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NS SAVANNAH SAFEGUARDS REPORT FOR 80-MW OPERATION

FEBRUARY 1963

THE BABCOCK & WILCOX COMPANY

ATOMIC ENERGY DIVISION

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NS SAVANNAH SAFEGUARDS REPORT FOR 80-MW OPERATION

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ABSTRACT

Operation of the NS Savannah has demonstrated that with certain sea conditions a reactor power greater than 69 MW is desirable. The design and operating characteristics of the power plant equipment have been reviewed and have been found to be adequate to permit operation at a steady-state reactor power of 80 MW. Transient operation and potential accidents to the power plant have been reviewed and demonstrate that the increasing of reactor power to 80 MW can be accomplished without significantly affecting the existing safety margins. Operation at 80 MW increases the maximum fission-product inventory of the core by 16% for the maximum credible accident. However, the accepted operating procedures for the ship limit the allowable operating reactor power as necessary to minimize the potential environmental hazard to the general public. In the open sea the ship and its occupants are under direct control of the crew. All passengers and non-essential crew members will be directed to the area of least radiation. Strict control of radiation dose will be enforced by personnel trained in health physics to limit an individual person's integrated dose to acceptable values. It is concluded, therefore, that the NS Savannah can be operated at 80 MW without compromising the existing safety of the ship.

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1. INTRODUCTION

This report presents the results of a study of the design and operation of the NS Savannah to demonstrate that the increasing of the operating power of the reactor from 69 to 80 MW does not present any undue hazard to the general public or to the occupants of the ship.

The ship's operational experience has indicated that increased operating power is desired. Though continuous operation at this increased power is not intended, a greater power margin is desirable to permit normal maneuverability while providing the increased electrical power and auxiliary steam required for ship operation under all sea conditions.

Previous analyses[1,2] and operational experience to date have demonstrated that the operation at 69 MW presents no appreciable hazard. This report reviews these analyses to demonstrate that the increase to 80 MW does not affect previous conclusions.

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2. EFFECTS OF 80-MW REACTOR POWER ON NORMAL PLANT OPERATION

2.1. General

Increasing the operating power of the reactor can be expected to affect the normal operating conditions of certain power-plant components. This section of the report examines the power plant to demonstrate that neither the safe operating capability of the individual components nor the combined capability of the power plant is exceeded by increasing the reactor power to 80 MW.

2.2. Steam-Plant Systems

The steam generators and the feedwater-system components are the major steam-plant components affected by the increased steam flow associated with a reactor power of 80 MW.

2.2.1. Steam Generators

Each steam generator was designed for a 37-MW full-power operating condition. Operation of these units at 40 MW each (153,700 lb/hr) presents only an 8.1% overload above the 37-MW (142,170 lb/hr) design condition. A study was made to determine the operating characteristics of the steam generators at 40 MW. Calculations show that, whereas the temperature difference across the tube sheet has increased by approximately 5 F above that at 37 MW, the difference is still more than 30 F less than the maximum allowable temperature difference of 100 F. The circulation ratio at 40 MW is sufficient so that the presently installed steam separating equipment is adequate to produce steam at or below the design maximum moisture content. Based on the results of this study, it is concluded that the steam generators are capable of steady-state 40-MW operation without exceeding their safe operating limits.

2.2.2. Feedwater Components

The major feedwater components affected by the increase in steam flow are the main-feed pumps, the deaerating feedwater heater, and the third-stage feedwater heater. At the increased flow rate two turbine-driven main-feed pumps are required; however, since each pump by itself has almost sufficient capacity to provide the total feedwater requirements, the combined capacity of both pumps is more than sufficient. The deaerating feedwater heater has a design maximum capacity of 310,000 pounds per hour, which is adequate to process the required feedwater at the increased flow rate. At this capacity the deaerator operates at its maximum efficiency and sufficiently reduces the oxygen concentration in the feedwater. The third-stage feedwater heater has the capability of operating at the increased flow rate. The only possible change would be a slight decrease in the final feedwater temperature. A lower feedwater temperature would lower the overall plant efficiency but would not be detrimental to plant operation. It is concluded that the feedwater-system components are capable of safe operation at the increased flow rate associated with 80-MW reactor operation.

2.3. Primary-Plant Systems

During normal startup, operation, and shutdown of the primary-plant systems, no significant operational differences exist if the reactor is operating at 80 MW, rather than 69 MW. The steady-state primary-system temperatures at 80 MW are 495.7 F reactor inlet, compared to 497.4 at 69 MW, and 520.3 F reactor outlet, compared to 518.6 at 69 MW. These slight temperature differences do not affect the normal operation of any of the primary-plant systems.

The emergency cooling system is the only primary-plant system whose operation is affected by 80-MW reactor operation. This system is not used for the normal shutdown of the plant but is provided in the event that all the normal electrical power is lost. Figure 1 presents the primary-system pressure as a function of time following a scram. This curve is based on infinite irradiation time at the power indicated and assumes that the emergency cooling system provides the only source of decay-heat removal. Figure 2 presents the temperature and pressure in the steam generators following a scram for the same conditions as Figure 1.

Under the assumed conditions and with the existing emergency-cooler flow rate of 40 gpm, the pressurizer safety valve lifts approximately 30 minutes after the scram. In addition, the pressure on the steam side of the steam generator approaches the set pressure of the safety valves. Though neither of these conditions presents any appreciable hazard, the lifting of safety valves is not desired. Therefore, the primary-coolant flow rate through the emergency cooler is increased to 80 gpm by adjustment of the existing flow-control valves to provide the same margin against safety-valve lifting as provided by a 40-gpm flow rate at 69 MW. The total primary-coolant flow rate through the core remains 200 gpm.

Operation of the emergency cooler with a primary-coolant flow rate of 80 gpm is within the safe operating capabilities of the unit. Operation of the emergency cooling system at the 80-gpm cooler flow rate after the peak primary-coolant temperature is reached cools the system at a maximum rate of approximately 40 F per hour. This is 10 F less than the maximum allowable cooldown rate.

It is concluded that steady-state operation of the primary-plant systems presents no additional hazard at 80 MW, compared to operation at 69 MW. The operating condition of the primary-plant systems during transients and abnormal conditions is analyzed in Section 3.

2.4. Instrumentation and Control Systems

2.4.1. Nonnuclear

The capabilities of the existing nonnuclear instrumentation and control components have been reviewed and are adequate to perform their required functions at the increased reactor power of 80 MW.

2.4.2. Nuclear

The existing nuclear instrumentation is capable of providing continuous neutron-level or reactor-power measurement from source level through 120 MW (150% of 80 MW).

2.5. Shielding

Essentially, radiation in the ship varies directly with the power density in the reactor core. Raising the reactor power to 80 MW presents no hazard from the standpoint of radiation dose rates in accessible locations in the ship. Whereas the dose rates increase almost directly by the factor 80/69 = 1.16, results of the shield test survey[3] indicate that full-time access design criteria would probably not be exceeded even if the reactor power were raised by a factor of 1.5. This indication applies even in the passenger-access areas, where the design dose-rate criterion is one-tenth of that for crew-access areas. It is concluded that the existing shielding is adequate for a power of 80 MW.

