SUPPLEMENT I

to the

1979 SAFETY ANALYSIS REPORT

for the

Nuclear Science Center Reactor
Texas A&M University System

30 March 1982

I. INTRODUCTION

Since the preparation of the current NSCR Safety Analysis Report in 1979 the situation regarding the damage to the TRIGA fuel has been resolved. This has necessitated some changes to the technical specifications, which are discussed in the supplement. In addition, several clarifications and changes resulting from numerous reviews of the original document are submitted.

II. TRIGA FUEL DAMAGE

Introduction

On September 27, 1976 three damaged fuel elements were discovered in the core of the NSCR. A report on the discovery and preliminary analysis was submitted to the Division of Reactor Licensing on November 1, 1976.[1] It was believed that no valid conclusion concerning the cause of the damage was possible without metallographic examination of the damaged fuel. Since Argonne National Laboratories (ANL-West) had TRIGA-FLIP fuel and the necessary facilities they agreed to perform these examinations. Because of delays due to Hot Cell modifications and the low priority of this project it was several years before data were collected. The evaluation of the work required an additional several years since it involved personnel from Texas A&M, Argonne National Laboratory, and General Atomic Company. The final report was issued in December 1981[2] with the proposed damage mechanism receiving concensus from those concerned. This report will not repeat the information contained in these reports but will elaborate on the damage mechanism to establish a new pulsing limit.

The Postulated Damage Mechanism

FLIP fuel was fabricated with a hydrogen to zirconium ratio of 1.6. This value was carefully selected to take advantage of several properties of zirconium hydride. Figure 1 reproduces a phase diagram of ZrH_x presented by Simnad.[3] Notice that a H to Zr atom ratio of 1.6 produces the delta phase which is a material that remains in this phase to well over 1000°C. This

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1. "Status Report on Damage to FLIP Fuel During Operation of the NSCR at Texas A&M University", a letter report to the Director, Division of Reactor Licensing, USAEC, from John D. Randall, Director, Texas A&M University Nuclear Science Center, November 1, 1976.

2. "Interpretation of Damage to the FLIP Fuel During Operation of the Nuclear Science Center Reactor at Texas A&M University", by M. T. Simnad, G. B. West, J. D. Randall, W. J. Richards, and D. Stahl. General Atomic Company - Report GA-A16613, December 1981.

3. "The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel", by M. T. Simnad. General Atomic Company - Report E-117-833, February 1980.

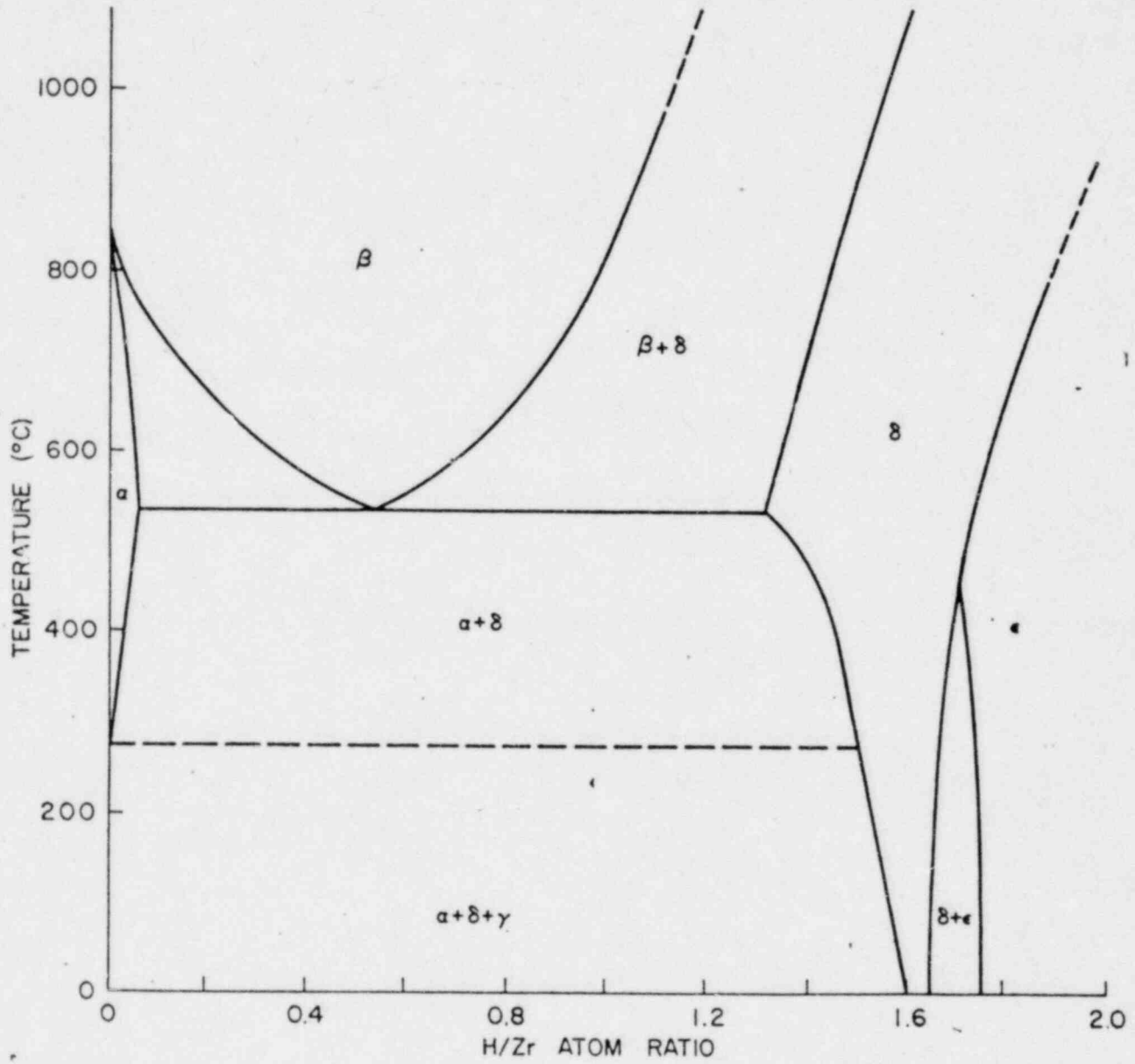

FIGURE 1
ZIRCONIUM HYDRIDE PHASE DIAGRAM

avoids substantial volume changes due to phase transformations that occur at approximately 530°C at lower hydrogen ratios.[4,5] It also produces a material that exhibits very small density changes with varying hydrogen content.[6] An additional advantage of the delta phase is that it has much greater creep strength than the beta phase.[7] Figure 2 reproduces data from Simnad on the creep strength of the beta and delta phases.[8] As can be seen the difference is quite considerable. However, if the fuel is used in a 1 Mw core with a high steady-state utilization factor the ideal properties and characteristics mentioned above will not be totally retained. When subjected to temperature gradients hydrogen will migrate from the hotter fuel regions to the cooler regions with the migration rate being greatly influenced by time, temperature, and the temperature gradient.[9] Figure 3 shows the radial temperature distribution in the highest temperature element in Core III-A operating at 1 Mw steady state. As can be seen there is an approximately 300°C temperature difference across only 1 cm of fuel. Thus, there will be a long term loss of hydrogen from the hotter regions to the cooler outer regions. Since there is negligible migration below 250°C[10] the outer skin of the element will retain the original hydrogen ratio of 1.6. The hydrogen loss in the inner part of the element will cause phase changes which could result in stress cracking.[11] Metallographic analysis showed the presence of alpha and delta phase mixtures in the porous regions of the damaged element with the outer skin of the fuel retaining the

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4. "Safety Analysis Report for the Torrey Pines TRIGA Mark III Reactor", Gulf General Atomic, Inc., Report GA-9064, pg. 3-2, Article 3.1.1, January 5, 1970.
5. "Stability of U-ZrH$_{1.7}$ TRIGA Fuel Subjected to Large Reactivity Insertions", by C. O. Coffer, J. R. Shoptaugh, and W. L. Whittemore, General Atomic Report - GA-6874, pp. 4-5, January 1966.
6. Ref. 3, pg. 2-19.
7. Ref. 3, pg. 2-31.
8. Ref. 3, pg. 2-32.
9. Ref. 3, pg. 2-16.
10. M. T. Simnad, Personal Communication.
11. Ref. 3, pg. 2-17.

