CLINCH RIVER

BREEDER REACTOR PROJECT

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PRELIMINARY SAFETY ANALYSIS REPORT

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15.3 UNDERCOOLING DESIGN EVENTS - INTRODUCTION

Of particular importance to the safe operation of the CRBRP is the determination of the response characteristics of the reactor to a group of postulated undercooling events. The reactor response to these undercooling events is characterized, in this section of the PSAR, by the resultant fuel rod hot spot cladding temperature. For these accident events either, 1) the resultant fuel rod cladding temperature will be presented, or 2) it will be shown that the primary or secondary Plant Protection System trip will shut down the reactor before resulting plant temperature changes can be transported to the core. The impact of these Accident Events on Plant Systems and components is less severe than the events presented in the Plant Duty Cycle List. Plant components have been designed to provide 30 year life for the Plant Duty Cycles.

61| 51| Based on the discussion presented in Section 15.1.2 a measure of the severity of these events can only in-part be ascertained by the resultant cladding temperatures of any one event. The true severity of the event on the cladding integrity is a function of the sum total of all the accumulated strains imposed on the cladding during its lifetime. Therefore, the severity of any event should be evaluated on a case by case basis using the cumulative damage function. In order to perform the evaluation process, the transients generated in this section are first compared to the guidelines
61|52|51| established in Section 15.1.2 and when necessary to the umbrella transients described in Section 4.2. If the accident transient falls within the time and temperature confines of the umbrella event, the conclusion can be made that the design life and safety objectives of the fuel assemblies has been conserved. If however, the resultant cladding temperatures exceed the
61|52|51| guidelines limits of Section 15.1.2 then supplementary analysis is required to determine the severity of the event.

The following is a list of the Thermal-Hydraulic initial conditions used for the accident events presented in this section;

Thermal Hydraulic Conditions

Thermal Power (MWT)	975**
Primary Flow (LB/Sec/Loop)	3842
Primary Hot Leg Temperature (°F)	1015*
Primary Cold Leg Temperature (°F)	750*
Intermediate Flow (LB/Sec/Loop)	3555
Intermediate Hot Leg Temperature (°F)	956*
Intermediate Cold Leg Temperature (°F)	671*
Hot Spot Clad Midwall Temperature (°F)	1365

*These values include an additional 20°F over their normal value to allow for instrument error and control dead band allowance.
**Power uncertainties are discussed in Section 4.4 for 3 loop operation.

Supplementing the above parameters, the following additional conservative assumptions and conditions were used for the analysis;

1. Maximum Decay Heat - The decay heat for the end-of-cycle condition corresponding to long term power operating history at full power was used. This included an added 25% conservative 2σ bias to cover uncertainties. The purpose was to provide maximum post-trip heat input to provide a conservatively high prediction of core maximum temperature and a conservative evaluation of heat input to the decay heat removal system.

2. Most rapid flow coastdown - The minimum vendor specified sodium coolant pump inertia and maximum system pressure drop are combined to generate a conservatively fast rate of flow reduction following a coolant pump trip. The purpose of this assumption is to provide a minimum prediction of net reactor coolant flow during the period from pump trip to the time of reaching pony motor flowrate. This results in minimum heat removal from the reactor during this period and hence a conservative maximum prediction of core temperature.

3. Full power thermal hydraulic design condition operating points - The full power thermal-hydraulic rated condition is at 975 MW reactor power. The thermal-hydraulic design operating temperatures have been conservatively increased by 20°F to allow for instrument error and the control dead band. The purpose of this assumption is to assure the most conservative prediction of severity for the events analyzed. The additional temperature bias for instrument error increases the conservatism of predicted reactor temperatures.

4. Shutdown Rod Worths with Maximum Worth Single Stuck Rod - The rod worth used to predict post trip negative reactivity insertions are the design expected values for the primary shutdown system control rods and the minimum expected values for the secondary shutdown system control rods (see Section
51| 15.1 for further details). For both sets of control rods, the single most reactive control rod is assumed to be stuck in the withdrawn position. The purpose of this assumption is to provide a realistic minimum prediction of shutdown reactivity and hence the slowest rate of power decrease. This provides a conservatively high prediction of reactor temperatures after shutdown.

5. A conservative 200 millisecond delay between the trip signal and the control rod insertion was used for these analyses. In Section 4.2.3 of the PSAR the requirement for the scram speed is that this delay be less than 100 milliseconds. The additional 100 plus millisecond delay over the required value results in higher clad temperatures and thus a worse condition.

6. Since the highest power fuel assembly and smallest Doppler coefficient occur at the beginning-of-equilibrium cycle (BOEC) the transients are analyzed for this particular worst period in core life. With burnup, the power generation and steady state temperature decrease (flows are constant) in the fuel assemblies and consequently, the temperatures due to the transients would decrease.

7. Three sigma (3σ) hot channel factors were used for all the analyses. The temperatures shown are at the midwall of the hot rod cladding at the highest temperature position both axially and circumferentially on the fuel rod (position is under wire-wrap).

In addition to the above conservative initial conditions assumed for the undercooling event analysis, additional conservatisms have been applied and are described under the specific cases presented. Also noted are those special cases for which the conservative assumptions stated above are not applicable.

The following is a Summary Table of the events considered in this section. Table 15.3-1 identifies; 1) the event, 2) the maximum midwall clad temperature resulting from a primary or secondary scram, and 3) comment on the severity of the event.

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TABLE 15.3-1

UNDERCOOLING EVENTS

Section No.	Event	Max. Clad Temp.* Primary Scram	Max. Clad Temp.* Secondary Scram	Comments
15.3	Undercooling Design Events			
15.3.1	Anticipated Events			
15.3.1.1	Loss of off-site electrical power	1410°F	1630°F	Primary shutdown within upset umbrella. Temperature spike associated with secondary shutdown is considerably less severe than the umbrella transient (See Section 15.3.1.1)
15.3.1.2	Spurious primary pump trip	1390°F	1445°F	Within the umbrella
15.3.1.3	Spurious intermediate pump trip	<1365°F	<1365°F	Core sees only normal trip
15.3.1.4	Inadvertent closure of one evaporator or superheater module isolation valve	<1365°F	<1365°F	Core sees only normal trip
15.3.1.5	Turbine trip	<1365°F	<1365°F	Temperature decreasing continuously
15.3.1.6	Loss of normal feedwater	<1365°F	<1365°F	Core sees only normal trip
15.3.1.7	Inadvertent actuation of the sodium/water reaction system	<1365°F	<1365°F	Core sees only normal trip
15.3.2	Unlikely events			
15.3.2.1	Single primary pump seizure	1400°F	1470°F	Within the umbrella

TABLE 15.3-1 (Continued)

Section No.	Event	Max. Clad Temp.* Primary Scram	Secondary Scram	Comments
15.3.2.2	Single intermedite loop pump seizure	<1365°F	<1365°F	Core sees only normal trip
15.3.2.3	Small water-to-sodium leaks in steam generator tubes	<1365°F	<1365°F	Core sees only normal trip
15.3.2.4	Failure of the steam bypass system	<1365°F	<1365°F	Core sees only normal trip
15.3.3	Extremely unlikely events			
15.3.3.1	Steam or feed-line pipe break	<1365°F	<1365°F	Core sees only normal trip
15.3.3.2	Loss of normal shutdown cooling system	<1365°F	<1365°F	Core sees only normal trip
15.3.3.3	Large sodium/water reaction	<1365°F	<1365°F	Core sees only normal trip
15.3.3.4	Primary heat transport system pipe leak	no effect	no effect	No effect on reactor core or primary system temperatures or pressures
15.3.3.5	Intermediate heat transport system pipe leak	no effect	no effect	Core temperatures would not increase

*Fuel pin cladding midwall temperature (under wire wrap)

15.3.1 Anticipated Events

15.3.1.1 Loss of Off-Site Electrical Power

15.3.1.1.1 Identification of Causes and Accident Description

The off-site power supply to the 13.8 KV buses is available from the generating switchyards and the reserve switchyard both of which are powered by outside sources as described in Chapter 8.0. Hence, the postulated loss of power would result only from simultaneous, multiple failures.

The loss of all off-site power trips all primary and intermediate sodium pumps, commencing a flow coastdown. It also initiates starting of the emergency diesel generators. Action of the Plant Protection System (PPS) trips the control rods thus limiting core over temperatures from reduced flow. The emergency diesels provide power to the primary and intermediate sodium pump pony motors and SGAHRS Auxiliary Feedwater Pumps for decay heat removal. To provide conservatism in the analysis, the most rapid core flow coastdown was assumed by using the minimum pump rotating kinetic energy and the maximum primary system flow resistance specified in the design.

The action of the Primary and Secondary Shutdown Systems (SDS) are as follows:

a. Primary trip - Loss of electrical power trip occurring in 0.5 seconds. The 0.5 second delay includes measurement and trip function lags.

b. Secondary trip - Flux-Total Flow trip occurring 2 seconds after loss of electrical pumping power. This lag includes time for the flow to coastdown as well as the measurement lags.

15.3.1.1.2 Analysis of Effects and Consequences

The loss of off-site electrical power event was analyzed with the DEMO computer code. The overall results of the analysis are summarized in Figures 15.3.1.1-1 and 15.3.1.1-2. As shown, the Primary PPS loss of electrical power trip limits the maximum core hot spot temperature to 1410°F.

In the event the primary shutdown system does not operate, Figure 15.3.1.1-1 shows that the secondary shutdown system limits the worst case clad hot spot temperature to 1630°F. While the transient temperature exceeds the design basis emergency transient envelope temperature by 30°F, the time above the normal operating temperature is only 6 second as compared to 150 seconds for the design basis transient (see Figure 15.3.1.1-3). Consequently, the cladding damage due to the transient is less than that due to the design basis transient for which, as shown in Section 4.2, cladding integrity limits are satisfied.

The capability of the CDF procedure to conservatively predict the results of Fuel Clad Transient Test (FCTT) is demonstrated below. The range of the FCTT temperatures and fluences considered exceeded the data base of the FURFAN CDF computer code. Despite this, the CDF analyses conservatively predicted the test results with peak cladding temperatures in excess of 1900^{o}F, and cladding fluence exposures in excess of 3×10^{22} n/cm^2.

The quantitative criteria in terms of Temperature versus Time for transient events which do not affect cladding integrity is shown in Figure 4.2-31. The shape of the emergency transient considered in this plot envelopes the loss of off-site electrical power with scram by the secondary PPS event. The minimum cladding lifetime is determined by the intersection of the peak transient cladding temperature versus time curve and the transient limit curve with maximum design temperatures and maximum uncertainty in properties. Note that the maximum peak cladding temperature occurs at beginning-of-life, and the cladding temperature increment due to the transient is assumed constant throughout life. Thus, for an emergency transient with a maximum peak cladding temperature of 1630^{o}F, the peak clad temperature versus time curve would lie parallel to and 30^{o}F above the peak clad temperature versus time curve shown in Figure 15.3.1.1-4. The intersection of this curve with the minimum transient limit curve gives a cladding lifetime of 450 days or 35 days less than the 1600^{o}F peak cladding temperature transient. In all calculations involved in generating Figure 15.3.1.1-4, cumulative cladding damage is continuously accounted for in the cladding property considerations.

It should be noted that the anticipated time temperature curve for the loss of off-site electrical power is considerably less than the time envelope used to develope the transient limit curves. Therefore, the above loss of 35 days due to the additional 30^{o}F is believed to be an overestimate of the transients actual effect. This not withstanding, the design lifetime based on the above analysis for the loss of offsite power is still in excess of the 411 day goal lifetime.

As discussed earlier, the most realistically severe combination of possibilities allowed in the design specifications were selected to analyze this event. Figure 15.3.1.1-2 shows the effects of a possible longer flow coastdown, enhanced secondary control rod dynamics, and using "minimum required" instead of "expected" primary control rod shutdown rates. Lower possible core flow resistances and higher pump rotating kinetic energies, decrease the core hot spot temperature 10^{o}F for a primary PPS trip and 15^{o}F for the secondary shutdown system trip. Additionally, increasing the initial secondary control rod insertion rate to match the primary rates decreases the clad temperature 35^{o}F for the secondary trip.

The effect of using "minimum required" primary control rod shutdown rate values instead of the "expected" values (both having the highest worth rod assumed to be stuck), is also shown in Figure 15.3.1.1-2. As indicated
61 in Section 15.1 the core temperatures described in this chapter for the primary system have been based on the expected rates of shutdown worth which give the more realistic evaluation of the transient. The secondary rod insertion rates used are the minimum rates. Figure 15.3.1.1-2 shows the hot spot cladding temperatures for the two cases. As can be seen, there would be about a 10°F increase from using the minimum rates. Thus, using minimum instead of expected primary rod insertion rates does not significantly change the nature of or effects of the transient.

15.3.1.1.3 Conclusions

The loss of off-site electrical power results in a simultaneous loss of sodium pump power and the consequent reduction in core flow. The primary shutdown system limits the clad midwall hotspot temperature to 1410°F. In the unlikely event that the primary shutdown system does not operate, the secondary shutdown system limits the hot spot midwall clad temperature to 1630°F. This is an acceptable result because analysis of
52 the transient has shown that the cladding damage (cumulative damage function)
51 does not exceed the limit for an emergency event.

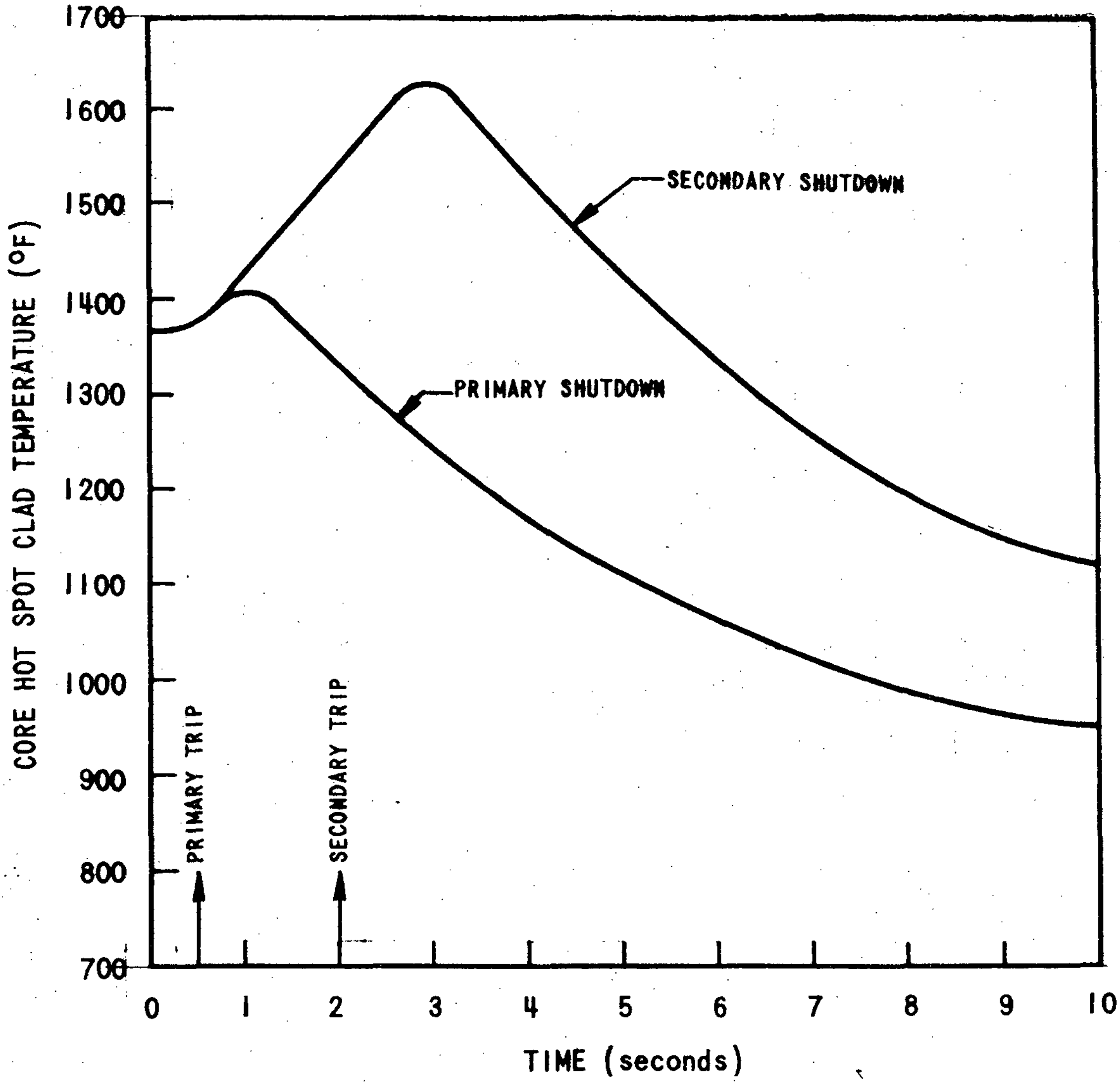

Figure 15.3.1.1-1. Core Hot Spot Clad Temperature Vs. Time After Loss of Off-Site Electrical Power

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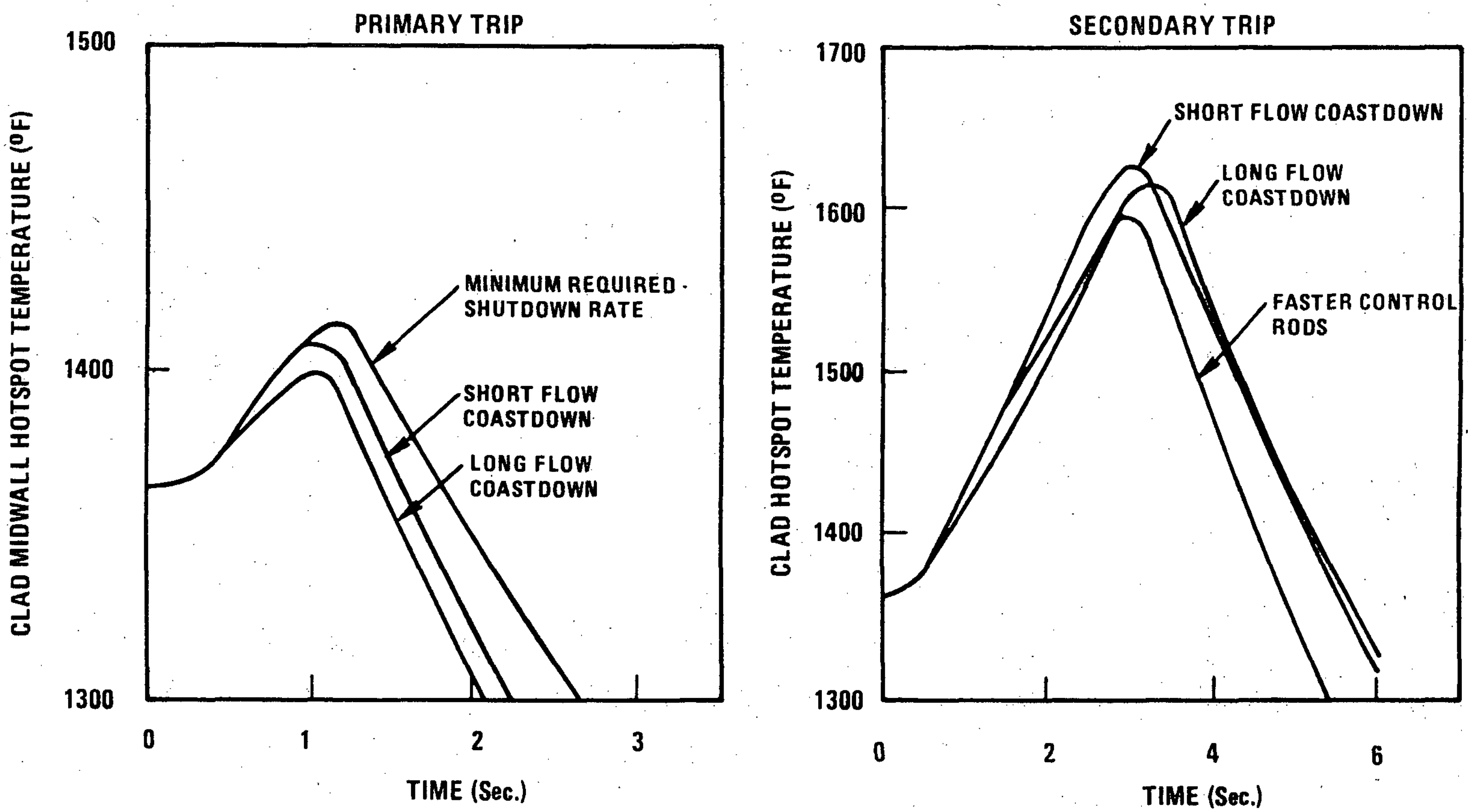

Figure 15.3.1.1-2. Clad Midwall Hotspot Temperature As a Function of Time After Loss of Off-Site Electrical Power for Varying Pump Coastdowns and and Control Rod Insertions