Figure 1. Pressurizer Pressure Vs Time After Scram for Two Emergency Cooler Flow Rates

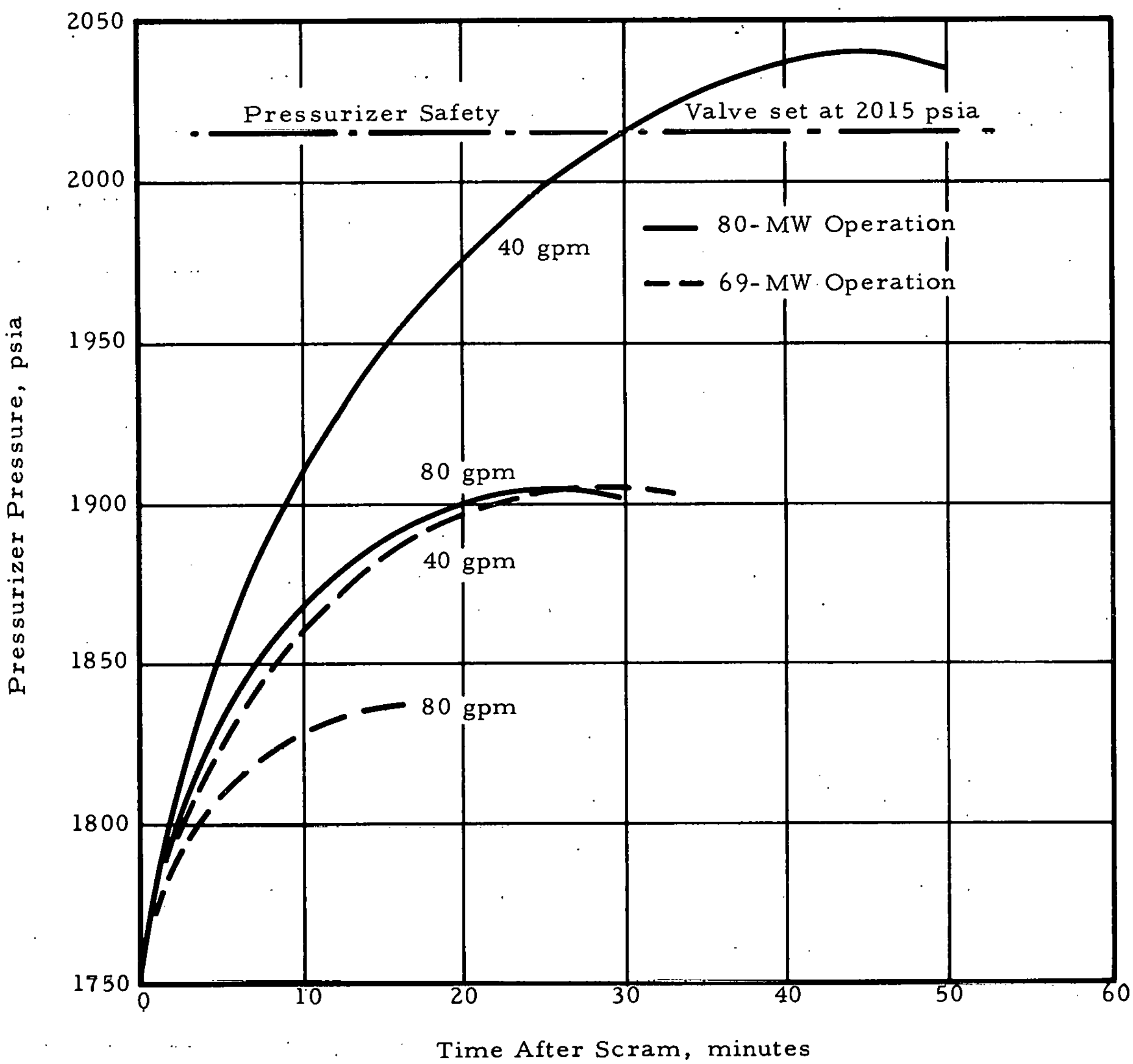
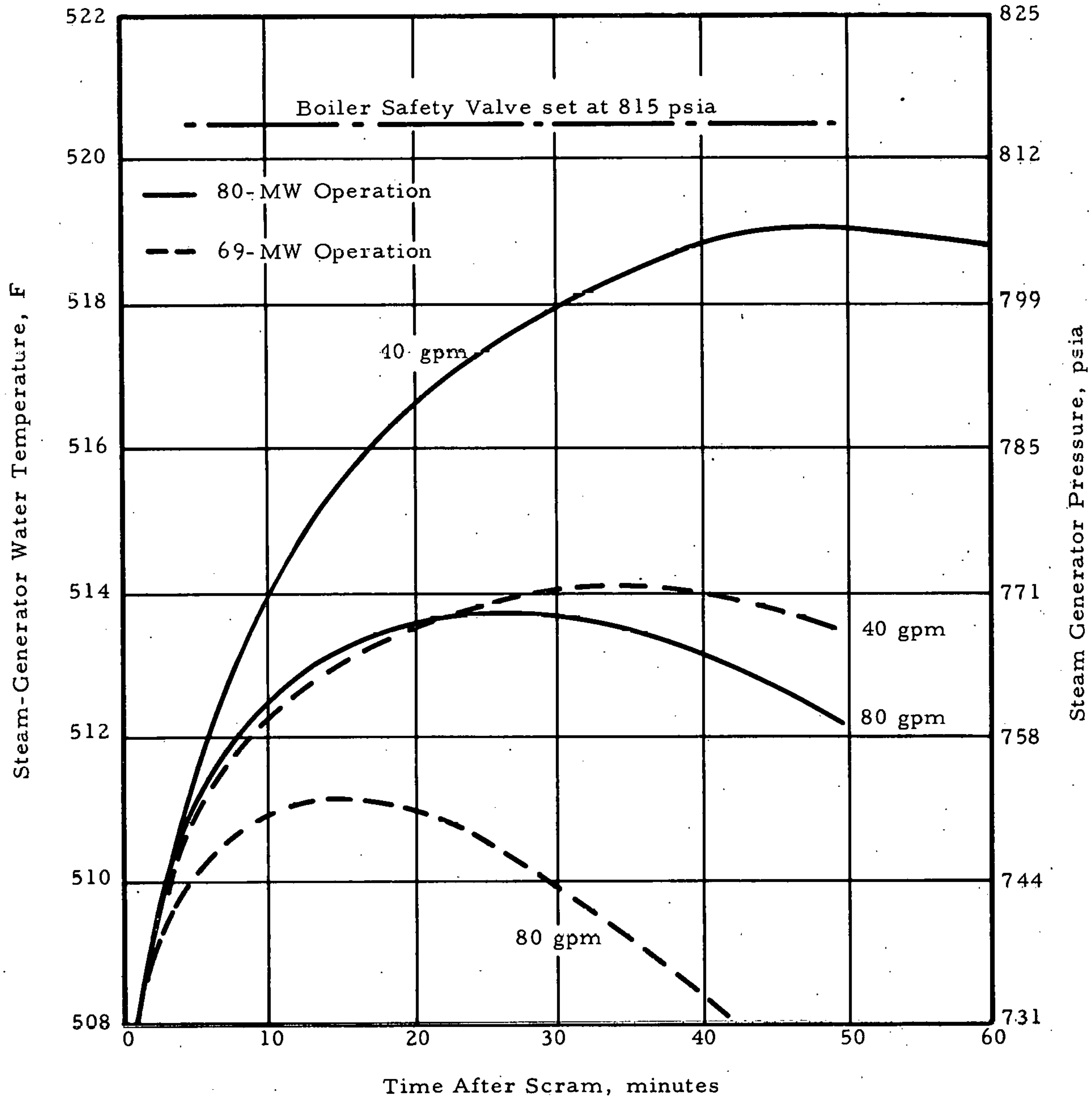