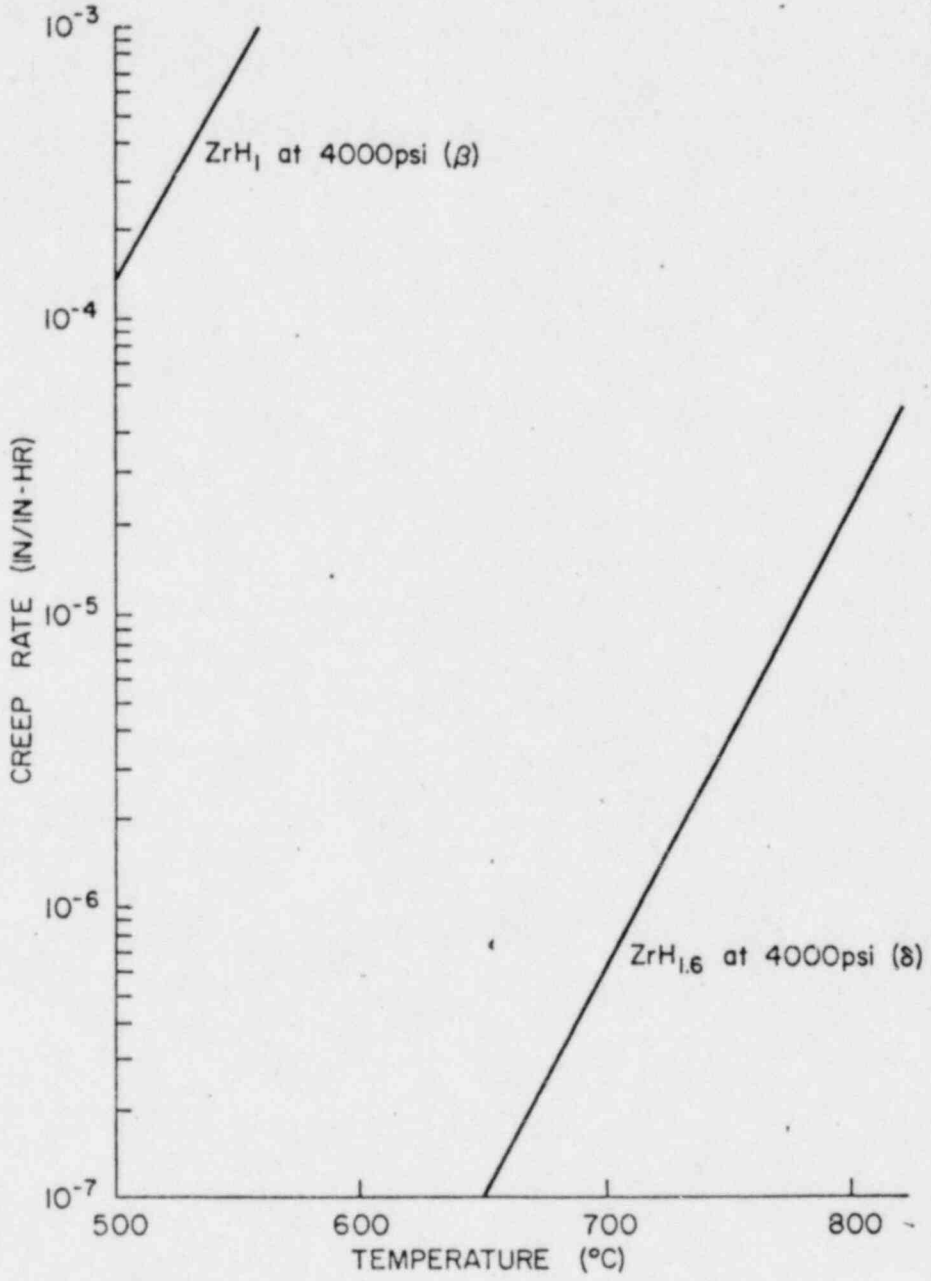

FIGURE 2
CREEP PROPERTIES OF ZIRCONIUM HYDRIDE

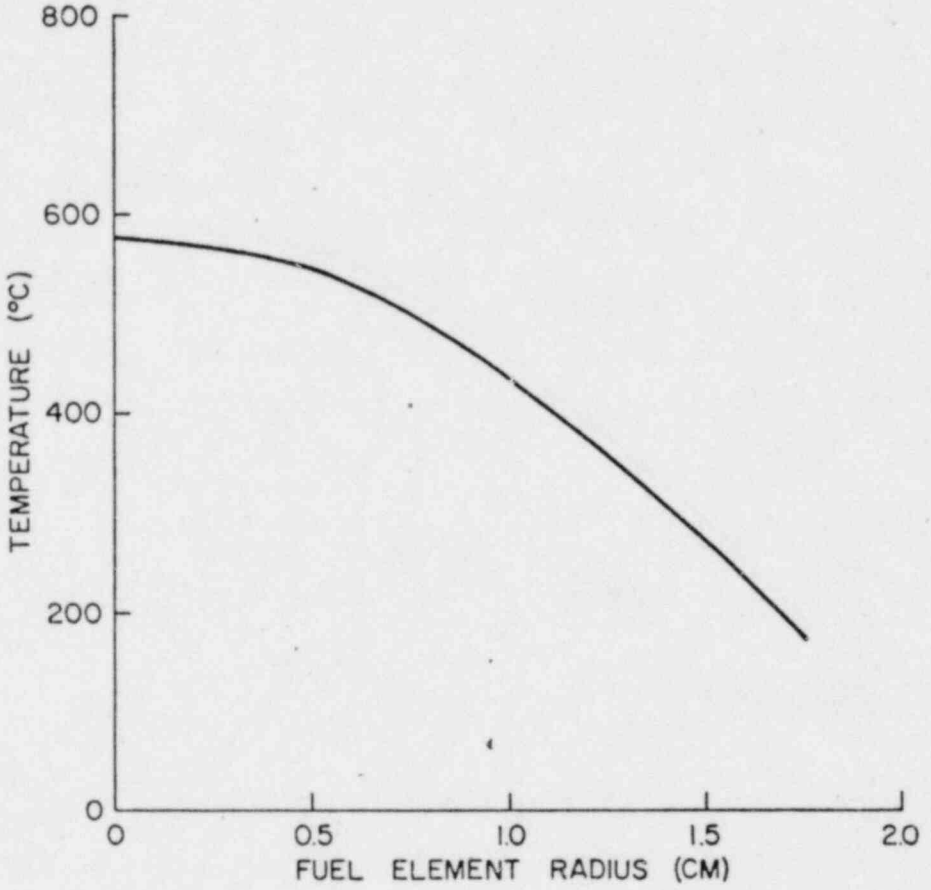

FIGURE 3
RADIAL TEMPERATURE DISTRIBUTION
IN MAXIMUM POWER ELEMENT AT 1Mw, CORE IIIA

delta phase. The porous regions are attributed to the high hydrogen pressures produced by large pulses. The porosity cannot be ascribed to fission product gases since the burnup is too low and they would migrate up the temperature gradient towards the center of the element and not towards the outside.