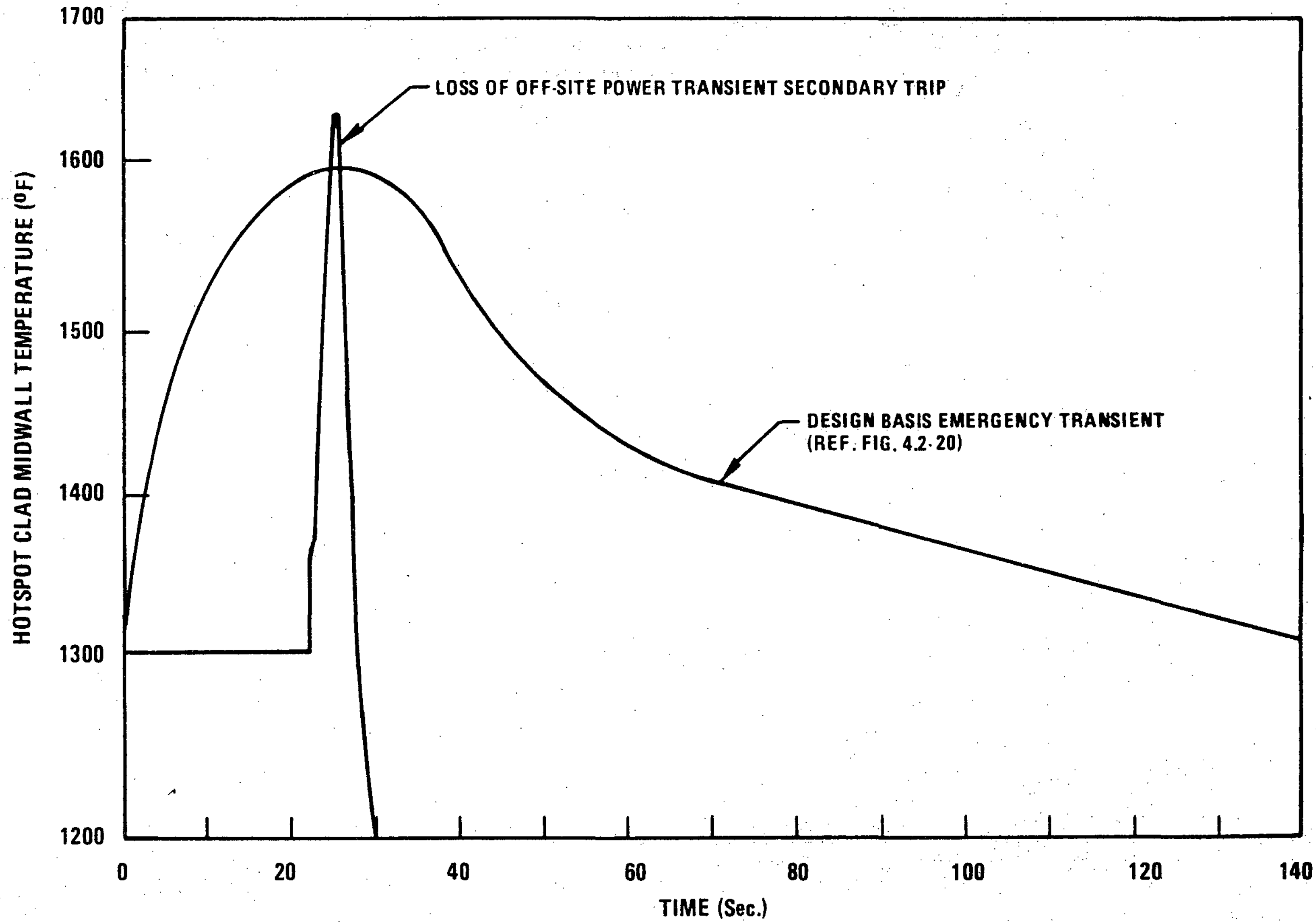

Figure 15.3.1.1-3. Comparison of Design Basis Emergency Transient and Loss of Off-Site Power Transient

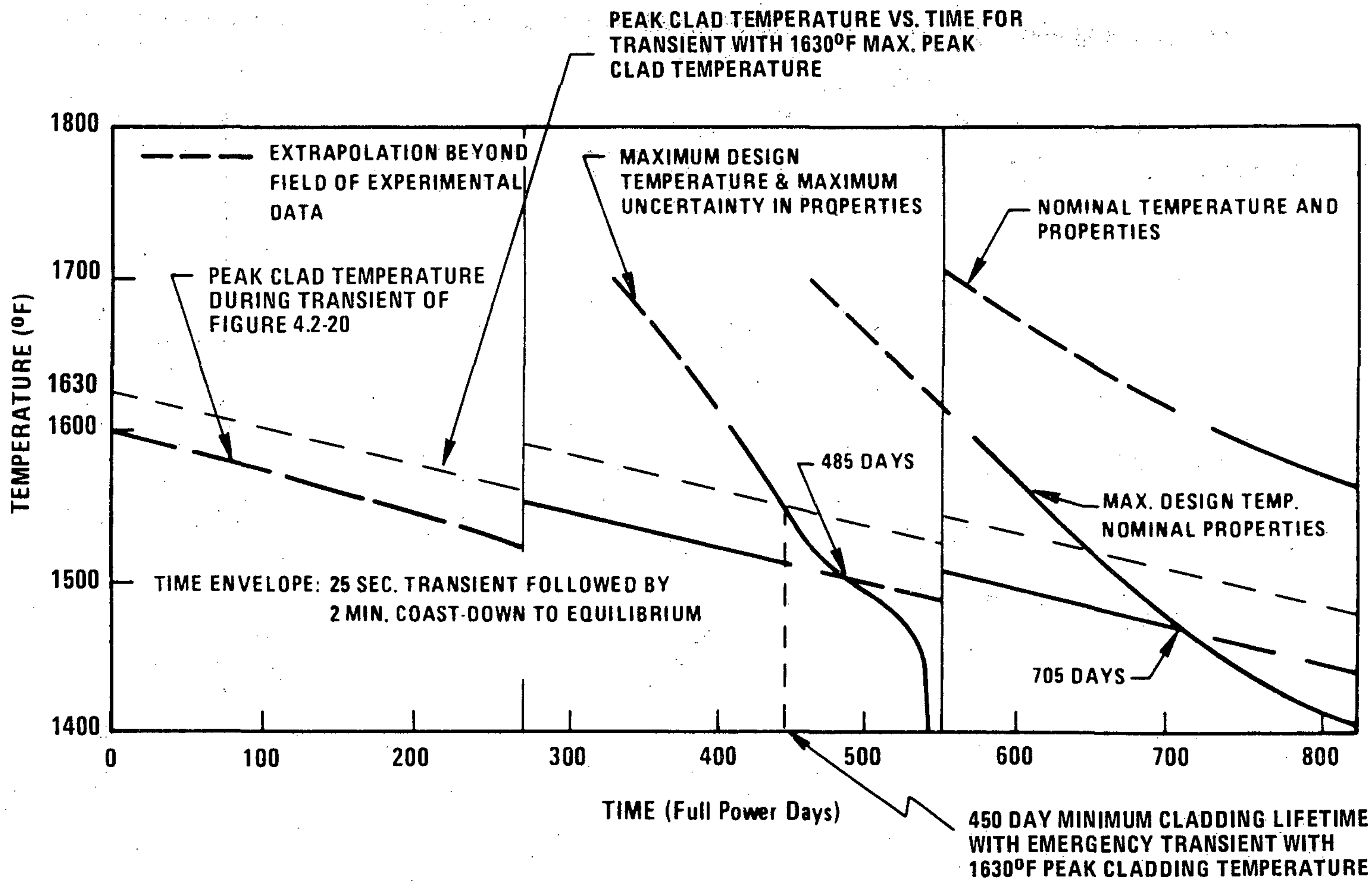

Figure 15.3.1.1-4 - Example of Use of Loss of Flow Transient Limit Curve Top of Fuel Stack Hot Fuel-Pin in CRBRP Assembly 6 Peak Cladding Temperature to Cause Failure During Transient

15.3.1.2 Spurious Primary Pump Trip

15.3.1.2.1 Identification of Causes and Accident Description

A spurious trip of a primary sodium pump may occur in one of two forms:

a. An A.C. bus fault in which both the primary loop sodium pump and the intermediate loop sodium pump on the same loop are tripped simultaneously, or

b. A malfunction only in the single primary pump or its circuits causing only that pump to trip.

In the first case, the trip function for the primary Plant Protection System (PPS) is based on the pump electrics (electrical feeder undervoltage coil or equivalent). For the second case, the measured primary pump speed to intermediate pump speed is used as the primary PPS trip. In both cases, the measured primary flow to intermediate flow ratio can operate the secondary PPS trip.

Results of the pump trip transient are affected by the rate of flow coastdown after the pump trip. Hence, they depend on the rotating kinetic energy of the pump and motor and on the loop flow resistance. Pump inertias corresponding to the range of rotating kinetic energies specified for pump design, along with a range of core pressure drops, have been examined to establish the most limiting cases.

The actions of the primary and secondary shutdown systems in the pump trip event are summarized as follows:

a. Bus fault (both primary and intermedite pumps tripped)

   Primary Shutdown System - pump electrics
   Secondary Shutdown System - flow ratio

b. Primary pump trip

   Primary Shutdown System - speed ratio
   Secondary Shutdown System - flow ratio

15.3.1.2.2 Analysis of Effects and Consequences

The DEMO code was used for analysis. The results of the most limiting cases for both the bus and pump faults are shown in Figure 15.3.1.2-1. As can be seen in the figure, the pump fault causes a more severe transient than the bus fault when the primary shutdown system operates. The reason for this is that the pump electrics trip which acts for a bus fault, occurs very rapidly compared to the speed ratio trip which acts for the pump fault. The maximum hot spot midwall clad temperature with a primary trip for the more severe event is 1390°F, 60°F below the guideline limit for an anticipated fault.

The secondary shutdown system utilizes the flow ratio trip for both a bus and pump fault. Thus, the bus fault, which results in a flow coastdown of both primary and intermediate loops, produces the more limiting transient. The secondary bus fault curve shown in Figure 15.3.1.2-1 corresponds to primary and secondary pump kinetic energies which more nearly match the primary and secondary flow coastdowns. This produces the longest delay of the flow ratio trip. The maximum hot spot midwall clad temperature with the secondary trip is 1430°F, 170°F below the guideline limit for an unlikely event.

The protective action and consequences of a sequential loss of flow are bounded by the cases of 1) Loss of all primary pumps (loss of offsite power Section 15.3.1.1) and 2) Trip of 1 primary pump (Section 15.3.1.2).

As discussed in Section 15.3.1, protection for this combined class of loss of flow accidents is provided by the following trip functions:

Primary Shutdown System
- Primary Pump Electrics
- Flux-Pressure
- Primary to Intermediate Speed Ratio

Secondary Shutdown System
- Flux-Total Flow
- Primary to Intermediate Flow Ratio

A study of the sequential loss of flow has been conducted. In this study the sodium pumps in one HTS loop were tripped at time t=0. The remaining pumps were tripped at a later time T. The delay T was varied from zero (identical to the loss of offsite power in 15.3.1.1) to 2.6 seconds (identical to the single primary pump trip in this section. A limiting analysis was done assuming only the secondary shutdown system. The intermediate pumps were tripped with the corresponding primary pump. This delayed the flow ratio trip producing the more limiting case.

The results of the analysis are shown in Figure 15.3.1.2-2. For a sequential loss of the remaining pumps from zero to 1.25 sec ($0<T<1.25$) the flux-flow function provides the trip. The peak temperatures are the same as for the loss of offsite power case. For $1.25<T<2.6$ sec. the primary to intermediate flow function trips the Ø-flow function. This results in lower maximum temperatures for larger T. Since the flow ratio function provides shutdown at 2.6 sec. for 1 loop loss of flow, a sequential pump trip at $T>2.6$ sec. is not meaningful.

It should be added that in any case a reactor trip occurs while substantial flow exists in the primary loop with the lowest flow. Consequently, without considering the check valve dashpot and acoustic filtering in the IHX and reactor vessel, there is no opportunity for 25 a check valve slam to cause a hydraulic disturbance which could initiate the trip of other pumps.

In summary, the consequences of sequential loss of flow are bounded by loss of offsite power event (all pumps tripped simultaneously) and the trip of 1 primary pump. 25

15.3.1.2.3 Conclusions

Action by either the primary or secondary shutdown systems is sufficient to prevent excessive core temperature and protect the reactor in case of the spurious trip of a primary sodium coolant pump. Either system shuts the reactor down early enough to terminate the initial core temperature rise within the limits identified for an operational incident, i.e., no loss of fuel lifetime.

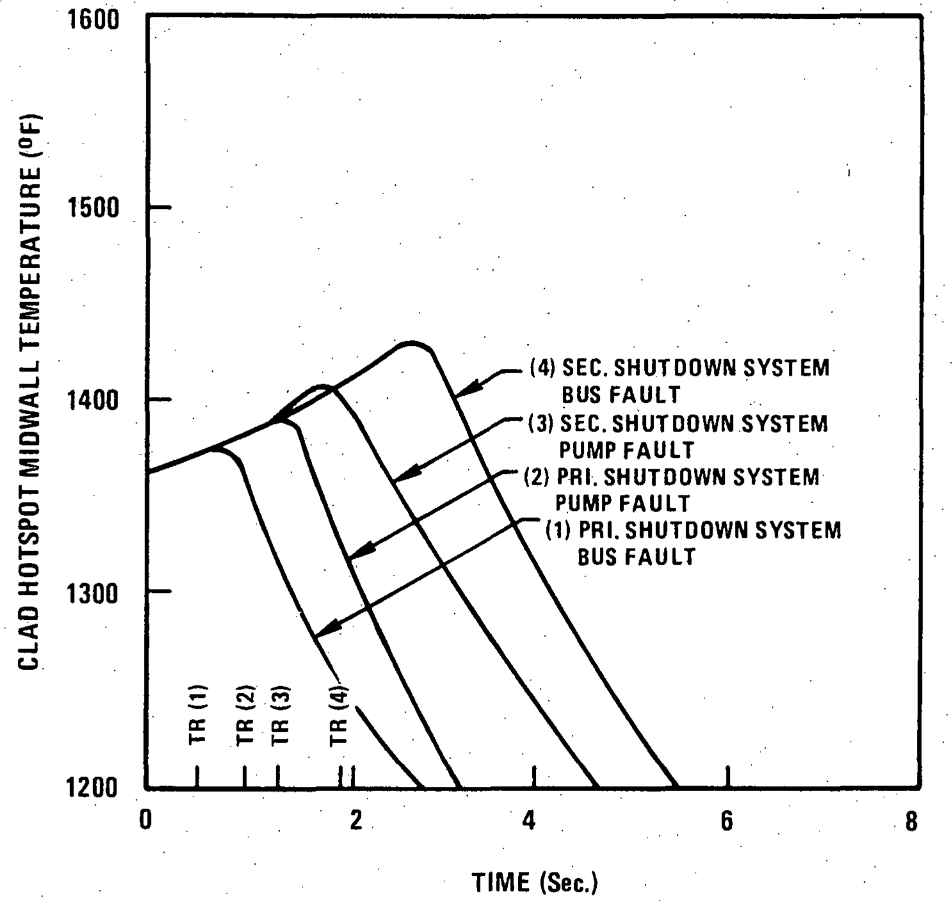

Figure 15.3.1.2-1. Hotspot Clad Midwall Temperature as a Function of Time After Pump Trip

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15.3.1.3 Spurious Intermediate Pump Trip

15.3.1.3.1 Identification of Causes and Accident Description

As described under the primary pump trip, Section 15.3.1.2, trip of an intermediate sodium coolant pump might result from either a pump bus fault or a fault in the pump drive itself. The former case would also trip the primary pump and has been covered in Section 15.3.1.2.

For a spurious trip of the intermediate pump caused by a pump fault, the intermediate flow will coast down to that supported by the pump pony motor. This will reduce heat removal from the intermediate side of the IHX. However, a reactor trip signal generated by either the primary or secondary Shutdown System will shut down the reactor before any temperature change can be transported back to the reactor. Consequently, the event as it occurs at the reactor and in the heat transport system is similar to a conventional plant trip except for the affected loop IHX, primary cold leg piping and steam generators. For these components, the resulting transient is within the design basis umbrella. 8

The following protective functions apply for the intermediate pump trip:

a. Primary - Primary to Intermediate pump speed ratio
b. Secondary - Primary to Intermediate flow ratio trip.

15.3.1.3.2 Analysis of Effects and Consequences

The pump trip analysis was conducted using the DEMO Code. Clad hot spot temperature resulting from the analysis is presented in Figure 15.3.1.3-1. For shutdown either by the primary or secondary PPS trip functions, no increase in core temperature is produced. The heat transport system experiences normal trip temperature. 8

15.3.1.3.3 Conclusions

In the event of a spurious trip of a single intermediate sodium coolant pump, either primary or secondary trips will occur rapidly. These trips will shutdown the reactor before the plant temperature changes can be transmitted back to the core. Hence, the resulting core and heat transport system temperature transients are similar to those resulting from a conventional plant trip.

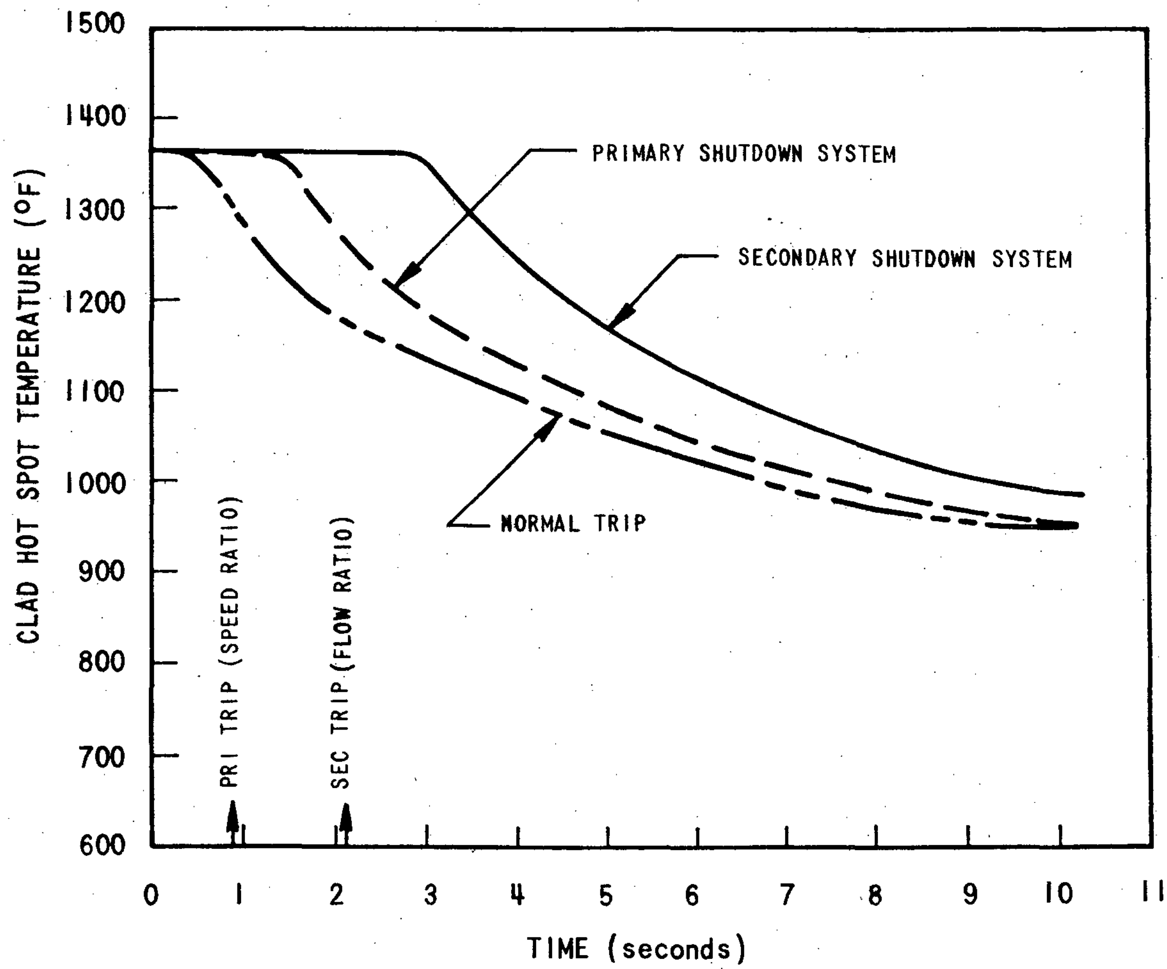

Figure 15.3.1.3-1. Cladding Midwall Hotspot Temperature Vs. Time After Spurious Intermediate Pump Fault Trip

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15.3.1.4 Inadvertent Closure of One Evaporator or Superheater Module Isolation Valve

15.3.1.4.1 Identification of Causes and Accident Description

The water side of each superheater module is provided with an inlet and outlet isolation valve and each evaporator with an inlet isolation valve. It is possible that a spurious signal or an operator error will close one of these during normal operation. The consequences will then depend on which of the valves is closed, as summarized in the following table:

TABLE 15.3.1.4-1

POTENTIAL EVENTS

Valve Closed	Result
Superheater exit	Loop steam flow to header ceases. Steam is relieved through superheater exit safety valve. Reactor and plant trip should follow from steam/feed flow mismatch.
Superheater inlet	Loop steam flow to header ceases. Superheater dries out and its sodium exit temperature increases. Steam is relieved through drum safety valves. Reactor and plant trip should follow from steam/feed flow mismatch.
Evaporator inlet	Inflow will cease to the affected evaporator module and it will dry out (with reversed flow into its exit prevented by the exit check valve). Its sodium exit temperature will rise. Water inflow to the remaining module in the loop will increase and tend to increase its load. Its sodium exit temperature will decrease. The net sodium exit temperature will be the mixed stream from the two modules but will increase on dryout of the affected module. The effective evaporator surface decrease will lower the drum pressure and reduce the steam flow to the header in the affected loop. A reactor and plant trip should follow from PPS action (high IHTS cold leg temperature trip at 720°F) in approximately 10 seconds.

