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LOSS OF FORCED COOLING FLOW IN THE PLUM BROOK REACTOR

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16. Abstract

The Plum Brook Reactor (PBR) experienced a temporary loss of forced-cooling flow in November, 1966. To provide a framework for understanding the occurrence, a brief description is given of the PBR, its electrical distribution system, and methods for maintaining flow. Then the occurrence is treated in three parts: causes; inspections and analyses conducted to evaluate the situation; and corrective action taken to avoid a recurrence. It is concluded in this case that an undesired rather than an unsafe condition existed.

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LOSS OF FORCED COOLING FLOW IN THE PLUM BROOK REACTOR*

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SUMMARY

Prevention of the "loss of flow" and "loss of coolant" accidents continues to have major technical and economic influence on nuclear power plant design and operation. Realistic assessment of the mechanisms and consequences of the accidents is being given much attention (e.g., the AEC LOFT facilities and analyses). The Plum Brook Reactor (PBR) experienced a temporary loss of forced-cooling flow. The analyses conducted and conclusions reached after conditions previously analyzed were exceeded may have application to other plants.

On November 22, 1966, following about 7 days of operation at full power of 60 megawatts (th), the Plum Brook Reactor experienced a temporary loss of forced-cooling flow, initiated by interruption of dc control power to the primary main and shutdown coolant pump breakers. The control power breaker was accidentally opened. An automatic pump interlock scram occurred within 1 second after the breaker was opened. Previous hydraulic testing demonstrated that forced flow from coastdown persisted for at least 30 seconds. Forced-cooling flow was restored within an additional 45 seconds.

The paper first gives a brief background description of the Plum Brook Reactor to provide a framework for understanding the occurrence. Then the causes of the occurrence are presented, followed by a description of the inspections, analyses, and evaluations conducted. The presence or extent of any damage was assessed, and safety for restart was assured. Other reviews and evaluations were conducted to determine the proper follow-up corrective action. The corrective actions taken to prevent a recurrence are summarized.

It is concluded in this case that an undesired rather than an unsafe condition existed. The investigation and actions taken have avoided recurrence to date, and it is believed are sufficient to avoid recurrence in the future.

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*Presented at ANS Topical Meeting on Reactor Operating Experience, San Juan, Puerto Rico, Oct. 1-3, 1969.

INTRODUCTION

Considerable attention is properly being given to a realistic assessment of the mechanisms and consequences of the "loss of flow" accident in nuclear power plants. Lack of complete information leads to designs with rather elaborate and redundant precautions against loss of flow, and conservative protection against the worst conceivable consequences. Thus, the "loss of flow" accident has considerable practical and economic impact on the design and operation of the nuclear power plant.

The Plum Brook Reactor (PBR) experienced a temporary loss of forced-cooling flow, and conditions which exceeded those previously analyzed were reached. To provide a framework for understanding the occurrence, we start with a brief description of the PBR, its electrical distribution system, and methods for maintaining flow. The paper then treats the occurrence in three parts, (1) causes; (2) inspections, analyses, and evaluations to assess the presence or extent of any damage and safety for restart, and to determine the proper corrective action to be taken; and (3) the corrective actions taken.

BACKGROUND DESCRIPTION

Reactor Core and Reactor Tank Assembly

The Plum Brook Reactor (PBR) is a 60 megawatt (th) pressurized water test reactor. Figure 1 illustrates the layout of the reactor core and instrument and experiment facilities. Within the core box can be seen the 3×9 array of fuel elements, surrounded on three sides by a single row of beryllium "L-piece" reflectors, and on the fourth side by four rows of beryllium "R-piece" reflectors. Normal cooling water flow has two upward paths, through the R reflector and past the experiment facilities; and all water passes downward through the L reflector and fuel elements. The fuel elements are of the MTR curved plate type, with 20 mils of aluminum cladding on a uranium-aluminum alloy. At the hottest spot in the fuel the nominal heat flux is 1.4×10^6 Btu per hour per square foot with a corresponding fuel element temperature of 325° F (less than saturation temperature). The steady-state departure from nucleate boiling ratio (DNBR) is greater than 2, and the transient DNBR is greater than 1.3.

