SAFEGUARDS REPORT

FOR THE SAXTON RE CTOR

OPERATING AT 35 MWt

December, 1966

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I. INTRODUCTION

A. PROGRAM OBJECTIVE AND SCOPE

The objectives of the Saxton Power Escalation program are to gain operating experience with a pressurized water reactor plant at the operating conditions of temperature and pressure being planned for large scale reactor plants presently in the construction and design stages and to verify the ability of the system to operate under such conditions.

The scope of the program covers the re-evaluation of the Saxton plant for operation at power levels up to 35 MWt, actual operation at these higher power levels, and evaluation of plant performance. Some of the specific areas to be evaluated include reactor transient response and behavior of in-core materials of construction.

B. PROGRAM DESCRIPTION

The general method used for the power increase can be divided into three phases: (1) plant preparation and checkout; (2) startup to 23.5 MWt utilizing the variable frequency set to provide power for the main coolant pump; and (3) power escalation from 23.5 MWt to 35 MWt.

The power will be increased according to the general procedure outlined below with detailed procedures to be prepared prior to the power increase:

1. Startup to 23.5 MWt will be accomplished according to the present operating instructions, staying within limits set forth in the presently existing license and technical specifications.

2. The reactor will be operated at 23.5 MWt equilibrium conditions prior to the power escalation.

3. Measurements will be made prior to the power increase to verify that pertinent reactor parameters are within normal operating limits.

4. Physics parameters will be measured periodically throughout core life up to the time of the power escalation, providing a well-founded base for comparison of design values with measured values to verify predictions of reactor behavior.

5. All important plant parameters will be closely monitored and compared to predicted values during the power rise. Departure

from the predicted values by a value larger than a predetermined tolerance will require a reduction in power until evaluation of the variation has been made by the responsible Westinghouse and SNEC personnel.

6. Power will be increased in two or more separate steps allowing enough time between the steps to evaluate reactor physics behavior and the reactor and secondary plant behavior.

7. When all parameters have been shown to be within acceptable limits and after the flux distribution has been evaluated, power will be increased to 35 MWt. Again, all important plant parameters will be closely monitored and compared with predicted values during the power rise. Departures from the predicted values by a value larger than a predetermined tolerance will require a reduction in power until evaluation of the variation has been made by the responsible Westinghouse and SNEC personnel.

8. The reactor will continue to operate at 35 MWt (or the maximum specific power of 19.1 kw/ft).

II. POWER ESCALATION PROGRAM

In order to implement the increase in reactor power and to maintain a DNB ratio greater than 1.3 based on the W-3 correlation, which is the presently accepted Westinghouse design criteria for departure from nucleate boiling, the reactor inlet temperature will be reduced as power is increased. This reduction in inlet temperature, coupled with a main coolant pressure increase, will increase the subcooling sufficiently to keep the calculated reactor transients within acceptable W-3 DNB ratios.

Part of the scope of the proposed test is to simulate future water reactor operating conditions. In order to do this, it is necessary to increase the main coolant operating pressure above the present 2000 psi operating pressure, but to less than the 2500 psi design pressure. The pressure which has been selected for the increased power operation is 2200 psi, a pressure which can be achieved with the relief valves in the plant, and one which is representative of the generation of pressurized water reactors now being designed and built. The resulting mean clad temperature at the hot spot in the core is expected to be 713°F with the reactor operating within the proposed license limitations for 19.1 kw/ft and at 2200 psi.

The calculation of specific power has several uncertainties associated with it. Each of these uncertainties is added to the calculated number before comparing the specific power to the licensed value. Experience has shown that this number is consistently high but is, nevertheless, used for purposes of license compliance. If these uncertainties are all maximized and combined by direct addition, the maximum calculated mean clad temperature at the hot spot would be 721°F.

During reactor operation at the proposed license level of 19.1 kw/ft and 2200 psi, approximately 107 fuel rods can be expected to operate with a mean clad temperature at some point on the rod greater than 700°F. All special test subassemblies shall be evaluated individually. In the worst case (if all of the uncertainties in the power calculation were considered to be maximum at the same time) the same 107 fuel rods could be expected to operate with a mean clad temperature greater than 707°F, but with the maximum hot spot, mean clad temperature still not exceeding 721°F.

For a maximum overpower condition of 114%, the overpower trip set point can be 107%. The 7% difference is caused by consideration of all of the pertinent instrumentation errors and allowances. If calibration curves of nuclear instrumentation errors versus rod positions are drawn and power range channels are set intentionally high by an amount equal to the maximum negative error that could occur during a rod withdrawal accident, then at the same maximum overpower condition of 114%, the overpower trip set point could be increased to 109%.

Moisture carryover through the moisture separators in the steam generator to the turbine inlet will be evaluated during normal 23.5 MWt operation and continuously during the initial rise to any new power level. Primary evaluation will be made by monitoring noise and vibration of the turbine.

The primary coolant will be monitored for fission product activity during rise to any new power level.

III. PLANT MODIFICATIONS

A. PRESSURIZER SAFETY VALVE ANTI-SIMMER DEVICES

In order to more closely simulate the operating conditions of the newest generation of pressurized water reactors, it is necessary to increase the main coolant operating pressure for Saxton during the high power operation from 2000 psia to 2200 psia. The higher operating pressure will allow a higher maximum clad surface temperature and, consequently, a higher maximum mean clad temperature. The Saxton primary system was designed for 2500 psia and 650°F so that the increased operating pressure will be well below the system design pressure.

The code safety valves on the pressurizers will be modified by the addition of an anti-simmer device which will eliminate the simmering which presently can occur on these valves at about 300 to 400 psia below the valve set pressure. The addition of the anti-simmer device to the safety valve will not affect the set pressure or operation of the valve.

Figure III-1 shows a schematic of the anti-simmer device installation. One air loading motor is used in each device to add approximately 200 psia to the spring force of the safety valve. The air supply to each device contains two separate three-way solenoid valves for redundancy. A pressure signal to the solenoids approaching the safety valve set point will cause the solenoid to release the air pressure from the anti-simmer air diaphragm so that only the spring force is holding the safety valve closed. Pressure signals to the solenoids are taken from two separate taps from the pressurizer.

In the event of loss of electrical power or pressure signals, the solenoid will failsafe and release the pressure from the anti-simmer air diaphragm. A relief valve is also placed in the air supply to each diaphragm to prevent excess pressure buildup.

B. REACTOR TRIP CIRCUIT MODIFICATIONS

1. Loss of Load

At the increased power operation, it is desirous to install two independent reactor trip signals which would be activated on complete loss of load. The pressurizer safety valves remain available for relieving system pressure, but maintenance considerations preclude their operation if it can be avoided. For this reason, the following trip circuit additions are proposed.

a. Install two pressure switches (63/AST1 and 63/AST2) in the turbine auto stop oil system to sense decrease in auto stop oil pressure whenever the turbine is tripped. Contacts from these switches shall be added to the reactor trip breaker circuits directly and via auxiliary relay contacts (63AX/AST1 and 63X/AST2) as shown on Figure III-2. Circuitry shall be arranged such that actuation of one pressure switch will trip one reactor trip breaker directly via the breaker's shunt trip coil and the other reactor trip breaker via an auxiliary relay acting on the second breaker's shunt trip coil. The second pressure switch circuitry will be the same except it will trip the second breaker directly and the first breaker via the second pressure switch's auxiliary-relay circuit. Both auxiliary relay circuits shall act upon both reactor trip breaker undervoltage trip devices.

b. Install a multi-contact position switch assembly, adjustable over full travel (3 3/4 inch PRV) on the four-inch pressure regulating valve in the main steam line to sense closing of the valve beyond approximately 50%. Contacts from this switch shall be added to the reactor trip breaker circuits directly as shown in Figure III-2.

2. Reactor Coolant Pump Power Supply

At the increased power operation, the reactor coolant pump will be supplied with power from the variable frequency motor generator set (See Section III-C). In order to assure that a loss of coolant flow could not occur from a less than normal coolant flow rate due to drift in the frequency control, the following reactor trip additions are proposed.

Install two underfrequency relays (81/UF1 and 81/UF2) in the variable frequency supply set to trip the reactor if the frequency should drop to 60 cycles per second. (Normal Operating frequency is 63 cycles per second.) Contacts from these relays

shall be added to the reactor trip breaker circuits via auxiliary relay contacts (81X/UF1) and (81X/VF2) as shown in Figure III-3. Circuitry shall be arranged such that actuation of one underfrequency relay will trip both reactor trip breakers via the auxiliary relays acting on the breakers' shunt trip coil. The second underfrequency relay will operate in the same manner as the first. Both auxiliary relay circuits shall act upon both reactor trip breaker undervoltage trip devices.

This arrangement will prevent reactor trip for normal small fluctuations in frequency while assuring that a loss of flow could not occur from a significantly reduced flow condition.

3. <u>Permissive Switch</u>

In order to over-ride the above reactor trips during low power operation or during startup, install a manually actuated permissive switch (69). This switch is under strict administrative control and would be used to over-ride the above trips only when the reactor is to be operated at a power of less than 23.5 MWt.

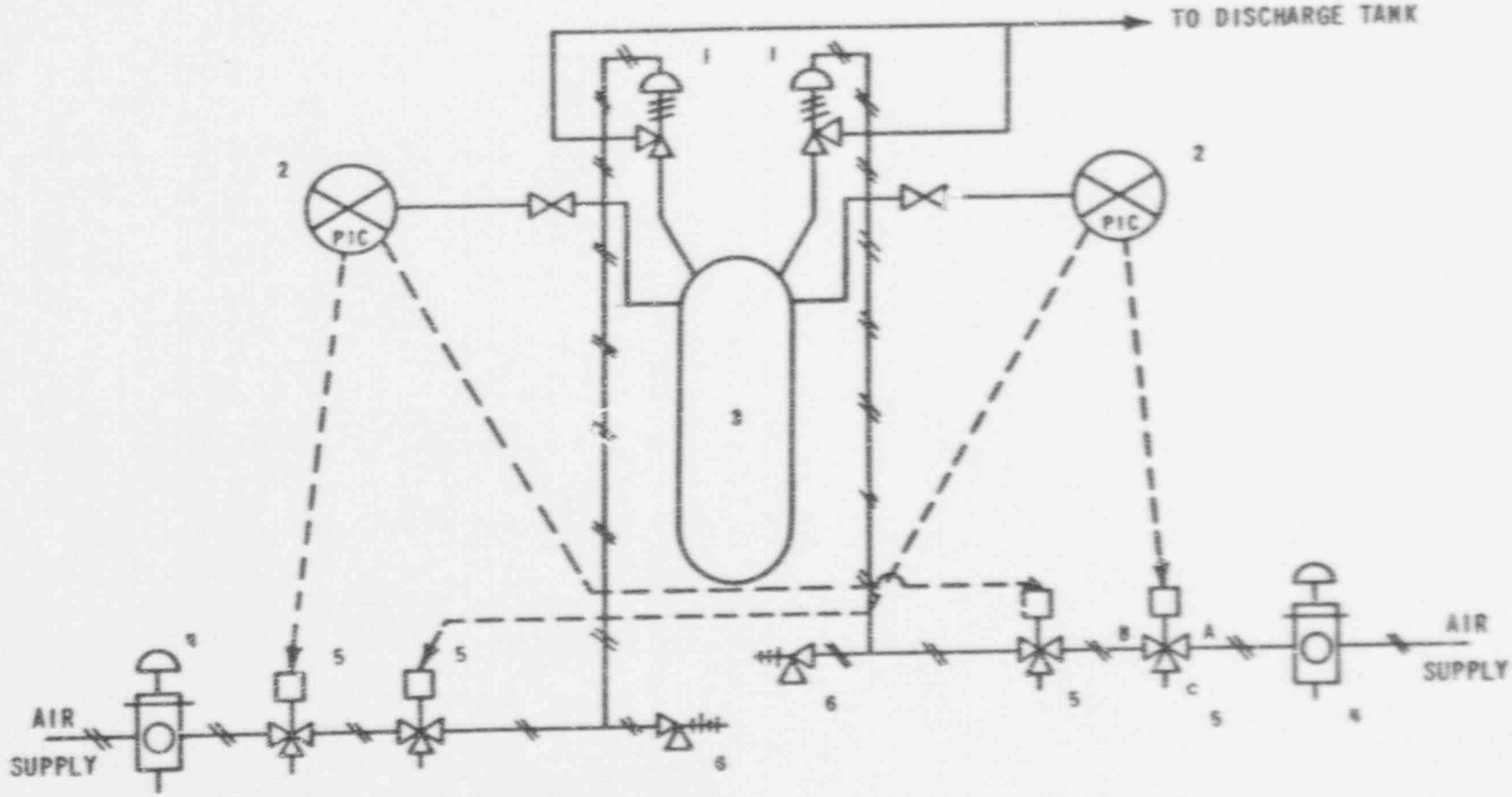
C. VARIABLE FREQUENCY MOTOR GENERATOR SET MODIFICATIONS

During the increa- - power operation of Core II, the reactor cool-ant pump will be supplied with power from the variable frequency motor-generator set. A description of the motor-generator set is given in Section 218 of the Saxton Final Safeguards Report. The motor-generator set will be used to assure an increased flow coast-down of the reactor coolant pump in the event of a loss of AC power to the plant. The modifications to be made to the plant electrical system to provide the increased coastdown are described below. Trip circuit additions to assure normal motor-generator set fre-quency are described in Section III-B-2. A one-line diagram of the modifications is shown in Figure III-3a.

Measurements of the reactor coolant pump flow coastdown with the coastdown energy of the generator have been made at Saxton and are the basis for the curve used in the loss of flow accident analyses presented in Section VI-C. In order to assure that the generator coastdown inertia will be available in the event of loss of power to the motor-generator set, the exciter field supply of the genera-tor will be automatically transferred from the A-C driven exciter to the 125V Station Battery Bus. The transfer will be accomplished using a live transfer scheme in which the generator exciter field supply will be connected to the station battery prior to the open-ing of the tie to the decaying normal exciter field supply.

The signal to accomplish the transfer will be initiated by a loss of voltage to exciter field control (See Figure III-3b). Redundant under voltage relays (Item 27-1 and Item 27-2) are provided to ini-tiate the transfer signal to the transfer equipment. Redundant transfer equipment (Items T-1 and T-2) are also provided to assure the completion of the transfer. The connection of the generator exciter to the battery bus is paralleled through the transfer equip-ment and the connection to the normal AC supply is in series through the equipment so that either set will perform the necessary transfer functions. The power required to operate the transfer equipment will also be taken from the 125V battery.

