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January 14, 1999

Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Request for Amendment of Facility Operating License No. R-37 for the Massachusetts Institute of Technology Research Reactor (MITR); Docket No. 50-20

Dear Sir or Madam:

On 3 October 1997, the Massachusetts Institute of Technology submitted an amendment request for the operation of a fission converter in conjunction with our research on neutron capture therapy. That submittal included a safety evaluation report, a proposed technical specification (No. 6.6), as well as minor charges to other technical specifications. On 20 July 1998, the Massachusetts Institute of Technology received a request for additional information relative to the amendment request. Enclosed is our response to that request together with the revised safety evaluation report (SER) and the revised wording to proposed MITR Technical Specification No. 6.6.

Your earliest attention to this request would be most appreciated. Correspondence concerning this request should be directed to Dr. Bernard.

Sincerely,

Lin Wen Hu, Ph.D.
Relicensing Engineer
MIT Nuclear Reactor Laboratory

John A. Bernard, Ph.D.
Director
MIT Nuclear Reactor Laboratory

JAB/lwh

Enclosure

cc: USNRC - Senior Project Manager,
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P. Busse, BIDMC Clinical Principal Investigator
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MITRSC

A020/1

**Response to Request for Additional Information**
**Massachusetts Institute of Technology**
**Docket No. 50-20; License No. R-37**
**(TAC No. M99844)**

1. The fission converter based epithermal neutron irradiation facility has been designed so that it can operate with $H_2O$ or $D_2O$ or even with a mixture of coolants. The $k_{eff}$ using either $H_2O$ or $D_2O$ is much less than 1.0, as shown in Table 2.1. Our neutronic calculations have shown that a somewhat better quality beam, at the target position, is obtained with $D_2O$. The heat removal system will function adequately with either $D_2O$ or $H_2O$. Our preference is to operate with $D_2O$, but should we not be able to obtain $D_2O$ coolant in a cost free loan, we will use $H_2O$. We do not intend to switch coolant between $D_2O$ and $H_2O$ routinely. There is of course a major difference in the precautions required for handling $D_2O$ versus $H_2O$. We will use our experience with the $D_2O$ reflector in the MITR to assure that the tritium hazard is adequately managed. However, because of the low power of the fission converter and its low capacity factor, less than 10%, the specific activity of the tritium in the fission converter coolant will be much lower than that in the $D_2O$ of the MITR reflector.

2. As discussed in question 13 below, the fission converter tank will be built to ASME Code Sections II and IX. Other aspects of the Fission Converter Facility, including piping, shielding, shutters, and the medical room, either have or will be designed with the involvement of licensed professional engineers (nuclear and mechanical) and reviewed by the reactor engineering staff and the Reactor SafeguardsCommittee.

3. The grid plate is designed to allow minimal clearance in order to reduce bypass flow. The amount of bypass flow is implicitly limited by TS# 6.6.2.1(2), which was used as the basis of the thermal hydraulic limit (SL and LSSS) calculations. The calculated SL and LSSS are valid as long as $F_f \times d_f > 0.8$, which takes into account both the effects of bypass flow and flow disparity among fuel elements as well as within the fuel element. The primary coolant flow through each fuel element will be measured during the initial startup test.

4. A step in the existing reactor startup procedure ensures that the cadmium curtain is fully inserted before and during reactor startup. This procedure will also be applied to the converter control shutter. In addition, there will be an interlock between CCS position and the reactor's withdraw permit circuit. This interlock will necessitate the CCS being in the fully closed position in order to commence reactor startup (TS# 6.6.4.7(e)). The interlock will be equipped with a key operated bypass to allow testing of the interlock and to allow bypassing of the interlock once the reactor is critical. Use of this bypass will be in accordance with existing MITR procedures. It should be noted that there is no safety significance in terms of either reactivity or radiation if a reactor startup is conducted with the CCS open. The concern is for the safety of the fission

converter and in particular that the CCS not be opened with the reactor at power and flow not established within the fission converter. Note: An additional safety feature that addresses the above concern is the interlock associated with the fission converter operation panel ON/OFF key switch. This interlock ensures that the CCS is closed whenever that key switch is in the OFF position. Also whenever the keyswitch is in the ON position, the low coolant level reactor scram is activated and either the fission converter primary coolant flow scram is also activated (forced convection) or a reactor scram at 15 kW fission converter power is operable (natural convection). Hence, it is not possible to open the CCS without adequate heat removal.

5. We have revised the proposed TS to utilize the reactivity worth for movable experiments as specified in existing TS# 6.1. We believe that the reactivity worth of the converter control shutter (CCS) will be substantially less than the existing limit (0.2% $\Delta K/K$).

6. These results are from two sets of MCNP calculations, the first set was done by W. Kiger and the second was by S. Sakamoto, both of whom were graduate students at MIT. Both were supervised closely by MIT faculty. Discrepancies are due to MCNP's random particle transport process and are less than 0.5% between these calculations. The tables have been revised to be consistent.

7. The purpose of the filter/moderator (F/M) is to moderate fission neutrons into the epithermal range (1 eV - 10 keV) and to reduce the intensity of fast neutrons. A large number of transport calculations have been carried out to examine a variety of F/M materials and configurations, see Ref. [2-1]. Satisfactory beams can be obtained with combinations of aluminum and aluminum oxide, aluminum and aluminum fluoride, aluminum and Teflon® and, aluminum and graphite. Furthermore, these materials have engineering properties such as mechanical strength, temperature resistance, radiation stability, and fabricability which make them suitable for this application. We chose to use aluminum/Teflon® because of its somewhat lower cost, ready availability, and very easy fabricability compared to the other material combinations. This additional information has been incorporated in the revised SER.

8. Nucleate boiling includes subcooled boiling and saturated boiling. The former refers to the condition where small bubbles are formed on the heated surface but collapse when they enter into the bulk coolant. The latter refers to the condition where the small bubbles coalesce with each other upon leaving the heated surfaces and form larger bubbles. The bubbles then collapse and form a vapor core with liquid film on the heated surface with higher heat fluxes. Onset of nucleate boiling (ONB) or incipient boiling defines the condition where small bubbles first start to

form on the heated surface. The definitions are included in the SER as well as in the basis of TS 6.6.1.

9. In the event of loss of fission converter primary flow, there is no provision for natural circulation flow because a natural circulation loop can not be established with the inlet pipes in the downcomers. A safety analysis was performed for this transient (see SER section 6.4). The results showed the maximum fuel temperature during this transient is well below the fuel softening point (450°C). With the reactor automatic scram, the maximum fuel temperature is 100°C and no boiling will occur. If the reactor automatic scram failed and the fission converter shuts down with automatic converter control shutter closure, the maximum fuel temperature is 139°C. Note that for the second scenario cold water will flow down from the top to wet the fuel surface with the vapor flowing upward when boiling occurs. This will provide adequate cooling for the fuel plates. (Ref: " The Effect of Flow Direction and Magnitude on CHF for Low Pressure Water in Thin Rectangular Channels", K. Mishima and H. Nishihara, Nuclear Engineering and Design 86(1985)). If low temperatures are desired, the cleanup loop could be used to provide cooling. Please see response to question #10.

10. For any given patient irradiation, the fraction of time spent in patient setup will likely exceed the time spent under the beam. According, for the anticipated duty cycle of the fission converter, the decay heat load will be small. There may, however, be other sources of heat that are deposited in the fission converter. This will include both gamma heating and fissions caused by leakage neutrons through the CCS. The temperature of the fission converter coolant is monitored by an alarmed instrument. If the alarm is received, the corrective actions might include use of the cleanup pump to lower the temperature. The SER has been modified to include this information.

11. There will be a burst disk or equivalent present on the cover gas system that will be set for 5 psig. The converter tank was hydrostatically tested at 10 psig. Hence, any prolonged operation at or below this value will have no consequences. In addition, the cover gas will have a feed and bleed control to compensate for pressure changes expected during normal startup and shutdown transients. This system will be set to bleed at 4 psig or lower.

12. Under normal operation, the fission converter primary system will operate at a pressure that is about 10 psi lower than that of the secondary system. Should an overpressure event occur, the burst disk will open before the primary pressure exceeds that of the secondary. The MITR-II secondary water radiation monitor will be operating and the secondary water will be sampled daily for tritium content. Any leakage to the fission converter primary system will be seen as a

level or a conductivity increase. The tritium concentration of the fission converter primary system using $D_2O$ will be at least two orders of magnitude lower than that of the reactor $D_2O$ system. Hence, the consequences of a leak into the secondary system will be much less than that from the reactor $D_2O$ system. Similarly, fission product activity will be much lower in the fission converter so the consequences of a leak will be much less than the reactor primary system. Leak detection will be present on and around the fission converter piping and will alarm in the reactor control room if a leak is detected. This will be in place regardless of which coolant is used. It is planned that the secondary flow will continue even if the fission converter is in a shutdown condition. Therefore, the secondary pressure will remain higher than the primary pressure. If this is not the case (maintenance activity to the secondary system, for example), the fission converter heat exchanger will be isolated and vented.

13. We are committing to the use of ASME Code sections II and IX and will document materials and welding specifications. No seismic loading considerations were made per se, although an analysis was made in SER Section 6.3 which showed that an instantaneous LOCA would not result in a fuel cladding failure. Any seismic damage would be bounded by this analysis.

14. The aluminum block placed between the fuel and the wall of the fuel tank, serves to minimize the coolant/moderator near the fuel. This removable aluminum block is planned to be used with either $H_2O$ or $D_2O$ coolant. It has been shown that if $H_2O$ coolant is allowed to fill the space occupied by the block, the beam will be overmoderated and a poor epithermal beam will result. TS# 6.6.4.5 was added to specify the associated requirements on this aluminum block. Calculations of $K_{eff}$ for the fission converter have been made for fission converter operation both with and without the aluminum block. The maximum $K_{eff}$ with the block removed using $H_2O$ and fresh fuel is $0.670 \pm 0.0012$, compared to $0.589 \pm 0.0012$ with the block installed. The calculated results for the hot channel factors and fission converter powers are summarized below (file memos attached in Appendix A):

| | Configuration | Hot Channel Factor | Total Power (kW) (MITR at 5 MW) |
|---|---|---|---|
| With Al Block | Fresh Fuel, $H_2O$ cooling | $1.47 \pm 0.013$ | $125.7 \pm 0.1$ |
| | Fresh Fuel, $D_2O$ cooling | $1.53 \pm 0.014$ | $104.9 \pm 0.1$ |
| Without Al Block | Fresh Fuel, $H_2O$ cooling | $1.55 \pm 0.007$ | $158.2 \pm 0.4$ |
| | Fresh Fuel, $D_2O$ cooling | $1.57 \pm 0.007$ | $141.5 \pm 0.4$ |

15. The low-power/natural-circulation operational mode of the fission converter is intended for the initial startup test when measurements require the lid to be off.

Because the tritium concentration is relatively low in fresh $D_2O$, tritium release to the building atmosphere should be extremely low. If the tank needs to be open for maintenance/refueling, an analysis will be performed prior to such operation to predict the associated stack release and address the procedures required for personnel radiation protection. This is the same approach currently followed for the MITR's $D_2O$ reflector tank.

16. Closure of the fission converter tank is provided by a bolted hatch with a rubber O-ring seal. It is both gas and water tight.

17. No. The minimum operational requirement of the recombiner was derived based on the assumption that the fission converter operates on a 100% duty cycle. Therefore, the recombiner can be operated any time within a month.

18. The following statements are added to section 3.5 of the SER:
" Onset of nucleate boiling (ONB, also called incipient boiling) defines the condition where bubbles first start to form on the heated surface. Because most of the liquid is still subcooled, the bubbles do not detach but grow and collapse while attached to the wall. Onset of significant voiding (OSV) describes the condition where the bubbles grow larger on the heated surface and start to detach regularly. "

19. No. The calculations will be redone only if the parameters used are less conservative than the measurements. For example, if the measured hot channel factor ($F_{HC}$) is higher than that used in the thermal hydraulic limit calculations, the calculations will be redone based on the measured value. TS# 6.6.5 has been added to address the initial startup reporting requirements for the fission converter.

20. The question is correct in that the fission converter SLs and LSSSs are based on 10 MW operation of the MITR. The question is also correct in that the nuclear hot channel factor obtained from Table 2-4 was determined for 5 MW operation. However, this factor is valid at any power level, including 10 MW, because it is the ratio of the fuel plate maximum power to the fuel plate average power. This ratio is independent of the actual power level provided that no power dependent feedback mechanism such as boiling has altered the thermal hydraulic behavior of the converter. This is in fact the case for the fission converter for a reactor power level up to at least 10 MW. Hence, the entire discussion on pages 3-22 to 3-25 is relevant to 10 MW operation. (File memo attached in Appendix F)

21. "top of fuel" means "top of fuel element". This has been corrected in the SER and TS.

22. The reviewer has raised a valid point that a margin should exist between LSSS and SL for natural convection cooling. Therefore, a 5°C margin has been added to

establish the LSSS curve. The LSSS setting for the fission converter power is thus lowered to 20 kW from 25 kW. A margin already existed between SL and LSSS for forced convection. Analyses for both forced convection and natural convection have been performed which show that the margins are adequate. Namely, the margins are sufficient so that automatic protective actions will correct an abnormal situation before a SL is reached. (File memo attached in Appendix G)

23. TS# 6.6.2.2 has been modified to utilize the reactivity worth for movable experiments. Please see the response to question #5. The opening speed of the converter control shutter (CCS) is therefore no longer an issue. The total reactivity effect of the CCS drop is the negative value of the reactivity worth of the CCS, which depends on the fuel/coolant combination of the fission converter system. Measurements will be made to confirm its reactivity worth during the initial startup testing.

24. These two figures were made based on the MITR power of 5 MW, using $D_2O$ coolant and spent fuel elements. The 251 kW figure was calculated for 10 MW with $H_2O$ coolant and fresh fuel elements. The captions for these two figures have been changed to indicate the conditions of these calculations.

25. We will not use a variable speed motor except possibly in the startup testing. During routine operation a fixed speed motor will be used. Provisions to protect against inadvertent movement of the converter control shutter (CCS) are listed below:

    a. An interlock ensures that the fission converter primary flow must be established in order to open the CCS (forced convection operation). (TS# 6.6.4.7 (a))
    b. An interlock ensures that the fission converter coolant level scram must be enabled in order to open the CCS. (TS# 6.6.4.7 (b))
    c. An interlock ensures that the water shutter and mechanical shutter will close automatically when the medical shield door is opened. The interlock will also prevent opening of the shutters if the shield door is open. (TS# 6.5.5 (a), (b))
    d. An interlock ensures that the CCS will close automatically when the CCS control panel key switch is in the OFF position. (TS# 6.6.4.7 (d))

    For additional discussion and an additional interlock for the CCS, please refer to our response to question #4.

26. Calculations of the dose rate in the medical room with all shutters closed, including the water shutter, indicate that water without dissolved B-10 will be adequate to reduce dose rates in the direct beam at the patient position to ~1 mrem/hr. B-10

may be added if a further reduction in neutron or hydrogen capture gamma dose rate is desired. The reactivity effect of B-10 is negligible. (file memo attached in Appendix B)

27. The remotely operated valve in the water shutter system will be either solenoid or air-operated. This valve will be normally open so that in the event of a loss of power, the shutter will fill. This valve is located outside the medical room. Hence, if manual operation were needed, no entry into the medical room would be required. In order to empty the water shutter, the valve will shut and the main pump, located outside the medical room, will start pumping water to the supply tank. Because the power failure mode of the valve is to keep the water shutter closed, there are currently no provisions to manipulate the valve manually.

28. The following list summarizes our planned shutter controls, their locations, and manual/automatic override if applicable.

A. Fission Converter Operation Control Panel (Located in Control Room)

A1. Fission Converter ON/OFF Key Switch
This key switch enables flow and level scrams as well as interlocks B1 through B3. It also energizes the fission converter operation control panel.

A2. Converter Control Shutter OPEN button

A3. Converter Control Shutter CLOSE button

A4. CCS Remote Open Permission Key Switch

The above controls come with indication lights.

B. Interlocks

The following interlocks are required by technical specifications.

B1. Fission Converter Primary Flow Scram (forced convection) / Converter Control Shutter Open Interlock (TS# 6.6.4.7 (a))

B2. Fission Converter Coolant Level Scram / Converter Control Shutter Open Interlock (TS# 6.6.4.7 (b))

B3. Medical Room Shield Door / Water Shutter and Mechanical Shutter Open Interlock (TS# 6.5.5 (a), (b))

B4. Loss of Electrical Power/ Water Shutter and Mechanical Shutter Close Interlock (TS# 6.5.5 (c))

B5. Medical Room Control Panel Key Switch Off / Water Shutter and Mechanical Shutter Close Interlock (TS# 6.6.4.7 (c))

B6. Reactor Startup / CCS Fully Closed Interlock (TS# 6.6.4.7 (e))

C. <u>Fission Converter Medical Room Control Panel (MRCP)</u>

C1. ON/OFF Key Switch
C2. Minor Scram button
C3. Converter Control Shutter (CCS) Close Button
C4. CCS OPEN/CLOSED Display
C5. Mechanical Shutter OPEN/CLOSE buttons and display
C6. Water Shutter OPEN/CLOSE buttons and display
C7. Med Room door OPEN/CLOSED display
C8. CCS OPEN Button
(operable only with remote permission key switch ON, see A4)

D. <u>Medical Room Emergency Control Panel (Inside the medical room)</u>
(TS 6.5.5 (e))

D1. Converter Control Shutter CLOSE button
D2. Mechanical Shutter CLOSE Button
D3. Water Shutter CLOSE button

Position Displays for the above.

E. <u>Emergency Manual Shutter Controls</u>
(TS 6.5.5 (d))

E1. Mechanical Shutter

Our answers to other sub-questions are listed as follows:

a. The CCS, water shutter, and mechanical shutter can be closed from inside the medical room. No significant dose rates exist in the medical room when all shutters are closed (~ 1 mrem/hr). Therefore, no scram button is required in the medical room. Also, the water and mechanical shutters are interlocked to close when the medical room door is opened. A fixed radiation monitor is provided so that the dose rate in the room is known prior to entry.

b. The table below shows the dose rates at the patient position in Rem/hour for various shutter conditions (adopted from Appendix H). In all cases, the dose rate exceeds the criteria for a high radiation area if one of the shutters fails to close with the reactor at power. The corrective action for such a situation is to lower the reactor power. This is the same approach that is used for the current MITR medical room shutters. Also, it should be noted there is a minor scram button on the medical room control panel.

| Condition of Shutters (MITR-II at 5 MW) | Dose Rate at Patient Position (Rem/hr) |
|---|---|
| CCS Open, Water and Mechanical Shutters Closed | 0.1 |
| Mechanical Shutter Open, Water Shutter and CCS Closed | 15 |
| Water Shutter Open, Mechanical Shutter and CCS Closed | 0.2 |

c. The medical room control panel key switch is interlocked with all three shutters. The shutters will close automatically when the key switch is in the OFF position.

d. Opening of the CCS is under the control of a licensed reactor operator. Please see #36 for detailed information.

