COMMONWEALTH EDISON COMPANY

GENERAL ELECTRIC COMPANY

GEAP-3053

DOCKET NO. 50-10

MASTER

AMENDMENT NO. 2 TO PRELIMINARY HAZARDS SUMMARY REPORT FOR THE DRESDEN NUCLEAR POWER STATION

By: Commonwealth Edison Company

General Electric Company

Submitted in accordance with provisions of Construction Permit No. CPPR-2

August 25, 1958

RANDOLPH 6-1200

Commonwealth Edison Company

72 WEST ADAMS STREET • CHICAGO 90 ILLINOIS

August 25, 1958

United States Atomic Energy Commission
Division of Licensing and Regulation
Washington 25, D. C.

Attention: Mr. Harold L. Price, Director

Gentlemen:

Commonwealth Edison Company for the purpose of supplementing the license application for the Dresden Nuclear Power Station contained in its proposal addressed to you under date of March 31, 1955, and to comply with paragraph 4, page 2 of the Construction Permit No. CPPR-2, submits herewith Amendment No. 2 to the portion of the Final Hazards Summary Report entitled Preliminary Hazards Summary Report.

Amendment No. 2, dated August 25, 1958, includes revisions to certain portions of Amendment No. 1 of the Preliminary Hazards Summary Report reflecting refinements in reactor physics data and the analysis of certain hypothetical reactor accidents, plus additional information on "start-up and coolant loss accidents" and on one of the fuel model heat transfer functions.

Respectfully submitted,

COMMONWEALTH EDISON COMPANY

By Murray Joslin
Murray Joslin
Vice President

Submitted and sworn to before me this 25th day of Aug., 1958 by said Murray Joslin.

George Barch
Notary Public
Cook County, Illinois

My commission expires:

Feb. 18, 1960

GEAP-3053

AMENDMENT NO. 2

To

PRELIMINARY

HAZARDS SUMMARY REPORT

For The

DRESDEN NUCLEAR POWER STATION

Submitted by: D P Ebright

D. P. Ebright
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G. Sege
Manager -- Safeguard Evaluations
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August 22, 1958

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ATOMIC POWER EQUIPMENT DEPARTMENT
GENERAL ELECTRIC COMPANY

CONTENTS

INTRODUCTION

This report is the second amendment to the Preliminary Hazards Summary Report for the Dresden Nuclear Power Station (GEAP-1044) submitted to the United States Atomic Energy Commission on September 3, 1957.

The present report consists of two parts. Part one revises certain portions of Amendment No. 1 (GEAP-3009) of the Preliminary Hazards Summary Report to reflect refinements in reactor physics data and the analysis of certain hypothetical reactor accidents.

Part two supplies the additional information requested by the Commission on July 25, 1958* with respect to the "start-up and coolant loss accidents", and one of the fuel model heat transfer functions presented in Amendment No. 1.

It is requested that the Commission consider this additional information in connection with its evaluation of the Preliminary Hazards Summary Report.

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* Reference: Meeting in Washington, D.C. on July 25, 1958, between Mr. H. L. Price, Director, Division of Licensing and Regulation, United States Atomic Energy Commission; members of his staff; and representatives of the Commonwealth Edison Company and the General Electric Company.

PART ONE

REVISIONS TO AMENDMENT NO. 1 OF THE PRELIMINARY

HAZARDS SUMMARY REPORT

REVISION NO. 1: NUCLEAR CHARACTERISTICS

1. Delete the entry "-0.8×10^{-5} (Δk/k)/°F of fuel temperature" appearing at the bottom of the right hand column of the table on page 6 and insert the following in its place:

   "-1.2×10^{-5} (Δk/k)/°F of fuel temperature"

2. On page 10 delete the last four entries in the left hand column beginning with the entry "Average strength per rod" and their corresponding entries in the right hand column beginning with the entry "0.0022 Δk"

   and insert the following in their respective places:

a.	"Average strength per rod"	"0.00195"
b.	"Maximum strength per rod"	"0.034 Δk"*
c.	"Maximum reactivity addition rate from withdrawal of a single control rod"	"0.0029 Δk/sec"*
d.	"Total worth of control rods"	"0.156 Δk"
e.	"End connector poison"	"0.02 Δk"
f.	"Total worth of control"	"0.176 Δk"

3. On page 33 delete the entry "-0.8×10^{-5}" appearing under each of the column headings "Cold", "Hot", and "Full Power" opposite the entry "(Δ k/k)/°F fuel" and insert the following in their respective places.