FIGURE III-1



1. AIR LOADED, SPRING OPERATED SAFETY VALVE

2. PRESSURE INDICATING CONTROLLER

3. PRESSURIZER

4. FILTER REGULATOR WITH GAUGE

5. 3-WAY SOLENOID VALVE
UPON POWER LOSS, VALVE VENTS B TO C

6. RELIEF VALVE
(MAINTAINS MINIMUM SET PRESSURE)

SAFETY VALVE ANTI-SIMMER DEVICE

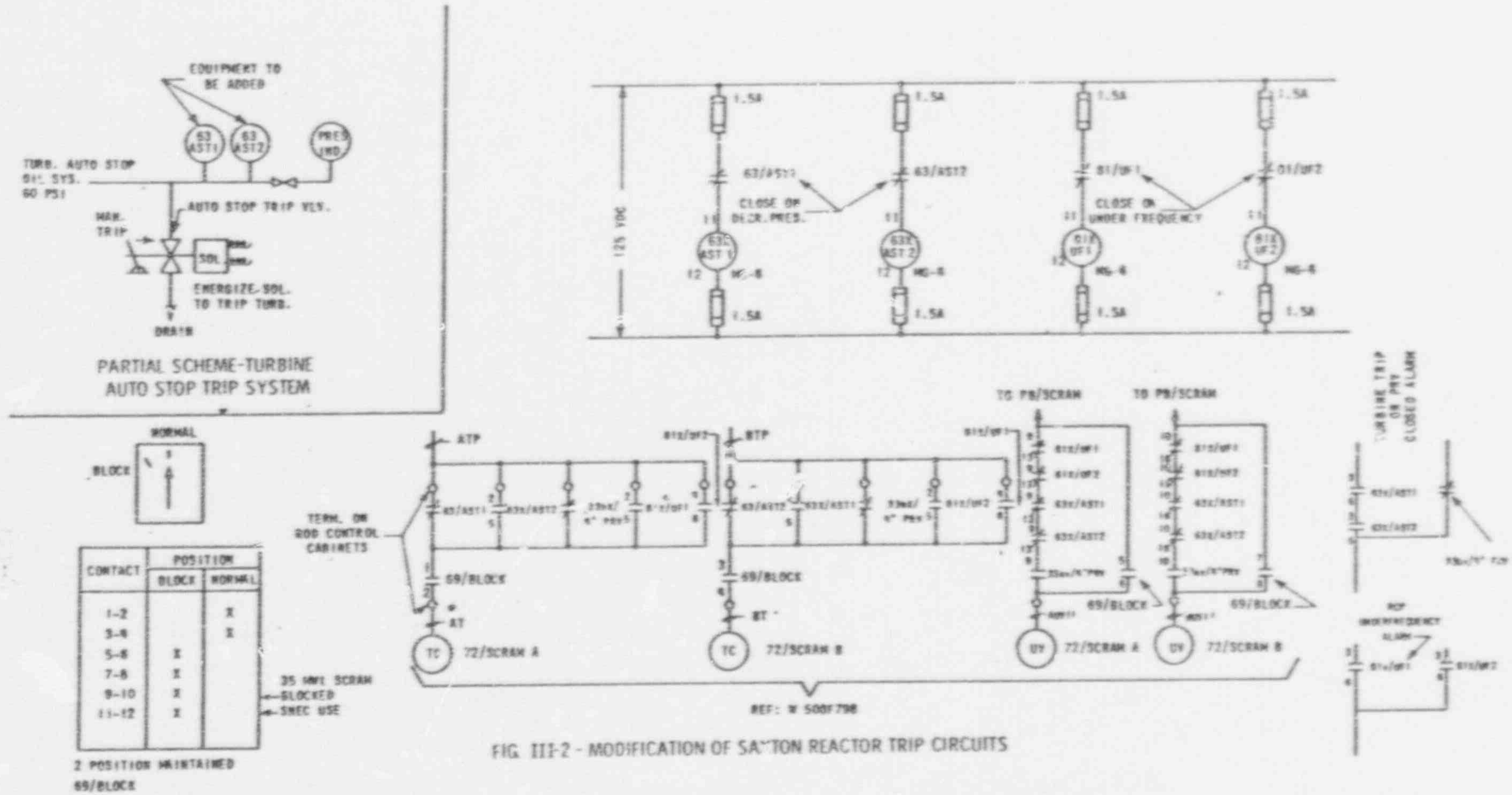
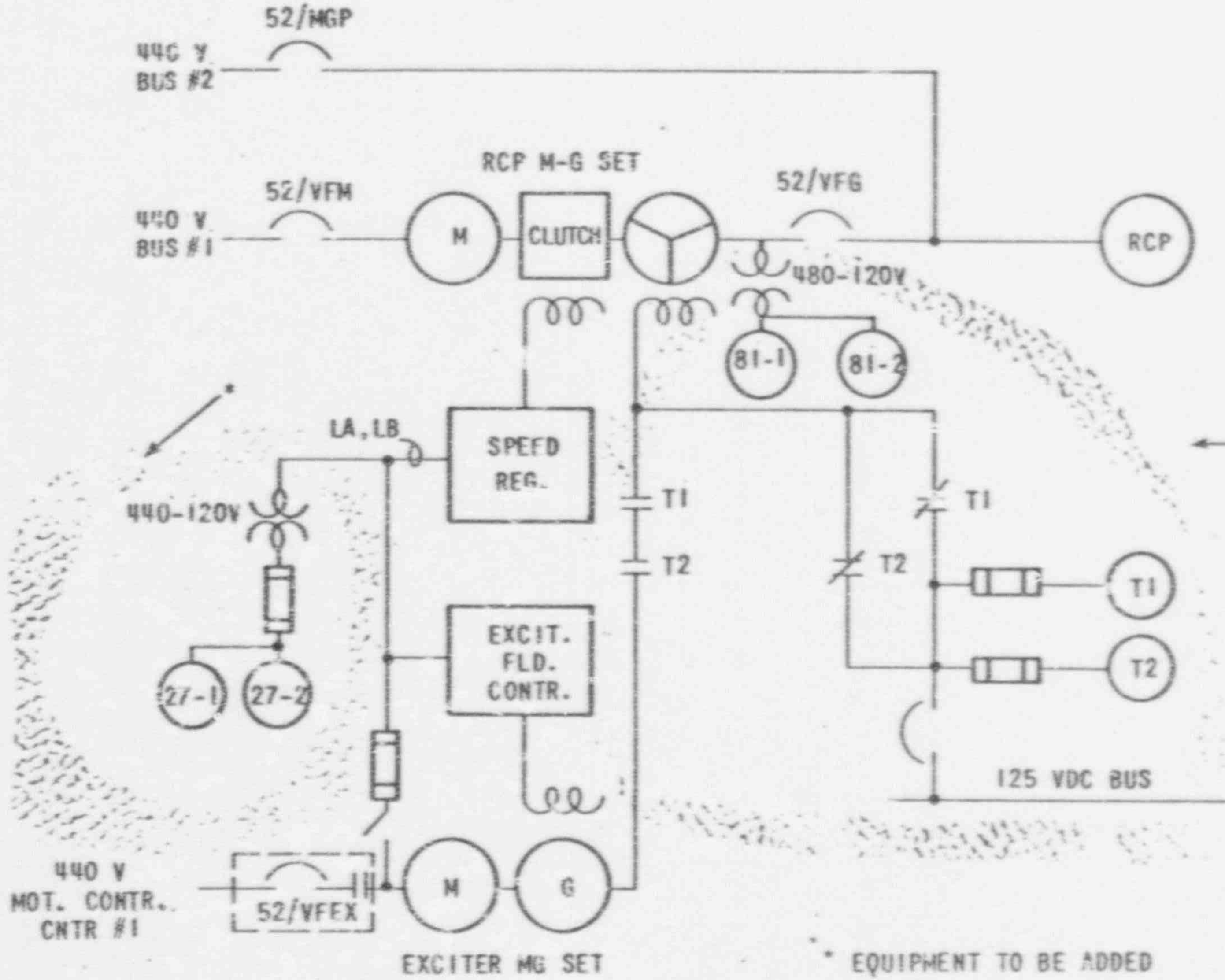
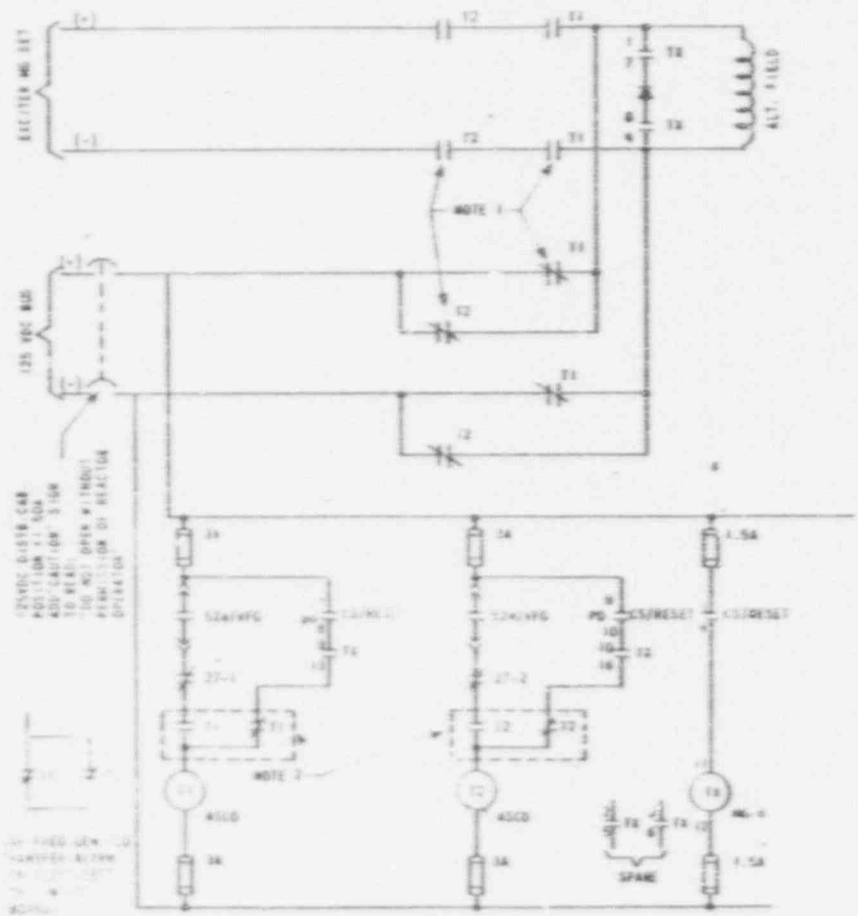


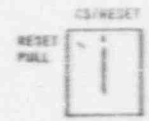
FIG. III-2 - MODIFICATION OF SA^TON REACTOR TRIP CIRCUITS

FIGURE III-3,a



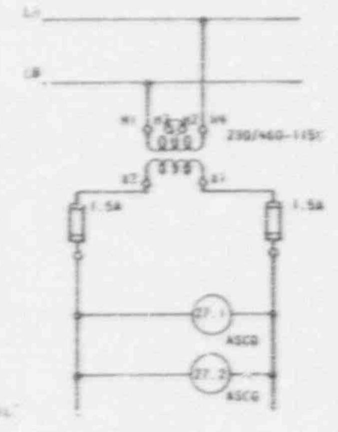
VARIABLE FREQUENCY SET GENERAL FIELD TRANSFER SCHEME ONE-LINE DIAGRAM





CONTACT	POSITION		
	PULL	RESET	NEUT.
3-4	X	X	
5-6	X	X	
7-8	X		
9-10	X		

2 POSITION SWITCH PLUS "PULL" TO RESET
SPRING RETURN FROM "PULL" TO "RESET" AND FROM "RESET" TO "NEUTRAL".



REF: DYNAMATIC DIAG. ED-14243

NOTES:
1. CONTACTS TO BE MADE BEFORE BREAK.
2. CONTACTS ARE MECHANICALLY ACTUATED BY TRANSFER SWITCH OPERATION.
3. T1 & T2 ARE MECHANICALLY HELD TRANSFER SWITCHES.
4. SCHEME SHOWN IN TRANSFER CONDITION.

FIG. III-3.D - VAR. FREQ. SET GEN. FLD. TRANSFER ELEMENTARY DIAGRAM

IV. NUCLEAR AND THERMAL AND HYDRAULIC EVALUATION

A. NUCLEAR EVALUATION

1. Summary of Core II Parameters

During the initial startup and low power operation of the Core II, a series of measurements were made to determine the operating characteristics of the core and to compare the experimental results with the design values. The following table presents some of the parameters measured for Core II and compares them to the design values presented in the Safeguards Report for Core II.

Table IV-1

Core Nuclear Parameters

Parameter	Experimental	Design Value
Reactivity Coefficients		
Power Coefficient	-0.99×10^{-4} $\Delta k/k$/% power at 20 MW	-0.8×10^{-4} $\Delta k/k$/% power at 20 MW
	-1.05×10^{-4} $\Delta k/k$/% power at 12 MW	-1.0×10^{-4} $\Delta k/k$/% power at 12 MW
Moderator Coefficient	-3.0×10^{-4} $\Delta k/k$/F° at 507°F, 800 ppm	-3.3×10^{-4} $\Delta k/k$/F° at 507°F, 800 ppm
	-2.6×10^{-4} $\Delta k/k$/F° at 520°F, 1250 ppm	-3.0×10^{-4} $\Delta k/k$/F° at 520°F, 1250 ppm
Pressure Coefficient	3.0×10^{-6} $\Delta k/k$/psi	3.5×10^{-6} $\Delta k/k$/psi
Installed Excess Reactivity	0.106 $\Delta k/k$ hot clean at power	0.103 ± .013 $\Delta k/k$ hot clean at power

Section 2 gives more details on the measurements to be taken during the power escalation to evaluate any changes in the core parameters that might be caused by the increased power level.

2. Core II Nuclear Hot Channel Factors

Hot channel factor measurements have been made for Core II to verify the core design and to assure compliance with the technical specification limits. Figure IV-1 presents the estimated change in the nuclear hot channel factors, Fq^N and $F^N_{\Delta H}$, as a function of hours of operation at 23.5 MWt. The base point for the curve is based upon measurements on March 1, 1966.

This curve shows an F_q^N of 2.86 as the best estimate and a value of 3.09 used in the safety evaluation which includes the instrumentation errors. For $F_{\Delta H}^N$ the best estimate was 2.17 with the safety evaluation using 2.34. This curve is the basis for the selection of the safety evaluation nuclear hot channel factors for the increased power operation of Core II. It was assumed that power operation at 35 MWt would commence after the core had experienced a burnup equivalent to 4800 hours operation at 23.5 MWt. This burnup was reached in the fall of 1966, with the exact time depending on the load factor at which the plant operates. With the above assumption, the hot channel factors for the start of the increased power operation are

	Best Estimate	Safety Evaluation
F_q^N	2.49	2.69
$F_{\Delta H}^N$	1.89	2.04

The above safety evaluation nuclear hot channel factors were combined with the engineering hot channel factors described in the thermal and hydraulic section to arrive at a total hot channel factor for core performance evaluation. Using these numbers, a core power level of 35 MWt will produce a maximum linear heat rating of 19.1 kw/ft in the hottest fuel rod. Hot channel factor measurements will be made during the initial period of increased power operations and the plant will be operated so that neither the power level of 35 MWt nor the maximum linear heat rating of 19.1 kw/ft will be exceeded at steady state.

B. THERMAL AND HYDRAULIC EVALUATION

1. General

The thermal and hydraulic evaluation of Core II for operation at 35 MWt is based on the same ground rules established for operation at 23.5 MWt. The evaluation of the Core II DNB conditions is now made using the Westinghouse W-3 correlation rather than the W-2 correlations used in the original design and evaluation of Core II. The details of the W-3 correlation are given in WCAP-5584, "DNB Predictions for an Axially Non Uniform Heat Flux Distribution."

The only change in the thermal and hydraulic criteria is the minimum DNB ratio for core operation and design transients. The minimum DNB ratio for the W-2 correlations was 1.25 and the new minimum ratio for the W-3 correlation is 1.30. These ratios correspond to the source statistical point in both correlations: that is, at this DNB ratio, there is a 95% probability that DNB will not occur with a confidence level of 95%. The higher minimum number for the W-3 correlation is due to a little more scatter in the experimental data upon which the correlation is based.

2. Engineering Hot Channel Factors

A description of the engineering hot channel factors is given in the Section III of the Core II Safeguards Report. Evaluations performed on Core II after the submission of the Safeguards Report have shown that a reduction in the enthalpy rise engineering hot channel factor, $F^E_{\Delta H}$ is possible. The heat flux engineering hot channel factor F^E_q, has not been changed and remains 1.045 as defined in the Core II Safeguards Report.

The changes to $F^E_{\Delta H}$ are as follows:

a. Statistical subfactor for pellet diameter density, enrichment and eccentricity and for fuel rod diameter, pitch and bowing: The previous design number was 1.14. Based upon measurements of the as-built Core II, this factor has been reduced to 1.08. The major improvement is realized through the use of the new, improved spring clip grid design described in the Core II Safeguards Report.

b. Inlet Flow Maldistribution Subfactor:

This factor is based upon previous analyses of in-core measurements and remains the same as before, 1.07.

c. Flow Redistribution Subfactor:

This factor is changed from a previous value of 1.05 to a new value of 1.02. The change is based upon revised computer code calculations which determine the effect of flow redistribution out of the hot channel.

d. Flow Mixing Subfactor:

This factor has been increased from a value of 0.95 to 0.96 based on an evaluation of the effect of the new grid design.

In summary, the new factor is:

Subfactor	Value
Pellet diameter, density enrichment and eccentricity; Fuel rod diameter, pitch and bowing	1.08
Inlet Flow Maldistribution	1.07
Flow Redistribution	1.02
Flow Mixing	0.96
TOTAL $F^E_{\Delta H}$ =	1.13

3. Thermal and Hydraulic Design Parameters

Table IV-2 presents the thermal and hydraulic parameters for the operation of Core II at 35 MWt.

Table IV-2

Thermal & Hydraulic Design Parameters

Total Core

Total Heat Output	35.0 MWt
Total Heat Output	119.5×10^6 Btu/hr
Heat Generated in Fuel	97.4 %
System Pressure - Nominal	2200 psia
System Pressure - Minimum - Steady State	2150 psia
System Pressure - Minimum - Transient	2050 psia
Total Flow Rate*	3.21×10^6 lb/hr
Effective Flow Rate for Heat Transfer	2.74×10^6 lb/hr
Flow Area for Heat Transfer Flow**	2.53 ft^2
Average Velocity Along Fuel Rods**	6.01 ft/sec

Coolant Temperatures

Nominal Inlet	480 F
Maximum Inlet Including Instrument Errors and Deadband	485 F
Average Rise in Vessel	32 F
Average Rise in Core	37 F
Average in Vessel	496 F
Average in Core	499 F

Heat Transfer

Active Heat Transfer Surface Area of Fuel Rods	510.8 ft^2
Average Heat Flux	228,000 Btu/hr ft^2
Average Thermal Output	6.83 kw/ft
Maximum Clad Surface Temperature at Nominal Pressure	654 F

* Assuming constant volume pump.