Figure 2. Steam Generator Pressure Vs Time After Scram for Two Emergency Cooler Flow Rates



3. ACCIDENT ANALYSIS

3.1. General

Previous analyses [1,2] of the potential accidents to the NS Savannah have demonstrated that it can be operated at a reactor power of 69 MW with no undue hazard to the general public. Section 2 has shown that the power plant is capable of being operated at a steady-state reactor power of 80 MW. This section evaluates the effects on potential accidents of increasing the operating power to 80 MW and increasing the overpower scram setpoint.

3.2. Reactivity Accidents

3.2.1. Startup Accident

The maximum potential reactivity-addition rate in a startup accident with the existing control-rod-drive system is 4×10^{-4} δk per second. Previous calculations and Special Power Excursion Reactor Test experiments with Savannah-type fuel pins have demonstrated that the initial power excursion reaches a maximum value and then levels off at some steady-state power entirely due to the inherent negative-reactivity effect of the oxide-fuel temperature (Doppler effect). On the assumption that all other safety features fail except the overpower scram, the following has been shown in BAW 1164:[1]

> ". . . inherent effects would halt the initial power rise approximately 150 milliseconds after the overpower trip is reached. Since the peak is passed before actual control rod motion takes place, small changes in scram delay time or a change in scram velocity would make little difference in the end result and certainly none in peak power".

For the purpose of analysis, raising the overpower scram setpoint is considered as an increase in the scram delay time. Furthermore, the overpower scram at 130% of full power is conservatively assumed to be the only effective safety action. Figure 2. 1-2 of Reference 1 shows that the difference in the time when the power is at 90 MW (130% of 69 MW) compared to 104 MW (130% of 80 MW) is approximately 10 milliseconds. It is concluded that the raising of the overpower scram setpoint to 104 MW does not affect the startup-accident analysis and, therefore, no additional hazard is introduced.

The maximum potential reactivity-addition rate in a startup accident with the replacement control-rod-drive system is 1×10^{-3} δk per second. Reference 2 has demonstrated that no undue hazard to the public exists with the replacement drive system. The same analysis applied to the existing drive system is used to demonstrate the effect of increasing the overpower scram setpoint on the replacement drive system. Figure 4-2 of Reference 2 shows that the difference in the time when the power is at 90 MW compared to 104 MW is also approximately 10 milliseconds. The raising of the overpower scram setpoint to 104 MW does not affect the conclusions regarding a startup accident with the replacement control-rod-drive system.

In summary, the mechanism that limits the power excursion in a startup accident is inherent to the oxide-fuel reactor core. The exact setpoint of the overpower scram has very little effect on the accident. No core damage can result nor hazard exist as a result of a continuous rod withdrawal from source level provided that the excursion is ultimately terminated by rod insertion shortly after the initial power peak is passed.

3.2.2. Rod Withdrawal in the Power Range

3.2.2.1. Introduction

Previous analyses[1,2] have demonstrated that the continuous-rod-withdrawal accident with the reactor operating at some initial power presents no hazard with either the existing or replacement control-rod-drive systems, provided that the overpower scram occurs at 90 MW (130% of 69 MW). In order to demonstrate that a higher overpower scram setpoint is adequate to protect the plant, the peak heat flux and maximum primary-system pressure during the accident are analyzed.

3.2.2.2. Analysis

The burnout limit of the core has been defined[2] as an average heat flux of 190,000 Btu/hr-ft^2 in the third pass when four pumps are operating and 120,000 Btu/hr-ft^2 when one pump is operating.

These are the average steady-state heat fluxes that would exist in the third pass before the hot spot in the core could approach a burnout condition. Reference 2 has further demonstrated that the peak value of the third-pass average heat flux is a maximum when the reactor is operating at full power prior to the accident. Figure 4-5 of Reference 2 shows that with an initial 80 MW, rather than 69 MW, the peak value for the maximum reactivity-addition rate is only 81,000 Btu/hr-ft^2, well below the burnout limit when either one or four pumps are operating. Therefore it is concluded that, for the maximum reactivity-addition rate available, a continuous-rod-withdrawal accident does not produce excessive heat fluxes.

The maximum reactivity-addition rates in a rod-withdrawal accident do not necessarily produce the worst operating conditions, particularly with respect to the rate of primary-system heating. For rapid reactivity-addition rates, the overpower scram setpoint is reached in a short period of time and the reactor is shut down with only a small increase in average primary-system temperature. For slow reactivity-addition rates, the moderator-temperature coefficient limits the actual power rise and ultimate shutdown is due to reaching the high-temperature scram setpoint. The reactivity-addition rate that produces the maximum rate of primary-coolant temperature rise occurs when the overpower scram setpoint and the high reactor-outlet-temperature scram setpoint are reached at approximately the same time after initiation of the rod withdrawal. The rate of primary-coolant temperature rise determines the rate of water flow into the pressurizer. If there is no spray, the required rate of steam relief is determined in this way. The capacity of the pressurizer relief valve is the limiting factor for the allowable rate of coolant temperature rise. Expressed in terms of temperature rise, the valve has a steam capacity of approximately 0.7 degrees per second. If the required steam relieving rate exceeds the installed capacity, over-pressurization of the primary system is possible.

Figure 3 presents the pressure buildup versus withdrawal rate for beginning of life conditions. With the overpower scram set at 104 MW (130% of 80 MW) a reactivity-addition rate of 2.25×10^{-4} δk per second produces the greatest pressure buildup in the primary system. With the overpower scram set at 90 MW a reactivity-addition rate of 2.0×10^{-4} δk per second produces the greatest pressure buildup. The following tabulation indicates the maximum primary-system pressures for these two reactivity addition rates if there is no spray action, no steam relief by the pilot-operated relief valve, and no net blowdown.

Scram settings	Beginning of life	End of life
104 MW and 540 F	2190 psia	2290 psia
90 MW and 540 F	2100 psia	2200 psia

The preceding tabulation shows that scram settings of 90-MW overpower and 540 F reactor-outlet temperature are adequate to prevent a pressure buildup above the maximum allowable by the ASME Code (2200 psia). With an overpower scram setting of 104 MW, the peak pressure at the end of life exceeds that allowed by the Code. For this condition, a high-pressure scram set at 2000 psia should be added to the circuitry of the safety system. With the addition of the high-pressure scram to the safety system, the primary-system pressure never exceeds that permissible by the applicable design codes. (See Figure 3a.)