29. The medical room shielding is designed to reduce the radiation levels, from the medical beam, on the outside of the room to $\lesssim$ 1 mrem/hr for all possible fission converter facility operating scenarios including MITR at 10 MW. Patients will be observed visually during irradiations using TV camera(s) and a shielded viewing window. This is the same as the observation of patients in the current medical room. A boron or lithium containing paint, plaster, or layer, e.g., boral will be used to reduce wall activation in the medical room. This approach is used in the existing MITR medical room. A permanent radiation monitoring system with readouts at the medical control panel outside the irradiation room will provide information concerning ambient radiation levels in the medical room. A similar system is currently used in the MITR medical room in the reactor basement. Note: This information has been added to SER section 4.3.

30. The MITR administrative procedures apply to the fission converter facility. PM 1.9.1 states that "If not part of an approved procedure, bypasses must be individually approved before implementation by the Duty-Shift-Supervisor or Reactor Superintendent. The bypass, authorizer's initials (or number of approved procedure) must be recorded on the bypass log sheet at the time of implementation." The medical room door bypasses will normally only be used for tests and experiments.

31. The maximum dose for a fuel plate melting accident is covered in section 6.1, "Maximum Hypothetical Accident" (four fuel plates melted). The following information has been added to the SER for clarification:
If the fission converter plate has been operating at 250 kW continuously, the whole body dose during the first two hours from the fission converter is (250 kW/5000 kW)(24 elements/11 elements) x 595 mrad= 65 mrad at 21 m (front fence), or 41 mrem at 8 m (back fence). The thyroid dose from containment leakage is (250 kW/5000 kW)(24 elements/11 elements) x 118 mrad=13 mrad. The calculated whole-body doses are significantly lower than the annual dose limit to a member of the general public (500 mR).
The 595 mrad and 118 mrad are the estimated doses, whole-body and thyroid respectively, at the nearest point of public occupancy during the first two hours of the MITR-II maximum hypothetical accident.

32. Situations of this type would be covered by the existing MITR emergency plan. In response to the sub-questions:

a. Dropping of a fuel element should not result in a radiation release because (1) the element would fall through water, which would cushion any impact and, (2) the fuel is a cermet which would limit the release of fission products should the clad be scratched or otherwise damaged. It should be noted that all MITR fuel handling tools have a safety lock feature that prevent this type of accident.

b. A large heavy load could not impact the converter because it would be deflected by surrounding shielding. Also, administrative procedures, which are followed for the reactor core, would preclude situations that could lead to this type of accident. Should such an accident somehow occur, the consequences would be limited because the fission converter operation cycle, even at a full patient load, will be a small fraction of a day. This is because most of the time spent on each patient will be for setup. Hence the fission product inventory will be low. Our calculation estimates that the fission product inventory is 31660 curies per fuel element 10 minutes after shutdown for a refueling. In the event that a fuel element was damaged in the converter, calculated results show that the total whole body dose would be 2.8 mrem at 8 meters (ground release) and 3.9 mrem (plume release) at 21 meters from the containment. The thyroid dose would be 0.5 mrem during the first two hours. The above figures are less than 0.5% than those calculated for the MITR (5 MW) design basis accident. (file memo attached in Appendix C)
In the event that all eleven fuel elements in the fission converter were damaged, the total whole body dose would be 30.8 mrem at 8 meters and 42.9 mrem at 21 meters from the containment, the thyroid dose would be 5.5 mrem during the first two hours.

33. The calculation has been redone based on a more realistic power function. A slight increase in the maximum fuel temperature from 132°C to 139°C was calculated using the power function shown in Figure 4.1 in the fission converter SER. (file memo attached in Appendix D)

34. The proposed fission converter overpower setpoint is 275 kW, which is 110% of the maximum anticipated design power, as shown in Tables 2.3 and 7.1. The automatic response that occurs at the overpower setpoint is CCS closure. An alarm will be provided at 110% or less of the fission converter's nominal operating power (TS# 6.6.4.8 and 6.6.2.5(3)). This alarm shall not exceed 275 kW. The fission converter's nominal operating power was calculated for combinations of reactor power, primary coolant, and fresh/spent fuel element using MCNP. The results will be verified during the initial startup test. The setpoints for fission converter protective actions are summarized in Tables 1 and 2 in our response to question #37. As stated above, one of the factors that determines the alarm setpoint is the reactor's power level (5 or 10 MW) which in turn affects the fission converter's nominal operating power. The setpoints for the other protective actions do not depend on the fission converter power level because the fission converter flow is always established for operation of the fission converter at 250 kW, which corresponds to a reactor power of 10 MW.

35. The function of the "design power" safety channel is to shut down the fission converter in case of an overpower condition. The design power is based on the LSSS (300 kW) regardless of the actual configuration (type of coolant and fuel) of the system. This safety channel is specified as "power" in Table 6.6.2.5-1 in TS# 6.6.2.5. The "nominal operating power" channel will provide an alarm in the reactor control room if the fission converter power exceeds 110% of its nominal operating power as specified in TS# 6.6.2.5 (3). The magnitude of the nominal operating power, which depends on the configuration of the system, was determined using MCNP and will be confirmed during the initial startup tests. The latter provides an extra margin to the design power. It is currently planned to use one neutron detector that will provide settings for both "design power" and "nominal operating power".
The following statement has been added to section 7.4 of the SER:
" The fission converter medical room control panel (MRCP) is a dedicated control panel for instruments, such as the shutters, that are related to the beam control and use of the fission converter medical room. The fission converter control panel (FCCP) houses the control and display of the process system instruments of the fission converter."

36. The converter control shutter (CCS) OPEN/CLOSE buttons will be located on a panel in the reactor control room. The panel is energized by an ON/OFF key

switch. Administrative controls will ensure proper possession of the key as with other keys associated with reactor operation. The CCS shall be opened from the control room by a licensed operator. In the event that the reactivity worth of the CCS is small and can be compensated using reactor automatic control, permission may be given to trained non-licensed personnel to open the CCS from the medical room control panel. This is enabled by turning a key switch on the same panel in the control room to the ON position in order to energize the CCS OPEN button on the medical room control panel. The authorized personnel will ask for the on-console operator's permission before operating the CCS. This is the current practice of using the $D_2O$ and $H_2O$ shutters for the M67 beam currently used for the BNCT clinical trial. In answer to specific sub-questions:

a. Pre-operational checks of the fission converter will be performed by licensed personnel (or a trainee under direct supervision) in the same way that reactor system pre-operational checks are conducted. Upon completion of these checks, all systems are operational and the converter can be placed on line by opening the CCS. This would be done by a licensed person (or a trained non-licensed person) upon direct receipt of authorization from the reactor console operator.

b. If non-licensed individuals wish to qualify on operation of the CCS, a written qualification program will be prepared. We have such a program for the existing MITR medical facility. It is structured along the lines of the existing MITR training programs for the reactor operators. Operation of the CCS by qualified non-licensed personnel meets the requirements of 10 CFR 50.54 because the licensed console operator retains direct control. Specifically, the person operating the CCS must request permission from the console operator to open the CCS. Moreover, this permission must be requested immediately before opening the CCS. Please refer to existing MITR TS# 6.5.16.

c. No. There are no fission converter operations that require the presence of a senior reactor operator.

d. The console operator's responsibilities for the fission converter will not be any different than they are now for experiments that have the potential to affect reactivity. The fission converter is not a distraction, in the sense that it is a separate activity distinct from the operation of the reactor. The two are coupled and the operator cannot properly monitor reactor operation unless he/she is also given the authority to direct fission converter operation. The sub-question implies that it might be desirable to split the responsibility. To do so would make it very difficult for the console-operator. One person, the console operator, should be in overall charge of anything that can affect the reactor.

e. In case of an emergency, the console operator's responsibility is either to close the CCS or scram the reactor as appropriate. Subsequent actions, which would take place outside the control room, would be the responsibilities of other licensed operators.

37. The interface of the fission converter safety system with the reactor control and safety system is shown schematically in Figure 1. Tables 1 and 2 outline the protective actions and proposed setpoints for corresponding major transients for forced convection and natural convection operation. A jumper (or equivalent) will be used to bypass the flow scram and CCS automatic closure for lower power (natural convection) operation. For forced convection cooling, protection against a fission converter overpower condition is provided by an alarm at 110% of nominal operating power and an automatic CCS closure at the over-power setpoint 275 kW. A reactor scram on fission converter overpower is not needed because the reactor itself will have already scrammed on high power. For natural convection cooling, protection against a fission converter overpower condition is provided by a reactor scram at the reactor power which corresponds to fission converter power 15 kW. This different approach is necessary because an overpower condition can occur on the fission converter during natural convection cooling even though the reactor itself is operating within its licensed operating power.

Table 1 Protective Actions for the Fission Converter Transients Related to Safety (Forced Convection)

| Transient | Automatic Reactor Scram | Automatic Converter Control Shutter Closure | Proposed Setpoints (LSSS) |
|---|---|---|---|
| Overpower | | X | 275 kW (300 kW) |
| Overtemperature | | X | 55 °C (60 °C) |
| Low Coolant Level | X | X | 2.4 m (2.1m) |
| Low Primary Flow | X | X | 50 gpm (45 gpm) |

## Fission Converter Safety System

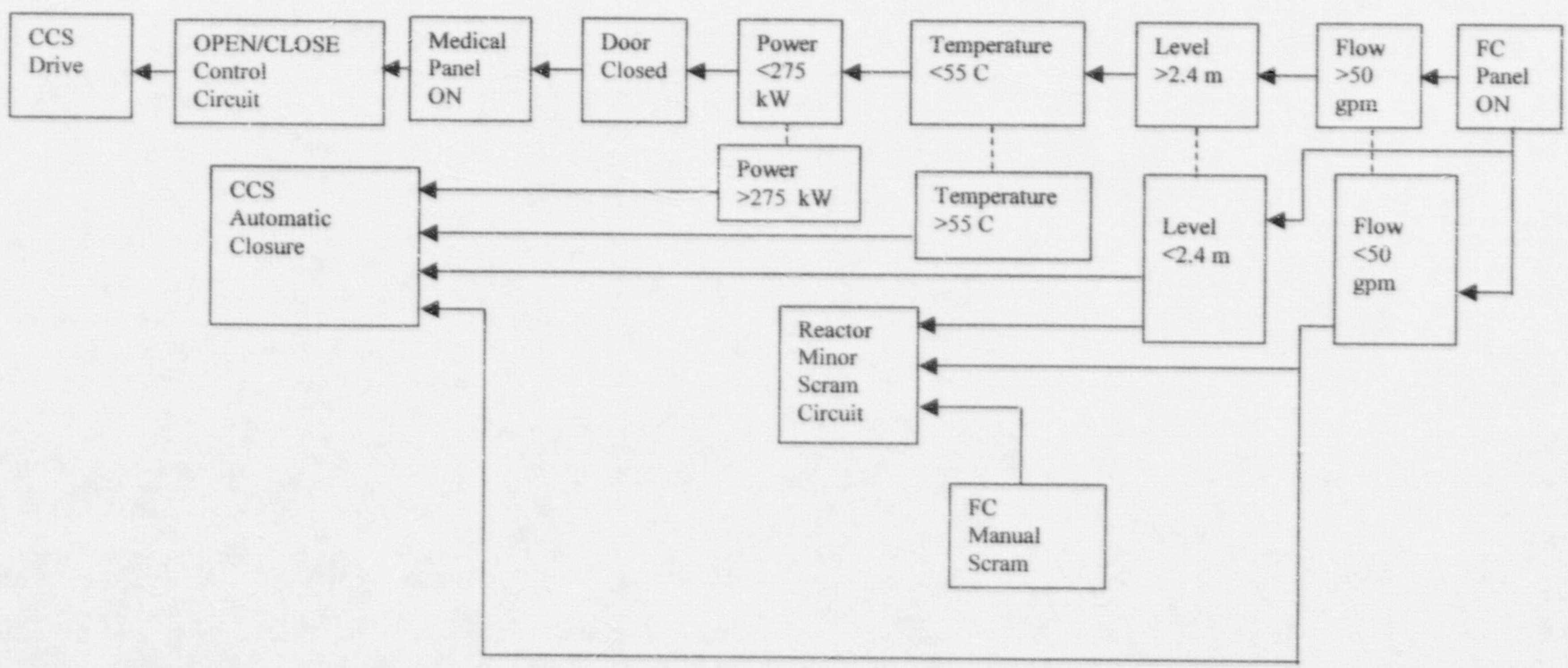


Table 2 Protective Actions for the Fission Converter Transients Related to Safety (Natural Convection)

| Transient | Automatic ReactorScram | Automatic ConverterControl Shutter Closure | Proposed Setpoints (LSSS) |
|---|---|---|---|
| Overpower | X | X | 15 kW (20 kW) |
| Overtemperature | | X | 55 °C (60 °C) |
| Low Coolant Level | X | X | 2.4 m (2.4m) |

38. The MITR Quality Assurance Program has been modified to include the fission converter. This means that all test procedures associated with the verification of the system design and its subsequent operation must be prepared in writing and be subject to a formal safety review. This review will examine the proposed test for (1) determination of the presence of an unreviewed safety question, (2) reactor safety issues, (3) industrial safety issues, and (4) ALARA considerations. In addition, the purpose of the procedure, any prerequisites, and required equipment are documented. Test results are filed formally and are available for review by cognizant regulatory authorities. The fission converter is treated under the MITR Quality Assurance Program in the same way that any major reactor component or system is addressed.

39. We anticipate that a series of calorimetrics (i.e. Power = $\sum \dot{m} c_p \Delta T$) will be performed to calibrate the nuclear instrumentation. The fission converter's heat removal system will be placed on line. The reactor's power will be adjusted to some low level, 500 kW for example. A calorimetric will then be performed and the result compared against the signals from the fission converter's nuclear instruments. The reactor's power will then be raised to some higher level, 1 MW for example, and the procedure repeated. The result will be a curve of nuclear instrument signal versus thermal power. This process is iterative. The "initial conservative operating condition" will be documented in the Q/A package that is prepared in conjunction with the startup testing of the fission converter.

40. This question is no longer relevant because the TS limit on the CCS operation has been changed to specify the reactivity worth.

41. After startup testing, a constant speed, AC motor driven at the 60 Hz line frequency will be used. The reactivity associated with opening the converter control shutter will be determined during the initial startup testing. Proposed TS#

6.6.2.2 has now been modified so that the opening speed of the CCS is no longer an issue.

42. "in-core temperature distribution" is changed to "temperature distribution in the fission converter plate"

43. The thermal hydraulic limits of the fission converter are set conservatively for a maximum power of 250 kW. This power level can only be achieved with use of light water, fresh fuel elements and reactor power of 10 MW. The fission converter can operate safely regardless of the reactor power as long as the thermal hydraulic limits are met. Therefore, it is not necessary to specify limits for 5 MW and 10 MW separately. However, there will be an alarm for the nominal operating power. Please see response to question # 34.

44. The reference point for water height should be the top of fuel element. This has been corrected in the SER and TS.

45. Please see response to question #22.

46. All calculations previously discussed in the SER that was provided to the NRC were done for eleven fuel elements. Current calculations show that the SL and LSSS are the same for ten fuel elements for forced convection but the differences are quite significant for natural convection (see Appendix E). TS# 6.6.2.1(4) has been modified to state the requirements for each operating condition.

47. Yes. TS# 6.6.2.1(8) is now changed to "...in place and sealed..".

48. "or its equivalent" was used with "cadmium curtain" in TS# 6.6.2.2 as a provision of possible material change of the curtain, such as using Boral instead of (or in conjunction with) cadmium. The term "cadmium curtain" is now changed to "converter control shutter" or CCS in the SER and associated TS to reflect the function of the shutter without referring to a specific material.

49. The technical specification for the CCS closing time has been deleted because a malfunction of the automatic reactor scram on low primary flow is not considered a credible event. Therefore, a surveillance requirement on the closure time for the CCS is not necessary.

50. TS# 6.6.2.4 specifies the $H_2/D_2$ concentration in helium. The MITR is an open system and has an air blanket over the primary coolant that is contained in the core tank. Therefore, TS# 3.4 was written for $H_2$ in air which has a higher $O_2$ concentration.

51. TS# 6.6.2.6 (4) has been added to address actions that should be taken in the event of out-of-specification water chemistry.

52. According to TS# 6.6.3, functional tests of the instrumentation shall be made prior to operation of the fission converter if the converter is not scheduled to be used and use becomes necessary. The functional tests will require bypass of the reactor scrams for fission converter primary flow and coolant level to prevent a reactor scram during these tests. "neutron flux level" in TS# 6.6.3 is now changed to "power level" to match TS# 6.6.2.5. Chloride content has been deleted from the primary coolant sampling requirement TS# 6.6.3 (4), the reason being that the conductivity will reflect an increase in the chloride level.
We do not think that a TS for initial startup of the fission converter is necessary. We will develop a startup plan and associated procedures. These documents will be available for review before the initial startup testing.

53. The sample assemblies that might be introduced into the fueled region of the converter are subject to TS# 6.6.4.4, which refer to TS# 6.6.2.1(4) and TS# 5.2. The former states that the requirements for sample assemblies and the latter specifies the criteria for MITR's in-core sample assembly design.

54. Wording in TS# 6.5 has been changed and now it applies to the neutron beams from both the basement medical room and the fission converter medical room. "Cadmium shutter" is the same as the "cadmium curtain". It has been renamed as "converter control shutter" (CCS), please see response to question #48. This has been changed in both the SER and TS for consistency. Please refer to response to question #36 for the CCS control.

55. Please refer to reply #36 for the CCS control. The CCS can be operated by any licensed operator, by any operator candidate who is in training and who is under the direct supervision of a licensed SRO, or by a non-licensed person who has completed a written qualification program on the fission converter. In all cases, the reactor console operator has overall authority and his/her permission will be required prior to any CCS operation. TS# 6.6.5 has been added to address the initial startup reporting requirements for the fission converter.