	Cold	Hot	Full Power
Δ k/k/°F fuel	"-0.9×10^{-5}"	"-1.2×10^{-5}"	"-1.4×10^{-5}"

*These new values are the result of recent investigations which have indicated that the worth of a particular rod is greatly affected by the position of adjacent rods.

REVISION NO. 2: SAFETY ANALYSIS

1. Delete items "a" through "e" in the answer to question E-1, on pages 77 to 78, in their entirety, and insert the following text in their place:

"a. The maximum rate of reactivity insertion that could occur as a result of the "cold water accident" (the inadvertent starting of one coolant-loop) in the Dresden plant is 0.004 Δ k/k per second. The maximum transient reactivity insertion in such an incident would be about 0.037 Δk.

These maximum reactivity values could be developed only when the reactor is operating at about 10% of full power and system pressure is about 150 psig. If the power were significantly lower, than 10% of rated at the time of the incident, there would be no voids in the core. Thus, the transient in this case would be much smaller since the major source of reactivity inserted in such accidents is contained in the voids. If the power of the reactor were significantly higher, the pressure of the system would have to be higher because of the limited flow capacity of the steam line valves. At these higher pressures, the reactivity contained per unit void volume decreases. Thus, again, the transient would be much smaller. Though the magnitude of the transient would be smaller, the maximum power level and fuel temperatures reached could be higher. In the event the reactor were operating at the maximum power possible at the time of such an incident (about 85% of rated is the maximum power that can be attained with only three recirculation pumps in service) the power would be momentarily increased to about 110% of rated. This would shortly settle out at about rated power. The maximum fuel temperature reached would be about 3900°F.

The arrangement of the piping in the recirculation system is an important factor in limiting the extent of this accident. Because the tie line (line number 0137 on Figure D-1 in reference (1)) between the two downcomer headers is always open to both headers and because there is constant suction on this line to supply water to the reactor clean-up demineralizers, there will be continual flow of hot water through the downcomers on the otherwise isolated and "cold" header.

The only water that can become "cold" is that in the ends of the isolated header and in the lines from there to the reactor via the recirculation pumps and secondary steam generators. Since it is not credible that both of the two recirculation pumps on the "cold" header could be started accidentally, i.e., through operator error, at the same time only one-half of the "cold" water in these lines was considered as contributing to this accident. Thus, under conditions for this problem, only about 20,000 pounds of water at a temperature of 100°F was considered available for the "cold water accident".

b. The minimum period resulting from this accident would be about 27 seconds.

c. The excursion in this incident would be self-limiting in that it would be terminated by void formation and depletion of the source of reactivity (cold water) well before the neutron flux reached a high enough level to require a scram. The peak flux reached would be about 65 percent of rated; thus, the operability of the scram protection devices is of little concern in this incident.

d. The maximum power level reached in this accident would be about 400 MW. The integrated power would be about 17,000 MW-sec.

e. The maximum "hot-spot" temperature reached in the fuel would be about 2000°F, thus no melting would occur."

2. Insert the following after the sentence in entry "g-iii" on page 79:

"(It takes five minutes for the discharge valve to open fully.)"

3. Delete item "h" in answer to question E-1 on page 80 in its entirety and insert the following in its place:

"h. As indicated by the previous discussion, this accident would not harm the reactor or even the fuel; thus, there can be no adverse effects to the public. If two pumps and their associated "cold water" were to be started simultaneously, an incredible occurrence, and the high neutron flux scram were to fail to operate, the peak power reached (starting with reactor power at 10% of rated) would not exceed about 1400 MW and the integrated power would be about 1.6×10^5 MW-sec. The maximum "hot spot" fuel temperature would be about 5300°F. Though some melting might occur in the center of a few fuel elements, it would not cause the cladding of these elements to fail."