The overpower scram setting of 104 MW protects the core from burnout even if the reactor operates continuously at 104 MW for four-pump, three-pump, and two-pump operation. For one-pump operation at 1500 psia the burnout power is 105 MW. If the reactor is to be operated on one pump, the overpower scram should be set at 96 MW to allow a margin for instrument and heat balance error in the event a rod-withdrawal accident were to occur. To avoid the necessity for resetting the overpower scram in the event that one-pump operation is required, the overpower scram should be set at 96 MW for all cases. Ship operation to date has demonstrated that the overpower margin above steady-state requirements is relatively small and that the 16-MW margin provided by the scram setting of 96 MW (120% of 80 MW) is adequate. This reduction in the overpower scram setpoint does not affect the conclusions reached for previous analyses using a scram setpoint of 104 MW; rather the safety margin is further increased.

Figure 3. Pressure Vs Reactivity-Addition Rate, Rod Withdrawal Accident

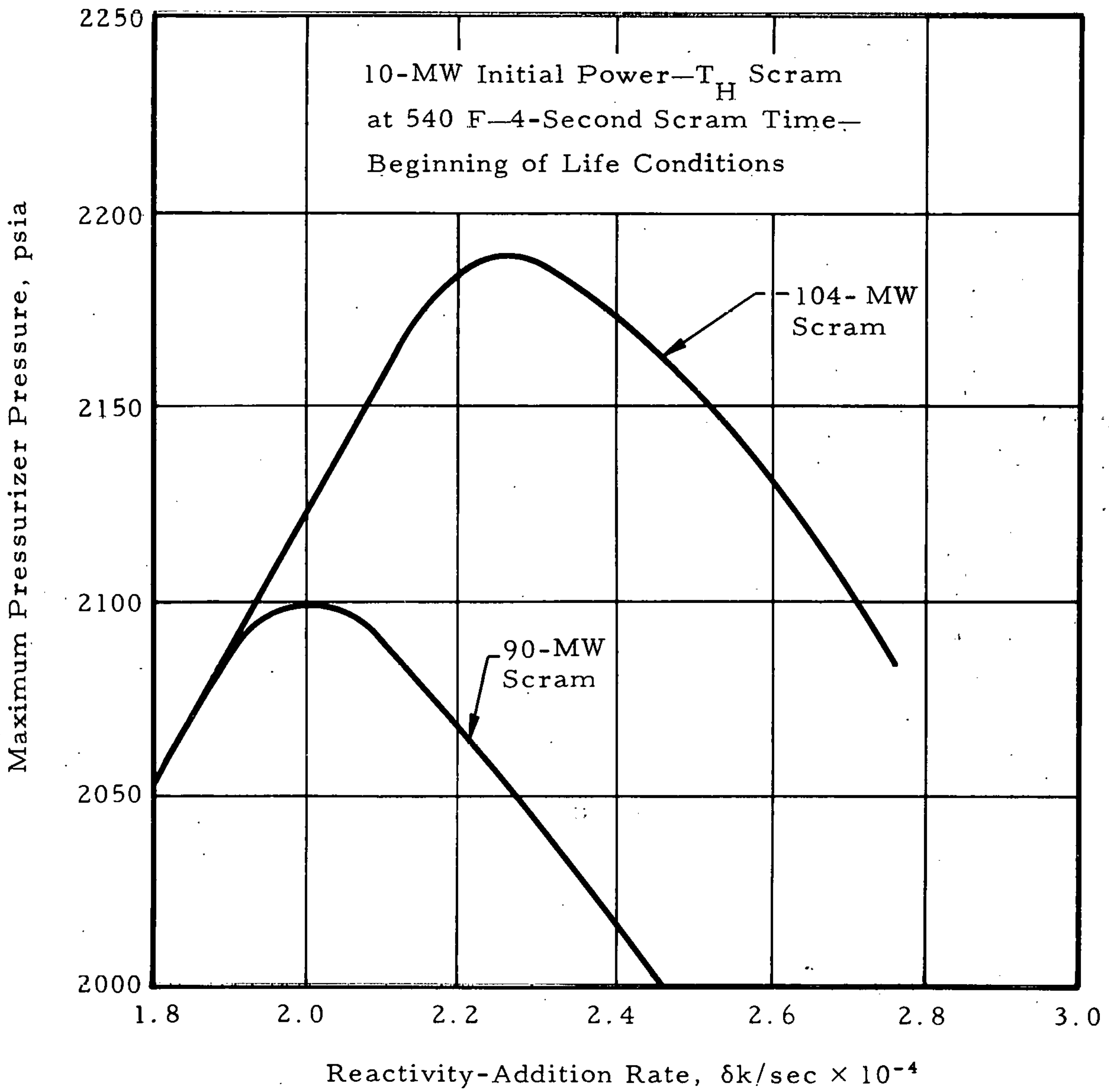
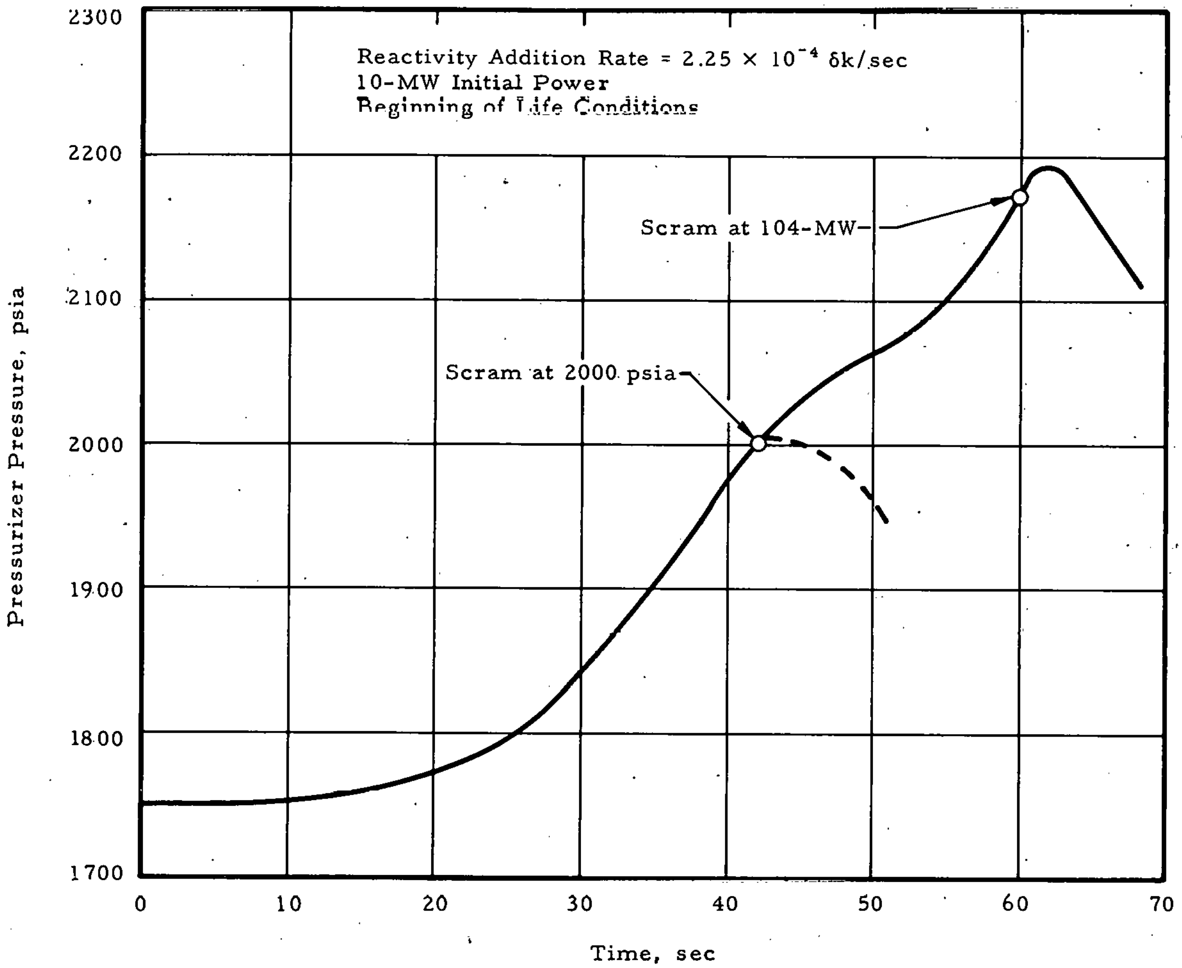