Figure 2 shows the reactor core located in the reactor tank. This provides a perspective view of the reactor, and the instrument and experiment facilities. The direction of normal flow can also be seen, upward past the experiment thimbles and through the R reflector, and downward through the L reflector and fuel elements.

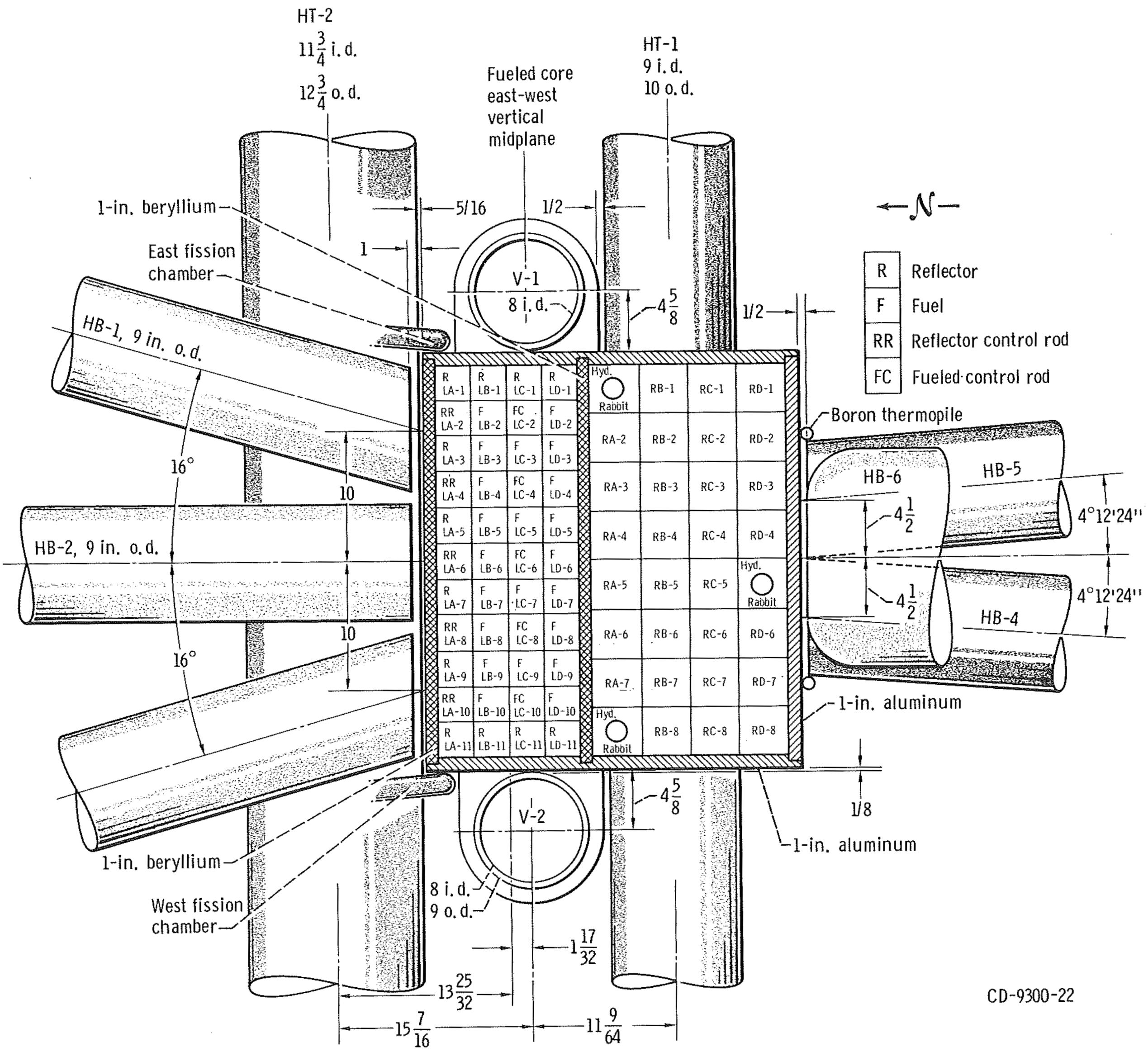

Figure 1. - Horizontal section of reactor core. (Dimensions are in inches.)