4. Delete:

a. The entry "40,000 MW-sec." in the second paragraph of item "c" on page 82.

b. The entry "4000°F" under item "d" on page 82,

and insert the following in their respective places:

a. "2400 MW-sec."

b. "1000°F"

5. Insert the following after the second paragraph in the answer to question E-4 on page 84:

"It is pertinent to note, with respect to the temperature drops in both the fuel and clad as shown on Figure E4-1, that the flux decreases very rapidly (within one second) at the start of the transient to a value of about 10% or less of rated for this case. It then recovers gradually within about four seconds to level off at a value of about 55% of rated."

PART TWO

INFORMATION REQUESTED BY THE A. E. C.

A. THE START-UP ACCIDENT

1. Sequential Removal of Control Rods at Start-Up

The "start-up accident" discussed in previous submittals (1, 2) was assumed to occur as a result of the continuous and sequential removal at random of control rods of average reactivity worth at the fastest design rate possible, starting with the reactor at source level. The analysis also assumed coincident failure of the principal device (period scram circuitry) to control such an incident, and it conservatively ignored the mitigating negative fuel-temperature (Doppler) effect. As presented in reference (2), the reactor would be safely shut down by the high neutron flux scram circuitry in such an accident and there would be no fuel melting or adverse effects to the reactor enclosure or the public.

2. Group Removal of Control Rods at Start-Up

The "start-up accident" discussed below is based on recent digital computer studies of the effect of removing control rods in local groups rather than at random. This is the worst start-up accident that is credible for this plant. The principal conservative assumptions made for this accident analysis include:

a. The withdrawal of the outer four rods of a cruciform 5-rod grouping so as to bring the reactor slightly subcritical.

b. The subsequent withdrawal of the fifth (center) rod at its maximum design rate.

c. A non-boiling clad-to-water heat transfer coefficient of 170 Btu/(ft^2)(hr)(°F). (Cf. the boiling heat transfer coefficient of 10,000 Btu/(ft^2)(hr)(°F)).

d. Coincident failure of the period scram circuitry.

e. Initiation of ramp reactivity insertion when reactor power was at 10^{-12} times rated power (equivalent to the spontaneous fission source level).

f. A moderator temperature of approximately 68°F.

g. All new fuel.

The reactivity insertion rate under the above conditions is calculated to be 0.0029 Δ k/k per second. On the basis of an experimentally determined 0.22 second flux scram delay time* and a calculated negative fuel-temperature coefficient of 0.9×10^{-5} $\Delta k/(k)(°F)$**, this accident would result in a localized flux peaking and possibly some melting in the center of a few of the fuel elements. This would not cause the cladding of these fuel elements to fail.

The flux peaking in this area is calculated to be restricted to a small area of the core, about two feet in diameter and six feet long. The temperature rises in the fuel in this area are calculated to be about as follows:

a. Overall average 900°F

b. Center average 1300°F

c. Center maximum 3800°F (for fuel of equilibrium exposure)

d. Center maximum 4700°F (for new fuel)

A plot of the flux, temperature rises and the sequence of events in this incident are shown in Figure 1.

In the event the flux scram delay time were as long as the specified design time of 0.50 seconds rather than the 0.22 seconds determined experimentally, the average center temperature would be increased by only about 200°F. The maximum "hot-spot" temperature rise would be about 4400°F for "equilibrium" fuel and 5000°F for new fuel. However, again it is not expected that even this temperature would cause any release of fission products to the coolant.

* Reference Figure C 3-1 in Reference (2).

** Reference data reported by Creutz, E. et. al., in JOURNAL OF APPLIED PHYSICS, Vol. 26, No. 3, March, 1955. (Calculations are also confirmed by reference to Reports by Pearch, R. M., and Walker, D. H., NUCLEAR SCIENCE AND ENGINEERING, Vol. 2, No. 1, February, 1957, and by Davies, M. V., JOURNAL OF APPLIED PHYSICS, Vol. 28, No. 2, February, 1957.)