The action of the Plant Protection System (PPS) in the above events is as follows:

Primary Trip - Steam-feedwater flow ratio
High IHX primary outlet temperature

Secondary Trip - High evaporator outlet temperature

In all of the above cases, the reactor is shutdown by PPS action before the steam-generator originated temperature transient reaches the reactor inlet and no core fuel or clad safety problem is created. Post-shutdown decay heat removal will be available through the unaffected loops (with some heat removal capability remaining in even the affected loop).

15.3.1.4.2 Analysis of Effects and Consequences

The reactor is assumed to be operating at rated conditions when one of the isolation valves in one steam generator water/steam side is inadvertently closed. The reactor is scrammed by one of the trips discussed in the previous section. After the reactor scrams, the core flow rate and the resultant core temperature are those for a normal scram for the first several minutes until the temperature transient due to the hot sodium from the affected steam generator module reaches the core.

For the limiting case of an evaporator module inlet isolation valve failure (maximum sodium temperature increase), the maximum hot channel core exit coolant temperature that occurs when the hot sodium reaches the core is more than 400°F below the normal operating temperature. If the evaporator sodium exit temperature trip is assumed to be inoperable, the reactor would be shutdown by the primary cold leg high temperature trip. The resulting core outlet coolant temperature in the hot channel would be about 40°F above the normal operating temperature.

The core coolant temperature transient for the inadvertent closure of a superheater isolation valve is less severe than that discussed above for the evaporator assuming a reactor trip on the steam-feedwater flow ratio trip. If the steam-feedwater flow ratio trip fails to shutdown the reactor following a superheater isolation valve closure, the increase in evaporator sodium exit temperature would not cause an immediate trip on either intermediate or primary high cold leg temperature. If it is conservatively assumed that the operator does not manually scram the reactor, the reactor would be scrammed by a delayed high temperature trip at 750°F in the intermediate cold leg after one or more transit times around the loop (transit time is about 2 minutes at rated flow). The increase in core inlet temperature would be less than 30°F (one-third of the increase in primary cold leg temperature at the time a trip on high intermediate cold leg temperature occurs). The resulting increase in core temperatures above normal operating temperatures would be less than 30°F.

15.3.1.4.3 Conclusion

Core temperatures following inadvertent closure of a steam generator module isolation valve are initially the same as a normal trip assuming a reactor shutdown from first level trips. Perturbations in core temperatures during the trip transient from these events are insignificant. These events are included in the overall plant duty cycle list that provides the basis for the thermal transient design conditions for the reactor and the main heat transport system.

Core temperatures for this failure assuming a reactor shutdown on second level trips remain within the limits discussed in Section 15.1.2 for an anticipated event with failure of the first level trip.

15.3.1.5 Turbine Trip

15.3.1.5.1 Identification of Causes and Accident Description

A turbine trip can be initiated from a number of causes in the steam plant and/or electrical generating system; for example, a mechanical failure in the large rotating components, loss of lube oil pressure, manual trip, or simply a spurious turbine trip signal.

In the normal case where a reactor trip does not follow from the turbine trip, the heat removal from the steam plant becomes less than the heat generated in the reactor, and a potential reactor undercooling event results.

To accommodate this and other load loss events, the plant design incorporates an 80% steam bypass capability (at rated pressure) vested in four turbine bypass valves.

The following actions occur in the turbine trip transient:

a. The turbine trip initiates immediate closing of the turbine inlet valve.

b. The turbine trip initiates opening of the steam bypass. The dump becomes fully open at 3 seconds after the trip. (In the analysis, appropriate lags are included in the steam bypass to represent dump controller and sensor lags and valve actuation time).

c. The normal reactor control system will initiate a reduction in reactor power at 3% per minute in response to the change in steam demand.

The above actions will bring the plant to a new operating point with load based on steam bypass. There will be sufficient time in this operating mode to bring the reactor to a safe and controlled shutdown if desired.

15.3.1.5.2 Analysis of Effects and Consequences

The analysis was performed using the DEMO code. Figure 15.3.1.5-1 shows the results of this analysis. As shown in the figure, no significant clad temperature transient results. Also shown in the figure are the reactor vessel and core inlet temperature transients. These transients show that there are no significant inlet temperature transients imposed on the reactor core as a result of the turbine trip without reactor trip.

Two hundred seconds after the turbine trip occurs, the reactor power has decreased to 90% and is decreasing at 3% per minute. This power reduction rate will continue to a new operating point with load based on steam bypass where sufficient time will exist to bring the reactor to a safe and controlled shutdown when desired.

The blanketing event concerning the radiological consequences of the accidents involving turbine trip involves assuming a complete release of the activity associated with the steam-water system. A complete, simultaneous release to the atmosphere of the activity associated with the deaerator, condenser hotwell, condensate and feedwater piping, condensate storage tank and steam generator loops has been assumed.

The radiological consequences of this postulated release were based on the presence of the maximum calculated tritum in the evaporated water at an activity of 0.25 $\mu Ci/gm$. The assumed release results in a conservatively calculated 2 hour site boundary dose of 100 mrem - approximately 99 mrem associated with a whole body dose and 1 mrem due to beta skin dose. The associated dose is less than 1% of the requirements of 10CFR100. 25

15.3.1.5.3 Conclusions

As shown in the above analysis, the turbine trip without reactor trip does not produce any significant transients within the reactor. The reactor power will be reduced in a controlled manner to a new operating point based on steam dump from which the reactor can be shut down if desired.

Radiological consequences of any transients and accidents producing a turbine trip with and without the assumption of a loss of offsite power would be within the results reported and well within 10CFR100 limits. 25

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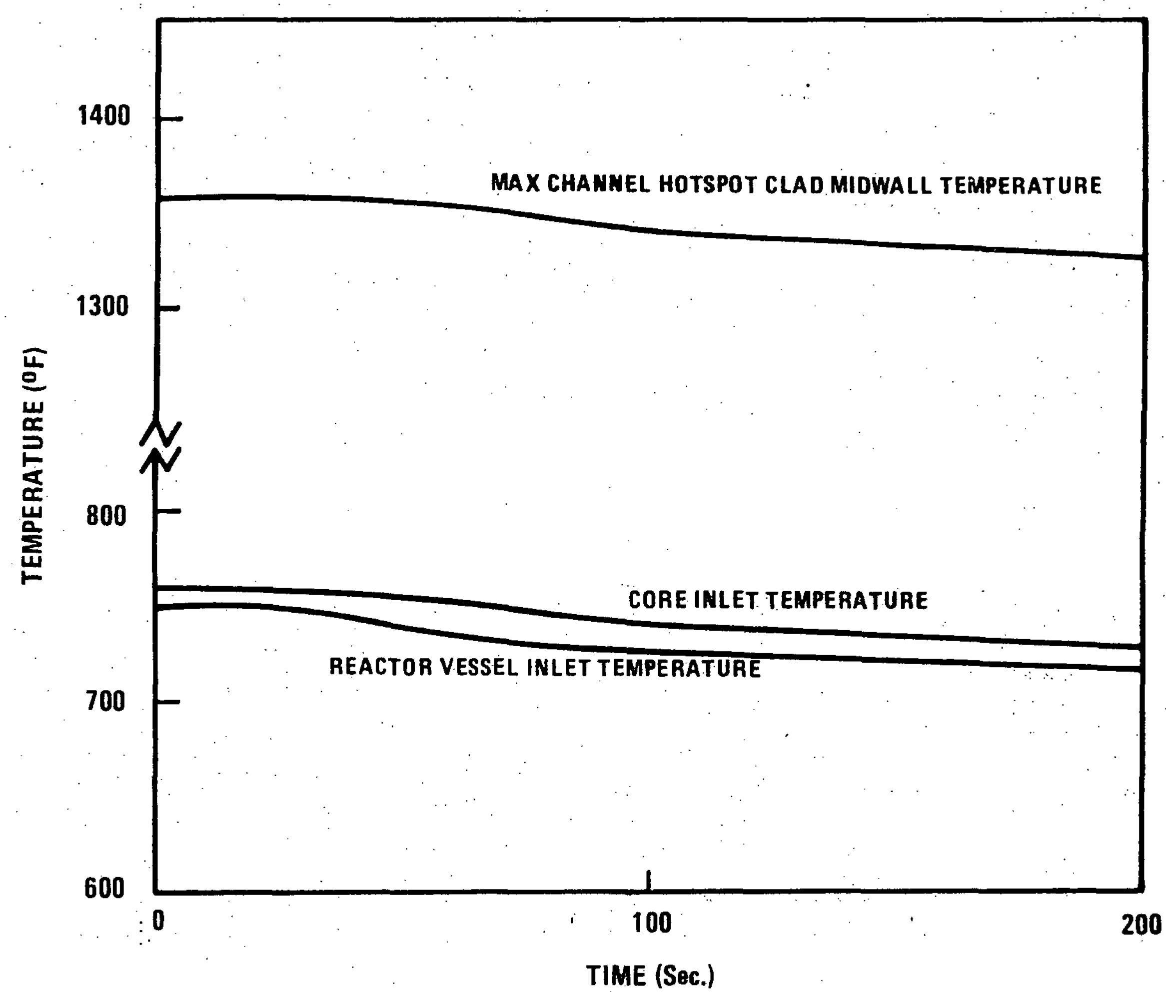

Figure 15.3.1.5-1. Turbine Trip

15.3.1.6 Loss of Normal Feedwater

15.3.1.6.1 Identification of Causes

Loss of normal feedwater flow could occur in either a single loop (e.g., malfunctions of a feedwater control or isolation valve in one loop) or in all three loops. Loss of normal feedwater to all loops would result from failure of one of the feedwater pumps or its power supply train. In this section, total loss of all feedwater is conservatively considered as a bounding case for this event. Upon loss of normal feedwater flow, the
49 reactor is tripped by the steam-feedwater flow ratio trip and SGAHRS is activated. Since the event is initiated by a loss of the feedwater pumps, the steam bypass system is not available and the steam line pressure will increase rapidly, causing the safety relief valves at the superheater exit to open. Since loss of off-site power de-activates the main circulating water pumps and the main condenser, bypassing of dump steam to the condenser is not considered. Upon SGAHRS initiation, the SGAHRS vent valves are opened. As system pressures decline, the safety relief valves close and the SGAHRS vent valve controls the steam drum pressure. The auxiliary feedwater from SGAHRS supplies make-up water to maintain steam drum level during the portion of the transient when steam is being vented to the atmosphere. The low steam drum level trip provides a backup SGAHRS activation signal to the steam-feedwater flow ratio trip.

The protected air-cooled condensers (PACC's) are activated by the same signal which activates the auxiliary feedwater pumps and the PACC's continue to remove a portion of the heat load delivered to the steam generator system by the intermediate system. When that heat load (the sum of both reactor decay heat and plant stored heat) decreases to the level at which the PACC's can fully dissipate it, the SGAHRS vent valves close, steam venting ceases and auxiliary feedwater control valves are closed. The PACC's continue to remove reactor decay heat and plant stored heat in
17 a closed loop fashion for as long as required.

15.3.1.6.2 Analysis of Effects and Consequences

The reactor is assumed to be operating at 115% of rated power when the normal feedwater supply is lost to all three loops. The reactor is scrammed by the steam-feedwater flow ratio trip. It is convenient to separate this event into short-term and long-term effects when considering its consequences. During the initial part of the transient, the main area of concern is the thermal transient experienced by the core and the main coolant system. For longer times, the adequacy of the SGAHRS to provide the required cooling is the most important consideration.

The thermal transient experienced by the core for either the first or
second level trips described above is essentially the same as for normal plant
trip. The only difference between the effects of the two incidents will be
that the loss-of-feedwater case will produce lower evaporator Na outlet tem-
peratures than the normal trip, due to the injection of a greater quantity of
highly-subcooled water from the protected water supply into the drum. This
7 effect will eventually be seen at the core as a lower inlet temperature than
for the normal trip case.

Pony motors are assumed to be operating at conservatively high flows for this incident (primary flow at 10% and intermediate flow at 12% of design value).* To evaluate the adequacy of SGAHRS for this incident, the analysis discussed in Section 5.6.1.3.9 was repeated for the same initial conditions.**

Results are similar to those shown in Figures 5.6-1 through 5.6-3 with the following differences:

- The peak heat load on SGAHRS for this case is significantly less than that for the loss-of-power case (Section 5.6.1.3.9):

  Loss-of-feedwater . . . . 16.0% of rated

  Loss-of-power . . . . 22.8% of rated

- The lower heat load produces a smaller peak AFW flow requirement and corresponds to a higher minimum Na temperature:

	Peak AFW (lb/sec/loop)	H_2O vented (ft^3)	Min. Evap. Na Outlet (°F)
Loss-of-feedwater . . .	35.8	2790	579
Loss-of-power	51.0	3130	425

The design basis event for sizing of the PWST (see Section 5.6.1.3.9) provides a tank with 9591 ft^3 which includes a volume of 5580 ft^3 for venting. Since this is considerably more than the venting requirements of the transient discussed above a large margin in the protected water supply is available for this event.

15.3.1.6.3 Conclusion

Core temperature following the loss of the normal feedwater supply is equivalent to the temperature following a normal plant trip, and large margins are available in the active systems required for auxiliary cooling. The reactor and heat transport systems are designed to accommodate this event.

* Recirculation water flow is assumed to be maintained at 100% of initial flow, since power remains available.

** "Stretch" conditions (115% power, high temperature) are conservative for evaluation of SGAHRS performance.

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15.3-24

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15.3.1.7 Inadvertent Actuation of the Sodium-Water Reaction Pressure Relief System

15.3.1.7.1 Identification of Causes and Accident Description

The Sodium-Water Reaction Pressure Relief System is designed to minimize the consequences of a large sodium-water reaction by separating the reacting components as quickly as possible and removing them and their reaction products from the system. This is accomplished by dumping the water/steam side of the affected steam generator to a water dump system, while draining the intermediate loop sodium to a sodium receiving tank. Rupture discs are employed at each module on the sodium side of the system which are broken by any overpressure resulting from a large sodium-water reaction. On the water/steam side, isolation valves are provided at the inlet of each evaporator and at the inlet and outlet of each superheater module. In addition, power-operated dump valves are connected inside the isolation valves on the modules. The dump valves are located at the inlet and outlet of each evaporator module and on the outlet of the superheater module.

A spurious actuation of the water/steam side of this system, without the corresponding dump of the intermediate sodium, immediately stops heat removal on the steam side of the affected loop but leaves the sodium circulating to transport the resulting temperature transient back to the cold leg components, IHX and reactor inlet. The following sequence of actions would occur:

a. The Sodium-Water Reaction Pressure Relief System actuation signal initiates closing of water/steam isolation valves on each steam generator module and opening of water/steam dump valves at the evaporator inlet and power relief valves at the evaporator and superheater outlets.

b. The superheater is dried out and depressurized very rapidly. Heat transfer falls off and the superheater sodium exit temperature begins to increase. This increase is subsequently transported to the evaporator sodium inlets.

c. The contents of the evaporators are dumped within 15 seconds or less. The initial liquid content of the evaporators is largely dumped; however, some flashing into steam takes place, removing additional heat from the sodium. The protection system action trips the reactor on steam feed flow mismatch plant; hence, intermediate sodium pumps are coasting down. The combination of reduced sodium flow and heat removed by the steam initially causes the sodium temperature leaving the evaporators to decrease slightly.

d. When the evaporator dryout occurs, heat removal is greatly decreased and the evaporator sodium exit temperature begins to increase toward the hot leg temperature.

The action of the Plant Protection System in this event is the following:

a. Primary - Steam flow-feed-flow-mismatch trip.

b. Secondary - Intermediate sodium cold leg temperature trip.

Either of the above trips will cause a reactor shutdown well before the temperature transient resulting from the water/steam isolation and dump can be transported back to the reactor inlet. Consequently, no reactor clad or fuel damage is involved with this event and the event is instead considered in the plant duty cycle for its thermal stress effects on the steam generator modules, IHX, cold leg components and reactor vessel inlet nozzles.

15.3.1.7.2 Analysis of Effects and Consequences

The analysis was performed with the DEMO Code. To assure a conservative analysis, the following have been applied:

a. The evaporator inlet dump valve area has been increased three times its design area. This increase also contributes to a conservatively fast sodium temperature transient and is expected to cover any uncertainties in future design changes in the dump system.

b. The transient is initiated from the 975 MWt full power thermal-hydraulic design operating condition.

c. Credit has been taken for only 75 percent of the tube metal heat storage in the heated sections of the units for its effect on slowing the temperature changes. No credit has been taken for heat storage in the shell or structure metal, nor for stagnant sodium or metal heat storage in the unheated sections. In addition to insure worst case transients on all cold leg components, no credit has been taken for the pipe mass or any mixing which may occur in the pipes.

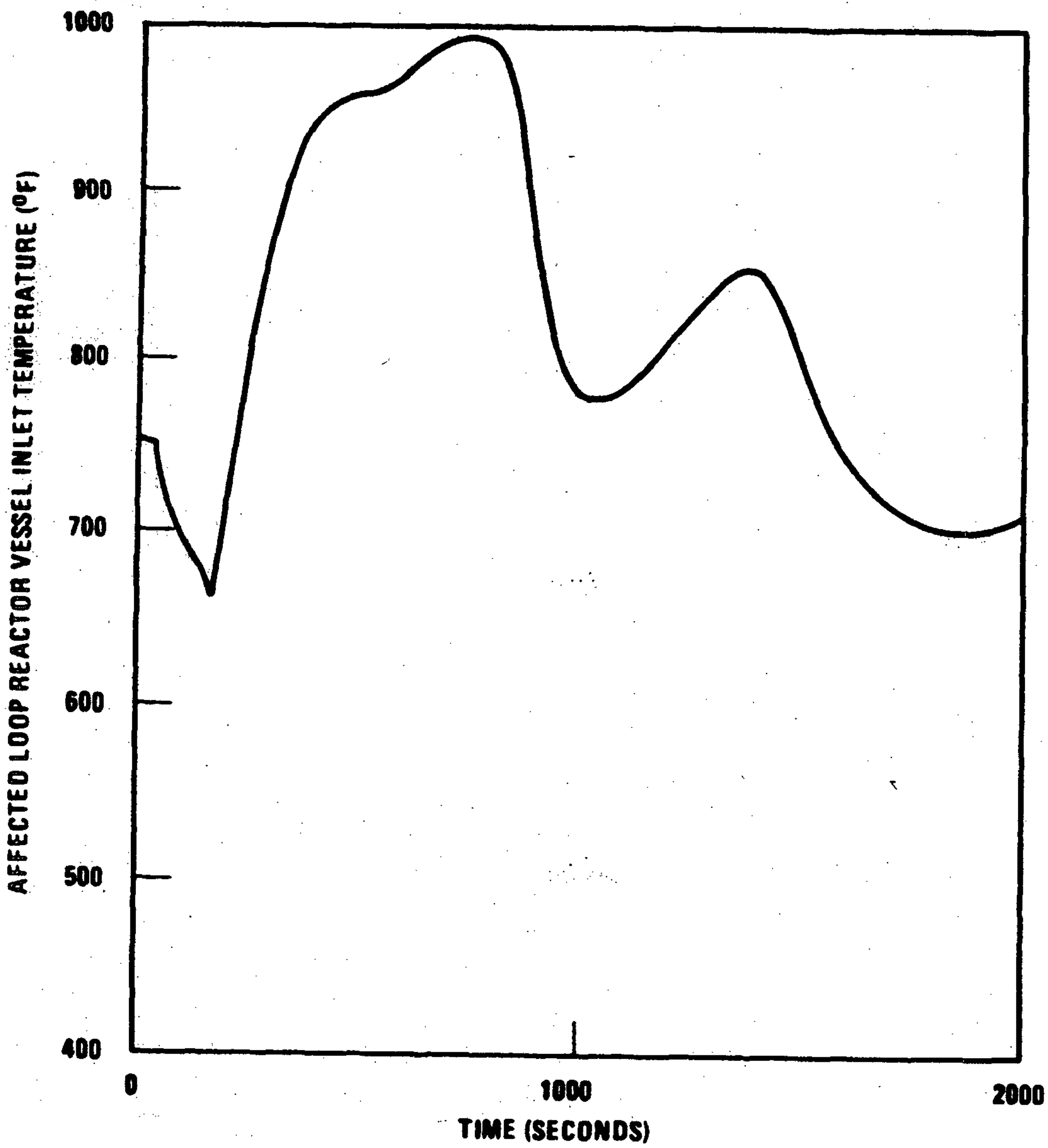
d. In the case shown, the pony motor was operated in the affected loop to produce conservatively fast temperature rates on the intermediate cold leg sodium temperature.

The isolation and dump causes the steam generator module pressures to fall rapidly to atmospheric (dump system back pressure is here assumed to be atmospheric to produce a conservatively fast blowdown). The drum has been isolated and its pressure remains high. The evaporator sodium exit temperature initially decreases, then rises when dryout occurs, eventually approaching hot leg temperature. The reactor, due to transport delays (more than 150 seconds at pony motor flow), does not immediately see the temperature changes, so that when the reactor trip occurs (less than 4 seconds), the transient at the reactor is the same initially as a conventional trip. The temperature transient originating at the evaporator outlet will subsequently be transported back to the IHX and then to the reactor inlet. By that time, however, the reactor has been shut down. The resulting reactor clad hot spot temperature is shown in Figure 15.3.1.7-1. The reactor inlet temperature begins increasing at about 190 seconds as a result of the cold leg temperature transient which occurs earlier at the evaporator outlet. The final reactor inlet temperature as a result of this transient will not jeopardize the core. This transient is included in the design bases of the plant.