Figure 3a. Pressure Vs Time, Rod Withdrawal Accident



3.2.2.3. Conclusions

Reference 2 has shown that rod-withdrawal accidents several times worse than the credible accidents do not cause excessive heat flux in the core. Increasing the overpower scram setpoint from 90 MW (130% of 69 MW) to 96 MW (120% of 80 MW) does not alter this conclusion. With the addition of a high-pressure scram set at 2000 psia, rod-withdrawal accidents do not produce excessive pressures in the primary system. Therefore, it is concluded that increasing the operating reactor power to 80 MW does not present any appreciable hazard from rod-withdrawal accidents.

3.2.3. Additional Reactivity Accidents

Both References 1 and 2 have analyzed the potential reactivity-addition mechanisms such as cold water accidents, excessive steam demand, control rod ejection, and xenon burnout transients. A cold water accident is prohibited by several interlocks and is not considered credible. Excessive steam demand from rupture of the secondary steam system or accidental opening of the turbine bypass valve results in reactivity-addition rates comparable to those used in normal control.[1] Even with no safety action fuel-element burnout does not occur. Increasing the reactor power to 80 MW does not affect this conclusion, since the reactivity-addition rate is not significantly changed. Ejection of a control rod has been shown to increase the peak value of the third-pass average heat flux by a constant amount independent of initial power.[2] With the reactor at 80 MW, ejection of the X-rod (1.2% δk) would produce an average heat flux of 115,000 Btu/hr-ft^2 in the third pass. Since burnout does not occur below an average heat flux of 190,000 Btu/hr-ft^2, no core damage would result. Xenon burnout transients introduce reactivity-addition rates that are considerably less than those possible from rod mishandling accidents and, therefore, do not produce any appreciable hazard.

3.2.4. Conclusions

From the above analyses, it is seen that the worst potential reactivity accident occurs from a continuous rod withdrawal in the power range. It has been demonstrated that the addition of a high-pressurizer-pressure scram to the safety system adequately protects the plant for this accident.

It is concluded, therefore, that potential reactivity accidents present no appreciable hazard and do not affect operation of the reactor at 80 MW with either the existing or the replacement control-rod-drive system.

3.3. Mechanical Failures

3.3.1. Fuel-Element Failures

Steady-state operation of the reactor core at 80 MW, rather than 69 MW, is not expected to affect the number of fuel-element failures. At steady state the burnout power has been calculated to be 186 MW and the power corresponding to central melting of the fuel has been calculated to be 110 MW.[1] Subsequent calculations indicate that 128 MW is necessary for central melting. At 80 MW neither central melting of the fuel nor burnout is approached during steady-state operation and, therefore, no increase in fuel-element failures is expected.

3.3.2. Primary-Coolant Leak

Reference 1 has analyzed the effects of various-sized leaks in the primary-coolant system. Increasing the reactor power to 80 MW does not affect the conclusions reached in this reference except for the large leak analyzed as the maximum credible accident. The maximum credible accident for operation at 80 MW is analyzed in Section 3.4.

3.3.3. Control-Rod Failure

Operation of the reactor at 80 MW, rather than 69 MW, does not significantly affect the reactivity balance of the core and the associated control-rod positions. The control-rod positions are essentially unchanged except that the active control-rod group is slightly further withdrawn to compensate for the additional Doppler deficit, 0.00096 δk. The conclusions reached in Reference 1 and 2 about the allowable number of disabled rods are not affected.

3.3.4. Loss of Power to Primary-System Pumps

3.3.4.1. Introduction

Reference 2 presents the latest information and most complete analysis of loss of forced circulation coolant flow caused by loss of power to the primary-system pumps. In this reference the analysis assumes an accident in which there is no safety action by the control rods, with reactor shutdown occurring due to inherent characteristics. The possibility of damage to the core is evaluated by comparing the maximum heat flux in the core with the heat flux necessary to cause a departure from nucleate boiling (DNB). If a DNB is indicated, the heat transfer coefficient at the clad surface is assumed to decrease to 100 Btu/hr-ft^2-F, and the maximum clad temperature is determined. Heat transfer conditions in both passes of the core are considered.

In the third-pass elements during the time that the flow is decreasing the flow coastdown analysis neglects any contribution to the flow caused by natural circulation. Thus, the assumed flow in the hot channel approaches zero and accordingly indicates a DNB. If at this time the heat flux is low enough that it can be safely removed by natural circulation, the DNB does not occur. Instead, heat continues to be removed by natural circulation within the vessel and no damage occurs. A study, which assumes natural circulation inside the reactor vessel only, indicates that for heat fluxes up to 94,000 Btu/hr-ft^2 a DNB does not occur. The flow coastdown analysis of the second-pass elements does take into account the natural circulation between the elements. This is demonstrated as a flow reversal in the course of the accident.

3.3.4.2. Analysis

The loss of coolant flow accident at 69 MW with four pumps at full speed was analyzed in Reference 1 for a 1-second scram time. This analysis demonstrated that no fuel-element damage would result and that the maximum clad-surface temperature would be 625 F. Because the scram time is very short, operation at 80 MW is not expected to affect these conclusions significantly. The accident at 80 MW with a 1-second scram time was not simulated because longer scram times are considered more significant. Some abnormal conditions, such as a reduced gravity effect caused by the ship's motion or a plugged lead screw, increase the scram time with the replacement control-rod-drive system. To analyze these conditions a scram time of 4 seconds was used. Figure 4 presents the results of the simulation of the loss of coolant flow accident for an initial power of 80 MW, four primary pumps initially at full speed, and a 4-second scram time. At 5 seconds a flow reversal occurs in the second pass. At this time, however, the ratio of burnout heat flux to actual heat flux is 4 and no DNB occurs. At approximately 12 seconds the ratio of burnout heat flux to actual heat flux approaches 1, thereby indicating a DNB. However, the maximum third-pass heat flux at this time is only 81,000 Btu/hr-ft^2. Since natural circulation is sufficient for heat fluxes up to 94,000 Btu/hr-ft^2, a DNB does not occur. Therefore, no fuel-element damage occurs from this accident.