** Assuming all control rods out, followers in.

Table IV-2 (Cont'd.)

<u>Central Core Region</u> (UO_2 - PUO_2 Fuel)

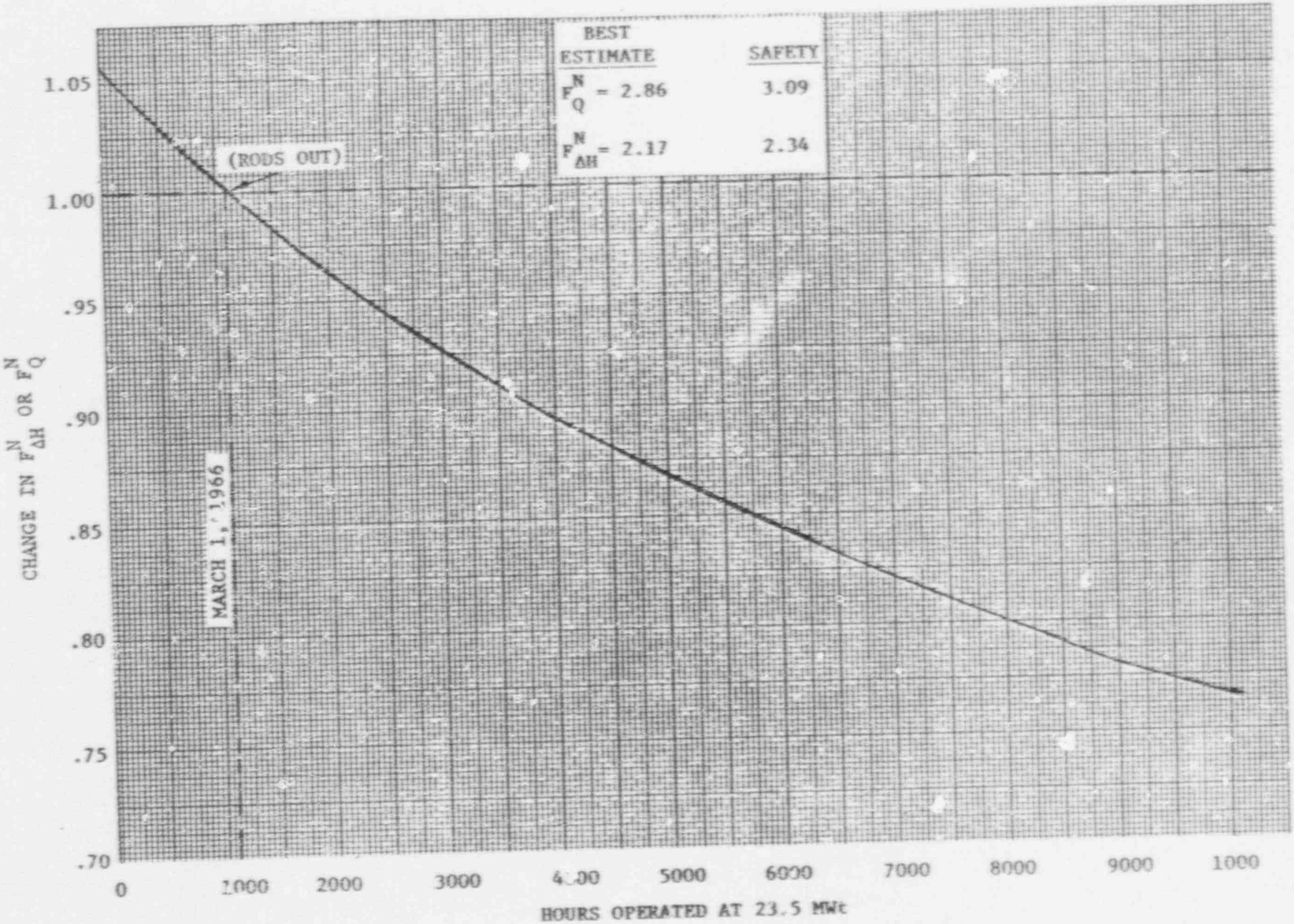
F_q Heat Flux Hot Channel Factor	2.81
$F_{\Delta H}$ Enthalpy Rise Hot Channel Factor	2.30
Nominal Outlet Temperature of Hot Channel	578 F
Maximum Outlet Temperature of Hot Channel	583 F
Maximum Outlet Enthalpy of Hot Channel	590.5 Btu/lb
Saturation Enthalpy at Minimum Steady State Pressure	689 Btu/lb
Maximum Heat Flux	640,000 Btu/hr ft^2
Maximum Thermal Output	19.1 kw/ft
W-3 DNB Ratio at 100% Power Nominal Conditions	1.52
W-3 DNB Ratio at 2050 psia, T_{in} max. 114% Power	1.32

4. Fuel Central Temperatures

Maximum fuel central temperatures for the core with a power level of 35 MWt and a maximum linear heat rating of 19.1 Kw/ft have been calculated. With the core loading of Core II, the hottest fuel rod is a zircaloy clad, pelletized PuO_2 - UO_2 rod. The maximum central temperature would occur for 19.1 Kw/ft condition and 35 MWt operation combination at the start of the high power operation. Using the basis outlined in the Core II Safeguard Report, a maximum central temperature of 4465°F is calculated which is below the fuel melting temperature. At the maximum over power condition of 114% power (39.9 MWt) the linear heat rating is 21.8 Kw/ft and the associated maximum central temperature is 4825°F which is also below the fuel melting temperature.

5. Fuel Clad Temperatures

At 19.1 Kw/ft (the maximum linear heat rating for 35 MWt) and 2200 psi, approximately 107 fuel rods can be expected to operate with a mean clad temperature at some point on the rod greater than 700°F. In the worst case, (if all of the uncertainties in the power calculations were considered to be maximum at the same time) the same 107 fuel rods could be expected to operate with a mean clad temperature greater than 707°F but with the maximum hot spot mean clad temperature still not exceeding 721°F.



CHANGE IN SAXTON HOT CHANNEL FACTORS

FIGURE IV-1

V. FUEL ELEMENT MATERIALS EVALUATION

A. INTRODUCTION

The problems associated with raising the power levels of Saxton fuel elements from 16 kw/ft to 19.1 kw/ft peak and operation with mean clad temperatures of 721°F peak have been analyzed from a materials standpoint. Three possible problem areas are discussed in relation to fuel element integrity: 1) fuel performance, 2) Zircaloy creep, 3) corrosion and hydriding, and 4) fuel swelling.

In addition, relevant Westinghouse experience with high power Zircaloy fuel rods is presented in the Appendix. Results of irradiation experiments in the WTR, Saxton, NASA Plumbrook, and GETR reactor are given which provide useful information on such parameters as specific power density, fuel-clad gap, diametral expansion, fuel melting, burnup, corrosion, coolant chemistry, etc.

Based on Westinghouse experience with fuel elements operating under comparable or more severe conditions than for the Saxton Power Escalation program, it may be concluded that the Saxton fuel elements can safely operate under the new design conditions from a materials standpoint.

B. FUEL PERFORMANCE

Operation of the Saxton fuel elements will not result in center melting either at nominal peak (19.1 kw/ft) or overpower conditions. Maximum fuel temperatures for the PuO_2-UO_2 pellet rods are tabulated in Section IV-B.

Westinghouse experience with extreme power ratings in fuel elements, (cf. Appendix, Section IX) shows that even gross center melting can safely be accommodated. Thus, design limitations based upon fuel melting are conservative.

Increased thermal expansion of the fuel pellets with the increased power levels will not jeopardize fuel cladding integrity. Axial expansion is minimized by dishing of the pellet endfaces. Initial (cold) diametral pellet-to-clad gaps of 0.005 inches and 0.0065 inches for the mixed oxide and uranium dioxide fuel rods, respectively, will be sufficient to prevent appreciable clad plastic deformation (in excess of 1/2% strain) or failure from radial expansion.

Westinghouse experience with vibratory-compacted fuel (cf. Appendix, Section VI) indicates there are no problems from fuel expansion.

Fuel swelling due to internal accumulation of fission products with burnup will also be safely accommodated by the initial fuel-clad gaps. Based upon initial fuel densities of 94%, and anticipated peak burnups of 50,000 MWD/MTU fuel swelling will occur at a sufficiently low rate [1] (0.0016 $\frac{\Delta V}{V}$ per 10^{20} f/cc) to limit swelling and subsequent clad failure. Although the control rod followers will experience a peak burnup of ~50,000 MWD/MTU, the anticipated fuel swelling can be accommodated by the initial fuel void volume and fuel-clad gap (a relatively large fuel-clad gap will exist near the end of life since the power rating of these fuel rods will be relatively low).

Higher peak burnups and average fuel temperatures will increase the amount of fission gas released by the fuel. However, the conservatively designed fission gas plenums in the Saxton rods will limit the accumulated internal gas pressure to less than the external coolant pressure.

Based upon the installed excess reactivity and measured boron worth, the expected life of Core II will slightly exceed the design life of 8400 hours of operation at 23.5 MWt (See Figure II-1 of the Core II Safeguards Report). Based upon this life, the maximum fuel rod internal gas pressure is less than the coolant system operating pressure of 2200 psia proposed for the 35 MWt operation of Core II.

C. ZIRCALOY CREEP RATE

The maximum mean clad temperature of 721°F in the 19.1 kw/ft rods is well below the critical region in which the permanent diametral creep rate of Zircaloy increases exponentially. It was calculated that tube-reduced Zircaloy would not fail for 7.4 years at the maximum (16,000 psia tensile or compressive) expected design stress level in pressurized water reactors. This calculation was based on the ultra-conservative assumption that the maximum clad stress exists throughout life.

It is concluded that no problem exists with creep in the fuel rods during the high power tests since stresses will be compressive and strains will be limited by the fuel pellet.

D. ZIRCALOY CORROSION AND HYDRIDING

Excessive corrosion and accompanied pickup of hydrogen leading to loss of cladding ductility is not a problem with Zircaloy fuel rods and is not expected to be at higher power levels and temperatures. Zircaloy-2 and Zircaloy-4 have been irradiated in Saxton at surface temperatures up to approximately 642°F with nucleate boiling.

Experience to date with Zircaloy-clad fuel rods operating in the borated environment shows:

a) Satisfactory in-pile performance of Zircaloy-clad rods. A lustrous black oxide was present in all cases, and there were no adverse effects related to exposure in the Saxton chemical-shim environment.

b) Good agreement of weight gain values with published out-of-pile results. There is no adverse effect on the corrosion rate due to irradiation or to the borated coolant.[2] Hydrogen absorption by the cladding also agrees well with published out-of-pile results and anticipated pickup levels.[3]

The possible extent of Zircaloy-4 corrosion and hydriding can be computed using the RB-COM code,[4] a combination of models representing forced convection (RUST) and nucleate boiling (BROIL) heat transfer conditions. Predicted curves for hydriding and corrosion for Saxton as a function of time agree well with actual in-pile data from this reactor. Based upon predicted values for similar or even more severe conditions than scheduled in the Saxton Power Escalation Program, increases in corrosion weight gains and hydriding of the Zircaloy clad will easily be within tolerable limits.

E. SUMMARY AND CONCLUSIONS

The performance and integrity of the Saxton core fuel elements under increased power conditions have been analyzed from a materials standpoint. The major possible problem areas of fuel performance, Zircaloy clad creep, and Zircaloy corrosion and hydriding have been examined. No problems are expected during operation of the Saxton fuel elements at the proposed power levels. The relatively modest magnitude of the proposed increase in specific power together with the initially conservative fuel element designs will permit expected changes in fuel behavior. Increased clad temperature and heat flux will not introduce problems of creep or corrosion and hydriding in the cladding.

These conclusions are substantially supported by Westinghouse irradiation experience with high power density fuel elements in pressurized water reactors.

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3. D. R. McClintock, "Effect of Surface Treatment on Zircaloy-Clad Fuel Rods in a Chemical Shim Environment," ANS Trans. 8, No. 1, pg. 17., June 1965.

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VI. ACCIDENT ANALYSIS

A. GENERAL

The increased power level (35 MWt as compared to the 23.5 MWt licensed operating power) and the corresponding changes in operating conditions and instrument settings require that some of the incidents previously reported (in the Saxton Final Safeguards Report and in the Safeguards Report for the Saxton Reactor Partial Plutonium Core II) be re-evaluated. Information in Section 502 of the Final Safeguards Report relating to the possible causes of incidents and the safeguards provided applies to the incidents analyzed for this report and will not be repeated.

B. REACTIVITY INCIDENTS

1. Rod Withdrawal Incident

An uncontrolled rod withdrawal is assumed to occur from an electrical or mechanical failure in the nuclear instrumentation and control systems or by operator error. In this unlikely event, an electrical interlock ensures that only one of the two control rod groups would be withdrawn. Assuming that the most reactive control rod group is withdrawn at its maximum rate (1.5 inches per minute) in its maximum worth region, the reactivity addition rate is limited to less than 7.2×10^{-5} Δk/sec. In Section VI of the Safeguards Report for the Saxton Reactor Partial Plutonium Core II, rod withdrawal transients are presented for cold and hot subcritical, and full power operation initial conditions. These analyses were based on a conservative high insertion rate of 2.5×10^{-4} Δk/sec and illustrate the effectiveness of the overpower trip in terminating a rod withdrawal transient. Startup from the hot or cold subcritical condition is of less consequence than rod withdrawal from power, so rod withdrawal from power is re-analyzed. The principal change affecting the transients starting from subcritical is the increase in the overpower trip setpoint from 115% of 23.5 MWt to 107% of 35 MWt.

The additional energy generated and, hence, the increase in fuel temperature in reaching the higher trip level is not significant. It should be noted that the analyses presented for rod withdrawal from subcritical in Section VI of the Safeguards Report for the Saxton Reactor Partial Plutonium Core II are highly conservative in that no credit is taken for either the startup rate trip (set at 2 decades/minute) or for the reduction in overpower trip setpoint (to 5 MWt) during zero power operation.

The transient for rod withdrawal from power is shown in Figure VI-1 based on the following conservative conditions:

a) Initial power level is 103% of the nominal (35 MWt) power to allow for calorimetric errors.

b) Initial primary coolant pressure is at its minimum value of 2150 psia to allow for instrument errors.

c) Initial coolant inlet temperature is at its maximum value of 485°F allowing for instrument error.

d) Minimum expected absolute value of negative fuel temperature (Doppler) coefficient: -1.0×10^{-5} $\Delta k/°F$.

e) Minimum expected absolute value of negative moderator temperature coefficient: -2.0×10^{-4} $\Delta k/°F$.

f) Reactor trip initiation due to overpower at 114% of the 35 MWt power (7% over the 107% reactor trip setpoint to allow for instrumentation errors).

With the nuclear flux and hot spot heat flux peak at 120% and 114% of their nominal full power values respectively, the minimum DNB ratio (calculated using the W-3 correlation) is 1.31 and occurs 3.7 seconds after initiation of the incident. This indicates that the power level reactor trip protection would prevent core damage in the improbable event of an uncontrolled rod withdrawal.

2. Steam Line Break Incident

Rupture of a secondary plant steam line is reflected into the primary system as a step load increase and results in a decreasing coolant inlet temperature. The negative moderator temperature coefficient causes reactivity and power to increase. For large breaks, the control capability of the plant is exceeded and the reactor protection system will automatically initiate a reactor trip by either the overpower or low pressure condition. Following trip, heat extraction exceeds heat generation and the coolant temperature decreases further. Due to the negative moderator temperature coefficient, the shutdown margin is reduced until heat removal is terminated.

The consequences of a steam line break for the proposed 35 MWt operating conditions are not changed from those reported

previously in Reference (1), (2) and (3). The lower secondary temperature, 420°F for 35 MWt operation as compared to 490° for 23.5 MWt operation) will result in a smaller steam mass flow rate through the break and will reduce the rate of cooldown in the primary system. Since the average coolant temperature is reduced to 500°F for the proposed operating conditions, the moderator temperature coefficient is less negative than that assumed in the reference analyses. Both of these effects will result in a less severe reactivity transient. The effect of the increased power level is to increase decay heat following reactor trip which reduces slightly the primary system cooldown. The operating conditions for 35 MWt result in less reduction in shutdown margin during blowdown of the steam generator contents.