15.3.1.7.3 Conclusions

The inadvertent dump of the water/steam side of a steam generator by spurious actuation of the Sodium-Water Reaction Pressure Relief System produces a large-magnitude temperature transient at the evaporator sodium outlet. The reactor, however, is shut down before any of the resulting temperature transient is transported to it. Since two unaffected loops remain available with pony motor flow, decay heat removal is available after shutdown.

Figure 15.3.1.7-1.a Temperatures of Pertinent Parameters as a Function of Time After Inadvertent Actuation of the Water/Steam Side of the Sodium/Water Reaction Pressure Relief System.



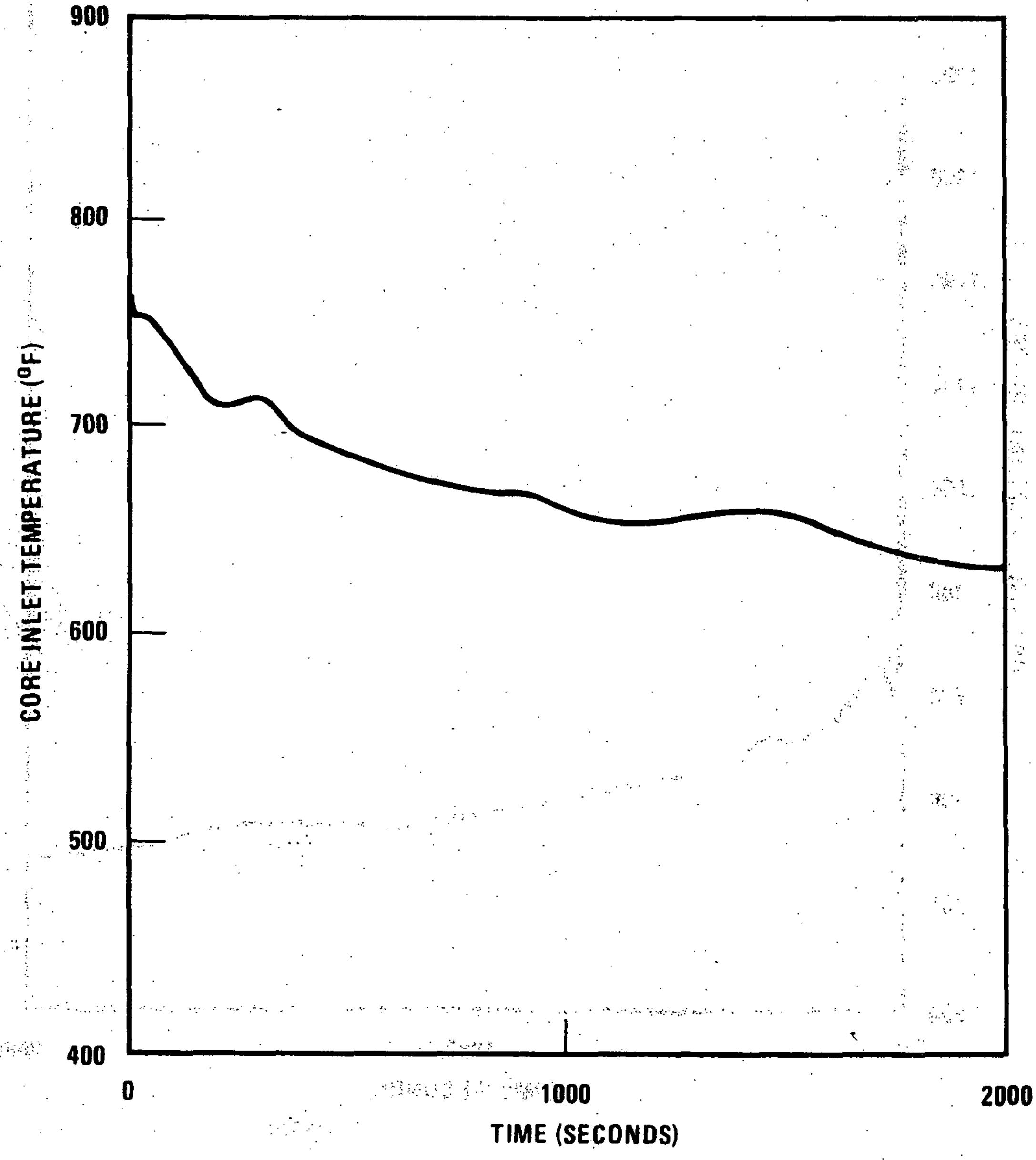


Figure 15.3.1.7-1B. Temperature of Core Inlet as a Function of Time After Inadvertant Actuation of the Water/Steam Side of the Sodium/Water Reactor Pressure Relief System

7203-17

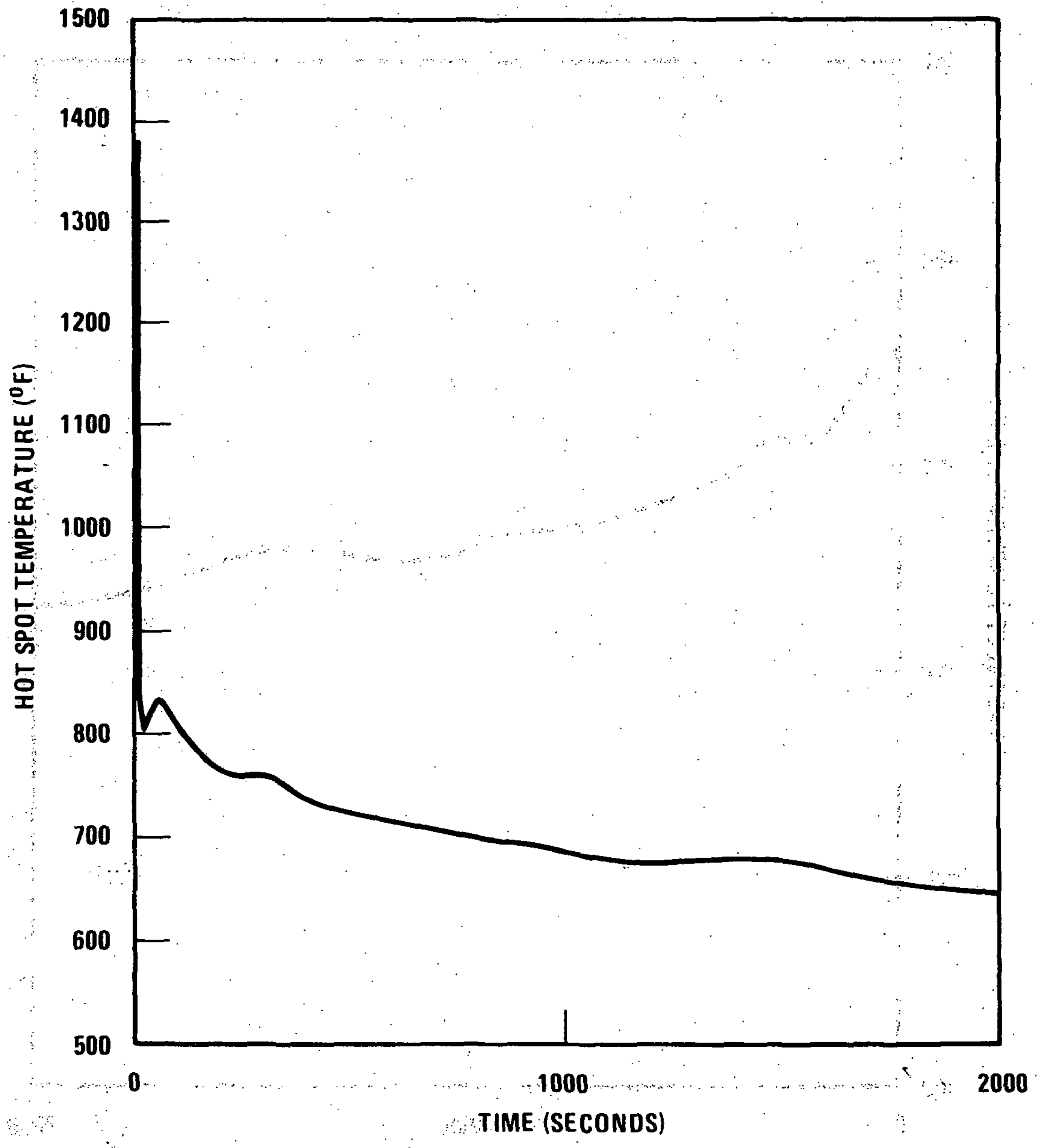


Figure 15.3.1.7-1C Temperature of Hot Spot as a Function of Time After Inadvertant Actuation of the Water/Steam Side of the Sodium/Water Reactor Pressure Relief System

7203-18

15.3.2 Unlikely Events

15.3.2.1 Single Primary Pump Seizure

15.3.2.1.1 Identification of Causes and Accident Description

The seizure of a primary sodium coolant pump is an unlikely event consequential to a mechanical failure in the pump, pump drive or pump motor. The resulting pump speed transient could occur at a rate that ranges from that similar to a normal pump coastdown to one that is nearly instantaneous, depending on the nature of the mechanical failure. This analysis, to assure conservative results, assumes that the pump impeller speed decreases instantly to zero.

A variation on the pump seizure transient is determined by whether or not the check valve closes in the affected loop after the pump seizure. Normally, the check valve will be expected to close. In the more unlikely event that it does not close, a flow reversal will occur in the affected loop, causing a larger decrease in reactor core flow.

The following Primary and Secondary Shutdown System (SDS) actions are applied in the pump seizure analysis:

a. Primary Trip

The basic primary trip for the seizure event is based on the measured speed ratio between primary and intermediate pumps. This trip provides a very rapid PPS actuation.

b. Secondary Trip

The secondary shutdown system trip for the pump seizure event is based on the measured flow ratio between the primary and intermediate coolant loops.

15.3.2.1.2 Analysis of Effects and Consequences

The analysis was performed using the DEMO Code. Cases with and without check valve failure were examined. Since it if not anticipated that the plant will be operated at steady state conditions in excess of 975 MWt (100% of power) the Nuclear Steam Supply System was analyzed at the peak (most adverse) power condition (100%) with appropriate margins for plus and minus error band and for the most pessimistic combination of pump capacities, pressure drops and heat transfer characteristics.

The West correlation, which follows is used for the CRBRP fuel assembly analysis:

$$Nu = 4 + 0.33\,(P/D)^{3.8}\,(Pe/100)^{0.86} + 0.16\,(P/D)^{5.0}$$

where:

Nu = Nusselt number
P/D = Pitch to diameter
Pe = Pelect number

This correlation and the basis for its selection are discussed in Reference 1. Undertainties in the heat transfer correlations are fully considered in the design through the use of hot channel factors. The uncertainty values are given and discussed in detail in Reference 2. 25

As shown in Figure 15.3.2.1-1 action of the primary shutdown system trip limits clad midwall temperature to $1400^{\circ}F$, $200^{\circ}F$ below the $1600^{\circ}F$ guideline for an unlikely event. In the event that the primary PPS trip fails, the secondary flow ratio trip limits the clad midwall temperature to $1470^{\circ}F$. This temperature is also within the guideline limit for maximum clad temperature. Both of these temperatures are for the case with check valve in the affected loop failed open. However, the trip times for the primary 0.05 sec.) and the secondary (0.65 sec.) shutdown systems are small enough that check valve failure does not affect the clad over temperatures.

The reactor inlet plenum acts as a large attenuator for any hydraulic shock waves entering the plenum from any of the piping loops. Our calculations show that less than two percent of any shock wave entering from one loop will be transmitted through the plenum to either of the other loops. The most severe shock wave conservatively postulated would be a check valve slam for which the shock magnitude is 50 psi. The impact on an unaffected loop would be less than 1 psi. The effect of such a small hydraulic perturbation on the operation of the remaining pumps is considered inconsequential. 16

15.3.2.1.3 Conclusions

Results for the primary seizure transient, even with instantaneous pump stoppage and check valve failure, show that both the Primary and Secondary shutdown systems prevent clad midwall hotspot temperatures from exceeding the guideline limits for an unlikely event.

Reference:

1. "Heat Transfer Correlation for Analysis of CRBRP Assemblies", WARD-D-0034, April, 1974.

2. "CRBRP Assemblies Hot Channel Factors Preliminary Analysis", WARD-D-0050, October, 1974. 25

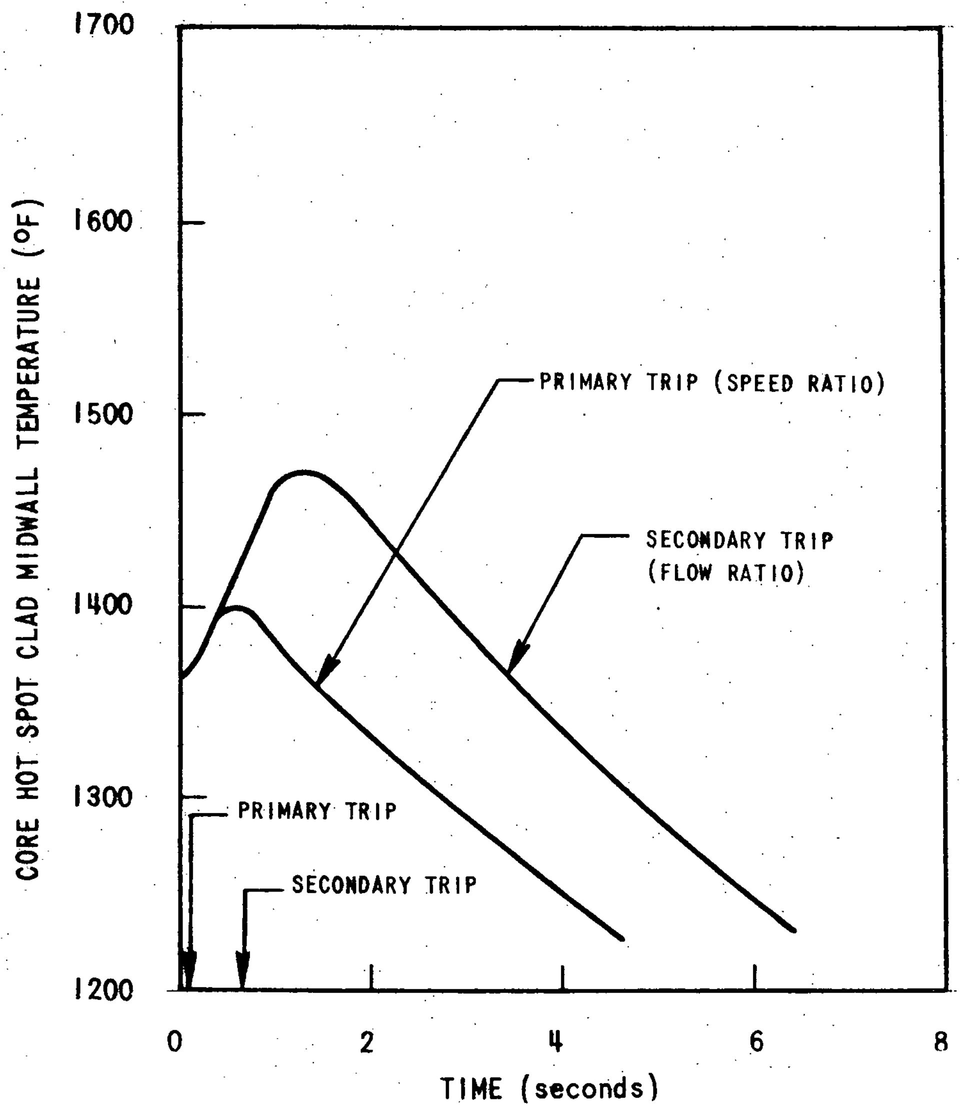


Figure 15.3.2.1-1. Core Hot Spot Clad Midwall Temperature Vs. Time After Primary Pump Seizure

6728-42

15.3.2.2 <u>Single Intermediate Loop Pump Seizure</u>

15.3.2.2.1 <u>Identification of Causes and Accident Description</u>

The seizure of an intermediate sodium coolant pump is an unlikely event; the potential causes are exactly analogous to the primary loop case in Section 15.3.2.1.

The results as seen by the reactor are less severe than the primary seizure since core flow is not directly affected and a relatively long time is required for temperature perturbations to be transmitted back to the reactor inlet. Flow in the intermediate heat transport system decreases to natural circulation. 8

The same shutdown system actions are applied in the intermediate pump seizure analysis as were applied for the primary pump seizure.

a. Primary trip - The speed ratio trip which is based on the measured speed ratio between primary and intermediate pumps in each loop.

b. Secondary trip - the flow ratio trip which is based on the ratio of measured primary to intermediate flow in each loop.

15.3.2.2.2 <u>Analysis of Effects and Consequences</u>

The intermediate pump seizure analysis was performed using the DEMO Code.

Results presented in Figure 15.3.2.2-1 for the maximum hotspot clad midwall temperature show that for actuation of either the primary or secondary trip, the core temperature does not exceed its initial steady-state value. The intermediate pump seizure, however, does result in a rapid decrease in the heat removed from the IHX (and delivered to the steam generator) in the affected intermediate loop. Consequently, the IHX outlet temperature rises on the primary loop side in the affected loop. This rise in temperature is inconsequential to the reactor core, since the reactor is shut down before the change at the IHX is transported back to the reactor inlet.

The other two heat transport systems operate normally to remove heat at pony motor flow after the trip and coast down, and the affected heat transport system continues to remove heat with its intermediate loop at natural circulation. As a result, the heat transport system undergoes temperature transients that are not significantly different from those of a normal scram except for the affected loop. The transients within the affected loop are enveloped by the umbrella transients.

15.3.2.2.3 <u>Conclusions</u>

The intermediate pump seizure event does not cause significant core or heat transport system temperature transients for either the primary or the secondary shutdown system trip function actuation. 8

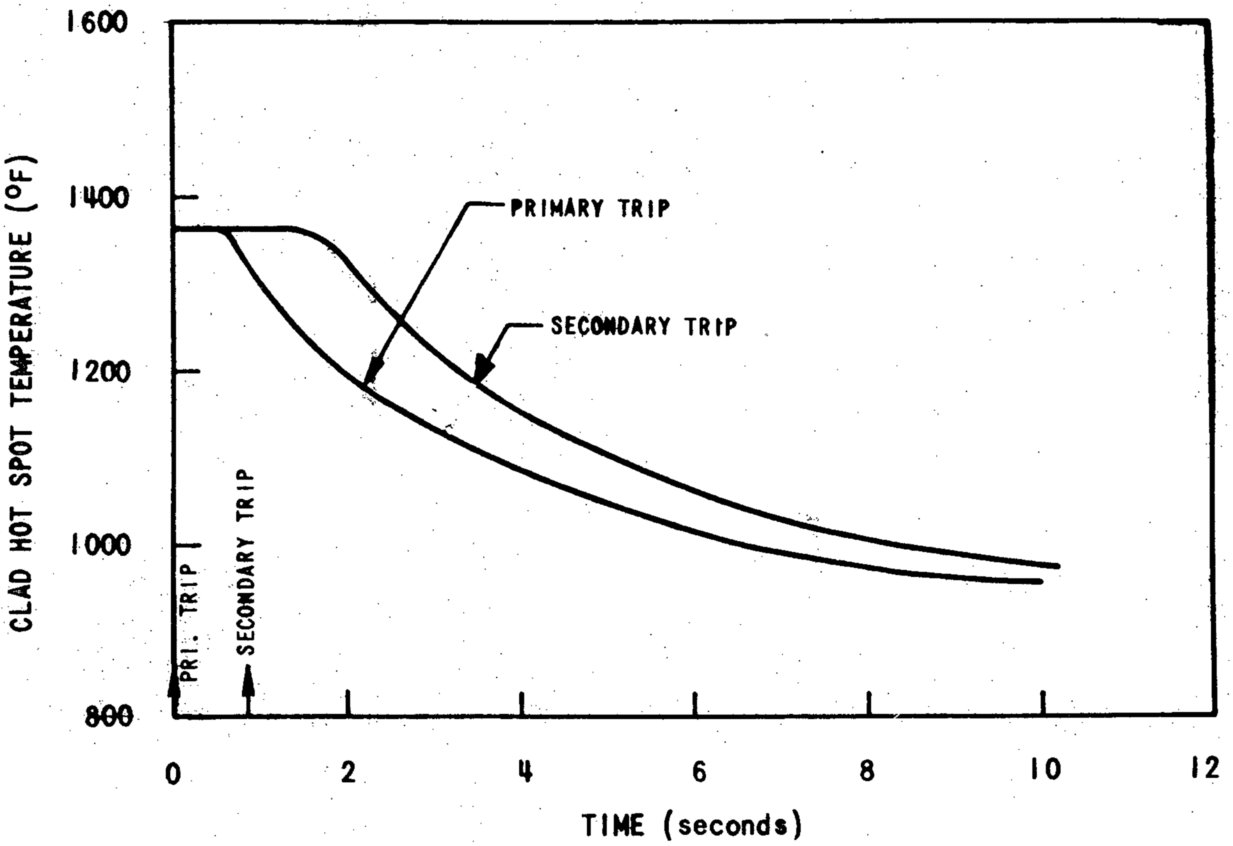


Figure 15.3.2.2-1. Hot Spot Cladding Midwall Temperature Vs. Time After Intermediate Loop Pump Seizure

6728-45

15.3.2.3 Small Water-to-Sodium Leaks in Steam Generator Tubes

15.3.2.3.1 Identification of Causes and Accident Description

The probability of a tube leak in the steam generators is expected to be quite small as a result of careful design supported by development and testing of the steam generators. However, the Steam Generator Leak Detection System, described in Section 7.5.5, has been provided to allow operator action to limit the consequences of a small leak in a steam generator tube.