To analyze the case of ship inclinations exceeding 45 degrees the loss of coolant flow accident for 80-MW operation with four pumps initially at full speed and no scram action of the control rods was simulated. See Figure 5. At 1-1/2 seconds a flow reversal occurs in the second-pass hot channel and the actual heat flux exceeds the burnout heat flux, indicating a DNB. After the flow reversal the upward flow increases quite rapidly, as indicated by the increasing burnout heat flux ratio. This increasing flow tends to cause recovery from the DNB. However, even if the fuel pin does not recover from the DNB, the maximum clad-surface temperature does not exceed 1400 F as shown in Figure 6. In the third-pass hot channel a DNB is indicated after approximately 12 seconds. The resultant maximum clad-surface temperature is approximately 1110 F. No clad melting occurs at these temperatures; however, some damage to the elements due to weakening of the brazed joints and possible misalignment may occur. The postulated accident can occur only if loss of flow is experienced with the ship inclined more than 45 degrees. Under these two conditions the rods would insert at their normal rate of 13.5 inches per minute with power supplied by the special battery source provided for this purpose.

Additional analyses were made for the pump combinations and reactor powers permitted by the Technical Specifications[4] and have been reported in Reference 2. None of the permissible operating combinations produce a DNB, even with an abnormal scram time of 4 seconds associated with the postulated condition of all lead screws plugged. When there is no scram action by the control rods, all other permissible pump combinations either do not produce a DNB or the resultant peak fuel temperature is less than the 4-pump 80-MW case.

3.3.4.3. Conclusions

Fuel melting and release of fission products do not occur even without a scram. The possibility of any core damage from excessive temperatures during an accident is extremely remote. The only situation in which any damage is possible depends on the simultaneous occurrence of the following:

1. Reactor at 80 MW.
2. Complete loss of all pumping power.
3. No safety action by the control rods.
4. The fuel pins do not recover from the DNB.

In order to lose all pumping, complete loss of electrical power to the main bus is required. Ship inclinations that exceed 45 degrees may cause complete power failure. However, ship operation at 80 MW under sea conditions that produce 45 degree rolls is not likely. In very rough seas general maritime procedures are to reduce speed and power to minimize the possibility of ship structural damage. However, even if all the unlikely circumstances were to occur simultaneously, fuel-element damage would involve only mechanical distortion, and certainly no hazard to the general public or to the occupants of the ship would result. Therefore, it is concluded that the loss of coolant flow accident with the reactor at 80 MW does not present any undue hazard.

3.3.5. Ship Capsize Accident

3.3.5.1. Introduction

A ship capsize accident with the existing control-rod-drive system installed does not present any appreciable hazard for reactor powers of 69 and 80 MW, since the source of energy for scram action is independent of ship attitude. A ship capsize accident with the replacement control-rod-drive system installed has been analyzed for a reactor power of 69 MW.[2] For this power no major failure of the core nor release of fission products occurs.

3.3.5.2. Analysis

The major areas of concern in a ship capsize accident are reactor shutdown, core burnout, and the integrity of the primary system. Reference 2 has demonstrated that the replacement control-rod-drive system is capable of reactor shutdown at any ship attitude through use of the emergency battery-power supply. Operation at 80 MW does not affect the reliability of this shutdown mechanism.

Figure 4. Heat Flux Ratio Vs Time, Four Pumps at Full Speed, 80 MW, Coastdown, 4-Second Scram

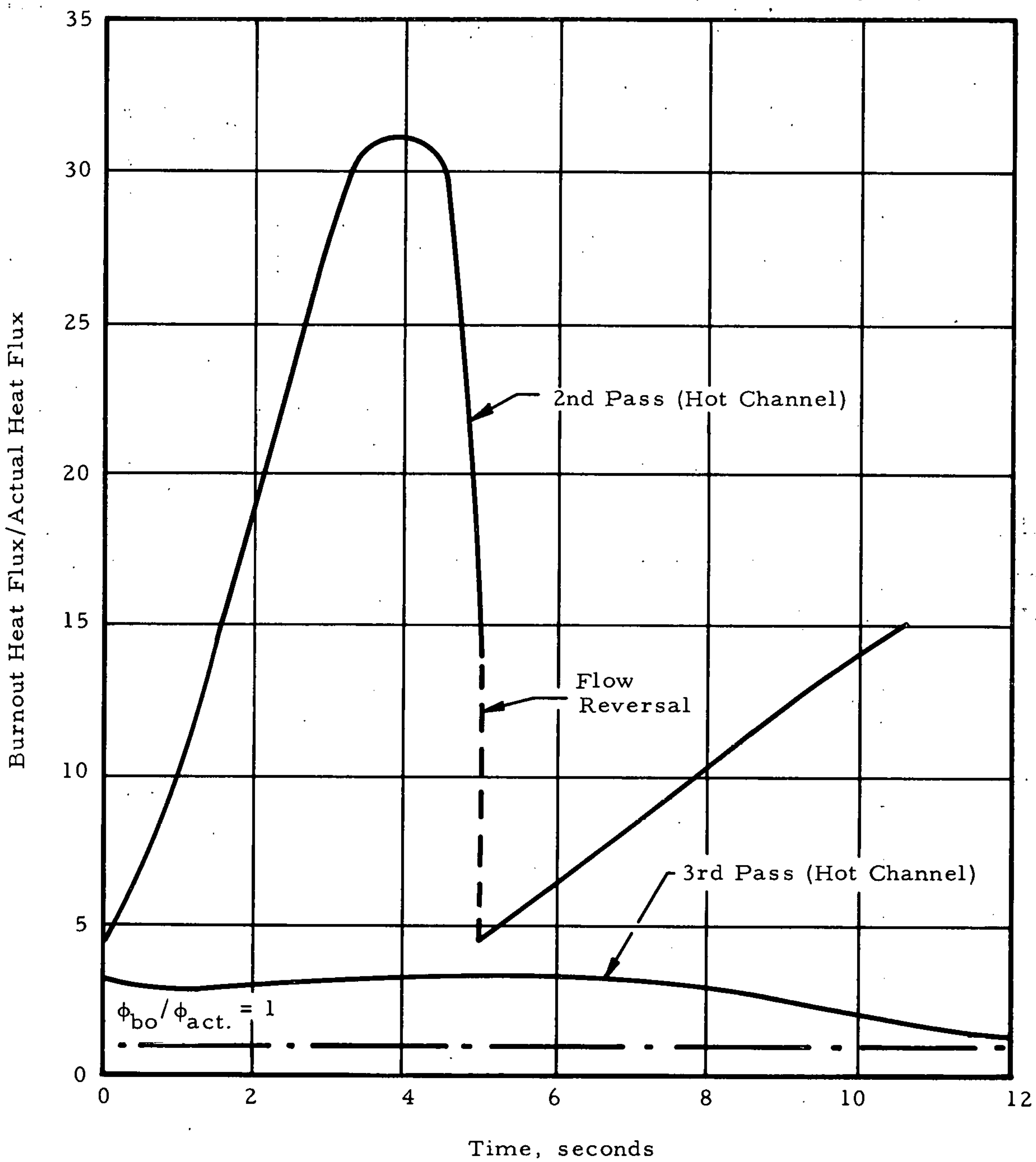


Figure 5. Heat Flux Ratio Vs Time, Four Pumps at Full Speed, 80 MW, Coastdown, No Scram