# APPENDIX A

# Memorandum

**To:** FCB Group

**CC:** File

**From:** K. Riley

**Date:** 08/12/98

**Re:** MCNP criticality calculations of the FCB fuel with the aluminum block removed from the fuel tank.

---

Several calculations to evaluate the effective multiplication constant (keff) of the fission converter with and without the aluminum block inserted in the fuel tank, and with various coolant and fuel loading conditions have been completed.

The MCNP model of the fuel tank matches the most recent design that was submitted to Artisan Engineering Inc. in January 1998. It includes the removable aluminum block that is placed between the fuel and the tank wall, on the patient side of the fuel. When the block is removed, it is replaced with either $H_2O$ or $D_2O$ coolant, as appropriate for the situation being modeled. Spent fuel contains 312 g $^{235}U$ per element, and fresh fuel contains 510 g $^{235}U$ per element.

The table below summarizes the results of the calculations. It is clear from the results that even with the block removed, using fresh fuel and $H_2O$ coolant, the system is still very subcritical and poses no credible risk from a criticality accident.

| Configuration | $k_{eff}$ |
|---|---|
| Spent fuel, $D_2O$ cooling, block in place | 0.2636 ± 0.0008 |
| Spent fuel, $D_2O$ cooling, block removed | 0.3910 ± 0.0010 |
| Fresh fuel, $H_2O$ cooling, block in place | 0.5893 ± 0.0012 |
| Fresh fuel, $H_2O$ cooling, block removed | 0.6698 ± 0.0012 |

It should be noted that the two reference cases shown in the table above (spent fuel, $D_2O$ cooling, block in place and fresh fuel, $H_2O$ cooling, block in place) differ slightly from the same calculations that were performed by W. S. Kiger [1] that are listed in the SER. Due to differences in the tank and coolant geometry, different $k_{eff}$ values have been calculated. The SER can be updated to reflect these new values, if so desired, but the differences are small and all values are much less than unity.

The configurations shown in the above table represent the lowest and highest expected $k_{eff}$ values when the aluminum block is removed. Calculations for other configurations can be performed if necessary.

---

1. W. S. Kiger III, *Neutronic Design of a Fission Converter Based Epithermal Beam for Neutron Capture Therapy*, Nuclear Engineers Thesis, MIT, 1996.

# Memorandum

**To:** FCB Group

**CC:** File

**From:** Kent Riley

**Date:** 08/31/98

**Re:** Hot channel factor and power calculations with the aluminum block removed from the FCB fuel tank

---

MCNP calculations have been completed to determine the power distribution for the case of fresh fuel (512 g $^{235}U$ per element) using light or heavy water coolant, with the aluminum block removed from the FCB fuel tank. The power in each fuel plate of each element was tallied, and a hot channel factor was calculated for each case.

The MCNP model of the fuel tank matches the most recent design that was submitted to Artisan Engineering Inc. in January 1998. The removable aluminum block that is placed between the fuel and the tank wall is removed for this calculation and is replaced with light water coolant. The output of the MCNP calculation is scaled to correspond to an MITR II reactor power of 5 MW. The MCNP filenames to reference for this calculation carry the prefix kjr505 (light water) and kjr506 (heavy water).

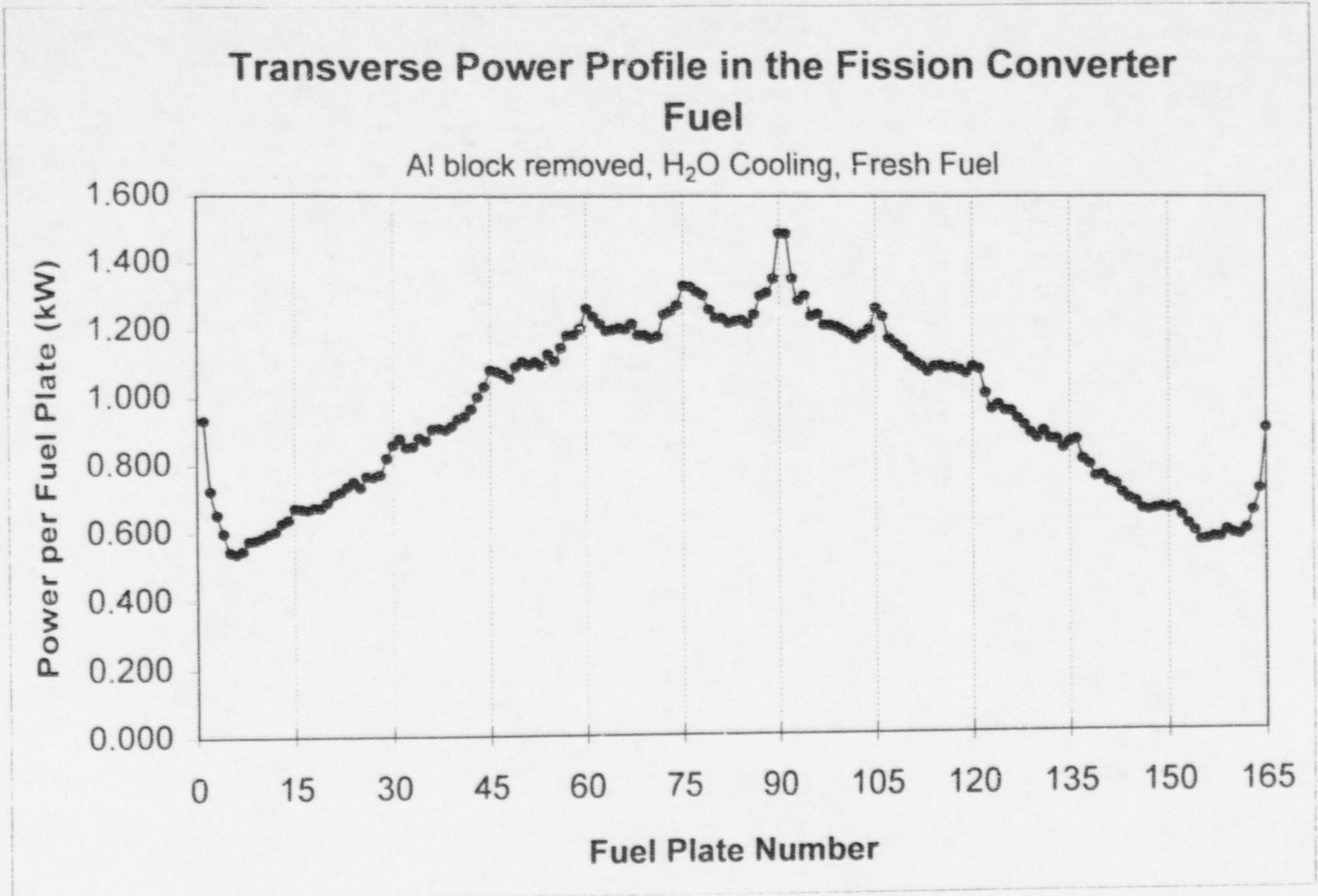
The table below summarizes the results from this calculation, as well as those calculated by S. Sakamoto (memo 1/13/97) under the same conditions, but with the aluminum block in place. The hot channel factors with the block removed are found to be slightly higher than those with the aluminum block in place. In all cases the plate with the highest power is plate 91, which is the first plate of element number 7.

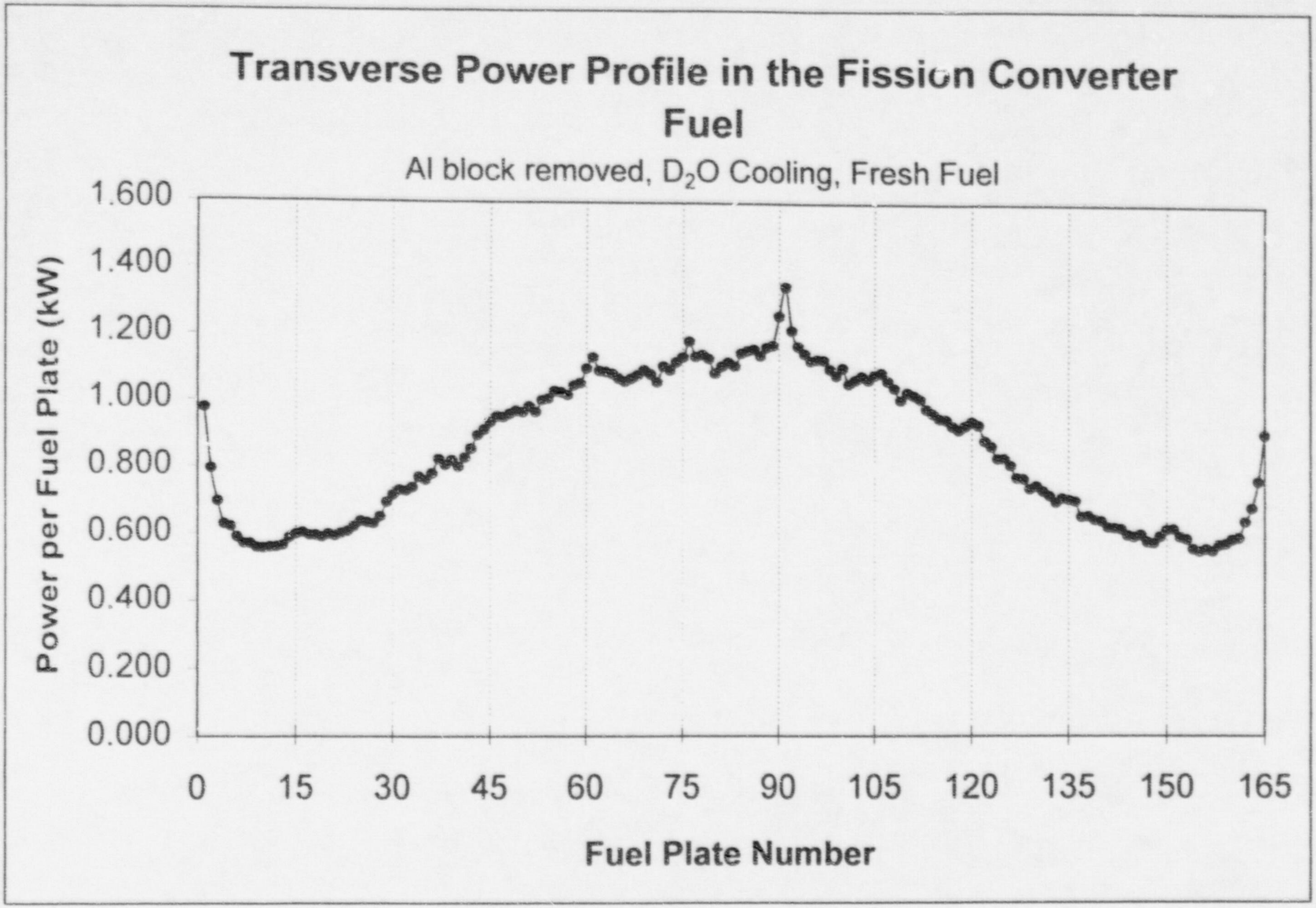
| Configuration | Hot Channel Factor | Total Power (kW) |
|---|---|---|
| Fresh Fuel, $H_2O$ cooling | 1.47 ± 0.013 | 125.7 ± 0.1 |
| Fresh Fuel, $D_2O$ cooling | 1.53 ± 0.014 | 104.9 ± 0.1 |
| Fresh Fuel, $H_2O$ cooling, block removed | 1.55 ± 0.007 | 158.2 ± 0.4 |
| Fresh Fuel, $D_2O$ cooling, block removed | 1.57 ± 0.007 | 141.5 ± 0.4 |

It is also important to notice in the above table that in both cases with the block removed, the total power of the fission converter is significantly higher. An earlier memo (K. Riley 8/13/98) reported a total power of 145.1 kW for the case of fresh fuel, light water cooling and the block removed. The value reported above is correct, as a mistake in the post processing of the MCNP output was discovered for the value reported earlier.

For either light or heavy water cooling, fresh fuel, aluminum block removed, and a reactor power of 10 MW, it appears that the nominal anticipated power of 250 kW will be exceeded (283 kW for $D_2O$ and 316 kW for $H_2O$).

Also attached to this memorandum are plots of the transverse power profile (power per plate) and a summary of the power distribution in each fuel element for the two cases described above.


Transverse Power Profile in the Fission Converter Fuel
Al block removed, $H_2O$ Cooling, Fresh Fuel
Power per Fuel Plate (kW)
1.600
1.400
1.200
1.000
0.800
0.600
0.400
0.200
0.000
0
15
30
45
60
75
90
105
120
135
150
165
Fuel Plate Number


Transverse Power Profile in the Fission Converter Fuel
Al block removed, D2O Cooling, Fresh Fuel
Power per Fuel Plate (kW)
1.600
1.400
1.200
1.000
0.800
0.600
0.400
0.200
0.000
0
15
30
45
60
75
90
105
120
135
150
165
Fuel Plate Number

## Al block removed, $D_2O$ cooling, fresh fuel

| Element 1 | | Element 2 | | Element 3 | | Element 4 | |
|---|---|---|---|---|---|---|---|
| Plate # | Power (kW) | Plate # | Power (kW) | Plate # | Power (kW) | Plate # | Power (kW) |
| 1 | 0.975 | 16 | 0.602 | 31 | 0.730 | 46 | 0.951 |
| 2 | 0.793 | 17 | 0.594 | 32 | 0.727 | 47 | 0.950 |
| 3 | 0.695 | 18 | 0.594 | 33 | 0.739 | 48 | 0.959 |
| 4 | 0.627 | 19 | 0.589 | 34 | 0.769 | 49 | 0.969 |
| 5 | 0.620 | 20 | 0.597 | 35 | 0.759 | 50 | 0.964 |
| 6 | 0.588 | 21 | 0.593 | 36 | 0.779 | 51 | 0.981 |
| 7 | 0.570 | 22 | 0.600 | 37 | 0.823 | 52 | 0.965 |
| 8 | 0.569 | 23 | 0.605 | 38 | 0.804 | 53 | 1.001 |
| 9 | 0.558 | 24 | 0.620 | 39 | 0.820 | 54 | 1.006 |
| 10 | 0.556 | 25 | 0.637 | 40 | 0.799 | 55 | 1.028 |
| 11 | 0.559 | 26 | 0.632 | 41 | 0.827 | 56 | 1.025 |
| 12 | 0.559 | 27 | 0.629 | 42 | 0.855 | 57 | 1.017 |
| 13 | 0.565 | 28 | 0.650 | 43 | 0.895 | 58 | 1.046 |
| 14 | 0.587 | 29 | 0.693 | 44 | 0.912 | 59 | 1.053 |
| 15 | 0.597 | 30 | 0.715 | 45 | 0.935 | 60 | 1.098 |
| TOTAL | 9.42 | | 9.35 | | 12.17 | | 15.01 |
| AVERAGE | 0.628 | | 0.623 | | 0.812 | | 1.001 |
| PF | 1.55 | | 1.15 | | 1.15 | | 1.10 |

| Element 5 | | Element 6 | | Element 7 | | Element 8 | |
|---|---|---|---|---|---|---|---|
| Plate # | Power (kW) | Plate # | Power (kW) | Plate # | Power (kW) | Plate # | Power (kW) |
| 61 | 1.131 | 76 | 1.181 | 91 | 1.346 | 106 | 1.087 |
| 62 | 1.091 | 77 | 1.134 | 92 | 1.214 | 107 | 1.063 |
| 63 | 1.088 | 78 | 1.140 | 93 | 1.167 | 108 | 1.041 |
| 64 | 1.082 | 79 | 1.128 | 94 | 1.145 | 109 | 1.004 |
| 65 | 1.068 | 80 | 1.084 | 95 | 1.123 | 110 | 1.027 |
| 66 | 1.057 | 81 | 1.105 | 96 | 1.127 | 111 | 1.018 |
| 67 | 1.067 | 82 | 1.118 | 97 | 1.126 | 112 | 1.007 |
| 68 | 1.078 | 83 | 1.106 | 98 | 1.098 | 113 | 0.977 |
| 69 | 1.093 | 84 | 1.145 | 99 | 1.078 | 114 | 0.963 |
| 70 | 1.080 | 85 | 1.152 | 100 | 1.101 | 115 | 0.948 |
| 71 | 1.057 | 86 | 1.158 | 101 | 1.052 | 116 | 0.945 |
| 72 | 1.104 | 87 | 1.138 | 102 | 1.065 | 117 | 0.926 |
| 73 | 1.092 | 88 | 1.163 | 103 | 1.077 | 118 | 0.914 |
| 74 | 1.118 | 89 | 1.170 | 104 | 1.065 | 119 | 0.927 |
| 75 | 1.132 | 90 | 1.258 | 105 | 1.081 | 120 | 0.940 |
| TOTAL | 16.34 | | 17.18 | | 16.86 | | 14.79 |
| AVERAGE | 1.089 | | 1.145 | | 1.124 | | 0.986 |
| PF | 1.04 | | 1.10 | | 1.20 | | 1.10 |

| Element 9 | | Element 10 | | Element 11 | |
|---|---|---|---|---|---|
| Plate # | Power (kW) | Plate # | Power (kW) | Plate # | Power (kW) |
| 121 | 0.929 | 136 | 0.702 | 151 | 0.618 |
| 122 | 0.882 | 137 | 0.655 | 152 | 0.595 |
| 123 | 0.866 | 138 | 0.658 | 153 | 0.588 |
| 124 | 0.829 | 139 | 0.646 | 154 | 0.560 |
| 125 | 0.830 | 140 | 0.642 | 155 | 0.553 |
| 126 | 0.809 | 141 | 0.623 | 156 | 0.559 |
| 127 | 0.772 | 142 | 0.622 | 157 | 0.554 |
| 128 | 0.769 | 143 | 0.619 | 158 | 0.571 |
| 129 | 0.739 | 144 | 0.600 | 159 | 0.574 |
| 130 | 0.748 | 145 | 0.594 | 160 | 0.587 |
| 131 | 0.733 | 146 | 0.601 | 161 | 0.592 |
| 132 | 0.719 | 147 | 0.580 | 162 | 0.641 |
| 133 | 0.698 | 148 | 0.576 | 163 | 0.683 |
| 134 | 0.711 | 149 | 0.596 | 164 | 0.763 |
| 135 | 0.707 | 150 | 0.615 | 165 | 0.904 |
| TOTAL | 11.74 | | 9.33 | | 9.34 |
| AVERAGE | 0.783 | | 0.622 | | 0.623 |
| PF | 1.19 | | 1.13 | | 1.45 |

