactor core. The tritium eventually released to the environs should not appreciably add to the severity of such a meltdown accident.

III. DISCUSSION

1. General

The test of the co-producer element will involve the irradiation of a group of test columns in a block to determine product yield data and fuel performance.⁽²⁾ The size of the block may range up to 49 columns. The block will be located in the lower, central part of the reactor, i.e. in the 8th through 14th horizontal tube rows (out of 33 total) half-way between the left and right sides.

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The use of fuel with normal 0.94% U²³⁵ inner fuel tubes and 1.25% U²³⁵ outer fuel tubes for spike enrichment has been previously reviewed. It is possible that both spike material and this co-producer test will be under irradiation at the same time. This possibility was recognized in the test criteria which require that compliance be ensured by specific study of the reactor condition with all such tests in plan.

The test fuel load will consist of an "outer" driver element and an "inner" target rod. The driver element utilizes a metallic uranium core enriched to 1.25% U²³⁵. The driver is coextruded in Zircaloy-2 cladding and utilizes the standard brazed end closure.

The target element for most of the load will utilize a lithium-aluminum alloy containing two per cent or less lithium enriched in Li⁶ so that the alloy contains 0.38 per cent Li⁶. The target core is coextrusion clad (unbonded) in high purity aluminum. Enclosure in aluminum is effected by welding aluminum end caps into the recessed ends of the assembly. The aluminum clad target is then canned in Zircaloy-2 with closure effected by welding Zircaloy-2 end caps into place.

In addition, the testing of a few columns utilizing an alternate target material is authorized subject to the limitation that the Li⁶ content will be the same as described and that the target pieces will have the same outer dimensions.

Physical dimensions of the assembly are as follows:

TABLE I

PHYSICAL DIMENSIONS OF TEST FUEL
(All dimensions are in inches)

Driver Element

Outer Diameter	2.292
Inner Diameter	1.441
Outer Clad Thickness	0.035
Inner Clad Thickness	0.025
Length	26.10

Target Element

Diameter	0.565
Zircaloy Clad Thickness	0.030
Aluminum Clad Thickness	0.035
Zircaloy Cap Thickness	0.250
Aluminum Cap Thickness	0.188

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The target element will be locked in the driver tube in much the same manner as normal inner fuel tubes are locked in outer fuel tubes.

2. Total Control

2.1 Criterion

Each of the two independent safety control systems must have adequate control strength to maintain the reactor subcritical under all normal operating conditions and credible accident conditions.

2.2 Compliance with Criterion

The reactivity behavior of the co-product element is, by design, very similar to that of the base case, 0.94%, fuel load. The co-product element is designed to provide the same equilibrium operating reactivity, at average exposure, as the 0.947% fuel. Differences in reactivity behavior occur only during transient conditions, due to differences in temperature coefficients of reactivity and the reactivity transients associated with the burn-out and buildup of fissile materials. The calculated reactivity behavior of the co-product fuel, the base 0.947% fuel, and the proposed mixed loading, under normal and transient conditions, is tabulated in Table II for unexposed fuel, which provides the most severe reactivity transients.

TABLE II

REACTIVITY PARAMETERS - BASE LOAD AND
CO-PRODUCT FUEL PILE REACTIVITY

	<u>Base Load</u>	<u>Reactivity with</u> <u>49-Tube</u> <u>Co-Product Block</u>
Equilibrium, K_{ex}	4.8 mk	5.5 mk
Turnaround, Xenon-Free	34.2 mk	36 mk
Turnaround, Plus Loss of Primary Loop Coolant	-22.5 mk	-17.6 mk
Turnaround, Plus Cold Water Injection	55.2 mk	56 mk
Horizontal Rod System Strength	67 mk	67 mk
Ball System Strength	64 mk	64 mk
Total Safety System Strength	115 mk	115 mk

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As can be seen from Table II, the behavior of the reactor when charged with the co-product block is, under all normal conditions and under cold water injection accident conditions, almost identical to the behavior of the base case. The slight reactivity increase (1-2 mk) observed is due to the assumption of unexposed fuel; exposure of the co-product fuel leads to a greater reactivity decrease than exposure of the base fuel, and the reactivity increase due to the presence of the block becomes a decrease when normal average fuel exposures are considered.