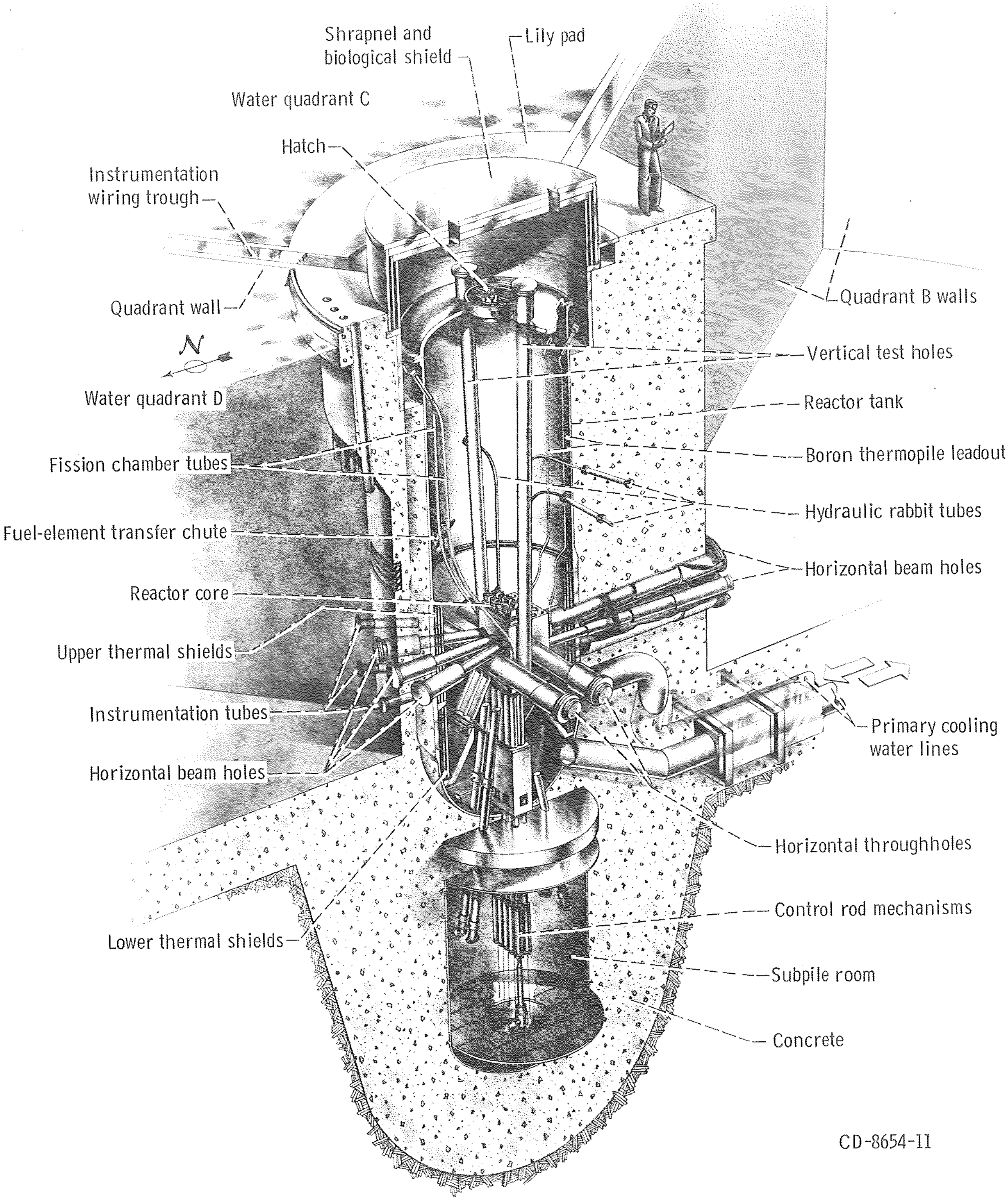

Figure 2. - Reactor tank assembly.