C. MECHANICAL INCIDENTS

1. Loss of Flow Incident

A loss of coolant flow could result from loss of electrical power to the reactor coolant pump motor or from mechanical failure in the pump motor or coupling between the pump motor and pump. A mechanical failure causing sudden seizure of the pump motor is not considered credible. Following loss of coolant flow, coolant temperature will increase because of reactor trip circuit delays which allow continued power generation while flow coastdown is occurring. If the heat generation is not terminated rapidly enough to prevent DNB, clad failure can result.

Power generation during a loss of coolant flow incident is terminated by an automatic reactor trip initiated from either a low voltage signal on the reactor coolant pump bus, a low frequency signal on the MG set control or from a low flow signal.

In the event of a loss of the station 480 V auxiliary electrical system, reactor coolant pump coastdown is extended (see Section IV) by automatic switching which transfers the field supply of the generator to the station battery. This extended coastdown

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(1) Saxton-Final Safeguards Report

(2) Safeguards Report for Phase I of Saxton Nuclear Experimental Corporation Five-Year Research and Development Program (December 1961).

(3) Saxton Nuclear Experiment Corporation - Safeguards Report for the Saxton Reactor Partial Plutonium Core II (March 1965).

is shown on figure VI-2. The loss of flow transient based on this coastdown was analyzed using the following conservative initial conditions:

a) Power: 35 MWt x 1.03 = 36.05 MWt

b) Inlet Temperature: 480°F - 485°F

c) Primary Pressure: 2200 psia - 50 psia = 2150 psia

d) Initial flow rate corresponding to MG set low frequency reactor trip setpoint: 60 cps.

The minimum DNB ratio in the transient is 1.31, occurring 1.4 seconds after initiation of the incident. Hence, there is no core damage for loss of power to the pump motor.

2. Loss of Load Incident

A loss of load could result from either turbine trip or steam line valve closure. Protective trips which sense "auto-stop" oil pressure and main steam line pressure regulating valve closure would trip the reactor immediately and there would be no consequence.

Loss of load without immediate reactor trip will result in heatup and pressurization of the primary system, and the reactor would be tripped by the hot leg temperature trip. The primary system pressure could increase to the safety valve setpoint which would cause steam release and limit primary pressure.

The loss of load incident is analyzed with the following assumptions:

a. Initial power was assumed to be 103% of 35 MWt to allow for calorimetric errors.

b. No immediate reactor trip.

c. No pressurizer spray action.

d. No pressurizer solenoid relief valve action.

e. Maximum expected value of the moderator reactivity coefficient: 2.0×10^{-4} $\Delta k/°F$.

f. Maximum expected absolute value of the fuel reactivity coefficient: -1.65×10^{-5} $\Delta k/°F$.

g. No hot leg temperature trip.

The transient response is shown on Figures VI-3 through VI-8. Figure VI-6 shows the power decrease resulting from the negative temperature coefficient. Figure VI-7 shows that the first safety valve opens in about 33 seconds to limit pressure to 2520 psia, a value well below the setpoint of the second safety valve (2575 psia). The maximum rate of steam release is 7720 lb/hr, well below the capacity of a single safety valve (20,000 lb/hr).

The above conservative analysis demonstrates that the plant is protected for this incident.

3. Loss of Coolant Incident

Loss of coolant through a rupture in the primary coolant system results in decreasing water level and decreasing system pressure. The reactor is tripped when the pressurizer water level decreases below 20 cu. ft. or when the system pressure decreases below 2050 psia whichever occurs first. The two safety injection pumps are started when system pressure decreases below 1000 psia to deliver borated water to the main coolant system.

Section 504-A of the Final Safeguards report analyzed the consequences of loss of coolant caused by rupture of the largest pipeline connecting to the reactor coolant system. That analysis, which is based on a reactor coolant system average temperature of 530°F, shows that the water addition by safety injection prevents the core from being uncovered,thereby preventing core damage. The reactor coolant temperature will be reduced to 500°F for the proposed 35 MWt operating power level. The effect of this reduction in average coolant temperature is a reduction in the saturation pressure of the system which in turn speeds delivery of borated safety injection water to the system. Hence, the analysis in the Final Safeguards report is conservative for the proposed 501°F average coolant temperature operating condition.

D. HYPOTHETICAL ACCIDENT

The existence of a situation with consequences more severe than those resulting from previously discussed credible incidents is postulated. This hypothesized accident is defined as an instantaneous blowdown of the contents of the reactor coolant system to the containment vessel resulting in meltdown of one hundred percent of the reactor core and subsequent release of fission products. Even though this situation is not considered credible, it is presented as representing an upper limit on the accidental release of radioactive material from which consequences to the public can be conservatively assessed.

The effect of flashing of the contents of the primary coolant system is analyzed in Section 506 of the Saxton-Final Safeguards report. In that analysis, the peak containment pressure reached 30 psig with a containment vapor temperature of 250°F. Since the reactor coolant average temperature is to be reduced from the present 530°F to 500°F for the proposed operating conditions, the pressure transient will be less severe, resulting in a peak of 24 psig with a containment vapor temperature of 205°F. This is considerably less than the 30 psig containment design pressure.

The off-site doses are calculated for one hundred percent core meltdown. The total core fission product inventory is computed based on operation at 35 MWt. The quantities of radioactive fission products released to the reactor containment during meltdown are estimated based on fission product release fractions reported in TID-14844. These fractions are:

100% of the noble gases
50% of the halogens
1% of the solids

Fifty percent of the iodine isotopes released to the containment are assumed to plate out on walls and structures within the containment. The remaining activity is assumed to be mixed uniformly throughout the containment volume and the direct gamma dose is calculated at various distances from the reactor containment.

In addition to the dose from activity confined in the containment, the dose to a receptor in the plume of fission products leaking from the containment is evaluated. The most significant aspect of this leakage is the thyroid dose caused by inhalation of the iodine isotopes in the plume.

A constant containment leak rate of 0.4% per twenty-four hours (the technical specification limit) is used with Sutton's meteorological model for determining activity concentrations downwind from the

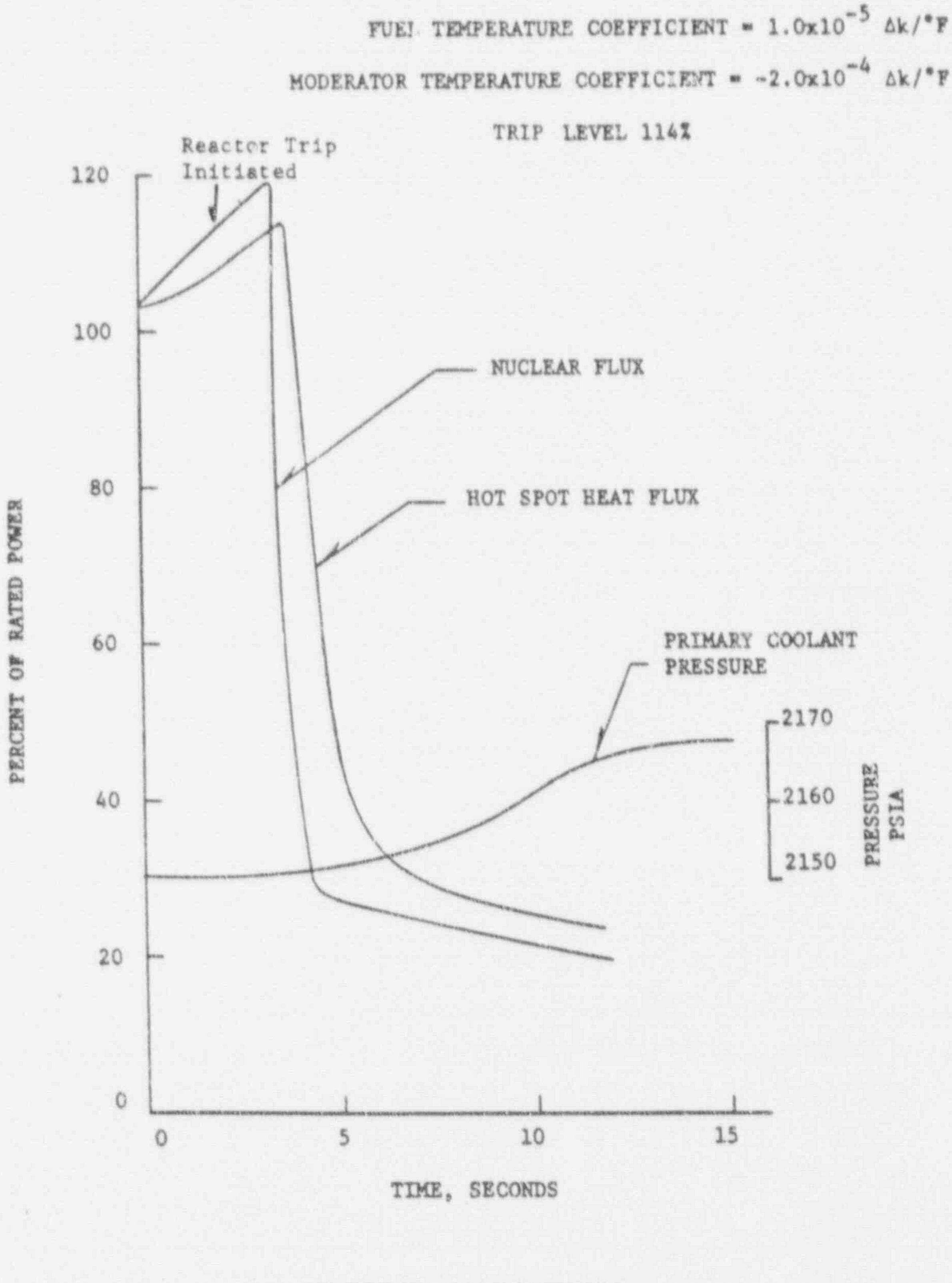
containment. The parameters used with Sutton's equation were deduced from measurements at the site and are representative of stable wind conditions. This model and the parameters are discussed in Section 103 of the Saxton-Final Safeguards Report. Two and twenty-four hour thyroid doses are computed based on inhalation of air containing the maximum (plume centerline) activity concentration.

Figures VI-9 and VI-10 show the two and twenty-four hour direct gamma dose and the two and twenty-four thyroid dose as a function of distance from the reactor containment.

The following table summarizes the computed doses.

Dose	Time in Which Dose Accumulated	Distance from Containment
25 Rem-whole body	2 hrs.	270 M
25 Rem-whole body	24 hrs.	375 M
300 Rem-thyroid	2 hrs.	240 M
300 Rem-thyroid	24 hrs.	720 M

The doses from this hypothetical accident comply with the limits set forth in 10 CFR 100 and do not represent undue hazard to the general public. The doses in any credible accident would be substantially less than those reported above.



CONTINUOUS ROD WITHDRAWAL

(REACTIVITY INSERTION RATE = 2.5 x 10^{-4} Δk/SECONDS)

FIGURE VI-1

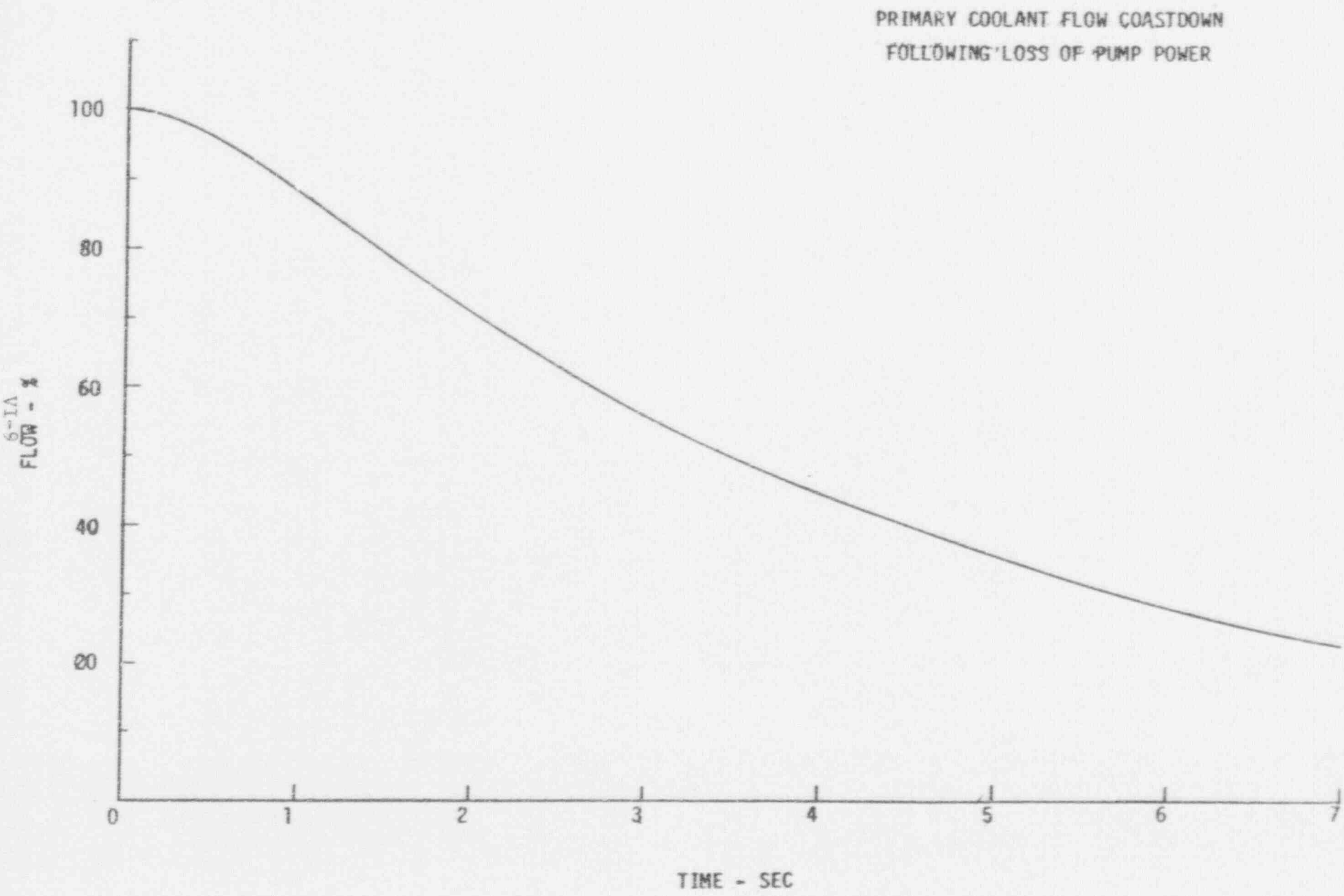
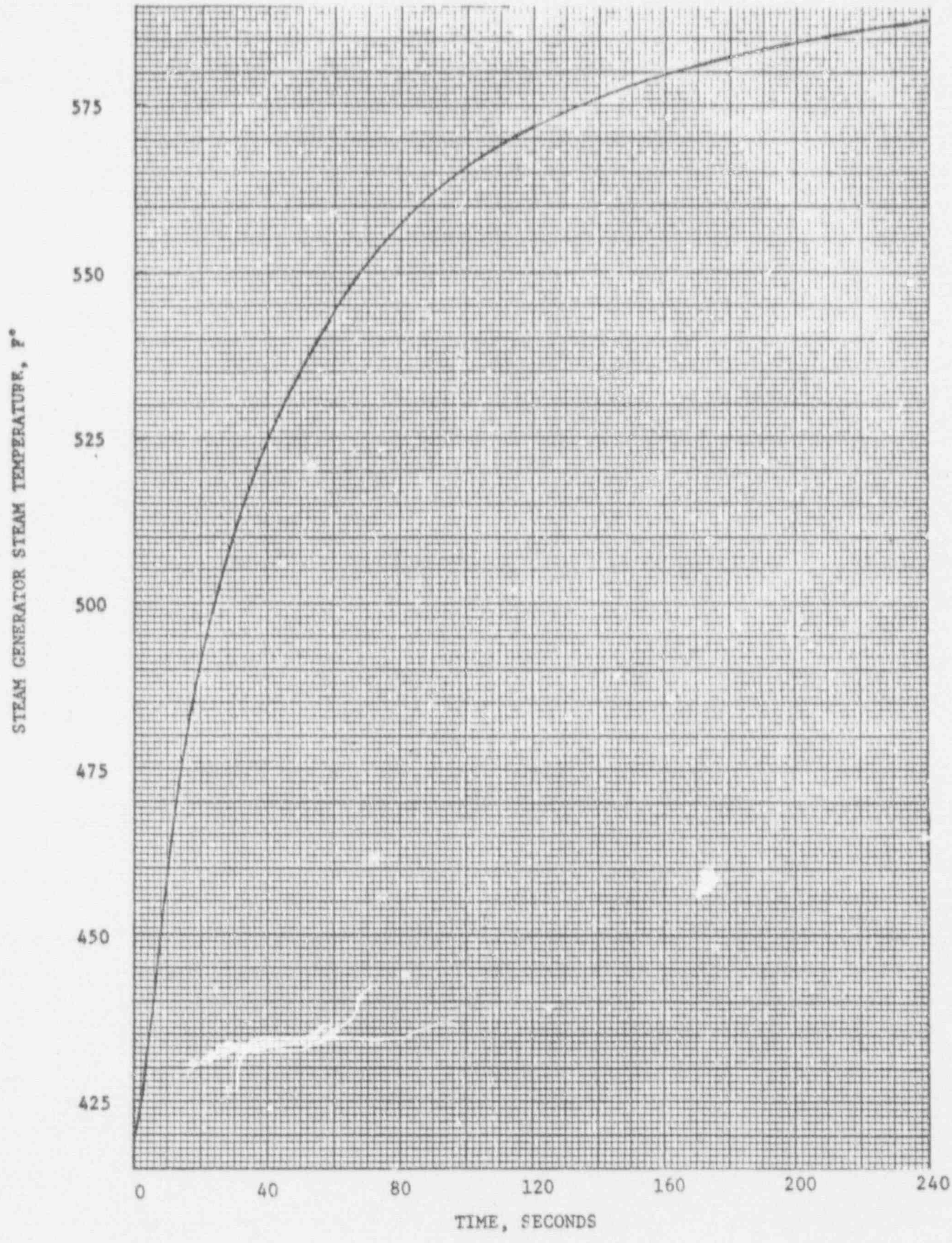
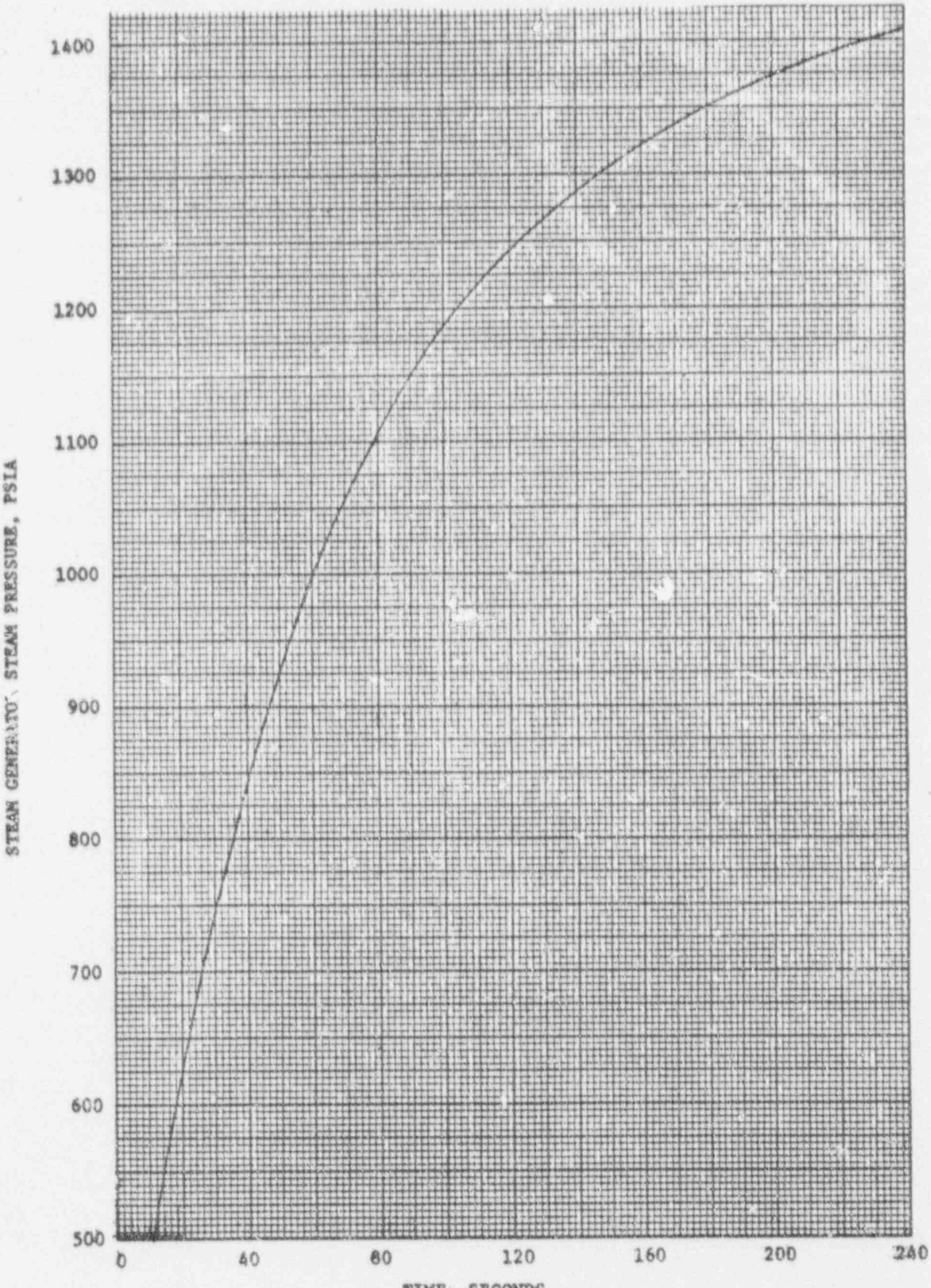