The lithium-aluminum target rod in the co-product fuel element melts at 600-625 C. This temperature will not be approached under normal operating conditions. To achieve these temperatures, it is necessary to postulate total loss of the primary loop coolant supply, including a failure of the last-ditch, emergency raw water system. Should this highly improbable event occur, calculations indicate that uranium melting temperatures will be exceeded in the inner tubes of the normal, 0.947%, tube-in-tube fuel elements, and fission product release would occur. With the co-product test block charged, the tritium burden contained in the target elements would also be released, and the lithium-aluminum alloy would be redistributed in the process tubes. If this alloy merely slumps to the bottom of the process tube, it would result in reactivity decrease; if it were somehow removed from the reactor core, the reactivity of the co-product fuel would increase, but the over-all pile reactivity would still be negative because of the reactivity loss associated with coolant loss in the base, 0.947%, fuel.

All of the data in Table II were calculated on the same basis; i.e., using the same basic computational technique. The absolute accuracy of the method is indicated, by comparison to the results of zero-power physics testing, to be excellent, predicting values within 1-2 mk for the turnaround reactivity and the cold water injection reactivity in the base case. The hazards associated with the charging of the co-product block, from a total reactivity control standpoint, are, therefore, not considered to be different than those associated with operation with the base, 0.947%, loading.

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It is probable that the reactor base load will include some columns of fuel enriched to 1.25% U²³⁵ to provide adequate operating reactivity. The value given in Table II for the base load equilibrium k_{ex} is for unexposed fuel; the reactivity transients associated with fuel exposure, primarily the buildup of samarium, will cause this value to decrease. To restore adequate operating reactivity, added enrichment would be charged. The effect of charging fuel columns enriched to 1.25% U²³⁵ would be to restore the reactor to the base load conditions shown, with only minor deviations resulting from differences in temperature coefficients of reactivity between 0.947% fuel and 1.25% fuel. Since the temperature coefficients of reactivity for 1.25% spike enrichment are calculated to be smaller than those for 0.947% fuel, the base case turnaround reactivity and the cold water injection case reactivity, would both be less than that indicated, provided that the equilibrium reactivity is maintained at about 5 mk.

3. Speed of Control

3.1 Criterion

The primary (horizontal rod) safety system must respond fast enough to prevent the consequences of any accident from being made worse by a nuclear excursion caused by that accident. If an accident, in itself, could not damage the reactor, the rod safety system must limit the consequences of the accident to mild fuel stressing.

The secondary (ball) safety system must respond fast enough to prevent fuel melting under all credible accident conditions where melting would not result from the accident itself.

3.2 Compliance with Criterion

Examination of the effects of the test loading on control system strengths and on reactor kinetics show that the net effect of the addition of the co-product block loading on the reactor kinetics will be to slightly decrease the power attained during an excursion, but at the expense of reaching higher uranium temperature in the co-product driver element. These findings are elaborated below.

3.2.1 Safety Control System Strengths

The strength of the safety control systems is very nearly proportional to the ratio of the migration areas when two separate cases are considered. The calculated migration areas in the hot, operating lattices for the base loading lattice and the co-product test lattice are 643 and 610, respectively. When these migration areas are weighed for the effective fractions of the reactor containing the base load and co-product load, an average migration area of approximately 641 is obtained, indicating a reduction in safety system strength of .2 mk, which is not significant. Hence, the speed of reactivity reduction will be largely unaffected by the presence of the test load.

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3.2.2 Prompt Temperature Coefficients of Reactivity

Previous kinetic studies of startup power excursions with the base fuel load indicated that the uranium temperature coefficient of reactivity (doppler) contributes more than 80 per cent of the total "prompt" negative reactivity effect associated with heatup of the fuel and coolant during a startup excursion, up to the point where the power excursion is terminated. The doppler coefficient of reactivity ($\Delta K/^\circ F$) of the co-product fuel is approximately 90 per cent as large as that in the base fuel load; however, the temperature rise in the co-product fuel element is calculated to be about 1.8 times as much as the temperature rise in the base fuel load at equivalent power levels. Thus, the negative reactivity effect of fuel heating during an excursion will be greater, for excursions to equivalent powers. While the temperatures attained in the co-product block would be greater than those in the base fuel load, they would not approach the point of uranium melting during excursions considered credible.