Electrical Distribution System

Figure 3 is a simplified schematic of the portion of the electrical distribution system that concerns reactor coolant flow, at the time of the temporary loss of forced-cooling flow. (Some changes that were unrelated to the occurrence have subsequently been made.) Commercial power comes in to two buses through the "North Line" and the "South Line." These two buses are normally separated, but automatically tie together on loss of either the North or the South Line. The Emergency Power bus is normally fed through one of two breakers, from either the North or the South Commercial Power bus. A key interlock prevents both breakers from being closed simultaneously and tying the North and South Lines. Diesel generators run partly loaded on the Emergency Power bus in parallel with commercial power. On loss of commercial power, the Emergency Power bus is automatically separated from the Commercial Power bus, and the Diesels power the necessary loads. The Guaranteed Power bus is a dc bus. It is normally fed through inverters-diverters from the Emergency Power bus. Additionally, however, batteries power this bus as a backup for the unlikely loss of emergency power.

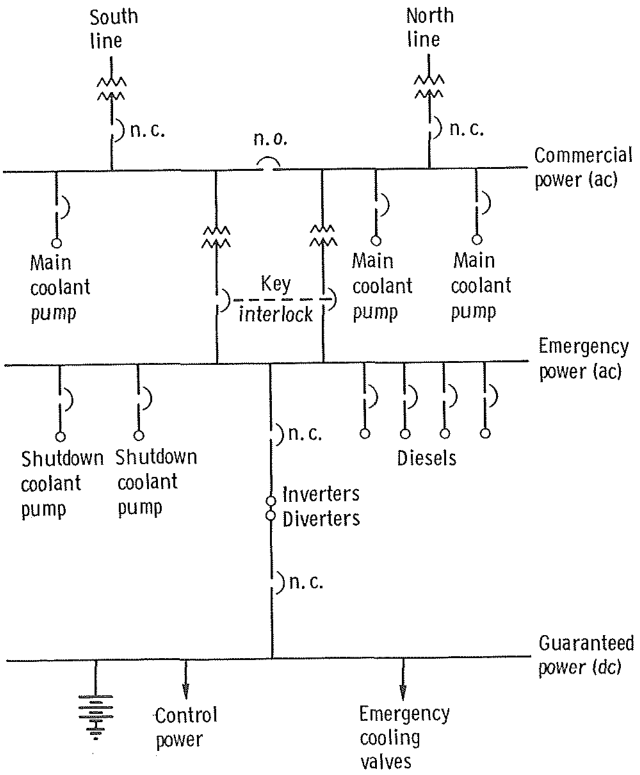

Figure 3. - Simplified electrical distribution system. (n. c., normally closed; n. o., normally open.)

Flow Systems

The PBR Primary Cooling Water System has two flow loops, a main and a shutdown. The primary main cooling water loop has three pumps, with two sufficient to provide full flow of 17 000 gpm for full reactor power. Two main coolant pumps are powered from the North Commercial Power bus, and one from the South Commercial Power bus. A normal operating mode is to run two pumps, one from each bus. The third pump is in standby and automatically starts if either of the running pumps is lost. The primary shutdown cooling water loop has two pumps, with either capable of providing sufficient flow (>1000 gpm) for cooling after a scram from prolonged full power operation. The shutdown coolant pumps are powered from the Emergency Power bus. One pump is run whenever the reactor is operating, and the other pump automatically starts if the first pump is lost. Control power for both the main and the shutdown coolant pumps comes from the dc Guaranteed Power bus.

A third source of flow, emergency flow, is available for the extremely unlikely loss of both main and shutdown cooling flow. This system provides water by gravity feed from an overhead storage tank through the reflector and core and out a drain to hot retention tanks. This system is activated by three buttons in the control room which actuate motor operated valves powered from the dc Guaranteed Power bus.