PRIMARY COOLANT FLOW COASTDOWN
FOLLOWING LOSS OF PUMP POWER
FLOW - %
100
80
60
40
20
0
1
2
3
4
5
6
7
TIME - SEC

FIGURE VI-2

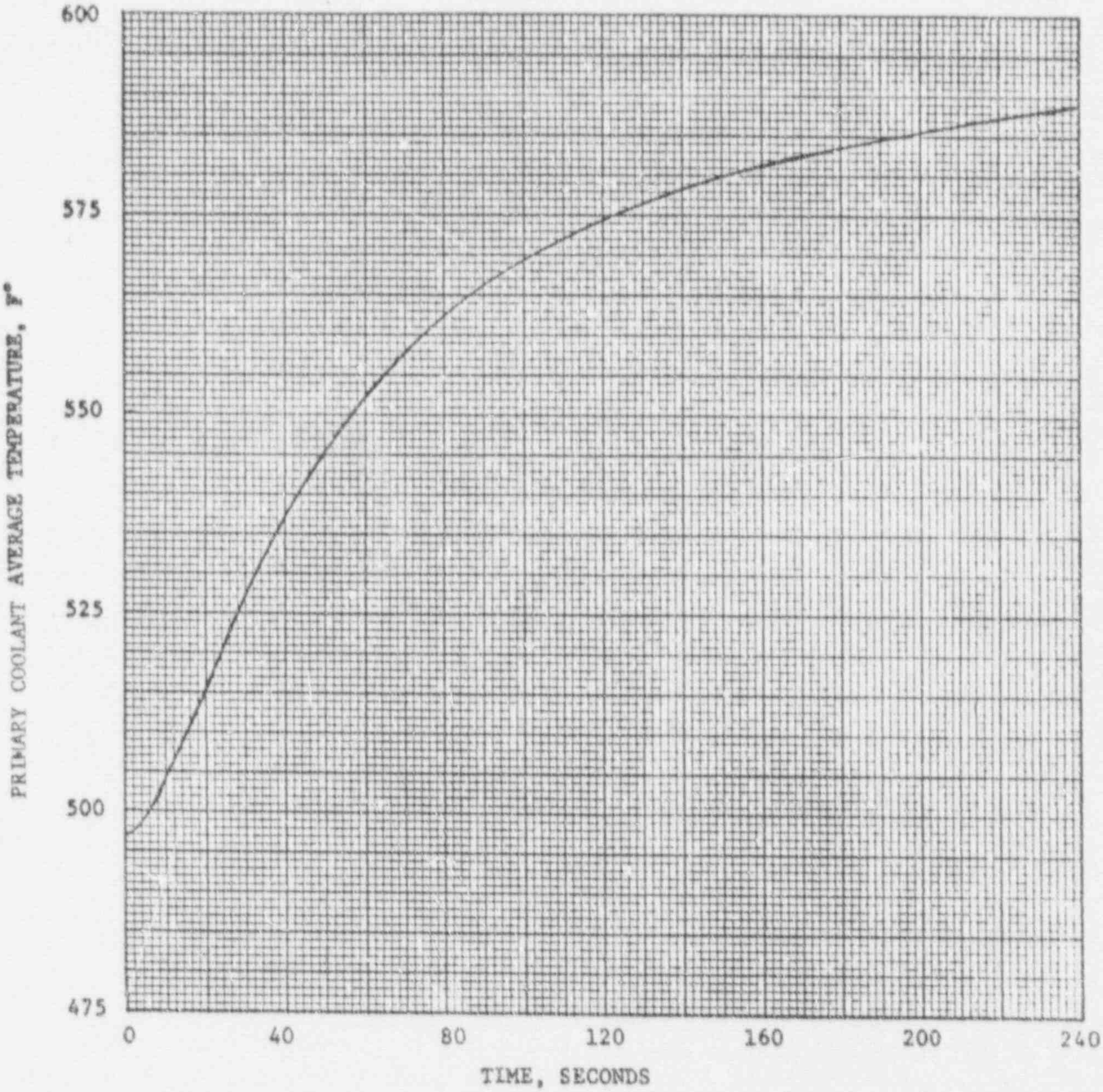


LOSS OF LOAD INCIDENT
FIGURE VI-3



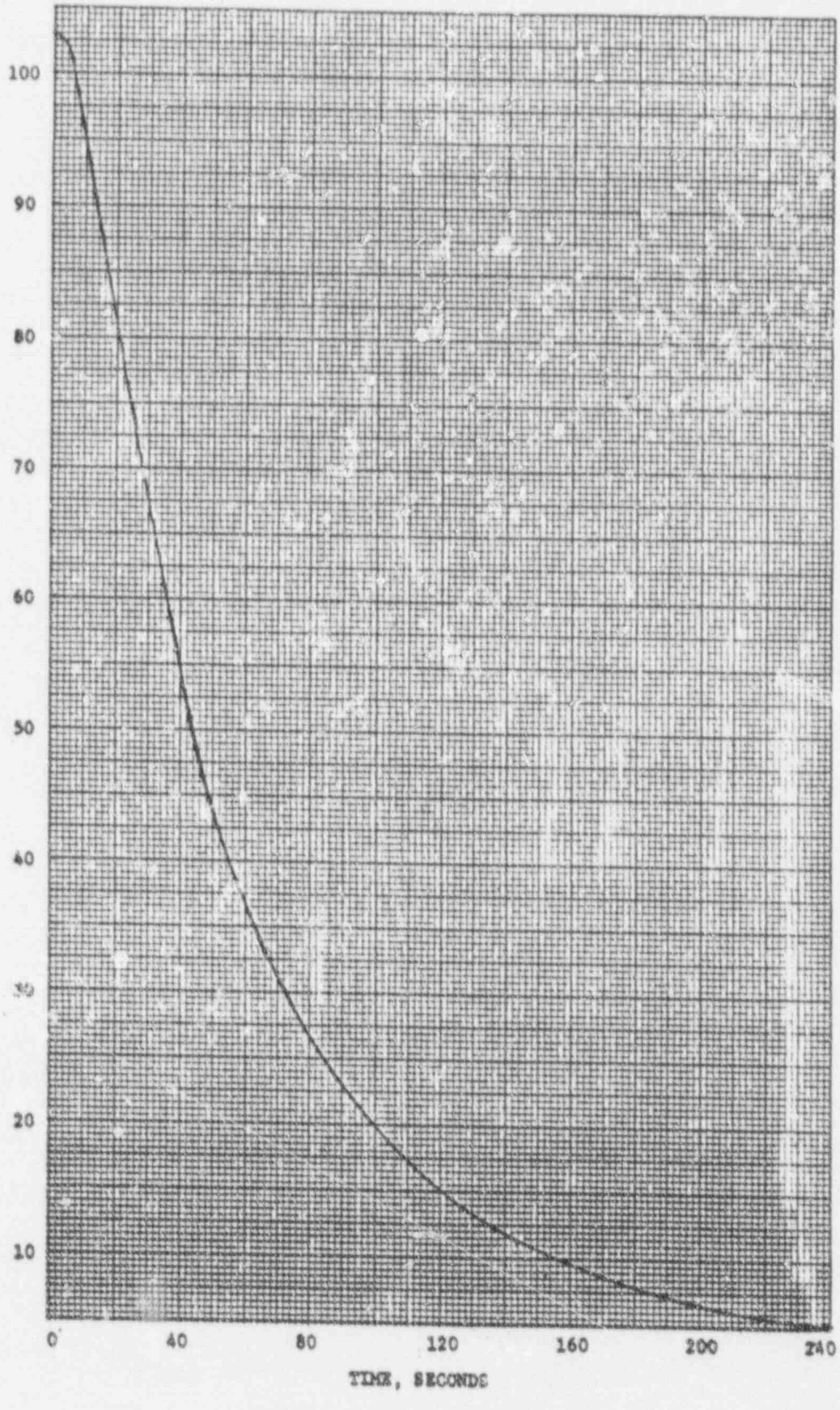
LOSS OF LOAD INCIDENT

FIGURE VI-4



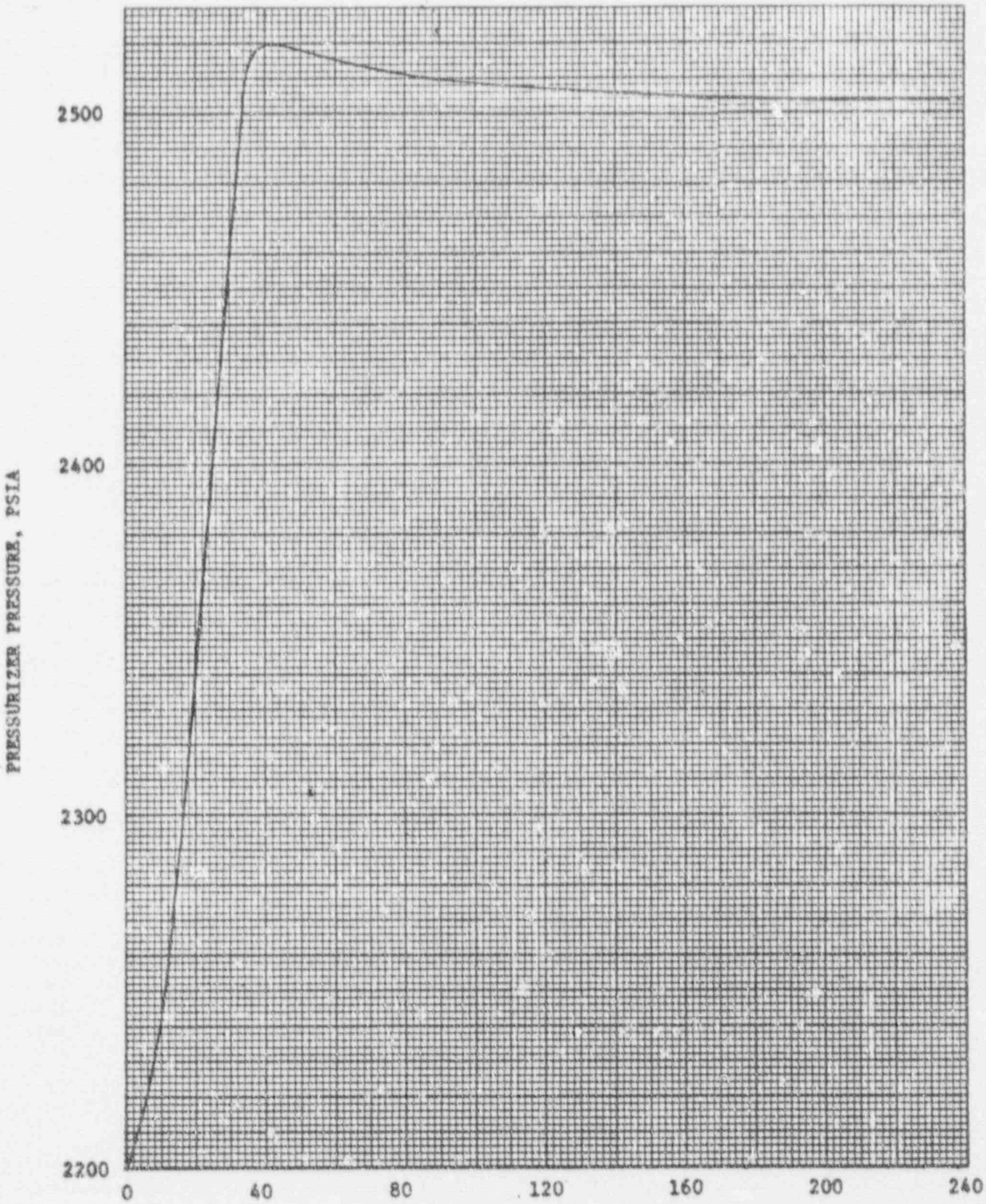
LOSS OF LOAD INCIDENT

FIGURE VI-5



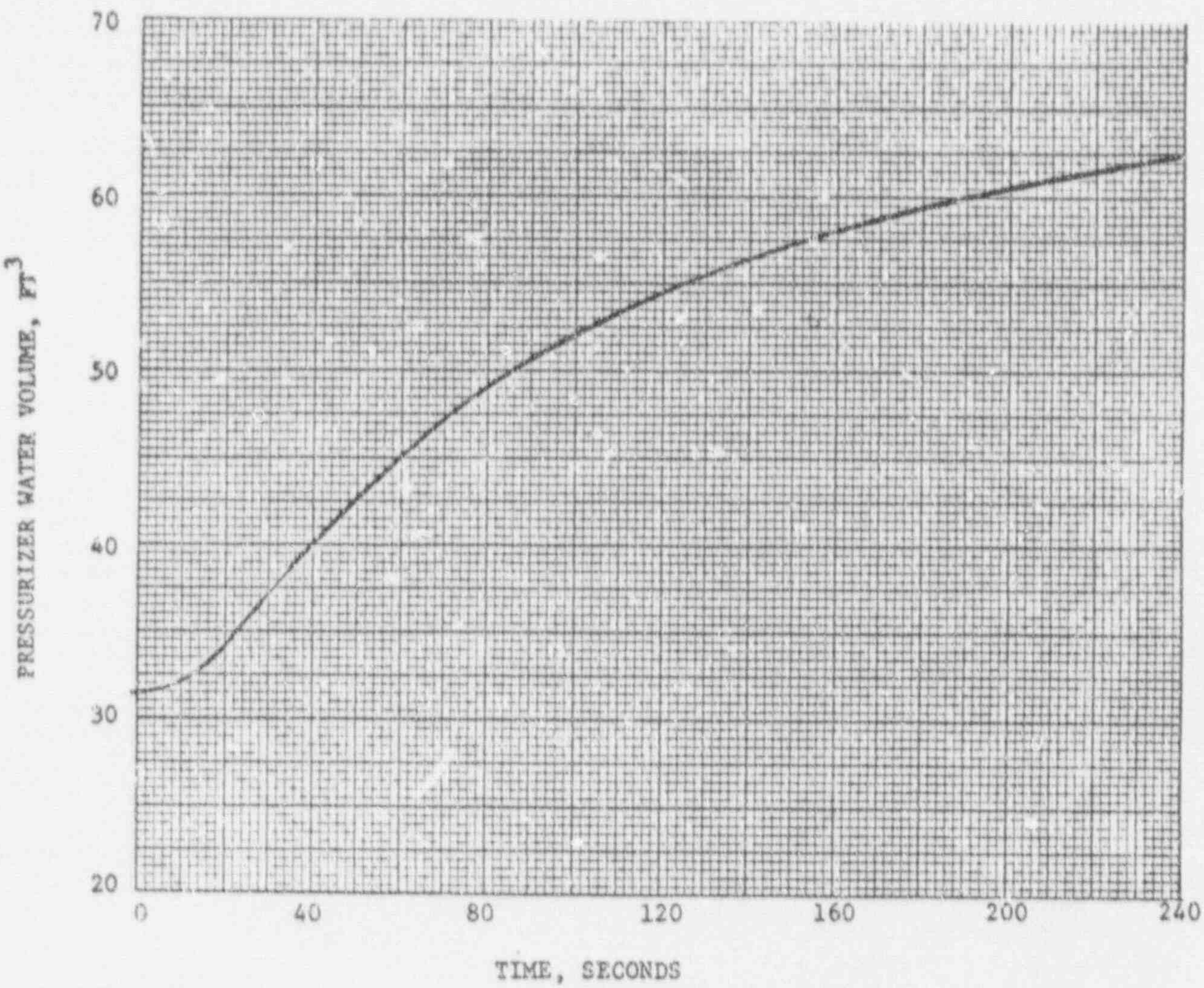
LOSS OF LOAD INCIDENT

FIGURE VI-6



LOSS OF LOAD INCIDENT

FIGURE VI-7



LOSS OF LOAD INCIDENT

FIGURE VI-8

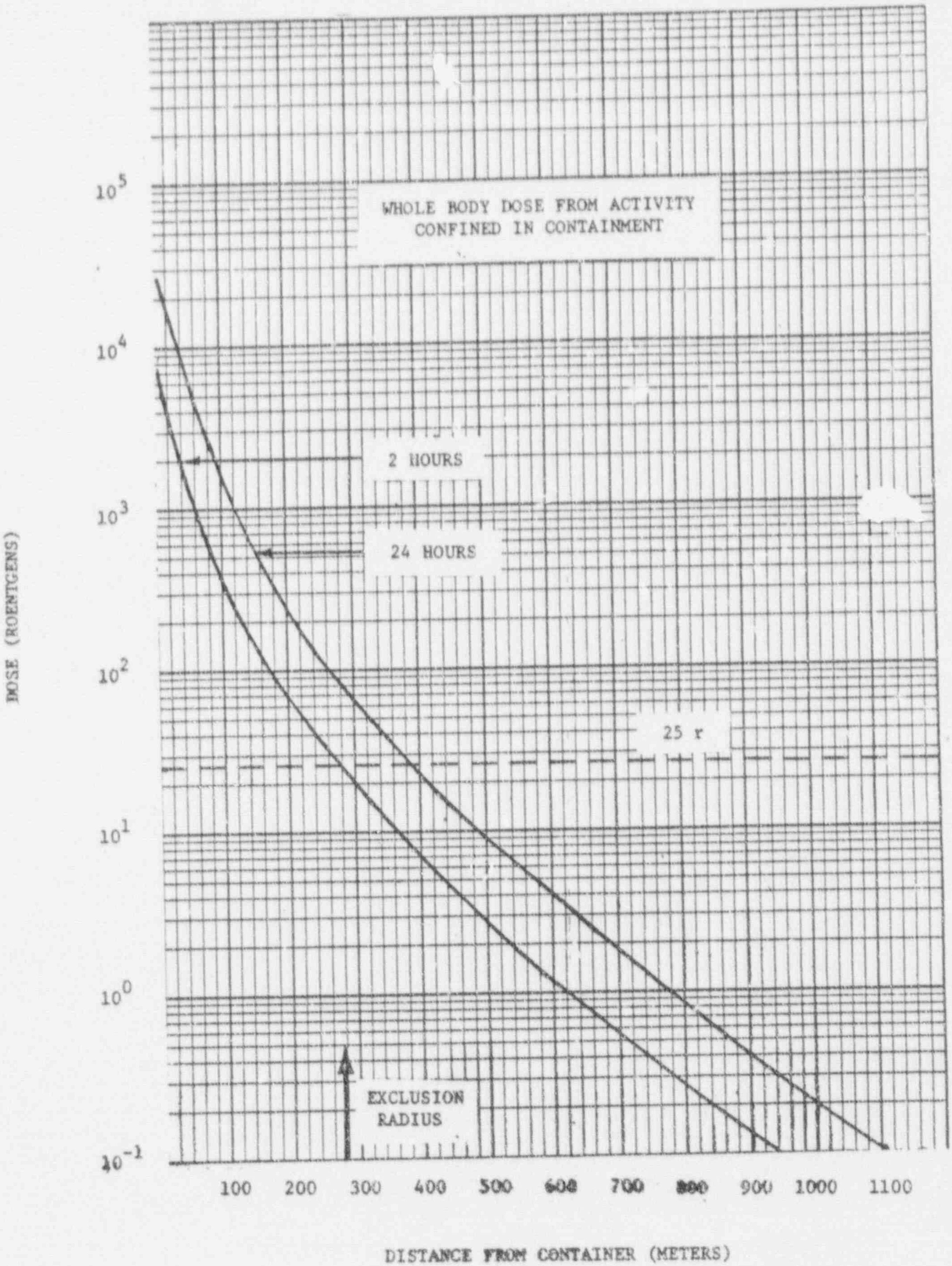
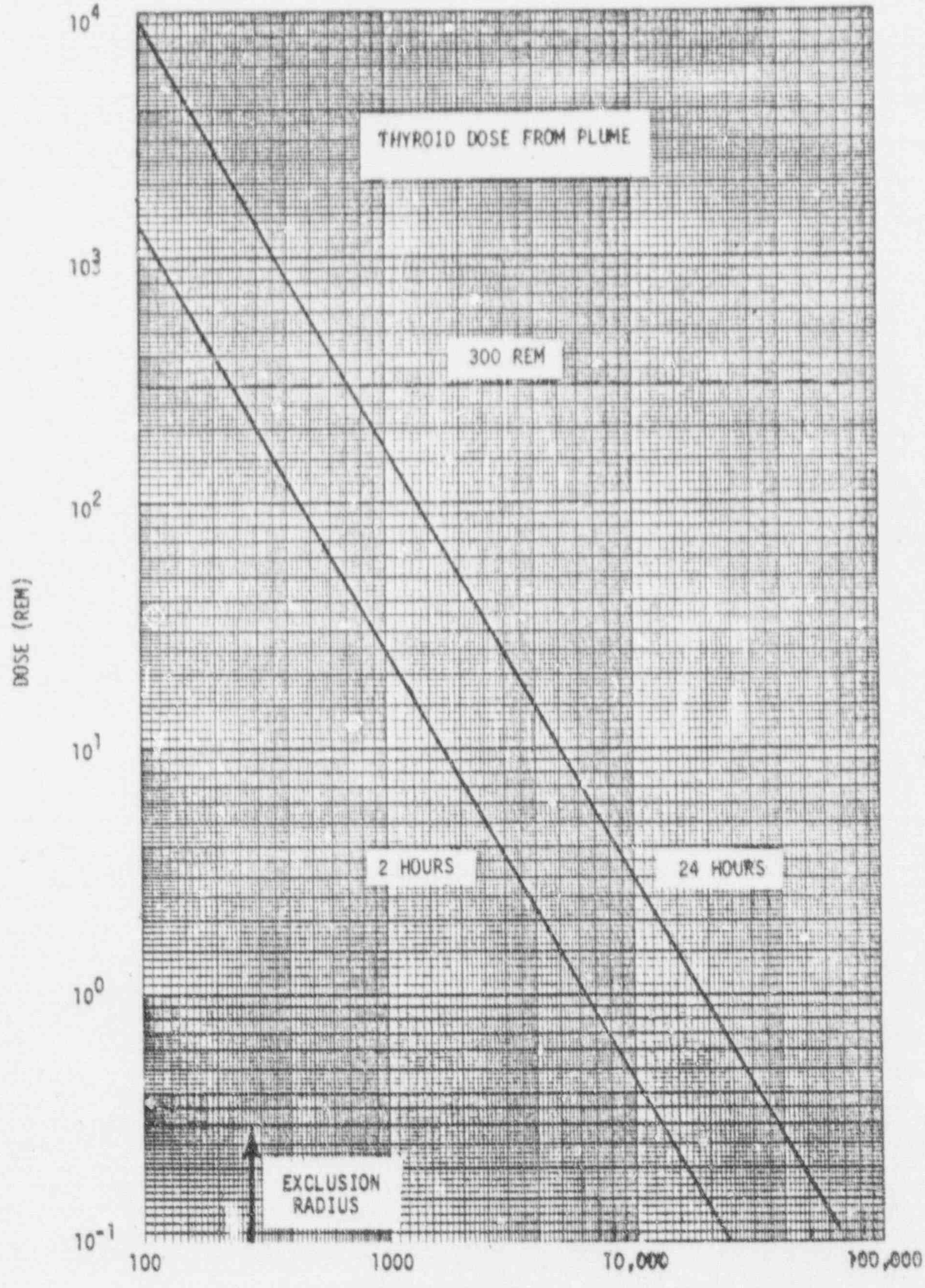

WHOLE BODY DOSE FROM ACTIVITY CONFINED IN CONTAINMENT
DOSE (ROENTGENS)
10^5
10^4
10^3
10^2
10^1
10^0
10^{-1}
2 HOURS
24 HOURS
25 r
EXCLUSION RADIUS
100
200
300
400
500
600
700
800
900
1000
1100
DISTANCE FROM CONTAINER (METERS)

FIGURE VI-9



THYROID DOSE FROM PLUME
300 REM
2 HOURS
24 HOURS
EXCLUSION RADIUS
DOSE (REM)
10^4
10^3
10^2
10^1
10^0
10^{-1}
100
1000
10,000
100,000
DISTANCE FROM CONTAINMENT (METERS)

FIGURE VI-10

VII. INSTRUMENTATION

In order to provide experimental measurements and continuous monitoring of conditions in the core, the Saxton Reactor was provided with a relatively extensive in-core instrumentation system. The system is capable of measuring outlet temperature, flow rate, pressure drop and the magnitude and axial distribution of the neutron flux. No modifications of the system are required or planned.

VIII. CONCLUSION

The plant modifications in conjunction with the changed operating condition and protection device setpoints will prevent core damage in the credible incidents. Analysis of a hypothetical accident much more severe than the credible incidents has shown that the plant design and the available separation distances result in off-site doses which comply with the siting regulations (10 CFR 100) when the operating power is 35 MWt.

IX. APPENDIX

WESTINGHOUSE EXPERIENCE

WITH

HIGH POWER LEVEL FUEL RODS

IX. APPENDIX

WESTINGHOUSE EXPERIENCE WITH HIGH POWER LEVEL FUEL RODS

Westinghouse experience with high power level fuel rods is illustrated in this section. Emphasis is given to the experiments developed with Zircaloy clad fuel rods. The main experiments with stainless steel clad fuel rods are only briefly summarized.

A. IRRADIATION OF SIX CAPSULES CONTAINING SAMPLES FROM THE CVTR CORE IN THE WESTINGHOUSE TEST REACTOR (MARCH AND NOVEMBER 1960)

1. General