In conclusion, it should be noted that in addition to the safeguard device (period scram circuitry) provided to avoid the "start-up accident" there are strong procedural controls to minimize the possibility of such an accident. That is, procedurally: (a) control rods will not be withdrawn in the configuration assumed for this incident, (b) a rod will not be continuously withdrawn at the maximum design rate possible, (c) the reactor will be brought up to critical in a shallow area near the top of the core rather than in the narrow and deep area necessary for the accident described above. Thus, it is not expected that the "start-up accident" will ever occur, but if it did, it is concluded that it would not cause any adverse effects to the reactor enclosure or the public.

From a safeguards standpoint, it is pertinent to note that even if both the period and flux level scram circuitry were to fail coincident with the above start-up accident (an incredible sequence), and there was no manual scram, the accident would be terminated by self-destruction of a portion of the core without any adverse effects to the public. It is calculated that about two and one-half percent of the core would be melted, but no significant amount of the fission products should be released from the primary system. The pressure rise in the reactor in this case (even in the unlikely event a zirconium water reaction were also to be initiated) would be very nominal, probably less than 60 psi. Figure 2 shows a plot of average fuel center temperature rises in such an incident. The times at which significant changes in state may occur are also shown.

FIGURE I

5-ROD START-UP ACCIDENT

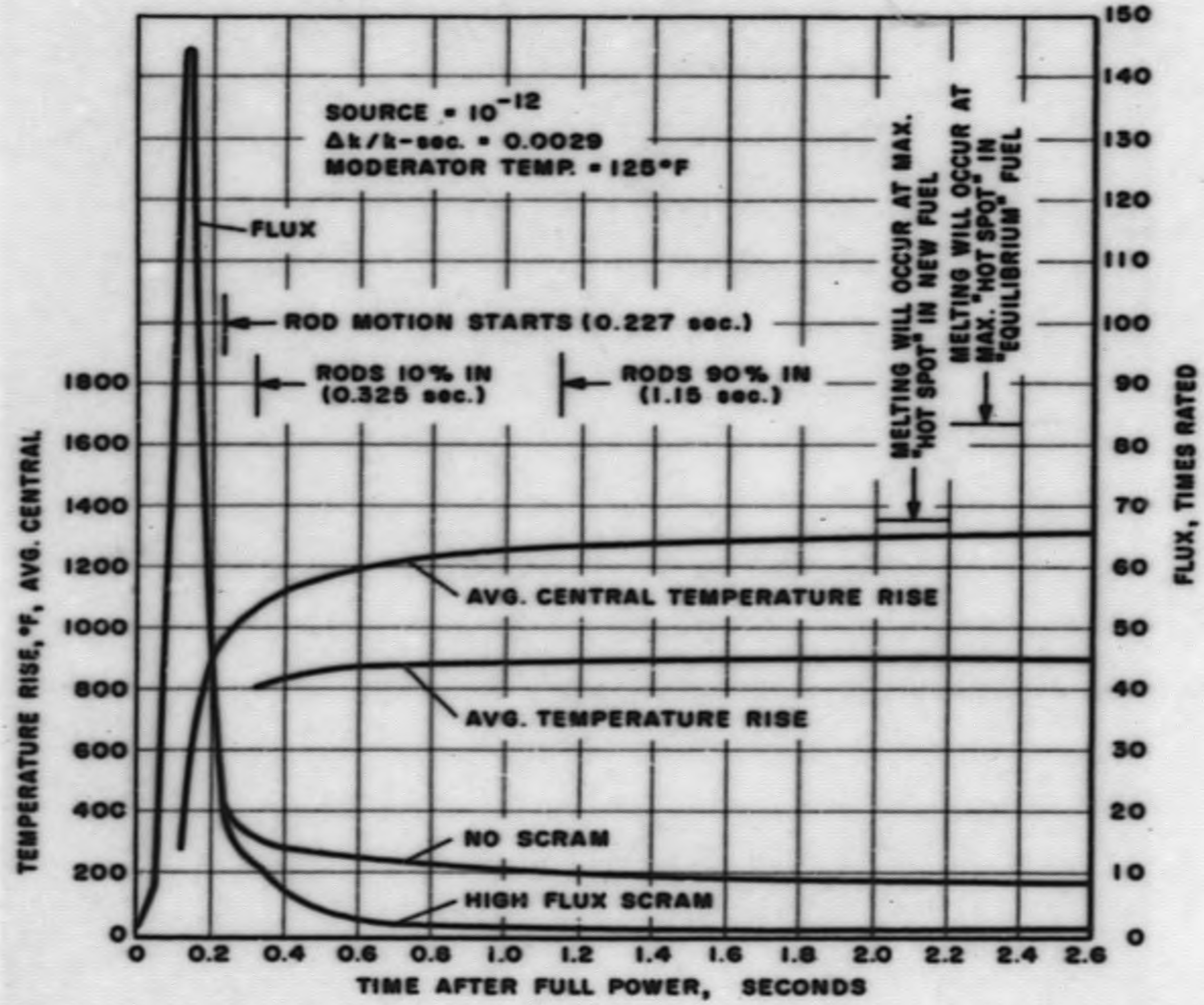

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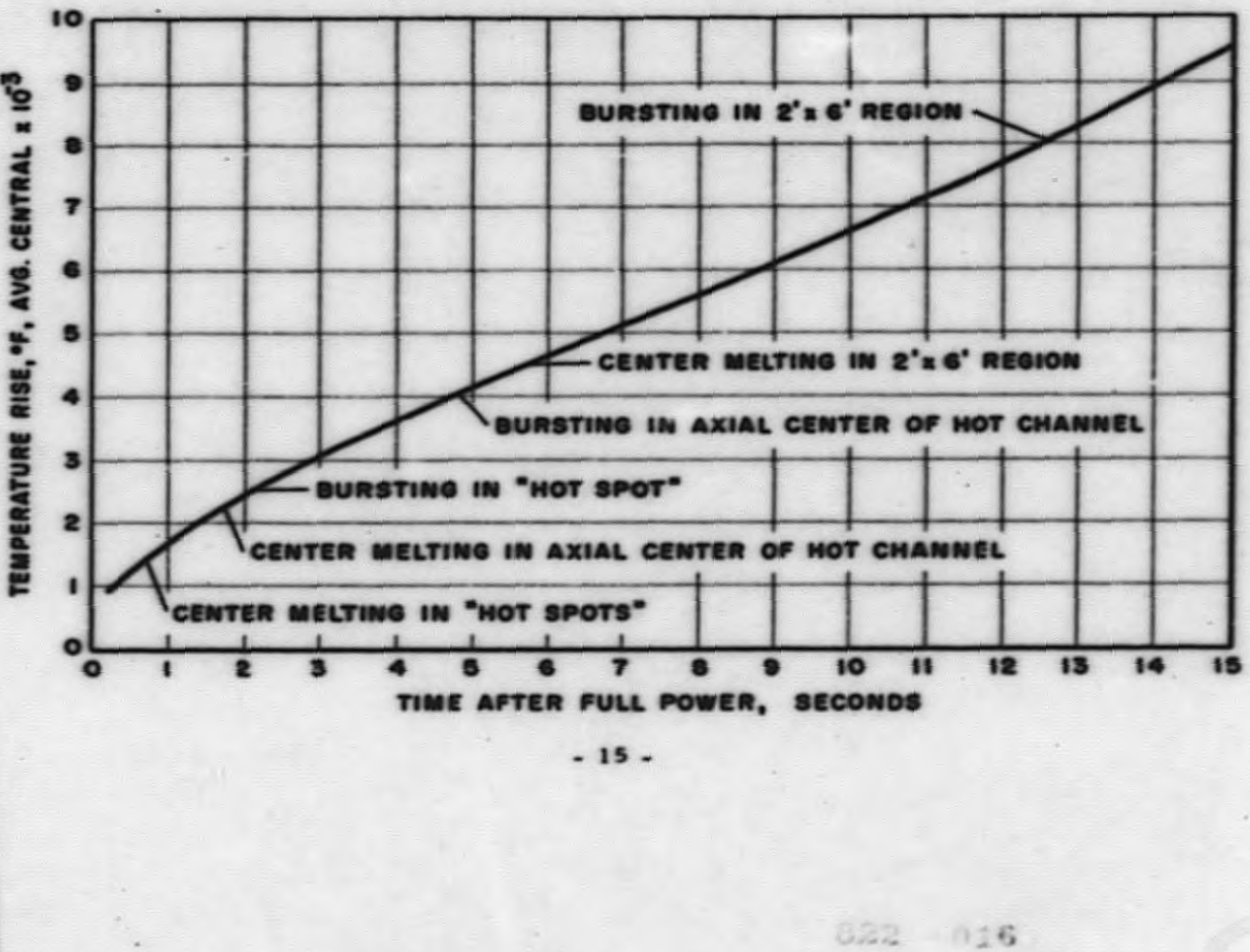
FIGURE 2