## Al block removed, $H_2O$ cooling, fresh fuel

| Element 1 | | Element 2 | | Element 3 | | Element 4 | |
|---|---|---|---|---|---|---|---|
| Plate # | Power (kW) | Plate # | Power (kW) | Plate # | Power (kW) | Plate # | Power (kW) |
| 1 | 0.932 | 16 | 0.669 | 31 | 0.878 | 46 | 1.078 |
| 2 | 0.722 | 17 | 0.666 | 32 | 0.852 | 47 | 1.069 |
| 3 | 0.652 | 18 | 0.672 | 33 | 0.854 | 48 | 1.056 |
| 4 | 0.598 | 19 | 0.671 | 34 | 0.880 | 49 | 1.091 |
| 5 | 0.543 | 20 | 0.688 | 35 | 0.872 | 50 | 1.106 |
| 6 | 0.536 | 21 | 0.710 | 36 | 0.905 | 51 | 1.096 |
| 7 | 0.546 | 22 | 0.719 | 37 | 0.908 | 52 | 1.105 |
| 8 | 0.575 | 23 | 0.735 | 38 | 0.903 | 53 | 1.093 |
| 9 | 0.577 | 24 | 0.748 | 39 | 0.915 | 54 | 1.128 |
| 10 | 0.585 | 25 | 0.732 | 40 | 0.936 | 55 | 1.107 |
| 11 | 0.596 | 26 | 0.766 | 41 | 0.944 | 56 | 1.146 |
| 12 | 0.603 | 27 | 0.762 | 42 | 0.966 | 57 | 1.182 |
| 13 | 0.627 | 28 | 0.770 | 43 | 1.001 | 58 | 1.183 |
| 14 | 0.637 | 29 | 0.820 | 44 | 1.032 | 59 | 1.202 |
| 15 | 0.671 | 30 | 0.858 | 45 | 1.082 | 60 | 1.263 |
| TOTAL | 9.40 | | 10.99 | | 13.93 | | 16.91 |
| AVERAGE | 0.627 | | 0.732 | | 0.928 | | 1.127 |
| PF | 1.49 | | 1.17 | | 1.17 | | 1.12 |

| Element 5 | | Element 6 | | Element 7 | | Element 8 | |
|---|---|---|---|---|---|---|---|
| Plate # | Power (kW) | Plate # | Power (kW) | Plate # | Power (kW) | Plate # | Power (kW) |
| 61 | 1.238 | 76 | 1.328 | 91 | 1.486 | 106 | 1.238 |
| 62 | 1.217 | 77 | 1.313 | 92 | 1.353 | 107 | 1.171 |
| 63 | 1.195 | 78 | 1.302 | 93 | 1.285 | 108 | 1.154 |
| 64 | 1.199 | 79 | 1.259 | 94 | 1.302 | 109 | 1.141 |
| 65 | 1.204 | 80 | 1.232 | 95 | 1.239 | 110 | 1.116 |
| 66 | 1.200 | 81 | 1.233 | 96 | 1.245 | 111 | 1.102 |
| 67 | 1.216 | 82 | 1.218 | 97 | 1.212 | 112 | 1.089 |
| 68 | 1.183 | 83 | 1.224 | 98 | 1.211 | 113 | 1.072 |
| 69 | 1.183 | 84 | 1.227 | 99 | 1.207 | 114 | 1.088 |
| 70 | 1.172 | 85 | 1.216 | 100 | 1.196 | 115 | 1.090 |
| 71 | 1.179 | 86 | 1.244 | 101 | 1.184 | 116 | 1.083 |
| 72 | 1.244 | 87 | 1.300 | 102 | 1.170 | 117 | 1.085 |
| 73 | 1.254 | 88 | 1.310 | 103 | 1.184 | 118 | 1.078 |
| 74 | 1.272 | 89 | 1.352 | 104 | 1.199 | 119 | 1.068 |
| 75 | 1.331 | 90 | 1.487 | 105 | 1.263 | 120 | 1.088 |
| TOTAL | 18.29 | | 19.24 | | 18.74 | | 16.66 |
| AVERAGE | 1.219 | | 1.283 | | 1.249 | | 1.111 |
| PF | 1.09 | | 1.16 | | 1.19 | | 1.11 |

| Element 9 | | Element 10 | | Element 11 | |
|---|---|---|---|---|---|
| Plate # | Power (kW) | Plate # | Power (kW) | Plate # | Power (kW) |
| 121 | 1.081 | 136 | 0.868 | 151 | 0.662 |
| 122 | 1.007 | 137 | 0.809 | 152 | 0.638 |
| 123 | 0.960 | 138 | 0.792 | 153 | 0.612 |
| 124 | 0.974 | 139 | 0.756 | 154 | 0.590 |
| 125 | 0.956 | 140 | 0.759 | 155 | 0.563 |
| 126 | 0.954 | 141 | 0.739 | 156 | 0.565 |
| 127 | 0.931 | 142 | 0.733 | 157 | 0.571 |
| 128 | 0.913 | 143 | 0.706 | 158 | 0.570 |
| 129 | 0.888 | 144 | 0.688 | 159 | 0.592 |
| 130 | 0.872 | 145 | 0.680 | 160 | 0.582 |
| 131 | 0.894 | 146 | 0.657 | 161 | 0.578 |
| 132 | 0.870 | 147 | 0.652 | 162 | 0.596 |
| 133 | 0.869 | 148 | 0.657 | 163 | 0.651 |
| 134 | 0.842 | 149 | 0.661 | 164 | 0.717 |
| 135 | 0.860 | 150 | 0.655 | 165 | 0.899 |
| TOTAL | 13.87 | | 10.81 | | 9.38 |
| AVERAGE | 0.925 | | 0.721 | | 0.626 |
| PF | 1.17 | | 1.20 | | 1.44 |

# APPENDIX B

# Memorandum

**To:** FCB Group

**CC:** File

**From:** Kent Riley

**Date:** 08/31/98

**Re:** Reactivity effect of $^{10}B$ in water shutter system.

---

Regarding Question 26 of the NRC response to the SER for the fission converter facility, the reactivity effect of adding $^{10}B$ to the water shutter system will be negligible.

The water shutter is located approximately 1 meter from the fuel in the fission converter facility. Also, a 0.020" layer of cadmium is placed in the beamline, after the filter/moderator, a location which is between the fuel in the fission converter tank and the water shutter in the collimator region. This layer of cadmium will prevent any neutrons that might be thermalized and/or backscattered in the water shutter from reaching the fuel in the fission converter facility. The addition of $^{10}B$ to the water shutter will have no impact on the flux of thermal neutrons that reach the fuel after scattering in the water shutter and will therefore have no effect on the reactivity of the fission converter fuel.

# APPENDIX C

**NUCLEAR REACTOR LABORATORY**
AN INTERDEPARTMENTAL CENTER OF
MASSACHUSETTS INSTITUTE OF TECHNOLOGY

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September 10, 1998

# MEMORANDUM

From: Lin-Wen Hu
To: Fission Converter Files
Subject: Radiation Doses after a Material Handling Accident in the Converter

1. A material handling accident in the fission converter is assumed to occur under the following scenario:
   a. The fission converter operated infinitely at 250 kW followed by 4 days of operation at 50 kW to prepare for refueling (TS# 6.6.2.4 (3a)).
   b. During refueling, a fuel element dropped in the converter tank and the whole element (15 plates) was damaged.

2. The fission product activity was estimated using DKPOWR. The total decay activity for the fission converter fueled region 10 minutes after shutdown is $3.483 \times 10^5$ Curies. Assume that the fueled region consists of 11 fuel elements, therefore the fission product activity for each element is $3.166 \times 10^4$ Curies.

3. A recent study (Q. Li, Estimate of Radiation Release for MIT Research Reactor During Design Basis Accident, MS Thesis, Nuclear Engineering Dept., MIT, 1998) analyzed the off-site radiation release from the MITR during design basis accident (DBA) using an updated source term. The calculated total fission product activity in 4 melted fuel plates is $7.16 \times 10^6$, activity for each fission product nuclei is listed in the attached table.

4. The radiation doses resulted from the above-described scenario are then obtained by using the doses calculated for 5 MW MITR and the ratio between these activities. Note that for the MITR DBA it was assumed that 4 fuel plated melted while for the fission converter it was assumed that a whole fuel element was damaged. The following results are estimated for two hours after the accident.

| | MITR (5 MW) | Fission Converter (250 kW) |
|---|---|---|
| Total whole-body, 8 m | 644 mR | 2.8 mR |
| Total whole-body, 21 m | 887 mR | 3.9 mR |
| Thyroid | 112 mR | 0.5 mR |

5. It is thus concluded that radiation doses resulted from a fission converter material mishandling accident are insignificant (< 0.5%) compared to those for the MITR design basis accident.

# Estimate of Radiation Release for MIT Research Reactor During Design Basis Accident

by

Qing Li

Submitted to the Department of Nuclear Engineering
in partial fulfillment of the requirements for the degree of

Master of Science in Nuclear Engineering

at the

MASSACHUSETTS INSTITUTE OF TECHNOLOGY

May 1998



Author ..........................................................
Department of Nuclear Engineering
May 8, 1998

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John A. Bernard
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Thesis Supervisor

Certified by ..........................................................
Jacquelyn C. Yanch
Professor, Nuclear Engineering Department
Thesis Supervisor

Accepted by ..........................................................
Lawrence M. Lidsky
Chairman, Department Committee on Graduate Students

Table A.1: Total Core Fission Product Inventory

| | Isotope | Half-life | $\lambda_i(\sec^{-1})$ | $Y_i$ (%) | $Q_s^i$ ($\times 10^5$ Ci) | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | 5MW | 6MW | 7MW | 8MW | 9MW | 10MW |
| Kr | 85m | 4.36h | 4.41E-5 | 1.5 | 0.6490 | 0.7788 | 0.9086 | 1.0384 | 1.1682 | 1.3000 |
| | 87 | 78m | 1.48E-4 | 2.7 | 1.1700 | 1.4040 | 1.6380 | 1.8720 | 2.1060 | 2.3400 |
| | 88 | 2.77h | 6.95E-5 | 3.7 | 1.6000 | 1.9200 | 2.2400 | 2.5600 | 2.8800 | 3.2000 |
| Xe | 131m | 12.0d | 6.68E-7 | 0.03 | 0.0130 | 0.0156 | 0.0182 | 0.0208 | 0.0234 | 0.0260 |
| | 133m | 2.3d | 3.49E-6 | 0.16 | 0.0692 | 0.0830 | 0.0969 | 0.1107 | 0.1246 | 0.1380 |
| | 133 | 5.27d | 1.52E-6 | 6.5 | 2.8100 | 3.3720 | 3.9340 | 4.4960 | 5.0580 | 5.6200 |
| | 135m | 15.6m | 7.40E-4 | 1.8 | 0.7780 | 0.9336 | 1.0892 | 1.2448 | 1.4004 | 1.5600 |
| | 135 | 9.13h | 2.11E-5 | 6.2 | 0.4130 | 0.4956 | 0.5782 | 0.6608 | 0.7434 | 0.8260 |
| | 138 | 17m | 6.79E-4 | 5.5 | 2.3800 | 2.8560 | 3.3320 | 3.8080 | 4.2840 | 4.7600 |
| I | 131 | 8.05d | 9.96E-7 | 2.9 | 1.2500 | 1.5000 | 1.7500 | 2.0000 | 2.2500 | 2.5100 |
| | 132 | 2.4h | 8.02E-5 | 4.4 | 1.9000 | 2.2800 | 2.6600 | 3.0400 | 3.4200 | 3.8100 |
| | 133 | 20.8h | 9.25E-6 | 6.5 | 2.8100 | 3.3720 | 3.9340 | 4.4960 | 5.0580 | 5.6200 |
| | 134 | 52.5m | 2.20E-5 | 7.6 | 3.2900 | 3.9480 | 4.6060 | 5.2640 | 5.9220 | 6.5700 |
| | 135 | 6.68h | 2.89E-5 | 5.9 | 2.5500 | 3.0600 | 3.5700 | 4.0800 | 4.5900 | 5.1000 |
| Br | 83 | 2.4h | 8.02E-5 | 0.48 | 0.2080 | 0.2496 | 0.2912 | 0.3328 | 0.3744 | 0.4150 |
| | 84 | 30m | 3.85E-4 | 1.1 | 0.4760 | 0.5712 | 0.6664 | 0.7616 | 0.8568 | 0.9510 |
| Cs | 134 | 2.0y | 1.10E-8 | 0.0* | 2.8600 | 3.4320 | 4.0040 | 4.5760 | 5.1480 | 5.7200 |
| | 136 | 13d | 6.17E-7 | 0.006* | 0.4140 | 0.4968 | 0.5796 | 0.6624 | 0.7452 | 0.8280 |
| | 137 | 26.6y | 8.27E-10 | 5.9 | 2.3100 | 2.7720 | 3.2340 | 3.6960 | 4.1580 | 4.6200 |
| Rb | 86 | 19.5d | 4.11E-7 | 2.8E-5* | 0.6120 | 0.7344 | 0.8568 | 0.9792 | 1.1016 | 1.2200 |
| Te | 127m | 90d | 8.82E-8 | 0.056 | 0.0242 | 0.0290 | 0.0339 | 0.0387 | 0.0436 | 0.0484 |
| | 127 | 9.3h | 2.07E-5 | 0.25 | 0.1080 | 0.1296 | 0.1512 | 0.1728 | 0.1944 | 0.2160 |
| | 129m | 33d | 2.43E-7 | 0.34 | 0.1470 | 0.1764 | 0.2058 | 0.2352 | 0.2646 | 0.2940 |
| | 129 | 72m | 1.60E-4 | 1.0 | 0.4320 | 0.5184 | 0.6048 | 0.6912 | 0.7776 | 0.8650 |
| | 131m | 30h | 6.42E-5 | 0.44 | 0.1900 | 0.2280 | 0.2660 | 0.3040 | 0.3420 | 0.3810 |
| | 131 | 24.8m | 4.66E-4 | 2.9 | 1.2500 | 1.5000 | 1.7500 | 2.0000 | 2.2500 | 2.5100 |
| | 132 | 77h | 2.50E-6 | 4.4 | 1.9000 | 2.2800 | 2.6600 | 3.0400 | 3.4200 | 3.8100 |
| | 133m | 63m | 1.83E-4 | 4.6 | 1.9900 | 2.3880 | 2.7860 | 3.1840 | 3.5820 | 3.9800 |
| | 134 | 44m | 2.63E-4 | 6.7 | 2.9000 | 3.4800 | 4.0600 | 4.6400 | 5.2200 | 5.8000 |

Table A.1: Total Core Fission Product Inventory

| | Isotope | Half-life | $\lambda_i(sec^{-1})$ | $Y_i$ (%) | $Q_s^i$ ($\times 10^5$ Ci) 5MW | 6MW | 7MW | 8MW | 9MW | 10MW |
|---|---|---|---|---|---|---|---|---|---|---|
| Sr | 91 | 97h | 1.99e-5 | 5.9 | 2.5500 | 3.0600 | 3.5700 | 4.0800 | 4.5900 | 5.1000 |
| Ba | 140 | 12.8d | 6.27E-7 | 6.3 | 2.7200 | 3.2640 | 3.8080 | 4.3520 | 4.8960 | 5.4500 |
| Ru | 103 | 41d | 1.96E-7 | 2.9 | 1.2500 | 1.5000 | 1.7500 | 2.0000 | 2.2500 | 2.5100 |
| | 105 | 4.5h | 4.28E-5 | 0.9 | 0.3890 | 0.4668 | 0.5446 | 0.6224 | 0.7002 | 0.7790 |
| | 106 | 1.0y | 2.20E-8 | 0.38 | 0.1640 | 0.1968 | 0.2296 | 0.2624 | 0.2952 | 0.3290 |
| Rh | 103 | 36.5h | 5.27E-6 | 0.9 | 0.3890 | 0.4668 | 0.5446 | 0.6224 | 0.7002 | 0.7790 |
| Tc | 99m | 6.04h | 3.19E-5 | 0.6 | 0.2590 | 0.3108 | 0.3626 | 0.4144 | 0.4662 | 0.5190 |
| Mo | 99 | 67h | 2.88E-6 | 6.1 | 2.6400 | 3.1680 | 3.6960 | 4.2240 | 4.7520 | 5.2800 |
| Sb | 127 | 93h | 2.07E-6 | 0.25 | 0.1080 | 0.1296 | 0.1512 | 0.1728 | 0.1944 | 0.2160 |
| | 129 | 4.6h | 4.32E-5 | 1.0 | 4.3200 | 5.1840 | 6.0480 | 6.9120 | 7.7760 | 8.6500 |
| Nd | 147 | 11.3d | 7.10E-7 | 2.6 | 1.1200 | 1.3440 | 1.5680 | 1.7920 | 2.0160 | 2.2500 |
| La | 140 | 40.2h | 4.79E-6 | 6.3 | 2.7200 | 3.2640 | 3.8080 | 4.3520 | 4.8960 | 5.4500 |
| Ce | 141 | 32d | 2.51E-7 | 6.0 | 2.5900 | 3.1080 | 3.6260 | 4.1440 | 4.6620 | 5.1900 |
| | 143 | 32h | 6.01E-6 | 6.2 | 2.6800 | 3.2160 | 3.7520 | 4.2880 | 4.8240 | 5.3600 |
| | 144 | 290d | 2.76E-8 | 6.1 | 2.6400 | 3.1680 | 3.6960 | 4.2240 | 4.7520 | 5.2800 |
| Zr | 95 | 63d | 1.27E-7 | 6.4 | 2.7700 | 3.3240 | 3.8780 | 4.4320 | 4.9860 | 5.5400 |
| | 97 | 17h | 1.13E-5 | 6.2 | 2.6800 | 3.2160 | 3.7520 | 4.2880 | 4.8240 | 5.3600 |
| Nb | 95 | 35d | 2.29E-7 | 6.4 | 2.7700 | 3.3240 | 3.8780 | 4.4320 | 4.9860 | 5.5400 |

Total: $71.64 \times 10^5$ Ci

Table A.2: Values of $N_s^i/N_{235}^o$ for Neutron-Capture Influenced Isotopes at $\phi_T = 4\times10^{13}$

| Isotope | $N_s^i/N_{235}^o$ |
|---|---|
| Xe 135 | $1.05 \times10^{-5}$ |
| Cs 134 | $1.4 \times10^{-1}$ |
| Cs 136 | $3.6 \times10^{-4}$ |
| Cs 137 | $1.5 \times10^{0}$ |
| Rb 86 | $8.0 \times10^{-4}$ |

Table 5.1: Total Dose at 5 MW

| Component of the Dose | Dose at 8m (Rem) | Dose at 21m (Rem) |
|---|---|---|
| Whole-body : | | |
| Containment Leakage | 1.38E-02 | 1.38E-02 |
| Steel Dome Penetration | 6.60E-03 | 5.09E-02 |
| Shadow Shield Penetration | 4.81E-02 | 2.31E-02 |
| Air Scattering | 2.21E-01 | 2.67E-01 |
| Steel Scattering | 3.54E-01 | 5.32E-01 |
| Total | 0.644 | 0.887 |
| Thyroid: | | |
| Containment Leakage | 1.12E-01 | 1.12E-01 |

The whole body dose which includes gamma and beta dose and the thyroid doses from all sources at the front and back fences are listed below. In the whole body dose, the scattering gamma doses contribute the highest portions, one or two orders of magnitude greater than those from other sources. The results are listed in Tables 5.1 through 5.6. The exclusion area doses as a function of reactor power are plotted in Fig. 5-1. The whole-body dose at 21 meters is greater than that at 8 meters. The thyroid doses at both distances are almost equal.