The migration of the hydrogen to the outer regions of the fuel during long term, hi-power operation resulted in H to Zr ratios greater than 1.6. The hydrogen gas pressures produced in the hydrogen rich regions during a large pulse were sufficient to cause high pressures and rapid hydrogen redistribution. "To produce pressures equivalent to those in $ZrH_{1.6}$ at 1150°C, the H/Zr ratio would have to increase to about 1.85 and be subjected to a temperature of about 880°C".[12] Both of these conditions are consistent with the observed condition of the damaged fuel and temperatures reached for $2.70 pulses.

Thus, the proposed mechanism that is deemed responsible for damaging the fuel is long term, high temperature steady state operation that causes some redistribution of the hydrogen by migration. Large pulses subsequent to the steady state operation result in sufficiently high pressures in the hydrogen rich areas to cause swelling, porosity, and rapid hydrogen redistribution leaving the region below average in hydrogen content. Temperatures above 530°C in this hydrogen depleted region can cause additional volume changes of up to 15%.[13] It is also possible that through continued operation hydrogen would be replenished by redistribution and the process repeated. This mechanism is independent of the erbium content of the fuel and is, therefore, applicable to FLIP and standard TRIGA fuels.

Proposed New Pulsing Tech Spec Limits

Since temperature is the parameter that ultimately determines the damage-free operation of the ZrH_x fuel, it is proposed to incorporate it into the technical specifications. The maximum temperature that a core will experience during pulsing is dependent on core composition and arrangement and on reactivity insertion. This maximum damage-free temperature can be determined most reliably by empirical observations. Consider Figure 4 which shows the positions of the damaged fuel elements closest to the transient rod. If the transient rod is called the "A" position, then those rods closest to it could be said to occupy the "B" ring. Those next farther out would be the "C" ring, using standard TRIGA terminology. Table 1 lists the calculated peak temperatures experienced by these

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12. Ref. 2, pg. 44

13. Ref. 2, pg. 2-23, Simnad Report

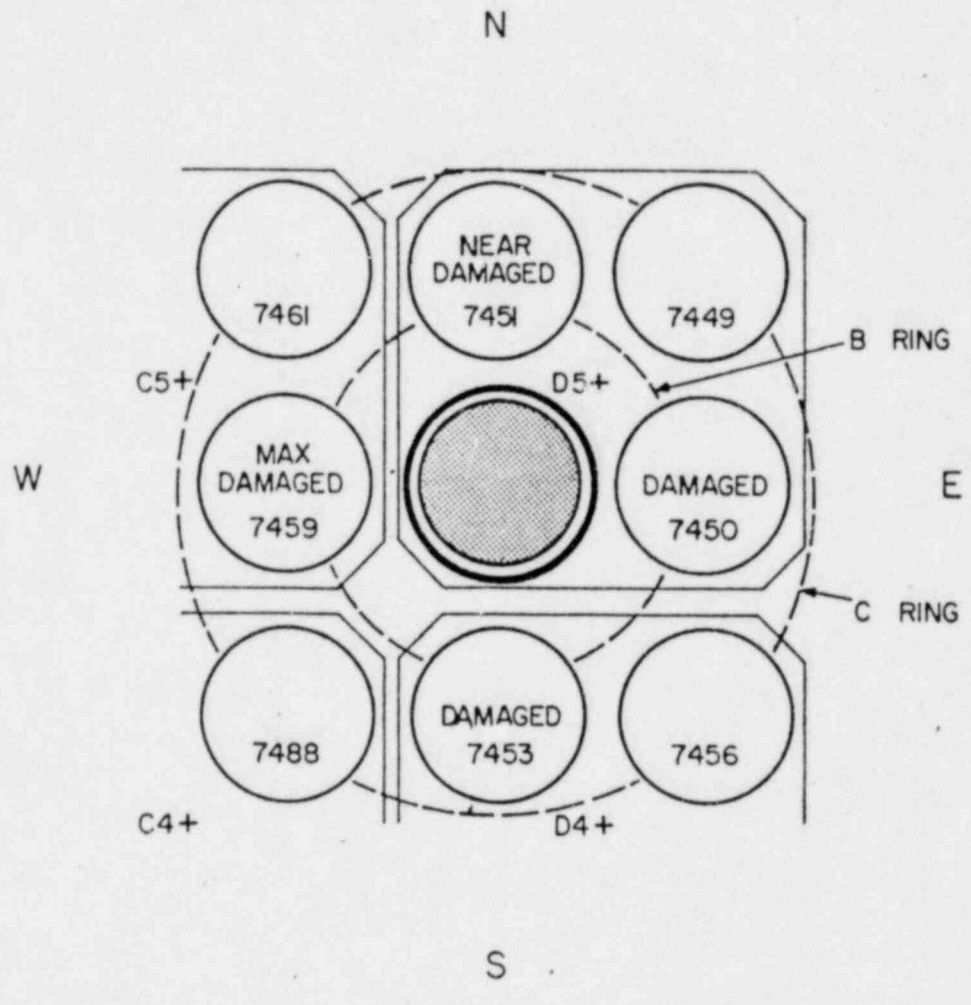

FIGURE 4
POSITION OF FUEL ELEMENTS NEAREST TRANSIENT ROD

TABLE 1

Condition of Fuel Elements Nearest
the Transient Rod

A Position Void Followed Transient Rod

"B Ring"

Fuel Element No.	Peak Temp, °C $2.70 Pulse	Element Condition
7459	908°	Maximum Damage
7453	874°	Damaged
7450	920°	Damaged
7451	890°	Small Bump (Passes go-no go Gauge)

"C Ring"

Fuel Element No.	Peak Temp, °C $2.70 Pulse	Element Condition
7461	659°	No Damage
7449	677°	No Damage
7456	653°	No Damage
7488	637°	No Damage

elements and their observed condition. The three most damaged elements in the "B" ring have calculated temperatures within a range of ± 23°C. All failed the go-no go gauge. The least damaged element passed the go-no go gauge but appeared to have a tiny deformation near the center. However, it has remained in steady state service and at the present time exhibits no external abnormalities. Thus, this element clearly operated near the limit for damage-free pulsing. The fact that the calculated peak temperature for this element was 11°C higher than the average peak temperature of the three damaged elements reflects on the uncertainties in the calculations of the fine structure of the power and temperature distributions generated across the very narrow water gaps. The "C" ring with its much lower temperatures exhibits no damage as would be predicted by the postulated damage mechanism.

Similar results are reported in Reference 2 where comparisons between the Texas A&M reactor and the General Atomic TRIGA are presented.[14] Measured maximum temperatures for both cores are extrapolated to the peak temperature using calculated peaking factors. These results indicate that the temperatures experienced in the G. A. core were approximately 20°C lower than those at Texas A&M. This result is significant in that no damage was experienced at General Atomic even though their core had achieved much more steady state burnup prior to pulsing.

Additional confidence is gained in the proposed mechanism by examining Washington State. As stated in Reference 2 the Washington State TRIGA pulsed with $2.50 insertions but used a water followed transient rod. Calculations at Texas A&M indicate that for a $2.50 pulse insertion the peak core temperature for a water followed rod would be 1010°C, whereas for the Texas A&M void follower it was 877°C. The lack of damage at Washington State is explained by the fact that the core had accumulated only 8 Mw-days of operation at that time. The migration of the hydrogen did not have sufficient time to produce a significant amount of redistribution.

This extreme sensitivity to temperature can be appreciated by examining Figure 5. This shows the equilibrium hydrogen pressure over $ZrH_{1.65}$ as a function of its temperature. Notice that at a temperature of 800°C the pressure is only 23 psi, whereas at 900°C it is 125 psi, and at 1000°C it increases to 425 psi! These pressures are not actually achieved during pulsing because the

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14. Ref. 2, pg. 41

pressure is more a function of the average temperature in the fuel, not the peak temperature, but the relative values are valid for comparison.