5-ROD START-UP ACCIDENT
WITHOUT SCRAM



B. COOLANT LOSS ACCIDENT

In the event of a large break in one of the bottom inlet lines to the reactor, which is the most severe of any credible coolant loss accidents, the principal sequence of events that would follow, assuming the reactor had been operating at or near rated power, are as follows:

a. The reactor would be shut down nearly simultaneously with the line rupture as a result of increased voids, or as a result of scram from low water in the steam drum and/or high pressure in the enclosure. (All three of these effects would be present, but any one of them would suffice to cause shutdown.)

b. Nearly all the water in the vessel would be ejected within one minute.

c. Melting of the zirconium cladding would progress about as follows:

   i. Initial melting would begin about 10 minutes after the rupture.

   ii. About one-third of the cladding would melt in the first hour.

   iii. All of the cladding would have melted in about 24 hours.

d. A zirconium water reaction might be initiated about 10 minutes after the line rupture between the molten zirconium and any water that might be present.

e. Melting of the UO_2 fuel would occur about as follows:

   i. Initial melting would begin about 16 minutes after the rupture.

   ii. About one-fifth of the fuel would melt in the first hour.

   iii. All of the fuel would have melted in about 40 hours.

f. Volatilization of fission products would commence 10-15 minutes after the line rupture.

Events "c" and "e" above have been calculated on the conservative assumption that essentially no heat is transferred to or from the fuel element or cladding following the line rupture. In this respect it should be noted that the principal mechanism by which heat would be carried away from the fuel elements upon loss of coolant would be by radiation. Each fuel assembly, however, is encased in a zirconium channel which would probably be an excellent reflector. Moreover, only the outer ring of channels could radiate to anything but the core itself. Accordingly, it is assumed, with respect to the initial melting of the fuel, that adiabatic conditions exist within the core following loss of coolant. However, it is expected that there would be subsequently a large heat loss by radiation from the molten pool of UO_2 that may form in the bottom of the reactor. For this reason vaporization of any significant portion of the fuel seems highly unlikely. Assuming that most of the fuel has melted, about 70 percent of it could be held in the bottom of the vessel below the lowest point of the inlet nozzles. In such a case, the heat loss by radiation from only the top of the "puddle" of fuel to the surroundings (conservatively assumed to be at 2000°F) would be ten times larger than the heat generation. Thus, it is not expected that any significant fraction of the fuel would be vaporized as a result of this accident. It is also not expected that there would be any rapid reaction between the molten UO_2 and water or air since the heat release from either reaction is small.

With respect to event "d" above, it is likely that the zirconium-water reaction would not take place, to a very substantial extent, under the conditions of this incident because:

a. The molten zirconium should be in a relatively large sized droplet upon contact with the water since there would be no mechanism to cause it to be finally dispersed. For this reason the percentage of zirconium entering into the reaction should be small if there is any reaction at all.

b. By the time the zirconium begins to melt, there should be but little water available within the reactor. This condition should at least limit the extent of the reaction.

However, even if the zirconium-water reaction did take place, it would not significantly affect the controlling parameters of the accident. By the time the zirconium-water reaction could be initiated (about ten minutes after rupture) the enclosure pressure would have been reduced to about one-half its initial peak by heat transfer to the enclosure walls and internal structures. The pressure contribution as a result of the energy released from the zirconium-water reaction would be negligible since only a fraction of a psi could be developed within the enclosure from this source at any given instant. Even the energy released from a reaction of 25 percent of the zirconium cladding would only increase the enclosure pressure by a few psi ($\sim$10). Even this would not raise the pressure in the enclosure above the initial peak value.