As part of the CVTR Research and Development Program, a series of capsule irradiation experiments [3] was devised to define more clearly the thermal performance capabilities of sintered UO_2 pellets contained in Zircaloy-2 cladding.

The philosophy of the program was to carefully control irradiation test in order to obtain unambiguous thermal performance data for use by reactor designers. The parameters to be evaluated at various fuel rod power levels included the effect of the initial cold diametral fuel-to-clad gap on UO_2 surface and center temperatures and on the cladding stress, due to the fuel-clad differential expansion.

The capsules were designed to minimize errors and potential problems which would lead to difficulty in interpreting the experimental results, and to minimize variation in both radial and axial thermal neutron flux.

2. Description of Experiment

Six capsules denoted R-1, R-2, R-4, R-5, R-6, R-8 and R-11, each containing three fuel rods, were irradiated at fuel rod power levels of from 11 to 24 kw/ft (360 to 785 w/cm).

The fuel rod configuration used was a Zircaloy-2 tube containing a column of UO_2 pellets, having a fuel length of about 5 inches. All UO_2 pellets used were right circular cylinders 0.430 inches in diameter with a nominal density of 10.3 g/cm^3

(94% of theoretical). The inside and outside diameters of the Zircaloy-2 cladding were varied to obtain the required cold diametral gaps; however, the cladding wall thickness was maintained at 0.032 inches in all cases. A 0.080 inch axial plenum was provided in all fuel rods to accommodate axial thermal expansion of the UO_2 pellet column relative to the Zircaloy-2 cladding.

Cold diametral gaps of 0.006, 0.012 and 0.025 inches were selected. Three fuel rods, each with a different cold diametral gap, were irradiated simultaneously to eliminate possible variations in fuel rod power level. Different fuel rod power levels were obtained by using UO_2 pellets of different U-235 content.

An irradiation time of 40 hours was chosen to allow fuel redistribution due to either sintering of the UO_2 or other time and temperature dependent phenomena. Gross thermal cycling of the UO_2 fuel was not desired. The 40 hours irradiation time was short enough to preclude the possibility of a major change in thermal neutron flux because of a reactor trip or significant changes in control rod positions.

Table 1 summarizes the basic fuel rod parameters used for the six rabbit capsule irradiations.

