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STUDY OF CORE COOLING
REQUIREMENTS DURING
NS SAVANNAH REFUELING

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ABSTRACT

The requirements for cooling the NS Savannah reactor core during refueling were investigated. The two-part study included analysis of the system required to provide coolant flow to the reactor and to remove the decay heat generated and analysis of flow and temperature characteristics in the reactor during refueling. Since previous investigations of the system requirements had been made, this study was concerned primarily with the flow and temperature analysis.

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1. INTRODUCTION

During normal cooldown of the NS Savannah's reactor plant, the primary pumps remove decay heat from the core, and the primary system is cooled with flow through the primary loop purification system and the buffer seal system. The primary system has a pressure of 125 psig or more to provide the driving force for the flow through the purification system and to maintain the desired conditions for operating the primary pumps.

Refueling of the NS Savannah requires depressurizing the primary system completely, opening the reactor vessel, and removing the upper flow baffle and upper grid plate. As a consequence, an alternate method of cooling the core during this time must be devised. The flow and temperature characteristics within the reactor during various phases of the refueling must be studied to determine whether the auxiliary cooling method prevents boiling in the core.

2. REACTOR FLOW AND TEMPERATURE ANALYSIS

2.1. Cases Studied

The refueling procedure was reviewed in detail to determine the various cases to be used to analyze the flow and temperature characteristics in the reactor. From this review, the following cases were studied:

Case I. The primary system was depressurized to below 50 psig, where the primary pumps cannot be operated normally. At this time, the normal flow path in the reactor was still intact.

Case II. The reactor vessel was open to the atmosphere, and the upper flow baffle had been removed. In this situation, the basic flow path in the reactor was still present, but the upper portions of the control-rod channel flow and the third-pass core flow had been altered.

Case III. The reactor vessel was open to the atmosphere, and the upper grid plate had been removed. For this case, the flow orifices were removed with the grid plate, and the normal flow channels were altered. Both the second and third core passes were open to the upper plenum, which became the entire upper portion of the reactor.

Case IV. The reactor vessel was open to the atmosphere, with the manipulator on the reactor flange, and a fuel element had been removed. This situation was studied in two parts: (a) with one second-pass element removed, and (b) with one third-pass element removed.

2.2. Conditions Used for Cases Studied

For the analysis of the four cases listed in the previous section, the following conditions were used:

1. Shutdown was initiated by a scram from full power at the end of core life.

2. Six days after the scram, the primary system was depressurized to the point that the primary pumps were shut off, and the core cooling system was placed into operation.

3. Six days following scram, the decay heat produced by the reactor was 850,000 Btu/hr, approximately 0.35% of full power, 68 MW. (See Figure 1.)

4. The decay heat generated in the core remained constant at 850,000 Btu/hr for all four cases. This, of course, was not a true situation, since the decay heat diminished with time.

5. The portable pump produced 53 gpm through the core cooling system and to the reactor.

6. The purification-system letdown coolers were capable of removing 850,000 Btu/hr.

7. The temperature of the water supplied to the reactor from the core cooling system was 110 F (maximum).

8. The third-pass hot channel produced 3.46 times as much power as an average channel and had an axial overpower factor of 1.5 at the hot spot.

9. The second-pass hot channel produced 3 times as much power as an average channel, had an axial overpower factor of 1.5 at the hot spot, and had reversed flow when the normal flow direction in the second pass was downward.

2.3. Calculations and Results

The basic flow-calculation models used for the study are shown in Figures 2 through 5.

The results of the analysis, based on the models described, are shown in Table 1 and lead to the following points of discussion:

1. The primary driving force, with respect to flow in the core, was not the portable pump in the core cooling system, but the density head developed in the core. This made the problem of core cooling one of natural circulation.

2. As the table indicates, there was no possibility of generating any net steam in the core, based on the calculation models. Likewise, the maximum temperatures in the hot channels indicated that no local boiling would occur.

3. It was thought that there might exist a short period of stagnation in a hot channel just prior to flow reversal. This condition was analyzed for a bulk temperature of 150 F. The maximum fuel-rod surface temperature derived in this manner was 163 F. The behavior of the channel in this condition was very similar to that of free convective heat transfer.

4. Since, in many cases, the flow was in the laminar range, the low flows necessitated a complete recalculation of flow path.

5. The small differences between the hot-channel temperatures and the average channel temperatures were a direct result of the low flows. The flows were so low that the amount of flow was primarily a function of heat input to the water, rather than friction and form losses. Thus, the flow rose almost in direct proportion to the power, which in turn kept the temperatures low.