With respect to the release of fission products, event "f" above, it is estimated that because of the high volatility of the noble gases and the halogens that they could be nearly 100% released by the end of the first 24 hours. The balance of the fission products, because of their much lower volatility, might be released to the extent of only about 10 percent in the first 24 hours. (Cesium and possibly tellurium, because of their intermediate volatility might be released to the extent of up to about 50 percent during this time.)

It is important to note that the fission product release figures given above should be considered as only an order of magnitude of the probable release because of the many uncertainties affecting such a release. However, even though these release values are only approximate, it is apparent, upon comparing them with the fission product dispersion assumed in the Preliminary Hazards Summary Report (1), that the radiological consequences of this accident would be a number of times less severe than those reported there. For example, in the Preliminary Hazards Summary Report, the entire fission product dispersion considered was assumed to occur coincident with the break in the primary coolant system, whereas in the coolant loss accident considered in this report the dispersion of fission products would occur slowly over the first day or so following the accident. In addition, it was assumed in the Preliminary Hazards Summary Report accident that all of the fission products released were initially completely dispersed to the enclosure atmosphere. In the coolant loss accident in this report, however, the fission products can only reach the "visible" enclosure atmosphere by a tortuous path through the broken pipe line, underground rooms, and passageways. As a result, many of the fission products, particularly those of lower volatility, would never reach a "visible" area as they will be deposited on the walls, floors and equipment surfaces by natural condensation or by action of the post-accident spray system.

As shown in the previous discussion of conceivable accidents, none involve a credible destructive nuclear excursion. In the absence of such an excursion, there is no credible incident that can lead to the overheating and fine dispersion of the zirconium cladding simultaneously with a primary system break. Thus, a simple coolant loss accident is the "maximum credible accident" for this plant. In this respect, it should be noted that the enclosure pressure and the radiological consequences resulting from the "worst reasonable accident" described in a previous submittal (1) and used as the basis for establishing the enclosure design are more severe than would be associated with the "maximum credible accident". Thus, the enclosure design is adequate to assure its integrity and adequate post-accident safety in the event of the "maximum credible accident".

To place this accident in proper perspective, it is important to note that a large coolant loss accident to the Dresden reactor is not likely because of the care in design, construction, and inspection of the plant.

C. FUEL MODEL HEAT TRANSFER DATA

A previous submittal (2) presented the fuel model as:

$$Q_{\substack{per\\unit}} = \Phi_{\substack{per\\unit}} \left[\frac{0.756}{1 + 16.8p} + \frac{0.136}{1 + 3.18p} + \frac{0.0624}{1 + 1.05p} + \frac{0.0156}{1 + 0.37p} + 0.03 \right]$$

In explanation of this model, it should be noted that the first four terms in the model are the first four terms of a rapidly converging infinite series derived for a cylindrical fuel element using Hankel Transforms. The assumptions involved in this derivation were the following:

a. No heat storage in the cladding but an equivalent heat transfer coefficient for the cladding of 3200 Btu/(hr)(°F)(ft^2).

b. A constant heat transfer coefficient between cladding and water of 10,000 Btu/(hr)(°F)(ft^2) for a boiling condition and 170 Btu/(hr)(°F)(ft^2) for non-boiling conditions.

c. A heat transfer coefficient of fuel to cladding of 1,000 Btu/(hr)(°F)(ft^2).

d. Due principally to gamma heating, 3% of the generated heat was assumed to be transferred to the water immediately, that is with no time delay.

e. A thermal conductivity in the fuel of 1.15 Btu/(hr)(ft^2)(°F/ft).

f. An average weighted specific heat for the fuel of 0.073 Btu/(lb °F).

g. A radial flux distribution within a fuel element of I_o (1.2r)

where:

I_o = Bessel function defining the power distribution

r = radius of fuel element.

REFERENCES

1. GEAP-1044 "Preliminary Hazards Summary Report for the Dresden Nuclear Power Station", by G. Sege, June 24, 1957.

2. GEAP-3009 "Amendment No. 1 to Preliminary Hazards Summary Report for the Dresden Nuclear Power Station", by D. P. Ebright, May 1, 1958.

END