3.2.3 Neutron Kinetics

The delayed neutron fraction and the prompt neutron lifetime of the co-product lattice are not significantly different from those of the base fuel load. The delayed neutron fraction of both lattices will become smaller as exposure increases, due to plutonium buildup; however, this effect is more significant in the base fuel load, and was considered in the original kinetics calculations.

3.2.4 Fuel Melting Temperatures

The co-product fuel element includes a lithium-aluminum target rod with a melting point of about 600-625 C. This is significantly lower than the minimum melting point of the base fuel load (uranium: 1133 C). However, the operating temperature of the target element is within a few degrees of the coolant temperature because of the low heat generation rate, and melting in the presence of coolant flow is precluded. Hence, the lower target melting temperatures do not affect speed-of-control factors.

4. Ex-Reactor Nuclear Safety

4.1 Criterion

Each type of test fuel must be studied to determine its ex-reactor criticality parameters. Once these parameters are known, safe handling and storage procedures must be provided in writing.

4.2 Compliance with Criterion

Criticality parameters have been provided for uranium enriched to 1.25% in tubular geometry. Criticality parameters are not available for the driver-target assembly; no experimental data exists on this geometry, and calculations in light-water-moderated arrays do not appear dependable. Critical mass limits for the co-product elements are, therefore, based on

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the criticality parameters of the driver element alone, allowing the presence of the target rod to be a significant safety factor, but one not known quantitatively.

TABLE III
CRITICALITY PARAMETERS
(1.25% Co-Product Driver Elements)

	<u>Safe Mass</u> (lbs.)	<u>Safe Slab</u> <u>Thickness</u> (inches)	<u>Safe Mass Per</u> <u>Unit Area</u> (lbs/ft ²)
Unirradiated	717	8.9	179
Irradiated	690	8.3	177

A 26-inch driver element contains 38.4 pounds of uranium; a column of 16 pieces contains 615 pounds of uranium. It will, therefore, be necessary to handle each column of fuel as an isolated batch, separated by three feet from all other fissile material meeting the safe mass limit or to store the unirradiated fuel in shipping boxes, with a base area of total loading which will meet the mass-per-unit-area limit. For example, shipping boxes with a base area of 9 square feet, with 42 or less pieces per box with stacking prohibited would be acceptable. When the shipping boxes are loaded to comply with the mass-per-unit-area limit, they may be stored adjacent to each other in arrays of unlimited extent. Once the co-product fuel elements are assembled into columns and placed into mono-tubes, the mono-tubes may be stored on the wall racks in compliance with the mass-per-unit-area rule; an array of mono-tubes 5 wide, spaced 9 inches between sets of 5, yields a mass-per-unit-area of 118 lb/ft².

Mono-tubes containing co-product fuel will be specially marked to prevent confusion with normal fuel charges. However, the accidental charging of co-product fuel instead of normal fuel does not present a significant nuclear hazard, inasmuch as the in-pile reactivity behavior of the co-product fuel would be no more severe than indicated in Sections 2 and 3 above.

Discharge of co-product fuel will be administratively restricted to single columns, which approach the safe mass limit in fuel weight. The safe mass limit is 45 per cent of the minimum critical mass in spherical geometry, so that an inadvertent double discharge would not result in a critical array. Further, the shape of the discharge chute precludes a spherical array; the minimum critical mass, when constrained to the chute shape, would be about 2230 pounds, or about 3-1/2 columns. In addition, the presence of the target rods provides an additional, but

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unknown safety factor. To prevent accidental discharge of the co-product fuel under 0.947% fuel limits, the rear nozzle cap bolts will be locked with keys or clips to prevent inadvertent removal through mis-identification

Once discharged, the co-product fuel must be segregated and stored under one of the limits given in Table III, using administrative procedures. Should these procedures fail, it would physically be possible to store the fuel in an array of up to 246 pounds per square foot in the storage basin (in batches of up to 1075 pounds which are separated by a minimum of 8 inches). This array would not be critical, since the batches would be less than the critical mass in spherical geometry, would not be spherical, and the 8 inch separation yields essentially infinite separation between batches.