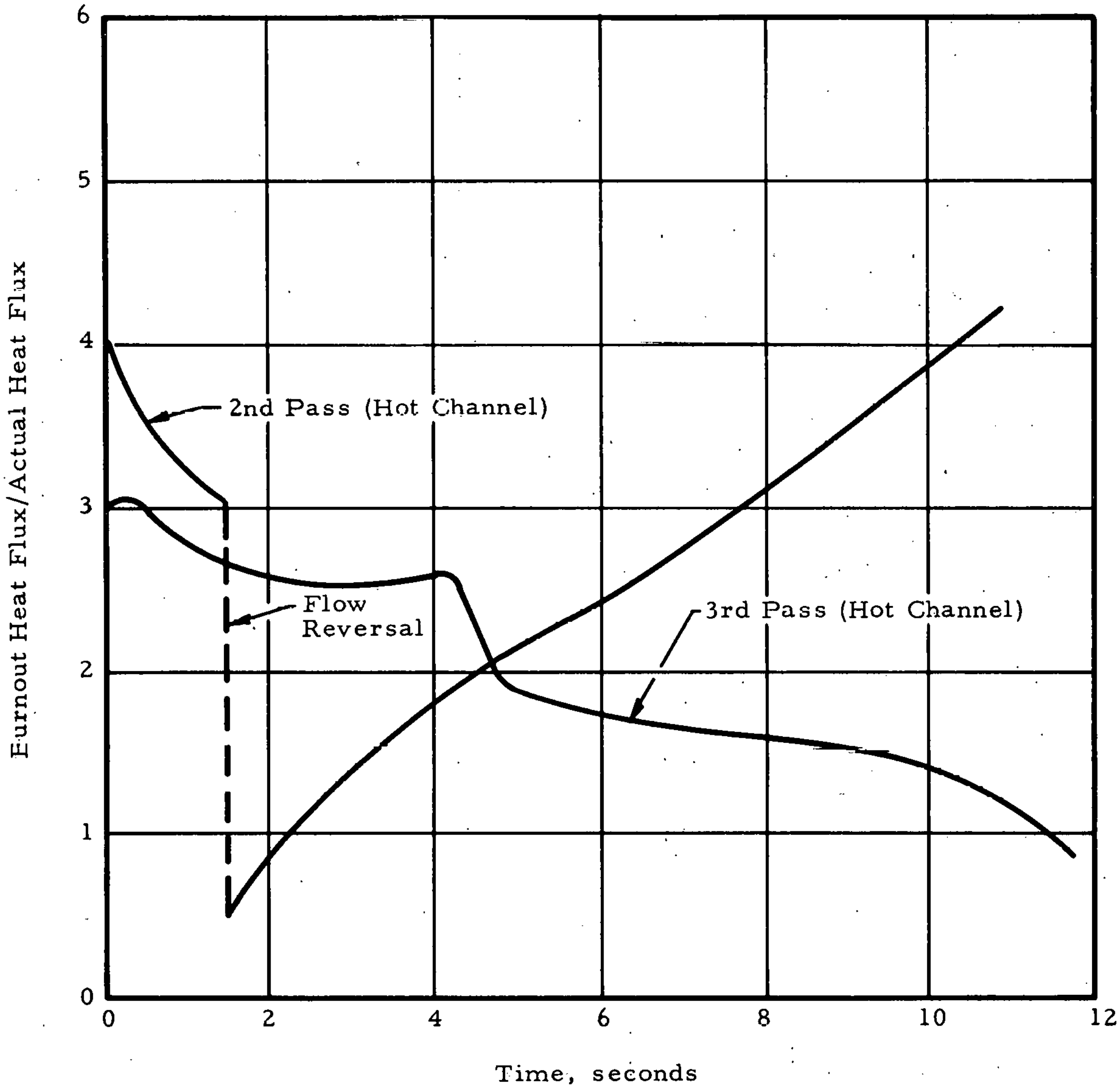
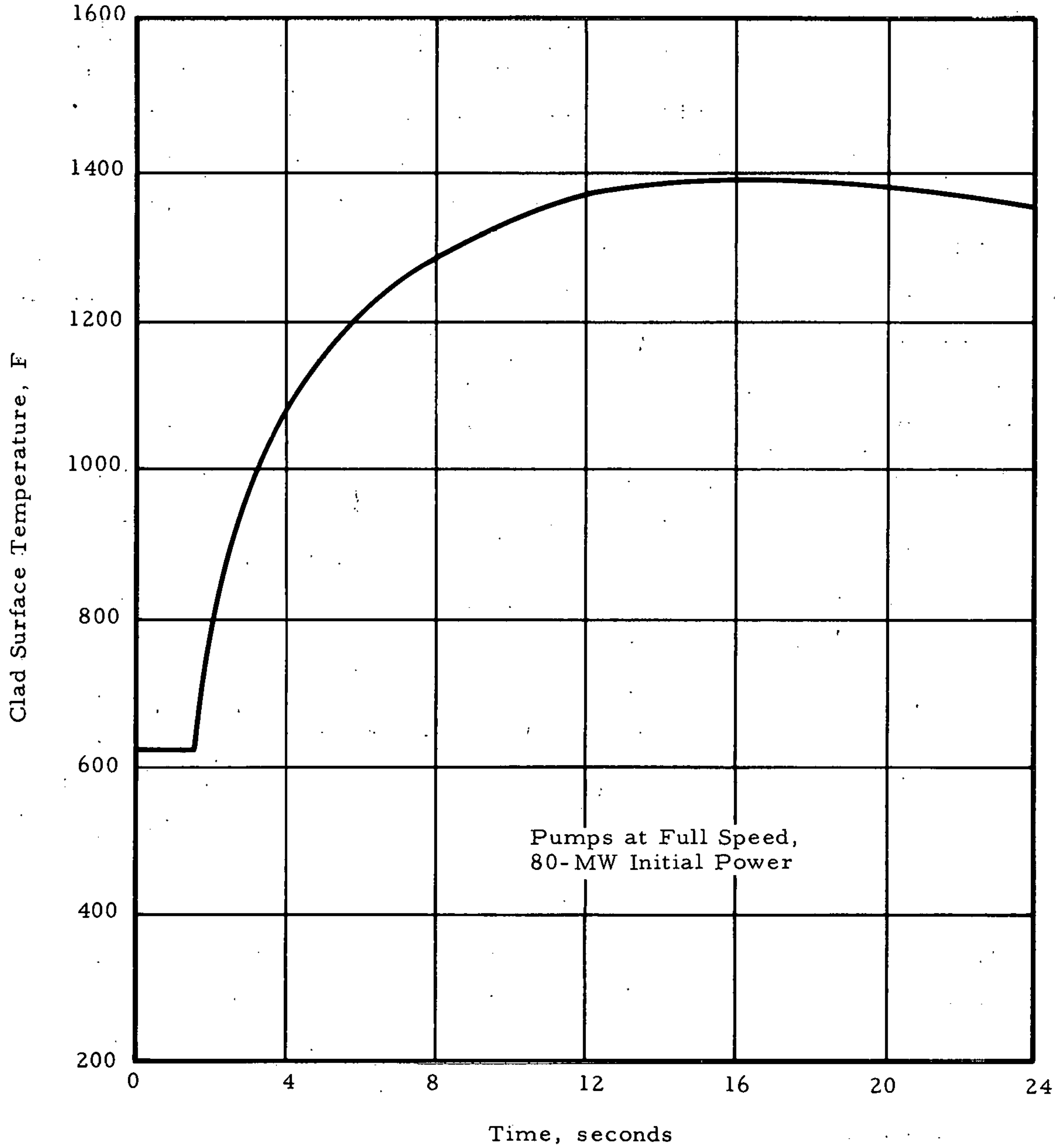


Figure 6. Second-Pass Clad-Surface Temperature Vs Time, Hot Channel, Coastdown, No Scram



Reference 2 has also shown that heat transfer conditions existing in a capsize accident are much better than in the pump coastdown accident. Since it has been shown that fuel-element burnout does not occur for the coastdown accident at 80 MW, a capsize accident at 80 MW does not cause burnout.