The water-to-sodium leak detection system is designed to alert the operator to the existence of very small leaks, as small as approximately 2×10^{-5} lb. water/sec. For initial very small leaks which can be realistically expected (up to about 5×10^{-3} lb. water/sec.), the reactor will be shut down normally followed by a controlled cooldown and depressurization of the affected steam generator. The affected IHTS loop would then be drained to allow repair of the steam generator.

However, in the unlikely event of a small leak exceeding approximately 5×10^{-3} lb. water/sec, the operator may elect to scram the reactor and isolate and blowdown all three steam generator modules in the affected loop. The operator would also drain the affected IHTS loop, resulting in flow stoppage in that loop.

15.3.2.3.2 Analysis of Effects and Consequences

It is assumed that the reactor is operating at rated conditions when a leak occurs in a steam generator of such a nature that the operator elects to manually shutdown the reactor, isolate and depressurize the water side of the affected loop, and drain the sodium side of that loop. Dynamic analyses have not been completed for this event; however, the primary system response can be conservatively bounded by assuming that all heat removal capability is instantaneously lost in the IHX of the affected loop at the time when intermediate flow stops. The IHX primary outlet temperature increases rapidly to the primary inlet temperature. Core flow rate and the resulting fuel cladding and core coolant exit temperatures are identical to those for normal scram until the hot sodium from the affected IHX reaches the core. This is calculated to occur about 60 seconds after reactor scram, assuming a normal flow coast down. The hot sodium from the affected loop mixes with the sodium from the other two loops in the reactor vessel inlet plenum. Assuming perfect mixing in the core inlet region, the core inlet temperature increases about 90°F. If this increase in core temperatures 60 seconds after scram is conservatively added to the hot-channel coolant exit temperature for the normal scram, the hot-channel clad temperature would increase from about

810°F to 900°F. This increase in temperature would be somewhat larger if incomplete mixing occurs in the reactor vessel inlet plenum. However, even for the extreme assumption of zero mixing, the hot-channel coolant temperature could increase a maximum of only 265°F to about 1125°F, still well below the normal steady-state operating value. The core exit temperatures would then decrease as the reactor was cooled by the operable loops.

If it is assumed that this event occurs following operation with the maximum undetected intermediate-to-primary sodium leak rate there will be insignificant radiological release. Leakage of primary sodium into the IHTS is prevented by pressurizing the IHTS such that a pressure differential across the IHX (intermediate-to-primary) of at least 10 psi exists during plant operation. This pressure differential could be lost during the sodium dumping process and it is possible that primary sodium could enter the IHTS. Leak rates of approximately 6 gph will be detected during normal operation (Section 7.5.5) and therefore only small amount of primary sodium could be introduced into the IHTS during the pump coast down. This small amount of primary sodium would mix with the intermediate sodium and either remain in the non drainable sections of the IHTS, steam generators, and IHX, or be drained to the sodium dump tank. Over pressurization of this tank is prevented by either the equalization line or the pressure relief valve, the gases vented through this system will be the inert gas displaced by the sodium entering the dump tank. No sodium will be released in this process and the radiological consequences of this event are insignificant. 22

15.3.2.3.3 Conclusion

Core temperatures following a steam generator tube leak are well within the normal operating temperature range for the fuel and core. Residual heat removal is provided by the operable loops. This event is included in the overall plant duty cycle list that provides the basis for the thermal transient design conditions for the reactor and the main heat transport system.

15.3.2.4 Failure of the Steam Bypass System

15.3.2.4.1 Identification of Causes and Accident Description

The turbine steam bypass system regulates the flow of steam to the main condenser following a turbine trip to maintain steam pressure at 1450 psig. The system contains four bypass valves.

A failure of a bypass valve to open following a turbine trip, may result in a pressure increase in the steam system to the power relief valve set point. The temperature transient at the core for this event is conservatively bounded by failure of all the bypass valves to open. In the event of a failure of all bypass valves to open, the main condenser would be unavailable for cooling. Flow in the main steamline would be interrupted resulting in a rapid increase in pressure until the power relief valves at the superheater exists opened. The reactor would be scrammed by any one of the three steam-feedwater flow ratio trips.

When the available normal feedwater supply is exhausted, the Steam Generator Auxiliary Heat Removal System (SGAHRS) would be activated by the low drum level trip and feedwater provided by the auxiliary feedwater pumps (see Section 5.6). A backup trip is provided by low steam drum level that occurs after the normal feedwater supply is exhausted.

A failure of the bypass system valves to the open position would result in increased steam flow to the condenser. The action of the shutdown systems would depend upon initial power level and the magnitude of the bypass flow. In the limiting case, at full power with the failure of all valves open, the steam-feedwater flow ratio quickly trips the plant.

15.3.2.4.2 Analysis of Effects and Consequences

A turbine trip with the plant operating at rated conditions is assumed to occur accompanied by a complete failure of the steam bypass system to operate. The steam line pressure increases and the superheater power relief valves open and blow steam to the atmosphere. The reactor trips on low steam-feedwater flow ratio in about two seconds. The resulting core temperatures are very similar to those for a normal trip from full power. After the normal feedwater supply has been exhausted (greater than 20 minutes) the Steam Generator Auxiliary Heat Removal System (SGAHRS) is actuated on low steam drum level and automatically maintains drum water level. The event is conservatively bounded by the Loss of Normal Feedwater (See Section 15.3.1.6).

If the reactor is assumed to trip on low drum level (rather than one of the three steam-feedwater flow ratio trips), the event is also similar to the Loss of Normal Feedwater. Long-term core temperatures for SGAHRS operation are very similar to those for a normal shutdown, since the steam generation system, designed to remove plant heat using SGAHRS after a reactor trip, maintains the cold leg temperatures near their normal values.

A failure resulting from the bypass valves failing open is bounded by the case of all the bypass valves failing open at power. In this event the reactor would be quickly tripped by any one of the three steam-feedwater flow ratio trips. The resultant thermal transients are conservatively bounded by a main steam line break (see Section 15.3.3.1).

15.3.2.4.3 Conclusion

Core temperatures following inadvertent opening of or failure to open a steam bypass valve are similar to a normal trip. These events are included in the overall plant duty cycle list that provides the basis for the thermal transient design conditions for the reactor and the main heat transport systems.

15.3.3 Extremely Unlikely Events

15.3.3.1 Steam or Feed Line Pipe Break

15.3.3.1.1 Identification of Causes and Accident Description

The breakage of a steam or feed pipe in the steam generator system is considered an extremely unlikely event. If such a break should occur, the resulting accident might have one of several forms, depending on where the break is located in the system, its size and whether or not it is insolatable. It should be noted that a reactor trip by the Plant Protection System will shut down the reactor before any of the steam system temperature changes have been transported back to the reactor core (at pony motor speed approximately 150 seconds) hence no problem results with immediate reactor safety. The event instead is considered in the plant design for its effect on plant component service life through thermal-transient-induced stress.

The plant has incorporated design features to protect against the steam line break. For instance the Superheater Outlet Isolation Valve and Superheater Bypass Valve in each loop are active valves and will close within 3 seconds following a steam line break. Closing of these valves in the failed loop will prevent blowdown of more than one loop through the postulated pipe break. The valves in the failed loop will close by either a Low Superheater Outlet Pressure (< 1100 psig) or a High Steam/Feedwater Flow Mismatch. When a high steam/feedwater flow ratio occurs, the Superheater Outlet Isolation Valves and Superheater Bypass Valves in the other two loops will close. A detailed description of the Outlet Steam Isolation Subsystem (OSIS) is presented in Section 7.4.2. The superheater Outlet Check Valve provides additional back-up to prevent blowdown but is not relied upon in any analysis. The Superheater Bypass Valve is normally closed during operation.

In the event of failure of an active valve to close, the Superheater Outlet and Bypass Valves in the other two loops preclude their blowdown.

Breaks at the following locations have been investigated:

a. Main steam line rupture.

b. Steam line from a superheater to the main steam header.

c. Saturated steam line between the steam drum and the superheater.

d. Feedline break.

e. Recirculation line break.

The saturated steam line break has been selected as the most severe thermal transients of the events presented above. Analysis results for this event are presented in Figure 15.3.3.1-1. All of the above cases are summarized as follows:

Main steam line rupture:

A steam break at the main steam header would, if not isolated, produce a severe cold leg temperature transient in all three loops consisting of a down transient due to initial excess cooling followed by an up-transient after dryout. It is not plausible, however, to assume that isolation would fail to occur in all three loops, hence for case (a) automatic isolation was assumed at three seconds with isolation initiated by the Plant Protection System (PPS).

Once the superheater outlet isolation valves close, the plant achieves a new operating point based on steam load through the safety valves and hence no other excessive plant temperatures are produced. As noted below, a reactor shutdown is initiated by the PPS based on either the primary shutdown system (steam/feed flow mismatch) or secondary system (Low Drum Level), terminating high power operation before excessive loss of water inventory. Either the high steam-to-feedwater flow ratio or the Low Steam Drum Water Level Trip also activates the steam generator auxiliary heat removal system (SGAHRS) as noted below and discussed in Section 5.6. All three loops would provide heat removal from the core. With the superheat steam line isolated, pressure in the steam system will build up to the relief setpoint. The drum water level will drop due to steam venting and the low steam drum water level trip will then activate SGAHRS if it has not been activated earlier in the transient by the High Steam to Feedwater Flow Ratio.

Rupture in a Steam Line Between a Superheater and the Main Steam Header:

This event results from a break occurring in the superheater exit steam line upstream of the isolation valve. A similar event follows from a break downstream of the isolation valve (including a break in the main steam line) if the isolation valve fails to close. For these cases, isolation can still be effectively accomplished by the superheater inlet isolation valve, either by manual initiation or automatically when steam drum pressure falls below 500 psig. Consequently, a break in the superheater-to-header line has an effect similar to the preceding main steam line break case, but its effects are limited to a single loop.

Saturated Steam Line Break:

In the saturated steam line break, case (c) above, the break may be located such that loss of water in the affected steam drum cannot be prevented. Isolation valves on the modules could still be closed, but safety valve outflow will still lead to module dryout. Consequently, no credit is taken for isolation in these cases.

As steam is removed from the system by the break, increased flashing of water into steam within the steam generator occurs, removing additional heat and causing the sodium temperature initially to decrease at the evaporator exit. A plant shutdown, when initiated by low steam feed flow, will cause coastdown of the intermediate sodium pump, and hence will amplify the initial decrease in evaporator exit temperature. Subsequently, when most of the moisture has been discharged from the steam generator, both evaporators and superheater will dry out, and the evaporator exit sodium temperature will increase to approach the intermediate hot leg temperature. The cold leg temperature increase will eventually be transported back to the reactor inlet, after being conducted through the IHX of the affected loop. Due to extended transport delays at pony motor flowrates, the temperature increase will not reach the reactor for 150 seconds or more, considerably after the time at which the reactor was safely shutdown by either the primary or secondary trip functions. These event results are presented in Figure 15.3.3.1-1.

Because the initial sodium temperature decrease followed by temperature increase produces the widest total span of temperature transient, this case has been selected as most limiting. The earlier primary PPS trip function is also applied since it produces a larger initial decrease in evaporator sodium exit temperature. It should be noted that a later SDS trip, while producing a sodium temperature transient of smaller span, may cause a larger temperature rate-of-change. This result, however, is bounded by the water/steam side isolation and dump event of Section 15.3.1.7.

Feed Line Break:

A break in a feed line between a steam drum and the feed line check valve will have consequence (i.e., dryout) in the affected loop similar to the preceding case but less severe in terms of total span of the sodium temperature transient. (A break upstream of the check valve causes a loss of normal feed to all units and is covered by Section 15.3.1.6). Since the feed line break initially discharges liquid from the drum for the most part, less energy is absorbed in changing liquid into steam. Consequently, the initial evaporator sodium exit temperature down-transient is small. The evaporator sodium exit temperature up transient follows dryout as above, but temperature rates are bounded by those of the water side dump of Section 15.3.1.7. Normal feedwater flow in the other two loops will stop because of low feedwater system pressure. The normal feedwater isolation valves will close and SGAHRS will be activated by a high steam-feedwater flow ratio signal. Core cooling will then be provided by SGAHRS in the two unaffected loops.

Recirculation Line Break:

A large break in a recirculation line quickly discharges the contents of the affected drum, similar to the feed break case. It also leads to rapid dryout of the evaporator modules, since the evaporators can discharge directly back to the break (with exit check valves closed). As a result, the initial evaporator sodium exit temperature down-transient is small. The dryout up-transient for this temperature is at a rate similar to that for the water-side dump found in Section 15.3.1.7.

An alternate location for this break is at the exit of one evaporator module. Closure of the other isolation valves, including the inlet valve on the affected module, would lead to a dryout of the generator similar to previous cases. If the inlet isolation valve on the module does not close, the contents of the drum would be dumped through the affected module, producing a severe temperature down-transient on that module. The remaining module will dry out and its resulting increase in sodium exit temperature will mix with that from the faulted module to attenuate the net intermediate cold leg temperature transient.

For the steam and feed break cases, the following conditions have been applied to assure a conservative analysis:

a. The largest possible break size is assumed, corresponding to the full guillotine severance of the pipe involved.

b. The earliest PPS trip is used to predict the largest span for the sodium temperature transient for cases in which the intermediate cold leg temperature is considered.

c. The transients were run from a starting point at the 1121 MWt reactor power design condition (stretch power).

d. Credit has not been taken for heat storage in shell and structural metal in active or unheated parts of the modules in mitigating the thermal transients. Credit was taken only for 75% of the tube metal in the heated part of the modules.

e. No isolation was performed on the affected unit during the drum to superheater break, feed break and recirculation line break cases and the steam generator was allowed to go to full dryout.

The action of the Plant Protection System (PPS) in the above cases is the following:

Primary Shutdown System

a. Reactor and plant trip - steam-feedwater flow ratio

Secondary Shutdown System

a. Reactor and plant trip - high evaporator outlet temperature

The following actions are also initiated based on the parameters identified:

a. Superheater steam isolation - high steam to feedwater ratio

b. Feed isolation - low drum pressure

c. SGAHRS initiation - steam-feedwater flow ratio

An alternate secondary PPS reactor and plant trip on low drum level is also available.

It should be noted that in keeping with the previous comment concerning a more conservatively wide span of sodium temperature transient for the earlier trip, the primary shutdown system function has been applied in cases for which the intermediate cold leg sodium temperature was being considered. With a trip by either the primary or secondary, the reactor was safely shutdown well before the time that the steam-generator-originated temperature transient reached the reactor inlet.

15.3.3.1.2 Analysis of Effects and Consequences

The break in a 14" saturated steam line between the steam drum and superheater has been selected as the most severe of the above cases. The analysis was performed using the DEMO Code. Results of this analysis are presented in Figure 15.3.3.1-1. For this case, the superheater exit check valve closes and the superheater dries out. The superheater sodium exit temperature then approaches the intermediate hot leg temperature. The evaporator sodium exit temperature initially decreases to 504°F as the break removes energy from the steam generator, then after dryout also increases to approach the intermediate hot leg temperature.

As shown in Figure 15.3.3.1-1, the resulting temperature transient does not reach the reactor inlet until after 150 seconds has elapsed. The reactor has been shutdown and the reactor temperature just after the shutdown is the same as that for a conventional reactor trip.

The steam or feedwater line break will result in the release of large quantities of steam and water. If this release is into one of the cells in the steam generator building a rapid pressurization will occur until the cell venting rate balances the release rate. Venting capacity and protective measures are provided, as necessary, to limit the internal cell pressure and prevent structural failure or mechanical damage which could result in propagation of the event to adjacent loops in the HTS required for decay heat removal.

15.3.3.1.3 Conclusions

Action of the Plant Protection System to shut down the reactor prior to the transport of steam-generator-originated temperature transients to the reactor core will preclude the possibility of core damage from the steam line or feed line break event as long as intact loops are available to remove reactor decay heat. Core temperatures following large steam or feedwater pipe breaks are only slightly perturbed from those that occur during a normal scram. As a result, none of the above transients have direct consequences to core safety. Long-term cooling after shutdown from these events is provided by the normal shutdown cooling system (steam bypass) or SGAHRS. The severe cold leg temperature transients produced for some of the breaks, however, must be considered in the design bases for plant components.

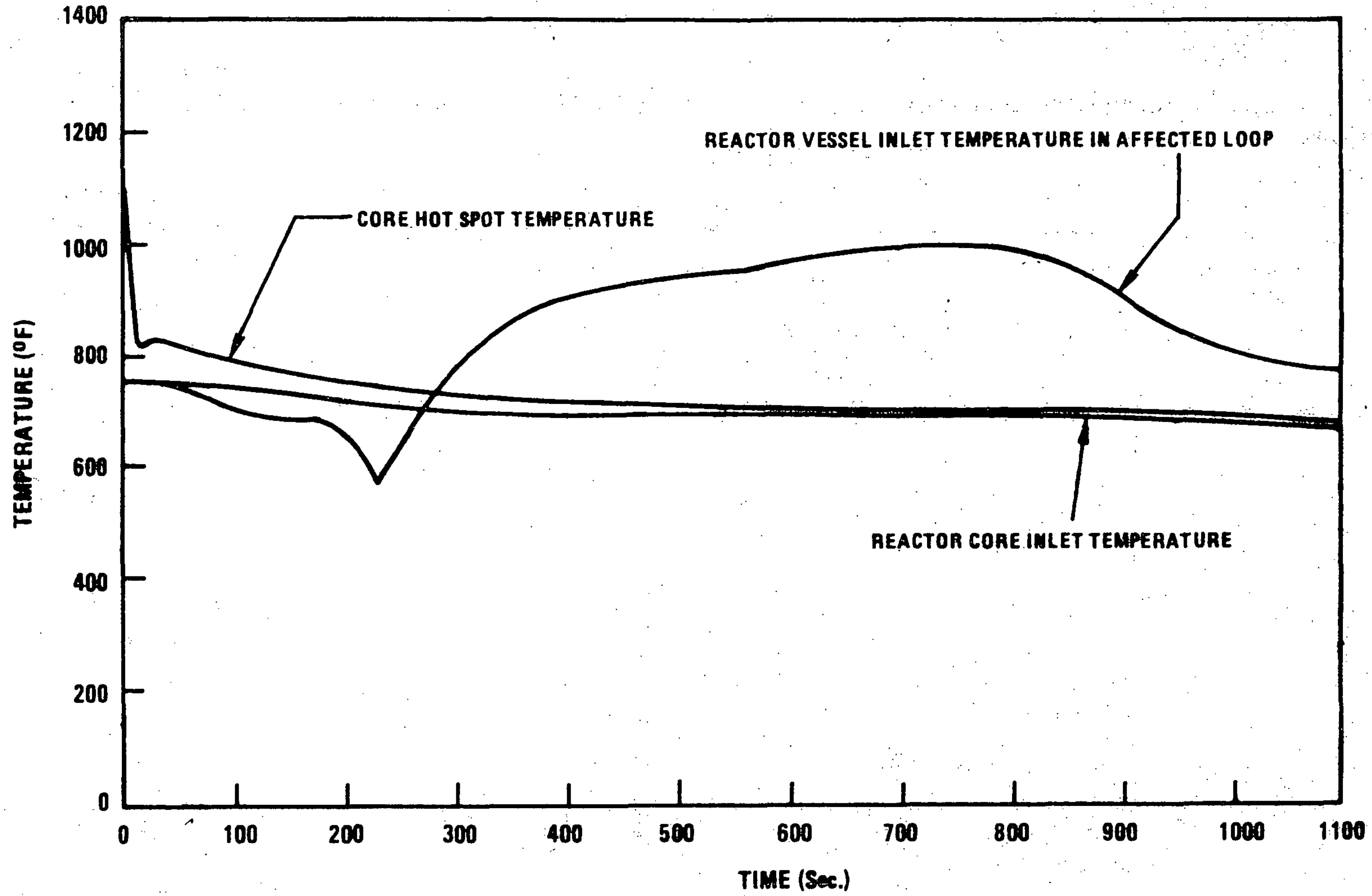


Figure 15.3.3.1-1. Temperatures (°F) of Pertinent Parameters as a Function of Time After a Saturated Steam Pipe Break (Between Drum & Superheater)