To determine a damage-free temperature limit for pulsing notice the arrows on Figure 5 which mark the calculated peak temperatures experienced by the damaged elements. The dashed line at 874°C is defined as the onset of damage since this is the lowest temperature observed for a damaged element and the near damaged element was calculated to reach temperatures 16°C higher. A factor of 2 reduction in equilibrium pressure from this value is considered a reasonable and prudent safety factor. This establishes a limit of 830°C for pulsing in a core that has had greater than 8 Mw-days of steady state burnup. The proposed Technical Specification is:

> Specification: The reactivity to be inserted for pulse operation shall not exceed that amount which will produce a peak fuel temperature of 830°C. In the pulse mode the pulse rod will be limited by mechanical means so that the reactivity insertion will not inadvertently exceed the maximum value.

To arrive at the pulse insertion that would produce the limiting peak core temperature, the peaking factors must be calculated. This would determine the limiting temperature as measured by the thermocouple for the given core configuration. Figure 6 shows the calculated peak core temperatures for Core III-A as a function of pulse insertions. The 830°C limit on peak core temperature indicates that the maximum allowable reactivity insertion would be $2.27. For an all FLIP core higher insertions would be permitted due to more favorable peaking factors.

It should be noted that this new temperature limit does not replace the safety limit. The safety limit is determined by the temperature that would initiate clad failure. Since the total hydrogen in the fuel element remains unchanged the safety limit temperature will also be the same.

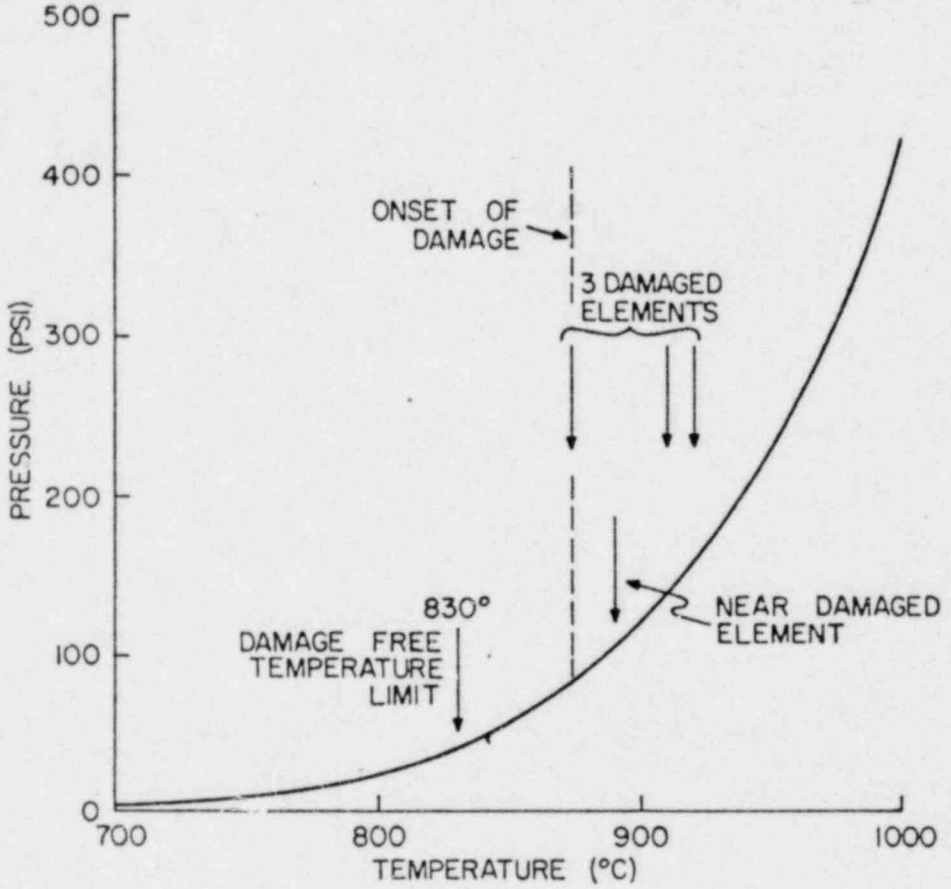

FIGURE 5
EQUILIBRIUM HYDROGEN PRESSURE OVER $ZrH_{1.65}$ vs TEMPERATURE
(FROM FIGURE 2-9, REF 3)

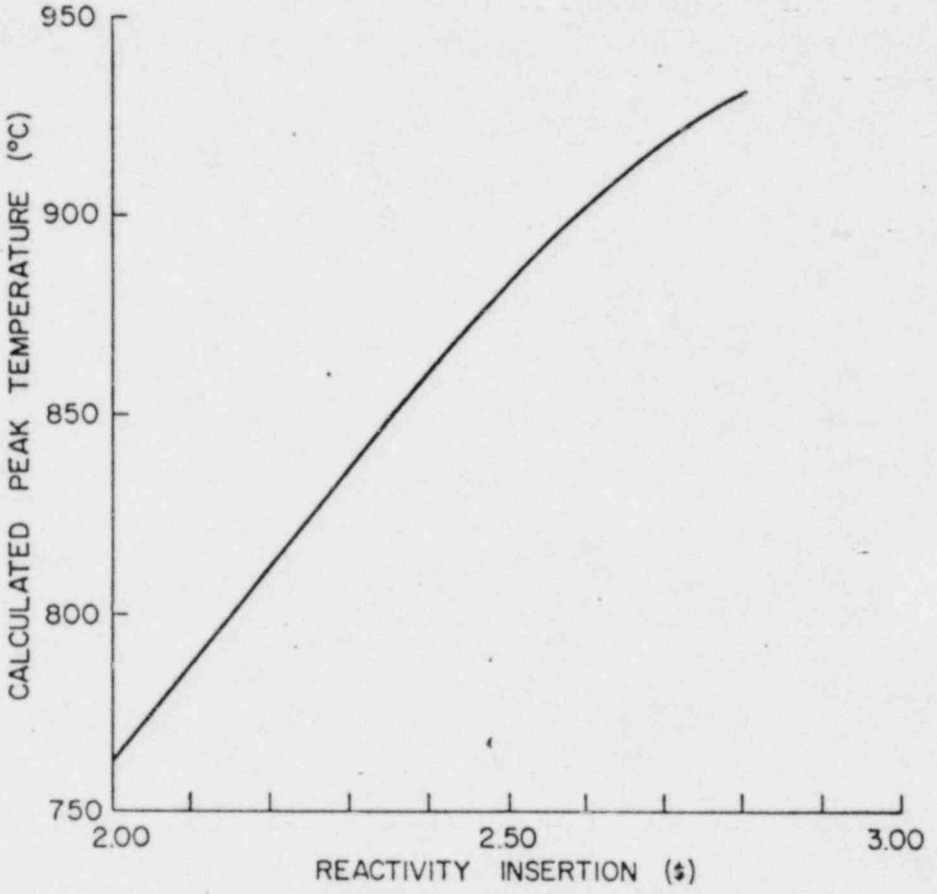

FIGURE 6
PEAK CORE TEMPERATURES vs REACTIVITY INSERTIONS, CORE III A

III. CLARIFICATIONS AND CHANGES TO THE 1979 SAR

The following clarifications and changes are submitted. Replace original pages with updated pages.

Figure 3-6 - This figure has been changed from a photograph to a drawing to improve reproduction and clarity.

Figure 3-24 - The "Full Width at Half Peak Power" Scale has been corrected.