5. Thermal Hydraulic Considerations

5.1 Criterion

The thermal hydraulic characteristics of the test fuel columns must be studied to determine safe single-tube operating limits. Specifically, there must be no boiling with net steam quality at any location within the process tube during normal operation. During all normal periods of operation, it is necessary that adequate cooling be provided to the test columns. Under accident conditions, inadequate cooling shall not occur in the test columns under conditions where normal columns are receiving adequate cooling. Adequate cooling is defined as a minimum as that required to prevent melting any of the contents of the process tube in question.

5.2 Thermal Hydraulic Characteristics

The thermal hydraulic characteristics of a tube containing the co-producer material compared to an adjacent tube containing regular fuel both at goal exposures have been calculated⁽⁴⁾ using the Flex-2 computer code and are shown in Table IV.

With green fuel, assuming the same base case conditions for a tube loaded with normal production material, a co-producer charge would be expected to operate at a power of about 4490 kw, with outlet temperatures of about 515 F.

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TABLE IV
COMPARISON OF THERMAL-HYDRAULIC CHARACTERISTICS
OF NORMAL AND CO-PRODUCER FUEL

(Goal Exposures)

<u>Parameter</u>	<u>Normal Fuel Load</u> <u>Base Case</u>	<u>Co-Producer</u> <u>Load</u>
Flow Rate, lb/hr	84,800	92,440
Tube Power, kw	4,349	4,406
Inlet Temperature, °F	367	367
Tube Bulk Outlet Temperature, °F	522	512
Heat to Outer Channel, kw	1,870	2,652
Heat to Middle Channel, kw	2,070	-----
Heat to Inner Channel, kw	410	1,754
Flow to Outer Channel, lb/hr	38,300	55,550
Flow to Middle Channel, lb/hr	39,700	-----
Flow to Inner Channel, lb/hr	6,800	36,890
Outlet Temperature, Outer Channel, °F	515	512
Outlet Temperature, Middle Channel, °F	525	-----
Outlet Temperature, Inner Channel, °F	547	511

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5.3 Compliance with Criterion

5.3.1 Accident Conditions

The criterion requires that under accident conditions, inadequate cooling shall not occur in test columns under conditions where normal columns are receiving adequate cooling. Accidents considered for the normal fuel load in the Hazards Summary Report include:

- a. Plugging events affecting individual flow channels in a single process tube.
- b. Process tube leaks.
- c. Reduction of flow to a single tube, e.g., reduction caused by a leak in an inlet connector or plugging upstream or downstream of the fuel charge.
- d. Events affecting large numbers of process tubes.

These potential accidents are covered in the following sections.

5.3.1.1 Plugging of Individual Flow Channels

Plugging or partial plugging of an individual flow channel could result from a fuel rupture or from foreign material entering the process tube channel. If such plugging occurs, flows in the affected channel would decrease, flow in the open channel would increase, but the total coolant flow in the tube would be reduced.

If the restriction became severe enough, boiling in the plugged channel would occur. As coolant temperatures in the affected channel increased, the percentage of heat transferred to the other channel would be increased and eventually, as the heat removal capability of the affected channel deteriorated, the shift of heat could produce boiling in the second channel.

It is important that in this sequence, either:

- a) The condition be detectable and an automatic flow trip scram the reactor before flow instability occurs, or
- b) If this condition is not detected until instability occurs (An unstable condition would result in a large flow loss which certainly would be detectable), it must be demonstrated that melting temperatures would not be attained during the time flow is unstable.

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It is planned to set flow trips so that a reactor scram would occur before plugging could induce boiling in either channel. This is a conservative way to implement the first alternative.

5.3.1.2 Process Tube Leaks

The Hazards Summary Report considers the effects of leaks in the active zone section of the Zircaloy-2 process tube. Such leaks divert flow from the outer coolant channel to the break, with a reduction in cooling of the outer fuel surface downstream of the failure.