The regulation gives a limitation of 300 rem for thyroid dose and 25 rem for whole-body dose. Our results show that the doses released in a postulated design basis accident of the MIT Research Reactor at a power level of 5 MW up to 10 MW are well below the limitation.

1

decay power calculation, epri code dkpowr

FISSION CONVERTER 250 KW OPERATION

```
nnuc   =   1   number of nuclides described with fission histories
                 (pos. value = fission rate input for each nuclide)
                 (neg. value = fission density input for each nuclide)

ntpow  =  -2   number of at-power time periods
                 (pos. value =power periods input in seconds)
                 (neg. value =power periods input in hours)

ntcool=    1   number of elapsed cooling times at which decay power desired
                 (pos. value = cooling times input in seconds)
                 (neg. value = cooling times input in hours)
                 (zero value = use built-in default cooling times)

jtunit=    1   output time units
                 1 =seconds
                 2 =hours
                 3 =days

jpunit=    2   output decay power units
                 1 =mev/s
                 2 =watts

jeunit=    2   output decay energy units
                 1 =mev
                 2 =joules
                 3 =watt-h
```

default fractional uncertainties used:

```
          u235 fission recoverable energy   .010
         pu239 fission recoverable energy   .010
          u238 fission recoverable energy   .010
                     reactor power level    .020
```

1

table i

Infinite operation @ 250KW

fission history

| | | #/s, for use with 1979 ans 5.1 standard | | | | #/s, for use with fits to cinder-10 summation calculations | | | | | | total |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| time step | delta time, hours | u235 thermal fission | pu239 thermal fission | u238 fast fission | u238 neutron capture | th232 fast fission | u233 thermal fission | u235 thermal fission | u238 fast fission | pu239 thermal fission | pu241 thermal fission | fission rate, #/s |
| 1 | 1.000E+10 | 7.803E+15 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 7.803E+15 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 7.803E+15 |
| 2 | 9.600E+01 | 1.561E+15 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 1.561E+15 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 1.561E+15 |

1

table ii

4 days @ 50KW

decay power calculated with the 1979 ans 5.1 decay power standard

| cooling | decay power, watts | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| time, | fission products | | | | limited heavy elements | | | f.p. decay power uncertainty, watts | | | |
| seconds | u235 | pu239 | u238 | total | gmax | u239 | np239 | total | u235 | pu239 | u238 | total ** |

| | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 6.0000E+02 | 1.795E+03 | 0.000E+00 | 0.000E+00 | 1.795E+03 | 1.030 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 3.356E+01 | 0.000E+00 | 0.000E+00 | 5.388E+01 |

the ans 5.1 standard limits use of the pulse functions to 10**9 seconds cooling after any power history. however, all of the standard pulse functions result from precise fits to decay power calculated with endf/b-iv data for cooling times to 10**13 seconds. calculated decay powers do not include the absorption correction g.

*gmax upper-bound absorption correction multiplier values are taken from table 10, ans 5.1 decay power standard, which lists gmax values at six points per decade (1.,1.5,2.,4.,6.,8.) for cooling times from 1 to 10**9 seconds. values resulting from linear interpolation or extension of ans 5.1 table 10 tabulated values are marked with an asterisk (*).

**the total uncertainty is calculated from algorithms included in the standard, incorporating the uncertainty in each pulse function, the uncertainty in the energy recoverable from each fissionable nuclide, and the uncertainty in the reactor power.

1 table iii

decay energy calculated with the 1979 ans 5.1 decay power standard

| cooling time, seconds | integrated decay energy from shutdown, joules | | | | | | |
|---|---|---|---|---|---|---|---|
| | fission products | | | | limited heavy elements | | |
| | u235 | pu239 | u238 | total | u239 | np239 | total |
| 6.0000E+02 | 1.239E+06 | 0.000E+00 | 0.000E+00 | 1.239E+06 | 0.000E+00 | 0.000E+00 | 0.000E+00 |

calculated decay energies do not include the effects of neutron absorption.

1 table iv

decay power calculated with pulse function fits to cinder-10 summation calculations using processed endf/b-v data,

fission product decay power, watts

| cooling time, seconds | th232 fast fission | u233 thermal fission | u235 thermal fission | u238 fast fission | pu239 thermal fission | pu241 thermal fission | total |
|---|---|---|---|---|---|---|---|
| 6.0000E+02 | 0.000E+00 | 0.000E+00 | 1.806E+03 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 1.8063E+03 |

calculated decay powers do not include the effects of neutron absorption.

1 table v

decay energy calculated with pulse function fits to cinder-10 summation calculations using processed endf/b-v data,

fission product decay energy, joules

| cooling time, seconds | th232 fast fission | u233 thermal fission | u235 thermal fission | u238 fast fission | pu239 thermal fission | pu241 thermal fission | total |
|---|---|---|---|---|---|---|---|
| 6.0000E+02 | 0.000E+00 | 0.000E+00 | 1.232E+06 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 1.2322E+06 |

calculated decay energies do not include the effects of neutron absorption.

1 table vi

decay activity calculated with pulse function fits to cinder-10 summation calculations using processed endf/b-v data, fission product decay activity, curies

| cooling time, seconds | th232 fast fission | u233 thermal fission | u235 thermal fission | u238 fast fission | pu239 thermal fission | pu241 thermal fission | total |
|---|---|---|---|---|---|---|---|
| 6.0000E+02 | 0.000E+00 | 0.000E+00 | 3.483E+05 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 3.4826E+05 |

calculated f.p. curies do not include the effects of neutron absorption.

1 table vii- 1

fission product decay spectra at cooling time of 6.0000E+02 seconds following shutdown calculated with pulse function fits to results of cinder-10 endf/b-v summation calculations and limited experimental data

beta spectrum, watts

| group | emin, mev | emax, mev | th232 fast fission | u233 thermal fission | u235 thermal fission | u238 fast fission | pu239 thermal fission | pu241 thermal fission | total watts |
|---|---|---|---|---|---|---|---|---|---|
| 1 | .00 | .20 | 0.000E+00 | 0.000E+00 | 7.791E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 7.7907E+01 |
| 2 | .20 | .40 | 0.000E+00 | 0.000E+00 | 9.737E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 9.7366E+01 |
| 3 | .40 | .60 | 0.000E+00 | 0.000E+00 | 9.358E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 9.3576E+01 |
| 4 | .60 | .80 | 0.000E+00 | 0.000E+00 | 9.584E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 9.5837E+01 |
| 5 | .80 | 1.00 | 0.000E+00 | 0.000E+00 | 8.666E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 8.6659E+01 |
| 6 | 1.00 | 1.20 | 0.000E+00 | 0.000E+00 | 7.441E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 7.4410E+01 |
| 7 | 1.20 | 1.40 | 0.000E+00 | 0.000E+00 | 6.047E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 6.0472E+01 |
| 8 | 1.40 | 1.60 | 0.000E+00 | 0.000E+00 | 5.209E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 5.2087E+01 |
| 9 | 1.60 | 1.80 | 0.000E+00 | 0.000E+00 | 4.447E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 4.4470E+01 |
| 10 | 1.80 | 2.00 | 0.000E+00 | 0.000E+00 | 3.625E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 3.6250E+01 |
| 11 | 2.00 | 2.20 | 0.000E+00 | 0.000E+00 | 2.798E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 2.7981E+01 |
| 12 | 2.20 | 2.40 | 0.000E+00 | 0.000E+00 | 2.218E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 2.2179E+01 |
| 13 | 2.40 | 2.60 | 0.000E+00 | 0.000E+00 | 1.696E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 1.6960E+01 |
| 14 | 2.60 | 3.00 | 0.000E+00 | 0.000E+00 | 2.212E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 2.2119E+01 |
| 15 | 3.00 | 4.00 | 0.000E+00 | 0.000E+00 | 2.426E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 2.4260E+01 |
| 16 | 4.00 | 5.00 | 0.000E+00 | 0.000E+00 | 3.547E+00 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 3.5465E+00 |
| 17 | 5.00 | 6.00 | 0.000E+00 | 0.000E+00 | 2.192E-01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 2.1919E-01 |
| 18 | 6.00 | 7.50 | 0.000E+00 | 0.000E+00 | 1.317E-02 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 1.3169E-02 |
| total | .00 | 7.50 | 0.000E+00 | 0.000E+00 | 8.363E+02 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 8.3631E+02 |

gamma spectrum, watts

| group | emin, mev | emax, mev | th232 fast fission | u233 thermal fission | u235 thermal fission | u238 fast fission | pu239 thermal fission | pu241 thermal fission | total watts |
|---|---|---|---|---|---|---|---|---|---|
| 1 | .00 | .10 | 0.000E+00 | 0.000E+00 | 8.300E+00 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 8.2998E+00 |
| 2 | .10 | .20 | 0.000E+00 | 0.000E+00 | 1.818E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 1.8183E+01 |
| 3 | .20 | .40 | 0.000E+00 | 0.000E+00 | 4.849E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 4.8487E+01 |
| 4 | .40 | .60 | 0.000E+00 | 0.000E+00 | 9.697E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 9.6965E+01 |
| 5 | .60 | .80 | 0.000E+00 | 0.000E+00 | 2.811E+02 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 2.8110E+02 |
| 6 | .80 | 1.00 | 0.000E+00 | 0.000E+00 | 1.148E+02 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 1.1480E+02 |
| 7 | 1.00 | 1.20 | 0.000E+00 | 0.000E+00 | 4.239E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 4.2390E+01 |
| 8 | 1.20 | 1.40 | 0.000E+00 | 0.000E+00 | 5.046E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 5.0459E+01 |
| 9 | 1.40 | 1.60 | 0.000E+00 | 0.000E+00 | 1.426E+02 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 1.4263E+02 |
| 10 | 1.60 | 1.80 | 0.000E+00 | 0.000E+00 | 2.349E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 2.3486E+01 |
| 11 | 1.80 | 2.00 | 0.000E+00 | 0.000E+00 | 1.973E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 1.9735E+01 |
| 12 | 2.00 | 2.20 | 0.000E+00 | 0.000E+00 | 2.139E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 2.1390E+01 |
| 13 | 2.20 | 2.40 | 0.000E+00 | 0.000E+00 | 1.975E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 1.9754E+01 |
| 14 | 2.40 | 2.60 | 0.000E+00 | 0.000E+00 | 2.062E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 2.0616E+01 |
| 15 | 2.60 | 3.00 | 0.000E+00 | 0.000E+00 | 1.270E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 1.2696E+01 |
| 16 | 3.00 | 4.00 | 0.000E+00 | 0.000E+00 | 1.210E+01 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 1.2099E+01 |
| 17 | 4.00 | 5.00 | 0.000E+00 | 0.000E+00 | 1.080E+00 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 1.0795E+00 |

| | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| 18 | 5.00 | 6.00 | 0.000E+00 | 0.000E+00 | 8.838E-02 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 8.8381E-02 |
| 19 | 6.00 | 7.50 | 0.000E+00 | 0.000E+00 | 1.876E-03 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 1.8761E-03 |
| total | .00 | 7.50 | 0.000E+00 | 0.000E+00 | 9.343E+02 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 9.3426E+02 * |

* the sum of beta and gamma group spectra may differ slightly from total decay power calculated in table iv for this cooling time. spectra do not include the effects of neutron absorption.

# APPENDIX D

NUCLEAR REACTOR LABORATORY
AN INTERDEPARTMENTAL CENTER OF
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August 17, 1998

## MEMORANDUM

From: Lin-Wen Hu
To: Fission Converter Files
Subject: Rev. Complete Loss of the Fission Converter Primary Coolant flow with Automatic Cadmium Curtain Closure

1. This calculation was updated using the calculated fission converter power as a function of the cadmium shutter height. A linear power profile was used previously.

2. The calculated was performed using MathCAD. All other assumptions remain the same as stated in the memo dated February 29, 1997.

3. The calculated maximum fuel temperature is 139.2°C at 20 seconds after initiation of the transient. The fuel temperature then decreases because of the decreasing fission converter power. Bulk boiling is predicted to occur about 10 seconds after the initiation of the transient. The equilibrium quality increases to about 0.39 at 60 seconds.

8/11/98

**Fission Converter Complete Loss of Primary Coolant Flow Analysis**

**assumptions:**

1. **Step change of Primary Coolant Flow to zero**
2. **Reactor fails to scram automatically, Cd curtain starts to drop after 1 s instrument delay time.**
3. **Initial power is 250 kW, primary flow is 100 gpm, Hot channel factor = 1.53**
4. **No convection cooling**
5. **Heat capabilities of fuel, clad, and coolant in the core region are the only heat sink.**
6. **Neglect the heat transfer to other structure materials**

**thermal conductivity (W/m C)**

$kfuel := 42.1 \quad kclad := 186 \quad kcrud := 2.08 \quad kl := 0.68 \quad kv := 16.1\cdot10^{-3}$

**thickness (m)**

$dfuel := 0.0381\cdot10^{-2} \quad dclad := 0.0508\cdot10^{-2} \quad dcrud := 2.54\cdot10^{-5} \quad dgap := 0.112\cdot10^{-2}$

**mass (kg)**

$Mfuel := \frac{0.083}{2} \quad Mclad := \frac{0.096}{2}$

**coolant channel volume (m^3)**

$Vchannel := \frac{8.243\cdot10^{-5}}{2}$

**heat transfer area on one side of fuel plate (assume no fin effect) (m^2)**

$Area := \frac{0.0635}{2}$

**density (kg/m^3)**

$\rho l(T) := 1000.1 + 0.0026863\cdot T - 0.0054424\cdot T^2 + 1.2324\cdot10^{-5}\cdot T^3 \quad \rho v := 0.7$

**heat capability (W/kg C) (Ref. McGuire's Thesis)**

$cpfuel(T) := \frac{1120.9\cdot T + 2.22\cdot10^6}{3675} \quad cpclad(T) := \frac{1187\cdot T + 2.41\cdot10^6}{2712.6}$

**enthalpy (J/kg)** $hg := 2682\cdot10^3 \quad hf := 444.46\cdot10^3 \quad hfg := hg - hf$

**effective heat transfer coefficient between fuel and coolant**

$$U(Xe) := \text{if}\left[Xe<0.0, \frac{1}{\left(\frac{dfuel}{kfuel}+\frac{dclad}{kclad}+\frac{dcrud}{kcrud}+\frac{\frac{dgap}{2}}{kl}\right)}, \frac{1}{\left[\frac{dfuel}{kfuel}+\frac{dclad}{kclad}+\frac{dcrud}{kcrud}+\frac{\frac{dgap}{2}}{(Xe\cdot kv+(1-Xe)\cdot kl)}\right]}\right]$$

$t\_shut := 60.0$ **Cd curtain closes in 1 minute** $U(0.0) = 1.18\cdot 10^3$ $U(1.0) = 28.73$

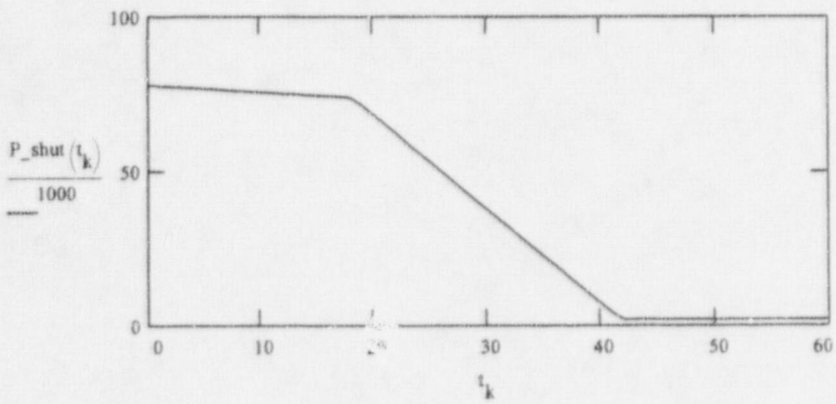
**The power changes as a function of shutter position is obtained from Kiger's thesis**

**closing rate** $Rate := \frac{164}{t\_shut}$

$t1 := \frac{82-32}{Rate}$ $t2 := \frac{82+32}{Rate}$ $t1 = 18.293$ $t2 = 41.707$

$$P\_shut(t) := \text{if}\left[t\le t1, \left(78-t\cdot\frac{4}{t1}\right)\cdot 10^3, \text{if}\left[t\le t2, \left[74-(t-t1)\cdot\frac{72}{(t2-t1)}\right]\cdot 10^3, 2\cdot 10^3\right]\right]$$

$k := 0..60$ $t_k := k$




$$P(t,t\_shut) := \text{if}\left[t\le t\_shut+1, \text{if}\left[t\le 1.0, \frac{250\cdot 10^3}{11\cdot 15\cdot 2}\cdot 1.53, \left(\frac{250\cdot 10^3}{11\cdot 15\cdot 2\cdot 78\cdot 10^3}\cdot 1.53\cdot P\_shut(t-1)\right)\right], 2.5\right]$$

$j := 1..6100$ $dt := 0.01$ $t_j := (j-1)\cdot dt$

**Initial conditions**

$Tw_0 := 64$ $Hw_0 := 4193\cdot Tw_0$ $Xe_0 := 0.0$

**(this is the hot channel outlet temeperature corresponding to the steady-state average core outlet temeperature of 55 C)**

$Tf_0 := Tw_0 + \frac{P(0.0,t\_shut)}{U(0.0)\cdot Area}$ $Tf_0 = 94.94$ **fuel temp increases upon loss of flow**