Page 114 - Section B.1 last sentence - Addition

"including drains from labs in laboratory building"

Figure 9-1 - A revised drawing of the "Radioactive Liquid Waste Disposal System" is submitted

Page 116 - Section C.2 last sentence - Remove

"if possible"

Section D.1 - New heading

"Generation of Airborne Radioactive Effluents"

Page 117 - Section 2 - New heading

"Gaseous Effluent Handling"

Section 3 - New heading "Gaseous Effluent Disposal", changed to read:

"Gaseous effluent is released to the environment under controlled conditions through the building stack which is 85 feet in height".

Section 4 - Addition

"Particulate Effluent"

"Experience has shown that particulate activity can be confined to the immediate area in cases of simple spills. Particulate effluent releases at the exhaust stack have remained well below MPC values during the last 5 years of operation at 1 Mw".

Page 119 - Section F - Third sentence of second paragraph changed to read:

"This monitor is equipped with Hi and Low alarm circuits which activate an audible alarm and a warning light indicating a circuit alarm. A Hi alarm causes an automatic shutdown of the air handling system to isolate the facility".

Section F - Change - Last sentence of fifth paragraph changed to read:

"An alarm on this system will automatically shutdown the air handling system to isolate the facility".

Figure 9-2 - Remove the following air monitoring system:

"Delayed Particulate ACT".

Page 121 - Section H.1 - Last sentence should read:

"Personnel monitoring devices will include but not be restricted to beta-gamma and neutron film badges or TLD badges and pocket ionization chambers.

Page 137 - The second sentence at top of page has been changed to read:

"Thus, if a value of 667°C were chosen for the LSSS, it would prevent peak core temperatures from exceeding 800°C in standard fuel elements and 950°C in FLIP fuel elements for all cores that comply with the Technical Specifications".

Figure 3, Appendix II - The ordinate scale on Figure 3 has been corrected to read:

"DECAY HEAT POWER, Watts/Kw OPERATION POWER"

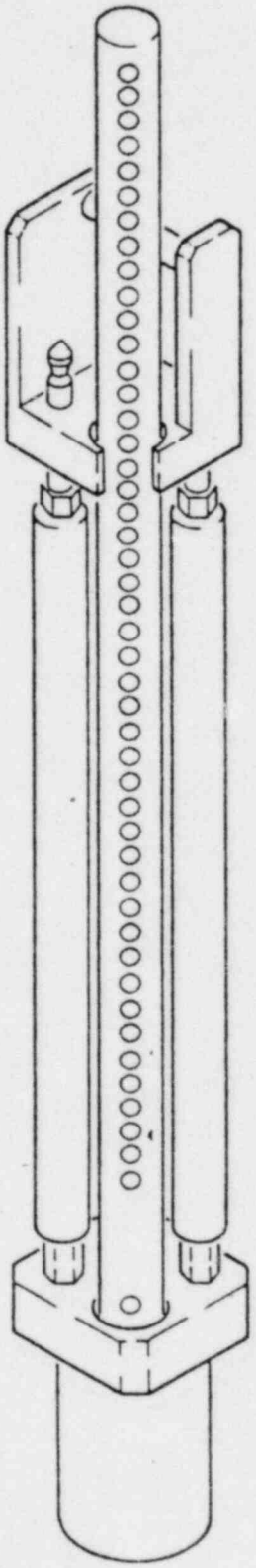

FIGURE 3-6 CONTROL ROD BUNDLE WITH GUIDE TUBE

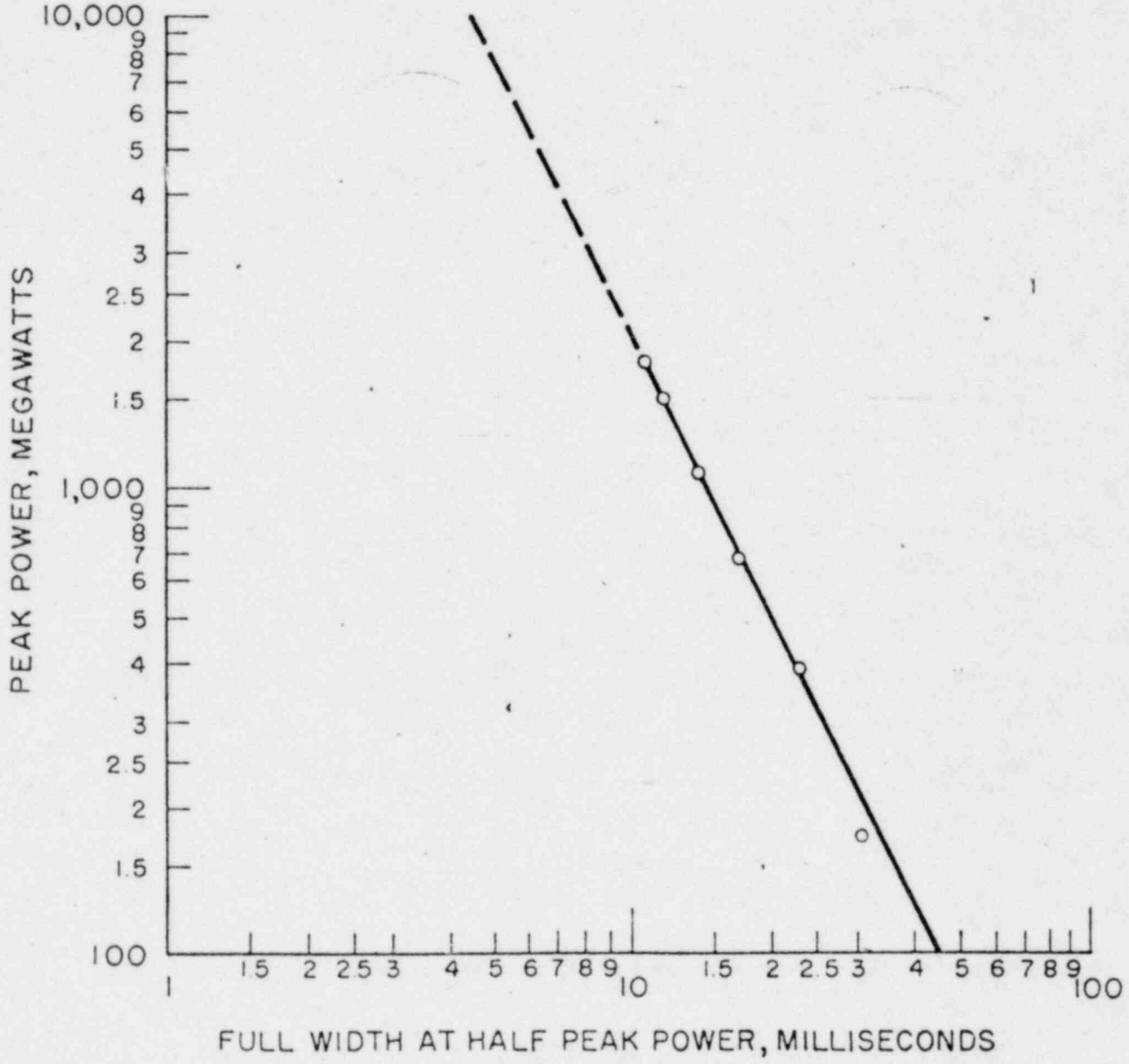

FIGURE 3-24 PEAK POWER VERSUS FULL WIDTH AT HALF PEAK POWER - PROTOTYPE REACTOR

IX. RADIATION PROTECTION AND RADIOACTIVE EFFLUENTS

A. Introduction

All activities will be conducted in such a manner as to comply with 10CFR20 "Standards for Protection Against Radiation". Exposure of individuals and release of radioactivity to the environment will be controlled to maintain compliance with all applicable sections of the regulations. Methods and instrumentation which are used to establish compliance are outlined in the following sections.