As mentioned above, the normal reactor operating mode is with two main coolant pumps and one shutdown coolant pump running. The reactor has the customary scrams from low flow and core ΔP. Additionally, anticipatory protection is provided for the reactor for off-normal flow conditions. If one main coolant pump is temporarily lost for any reason, all the control rods are automatically inserted at a fast rate. If both main coolant pumps are lost, a pump interlock reactor scram is actuated in just less than 1 second. The fast rod insertion and the 1 second delay of the scram avoid unnecessary reactor shutdowns for transients or conditions that can be quickly corrected by automatic switching (e. g., tie breaker closure or start of a standby pump). Thus, if for any reason all main cooling flow is lost, the reactor scrams; and shutdown cooling flow is provided by a shutdown coolant pump. In the unlikely event shutdown cooling flow is also lost, the reactor operator institutes emergency flow by pressing the three buttons in the control room. Since previous hydraulic testing demonstrated that forced flow from coastdown persists for at least 30 seconds, the operator has 30 seconds after a scram to verify the presence of flow or to actuate emergency cooling.

LOSS OF FORCED COOLING FLOW

On November 22, 1966, following about 7 days of operation at full power of 60 megawatts (th), the Plum Brook Reactor experienced a temporary loss of forced cooling flow, initiated by interruption of dc control power to the primary main and shutdown coolant pump breakers. The control power breaker was accidentally opened. An automatic pump interlock scram occurred within 1 second after the breaker was opened. Previous hydraulic testing demonstrated that forced flow from coastdown persisted for at least 30 seconds. Forced cooling flow was restored within an additional 45 seconds.

Causes

This potentially serious occurrence followed the classic pattern, where the consequences of an improbable event are compounded by multiple additional errors. The occurrence was caused by a combination of design errors and operator errors.

There were three design errors. (1) Too many vital control power circuits on one breaker. Dc control power to all of the main and shutdown coolant pump breakers originated from one breaker on the Guaranteed Power bus. (2) Improper breaker action on loss and restoration of breaker control power. Although the control power breaker was reclosed within a few seconds after its accidental opening, restoration of the control power did not restart any pumps. The reactor operator had to restart pumps by pressing buttons in the control room. (3) Inadequate physical protection for the breaker handles on some vital breakers. This single vital control power breaker handle had no physical protection covering it to prevent inadvertent operation.

Three operator errors were evidenced. (1) Carelessness by the electrical operator who inadvertently tripped the breaker. (2) Failure of the electrical operator to report the condition to the reactor control room, even though he immediately reclosed the breaker. (3) Failure of the reactor operators to recognize immediately that a loss of flow condition had occurred, and therefore failure to take immediate proper emergency action. It should be noted that although operator error contributed to the occurrence, it was good operator performance that restored conditions to normal and prevented the occurrence from lasting any longer than it did.

Inspections, Analyses, and Evaluations

The inspections, analyses, and evaluations of the occurrence can be divided into two phases, (1) the immediate, necessary to assess the presence or extent of any damage, and the safety for restart, and (2) the subsequent or follow-up, to determine all necessary corrective action to prevent recurrence and any other important corollary changes or actions suggested by the occurrence.

The first phase of the evaluation included the following. It was first determined that there was no increase in the primary cooling water fission product activity following the occurrence. To confirm that no damage had occurred, we performed significant analyses and inspections to systematically consider all of the reactor core components, and instruments and experiments (see figs. 1 and 2). The exact model for convective core cooling was not known. Therefore, we calculated the minimum departure from nucleate boiling ratio (DNBR) assuming free-convective flow. This analysis yielded a DNB ratio of at least 2. Four fuel elements located in the central region of the core in locations of the highest heat flux were visually inspected in the reactor tank. There was no evidence of warpage, blistering, coolant channel blockage, loss of clad integrity, or abnormal discoloration. Subsequently, all 22 standard and five control follower elements were removed without difficulty, visually inspected, and placed in storage. Analysis of the beryllium reflector pieces, assuming no heat removal, established their maximum possible temperature at 600° F. Visual inspection of a beryllium piece showed no damage. The cadmium absorber section of the central fueled shim safety control rod (no. 3) was examined in the Hot Laboratory. There was no warpage, deformation, discoloration, or indication of any abnormality. One of the experiment thimbles (HB-3) contains a collimator. A helium leak check of the collimator and shutter system revealed no leakage. Inspection and/or analysis of a titanium clad tungsten shield in the HB-2 experiment thimble, of an instrument thimble bismuth shield, and of experiments evidenced no damage. Selected reactor material irradiation surveillance specimens were removed for testing. There were no measurable changes in the specimens attributable to the loss of flow occurrence. Our Safeguards Committee concurred that the actions taken were sufficient for restart. Prior to restart with a new core, rod drop tests confirmed normal rod drop times. Rod worth measurement on restart yielded proper values.