**Results**

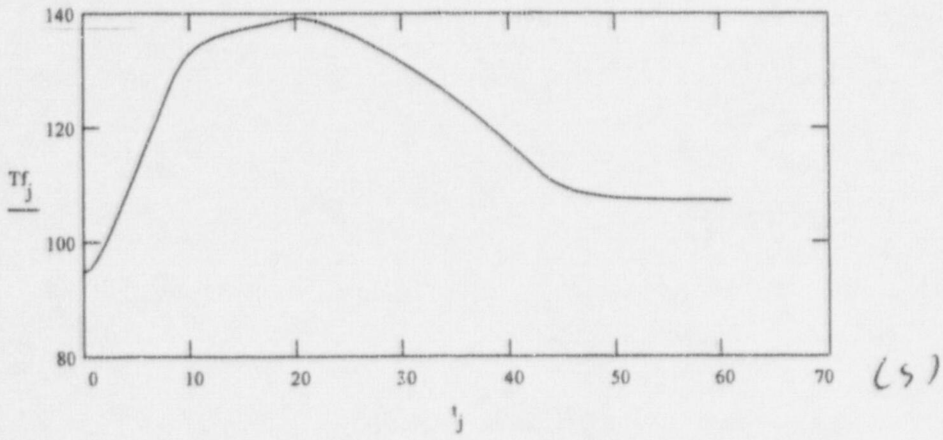



**Figure 1 Fuel Temperature**

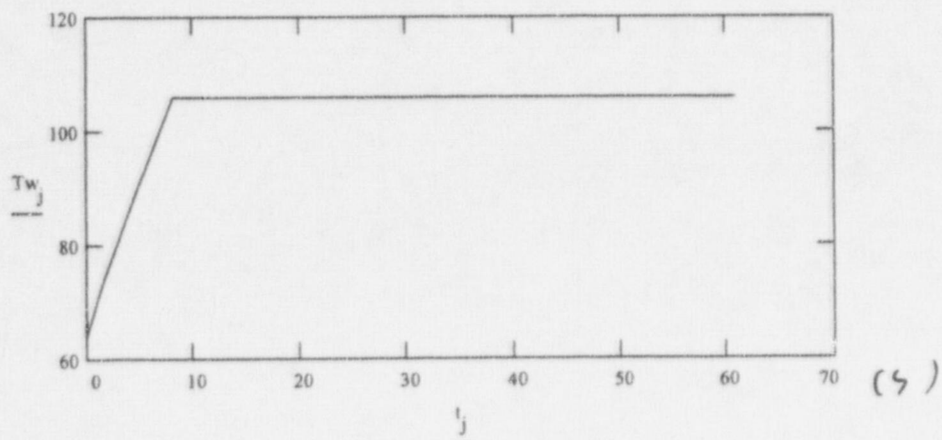



**Figure 2 Coolant Temperature**

$$\text{mass}_j = \left[\rho l\left(Tw_j\right)\cdot\left(1 - Xe_j\right) + \rho v\cdot Xe_j\right]\cdot \text{Vchannel}$$

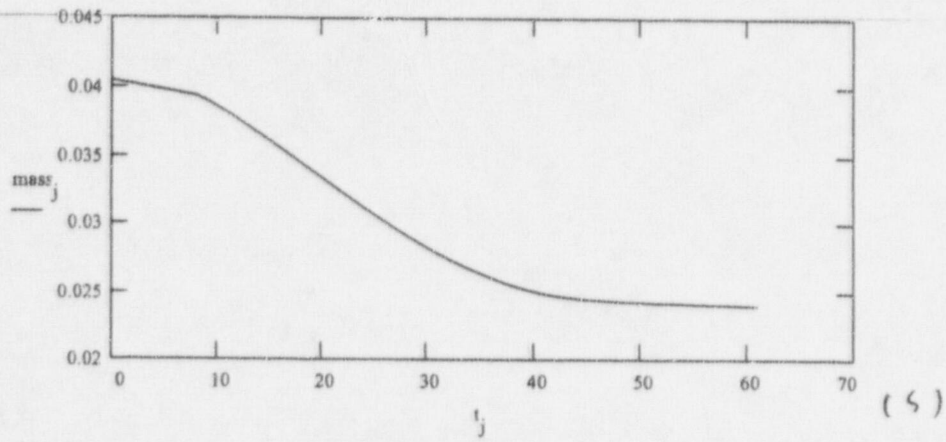



**Figure 3 Coolant Mass in the channel**

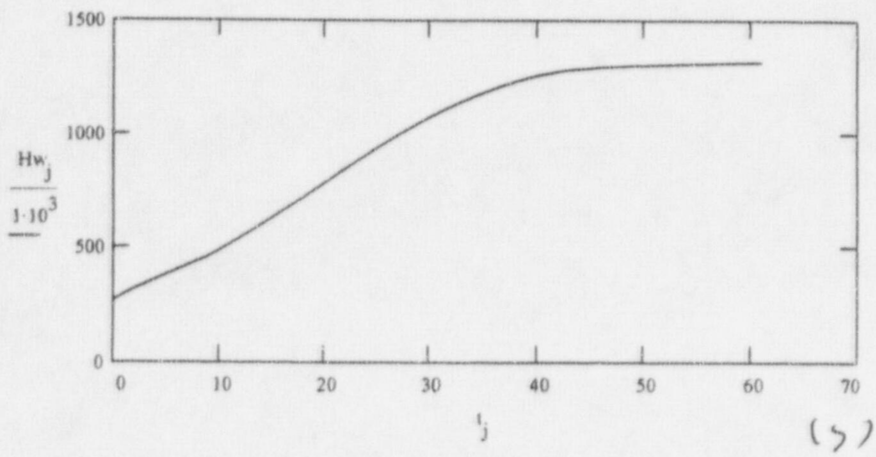



**Figure 4 Coolant Specific Enthalpy**

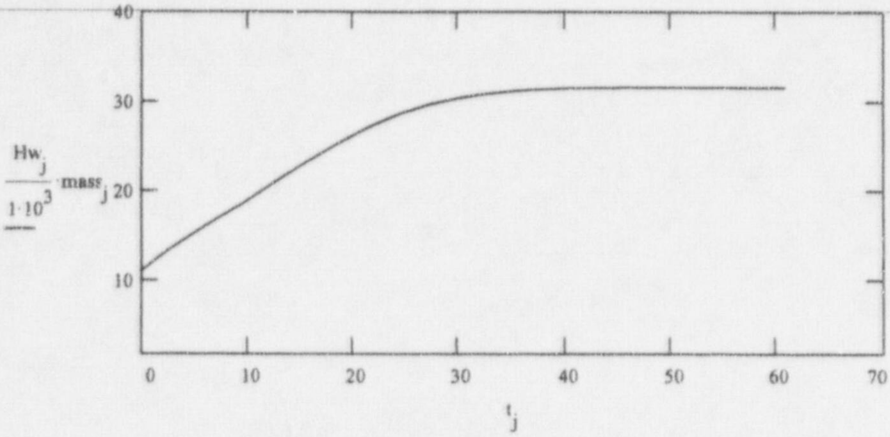



**Figure 5 Coolant Enthalpy**

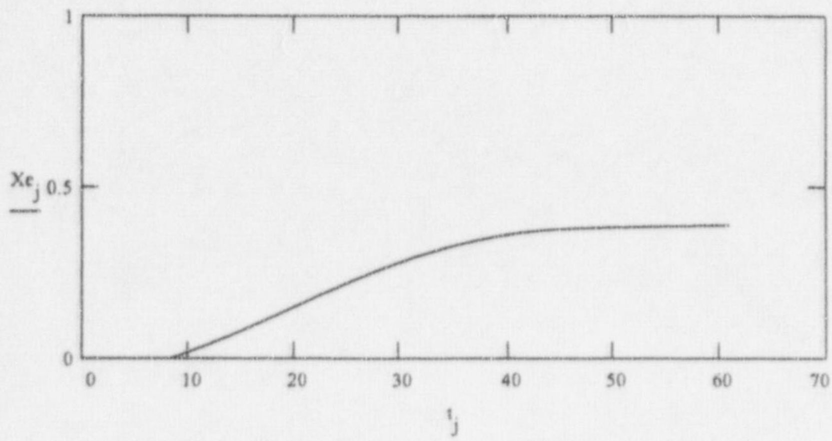



**Figure 6 Coolant Equilibrium Quality**

## Check for total energy balance

### Total energy deposition

$$Qd := \int_{0}^{t\_shut} P(t, t\_shut)\, dt \qquad Qd = 3.555 \cdot 10^{4}$$

### Total energy absorbed in the system

$$Qfuel := \int_{Tf_0}^{Tf_{6100}} (Mfuel \cdot cpfuel(T) + Mclad \cdot cpclad(T))\, dT$$

$$Qcoolant := \sum_{j=1}^{6100} \left[ \left(Hw_j - Hw_{j-1}\right) \cdot \left[ Xe_j \cdot \rho v + \left(1 - Xe_j\right) \cdot \rho l\left(Tw_j\right) \right] \right] \cdot Vchannel$$

$$Qa := Qfuel + Qcoolant$$

$Qfuel = 877.443$ $\qquad$ $Qcoolant = 3.471 \cdot 10^{4}$ $\qquad$ $Qa = 3.559 \cdot 10^{4}$

$$Dev := 1 - \frac{Qd}{Qa}$$

$Dev = 8.932 \cdot 10^{-4}$

### Write results to a data file

$i := 0..610$

$A_{i,0} := t_{i \cdot 10}$ $\qquad$ $A_{i,1} := Tf_{i \cdot 10}$ $\qquad$ $A_{i,2} := Tw_{i \cdot 10}$ $\qquad$ $A_{i,3} := mass_{i \cdot 10}$ $\qquad$ $A_{i,4} := Hw_{i \cdot 10} \cdot mass_{i \cdot 10}$

$A_{i,5} := Hw_{i \cdot 10}$ $\qquad$ $A_{i,6} := Xe_{i \cdot 10}$

WRITEPRN(FC) := A

**Solve for energy equations**

$$\begin{bmatrix} Tf_j \\ Hw_j \\ Tw_j \\ Xe_j \end{bmatrix} = \begin{bmatrix} Tf_{j-1} + \left[P\left(t_j, t\_shut\right) - U\left(Xe_{j-1}\right)\cdot Area\cdot\left(Tf_{j-1} - Tw_{j-1}\right)\right]\cdot\dfrac{dt}{Mfuel\cdot cpfuel\left(Tf_{j-1}\right) + Mclad\cdot cpclad\left(Tf_{j-1}\right)} \\ Hw_{j-1} + \dfrac{dt}{Vchannel\cdot\left[Xe_{j-1}\cdot\rho v + \left(1 - Xe_{j-1}\right)\cdot\rho l\left(Tw_{j-1}\right)\right]}\cdot U\left(Xe_{j-1}\right)\cdot Area\cdot\left(Tf_{j-1} - Tw_{j-1}\right) \\ if\left(Hw_{j-1} \le hf, \dfrac{Hw_{j-1}}{4193}, 106\right) \\ if\left(Hw_{j-1} \le hf, 0.0, if\left(\dfrac{Hw_{j-1} - hf}{hfg} \le 1.0, \dfrac{Hw_{j-1} - hf}{hfg}, 1.0\right)\right) \end{bmatrix}$$

# APPENDIX E

# NUCLEAR REACTOR LABORATORY

AN INTERDEPARTMENTAL CENTER OF
MASSACHUSETTS INSTITUTE OF TECHNOLOGY

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Activation Analysis
Coolant Chemistry
Nuclear Medicine
Reactor Engineering

September 28, 1998

## MEMORANDUM

From: Lin-Wen Hu
To: Fission Converter Files
Subject: Thermal Hydraulic Limits with Ten fuel Elements Operations

1. The fission converter operation with one sample assembly in the fueled region (or ten fuel elements) has been analyzed based on the thermal hydraulic limit calculations. It is assumed that no excessive bypass flow is caused by the sample assembly. The LCO for flow disparity ($F_f \times d_f \geq 0.80$) should be satisfied. The LCO for the hot channel power generation ($F_{HC} \times F_p \leq 1.53$ ) also applies.

2. For forced convection, there is not significant change in SL and LSSS for ten fuel elements. For 300 kW, the $\Delta T$ is less than 0.2°C for both SL and LSSS. This is because both the flow rate and the power increase in the hot channel. The results are attached with the memo. It is thus concluded that the thermal hydraulic limits for forced convection remain the same for 10 or 11 fuel elements.

3. For natural convection, the $\Delta T$ is about 5°C for LSSS. This is because the natural convection flow rate is driven by the inlet/outlet temperature difference. With reduced total flow area, a higher buoyancy force is needed to offset the higher friction pressure drop. The current SL and LSSS for natural convection thus cannot be used for 10 fuel elements.

Limit calculation (mp = 45 gpm; H = 2.1
10 fuel elements)

```
FC LIMITS CALCULATION (new flux dist, new coolant H, 45 gpm) 7/19/97
----------------------------------------------
50.0e3,25.2,2.83,3.77,2.10                      LSSS coolant height is 2.10m
1
0.1,0.1
0.
0.,0.
----------------------------------------------
-1.87,0.41,2.95,-0.68,0.5136,1.492              pump coast down
2.03e-3,9.3e-3,0.0508,-1.02,0.0,1               HL primary
3.8870e-5,1.6792e-4,7.04e-3,0.0,7.30,500        HX primary
2.03e-3,1.8e-2,0.0508,1.02,0.0,2                CL primary
0.134,0.135,0.363,-1.007,0.0,2                  DC 1
0.014,0.037,0.119,-2.65,0.0,2                   DC 2
0.039,0.027,0.162,-0.379,0.0,2                  DC 3
0.00258,0.00684,0.04064,2.65,0.5,2              Bypass channel
1.2490e-4,8.2434e-5,2.1864e-3,0.66,2.05,150     Core
0.114,0.226,0.187,1.99,0.0,1                    FG
0.357,0.359,0.484,1.007,0.0,1                   Mixing Area
0.032,0.427,0.203,-7.08,4.58,1                  CL secondary
9.003e-5,3.8895e-4,3.010e-3,0.0,7.3,500         HX secondary
0.032,0.468,0.203,6.97,2.17,1                   HL secondary
----------------------------------------------------
11,7.62e-4,5.08e-4,2.54e-5,0.05588,0.5683,1.9
0.9205                                          ! 0.9205
2.0E-4
-------------------------------------------------------
1.0,0.0
1.53                                            ! for D2O
0.852,0.892,0.989,1.146,1.174,1.137,1.08,1.039,0.882,0.809     Flux distribution
0.852,0.892,0.989,1.146,1.174,1.137,1.08,1.039,0.882,0.809
1.107,1.05,1.063,1.042,1.027,1.019,1.019,1.032,1.123,1.313     Local Peaking
1.107,1.05,1.063,1.042,1.027,1.019,1.019,1.032,1.123,1.313
0.864                                           ! MITR-II data
1.154,1.265,1.123                               ! new e-factors 1/31/97
1.5
-------------------------------------------------------


1.216,1.271,1.224,1.0

95.0e-2,5.0e-2,0.0e-2,0.0e-2

1.0,1.0,1.0,1.0
```

LSSS calcul[illegible]ons ($\dot{m}_p$ = 45 gpm (2.83 kg/s); H = [illegible] m)

10 fuel elements

Forced convection

| | | ONB | OSV | | |
|---|---|---|---|---|---|
| .260 | 69.369 | 47.458 | 39.800 | 1 | 0 |
| .270 | 67.947 | 45.193 | 37.000 | 1 | 0 |
| .270 | 70.588 | 47.834 | 39.900 | 1 | 0 |
| .280 | 66.468 | 42.872 | 34.100 | 1 | 0 |
| .290 | 71.715 | 48.119 | 39.900 | 1 | 0 |
| .290 | 65.026 | 40.587 | 31.200 | 1 | 0 |
| .290 | 72.388 | 47.949 | 39.400 | 1 | 1 |
| .300 | 63.533 | 38.251 | 28.200 | 1 | 0 |
| .300 | 71.262 | 45.980 | 36.900 | 1 | 1 |
| .310 | 62.095 | 35.970 | 25.300 | 1 | 0 |
| .310 | 70.165 | 44.040 | 34.400 | 1 | 1 |
| .320 | 60.575 | 33.608 | 22.400 | 1 | 0 |
| .320 | 69.009 | 42.042 | 31.800 | 1 | 1 |
| .330 | 59.076 | 31.266 | 19.500 | 1 | 0 |
| .330 | 67.799 | 39.989 | 29.100 | 1 | 1 |
| .340 | 57.598 | 28.945 | 16.600 | 1 | 0 |
| .340 | 66.710 | 38.058 | 26.500 | 1 | 1 |
| .350 | 56.142 | 26.646 | 13.700 | 1 | 0 |
| .350 | 55.567 | 36.072 | 24.000 | 1 | 1 |
| .360 | 54.618 | 24.280 | 10.700 | 1 | 0 |
| .360 | 64.438 | 34.100 | 21.50C | 1 | 1 |
| .370 | 53.119 | 21.938 | 7.700 | 1 | 0 |
| .370 | 63.235 | 32.054 | 18.900 | 1 | 1 |
| .380 | 51.911 | 19.887 | 5.000 | 1 | 0 |
| .380 | 62.141 | 30.117 | 16.400 | 1 | 1 |
| .390 | 53.114 | 20.247 | 5.000 | 1 | 0 |
| .390 | 60.975 | 28.108 | 13.800 | 1 | 1 |
| .400 | 54.315 | 20.606 | 5.000 | 1 | 0 |
| .400 | 59.828 | 26.119 | 11.200 | 1 | 1 |
| .410 | 55.515 | 20.963 | 5.000 | 1 | 0 |
| .420 | 57.506 | 22.112 | 5.900 | 1 | 1 |
| .430 | 57.911 | 21.674 | 5.000 | 1 | 1 |
| .440 | 59.107 | 22.027 | 5.000 | 1 | 1 |
| .450 | 60.302 | 22.379 | 5.000 | 1 | 1 |
| .460 | 61.496 | 22.730 | 5.000 | 1 | 1 |
| .470 | 62.688 | 23.080 | 5.000 | 1 | 1 |
| .480 | 63.879 | 23.428 | 5.000 | 1 | 1 |
| .490 | 65.069 | 23.776 | 5.000 | 1 | 1 |
| .500 | 66.258 | 24.122 | 5.000 | 1 | 1 |
| .510 | 67.446 | 24.467 | 5.000 | 1 | 1 |
| .520 | 68.632 | 24.811 | 5.000 | 1 | 1 |
| .530 | 69.818 | 25.153 | 5.000 | 1 | 1 |
| .540 | 71.002 | 25.495 | 5.000 | 1 | 1 |
| .550 | 72.186 | 25.835 | 5.000 | 1 | 1 |
| .560 | 73.368 | 26.175 | 5.000 | 1 | 1 |
| .570 | 74.549 | 26.513 | 5.000 | 1 | 1 |
| .580 | 75.729 | 26.851 | 5.000 | 1 | 1 |
| .590 | 76.908 | 27.187 | 5.000 | 1 | 1 |
| .600 | 78.086 | 27.522 | 5.000 | 1 | 1 |