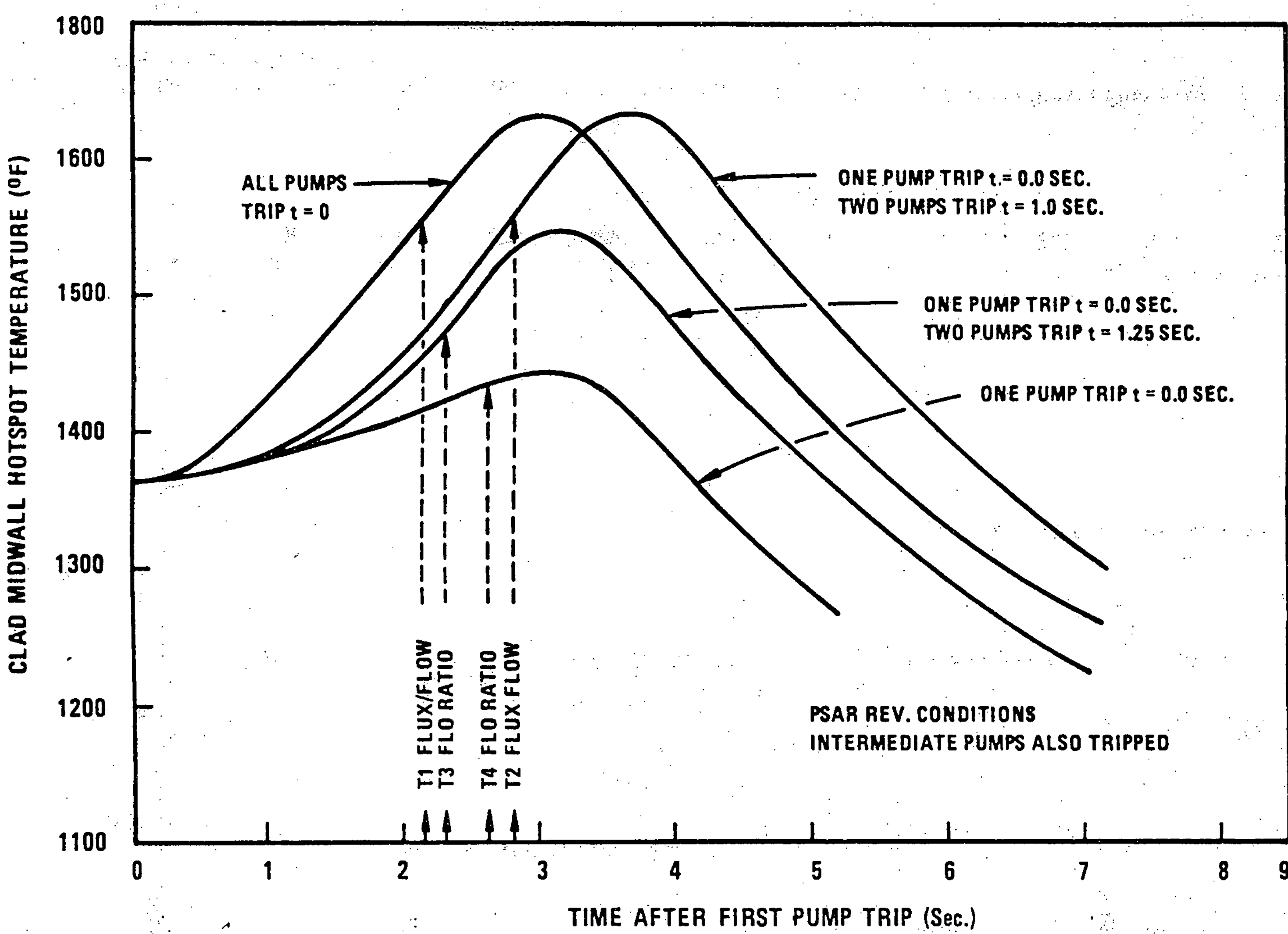


Figure 15.3.1.2-2 Clad Hotspot Temperatures For Various Sequential Loss of Flow Cases (Secondary Shutdown System Only)

15.3.3.2 Loss of Normal Shutdown Cooling System

15.3.3.2.1 Identification of Causes and Accident Description

Loss of normal shutdown cooling will occur following loss of the main condenser, since the heat sink for the normal shutdown cooling mode is provided by the main condenser. Other conditions that affect the ability of the main condenser to provide shutdown cooling include loss of normal feedwater (Section 15.3.1.6), failure of the steam bypass system (Section 15.3.2.4), and main steam line pipe break (Section 15.3.3.1). In the event of a loss of the condenser, the reactor will be scrammed. Since the steam bypass system is prevented from operating in the event of loss of condenser, a loss of condenser would result in a sequence of events similar to that for failure of the Steam Bypass System (Section 15.3.2.4).

15.3.3.2.2 Analysis of Effects and Consequences

The consequences of a loss of condenser are slightly less severe than that for the failure of the steam bypass system (Section 15.3.2.4) since a reactor trip occurs somewhat earlier in the transient. Core temperatures are similar to those for a normal scram.

15.3.3.2.3 Conclusions

Core temperatures following loss of normal shutdown cooling are similar to a normal trip.

15.3.3.3 Large Sodium-Water Reaction

15.3.3.3.1 Identification of Causes and Accident Description

A large leak in a steam generator tube will result in injection of high pressure steam and/or water into the IHTS sodium. The resulting sodium-water reaction (SWR) will generate higher than normal pressures and temperatures in the IHTS. As discussed in Section 15.3.2.3, Steam Generator Tube Leak, the probability of a leak in a tube in the steam generators is expected to be quite small as a result of careful design supported by development and testing of the steam generators. However, a leak detection system, described in Section 7.5.5, has been provided to allow operator action to limit the consequences of a leak in a steam generator tube. The leak detection system will alert the operator to the existence of a leak rate as low as 2×10^{-5} lb. water/sec. For initial leak sizes which can be realistically expected (up to about 10^{-2} lb. water/sec.) there will be sufficient time for operator action to limit damage to the steam generator and to prevent a significant increase of the leak rate. Should a leak occur of such magnitude that operator action as described above is not effective, the Sodium-Water Reaction Pressure Relief Subsystem (SWRPRS) will provide sodium side pressure relief by operation of the rupture discs in the IHTS so that integrity of the IHTS piping and components, e.g., pump and the Intermediate Heat Exchanger (IHX) will be maintained. No operator action is required for the SWRPRS to perform its design function. A description of the SWRPRS is given in Section 5.5.

Large leaks might occur due to sudden rapid propagation of a large flaw in a tube. In this event, the leak could develop in a very short time and in the limit could approach the instantaneous double ended guillotine failure assumed. A second mechanism for developing a large leak is through wastage from a small leak. The latter mechanism is believed to have the higher probability of occurrence. An estimate of the time required for the development of a significant leak due to wastage can be obtained from the results of small SWR leak development and wastage data. Leaks in the range of 10^{-6} to 10^{-3} lb/sec have been observed to self-enlarge as indicated in
59 Figure 15.3.3.3-1. A leak of the order of 10^{-5} lb/sec could over the period of several hours suddenly increase in size to the order of 10^{-3} to 10^{-2} lb/sec. A leak of this magnitude, directed through a drilled hole (an idealized, conservative leak geometry) onto an adjacent target (representing an adjacent steam generator tube) has been observed to cause wastage rates on the target of 1 to 5 mils per second (Ref. 1). At these wastage rates, failure of a steam generator tube adjacent to the leaking tube could occur within about twenty seconds. Definitive data on the ultimate leak size resulting from wastage failure does not exist, however, wasted areas exhibit configurations ranging from cone shaped craters to irregular and diffuse wastage regions. The area of the failure in the adjacent tube wall, if the wasted area is cone shaped, would be small relative to a double ended failure area. If the jet emanating from the original leak is diffuse (as opposed to a concentrated jet) the resultant leak area on the adjacent tube could be larger but would not be expected to approach that of a double ended guillotine.

Based on the foregoing discussion, the largest expected steam generator failure is the double ended guillotine failure of a single tube. However, as explained in detail in Section 5.5.3.6, a more severe event has been postulated to ensure adequate design margin. This DBL is defined as an Equivalent Double Ended Guillotine (EDEG) failure of a steam generator tube which is followed by two additional single DEG failures, spaced at 22
1.0 second intervals, for a total of 3 DEG. This sequence is superimposed on a system which has been pressurized by an undetected moderate sized leak to just below the rupture disk burst pressure. 30

59 The injection of water into sodium results in high IHTS pressure pulses from the sodium-water reaction. This pressure is relieved by the rupture discs in the SWRPRS. Sodium reaction products and hydrogen are expelled from the IHTS into the SWRPRS where hydrogen is separated from the particulate and liquid matter. The hydrogen is vented to the atmosphere through a flare stack and liquid and particulate are contained in the reaction
59 products separator tanks under an inert atmosphere. Operation of the rupture discs automatically isolates and depressurizes the water side of the steam generators to limit damage to the system. The remaining sodium in the affected IHTS loop would be drained by operator action.

The action of the Plant Protection System (PPS) in this event is the following:

a. Primary Shutdown System - Trip on steam flow - feed flow mismatch.

b. Secondary Shutdown System - Trip on sodium water reaction.

Either of the above trips will cause a reactor shutdown well before the temperature transient resulting from the water/steam isolation and dump can be transported back to the reactor inlet. Consequently, no reactor clad or fuel damage is involved with this event.

Details of the resultant pressure pulses and their impact on the adjacent steam generators, IHX and pump can be found in Section 5.5.3.6 of this PSAR. This includes evaluation of various sizes of failures, including discussions of the probable development sequences of various leaks, up to and including the DBL in the evaporator and the superheater modules. 30

15.3.3.3.2 Analysis of Effects and Consequences

The analysis of the effects of the DBL on the steam generators and associated components in the IHTS and SWRPRS will be carried out using the
59 Transwrap computer code, as discussed in Section 5.5.3.6.

The impact of this event on the reactor core is similar to the event analyzed in Section 15.3.1.7 (Inadvertant Actuation of the Sodium-Water Reaction Pressure Relief System) by the DEMO Code. The reactor, due to the long transport delay associated with this event, does not immediately see the temperature changes, so that when the reactor trip occurs on steam-flow - feed flow mismatch (less than 4.0 seconds), the transient at the reactor is the same initially as a conventional trip. The core hot spot temperature will decrease quite rapidly and remain below normal operational temperatures throughout the course of the accident event.

If it is assumed that this event occurs following operation with the maximum undetected intermediate-to-primary sodium leak rate, the possibility of a radiological release resulting from venting of the sodium water reaction products must be considered. Leakage of primary sodium into the IHTS is prevented normally by pressurizing the IHTS such that a pressure differential across the IHX (intermediate to primary) of at least 10 psi exists during plant operation. This pressure differential could be lost following bursting of the SWRPRS rupture discs and it is possible that primary sodium could enter the IHTS. During normal operation
59 (Section 7.5.5) peak rates in excess of approximately 6 gph will be detected and therefore only small amounts of primary sodium could be introduced into the IHTS during the depressurization transient.

Section 15.6.1.5 looks at a more severe incident in which the 24 inch IHTS pipe is severed between the IHTS pump and the IHX. This results in immediate IHTS depressurization and all IHTS sodium spilled onto the cell floor along with 1.4 pounds of primary sodium leaked across the IHX. This 1.4 pounds of primary sodium represents a conservative envelope of the amount that can be leaked across the IHX during the SWRPRS actuation event. Regardless of the location (superheater or evaporator) of the sodium-water reaction, the
59 type of initiating leak, and the number of secondary tube failures in the sodium-water reaction incident will result in less primary sodium entering the IHTS. This is true because for the sodium-water reaction, there is no sudden depressurization of the IHTS as occurred when the 24-inch pipe was severed.

Primary sodium that leaks across the IHX may be transported to the SWRPRS tank if there is sufficient flow available to move sodium from the IHX to the superheater inlet. The maximum IHTS sodium available to transport primary sodium is calculated by summing: (1) the integral of pump flow as a function of time for pump coastdown and (2) expansion tank and pump tank cover gas expansion down to 28 psia (minimum pressure to elevate IHTS sodium up to IHX inlet). This is very conservative since during pump coastdown, some of the sodium flowing from the expansion tank and pump tank will probably reverse at the pump inlet and will flow towards the evaporator exit rupture disc. The sodium that flows in this direction will not be available to transport the primary sodium.

The primary sodium that reaches the superheater inlet is assumed to react with the water/steam and be swept into the SWRPRS tank. All primary sodium particles are assumed to be airborne and are swept up and stack with the hydrogen gas. The separator will remove approximately 95% of these particles but 5% will escape to the atmosphere as the hydrogen gas is burned. As a result of the burning of the hydrogen gas, the sodium particles will be carried to heights much higher than the actual stack height. From references 3 and 5 the effective release height may exceed 1000 meters depending on wind velocity and quantity of hydrogen gas burned. For conservatism the effective release height is assumed to be 300 meters (Reference 3). Reference 4 contains procedures for calculating centerline doses at various distances from the point source (puff release) and for various effective release heights. The X/Q values calculated here are consistent with site meteorology and an effective release height of 300 meters. 22

The site boundary dose was calculated using the following conservative assumptions:

1. Pasquill stability type A and wind speed of 1 m/s (CRBRP frequency 3.5%). This is the meteorological type that yields the largest dose at site boundary for an effective release height of 300 meters.
2. No radioactive decay during transport time from IHX to site boundary.
3. Inhalation rate for exercising man (3.427×10^{-4} lung fraction/sec).

The results of these calculations show that with 1.4 pounds of primary sodium reaching the SWRPRS tank, the site boundary whole body and lung doses are 0.006 mrem and 0.04 mrem, respectively. This is much less than the 10CFR20 short term limit of 2 mrem. Bone dose is about 50% that for lung dose. Therefore, the radiological consequences of a sodium-water reaction with a leaking IHX will not exceed 10CFR20 limits at the CRBRP site boundary. 22

The off-site radioactivity consequences of venting the total water/steam inventory in a steam generator loop are given in Section 7.1.2.5.2 of the CRBRP Environmental Report. As given in that section, the maximum off-site skin and whole body doses for the postulated total release of the water/steam inventory are 7.2×10^{-5} and 6.3×10^{-3} mrem, respectively. If these rates should be concurrent with radioactivity resulting from the leakage of primary sodium into the IHTS and escape of the primary sodium to the atmosphere, as calculated above, the total exposure rates will still be far below 10CFR20 short term limits.

The limit type analyses for the release of radioactivity by a sodium-water reaction, provides the maximum effects which would result from radioactivity in the primary sodium leaked into the IHTS, and from tritium activity buildup in the water/steam. 30

15.3.3.3.3 Conclusions

The large sodium-water reaction event generates sizeable pressure pulses on the steam generators and associated components in the IHTS. As discussed in Section 5.5.3.6 the consequences of this event are within the design margin of the IHTS and components. The reactor is shutdown before any of the resulting temperature transients are transported to it. A large margin exists between the potential offsite doses and the applicable guideline limits. Therefore, it is concluded that this event does not present any safety problems. 22

References

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26 * 2. F. J. Moddy, "Maximum Two-Phase Vessel Blowdown from Pipes", APED-4827, General Electric Company Nuclear Energy Division, April, 1965.

3. R. K. Anand, "Overall Separation Efficiency Requirement of the Centrifugal Separator for the LMFBR Sodium Water Reaction Pressure Relief System (SWRPRS), Section 3.3.

4. David H. Slade, Editor, "Meteorology and Atomic Energy". TID-24190, 1968.

5. H. Moses, J. F. Carson et. al., "Calculations of Effective Stack Height". 22

* References annotated with an asterisk support conclusions in the Section. Other references are provided as background information.

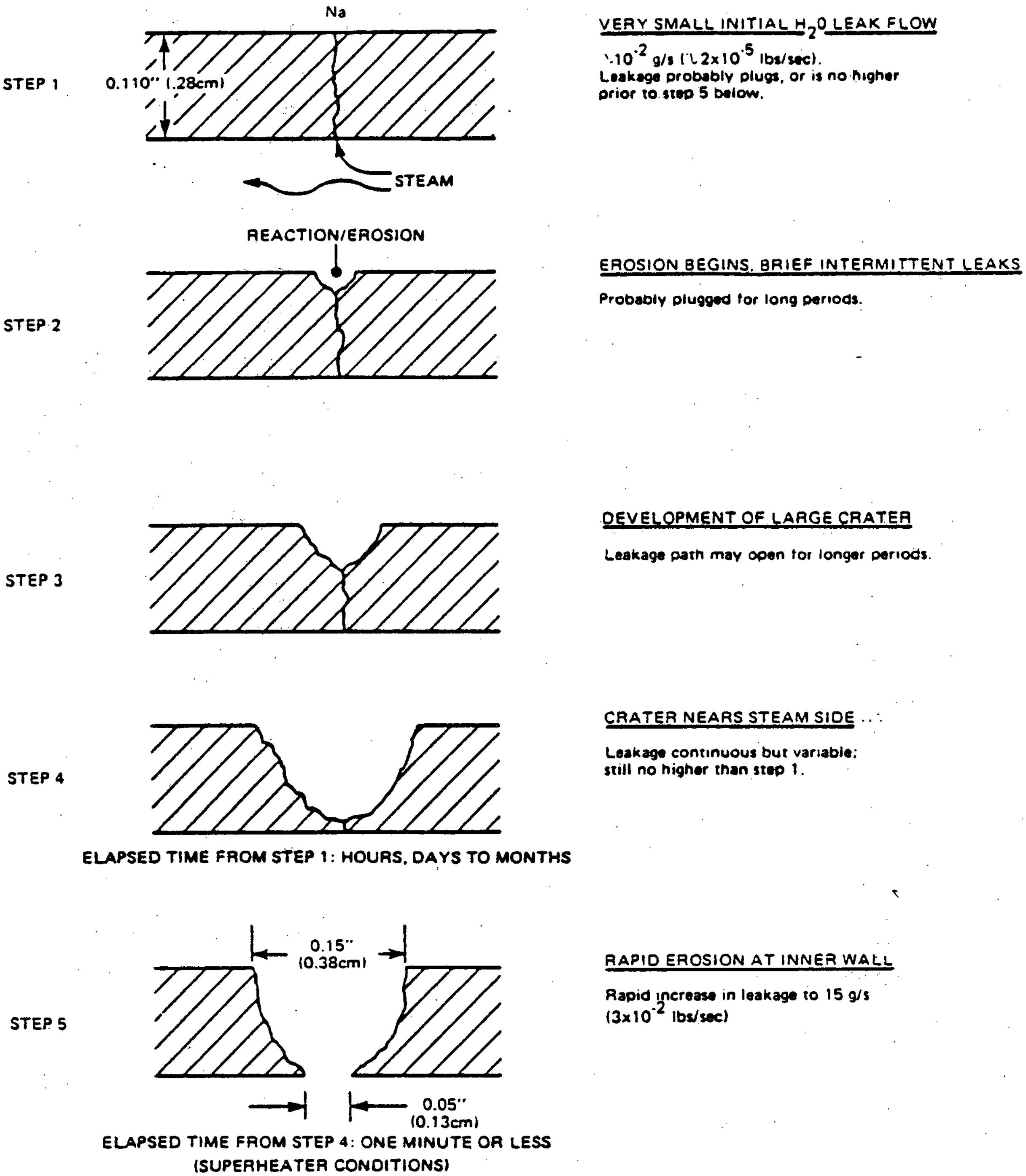

Figure 15.3.3.3-1. Development of a Large Leak From a Small Steam Leak in 2 1/4Cr-1Mo Tubing Exposed to Sodium

15.3.3.4 Primary Heat Transport System Pipe Leak

15.3.3.4.1 Identification of Causes and Accident Description

Small sodium leaks have occurred several times in sodium testing facilities and in operating reactors. As a result, PHTS leaks are considered in the design and evaluation of the plant to assure that the design has adequate capabilities from the standpoint of core thermal transients. This particular section will address the PHTS pipe leak as an undercooling event while Section 15.6.1.4 provides a detailed discussion of the PHTS pipe leak and its consequences with regard to cell pressure and temperature transients and radiological effects.

Based on a detailed evaluation of the PHTS piping structural integrity, presented in Reference 2 of Section 1.6 of the PSAR, a 4-inch crack was chosen to establish the design basis leak (DBL) for the functional performance of the heat transport system (see PSAR Section 3.6.1.1). The maximum leak rate corresponding to the 4-inch crack is 8 gal/min. As indicated in Section 7.5.5.1, the liquid metal-to-gas leak detection system is designed to provide detection capability for leaks as small as 100 gm/hr (5.3×10^{-4} gal/min).

15.3.3.4.2 Analysis of Effects and Consequences

A 8 gal/min leak would not result in any measurable core transient. An automatic reactor trip would not be required and adequate time (significantly greater than 1/2 hour) would be available for the operator to manually shutdown the reactor. Therefore, a leak from the PHTS is not a design basis event for the Plant Protection System. A normal reactor shutdown would be accomplished following indications from the leak detection system. The primary PPS includes reactor vessel sodium level and flux/pressure trip functions, that would provide margin capability to scram the reactor in the event of a leak significantly greater than the DBL.

Following an indication of a leak, the reactor would be shutdown and the coastdown of the pumps would reduce the system pressure. After pump coastdown (<1 minute), the leak rate would be reduced to a fraction of the 8 gal/min leak rate used for the event because of the pressure reduction and the system would then continue to drain until static equilibrium of the fluid in the system is reached, assuming no operator action to reduce the amount of sodium released. The quantity of sodium which could potentially leak from the system during this period is dependent on the location of the leak and the action that the operator takes. Once the plant is shutdown, the leakage rate becomes so small that the operator would have several days to select a method for further reducing the sodium leakage. Even if no further action were taken, the system design (guard vessels and elevated piping) would assure that long term core cooling would be provided.

The 8 gal/min leak rate is orders of magnitude below the leak rate that could cause a significant core transient. Conservative analysis indicates that for 3-loop operation, a transient maximum loss rate of over 50,000 gal/min would be required for the core sodium temperature to approach the saturation value, and this would require a rupture of more than 1 square foot at the reactor inlet nozzle. At other postulated primary heat transport system locations, even larger rupture areas would have to be postulated to challenge core cooling. Separate best-estimate margin analyses have demonstrated that even leaks as large as a double ended rupture can be accommodated without a loss of core coolable geometry. The results of this analysis were confirmed in Reference 16 of Section 1.6 of the PSAR.