B. Liquid Waste

1. Generation of Liquid Waste

Low level liquid waste originates from four primary sources at the Nuclear Science Center. These sources are: (1) floor drains, shower and radio chemistry laboratory on the lower research level; (2) demineralizer room filter and ion bed; (3) condensate from air handling units on mechanical chase; and (4) valve pit sump in cooling equipment room (including drains from labs in laboratory building).

2. Liquid Waste Handling System

Liquid waste flows through common headers to a liquid waste sump located below the grade of the lower research level (See Figures 4-4 and 9-1). Waste is transferred by a sump pump to one of three storage tanks located above grade 200 feet northwest of the building. These tanks have a total storage capacity of 22,000 gallons. Each tank is equipped with an inlet valve, outlet valve, volume indicator and sampling line. There is a valve on the master outflow line which is secured with a keyed supervisor lock. Fresh water is available to the master outflow line for diluting and flushing the liquid waste being dumped to the unrestricted environment.

When a tank is filled, it is valved off, mixed to insure uniformity, sampled, and the activity determined. Care must be taken in the collection and preparation of the liquid waste sample to insure that the concentrations determined are representative of what is being released. Thus the tank must be isolated from incoming water prior to sampling, the liquid should be thoroughly mixed when sampled, and care must be

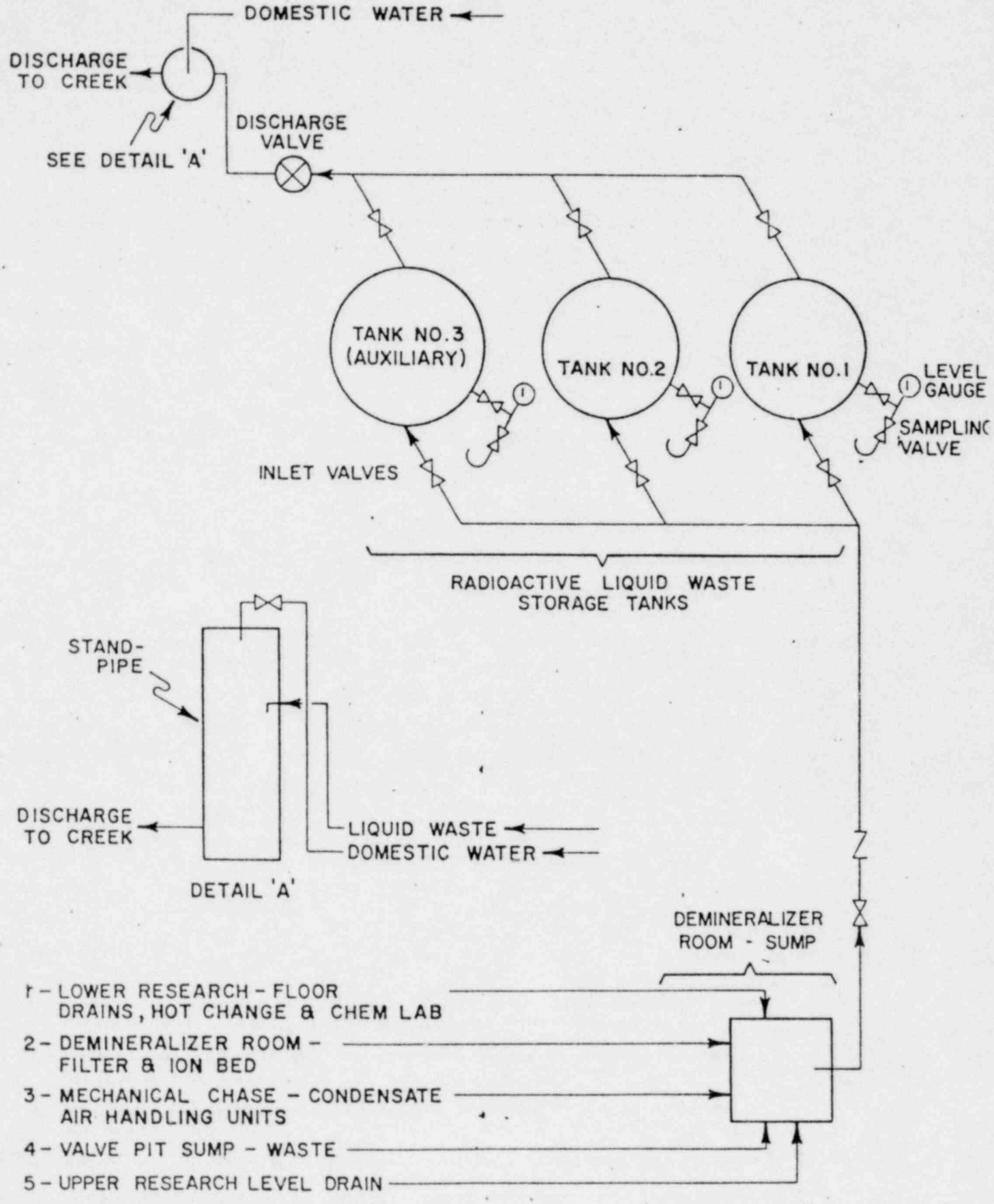

FIGURE 9-1 RADIOACTIVE LIQUID WASTE DISPOSAL SYSTEM

taken in sample preparation to prevent foreign contamination. Based on the activity determination, the waste will either be drained, stored for decay, or diluted with fresh water to release levels in compliance with 10CFR20 limits and drained.

C. Solid Waste

1. Generation of Solid Waste

Solid radioactive waste in the form of rags, paper towels, used laboratory equipment, sample containers, aluminum, etc., is generated in normal operation of the Nuclear Science Center Reactor. Activated materials such as used experimental hardware also are generated.

2. Solid Waste Handling and Disposal

Low level solid waste is accumulated in plastic-lined waste containers located at strategic points throughout the facility. When filled these containers are monitored, the plastic liner sealed and removed, and the waste stored in the radioactive waste storage building (See Figure 2-2). This waste is transferred to the Texas A&M University by-product material license for disposal.

Activated equipment is normally stored in the high level waste storage area adjacent to the outside wall of the irradiation cell. If equipment cannot be reused, it may be transferred in the same manner as low level waste under state regulations.

D. Gaseous and Particulate Waste

1. Generation of Airborne Radioactive Effluents

Production of radioactive gases, primarily ^{41}Ar, comes from dissolved gases in the cooling system, irradiation of air in open beam port tubes, dry tube, pneumatic irradiation systems, and irradiation of air in the irradiation cell.

Nuclear Science Center Technical Report No. 32, "Determination of Argon-41 Production at the Texas A&M Nuclear Science Center Reactor" documents that approximately 4.7 Ci of ^{41}Ar are released on an annual basis. Applying a dilution factor of 5×10^{-3}, the releases produce approximately .8% of the permissible concentration specified in 10CFR20. The results of 4.7 Ci was based on 100 Mw-day operation of the NSCR.

Several important findings were made concerning ^{41}Ar production at the NSC. On a long term basis, the pool accounts for more than 95% of the facility's production. The average cell concentrations were measured versus time for the cell exhaust on and off for 4 hours at 1 Mw. As expected the exhaust-on values were lower than the exhaust

off, with peak values of 6.7×10^{-5} μCi/cc and 1.2×10^{-4} μCi/cc, respectively. The pneumatic system was examined for absolute production on each firing as a function of time at 1 Mw before the first firing, and results showed that the release increased from 6.8 μCi present before reactor startup to a plateau value of about 208 μCi after 6 hours at 1 Mw. In all cases the system was purged of argon after 5 firings. As expected, the dry tube showed no contribution to release, but the beam port measurements showed a level of 2.15×10^{-3} μCi/cc at 1 Mw in Beam Port #1, closest to the core. Thus, although the pool is the major production source in the long run, the other sources can rival the pool release rate on occasion.