limits calculation ( v = 45 gpm, H = 2.1 m, 75 fuel plates, or ~~8~~ elements
5 )

| | | ONB | OSV | | |
|---|---|---|---|---|---|
| .260 | 68.362 | 46.451 | 38.700 | 1 | 0 |
| .270 | 66.859 | 44.106 | 35.800 | 1 | 0 |
| .280 | 65.302 | 41.706 | 32.800 | 1 | 0 |
| .290 | 63.696 | 39.256 | 29.700 | 1 | 0 |
| .290 | 71.934 | 47.495 | 38.900 | 1 | 1 |
| .300 | 62.218 | 36.936 | 26.700 | 1 | 0 |
| .300 | 70.813 | 45.531 | 36.400 | 1 | 1 |
| .310 | 60.700 | 34.575 | 23.800 | 1 | 0 |
| .310 | 69.631 | 43.507 | 33.800 | 1 | 1 |
| .320 | 59.096 | 32.129 | 20.800 | 1 | 0 |
| .320 | 68.481 | 41.514 | 31.200 | 1 | 1 |
| .330 | 57.606 | 29.796 | 17.900 | 1 | 0 |
| .330 | 67.277 | 39.467 | 28.500 | 1 | 1 |
| .340 | 56.046 | 27.393 | 14.900 | 1 | 0 |
| .340 | 66.157 | 37.504 | 25.900 | 1 | 1 |
| .350 | 54.510 | 25.014 | 11.900 | 1 | 0 |
| .350 | 65.014 | 35.519 | 23.400 | 1 | 1 |
| .360 | 52.907 | 22.569 | 8.800 | 1 | 0 |
| .360 | 63.797 | 33.459 | 20.800 | 1 | 1 |
| .370 | 51.420 | 20.239 | 5.800 | 1 | 0 |
| .370 | 62.689 | 31.508 | 18.300 | 1 | 1 |
| .380 | 51.911 | 19.887 | 5.000 | 1 | 0 |
| .380 | 61.507 | 29.484 | 15.700 | 1 | 1 |
| .390 | 53.114 | 20.247 | 5.000 | 1 | 0 |
| .390 | 60.345 | 27.479 | 13.100 | 1 | 1 |
| .400 | 54.315 | 20.606 | 5.000 | 1 | 0 |
| .400 | 59.203 | 25.494 | 10.500 | 1 | 1 |
| .410 | 55.515 | 20.963 | 5.000 | 1 | 0 |
| .410 | 57.991 | 23.439 | 7.800 | 1 | 1 |
| .420 | 56.714 | 21.319 | 5.000 | 1 | 0 |
| .420 | 56.802 | 21.407 | 5.100 | 1 | 0 |
| .420 | 56.890 | 21.495 | 5.200 | 1 | 1 |
| .430 | 57.911 | 21.674 | 5.000 | 1 | 1 |
| .440 | 59.107 | 22.027 | 5.000 | 1 | 1 |
| .450 | 60.302 | 22.379 | 5.000 | 1 | 1 |
| .460 | 61.496 | 22.730 | 5.000 | 1 | 1 |
| .470 | 62.688 | 23.080 | 5.000 | 1 | 1 |
| .480 | 63.879 | 23.428 | 5.000 | 1 | 1 |
| .490 | 65.069 | 23.776 | 5.000 | 1 | 1 |
| .500 | 66.258 | 24.122 | 5.000 | 1 | 1 |
| .510 | 67.446 | 24.467 | 5.000 | 1 | 1 |
| .520 | 68.632 | 24.811 | 5.000 | 1 | 1 |
| .530 | 69.818 | 25.153 | 5.000 | 1 | 1 |
| .540 | 71.002 | 25.495 | 5.000 | 1 | 1 |
| .550 | 72.186 | 25.835 | 5.000 | 1 | 1 |
| .560 | 73.368 | 26.175 | 5.000 | 1 | 1 |
| .570 | 74.549 | 26.513 | 5.000 | 1 | 1 |
| .580 | 75.729 | 26.851 | 5.000 | 1 | 1 |
| .590 | 76.908 | 27.187 | 5.000 | 1 | 1 |
| .600 | 78.086 | 27.522 | 5.000 | 1 | 1 |

LS[illegible]alculations ($\dot{m}_p$ = 45 gpm (2.83 kg/s); H[illegible]1m) 11 fuel elements

| | | | | ONB | OSV |
|---|---|---|---|---|---|
| .270 | 68.038 | 45.284 | 37.100 | 1 | 0 |
| .280 | [illegible] | 43.052 | 34.300 | 1 | 0 |
| .290 | 65.204 | [illegible] | [illegible] | 1 | 0 |
| .290 | 72.478 | 48.039 | 35.500 | 1 | 1 |
| .300 | 63.709 | 38.427 | 28.400 | 1 | 0 |
| .300 | 71.352 | 46.070 | 37.000 | 1 | 1 |
| .310 | 62.281 | 36.157 | 25.500 | 1 | 0 |
| .310 | 70.165 | 44.040 | 34.400 | 1 | 1 |
| .320 | 60.760 | 33.793 | 22.600 | 1 | 0 |
| .320 | 69.009 | 42.042 | 31.800 | 1 | 1 |
| .330 | 59.260 | 31.450 | 19.700 | 1 | 0 |
| .330 | 67.886 | 40.076 | 29.200 | 1 | 1 |
| .340 | 57.781 | 29.128 | 16.800 | 1 | 0 |
| .340 | 66.710 | 38.058 | 26.500 | 1 | 1 |
| .350 | 56.323 | 26.828 | 13.900 | 1 | 0 |
| .350 | 65.567 | 36.072 | 24.000 | 1 | 1 |
| .360 | 54.889 | 24.550 | 11.000 | 1 | 0 |
| .360 | 64.438 | 34.100 | 21.500 | 1 | 1 |
| .370 | 53.387 | 22.206 | 8.000 | 1 | 0 |
| .370 | 63.326 | 32.145 | 19.000 | 1 | 1 |
| .380 | 51.911 | 19.887 | 5.000 | 1 | 0 |
| .380 | 62.141 | 30.117 | 16.400 | 1 | 1 |
| .390 | 53.114 | 20.247 | 5.000 | 1 | 0 |
| .390 | 61.065 | 28.198 | 13.900 | 1 | 1 |
| .400 | 54.315 | 20.606 | 5.000 | 1 | 0 |
| .400 | 59.918 | 26.208 | 11.300 | 1 | 1 |
| .410 | 55.515 | 20.963 | 5.000 | 1 | 0 |
| .410 | 58.790 | 24.238 | 8.700 | 1 | 1 |
| .420 | 56.714 | 21.319 | 5.000 | 1 | 0 |
| .420 | 57.594 | 22.200 | 6.000 | 1 | 1 |
| .430 | 57.911 | 21.674 | 5.000 | 1 | 1 |
| .440 | 59.107 | 22.027 | 5.000 | 1 | 1 |
| .450 | 60.302 | 22.379 | 5.000 | 1 | 1 |
| .460 | 61.496 | 22.730 | 5.000 | 1 | 1 |
| .470 | 62.688 | 23.080 | 5.000 | 1 | 1 |
| .480 | 63.879 | 23.428 | 5.000 | 1 | 1 |
| .490 | 65.069 | 23.776 | 5.000 | 1 | 1 |
| .500 | 66.258 | 24.122 | 5.000 | 1 | 1 |

L[illegible]s calculation (natural convection, 1[illegible] elements)

| | | ONB | OSV | | |
|---|---|---|---|---|---|
| 25.000 | 60.000 | 98.353 | .275 | 0 | 1 |
| 25.000 | 61.000 | 99.042 | .277 | 1 | 1 |
| 25.000 | 62.000 | 99.742 | .280 | 1 | 1 |
| 25.000 | 63.000 | 100.451 | .282 | 1 | 1 |
| 25.000 | 64.000 | 101.169 | .284 | 1 | 1 |
| 25.000 | 65.000 | 101.897 | .286 | 1 | 1 |
| 25.000 | 66.000 | 102.632 | .288 | 1 | 1 |
| 25.000 | 67.000 | 103.376 | .290 | 1 | 1 |
| 25.000 | 68.000 | 104.128 | .292 | 1 | 1 |
| 25.000 | 69.000 | 104.886 | .294 | 1 | 1 |
| 25.000 | 70.000 | 105.651 | .296 | 1 | 1 |

$\Delta T = 1°C$

Note: 25 kW at 65°C for 11 fuel elements. See attached



ICT4PRINT

FC SL & LSSS calculations (see input file for parameters)
ΔT c for natural circulation

| | | | | ONB | OSV |
|---|---|---|---|---|---|
| 13.000 | 79.000 | 101.196 | .247 | 0 | 1 |
| 13.000 | 80.000 | 102.049 | .249 | 1 | 1 |
| 14.000 | 78.000 | 101.454 | .252 | 0 | 1 |
| 14.000 | 79.000 | 102.226 | .254 | 1 | 1 |
| 14.000 | 80.000 | 103.024 | .257 | 1 | 1 |
| 15.000 | 77.000 | 108.405 | .202 | 1 | 1 |
| 15.000 | 78.000 | 108.863 | .205 | 1 | 1 |
| 15.000 | 79.000 | 109.135 | .210 | 1 | 1 |
| 15.000 | 80.000 | 109.061 | .218 | 1 | 1 |
| 16.000 | 76.000 | 101.002 | .270 | 0 | 1 |
| 16.000 | 77.000 | 101.840 | .272 | 1 | 1 |
| 16.000 | 78.000 | 102.689 | .274 | 1 | 1 |
| 16.000 | 79.000 | 103.531 | .275 | 1 | 1 |
| 16.000 | 80.000 | 104.374 | .277 | 1 | 1 |
| 17.000 | 75.000 | 100.930 | .277 | 0 | 1 |
| 17.000 | 76.000 | 101.757 | .279 | 1 | 1 |
| 17.000 | 77.000 | 102.585 | .280 | 1 | 1 |
| 17.000 | 78.000 | 103.420 | .282 | 1 | 1 |
| 17.000 | 79.000 | 104.258 | .284 | 1 | 1 |
| 17.000 | 80.000 | 105.097 | .286 | 1 | 1 |
| 18.000 | 74.000 | 100.816 | .283 | 0 | 1 |
| 18.000 | 75.000 | 101.638 | .285 | 1 | 1 |
| 18.000 | 76.000 | 102.473 | .287 | 1 | 1 |
| 18.000 | 77.000 | 127.597 | .150 | 1 | 1 |
| 18.000 | 78.000 | 126.827 | .156 | 1 | 1 |
| 18.000 | 79.000 | 104.959 | .293 | 1 | 1 |
| 18.000 | 80.000 | 105.790 | .295 | 1 | 1 |
| 19.000 | 72.000 | 131.446 | .135 | 1 | 1 |
| 19.000 | 73.000 | 130.676 | .139 | 1 | 1 |
| 19.000 | 74.000 | 129.906 | .144 | 1 | 1 |
| 19.000 | 75.000 | 102.340 | .293 | 1 | 1 |
| 19.000 | 76.000 | 103.186 | .295 | 1 | 1 |
| 19.000 | 77.000 | 104.021 | .297 | 1 | 1 |
| 19.000 | 78.000 | 104.811 | .299 | 1 | 1 |
| 19.000 | 79.000 | 105.641 | .301 | 1 | 1 |
| 19.000 | 80.000 | 106.472 | .303 | 1 | 1 |
| 20.000 | 67.000 | 135.294 | .124 | 1 | 1 |
| 20.000 | 68.000 | 134.524 | .127 | 1 | 1 |
| 20.000 | 69.000 | 133.755 | .130 | 1 | 1 |
| 20.000 | 71.000 | 99.798 | .293 | 0 | 1 |
| 20.000 | 72.000 | 100.597 | .295 | 1 | 1 |
| 20.000 | 73.000 | 101.400 | .297 | 1 | 1 |
| 20.000 | 74.000 | 102.209 | .299 | 1 | 1 |
| 20.000 | 75.000 | 103.022 | .301 | 1 | 1 |
| 20.000 | 76.000 | 103.839 | .303 | 1 | 1 |
| 20.000 | 77.000 | 104.659 | .305 | 1 | 1 |
| 20.000 | 78.000 | 105.482 | .307 | 1 | 1 |
| 20.000 | 79.000 | 106.311 | .309 | 1 | 1 |
| 20.000 | 80.000 | 107.139 | .311 | 1 | 1 |
| 21.000 | 70.000 | 99.690 | .299 | 0 | 1 |
| 21.000 | 71.000 | 100.478 | .301 | 1 | 1 |
| 21.000 | 72.000 | 101.273 | .303 | 1 | 1 |
| 21.000 | 73.000 | 102.073 | .305 | 1 | 1 |
| 21.000 | 74.000 | 102.878 | .307 | 1 | 1 |
| 21.000 | 75.000 | 103.687 | .309 | 1 | 1 |
| 21.000 | 76.000 | 104.500 | .311 | 1 | 1 |
| 21.000 | 77.000 | 105.316 | .313 | 1 | 1 |
| 21.000 | 78.000 | 106.136 | .315 | 1 | 1 |
| 21.000 | 79.000 | 106.959 | .317 | 1 | 1 |
| 21.000 | 80.000 | 107.784 | .319 | 1 | 1 |
| 22.000 | 69.000 | 99.578 | .304 | 0 | 1 |
| 22.000 | 70.000 | 100.357 | .306 | 1 | 1 |
| 22.000 | 71.000 | 101.142 | .308 | 1 | 1 |
| 22.000 | 72.000 | 101.932 | .310 | 1 | 1 |
| 22.000 | 73.000 | 102.728 | .312 | 1 | 1 |
| 22.000 | 74.000 | 103.529 | .315 | 1 | 1 |
| 22.000 | 75.000 | 104.334 | .317 | 1 | 1 |
| 22.000 | 76.000 | 105.143 | .319 | 1 | 1 |
| 22.000 | 77.000 | 105.956 | .321 | 1 | 1 |
| 22.000 | 78.000 | 106.772 | .323 | 1 | 1 |
| 22.000 | 79.000 | 107.591 | .325 | 1 | 1 |
| 22.000 | 80.000 | 108.412 | .327 | 1 | 1 |
| 23.000 | 68.000 | 99.464 | .309 | 0 | 1 |
| 23.000 | 69.000 | 100.233 | .311 | 1 | 1 |
| 23.000 | 70.000 | 101.008 | .313 | 1 | 1 |
| 24.000 | 67.000 | 99.351 | .313 | 0 | 1 |
| 24.000 | 68.000 | 100.108 | .316 | 1 | 1 |
| 24.000 | 69.000 | 100.872 | .318 | 1 | 1 |
| 24.000 | 70.000 | 101.643 | .320 | 1 | 1 |
| 25.000 | 66.000 | 99.238 | .318 | 0 | 1 |
| 25.000 | 67.000 | 99.986 | .320 | 1 | 1 |
| 25.000 | 68.000 | 100.738 | .322 | 1 | 1 |
| 25.000 | 69.000 | 101.499 | .325 | 1 | 1 |
| 25.000 | 70.000 | 102.265 | .327 | 1 | 1 |
| 26.000 | 65.000 | 99.132 | .322 | 0 | 1 |
| 26.000 | 66.000 | 99.865 | .324 | 1 | 1 |
| 26.000 | 67.000 | 100.605 | .327 | 1 | 1 |
| 26.000 | 68.000 | 101.355 | .329 | 1 | 1 |
| 26.000 | 69.000 | 102.111 | .332 | 1 | 1 |
| 26.000 | 70.000 | 102.874 | .334 | 1 | 1 |
| 27.000 | 64.000 | 99.027 | .325 | 0 | 1 |
| 27.000 | 65.000 | 99.747 | .328 | 1 | 1 |
| 27.000 | 66.000 | 100.476 | .331 | 1 | 1 |
| 27.000 | 67.000 | 101.212 | .333 | 1 | 1 |
| 27.000 | 68.000 | 101.957 | .336 | 1 | 1 |
| 27.000 | 69.000 | 102.710 | .338 | 1 | 1 |
| 27.000 | 70.000 | 103.470 | .341 | 1 | 1 |
| 28.000 | 62.000 | 98.230 | .326 | 0 | 1 |
| 28.000 | 63.000 | 98.926 | .329 | 0 | 1 |
| 28.000 | 64.000 | 99.633 | .332 | 1 | 1 |
| 28.000 | 65.000 | 100.504 | .333 | 1 | 1 |
| 28.000 | 66.000 | 101.225 | .336 | 1 | 1 |
| 28.000 | 67.000 | 101.933 | .338 | 1 | 1 |
| 28.000 | 68.000 | 102.661 | .341 | 1 | 1 |
| 28.000 | 69.000 | 103.375 | .344 | 1 | 1 |
| 28.000 | 70.000 | 104.114 | .347 | 1 | 1 |
| 29.000 | 61.000 | 98.436 | .327 | 0 | 1 |
| 29.000 | 62.000 | 99.111 | .330 | 1 | 1 |
| 29.000 | 63.000 | 99.763 | .333 | 1 | 1 |
| 29.000 | 64.000 | 100.437 | .336 | 1 | 1 |
| 29.000 | 65.000 | 101.103 | .339 | 1 | 1 |
| 29.000 | 66.000 | 101.813 | .342 | 1 | 1 |
| 29.000 | 67.000 | 102.545 | .344 | 1 | 1 |
| 29.000 | 68.000 | 103.240 | .347 | 1 | 1 |

inlet hot channel

# APPENDIX F

# Memorandum

**To:** FCB Group

**CC:** File

**From:** Kent Riley

**Date:** 12/02/98

**Re:** Calculation of Nuclear Hot Channel Factors

---

A Monte Carlo model of the FCB fuel tank and beamline, developed by W.S. Kiger (1), has been extensively used to carry out the neutronic design for the MIT Fission Converter Based Epithermal Neutron Beam. This model (developed using MCNP 4a) contains accurate and detailed geometry for all of the major FCB components, including the fuel elements, coolant, fuel tank, as well as beamline components such as the filter/moderator, collimator and photon shield. The MCNP code makes use of continuous energy cross section libraries; energy group approximations are not required. A portion of the MITR-II reactor is also included in the model to accurately account for any albedo effect from the graphite and $D_2O$ reflector region. The source term that drives the FCB model is generated from a similar Monte Carlo model of the 5 MW MITR-II reactor. Developed by E. L. Redmond II (2), this model includes the full MITR-II reactor core (with detail in each fuel element), the light water coolant, the heavy water and graphite reflector regions, as well as some of the MITR-II experimental facilities. The MITR-II model has been benchmarked via thermal neutron flux measurements in several experimental facilities of the MITR-II, including the region in which the FCB will be installed. W.S. Kiger made use of the MITR-II model to determine the neutron source in this region. This source, which is from the 5MW MITR-II, has been used for many design calculations, including the calculation of hot channel factors reported in earlier memoranda (K. Riley 8/31/98, S. Sakamoto 1/13/97).