15.3.3.4.3 <u>Conclusions</u>

The improbable occurrence of a leak, on the order of 8 gal/min in the PHTS piping would lead to an inconsequential transient in the reactor. Activation of several leak detection systems would result in corrective action including manual plant shutdown. The consequences would be limited to an economic penalty for plant downtime, sodium cleanup, and piping repair. Moreover, a leak several orders of magnitude greater than the 8 gal/min leak would not cause hot channel coolant temperatures to approach saturation.

15.3.3.5 Intermediate Heat Transport System Pipe Leak

15.3.3.5.1 Identification of Causes and Accident Description

Sodium leaks associated with the Intermediate Heat Transport System (IHTS) are being considered on a different basis from the Primary Heat Transport System (PHTS). Design measures lead the project to believe that ruptures in the IHTS are of very low probability. However, even though the same stringent Quality Assurance measures, fracture mechanics tests and analysis, codes and criteria apply equally to the two systems, there are some differences that must be taken into consideration. Among those characteristic that set the IHTS apart from the PHTS is the fact that a considerable portion of the intermediate system lies outside containment in an air atmosphere. In such an environment, the piping is subject to the corrosion rate associated with a sodium leak in an air atmosphere.

Also, the IHTS employs mixing tees as part of the piping configuration. At this stage in the design of the IHTS, careful considerations are being given to these mixing tees to determine the thermal stresses and mixing induced vibratory loads that could be expected to occur during the design life of the component.

Development programs are being formulated to evaluate these two characteristics of the IHTS piping system. One program has been formulated to define more precisely the effects of high temperature sodium leaking from a stainless steel pipe in an air environment. A complete description of this corrosion program is given in Section 1.5 of this PSAR. The information obtained from that program will be used in the fracture mechanics analysis along with information on leak detectability to prove that a large pipe rupture will not occur in the IHTS piping system.

The second development program concerns the evaluation of mixing components in sodium under prototypic CRBRP service conditions. Presently mixing tees are being tested to establish design criteria at ANL and HNL (ORNL). The data from these programs are not considered critical to establish the incredibility of Intermediate Pipe Rupture, but are oriented toward establishing design bases. Should the results of these tests at ANL and ORNL suggest further data on mixing tees are required to preclude overstressing, the option is available to extend these programs.

Besides the development programs discussed to insure pipe integrity in the IHTS piping system, the ability to do regular, extensive inservice inspection will add to the assurance that pipe integrity can be maintained.

It is expected that results from the above listed development programs along with inservice inspection considerations, pipe fabrication quality assurance measures, fracture mechanics analysis and tests, and leak detectability will lead to the conclusion that a large pipe rupture equivalent to complete severance of the pipe is not credible in the IHTS piping system. For this reason, and because the necessary design modifications (indicated below and in Section 1.5) could readily be incorporated at a late date, an IHTS pipe break has not been used as a design basis event at this time. Since the data currently available on the corrosion rate of stainless steel piping from leaking sodium and on the behavior of mixing components under CRBRP service conditions are not conclusive at this time, it was determined that a prudent approach to analyzing the potential problems associated with an IHTS leak was to examine the limiting case, namely, a leak of the same magnitude as would result from complete severance of the pipe. It is fully expected that at a later date, the development program examining leaking sodium in an air atmosphere will provide specific corrosion rates, thus enabling an analysis of this event based on a specific leak size, rather than the current limiting case approach.

Based on the previous discussion, an evaluation of potential leak sizes and locations was carried out for the IHTS. These evaluations indicate that in terms of the magnitude of the temperature increase in the primary cold leg of the affected loop, the worst leak size and location which may be postulated, is a severance occuring in the 24 inch IHTS piping between the flow meter and the IHX. Large leaks at this location may not result in an immediate low flow trip, as discussed below, allowing reactor operation to continue for a period of time without heat removal capability of the IHX of the affected loop. Stoppage of IHTS flow due to loss of sodium and subsequent loss of pump suction after inadvertent opening of dump valves can also result in conditions similar to those of the worst case large leak in the IHTS (Section 15.3.3.5.2). The dump flows are relatively small, and no significant temperature transients occur before cessation of flow. 8

Large leaks at other locations in the IHTS would be detected by the primary-to-intermediate flow ratio trips, causing a reactor trip coincident with a significant (>20%) reduction in IHTS flow through the IHX. For these cases, core exit coolant and fuel clad temperatures would not be significantly different than for a normal reactor trip and the core would be cooled by the remaining IHTS loops.

Small leaks at any location in the IHTS would not result in the large (300°F maximum) primary cold leg temperature increases discussed in Section 15.3.3.5.2 before a primary cold leg high temperature trip would cause a reactor shutdown. However, smaller leaks of a specific size and at a particular location could result in an increase in the primary cold leg sodium temperature in the affected loop to just below the trip level setting (120°F above the normal cold leg temperature). This would result in an increase in core coolant exit temperature. The resultant increase in primary system hot leg temperature, when propagated through the IHX, would cause a reactor trip from a delayed high temperature trip in the primary system cold leg. In all cases, post-shutdown cooling of the core would be provided by the unaffected loops.

For a large leak between the flowmeter and the IHX, the increased pump output (due to reduced flow resistance) might not produce a reactor trip on low primary-to-intermediate flow ratio, or primary-to-intermediate speed ratio. This is a very conservative assumption because it is expected that an immediate trip will take place. In the event of no trip, IHTS flow through the IHX of the affected loop could be significantly reduced, and the cold leg primary coolant temperature at the IHX exit would begin to increase toward the primary hot leg temperature until the high temperature primary cold leg trip point (120°F) above the normal primary cold leg temperature is reached. Reactor trip and automatic tripping of the main coolant pumps would then occur. Post-shutdown core cooling would be provided by the unaffected loops.

15.3.3.5.2 Analysis of Effects and Consequences

The large leaks in the IHTS discussed in the previous section are assumed to occur with the reactor operating at rated conditions. Dynamic analyses have not been completed for these events. However, the primary system response for the worst case large leak equivalent to complete severance of the pipe in the IHTS can be conservatively bounded by assuming that all heat removal capability is instantaneously lost in the IHX of the affected loop at the time the leak occurs, i.e., the intermediate side is instantaneously voided of sodium. The IHX primary exit temperature increases rapidly to the primary cold leg high temperature trip level and the temperature of sodium in the IHTS piping near the IHX rises toward the primary hot leg temperature. A reactor trip occurs about one second after the primary cold leg trip point is reached. The main 8
coolant pumps are automatically tripped, and coast down to pony motor speed. Core flow rate and the resulting fuel cladding and core coolant exit temperatures are identical to those for a normal scram, until the hot sodium from the affected IHX reaches the core. This is calculated to occur about 15 seconds after reactor scram, using the delay times to scram the reactor and trip the pumps and assuming a normal flow coast down rate. The hot sodium from the affected loop mixes with the sodium from the remaining loops in the reactor vessel inlet plenum. If the temperature increases in the affected loop to the initial steady state hot leg temperature (300°F increase) and assuming perfect mixing in the core inlet region, the core inlet temperature increases about 100°F. If this increase in core inlet temperatures 15 seconds after trip is conservatively added to the hot-channel coolant exit temperatures for the normal scram, the hot-channel coolant exit temperatures for the normal scram, the hot-channel coolant exit temperature would increase from about 830°F to 930°F. This temperature increase would be somewhat larger if incomplete mixing occurs in the reactor vessel inlet plenum. However, even for the extreme assumption of zero mixing, the hot-channel coolant temperature culd only increase a maximum of 300°F (to about 1130°F), still well below the normal steady state operating value. The core exit temperatures would then decrease as the reactor is cooled by the operable loops. For the special case of the small 28
leak that results in an increase in the primary cold leg temperature in the affected loop of less than 120°F (to just below the trip point), the hot-channel core coolant exit temperature would increase less than 40°F (assuming perfect mixing at the core inlet). Even for the limiting case leak condition of zero mixing, the maximum hot-channel temperature would increase less than 120°F (from 1340°F to about 1460°F).

15.3.3.5.3 Conclusions

The hot-channel coolant temperature following the worst case, equivalent to complete severance of the pipe in the IHTS, remains significantly less than the normal steady state operating temperature of 1340°F. Inadver- 8 tent opening of intermediate sodium dump valves is included in the overall plant duty cycle list that provides the basis for thermal transient design conditions. Since conditions resulting from occurrence of this duty cycle event can conservatively be the same as those of the worst case IHTS large leak, the reactor and heat transport system are designed to accommodate both events. For the case of a smaller leak that does not cause an immediate reactor scram, the maximum hot-channel coolant temperature of 1380°F is more than 300°F below the saturated temperature. 8

15.4 LOCAL FAILURE EVENTS - INTRODUCTION

Fuel failure is defined as a loss of fuel pin cladding integrity such that mass transport can occur across the fuel pin boundary. Mass transport implies the egress of fission gas or fuel and solid fission products from the fuel pin into the coolant or the ingress of sodium into the pin.

Local fuel failure implies a failure which is initiated within a single fuel assembly (which in CRBRP, consists of a bundle of 217 fuel pins surrounded by a hexagonal duct and includes axial blankets and a coolant orifice).

Pin to pin failure propagation would be defined as a failure in one pin initiating failure in an adjacent pin. Such propagation may be either self-limiting, in which case the damage is confined to a region of the pin bundle, or progressive, with the potential for involving the whole assembly. Assembly to assembly propagation would be a very remote occurrence of an assembly with an initial failure initiating damage in a neighbor assembly.

Examples of postulated local failure-initiating mechanisms are excess power in a single pin, insufficient flow within a fuel assembly, insufficient fuel pin heat transfer, and stochastic fuel pin failure. These events are in contrast to those considered in Sections 15.2, Reactivity Insertion Events and 15.3, Heat Removal Reduction Events, where the entire core is involved in the power increase or flow reduction.

This section shows that local failures, even if they were to occur, could lead only to minor disturbances which would be confined to the fuel assembly in which they occur. It is demonstrated that the effects of local failures would remain localized due to inherent characteristics of the sodium-cooled, mixed-oxide-fueled core and due to the ducted design of the CRBRP core assemblies.

It will be shown that pin-to-pin failure propagation would be very remote in CRBRP for an initiating event such as stochastic fuel pin failure or even for the postulated event of a small release of molten fuel or a postulated local flow blockage in the fuel assembly.

It is even more difficult to find a potentially realistic sequence of events which could cause damage to a neighboring fuel assembly. It is concluded that assembly-to-assembly propagation is highly improbable in CRBRP. The salient points which provide the basis for the position that local failures remain confined to the assembly in which they are postulated to occur, are discussed in detail in the following sections.

15.4.1 Fuel Assembly

15.4.1.1 Stochastic Core Fuel Pin Failure

A stochastic failure is a random pin failure that is unpredictable. Such a failure could result from a random cladding defect which goes undetected during manufacturing. Conservative design philosophy provides margins that minimize stochastic failures.

Stochastic failure of fuel pins is an anticipated occurrence for the CRBRP and such failures can be accommodated easily. Experiments in support of FFTF supplemented by supporting analyses have shown that for any postulated fission gas release mode, there are no serious thermal effects on adjacent pins or structures within the core. A transient jet of gas could not produce cladding overheating sufficient to cause failure of a neighboring pin, and even a steady jet could not cause failures because of the internal flow resistance in the failed pin. It has also been shown (Ref. 1 and 2) that gas blanketing could not cause significant cladding overheating. Furthermore, a volumetrically large gas release from a pin could not stop local coolant flow long enough to cause significant cladding overheating. Based upon experience with stochastic failures in other sodium cooled plants (Ref. 1), there should be no adverse mechanical effects from the expected mode of slow gas release through a small hole. For a postulated burst-release mode through a large rip in the gas plenum, no mechanical damage to the neighboring fuel pins or fuel assembly duct would occur. Finally, no adverse long-term effects of fuel pin failure are expected to occur even if some sodium logging of the fuel and some leaching of fission products would occur.

15.4.1.1.1 Prevention and Detection

The failure of a single fuel pin in a fuel assembly at nominal steady state full power conditions (975 MW) should not occur during the lifetime of the pin because of the margins in the design of the fuel and its cladding. The QA/QC procedures in the manufacturing process assure that the fuel pin will be fabricated in accordance with the design specifications. As shown in Section 4.2.1.3, the burnup goal of 80,000 MWD/T peak, hot spot temperature, fission gas release, cladding wastage and creep rates are applied simultaneously in the evaluation. To give an indication of the design margin against failure, the fuel rod peak burnup could be extended to 137,000 MWD/T for nominal design conditions. Furthermore, it was shown that pin failure should not even occur during the worst emergency heat removal reduction transient because the action of the Plant Protection System is sufficient to prevent failure of even the statistical hot pin i.e., the pin which has the highest temperatures with 99% confidence. For a stochastic pin failure to occur, a defect which goes undetected during manufacturing or a condition must be present which is outside that expected for the statistical hot pin. Should stochastic pin failure occur in either a rapid transient as cited above or a slow transient such as drift outside of the normal power range operating band, no molten fuel would be present because the transient is terminated by PPS action before incipient melting is reached in even the statistical peak power pin. The probability of the presence of molten fuel will be discussed fully in Section 15.4.1.2 where a postulated overpower pin is discussed.

Stochastic cladding failure should not occur due to reduction of heat transfer at the cladding surface. Deposition of foreign material on fuel cladding surfaces would not be expected because mass-transport in sodium systems occurs in the direction of hot surfaces to cold surfaces. Thus, material in the coolant should tend to deposit first in the cold end of the intermediate heat exchanger. Deposition at the inlet to the core would be of insignificant consequence since this is not a critical location for cladding integrity. In general, the cladding thickness may be reduced by coolant corrosion in the core but no material deposition will occur. The thickness of the cladding is chosen to ensure sufficient strength throughout life in spite of the cladding wastage process. Although the heat transfer coefficient of the pin outer surface would improve with wastage, it is not considered in the analysis because fuel-cladding chemical reaction may reduce the heat transfer coefficient at the inner cladding surface.

Stochastic pin failure may occur due to a random cladding defect which goes undetected during manufacturing inspection and/or localized random thermal, hydraulic, or mechanical conditions within the fuel assembly.

Even though stochastic fuel pin failure cannot be precluded it will be easily accommodated by the core with no change in operation and no loss in lifetime. An important goal of CRBRP is to achieve high burnup and this will be accomplished with the expectation that random pin failures could occur and can be tolerated easily. Fission gas release from failed fuel will be detected by a continuous, on-line gamma-energy gas analysis system (see Section 7.5.4, Fuel Failure Monitoring).

Fuel pin failures which exhibit only fission gas releases will not be removed from the core. The identified assembly is to be allowed to operate as long as additional failures will not mask the tag gas location system or obstruct detection capabilities. Because of the developmental nature of failed fuel detection systems, quantitative requirements cannot be delineated at this time. Such requirements can best be developed after evaluations based on the preliminary position (and development work) have been performed. It is noted that the fission product clean-up system is designed to accommodate 1% failures and thereby provides another limit.

The purpose of the development program is to upgrade the performance of detection and location equipment developed for FFTF to enable detection of leakers in the presence of a greater number of leakers and provide location of a larger number of core assemblies while accommodating a degree of simultaneous fuel rod gas leakage. 25

The reactor operational plan is to operate the reactor with failures of the "pin hole" type so long as fuel contact criteria are not exceeded and the cover gas activity is within specification. If the fuel failure rate is within the detection and location capability of the fuel failure monitoring system, the failed assemblies could be removed during a conveniently scheduled shutdown. If a point is reached where prior to a refueling shutdown, subsequent "pin hole" failures will exceed the detection and location capability of the failed fuel monitoring system but other criteria (technical specifications) are not exceeded, a decision will be made on an economic basis whether to shut down the reactor and remove known failures or to continue operation. If the latter course is chosen, and subsequent "pin hole" failures occur, locating all leakers would become more difficult and time consuming if it becomes necessary to do so. However, such action is not absolutely necessary unless other limiting criteria are reached. Whichever approach is taken, the fuel contact criteria and cover gas activity specification will not be exceeded.

If, therefore, the development program falls short of its quantitative objectives, depending on actual fuel failure rates, only an adverse impact on the amount of shutdown time required to locate and remove failed assemblies could result. 25

The CRBR is being designed for operation with limited amounts of failed fuel, as described in Chapter 11. Fuel or blanket rod failures which exhibit only fission gas releases will not be removed from the core as they present no safety problem provided that fission gas process limits are not exceeded. Fuel assembly failures having concurrent or subsequent indications of fuel exposure to sodium beyond a defined limit, are to be removed from the core. Development of this limit is dependent on the development of appropriate technology up to a limit consistent with applicable system bases and/or safety considerations.

15.4.1.1.2 Local Flow Reduction Due to Fission Gas Release

There are several potential mechanisms for adverse effects to adjacent pins resulting from the thermal effects of fission gas release from a failed fuel pin. These include flow reversal resulting from rapid gas release, gas jet blanketing of a neighboring pin, and downstream flow starvation. The flow reduction effects including flow reversal and downstream flow starvation will be discussed in the present section, while gas jet blanketing of a neighboring pin will be discussed in Section 15.4.1.1.3.

With regard to the effect of flow reversal resulting from rapid gas release, a conservative analysis was done to ascertain how long a fuel pin would have to be completely insulated before the failure temperature (assumed to be 1600°F as discussed in Section 15.1.2) would be reached. The length of time required for the cladding temperature to reach a specified failure point depends on the fuel pin power generation rate in the area covered by the gas, the size of the area covered, and the rate of cooling by the gas and any entrained liquid in the gas. An analysis was performed conservatively assuming that the gas surrounded the pin (360° angular coverage) totally eliminating heat removal. Calculations were made for the statistical hot pin (3σ hot channel factors) at 115% power, beginning of equilibrium cycle conditions, at which time the cladding temperatures and linear power rating have their highest values. The heat flux into the cladding was conservatively assumed to remain constant, even though it would actually decrease as the cladding temperature increased. The worst location was found to be at an elevation of 0.75 of the core height, and the time required for the cladding midwall temperature at that elevation to reach 1600°F was 0.12 seconds after gas insulation for these conservative assumptions.

Similar calculations were made at EOL (End of Life) conditions for the initial core design (burnup 80,000 MWd/Te) when the gas plenum pressure would be at its maximum value of about 800 psi. For this case, the cladding thickness was reduced to 0.010 inch, to allow for corrosion and fretting. For this case, the time required for the cladding midwall temperature to reach 1600°F was also 0.12 seconds because of the reduced linear heat rating. Therefore, at least 0.12 seconds of total pin insulation would be required before the cladding could reach 1600°F and before a further cladding failure might occur.

High internal gas flow resistances will prevent gas ejection rates from a cladding failure in the fuel region from being sufficiently rapid to cause flow reversal. The most severe case of gas bulk ejection would be for a large cladding rupture at the bottom of the fission gas plenum region at end of life. In this case, the only significant flow resistance would be that across the rupture. The formation of a gas bubble in the coolant and the transient flow behavior would be governed entirely by liquid inertia effects. Experiments have been conducted (Ref. 3) and an analytical model developed to allow evaluation of such an event. The experiment consisted of a 19 pin assembly with water as the test fluid. The device allowed rapid gas release and the measurement of inlet and exit flows in the test section. An analytical model was developed to allow extrapolation of the effects of rapid gas release to the 217 pin CRBRP fuel bundles. This model was based on the experiment and good agreement with the test results was obtained. The principal assumptions of the model are the following:

(A) The fission gas spreads uniformly over the entire fuel assembly cross section.

(B) The coolant is incompressible.

(C) The internal pin resistance to gas flow is negligible.

(D) The rupture is so large that its resistance to gas flow is negligible.

(E) The inertia and frictional effects of the fluid between the inlet and exit of the assembly are included.

The following parameters were used in the case of interest:

Initial Gas Pressure	1750 psia
Initial Coolant Velocity	21 ft./sec.
Rupture location	Top of upper blanket

The plenum pressure is the EOL value. The results of the cases analyzed are shown in Figure 15.4.1.1-1. For the case where 10 pins were postulated to rupture simultaneously, it was found that there was a small amount of flow reversal but that the lower gas-liquid interface did not reach the top of the fuel pins. A simultaneous rupture of 20 or more pins is required for the bubble to reach the fuel region. The calculations terminated when the upper gas liquid interface reached the top of the assembly. At later times, it would be expected that the gas bubble would be rapidly expelled out the top of the assembly and normal flow would resume. For simultaneous rupture of between 20 and 217 pins, this occurs in less than 0.05 seconds; for a simultaneous rupture of 10 pins, it occurs in less than 0.07 seconds. Similar calculations were made for the case of a single pin rupture. For this case there was no flow reversal. For all cases analyzed, flow reversal and gas blanketing was less than the 0.12 seconds required for the cladding temperature to reach 1600°F under worst conditions with complete insulation. Therefore, even if the break were to occur at the worst location in the fuel region and the internal gas flow resistance between the gas plenum and the break were neglected, the cladding on any adjacent affected pins would not reach the temperature at which further cladding failures might occur.