6. The hot-channel surface temperatures were calculated, taking into account the higher hot-channel flows, for three points: midway up the channel, two thirds the way up the channel, and at the outlet. The highest surface temperature was, in all cases, at the channel outlet.

7. In Cases I and II there was a distinct possibility of local flow reversals in the downflowing second pass. The specific condition considered was one in which a fuel rod was producing three times the power of the channel next to it. It was judged that, under these circumstances, the flow in the channel surrounding the higher powered pin reversed to upflow.

Once again it was found that, due to the low flow rates, the flow in a given channel was more of a function of the power to the coolant than of the channel friction losses. As in the other channels, the flow followed the power and caused smaller rises in coolant and rod surface temperatures than normally expected.

8. A considerable amount of conservatism was introduced into the analysis by assuming that the decay heat input to the water was constant, i. e. , 850,000 Btu/hr for all of the cases. In reality, this input continually drops so that, if the last step (Case IV) were taken 10 days from shutdown, the power to the water would have decreased by about 15%. However, considering the uncertainties in the schedule for removing the elements, the method used was believed to be better.

9. Although the study was made with one second-pass fuel element removed (Case IV a) and with one third-pass fuel element removed (Case IV b), the analysis showed that the core would be adequately cooled if any combination of fuel elements removed from the second or third passes were considered. Consequently, the core will be adequately cooled if it is reshuffled in 1965, or if it is completely changed during any future refueling operation.

10. The analysis was based on 68-MW full power, end of core life, and 6 days after shutdown. If it is desired to use 80 MW as the full power, this analysis holds true for 9 days after shutdown.

Table 1. Results of Reactor Temperature and Flow Analysis

Case number	Flow direction	Average flows, (a) lbs/hr-element		Core coolant temperatures, F			Hot channel maximum surface temperature, F
		2nd pass	3rd pass	Inlet	Avg. outlet	Hot channel outlet	
I	2nd pass—down 3rd pass—up	4,220	4,220	137.2	150	159.7	163
I(a)	2nd pass—re-versed locally	16,880	--	143.6	--	154.0	157
II	Same as I	3,590	3,590	135	150	159.5	163
II(a)	Same as I(a)	14,550	--	142.5	--	154.5	158
III(a)	2nd & 3rd passes—up	890	890	120	150	153.5	156.5
III(b)	8 second pass + 4 third pass elements—down	7,040 in the downcomers		140	150	160	163
	8 second pass + 12 third pass elements—up	4,875 in the risers					
IV(a)	2nd & 3rd passes—up	1,077	1,077	127	150	155.8	158.8
IV(b)	----- Same as Case IV(a) -----						

(a) Leakage flows were calculated and their effects were included in all cases. They have been omitted for the sake of simplicity.

Figure 1. Ratio of Decay Power to Power at Scram at the End of Core Life Vs Time After Scram

Figure 2. Case I Flow-Calculation Model

The first case, with all internal parts intact, had essentially the same model that was used in all previous calculations. Flow was down through the second pass and up through the third pass to the upper plenum. The letdown coolers received 26,500 lb/hr, with the rest of the flow going down through the flow orifices in the upper flow baffle and grid plate.

Figure 3. Case II Flow-Calculation Model

In the second case, the upper flow baffle had been removed. Whereas the basic flow path was still intact, due to the upper grid plate, essentially all of the resistance in the upper portions of the control rod channel and third pass was removed with the baffle.

Figure 4. Case III Flow-Calculation Model

For Case III the upper grid plate was removed. The flow orifices were removed with the grid plate, and the old flow channels were destroyed. The second and third passes opened into the upper plenum, which became the entire upper portion of the reactor. A new flow channel (designated as "Feed Channel" in the diagram above) was opened up, however, by the removal of the grid plate. In case III two possibilities were considered: (a) The entire second pass had reversed flow, with the only downflow coming through the "feed channel", and (b) twelve elements were downflowing along with the "feed channel".

Figure 5. Case IV Flow-Calculation Model

For Case IV (a) a fuel element had been removed from the second pass, and this, along with the "feed channel", was downflowing and feeding the entire second and third passes, which were upflowing. Case IV (b) was the same as IV (a) except that a third-pass fuel element rather than a second-pass element had been removed.

3. ANALYSIS OF CORE COOLING SYSTEM

Figure 6 shows the core cooling system developed in previous work.¹ This system removes the decay heat generated in the core by using the present primary purification system letdown coolers and the present portable pump in a line between the flange connection located downstream of the coolers and the flange on the buffer seal system make-up line. To reaffirm satisfactory performance of this system, an independent analysis was made of pressure drop in the lines between the primary purification system connection at the primary system piping and the suction of the portable pump. This analysis verified that sufficient net positive suction head (NPSH) was available for the pump. The results obtained in this independent analysis were in complete agreement with those of the previous work. The results in both cases were generally as follows:

1. Sufficient NPSH is available to pass 53 gpm through the core cooling system.
2. In order to obtain this flow, two sets of letdown coolers (all four) must be in operation, and the portable pump must be located at or below the grating behind the letdown coolers.