2. Gaseous Effluent Handling

The ^{41}Ar which is produced in the beam ports and in the irradiation cell is exhausted directly to the building central exhaust, and thus, through the stack. The ^{41}Ar that is produced by activation of the air which is in the pool water is transferred through the building ventilation system to the central exhaust.

3. Gaseous Effluent Disposal

Gaseous effluent is released to the environment under controlled conditions through the building stack which is 85 feet in height.

4. Particulate Effluent

Experience has shown that particulate activity can be confined to the immediate area in cases of simple spills. Particulate effluent releases at the exhaust stack have remained well below MPC values during the last 5 years of operation at 1 Mw.

E. Dilution Factor Calculations

The equations used in developing the dilution factors calculated below are those presented by F. A. Gifford, Jr.[13,14] These calculations are based on release at ground level and utilize the building dilution factor $D_B + cAu$, where A is the cross sectional area of the building normal to the wind and u is wind speed in meters/second. From reference 13, C is estimated to be 0.5. The cross sectional area of the Nuclear Science Center is 357 m^2.

The equation for the atmospheric dilution factor is:

$$X = \pi \frac{Q}{\sigma_y \sigma_z u} \exp\left\{-1/2 \left(\frac{y^2}{\sigma_y^2} + \frac{h^2}{\sigma_2^2}\right)\right\}$$

13. F. A. Gifford, Jr., Nuclear Safety, December, 1960.

14. F. A. Gifford, Jr., Nuclear Safety, July, 1961.

The calculations presented in this section clearly show that a dilution factor of 200 can be utilized by the Nuclear Science Center for stack release without endangering the public health and safety.

F. Facility Air Monitoring System

Argon-41 activity is monitored with a gas detector which utilizes a 3" NaI (Tl) scintillation crystal and a gamma spectrometer. The detector which is calibrated for ^{41}Ar activity, continuously samples air from the building exhaust plenum. The system is equipped with an adjustable contact which provides an audible alarm on the console and a warning light on the console and in the reception room. The system is shown schematically in Figure 9-2.

Stack particulate activity is monitored with a moving tape type, continuous air monitor. This monitor samples air from the building exhaust plenum. This monitor is equipped with Hi and Low alarm circuits which activate an audible alarm and a warning light indicating a circuit alarm. A Hi alarm causes an automatic shutdown of the air handling system to isolate the facility.

Building gas activity is monitored by a gas detector which is calibrated for ^{41}Ar activity. Air is sampled on the chase level by this monitor. An alarm circuit actuates an audible alarm when preset alarm levels are reached and a warning light is actuated.

Building particulate activity is monitored with a moving tape type, continuous air monitor. Air is sampled on the chase level by this monitor. An alarm circuit actuates an audible alarm when preset alarm levels are reached and a warning light is actuated.

A fission product monitor with a low sensitivity for the detection of gases is used to essentially eliminate high detector backgrounds due to ^{41}Ar gas. The air sampling region is located approximately one foot above the pool surface at the reactor bridge. Air is drawn through the line and through the monitor filter paper using an air suction pump. A G.M. detector monitors the filter paper 180° from the point of collection. The monitor primarily detects particulates that are produced by decay of fission product gases collected in the sampling line. This monitor is equipped with an alarm circuit which activates an audible alarm and a warning light. An alarm on this system will automatically shut down the air handling system to isolate the facility.

Area Radiation Monitors

The area radiation monitoring system provides a continuous indication at the reactor console and in the reception room of the radiation level in each of the monitored areas. An adjustable contact on each indicating meter provides an alarm on the console annunciator panel. A red light on the indicating meter and on the detector identify

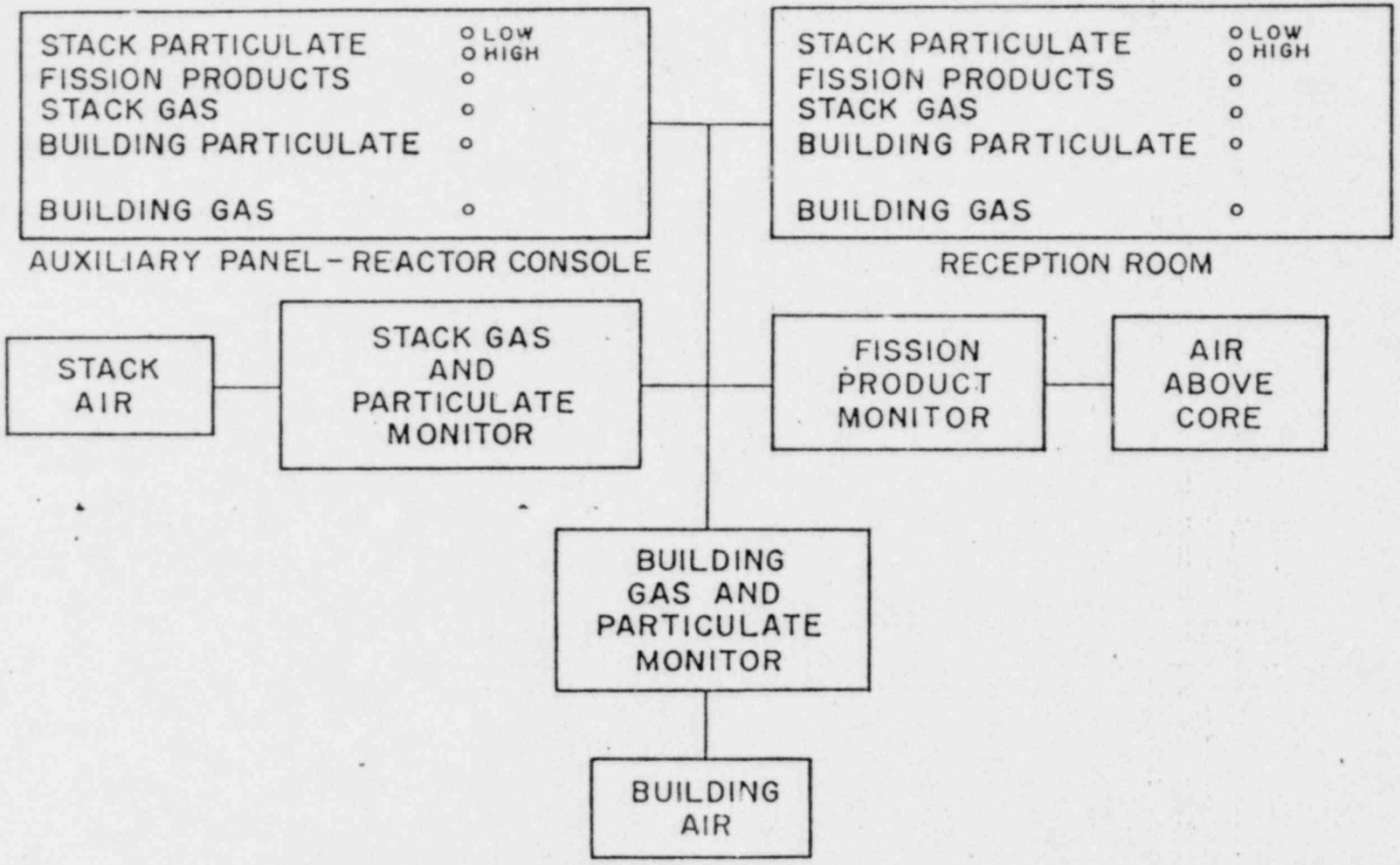

FIGURE 9-2 FACILITY AIR MONITORING SYSTEM

the particular area. A block diagram of a typical system is shown in Figure 9-3.

The area radiation monitors are located at strategic points throughout the building where the radiation levels might increase and reflect an abnormality or hazard in operations.

H. Health Physics

1. Personnel Monitoring

All personnel entering the facility will be provided with appropriate personnel monitoring devices. Personnel monitoring devices will include but not be restricted to beta-gamma and neutron film badges or TLD badges and pocket ionization chambers.