3. Post Irradiation Examination

After irradiation, the rabbit capsules were examined in the WTR hot cells. The capsules were disassembled and the fuel rod samples removed. Length and diameter measurements were taken on all the fuel rod samples to detect any deformation of the Zircaloy-2 cladding which may have occurred because of interaction with the UO_2 or because of internal gas pressure buildup as the result of excessive fission gas release.

a. Fuel Rod Dimensional Changes

The diameter and length of all the fuel rods were measured after irradiation to establish whether swelling or other deformation of the cladding had occurred during irradiation.

Table IX-1

RABBIT CAPSULE NUMBER	FUEL ROD NUMBER	FUEL ENRICHMENT	NOMINAL COLD DIAMETRAL CLEARANCE	POWER LEVEL	LINEAR POWER OUTPUT	SURFACE HEAT FLUX AT FUEL ROD O.D.
		% U-235	Inches	kW/Ft	W/Cm	$\frac{Btu}{Hr, Ft^2} \times 10^3$
R-1	1-1	2.6	0.006			286
	1-2	2.6	0.012	11.0 ± 0.5	360	283
	1-3	2.6	0.025			277
R-2	2-1	2.6	0.006			286
	2-2	2.6	0.006	11.0 ± 0.5	360	286
	2-3	2.6	0.025			277
R-4	4-1	3.8	0.006			417
	4-2	3.8	0.012	16.0 ± 0.8	525	412
	4-3	3.8	0.025			402
R-6	6-1	5.2	0.006			470
	6-2	5.2	0.012	18.0 ± 1.0	590	464
	6-3	5.2	0.025			454
R-8	8-1	5.2	0.012			464
	8-2	5.2	0.012	18.0 ± 1.0	590	464
	8-3	5.2	0.012			464
R-11	11-1	7.6	0.006			624
	11-2	7.6	0.012	24.0 ± 1.2	785	618
	11-3	7.6	0.025			602

Note: UO_2 pellets 0.430 inch diameter, nominal 94% dense, O/U ratio 2.00-2.01
Zircaloy cladding dimensions varied to give required cold diametral clearance
Irradiated 40 hours in the WTR rabbit tube facilities.

The following measurements were made:

-Overall length: two readings for each measurement;

-Diameters at the following positions:

One-inch from top: two measurements, 90° apart,

Center: as above,

One-inch from bottom: as above.

The diameter measurements taken 90° apart were in agreement with each other within the recognized precision limits ($\pm$ 0.0005 inches).

Table 2 summarizes the pre- and post-irradiation measurements for the fuel rod samples from the rabbit capsules.

No significant dimensional changes occurred during irradiation of the fuel rod samples. In some cases; i.e., fuel rods with initial diametral gaps of 0.006 inches operated at the higher fuel rod power levels (fuel rods No. 4-1, 6-1 and 11-1, as listed in Table 1), the radial thermal expansion of the UO_2 fuel relative to the cladding should have resulted in a zero diametral gap during irradiation. No significant diametral changes were noted on these fuel rods indicating that the interfacial pressures between the UO_2 fuel and the Zircaloy-2 cladding were not sufficient to plastically deform the clad.

b. Metallography of Fuel

Selected cross sections of UO_2 samples taken from all fuel rods were pressure mounted in an epoxy resin for subsequent metallographic preparation. The radii corresponding to various microstructural features such as equiaxed grain growth, columnar grain formation, and changes associated with the solidification of molten UO_2 were measured to establish radial temperature profiles. No fuel rod (0.006, 0.012 and 0.025 initial diametral gap fuel rods) experienced center melting while operating at 11, 16 and 18 kw/ft. Only the fuel

Table IX-2

DIMENSIONAL MEASUREMENTS ON FULL ROD SAMPLES FROM THE RABBIT CAPSULES

(Precision ± 0.0005 inches)

Rabbit Capsule No.		R-1			R-2			R-4		
		Pre Inches	Post Inches	Change Inches	Pre Inches	Post Inches	Change Inches	Pre Inches	Post Inches	Change Inches
Fuel Rod Number		1-1			2-1			4-1		
*Length	l	6.412	6.413	+0.001	6.412	6.411	-0.001	6.419	***6.410	***-0.009
**Diameter	d_1	0.500	0.502	+0.002	0.501	0.501	0	0.501	0.501	0
	d_2	0.502	0.502	0	0.502	0.502	0	0.502	0.502	0
	d_3	0.501	0.502	+0.001	0.501	0.501	0	0.501	0.502	+0.001
Fuel Rod Number		1-2			2-2			4-2		
Length	l	6.416	6.418	+0.002	6.408	6.409	-0.001	6.417	***6.409	***-0.008
Diameter	d_1	0.505	0.505	0	0.501	0.501	0	0.505	0.505	0
	d_2	0.506	0.505	0	0.501	0.502	+0.001	0.506	0.506	0
	d_3	0.506	0.506	0	0.501	0.501	0	0.506	0.506	0
Fuel Rod Number		1-3			2-3			4-3		
Length	l	6.414	6.414	0	6.411	6.409	-0.002	6.410	6.409	-0.001
Diameter	d_1	0.518	0.519	+0.001	0.519	0.519	0	0.518	0.519	+0.001
	d_2	0.518	0.519	+0.001	0.518	0.519	+0.001	0.519	0.519	0
	d_3	0.518	0.518	0	0.518	0.519	+0.001	0.518	0.518	0

Rabbit Capsule No.		R-6			R-8			R-11		
		Pre Inches	Post Inches	Change Inches	Pre Inches	Post Inches	Change Inches	Pre Inches	Post Inches	Change Inches
Fuel Rod Number		6-1			8-1			11-1		
Length	l	6.4240	6.4210	-0.003	6.4140	6.4120	-0.002	6.4110	6.4110	0
Diameter	d_1	0.4985	0.4995	+0.0010	0.5035	0.5036	+0.0001	0.4984	0.4986	+0.0002
	d_2	0.4994	0.5004	+0.0010	0.5039	0.5046	+0.0007	0.4993	0.5002	+0.0009
	d_3	0.4990	0.5006	+0.0016	0.5036	0.5043	+0.0007	0.4992	0.5001	+0.0009
Fuel Rod Number		6-2			8-2			11-2		
Length	l	6.4160	6.4130	-0.003	6.4100	6.4094	-0.0006	6.4140	6.4150	+0.001
Diameter	d_1	0.5034	0.5033	-0.0001	0.5033	0.5034	+0.0001	0.5032	0.5038	+0.0006
	d_2	0.5039	0.5051	+0.0012	0.5046	0.5048	+0.0002	0.5041	0.5048	+0.0006
	d_3	0.5035	0.5046	+0.0011	0.5041	0.5043	+0.0002	0.5037	0.50[illegible]7	+0.0010
Fuel Rod Number		6-3			8-3			11-3		
Length	l	6.4090	6.4070	-0.002	6.4060	6.4050	-0.001	6.4070	6.4040	-0.003
Diameter	d_1	0.5157	0.5167	+0.0010	0.5030	0.5036	+0.0006	0.5159	0.5165	+0.0006
	d_2	0.5164	0.5173	+0.0009	0.5041	0.5045	+0.0004	0.5161	0.5160	-0.0001
	d_3	0.5161	0.5171	+0.0010	0.5035	0.5045	+0.0010	0.5159	0.[illegible]68	+0.0009

* Length: Overall length, two readings made for each measurement. Number listed is the average of the two readings.

** Diameter: For each diameter measurement, two readings were made each 90° apart. Number listed is the average of both measurements.

d_1 - measurements made 1 inch from fuel rod top

d_2 - measurements made at center

d_3 - measurements made 1 inch from fuel rod bottom

*** Burrs formed during disassembly filed off prior to making length measurements.

rod with 0.025 inches initial diametral gap experienced melting in the central region. But no significant diametral change was measured, as shown in Table 2. The fuel rods with 0.006 and 0.012 inches initial diametral gap and which operated at 24 kw/ft experienced no center melting. It can be concluded that the surface and center temperatures of sintered UO_2 fuel can be lowered by decreasing the initial diametral ga and thus allowing reactor operation at 24 kw/ft without central melting.

B. IRRADIATION OF TWO CAPSULES CONTAINING SAMPLES FROM THE CVTR CORE IN THE WESTINGHOUSE TEST REACTOR (MARCH-JULY 1962)

1. General

Capsule irradiations [4] of UO_2 fuel rods were performed to evaluate the effect of fuel rod power level on cladding dimensional changes and fission gas release.

Two capsules, designated A-2 and A-4, were irradiated in the Westinghouse Test Reactor. On capsule contained three fuel fuel rods with a 38 inch fuel length and was irradiated at peak fuel rod power levels of 19 kw/ft to a maximum fuel burnup of 3,450 $\frac{MWD}{MTU}$. The other capsule contained four fuel rods with 6 inch fuel lengths. Peak fuel rod power levels of 22.2 kw/ft were measured with irradiation to 6,250 $\frac{MWD}{MTU}$.

2. Description of Experiment

To evaluate the effect of the initial cold diametral fuel-to-clad gap on the radial and axial thermal expansion of the UO_2 fuel relative to the cladding, a range of gaps was selected so that high interfacial pressure would exist during operation in some of the fuel rods while a finite hot gap should exist at all times in others when operated at the same power level.

Cold diametral gaps of 0.002, 0.005 and 0.012 inches were used. The Zircaloy-2 cladding dimensions were varied to obtain the desired diametral clearances while keeping the wall thickness constant at 0.032 inches and the UO_2 pellet diameter constant at 0.430 inches.

The various initial cold diametral gaps used in the fuel rods also resulted in different UO_2 fuel surfaces and center temperatures allowing the fractional fission gas release from UO_2 to be measured at different average UO_2 fuel temperatures.

Two capsules, A-2 and A-4, were irradiated in the WTR. Table 3 summarizes the parameters used for the design of the fuel rods for the two capsule irradiations.

a. A-2 Capsule Design

Capsule A-2 was designed primarily to evaluate UO_2 thermal expansion relative to the Zircaloy-2 cladding with different initial pellet-to-clad gaps. The three fuel rods contained a 38 inch long column of UO_2 pellets. The fuel enrichment was varied along the length of the fuel rods to maximize the length of fuel operating at high temperatures.

All UO_2 pellets used were right circular cylinders 0.430 inches in diameter with a nominal density of 10.3 g/cm^3 (94% of theoretical). The inside and outside diameters of the Zircaloy-2 cladding were varied to obtain the various cold diametral gaps; however, the cladding wall thickness was maintained at 0.032 inches in all cases.

b. A-4 Capsule Design

Capsule A-4 was designed primarily to evaluate the fission gas release from sintered UO_2 in fuel rods operating with high UO_2 center temperatures.

The various initial cold diametral gaps used would result in different UO_2 fuel surface and center temperatures. The fuel rod configuration used was a Zircaloy-2 tube containing a column of seven UO_2 pellets, 0.860 inches long, giving a fuel length of about 6 inches. A 0.100 inch axial gap was provided in the fuel rods to accommodate any axial expansion of the UO_2 pellet column relative to the cladding.

Capsule A-4 contained four fuel rods with fuel of the same enrichment. The fuel rods had the same initial pellet-to-clad gaps as the A-2 capsule fuel rods.

Table IX-3

SUMMARY OF FUEL CAPSULE IRRADIATION EXPERIMENT PARAMETERS

Capsule Number	Fuel Rod Number	Fuel Column Length	Fuel Enrichment	Nominal Initial Diametral Clearance	Peak Fuel Rod Power Level (Actual)		Fuel Rod Nominal O.D.	Maximum Surface Heat Flux at Fuel Rod O.D.
		inches	% U-235	inches	Kw/ft	W/cm	Inches	$\frac{Btu}{hr\ ft^2} \times 10^3$
A-2	2-1	38	Variable	0.002	19.0	624	0.496	498
	2-2	38	Along Length	0.006	19.0	624	0.500	495
	2-3	38	5.7 to 8.5%	0.012	19.0	624	0.506	488
A-4	4-1	6	4.5	0.002	22.2	727	0.496	580
	4-2	6	4.5	0.006	22.2	727	0.500	576
	4-3	6	4.5	0.012	22.2	727	0.506	568
	4-4	6	4.5	0.012	22.2	727	0.506	568

Note: UO_2 pellets 0.430 inch diameter, nominal 98% dense, O/U ratio 2.00 - 2.01

Zr-2 cladding dimensions varied to give initial diametral clearances

3. Post Irradiation Examination

After irradiation the A-2 and A-4 capsule fuel rods were examined in the WTR hot cells.

The capsules were disassembled and the fuel rod samples removed. Diameter and overall length measurements were taken on all the fuel rod samples to detect any deformation of the Zircaloy-2 cladding which may have occurred because of interaction with the UO_2 or because of internal gas pressure as the result of high fission gas releases.

a. Capsule A-2 Fuel Rod Dimensional Measurements

The post irradiation diameters of the three A-2 capsule fuel rods were measured at various positions along the lengths of the fuel rods. The overall fuel rod lengths were also measured. Table 4 summarizes the diameter and length measurements made on the A-2 capsule fuel rods.

Within the accuracy of the measurements no significant diameter or length changes occurred during the irradiation of fuel rods A-2-2 and A-2-3 which had initial cold diametral clearances between the UO_2 pellets and the Zircaloy-2 cladding of 0.006 and 0.012 inches, respectively.

In the case of fuel rod A-2-1, which had an initial cold diametral gap of 0.002 inches, fuel rod diameter increases were found near the lower end of the fuel rod and some elongation of the fuel rod has occurred.

The Zircaloy-2 cladding used to fabricate this fuel rod had an inside diameter of 0.432 inches and 0.032 inch nominal wall. The UO_2 pellets used in all fuel rods were 0.430 inches in diameter.

These diameter changes are attributed to deformation of the Zr-2 cladding by the UO_2 pellets as they thermally expand radially against the cladding.

Fuel rods with an initial 0.006 inch cold diametral gap or greater showed no measurable diameter change when irradiated at comparable power levels.

Table IX-4

DIAMETER AND LENGTH MEASUREMENTS ON FUEL ROD SAMPLES FROM CVTR CAPSULE A-2

Fuel Rod Number	Initial Cold Diametral Gap (inches)	Diameter Measured (1)	Pre (2) (in.)	Post (2) Precision ± 0.0005 (in.)	Change in Diameter mils	Change in Diameter inches	Fuel Rod Lengths ± 0.002 Pre-Irradiation (inches)	Fuel Rod Lengths ± 0.002 Post-Irradiation (inches)	Overall Length Change (in.)	Uniform Change Over Entire Length $\frac{\text{In. x } 10^{-3}}{\text{In. of Length}}$	Uniform Change (3) Over 20-In. of Length $\frac{\text{In. x } 10^{-3}}{\text{In. of Length}}$
A-2-1	0.002	d_1	0.4963	0.4958	-0.5	-0.0005	40.943	40.960	+0.017	0.43	0.87
		d_2	0.4960	0.4958	-0.2	-0.0002					
		d_3	0.4958	0.4980	+2.2	+0.0022					
		d_4	0.4943	0.4973	+3.0	+0.0030					
A-2-2	0.006	d_1	0.5019	0.5014	-0.5	-0.0005	40.955	40.959	+0.004	0.09	----
		d_2	0.5020	0.5012	-0.8	-0.0008					
		d_3	0.5021	0.5019	-0.2	-0.0002					
		d_4	0.5014	0.5020	+0.6	+0.0006					
A-2-3	0.012	d_1	0.5066	0.5052	-1.4	-0.0014	40.965	40.971	+0.006	0.15	----
		d_2	0.5066	0.5064	-0.2	-0.0002					
		d_3	0.5061	0.5065	+0.4	+0.0004					
		d_4	0.5051	0.5051	0	0					

(1) d_1 - Measurement made 14 inches from top of fuel rod
d_2 - Measurement made 26 inches from top of fuel rod
d_3 - Measurement made 32 inches from top of fuel rod
d_4 - Measurement made 38 inches from top of fuel rod

(2) For each diameter measurement both pre- and post-irradiation, two readings were taken 90° apart. Number listed is the average of both measurements.

(3) Interaction between UO_2 pellets and Zircaloy-2 cladding in fuel rod A-2-1 was great enough to cause radial cladding deformation over about 20 inches of length.

b. Capsule A-4 Fuel Rod Dimensional Measurements

Table 5 summarized the diameter and length measurements made on the A-4 capsule fuel rods.

Within the accuracy of the measurements, no significant dimensional changes were noted except in the case of fuel rod A-4-1 which had an initial 0.002 inch cold diametral pellet-to-clad clearance. The diameter of this fuel rod increased slightly due to the radial thermal expansion of the UO_2 pellets against the cladding.

c. Metallography of Fuel

Selected cross sections of UO_2 samples taken from the A-2 and A-4 capsule fuel rods were pressure mounted in an epoxy resin for subsequent metallographic preparation. The radius corresponding to various micro-structural features of the UO_2 such as equiaxed grain growth and columnar grain information was measured to establish radial temperature profile.