¹ J. W. Merchant, Core Cooling During Reactor Shutdown, Letter to W. F. Long dated January 3, 1963, File 70-0002-12, transmitted to R. D. Schilling, January 9, 1963.

Figure 6. Engineering Flow Diagram for Shutdown Circulating System

4. ALTERNATE COOLING METHODS

A parallel study was made to determine what other methods could be used to cool the core during refueling if the core cooling system did not prove to be adequate. The primary pumps were first investigated to see whether they could be operated with the reactor vessel open to the atmosphere. Pressure drop calculations showed that the critical NPSH requirement for a pump was not available if the pump were normally operated at half speed, i.e., no throttling. (The critical NPSH data was obtained from pump vendor test results.) However, the calculations did indicate that the available NPSH would be higher than the critical NPSH requirement if the pump were operated at half speed with flows between 1500 and 2500 gpm. For this flow range, of course, the primary gate valve (reactor inlet gate valve) would have to be throttled. The pump vendor and the valve vendor were consulted, and they stated that it would be satisfactory to operate the pump and the gate valve under these conditions for the estimated duration of the refueling cycle. The pump vendor recommended, however, that the pump flow should be maintained at about 2000 gpm with accurate flow measurements to prevent operating near the "nonoperating" point of 1500 gpm or near the point where the available NPSH is lower than the critical NPSH.

A second method briefly considered for reducing the pump flow, and thus eliminating the necessity of throttling the gate valve, was the possibility of reducing the pump speed by operating the auxiliary 750-KW diesel generators at a reduced electrical frequency. Calculations indicated that the pump would have to operate at a speed of about 745 rpm to pump 2000 gpm with no throttling of the valve. The electrical operation under these conditions appeared to be satisfactory; however, mechanical problems could exist with the pump: (1) the auxiliary pump impeller in the pump might not circulate enough water

through the motor cavity to properly lubricate the thrust bearings, and (2) the low water flow in the motor cavity might cause excessive heating of the cavity.

It is further pointed out that, when the upper flow baffle has been removed and the upper grid plate is still in place, there might be problems with the grid plate floating out of place owing to force created by the flow from a primary pump.

The alternate investigation was carried further to consider the possibility of using a boiler to dissipate the decay heat. In this arrangement a vacuum would be maintained in the boiler with the main or auxiliary condensers. The heat would be transferred from the primary system to the surface of the water in the boiler drum by natural circulation, and then would be carried to the condenser or condensers by the boiling action at the surface of the water in the steam drum. This method appeared to be feasible, but when it was compared with the core cooling system shown in Figure 6, the latter method had the following advantages:

1. The vacuum method required the operation of a great deal of equipment, whereas the operation of the core cooling was very simple.

2. The major equipment (coolers and pump) was already available for the core cooling system. Only a small amount of piping, fittings, and instrumentation was required to complete the permanent installation of the system, and it was not necessary to cut into existing piping.

5. CONCLUSIONS

The results of the study are summarized in the following conclusions.

1. Based on the shutdown circulating system shown in Figure 6, a flow rate of about 53 gpm can be obtained through the purification system letdown coolers with the portable pump connected downstream of the coolers. In order to obtain this flow, however, both sets of letdown coolers must be in operation and the pump must be located on or below the grating behind the coolers.

2. Based on a scram from full power at the end of core life with the primary system depressurized in six days after scram, the 53 gpm circulated through the reactor by the portable pump will adequately remove the decay heat generated during all phases of the refueling operation.

3. For the conditions described in Conclusion 2 above, the maximum coolant temperature (hot-channel outlet temperature) created in the reactor for all phases of the refueling operation is 160 F, while the maximum fuel-rod surface temperature created is 163 F. Consequently, no net steam is generated in the core.

4. With two sets of letdown coolers in service and with 53 gpm of coolant flowing, the shutdown circulating system is capable of handling the decay heat generated six days after scram from full power at the end of core life with a 150 F inlet to the letdown coolers and with a 110 F outlet from the coolers.

5. When the primary system is totally depressurized, or when the reactor vessel is open to the atmosphere, a primary pump can be operated at half speed with the flow throttled to about 2000 gpm by the reactor inlet gate valve. However, the reactor flow and temperature analysis shows that this mode of operation is not necessary.

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