While the precise chain of events has not been evaluated for either the normal fuel load or a co-producer load, the following factors are pertinent and serve to put the accident into perspective.

- a) It is highly improbable that a leak would occur of a size large enough to produce a serious reduction in cooling of the outer fuel surface, yet at the same time be small enough to prevent a high-flow trip.
- b) Even if such a leak did occur, the hydraulic transients from boiling in the outer channel and, perhaps later in the inner channel, would be expected to cause a low-flow trip.
- c) Leak of water into the reactor core at a significant rate should be detected as a gas atmosphere pressure increase, and would signal a manual reactor scram.

Hence, while not treated in a quantitative fashion, it would appear that the hazards associated with a tube leak in the active zone part of the tube are of the same order as those already deemed acceptable with normal fuel loadings.

5.3.1.3 Reduction of Flow to a Single Tube

It is possible that plugging could occur upstream or downstream of the fuel charge or a leak would occur upstream of the fuel charge reducing flow to both coolant channels.

Protection to be provided for the case of plugging of either channel will serve to detect a general loss of flow to the tube due to plugging well before boiling in the tube is initiated. A leak occurring upstream of the fuel charge would produce analagous results except that in this instance a high-flow trip would occur should the leak be enough to cause tube effluent coolant temperatures to approach the boiling point.

5.3.1.4 Events Affecting Large Numbers of Tubes

The consequences of primary coolant line failures and actuation of the emergency raw water system was covered in the Hazards Summary

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Report and in later studies. In an accident of this type, the primary coolant would be expelled from the primary coolant system through the dump valves, and through the line failure if one had occurred. Reactor cooling would then be provided by the diesel-powered emergency coolant system. Heat generation rates following the accident and, hence, cooling requirements are essentially the same in both cases. However, the tube containing the test loads offer reduced hydraulic resistance and should receive adequate cooling without significantly diverting flow from adjacent normal tubes. It is concluded that tubes containing the co-producer material should behave in about the same fashion as tubes containing normal fuel.

5.3.2 Normal Operating Conditions

The thermal hydraulic conditions occurring in the process tube during normal, non-accident, conditions have been determined.

Figure 1 shows coolant and surface temperatures along the fuel column for a charge operating at "hot channel" powers and design coolant flow rates. Figure 2 is a similar figure, this time assuming a tube flow reduced to near low-flow trip conditions. Also shown are saturation temperatures assuming two different tube outlet pressures. The point labeled "Ts" show the required surface temperatures required for the initiation of nucleate surface boiling. It is noted that at conditions approaching a reactor scram point, nucleate boiling would be expected on the inner surfaces over a fraction of the fuel elements in a charge operating at design hot channel powers. Nucleate boiling is allowed for normal tubes when the scram point is approached.

In all cases, there remains a considerable margin of boiling point suppression for the bulk coolant flow.

An analysis was also made of the "burnout margin" at trip conditions. Results are shown in Figures 3 and 4 where actual and burnout heat fluxes are depicted along the fuel columns. The burnout flux is shown to increase up to a point upstream from the rear of the fuel charge. The horizontal representation of the burnout flux for regions upstream of that point represents a conservative estimate in regions where experimental data are lacking. The ratio of burnout flux to actual flux for the two surfaces is shown in Figures 5 and 6. The minimum ratio is about two, greater than is normally considered necessary in assessment of cooling requirements.

6. Confinement

6.1 Criterion

The operating principle of the confinement system is that the relatively non-contaminated steam would be released to the atmosphere if a primary loop rupture occurs. The confiner would then be resealed to provide

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a filtered release should the emergency coolant system also fail. It is thus necessary that resealing be effected prior to the melting of fission product-containing fuel or release of any biologically significant quantities of product material. The presence of the test material must not create conditions beyond the ability of the confinement system to fulfill this function.

6.2 Compliance with Criteria

6.2.1 Elapsed Time to Melting Following Accident

If a large coolant line in the primary coolant system failed during reactor equilibrium operation, and if the accident were compounded by failure of the emergency raw water cooling system, a fraction of the test load, including both target and driver elements, would melt.