2. Protective Clothing and Equipment

Protective clothing including coveralls, boots, shoe covers, and gloves are available for use at the NSC. Use of protective clothing will be as prescribed by the health physics staff. Respiratory protective equipment is also available for emergency use. However, no allowance for its use will be taken in determining exposure of individuals to airborne radioactive material without specific USNRC authorization.

3. Change Room Facility

A change room is provided on the upper research level for use by personnel. Lockers and a shower are provided. A shower connected to the "hot" drain is provided on the lower research level for decontamination of personnel. Laundering of contaminated clothing can be accomplished on the lower research level where the drain from the washing machine is connected to the "hot" drain.

4. Radioactive Materials Handling Area

A radioactive materials handling area is located adjacent to the reactor on the upper research level. This area is used for processing and packaging radioactive materials. Protective clothing and equipment are available for use in this area. Access to the area is controlled by internal procedures. The area is posted in accordance with 10CFR20 requirements.

5. Laboratory Facility

A standard radiochemistry laboratory on the lower research level is available for research experiments and health physics use. Equipment for routine radiochemical procedures is maintained. Laboratory procedures as required for fulfillment of the radiation protection regulations will be developed as needed.

For steady state operation the all standard core has the lowest LSSS which is due to the lower safety limit. Thus, if a value of 667°C were chosen for the LSSS, it would prevent peak core temperatures from exceeding 800°C in standard fuel elements and 950°C in FLIP fuel elements for all cores that comply with the Technical Specifications.

3. The LSSS and Pulsing

The temperature scram that will occur when the LSSS is reached cannot prevent a pulse from causing the safety limit to be exceeded. This control is achieved by limiting the allowed reactivity insertion. However, a fuel element failure could possibly be prevented in the event of a pulsing accident if a scram occurred when the peak fuel temperature reached 950°C for mixed cores or 800°C for a standard core. This would reduce the total energy produced in the element by clipping the pulse "tall." Since establishing the LSSS on this basis does not limit steady state operations this conservative approach is considered to be the prudent thing to do.

The minimum LSSS for pulsing was determined for the thermocouple location adjacent to the central bundle that establishes the largest value for the power ratio ($\hat{P}/P_{TC}$) for each core examined. This power ratio is the ratio of the maximum core power, $\hat{P}$, to the power at the thermocouple location, P_{TC}. It establishes the relation between the peak core adiabatic temperature rise and the adiabatic thermocouple temperature rise. A sample calculation of the pulsing LSSS is presented for the 35 FLIP element core. Values of the LSSS for additional cores were obtained in the same manner.

The limiting peak core temperature rise, $\Delta\hat{T}_{core}$, for a peak core temperature of 950°C and an ambient temperature of 37°C is

$$\Delta\hat{T}_{core} = \hat{T}_{core} - 37^{\circ}C = 913^{\circ}C$$

The energy density required to raise the FLIP fuel to 913°C is 3580 watt-sec/cc. Applying the power ratio, the energy density at the thermocouple becomes

$$E_{TC} = \frac{3580 \text{ w.sec/cc}}{2.85} = 1256 \text{ w.sec/cc}$$

The corresponding adiabatic temperature rise ΔT_{TC} at the thermocouple is 438°C. The LSSS is obtained by determining the observed thermocouple reading. The observed reading differs from the adiabatic value due to heat flow during the first few seconds after the pulse.

The adiabatic temperature distribution corresponds to the power density distribution where the surface value is considerably higher than that at the element center. After the pulse the heat not only flows out of the element but initially some will flow towards the lower, central temperatures. This will cause an increase in the indicated thermocouple reading. The maximum thermocouple reading must therefore be used. The accepted value for the ratio of the peak temperature reading to the

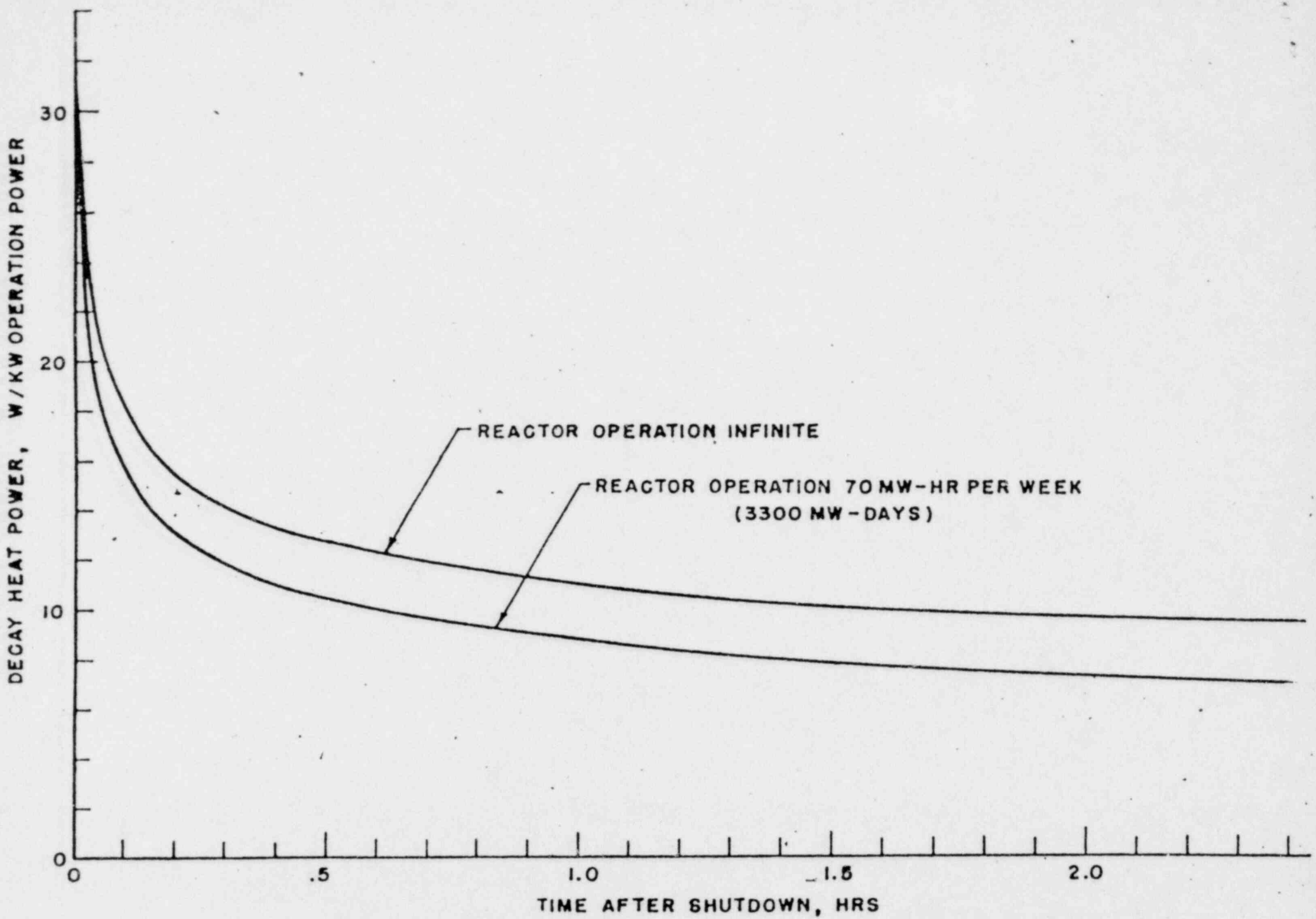

FIGURE 3 DECAY HEAT POWER GENERATION FOLLOWING LOSS OF COOLANT FOR INFINITE REACTOR OPERATION AND PERIODIC REACTOR OPERATION