Since an MCNP model of the proposed 10 MW MITR-II core is not readily available, the reported results for the FCB driven by a 10 MW core have simply been scaled from the 5 MW calculations, where applicable. For example, the magnitude of the power and the flux levels in the FCB are expected to increase in proportion with the power of the MITR reactor core. Although the intensity of the source used in the FCB model will change when the reactor is changed to 10 MW, the energy and spatial distribution of the neutrons is not expected to appreciably change. The shape of the power profile in the fuel plates for the FCB is therefore not expected to change with an increase in the power of the reactor. The hot channel factors for a

10 MW MITR core are therefore identical to those that have been calculated for the 5 MW MITR core.

---


1. W. S. Kiger III, *Neutronic Design of a Fission Converter Based Epithermal Beam for Neutron Capture Therapy*, Nuclear Engineers Thesis, MIT, 1996.

2. E. L. Redmond II, J. C. Yanch, O. K. Harling, "Monte Carlo Simulation of the Massachusetts Institute of Technology Research Reactor," *Nuclear Technology*, : 106, 1-14, (April 1994).

# APPENDIX G

**NUCLEAR REACTOR LABORATORY**
AN INTERDEPARTMENTAL CENTER OF
MASSACHUSETTS INSTITUTE OF TECHNOLOGY

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Activation Analysis
Coolant Chemistry
Nuclear Medicine
Reactor Engineering

November 24, 1998

**MEMORANDUM**

From: Lin-Wen Hu
To: Fission Converter Files
Subject: Determination of SL and LSSS

1. The fission converter safety limits (SL) and limiting safety system settings (LSSS) were determined based on onset of significant voiding (OSV) and onset of nucleate boiling (ONB), respectively. Note that even the actual SL is defined to prevent fuel overheating, OSV is chosen conservatively because it is easier to calculate. It is believed that adequate margin exists between LSSS and SL so that automatic protective actions will terminate the abnormal situation before a safety limit is reached. The following analyses were performed to support that argument.

2. For forced convection operation, the most limiting transient is loss of primary coolant flow. It is assumed that the reactor scrams in 2 seconds after the LSSS condition (300 kW, 60°C, and no forced flow) is reached. One second is due to instrument delay, and one second due to shim blade drop. Calculated results show that the maximum fuel temperature is about 112°C, significantly below the cladding softening temperature 450°C. The coolant temperature in the hot channel will remain below the saturation temperature.

3. For natural convection operation, the most limiting transient is an overpower condition. It is assumed that the operator raises the reactor power at 100 s period inadvertently and cause the CCS to close automatically because of high power. The initial condition is assumed to be 20 kW and 60°C (LSSS) and a one-second instrument delay time is taken into account. The fission converter power is calculated based on the effects of the 100 s period and CCS closure. The maximum calculated fission converter power is 23 kW and it occurs at about 19 seconds. This is lower than the SL power of 27 kW at 60°C. Note that the change in coolant temperature ($T_{mix}$) during this transient is insignificant because of the large coolant volume in the fission converter tank. It is therefore assumed constant in this analysis.

4. The calculations are attached with this memo.

## Fission Converter Complete Loss of Primary Coolant Flow Analysis

**assumptions:**

1. **Step change of Primary Coolant Flow to zero**
2. **Reactor scrams automatically in 2 seconds (1s instrument delay, 1s blade drop)**
3. **Initial power is 300 kW, hot channel outlet temperature is 69 C (this corresponds to average core outlet temperature 60 C)**
4. **Hot channel factor = 1.53**
5. **No convection cooling**
6. **Heat capabilities of fuel, clad, and coolant in the core region are the only heat sink.**
7. **Neglect the heat transfer to other structure materials**

**thermal conductivity (W/m C)**

$kfuel := 42.1$ $\quad$ $kclad := 186$ $\quad$ $kcrud := 2.08$ $\quad$ $kl := 0.68$ $\quad$ $kv := 16.1 \cdot 10^{-3}$

**thickness (m)**

$dfuel := 0.0381 \cdot 10^{-2}$ $\quad$ $dclad := 0.0508 \cdot 10^{-2}$ $\quad$ $dcrud := 2.54 \cdot 10^{-5}$ $\quad$ $dgap := 0.112 \cdot 10^{-2}$

**mass (kg)**

$Mfuel := \frac{0.083}{2}$ $\quad$ $Mclad := \frac{0.096}{2}$

**coolant channel volume (m^3)**

$$Vchannel := \frac{8.243 \cdot 10^{-5}}{2}$$

**heat transfer area on one side of fuel plate (assume no fin effect) (m^2)**

$$Area := \frac{0.0635}{2}$$

**density (kg/m^3)**

$\rho l(T) := 1000.1 + 0.0026863 \cdot T - 0.0054424 \cdot T^2 + 1.2324 \cdot 10^{-5} \cdot T^3$ $\quad$ $\rho v := 0.7$

**heat capability (W/kg C) (Ref. McGuire's Thesis)**

$cpfuel(T) := \frac{1120.9 \cdot T + 2.22 \cdot 10^6}{3675}$ $\quad$ $cpclad(T) := \frac{1187 \cdot T + 2.41 \cdot 10^6}{2712.6}$

**enthalpy (J/kg)** $\quad$ $hg := 2682 \cdot 10^3$ $\quad$ $hf := 444.46 \cdot 10^3$ $\quad$ $hfg := hg - hf$

**effective heat transfer coefficient between fuel and coolant**

$$U(Xe) := \text{if}\left[Xe<0.0, \frac{1}{\left(\frac{dfuel}{kfuel}+\frac{dclad}{kclad}+\frac{dcrud}{kcrud}+\frac{\frac{dgap}{2}}{kl}\right)}, \frac{1}{\left[\frac{dfuel}{kfuel}+\frac{dclad}{kclad}+\frac{dcrud}{kcrud}+\frac{\frac{dgap}{2}}{(Xe\cdot kv+(1-Xe)\cdot kl)}\right]}\right]$$

$t\_scram := 2.0$ **Reactor scrams in 2 seconds** $U(0.0) = 1.18\cdot 10^{3}$ $U(1.0) = 28.73$

$$P(t,t\_scram) := \text{if}\left(t < t\_scram, \frac{300\cdot 10^{3}}{11\cdot 15\cdot 2}\cdot 1.53, 0.0\right)$$

**assume FC power changes linearly after 1s instrument delay time**

$j := 1..300$ $dt := 0.01$ $t_j := (j-1)\cdot dt$

**Initial conditions**

$Tw_0 := 69$ $Hw_0 := 4193\cdot Tw_0$ $Xe_0 := 0.0$

**(this is the hot channel outlet temeperature corresponding to the steady-state average core outlet temeperature of 60 C)**

$Tf_0 := Tw_0 + \frac{P(0.0,t\_scram)}{U(0.0)\cdot Area}$ $Tf_0 = 106.128$ **fuel temp increases upon loss of flow**

**Solve for energy equations**

$$\begin{pmatrix} Tf_j \\ Hw_j \\ Tw_j \\ Xe_j \end{pmatrix} := \begin{bmatrix} Tf_{j-1} + \left[P\left(t_j,t\_scram\right) - U\left(Xe_{j-1}\right)\cdot Area\cdot\left(Tf_{j-1} - Tw_{j-1}\right)\right]\cdot\frac{dt}{\left(Mfuel\cdot cpfuel\left(Tf_{j-1}\right) + Mclad\cdot cpclad\left(Tf_{j-1}\right)\right)} \\ Hw_{j-1} + \frac{dt}{Vchannel\cdot\left[Xe_{j-1}\cdot\rho v + \left(1 - Xe_{j-1}\right)\cdot\rho l\left(Tw_{j-1}\right)\right]}\cdot U\left(Xe_{j-1}\right)\cdot Area\cdot\left(Tf_{j-1} - Tw_{j-1}\right) \\ \text{if}\left(Hw_{j-1} \le hf, \frac{Hw_{j-1}}{4193}, 106\right) \\ \text{if}\left(Hw_{j-1} \le hf, 0.0, \text{if}\left(\frac{Hw_{j-1} - hf}{hfg} \le 1.0, \frac{Hw_{j-1} - hf}{hfg}, 1.0\right)\right) \end{bmatrix}$$

**Results**

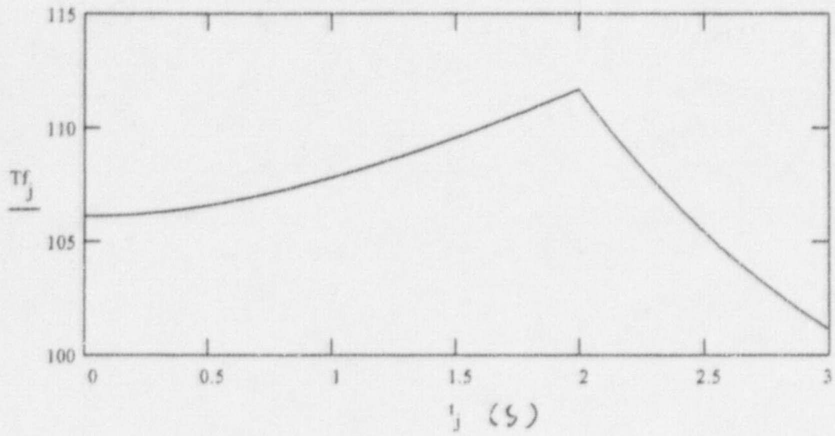



**Figure 1 Fuel Temperature**

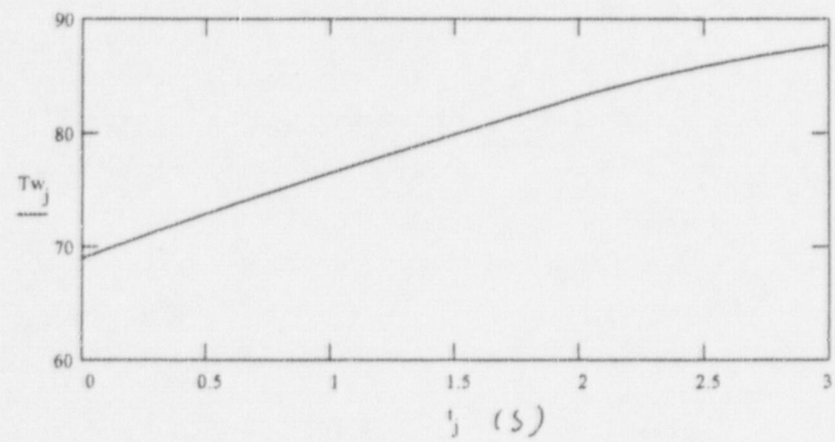



**Figure 2 Coolant Temperature**

## Fission converter maximum power during an overpower transient

t_shut := 60.0 **(time to fully close the CCS)**

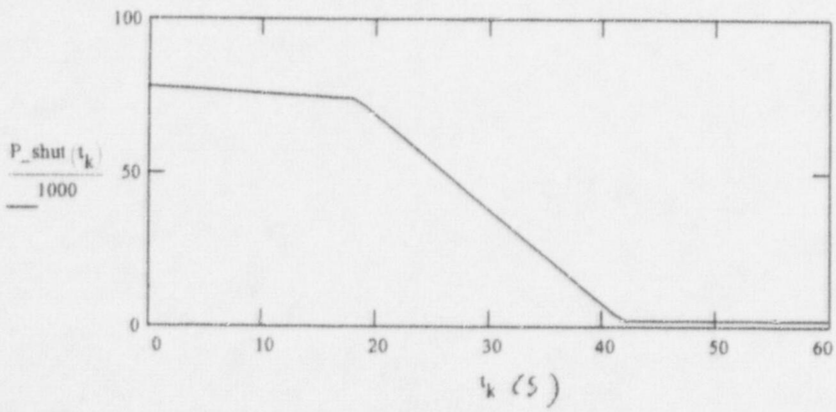
**The power changes as a function of shutter position is obtained from Kiger's thesis**

**closing rate** $\quad Rate := \dfrac{164}{t\_shut}$ **(constant speed)**

$t1 := \dfrac{82 - 32}{Rate} \qquad t2 := \dfrac{82 + 32}{Rate} \qquad t1 = 18.293 \qquad t2 = 41.707$

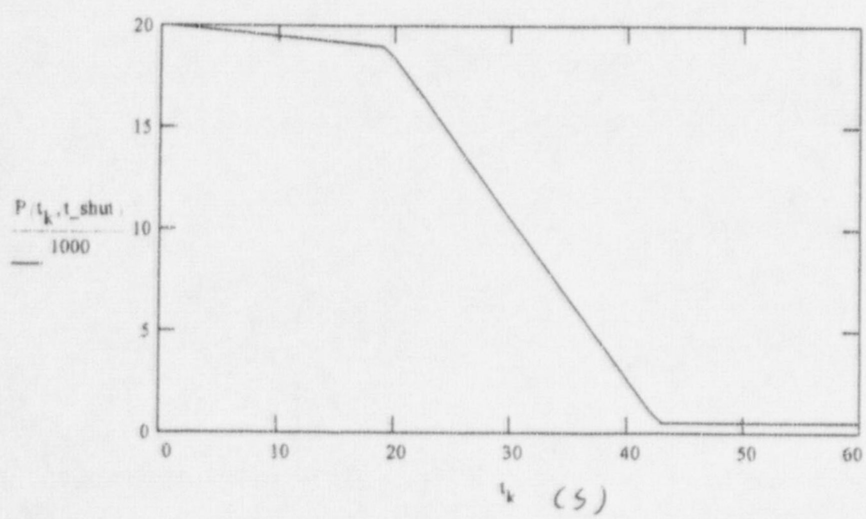
$$P\_shut(t) := \mathrm{if}\left[t \le t1, \left(78 - t\cdot\frac{4}{t1}\right)\cdot 10^3, \mathrm{if}\left[t \le t2, \left[74 - (t - t1)\cdot\frac{72}{(t2 - t1)}\right]\cdot 10^3, 2\cdot 10^3\right]\right]$$

$k := 0..60 \qquad t_k := k$



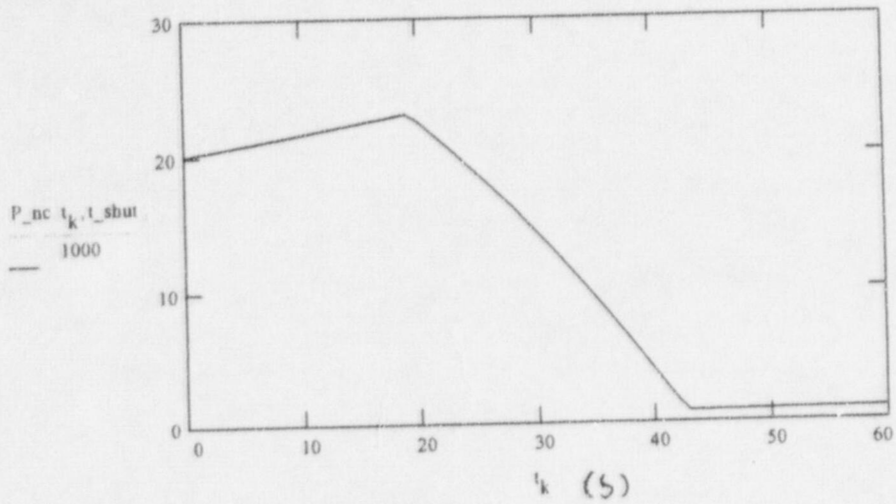

**Normalize the power to 20 kW**

$$P(t, t\_shut) := \mathrm{if}\left[t \le t\_shut + 1, \mathrm{if}\left[t \le 1.0, 20\cdot 10^3, \left(\frac{20\cdot 10^3}{78\cdot 10^3}\cdot P\_shut(t-1)\right)\right], 200\right]$$



**Fission converter power with reactor at 100 s period and CCS closing**

$$P\_nc(t, t\_shut) := P(t, t\_shut) \cdot \exp\left(\frac{t}{100}\right)$$




$i := 15..22$

$t_i := i$

$$\frac{P\_nc(t_i, t\_shut)}{1000}$$

| |
|---|
| 22.325 |
| 22.483 |
| 22.643 |
| 22.803 |
| 22.965 |
| 22.494 |
| 21.748 |
| 20.984 |

**maximum fission converter power during this transient is 23 kW**

# APPENDIX H

# Memorandum

**To:** FCB Group

**CC:** File

**From:** Kent Riley

**Date:** 12/04/98

**Re:** Estimates of dose rates inside FCB medical room with failure of one shutter

---

MCNP calculations of the dose rate inside the medical room for all possible shutter configurations are currently being completed. Though these calculations are still in progress, this memorandum provides an estimate for the dose rate at the patient position with the reactor operating and the failure of one shutter (two shutters closed). MCNP calculations with each shutter closed (and the other two open) have been completed. The values in the table below have been estimated by attenuating the single shutter closure results by the amount appropriate for the other shutter.

| Condition of FCB Shutters (MITR-II at 5 MW) | Dose Rate at Patient Position (Rem/hr) |
|---|---|
| CCS Open, Water and Mechanical Shutters Closed | 0.1 |
| Mechanical Shutter Open, Water Shutter and CCS Closed | 15 |
| Water Shutter Open, CCS and Mechanical Shutter Closed | 0.2 |

Although the MCNP results for single shutter closures are statistically accurate to 10% or better, the uncertainty in the above numbers is much higher. The attenuation of two shutters together can be different than the product of their attenuation when acting alone. It is also important to remember that the above numbers are calculated in the path of the direct beam; general area room dose rates and dose rates outside the beam path will be much lower. Nevertheless, the table above can be used for guidance in identifying failure scenarios that might prohibit access to the room. Failure of the mechanical shutter may prohibit access to the room (without a reactor scram). However it is important to remember that if the mechanical shutter can even be partially closed, dose rates will be dramatically reduced. In any case, a reactor scram will sufficiently reduce dose rates to permit room entry.