Subsequent review and evaluation included the following. All important electrical components were reviewed for optimum separation of control power sources and breakers. The shutdown coolant pump and other important breakers were reviewed for proper breaker action on loss and restoration of control power. All vital breakers were reexamined for physical protection needed to prevent inadvertent tripping (but not to hamper deliberate operation). Operating procedures were reviewed for any necessary changes. The layout of the facility alarm panels was reviewed for any changes which would facilitate the operator's interpretation or recognition of the most important abnormal situations. Emergency actions associated with the relatively few fundamental and vital elements of reactor safety (cooling water flow and pressure, reactor power, control rod position) were reviewed for any possible improvements, with particular emphasis given to methods for achieving almost instinctive operator reaction. The condition of the fuel elements in the core during the occurrence was assessed.

Corrective Actions

As a result of the review and evaluation occasioned by the loss of forced cooling flow occurrence, we took the following actions. The dc control power for the three main and two shutdown coolant pumps was rewired so that each pump breaker and the automatic switching circuitry received control power through separate breakers. We separated the ac feed of the two shutdown coolant pumps so that they were not both fed from a single breaker. We provided for automatic restart of the shutdown coolant pumps when power is restored. The dc control and operator power sources for each of the emergency coolant valves were combined, and each of these valves was supplied from a separate breaker. All vital circuit breakers were provided with physical protection to prevent inadvertent tripping. A single switch was installed on the control room console to activate all of the valves required for emergency cooling. A flow limiting orifice was installed downstream of the reactor tank drain valve to automatically limit the flow of emergency cooling water, avoiding the need to open the drain valve a specified amount. The control room annunciator windows for "primary main flow off-normal" and the "primary shutdown flow off-normal" were outlined in red to make them stand out from the other alarms. We ran several additional drills for each shift team to insure that all personnel recognize and know what to do if emergency cooling should become necessary or other abnormalities occur in the relatively few fundamental elements of reactor safety. We discussed with all reactor operators the circumstances and potential consequences of this occurrence.

Disposition of the 27 fuel elements which comprised the core during the loss of forced-cooling flow was determined. Our analysis of possible damage to these fuel elements was inconclusive. Further, we did not believe that visual and nondestructive examinations or additional analysis would provide more conclusive data. The remaining dollar value of the elements was about $10 000. This worth was not sufficient to justify significant risk in their reuse for two reasons. (1) If one or more of these fuel elements failed in operation, the cost of cleanup and downtime could be more than the potential savings from reusing the elements. (2) Any fission product leakage or gross contamination of the primary cooling water would confuse our surveillance of elements being burned up to higher than previous values. Therefore, because we could not prove them unchanged, we did not reuse them.

CONCLUSIONS

The PBR was designed with redundant protection against loss of flow. Nevertheless, a combination of design and personnel errors resulted in a temporary loss of forced-cooling flow. Previously analyzed conditions were exceeded. The inspections and analyses conduced to evaluate the situation confirmed that an unsafe condition had not been reached. Although much can be learned from a systematic evaluation of this kind of an occurrence, a precise determination of all conditions cannot yet be made. Because of the difficulty and effort required for such an analysis, and the potential for severe damage, even a temporary loss of forced-cooling flow is an undesired condition. The corrective actions which were taken to avoid a recurrence have proven to be effective to date and are believed sufficient for the future.

Lewis Research Center,
National Aeronautics and Space Administration,
Cleveland, Ohio, October 20, 1969,
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