No evidence of melting in the UO_2 can be seen in any of the samples irradiated in the A-2 and A-4 capsules.

4. Conclusions

The results from the A-2 and A-4 capsule irradiation experiments enable some general conclusions pertaining to the design of sintered UO_2 fuel rods.

The results obtained from the examination of the Zircaloy-2 clad UO_2 fuel rods indicate that extended operation at fuel rod power levels of 18-22 kw/ft can be achieved without failure or fuel rod dimensional changes if the initial fuel-to-clad gap is large enough to accommodate the relative radial expansion of the UO_2 fuel against the cladding. The initial diametral gap between the UO_2 and the cladding selected for the Saxton rods will not result in cladding diameter increases due to thermal expansion of the UO_2.

Table IX-5

DIAMETER MEASUREMENTS ON FUEL ROD SAMPLES FROM CVTR CAPSULE A-4

Fuel Rod Number	Initial Cold Diametral Gap (inches)	Diameter Measured (1)	Pre (2) (inches)	Post (3) (inches)	Change in Diameter (mils)	Change in Diameter (inches)
A-4-1	0.002	d_1	0.4940	0.4956	+ 1.6	+ 0.0016
		d_2	0.4940	0.4955	+ 1.5	+ 0.0015
		d_3	0.4940	0.4956	+ 1.6	+ 0.0016
A-4-2	0.006	d_1	0.5017	0.5020	+ 0.3	+ 0.0003
		d_2	0.5017	0.5012	- 0.5	- 0.0005
		d_3	0.5017	0.5016	- 0.1	- 0.0001
A-4-3	0.012	d_1	0.5064	0.5064	0	0
		d_2	0.5064	0.5059	- 0.5	- 0.0005
		d_3	0.5064	0.5063	- 0.1	- 0.0001
A-4-4	0.012	d_1	0.5064	0.5067	+ 0.3	+ 0.0003
		d_2	0.5064	0.5054	- 1.0	- 0.0010
		d_3	0.5065	0.5065	+ 0.1	+ 0.0001

(1) d_1 = Measurements made 1 inch from top of fuel rod.
d_2 = Measurements made at center of fuel rod.
d_3 = Measurements made 1 inch from bottom of fuel rod.

(2) Average of all measurements made prior to irradiation.

(3) For each diameter measurement after irradiation, two readings were taken 90° apart. Number listed is average of both measurements.

Precision ± 0.0005 inches

C. LRD IN-PILE TESTS PROGRAM IN THE SAXTON REACTOR

1. General

The purpose of this program has been:

a) To perform in-pile proof tests to verify technical feasibility of prototype designs, materials, and fabrication variables proposed for use in a large plant chemical shim environment, and

b) to perform fuel and cladding experiments aimed at reducing overall fuel cycle and plant costs.

A series of subassemblies, which in most cases represents a combination of these objectives, has been irradiated in the Saxton Reactor. The present status of each experiment and significant results to date are detailed in Table 6.

In the subsequent section the performance of Zircaloy clad fuel rods is examined.[5] [6]

2. Description of Experiments and Post Irradiation Examination of High Power Level Zircaloy Clad Fuel Rods

Table 7 summarizes the Zircaloy portion of the LRD irradiations program.

The type of Zircaloy used as cladding material, the peak power level and the peak burnup are reported for each experimental fuel rod.

a. Zircaloy Clad Fuel Rods from Saxton Modified 3x3 Subassembly No. 503-4-23

Evaluation of the in-pile performance of the Zircaloy clad fuel rods irradiated as part of 3x3 subassembly No. 503-4-23 was completed in the period April-June 1965. Two as-pickled and three autoclaved pre-oxidized Zircaloy clad fuel rods designed to operate at 16 kw/ft (530,000 BTu/hr-ft^2) were irradiated. The rods operated as part of the Saxton core for a total of approximately 58 effective full power days at a maximum clad surface temperature of 640°F. During this time the rods achieved a burnup of approximately 3000 MWD/MTU.

Table IV-6

[illegible] PROOF TESTING OF ADVANCED FUEL ASSEMBLY DESIGNS

SUMMARY OF IN-PILE EXPERIMENTS

Ass'y Ident.	Assembly Description	Objective(s) of Test	Date In-Core	Date Out-of-Core	In-Core Location	Est. Effective Full Power Days Exposure	Estimated Peak Burnup (MWD/MTU)	Estimated Peak Exposure (nvt > 1 MeV)	Significant Results	Present Status and Location
	[illegible] Loop Experiment	1. Demonstrate compatibility of S/S clad fuel rods and chem shim environment 2. Performance of candidate control rod materials 3. Dished vs flat end fuel pellets 4. Determine operating fuel center temperature	10-62	3-63	Core	93	[illegible]	1×10^{21}	1. No adverse effects on S/S clad [illegible] 2. [illegible] Ag-In-Cd control rod materials must be diffusion bonded prior to exposure 3. Operating fuel center temp. was successfully measured 4. Axial expansion surface did not function successfully	Exam complete. Stored at [illegible] until disposal
[illegible]-4-21	Modified Saxton 3 x 3 Assembly	1. Performance of thin-walled S/S clad 2. Performance of furnace pre-oxidized Zr clad in chem shim environment 3. Comparison of crud buildup between Zr and S/S	3-63	12-63	Center	190	[illegible]	1.1×10^{21}	1. Satisfactory performance of S/S clad although subjected to severe local plastic deformation. Fretting due to rod design was observed 2. Satisfactory performance of Zr clad in chem shim environment. Corrosion and hydrogen pickup equivalent to out-of-pile data.	Exam complete. Stored at [illegible] until disposal at completion of program.
[illegible]-4-2[illegible]	Modified Saxton 3 x 3 Assembly	Same as above	5-63	12-63	Periphery	190	[illegible]	4×10^{20}	Same as above	Same as above
[illegible]-4-23	Modified Saxton 3 x 3 Assembly (RCC Test)	1. Proof test static simulated RCC elements in chem shim environment 2. Compare performance of as-pickled and autoclave tested Zr clad rods operated at [illegible] kw/ft in chem shim 3. Performance of Zr with nucleate boiling	1-64	5-64	Center	60	[illegible]	3.5×10^{20}	1. RCC elements performance satisfactory; no crud buildup or change in force required to withdraw rods 2. Zr clad rods performance satisfactory. No difference between as-pickled and autoclave tested rods 3. No fretting 4. Light crud deposition	Same as above
[illegible]-4-24	Modified Saxton 3 x 3 Assembly (Defect Test)	1. Determine if boric acid in coolant enhances a [illegible] reaction or hydriding on the inside surface of Zr clad as a function of burnup and exposure 2. Compare performance of Zr and S/S clad [illegible] fuel elements 3. Extend exposure on Zr clad in chem shim environment 4. Performance of sensitized 304 S/S clad	6-64	10-64	Center	56	[illegible]	3.7×10^{20}	1. In-pile performance satisfactory 2. No waterlogging, erosion or [illegible] reaction in vicinity of defect. No hydriding of I.D. surface of Zr clad 3. Defect in previously irradiated rod did not completely penetrate clad wall 4. Slight surface attack of sensitized 304 S/S clad. No intergranular cracking observed 5. No fretting 6. Light crud deposition	Same as above
[illegible]-4-25	Modified Saxton 3 x 3 Assembly (Improved Grid Test)	1. Proof test improved grid design 2. Expose new grid and braze materials to chem shim environment	10-64	1-65	Center	60	[illegible]	8.5×10^{20}	1. Satisfactory in-pile performance to date 2. No fretting observed 3. No cracking or attack observed on grid, braze or cladding materials 4. Heavy crud deposition following [illegible]	In-pile
		3. Performance of mechanically prepared Zr clad rods	3-65	8-65	Periphery	[illegible]	[illegible]	1.4×10^{20}		
		4. Repeat defect test of previously irradiated Zr clad rod	[illegible]	7-66	Periphery	[illegible]	[illegible]	2.5×10^{20}		
[illegible]-5-1	Special [illegible] Assembly	1. Determine fuel and fuel element behavior at high specific ratings	[illegible]	[illegible]	Periphery	[illegible]	[illegible]	8.5×10^{20}	1. Satisfactory in-pile performance to date 2. Light crud deposition	In-pile
			[illegible]	[illegible]	Center	[illegible]	[illegible]	4.0×10^{20}		
			[illegible]	[illegible]	Periphery	[illegible]	[illegible]	1.4×10^{20}		
			[illegible]	[illegible]	Center	[illegible]	[illegible]	4.7×10^{20}		
[illegible]-10-1	Special 9 x 9	1. Obtain statistically significant irradiation data to verify prototype designs, materials, and fabrication variables considered for use in large plant chem shim environment (see attachment)	6-64	8-65	Center	[illegible]	[illegible]	1.8×10^{21}	1. Satisfactory in-pile performance 2. No fretting, cracks, or abnormalities observed 3. No increase in RCC element withdrawal	Stored in Saxton [illegible] until completion of program

SUMMARY OF ZIRCALOY CLAD IN SAXTON

Saxton Assembly No.	Assembly Type	No. of Zirc. Rods	Material	Condition	Surface Treatment	Full Power Days Exp.	Peak Power (kw/ft)	Peak Heat Flux (Btu/hr/ft)	Peak Burnup (MWD/MTU)
503-4-21	3x3	4	Zr-2 (Ni-free)	Annealed	Furnace Pre-oxidized)	190	13.4	450,000	9,500
503-4-23	3x3	4	Zr-2 (Ni-free)	Annealed	Furnace Pre-oxidized	190	4.5	150,000	3,400
503-4-23	3x3	3	Zr-2 (Ni-free)	Annealed	Autoclave Pre-oxidized	58	16.0	550,000	2,900
503-4-23	3x3	2	Zr-2 (Ni-free)	Annealed	As-pickled	58	16.0	550,000	2,900
503-4-24	3x3	3(a)	Zr-4	10% C.W.	Autoclave Pre-oxidized	56	14.0	465,000	2,900
503-4-24	3x3	1(b)	Zr-2 (Ni-free)	Annealed	Furnace Pre-oxidized	246	13.5	450,000	12,300
503-4-24	3x3	1(c)	Zr-2 (Ni-free)	Annealed	Furnace Pre-oxidized	246	13.5	450,000	12,300
503-4-24	3x3	1(d)	Zr-4	10% C.W.	Furnace Pre-oxidized	56	13.5	450,000	2,800
503-4-25	3x3	1	Zr-4	10% C.W.	Surf. Prep & Autoclave Preoxidized	99	14.0	465,000	3,300
503-4-25	3x3	1	Zr-4	10% C.W.	As-Surf. Prep.	99	14.0	465,000	3,300
503-4-25	3x3	1(e)	Zr-2 (Ni-free)	Annealed	Furnace	336	13.5	450,000	15,600
503-4-25	3x3	1(f)	Zr-2 (Ni-free)	Annealed	Autoclave	150	16.0	550,000	6,200
503-10-1	9x9	3	Zr-4	10% C.W.	Autoclave Pre-oxidized	155	12.0	400,000	8,000
503-10-1	9x9	3	Zr-4	10% C.W.	Autoclave Pre-oxidized	155	13.5	450,000	8,000
503-10-1	9x9	3	Zr-2	10% C.W.	Autoclave Pre-oxidized	155	14.0	465,000	8,000
503-10-1	9x9	3	Zr-2 (Ni-free)	10% C.W.	Autoclave Pre-oxidized	155	14.0	465,000	8,000
503-10-1	9x9	3	Zr-4	10% C.W.	Autoclave Pre-oxidized	155	14.0	465,000	8,000
503-10-1	9x9	3	Zr-4	10% C.W.	As-pickled	155	14.0	465,000	8,000
503-10-1	9x9	3	Zr-4	10% C.W.	Furnace Pre-oxidized	155	14.0	465,000	8,000
503-10-1	9x9	3	Zr-4	10% C.W.	Autoclave Pre-oxidized on O.D. & I.D.	155	14.0	465,000	8,000
503-10-1	9x9	3	Zr-4	10% C.W.	Autoclave Pre-oxidized	155	16.0	550,000	8,000

NOTES:

(a) Contain loose oxide UO_2 fuel.

(b) Previously irradiated in Assembly No. 503-4-21.

(c) Previously irradiated in Assembly No. 503-4-21. (Contains 15 mil dia. defect.)

(d) Contains 15 mil diameter defect.

(e) Previously irradiated in Assembly Nos. 503-4-21 and -24.

(f) Previously irradiated in Assembly No. 503-4-23.

The main purpose of the experiment was to determine the effect of preirradiation surface treatment on Zircaloy-clad fuel rods when exposed to nucleate boiling heat transfer conditions in a chemical shim PWR environment.

The post irradiation examination indicated satisfactory in-pile performance of both the pre-oxidized and as-pickled fuel rods irrespective of surface treatment prior to irradiation. No dimensional changes or abnormalities were observed on the fuel rod surfaces.

b. Saxton In-Pile Defect Test 3x3 Subassembly No. 503-4-24

Visual examination of the intentionally defected Zircaloy clad rods was completed at the Post Irradiation Facility hot cells. The examination yielded the following observations:

1) No fretting was observed.

2) Crud deposition was very light.

3) No indication of attack of the cladding or fuel was observed in the vicinity of the defect.

Therefore, even if a defect were to occur no further undesirable problem, such as defect enlargement, clad bursting, etc., is expected.

c. Saxton Special 9x9 Assembly No. 503-10-1

The operating characteristics of this assembly are shown in Table 7. In-core examination was successfully performed with the aid of a boroscope upon completion of the crud test. The assembly has experience approximately 8000 MWD/MTU. No indication of failure, cracking, or attack of either the Zircaloy or stainless steel cladding was observed.

d. Saxton Advanced Fuel Assembly No. 503-4-25

The operating data are shown on Table 7. The present general appearance of the subassembly is satisfactory. No fretting, cracking, or attack of the grid or cladding material was observed.

D. SAXTON SPECIAL 2x2 STAINLESS STEEL FUEL RODS SUBASSEMBLY NO. 503-9-1

Four fuel rods have operated successfully in the Saxton core at a peak power rating of 25 kw/ft to a burnup exceeding 6500 MWD/MTU. There was no indication of cracking or swelling of the clad and the overall appearance of the rods was excellent.

E. NASA-PLUM BROOK REACTOR-HIGH POWER-HIGH BURNUP-IRRADIATION PROGRAM

UO_2 fuel capsules are being irradiated in the NASA - Plum Brook Reactor as part of the High-Power, High-Burnup Irradiation Program.(7) Fuel pins containing 0.300 inch diameter pellets 96% dense with a 6 inch fuel column are clad in Type 304 stainless steel. The capsules are being irradiated at mean linear heat ratings of 20 to 47 kw/ft, to a maximum burnup of 80,000 MWD/MTU. Four capsules have been irradiated to 10,000 MWD/MTU at a peak power rating of 49.6 kw/ft. Three of these capsules experienced no clad deformation even though they were exposed for a long time at a very high power level causing large fragmentation in some fuel pellets and center cavities with a diameter as large as 20% of the fuel diameter in others. All of the fuel pellets had experienced center melting.

In the fourth capsule, over 75% of the cross sectional area of the pellets melted due to exposure at extremely high power levels. Part of the clad melted (possible because molten uranium was momentarily in contact with the cladding). The center of the pellets shifted about 13% of the fuel radium toward the molten clad zone and an internal cavity, whose diameter was about 45% of the fuel diameter was formed. However, no expulsion of uranium into the coolant or excessive clad deformation occurred.

Three other capsules were irradiated in the Plum Brook Reactor in a program designed to measure the thermal conductivity of UO_2 at temperatures up to 2300°C.(8) The UO_2 fuel columns were 4-1/2 inches long and 1-1/4 inches in diameter. They were successfully irradiated at rod powers of 22-25 kw/ft.

F. FUEL PIN IRRADIATION IN THE GETR

Four vibratory compacted and two pelleted fuel pins were successfully irradiated in the GETR at peak rod power of 21 kw/ft.(9) The pins were 5.2 inches long, had an active fuel diameter of 0.56 inches and were 304 stainless steel cladded. The pelleted UO_2 was 88.3% dense while the vibratory compacted UO_2 was from 80.4 to 86.7% of theoretical density. Residual diametral expansion was observed in both pellet rods, which had extremely small diametral gaps of 0.001 inch.

However, no diametral increase was observed in the vibratory compacted rods, operating at comparable power ratings. The difference is attributed to the ability of the vibratory compacted fuel to utilize internal volume space to compensate for thermal expansion.

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