In such an accident it is important that fission products not be released to the confiner until the venting of steam from the confiner has been completed and a filtered release established.

Failures of coolant lines on the inlet side of the reactor would produce the shortest elapse of time to fuel melting because such failures could interrupt all reactor cooling a very short time (few seconds) after the accident occurred. Calculations show that following such accidents as long as five minutes might be required before a filtered release could be established. It is important, therefore, that melting of the driver elements not occur in a much lesser time. (Some time would be required for transport of fission products from the reactor core to the line rupture).

The entire question of fuel melting times for both normal and test loads is in the process of being re-examined. This re-evaluation will use data obtained from reactor operating experience, and test results from thermal-hydraulic studies. These advances in turn justify a more exact analysis of the complex heat transfer process from the fuel to the tube to the ultimate graphite cooling system sink.

In the meantime, results are available from a less exact but very conservative calculation of melting times for two tube powers based on techniques similar to those used for the normal fuel elements. These results are shown on Figure 7. A straight-line extrapolation shows that tube powers would need to be restricted to about 4000 KW in order to ensure a five minute delay in the start of driver element melting. Actually, a straight-line extrapolation is conservative inasmuch as melting would be avoided altogether at some reduced tube power, probably in the range of 2000 to 4000 KW and the curve would actually be concave. Hence, the 4000 KW limit is very conservative.

This 4000 KW tube power limit for this test load will be established and adhered to until a firm basis for change is derived from the re-evaluation just discussed.

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The target element will melt after a shorter time delay, perhaps as soon as 100 seconds following the accident for 4000 KW tube powers. However, as discussed in the next section, the tritium inventory available for release would be relatively modest and, further, there would be no particular benefit from passing tritium through the filters. Hence, the earlier target melting times is not of much significance for the view point of protection of the environs.

6.2.2 Significance of Test Load Meltdown

An accident which would involve meltdown of the test load would almost certainly cause melting of some fraction of the normal load. The differences in consequences of such a meltdown would mainly arise from the tritium released from the target material. The quantity of fission products released during the accident would not be significantly altered by the presence of the driver elements. For this reason, the balance of the discussion is centered on the potential tritium release.

The test of the co-producer material envisions the irradiation of up to 49 columns of test material to average exposure of 1100 MWD/T. It is anticipated that this will result in the production of about 45 grams of tritium or about 455,000 curies.

The lithium-aluminum material is coextrusion clad (unbonded) in aluminum. The aluminum clad target is then canned in Zircaloy-2. If, following an accident, the target and aluminum cladding melted, there would be an opportunity for the tritium product to react with the Zircaloy-2 and such a reaction would be expected at these temperatures.

The weight of Zircaloy-2 used for encasing the target material in the 49-tube test load will be about 225 pounds. Hence, if the entire tritium inventory reacted with the Zircaloy components, the resulting hydride content would be about 440 ppm. It is not certain how much tritium would react with the Zircaloy-2 target cladding; however, it does appear that the Zircaloy components could represent a "sink" for most of the total tritium inventory.

It is quite probable that the Zircaloy cladding would fail and unreacted tritium would be released into the flow channel. Further reaction with other Zircaloy-2 components could occur, but the release of some fraction of the tritium inventory to the confiner would be expected.

The 455,000 curies of tritium, though not negligible, would not add significantly to the consequence of accidents which would lead to its release.

7. Acknowledgement

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FIGURE 1

FIGURE 2

FIGURE 3

BURNOUT RATIO DIAGRAM FOR INNER CHANNEL

* Tube power higher because of power in other lattice materials.

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FIGURE 4

BURNOUT RATIO DIAGRAM FOR OUTER CHANNEL

* Tube power higher because of power generated in other lattice materials.

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FIGURE 5

BURNOUT RATIO FOR OUTER CHANNEL

FIGURE 6

BURNOUT RATIO - INNER CHANNEL

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FIGURE 7

MELTING TIME - 1.25 CO-PRODUCER DRIVER ELEMENT

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