The mechanical integrity of the primary system during the course of the ship capsize accident has been demonstrated for initial power of 69 MW and the negative reactivity-addition rate of 3×10^{-4} δk per second associated with the capsize scram. If the initial reactor power is 80 MW, the rod insertion also must compensate for the additional Doppler deficit of 0.00096 δk. At a rate of 3×10^{-4} δk per second, the power is reduced to 69 MW in 3.2 seconds. At an assumed average power of 74.5 MW during this period an additional 225,000 Btu is added to the primary system. This quantity of heat raises the average primary-system temperature 3 F. For the purpose of analysis of the 80 MW accident, the transient temperatures obtained in the 69 MW accident can be increased by 3 F.

If the 80-MW accident occurs with no safety action, the average primary-system temperature is increased from 508 to 547 F. When the primary-system pressure reaches 2000 psig, the rate of change of primary-system average temperature is 0.65 F per second. Under normal operating conditions of pressurizer and relief valve, the primary system is not overpressurized, since the pressurizer is not filled, and the rate of temperature rise is within the relief valve capacity (0.7 F per second).

Because of the nature of the capsize accident proper functioning of the pressurizer and relief valve cannot be assured, and the relief valve may relieve water rather than steam if the ship is severely inclined. A capsize scram removes reactivity at a rate of 3×10^{-4} δk per second. When reactivity is removed at this rate, the maximum pressure does not exceed 2000 psig, since the relief valve has a water flow capacity of 200 gpm. This capacity is about 110% of that required to accommodate expansion of the primary system after the pressure reaches 2000 psi.

3.3.5.3. Conclusions

Operation of the reactor at 80 MW, rather than 69 MW, does not affect the shutdown capabilities of either the existing or replacement control-rod-drive systems in the event of a ship capsize accident. The accident does not cause core burnout, and the integrity of the primary system is not affected. It is concluded that no major failure of the core nor release of fission products occurs even if the reactor is operating at 80 MW at the time the ship capsizes.

3.4. Maximum Credible Accident

In Reference 1 the maximum credible accident to the NS Savannah has been postulated as the complete loss of primary coolant followed by melting of the reactor core and the resultant release of fission products to the containment vessel. Increasing the reactor power to 80 MW does not raise the resultant containment pressure but does increase the quantity of fission products in the containment vessel by 16% following the accident. However permissible dose rates are not exceeded.

3.4.1. Containment Pressure

Table 1 shows the energy released during the maximum credible accident at 69 MW from Reference 1 and compares the energy released at 80 MW using the same assumptions except for power level.

Table 1. Total Heat Released During Blowdown Period

	69-MW reactor power	80-MW reactor power
Primary system	33,436,000	33,436,000
Secondary system	3,560,000	3,485,000
Decay heat	189,000	219,000
Full power for 5 seconds	327,000	379,000
Stored heat in fuel pins	426,000	494,000
Total	37,938,000	38,013,000

The total difference in heat input at 80 MW is only 75,000 Btu, an increase of approximately 0.2%. This additional heat input is insignificant. The resultant containment pressure remains the same and, therefore, does not present any additional hazard.

3.4.2. Environmental Hazards

3.4.2.1. Fission-Product Inventory

Operation of the reactor at 80 MW increases the fission-product inventory of the core. For environmental analysis considerations of the maximum credible accident, continuous operation at 80 MW raises both the fission product inventory and accident-case dose rates by approximately 16% over values for 69-MW operation. However, the accepted operating procedures for the ship limit the allowable reactor operating power as necessary to minimize the potential environmental hazard to the general public. With this means for controlling the potential environmental hazard, operation of the reactor at 80 MW, rather than 69 MW, does not affect the general public.

3.4.2.2. Shipboard Exposures at Sea

Increasing the reactor power to 80 MW can be expected to increase the shipboard exposure by a maximum factor of 1.16. After a reactor accident the passengers and crew members not required for emergency procedures will be directed by the Senior Deck Officer in command to remote areas of the ship as advised by the health physicist. The dose rate at the after docking station is approximately 10 mrem/hr at 69 MW. At 80 MW the dose rate is 11.5 to 12 mrem/hr. This slight increase would not present any appreciable hazard to the passengers. The crew members responsible for emergency procedures will be continuously monitored to ensure that none receive an excessive dose. Increasing the reactor power level to 80 MW only reduces the total time that a specific crew member can work in the radiation field.

3.4.3. Conclusions

It has been shown that increasing the reactor power to 80 MW does not appreciably increase the resultant pressure within the containment vessel following the maximum credible accident. Since reactor power can be adjusted prior to approaching densely populated land areas, 80-MW operation does not necessarily affect the environmental hazard to the general public. At sea, exposure dose rates are increased by a factor of 1.16; however, procedures will be followed to minimize the total exposure received by the occupants of the ship. Therefore, it is concluded that operation of the reactor at 80 MW does not significantly affect the hazards associated with the maximum credible accident.

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4. SUMMARY AND CONCLUSIONS

4.1. Required Modifications for 80-MW Operation

Sections 2 and 3 of this report have analyzed the steady-state and transient operating characteristics of the NS Savannah power plant for a reactor power of 80 MW. Modifications required to ensure safe operation at this power were discussed and are summarized below.

1. The primary-coolant flow rate through the emergency cooler should be increased from 40 gpm to 80 gpm.

2. The power-range instrumentation trips should be set to assure that: with the "Start-Run" switch in the "Start" position the overpower scram will occur at 22.4 MW (28% of 80 MW) or less, and with the switch in the "Run" position the overpower scram will occur at 96 MW (120% of 80 MW) or less.

3. A scram input should be added to the safety system set at a primary-system pressure of 2000 psig.

4.2. Conclusions

With the required modifications completed the reactor may be safely operated at 80 MW. The accident analysis section of this report has demonstrated that either the existing control-rod-drive system or the replacement control-rod-drive system may be used at this increased reactor power. The environmental analysis of 80-MW operation has shown that for all accidents, including the maximum credible, no appreciable difference exists from the hazards associated with 69-MW operation. Therefore, the NS Savannah can be operated at a reactor power of 80 MW without undue hazard to the general public or to the occupants of the ship.

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