Experimental data are available and analytical models have been developed which show that the gas release rates from a failure in the fuel region of irradiated fuel pins are too low for downstream flow starvation or for gas jet blanketing to result in fuel failure propagation. Gas release data were obtained from an in-pile failed rod B3B (Ref. 4). This rod was irradiated in a natural circulation capsule in the GETR and failed at a burnup of about 18,000 MWD/T at a peak power of 22 kW/ft. The fuel column consisted of 90% T.D. (0.25 Pu-0.75U)O_2 pellets and was 23 inches long. Gas void space was 16.8 cm^3, cladding ID was 0.220 inches, and average as-fabricated cold diametral gap was 2.8 mils. Failure began about 7 inches below the top of the fuel column. Plenum pressure was 76 psia at the time of the failure, and total depressurization time was 4.5 hours.

Some additional out-of-pile data were obtained from two fuel rods, F3B-2 and F3B-4, irradiated in EBR-II to 5.5 a/o burnup at peak powers of 15.7 kW/ft. (Ref. 5). The fuel column consisted of 91% T.D. (0.25 Pu-0.75U)O_2 pellets and was 13.5 inches long. Cladding ID was 0.250 inches and the as-fabricated cold gap was 3.9-5.0 mils for F3B-2 and 3.8-6.4 mils for F3B-4. The F3B-2 cladding was punctured in the lower insulator region and the F3B-4 cladding was punctured 2.6 inches below the top of the fuel column.

An analytical model was developed based on the flow resistance in the fuel/cladding gap (Ref. 6) and showed very good agreement with simulated fuel pin flow data. This model also showed very good agreement with the B3B depressurization data when an effective diametral gap of 0.4 mils was used. For F3B-2, the calculated effective diametral gap was 0.4 mils while for F3B-4 it was 0.8 mils. The latter two effective gaps calculated for cold irradiated fuel rods would be expected to be even smaller during operation.

Another model was developed based on flow through a porous medium (Ref. 7, 8). This model showed very good agreement with the B3B data for an effective permeability of about 5.1 millidarcys, and with the F3B-2 data for a permeability of about 13.7 millidarcys.

These models and experimentally determined effective gaps or permeabilities were applied to the CRBRP advanced* design fuel pin at the end of life, with cladding failure at the bottom of the fuel column. Assuming the gas plenum and flow path at 1200°F, plenum pressure 1720 psia, and sodium pressure at bottom of the core 68 psia, the calculated fission gas (xenon) leak rates are:

Data	Equivalent Gap, mils	Porosity, millidarcys	Gas flow rate, lbm/sec.
B3B,F3B-2	0.4	-	1.29×10^{-5}
F3B-4	0.8	-	10.36×10^{-5}
B3B	-	5.1	0.51×10^{-5}
F3B-2	-	13.7	1.36×10^{-5}

*Initial fuel pin design is not planned to exceed 80,000 MWD/T burnup or approximately 1000 psi gas plenum pressures; advanced design refers to the fuel pin design for equilibrium conditions (i.e., burnups, temperatures, plenum gas pressures).

Flow starvation effects were analyzed for an FFTF fuel assembly which has a similar core geometry to the CRBRP (Ref. 9). The model includes coolant flow reduction resulting from two-phase flow in the accident subchannel, and conduction between the accident subchannel and the three neighboring subchannels. Conservatively assuming the gas to be confined to a single subchannel, it was found that the worst location for the leak was the bottom of the fuel column, with the maximum temperature occurring at the top of the fuel column. The calculations were performed for a subchannel with an inlet coolant temperature of 600°F and a normal exit temperature of 900°F, for various gas release rates. Scaling these results to the hottest 3σ + overpower equilibrium cycle BOL CRBRP assembly results in a limiting maximum cladding midwall temperature of 1600°F at an initial gas flow rate of 0.00107 lbm/sec. The maximum leak rate was previously calculated as 0.000104 lbm/sec. based on F3B-4 data, and would be an order of magnitude less based on the more prototypic B3B data. Thus, there is at least a factor of 10 margin between the acceptable leak rate and the maximum expected leak rate. These results were conservatively calculated based on BOL temperatures and EOL pressures. For the initial core design for which the maximum plenum pressure is expected to be about 800 psia, the margin is about a factor of 50.

The criterion for cladding failure was taken as a midwall temperature of 1600°F. This limit is a conservative criterion established for FFTF and based in part on cladding burst tests. However, a principal reason for using this limit is that data for the material properties of the cladding at temperatures greater than 1600°F are sparse. There is evidence (Ref. 2 and 10) that the cladding may withstand considerably higher temperatures during short term transients without failure occurring.

The above discussion demonstrates that pin failure propagation resulting from flow starvation is very unlikely. Furthermore, if neighboring pins were to fail, the failures would tend to be in the same subchannel as that in which the initiating failure occurred and so the failures would be self-limiting. Also, flow starvation failures would occur downstream of the initiation failure, and therefore, any propagation would terminate when the failure elevation reached the top of the core.

15.4.1.1.3 Gas Blanketing of Adjacent Pins

Another potential consequence of fission gas release is gas jet blanketing of adjacent fuel pins. A summary of some of the available experimental data which provide information on steady gas jet release and blanketing follows:

The effects of steady state gas jet blanketing of heated fuel pins have been experimentally determined in a sodium loop using three electrically heated pins (Ref. 11). The pins (O.D. = 0.23 inch) were arranged in an equilateral triangular configuration surrounded by a triflute shroud which, in turn, was surrounded by a cylindrical pressure chamber. Wire-spacers

(O.D. = 0.0602 inch), wrapped around the pins at a pitch of 12 in. were used for spacing. The wire wraps and the pins were provided with internal thermocouples and the angular orientation of the pins was such that the internal thermocouples faced the central coolant subchannel. For most of the test, argon, heated to 950°F, was released from a needle (I.D. = 0.023 in.) protruding through the test section pressure chamber and the triflute shroud, between two pins. The end of the needle was adjusted to be at a distance of $\sim$0.056 inch from the pin upon which impingement took place. The linear power was 7.1 kW/ft., system pressure was 58 psia, and the ratio of gas plenum to system pressure, P_g/P_s, was varied from $\sim$1.2 to 14. The tests were performed at steady state so as to eliminate the effects of flow and temperature transients. Check runs were also made at a linear heat rating of 14.0 kW/ft., needle I.D. of 0.013 inch and 0.033 inch, xenon instead of argon, and a gas temperature of 1328°F.

Earlier experimental data dealt primarily with gas release under operating conditions far into the sonic range. This work showed that for $P_g/P_s > \sim 3$, considerable spray formation takes place at the gas-liquid interface of the gas jet, so that cooling in the impingement area is predominantly due to coolant spray (Ref. 12 to 16). For $Pg/Ps < \sim 3$, spray formation takes place to a lesser degree and higher temperature increases are observed peaking at about $P_g/P_s \simeq 2$ and dropping again as the pressure ratio is reduced to $P_g/P_s \simeq 1.2$. The maximum cladding temperature increase is about 432°F at a linear heating rate of 14.0 kW/ft., and is proportional to heating rate. The maximum effect was observed for the hole diameter of 0.023 inch. For both the larger and the smaller holes tested, the temperature rise was reduced by about a factor of 2.

Applying these aforementioned data to the CRBRP fuel pin, it was found that the worst location for gas jet impingement is at an elevation of about 0.7 of the core height. At the worst location, for the worst hole size and pressure ratio at equilibrium cycle BOL (Beginning Of Life) conditions at 100% power for steady state gas impingement, the maximum (3σ hot channel factor) local (hot spot under wire wrap) cladding midwall temperature is approximately at the limiting value of 1600°F. It will decrease more than 200°F at EOL (End Of Life) conditions.

These results are at full reactor power and are very conservative because they neglect the internal resistance to gas flow. Under overpower conditions, this very conservative approach yields cladding temperatures about 1700°F, and therefore, a more detailed analysis was required to demonstrate that the cladding temperature would not exceed 1600°F even under 3σ plus overpower conditions. An analysis (Ref. 2) performed using the experimentally determined internal flow resistance for irradiated fuel pin F3B-4 which gave the highest gas flow rate, as discussed in Section 15.4.1.1.2, (using the more prototypical B3B data would reduce the gas flow rates by an additional order of magnitude). Calculations were made for 3σ plus 15% overpower conditions at various locations along the pin length. Cladding temperatures decrease during the pin lifetime. The time was determined during the pin lifetime when the local cladding midwall temperature with worst case jet impingement dropped to 1600°F. The gas plenum pressure at the time was used to calculate the gas flow rate to the failure location. For the worst hole diameter (0.023 inch) and pressure ratio ($P_g/P_s = 2$), the calculated flow rate was lower by a factor of ten than the flow rate required to give clad temperatures greater than 1600°F.

It was also confirmed by using conservative combinations of hole diameter, pressure ratio and time during pin lifetime that there was always at least a factor of 2 margin between the available gas flow rate and that required for gas jet blanketing to cause failure at 3σ plus 115% power conditions.

These calculations are all based on the internal flow resistance for irradiated fuel pins, for which the initial fuel cladding gap has closed. The HEDL-P-20 burnup data show that this occurs prior to 0.3 atom percent burnup. At beginning of life, the gas plenum pressure is high enough for jet blanketing to occur. However, the hoop stress in the cladding is less than 1000 psi. The jet blanketing transient can last no more than a few seconds so that the high strength of the 20% CW-316 stainless steel will not be annealed out. Although yield strength data are not available above 1600°F, extrapolating the existing data to 1700°F indicates that it is very unlikely that the neighboring pin would fail at the low cladding stress existing at beginning of life.

The preceding analyses were all based on steady state conditions. A transient analysis was also performed to determine the effect of the rapid decay of the plenum gas pressure through the narrow range over which blanketing was significant. It was found that this resulted in the 3σ plus 115% power cladding midwall temperature never exceeding 1600°F.

In summary: Gas jet blanketing is significant over only a narrow range of hole diameters approximately equal to 0.023 inch. Peak blanketing takes place over a narrow pressure ratio range of $\sim 1.3 < P_g/P_s < \sim 3$. Internal resistance to gas flow for irradiated fuel pins will prevent sufficient gas flow for jet blanketing to cause cladding failure at 3σ 115% power conditions. At beginning-of-life when internal gas flow resistance may be low, the low cladding hoop stress makes it very unlikely that a pin failure will propagate. Transient analyses show that an adjacent pin cladding is not likely to reach the limiting temperature for failure.

If failures do occur, they would tend to be self-limiting as the jets from subsequent failed pins would tend to be directed back to the initially failed pin.

As indicated above, there is always at least a factor of two between the available gas flow rate and that required for failure due to gas jet blanketing. If failure due to flow starvation effects are considered, there is at least an order of magnitude margin between the flow available and that required to potentially induce failure (Section 15.4.1.1.2). These conclusions apply for the cases in which the fission gas release is concentrated on a single pin or a single flow 25 channel. Therefore, if the fission gas from a single failure is dispersed, the margin between available gas flow and that required to produce unacceptable consequences must increase thus making a second failure even less likely. Experiments in 19 pin water cooled bundles showed that gas dispersion was fairly homogeneous across the test section (Reference 68). This substantiates for gas release the observations from dye injection and sodium nitrate injection experiments (References 69, 70 and 71) that there is significant cross flow and dispersion in wire wrapped bundles.

If a second failure is to result from an initiating pin failure, then it must occur in the immediate vicinity of the failed rod. Consideration of local gas blanketing or jetting of hot gases or flow starvation would indicate that the hottest portion of pins adjacent to a failed pin would face the failure pin. Therefore, if a second failure occurs, it would be expected to be in the same flow channel and potential damage would be directed back towards the initiating failure. In this latter case there is the potential for three failures in the pins associated with a flow channel. These basic considerations lead to the conclusion that fuel pin failures tend to be self-limiting.

There is no direct experimental confirmation that additional failures, if they should occur, would be self-limiting since all experiments (e.g., Reference 72) and operation to cladding breach have never resulted in a single additional failure due to fission gas release. 25

15.4.1.1.4 Mechanical Effects of Fission Gas Release on Pins and Duct Walls

Fuel pin cladding failure results in the coolant adjacent to the rupture being pressurized. Conceptually, at least, high internal pin pressure and a large rupture could allow a large pressure pulse to be applied to the coolant and fuel pins adjacent to the rupture. A discussion of the mechanical response of surrounding structures to a limiting-case pressure pulse follows.

An analysis was performed (Ref. 1) in which the magnitude of limiting loading for pin rupture was determined. The fuel pin was considered to be a thin-walled, simple-supported cylinder. This is conservative because the strength imparted by the supports to the rest of the tube are neglected. The presence of fuel inside the simulated cladding and sodium around the cladding were conservatively ignored. The conservative spatial variation assumed was axially uniform over the length and $P(\Theta) = P \cos\Theta$ for $-\frac{\pi}{2} \leq \Theta \leq \frac{\pi}{2}$, and $P(\Theta) = 0$ for $\frac{\pi}{2} \leq \Theta \leq \frac{3\pi}{2}$, i.e., pressure on only one side of the tube. The pressure was also conservatively assumed to instantaneously rise to P_0 and to decay according to $P(t) = P_0 e^{-\alpha t}$.

The cladding tube could rupture if the buckling moment, M, is greater than the critical buckling moment, M crit. It was found that the exponentially decaying pressure pulse would cause the applied moment to exceed the tube critical buckling moment only for high initial pressures and slowly decaying pulses. Figure 15.4.1.1-2 shows the condition for which the critical moment could be exceeded. As an example, an initial pressure of 500 psi and time constant 1.25 milliseconds or larger might cause tube rupture (M > M crit).

The pressure available may be inferred from out-of-pile cladding burst experiments conducted in connection with EBR-II (Ref. 17,18 and 19). In these tests, a limited volume of high-pressure gas was released in a standard EBR-II hexagonal duct. Transient pressures were usually measured inside the tube and inside the duct which was immersed in a drum of water. Duct deformations were measured after each test.

One of the conclusions that may be drawn from these tests is that the ratio of the peak pressure inside the duct to the initial pressure inside the tubes was usually much less than unity (typically ~0.2). An analytical model (Ref. 20) was developed in which all of the resistance to gas flow from the pin was assumed to occur at the rupture. This corresponds to a cladding failure in the plenum region and is very conservative if a failure in the fuel region is being considered. The gas bubble was assumed to be spherical and expanding within an infinite sea of incompressible liquid. This simplified approach neglects the effects of the solid surfaces present. The model showed very good agreement with the experimental data in determining the peak bubble pressure, although it predicted more rapid pressure decay after the peak was reached, presumably because of the neglect of the solid surfaces. The peak pressure decreased slightly as the rupture area was increased from 0.02 to 0.2 in^2, and increased slightly as the plenum gas volume was increased from 4 to 26 cm^3. The main effect was that of initial gas plenum pressure, and the relation between peak bubble pressure and initial gas plenum pressure is shown in Figure 15.4.1.1-3. CRBRP will have a maximum fuel pin pressure of ~800 psia for the initial core design and ~1720 psia for the advanced core design. Hence, gross cladding rupture in the gas plenum region would generate peak pressures of about 180 psi for the initial core design and about 300 psi for the equilibrium core. 29

The mechanical strength of the duct was conservatively evaluated for the limiting case of rapid fission gas release. The maximum duct temperature at end of life is about 1100°F for the initial core design and about 1000°F for the equilibrium core. Since irradiation of stainless steel
causes an increase in strength but a loss in ductility, properties of fully 29
irradiated (~2 x 10^{23} n/cm^2) 20% CW 316 SS were used. The stress-strain curve shown in Figure 15.4.1 1-4 was derived (Ref. 21) by extrapolating to the goal fluence of 2 x 10^{23} n/cm^2 using data from samples irradiated to about 10^{22} n/cm^2. The material samples were tested at very low strain rates (2 x 10^{-3}/min.) in the condition considered. Since higher strain rates yield improvements in both ductility and strength of unirradiated CW 316 SS (Ref. 22) and as it is expected that similar behavior could occur for irradiated CW316SS, the stress-strain data used here are considered to be very conservative. It is seen that the ultimate tensile strength and corresponding strain are 42,000 psi and 1.4%, respectively.

For the present analyses, it is assumed that the ultimate tensile strength is the failure threshold for the duct even though reaching the ultimate tensile strength at the duct surface would not necessarily result in rupture.

The analysis of the structural response of the similar FFTF duct was performed with the ANSYS code (see Appendix A of this PSAR).

To calculate the strength of the duct to withstand internal pressure, a two-dimensional model was constructed using elastic-plastic beam elements. Assuming spatial uniformity of the internal pressure loading, one-twelfth of the hexagon was modeled which was comprised of one-half of a duct flat and one-half of a corner. The validity of using a two-dimensional model was established by testing a 3-D model with an azimuthally uniform, axially-rectangular pressure distribution. It was concluded that slightly conservative results were obtained by neglecting the axial dependence of the pressure distribution. The additional strength from surrounding sodium and fuel assemblies was neglected although the fuel assemblies may be well-coupled by the sodium in the interassembly gaps (nominal gap is 0.14 inch) during rapid dynamic loading. The following results were obtained: (a) for a maximum duct temperature of 1000°F, the duct can withstand a steady uniform internal pressure of about 550 psi and (b) for a maximum duct temperature of 1200°F, the duct can withstand a steady uniform internal pressure of about 300 psi. Since the peak duct pressure for the initial core is about 180 psi with a maximum duct temperature of 1100°F, and the peak duct pressure for the equilibrium core is about 300 psi with a maximum duct temperature
about 1000°F, duct failure would not occur. 29

The deformation of the duct and, in particular, the deformation at the center of the duct wall would be significantly less (less than 0.02 inch for the 1200°F case) than the distance between adjacent ducts (0.14 inch) so no contact would occur between two ducts as the result of a pressure pulse due
to a loss of fuel pin cladding integrity. 29

Since the strength of the duct was calculated for an axially uniform loading and the gas pressure from a failed pin could only be applied over a very short axial length, a large amount of conservatism is built into the calculations. Again, it should be reiterated that cladding failure would be expected to result only in small ruptures, slow gas release, and no significant increase in local coolant pressure. Thus, the conditions above are limiting extreme cases and duct failure, therefore, is not expected to occur.

Because of the high acoustic velocity in sodium of the order of 7000 to 8000 ft. per second, a pressure pulse with a risetime of the order of microseconds would be required to develop a significant pressure differential across the pin. The pressure pulse risetimes found in the EBR-II duct tests were several hundred microseconds and therefore no appreciable pressure differential across the pin is expected from this mechanism. However, the case was considered of a force acting on a pin as a result of a fission gas jet impinging on the pin. Assuming the jet to be deflected at right angles, the maximum impulse imparted to the adjacent pin is 0.258 lbm-sec. for a gas plenum pressure 1720 psi and 0.120 for a gas plenum pressure of 800 psi. From the results shown in Figure 15.4.1.1-3, and assuming that the maximum moment with a concentrated dynamic load at the center of a simply supported beam is twice that for the same load uniformly distributed, it was determined that the critical bending moment is reached for an impulse of 0.318 lbm-sec. This is higher than the maximum impulse which can be imparted to an adjacent pin in either the initial core or in the equilibrium core. Thus, no additional pin failures would occur as a result of the mechanical effects of fission gas release.

Assuming that gas does not communicate with the plenum, the gas pressure within the fuel column would be significantly higher than used in design calculations and the cladding would fail at higher pressures than those assumed for the stochastic pin failure analyses. This does not affect the previous evaluation of steady state gas jet blanketing of the adjacent pin because that evaluation was made for the internal pressure which resulted in the maximum cladding temperature increase of the adjacent pin. Higher pressures result in lower temperature increases.

For the present case, with the gas at a higher pressure in a smaller volume, the transient temperature resulting from gas jet blanketing of the adjacent pin will be lower than those calculated previously.

The maximum gas jet impulse available from a failed fuel pin is not affected by having the same mass of gas at a higher pressure in a smaller volume. Therefore, the previous conclusion reported in this section, that no additional pin failures would occur as a result of the mechanical effects of fission gas release, is still applicable.

It is concluded that if the fission gas released in the fuel region does not communicate freely with the gas plenum, the probability of cladding breaches occurring would increase, but the thermal and mechanical effects of the cladding breach would be accommodated and not result in rod to rod failure propagation.

Since cladding breach can be accommodated without rod to rod failure propagation, normal operation policy stated in Paragraph 4.2.1.1.3.8 will not be affected by the assumed conditions. 16

15.4.1.1.5 Effects of Fuel Particle Release

Fuel particle release from failed fuel could:

a. contaminate the primary system with non volatile fission products, plutonium and uranium, and

b. potentially form coolant channel flow blockages which could result in overheating in the fuel rod bundle. Experience has shown this to be exceedingly low in probability.

Contamination of the primary system sodium by non volatile fission products,uranium and plutonium presents a potential radiation and health hazard, as well as a cleanup problem. Preliminary indications are that cladding defects (in the fuel zone region) of 0.1% of the fuel rods during the entire 30 year plant life would result in an end of life plutonium concentration of 0.1 ppm in the primary system sodium.

Section 4.2.1.1 evaluates the reactions between the fuel or blanket materials and the sodium coolant. These reactions form a product which is much less dense than the original fuel or blanket pellets. Fuel-sodium reaction products can cause small, partial flow blockages by either expanding the cladding to a larger diameter or depositing on the outside surface of the fuel rod. The consequences of a partial flow blockage with a heat generating material on fuel rod cladding temperatures is presented in Section 15.4.1.3.4. Effects of a cladding temperature increase on reducing the cladding lifetime is discussed in Section 15.4.1.3.6.

Irradiation experience with mixed oxide fuel pins in LMFBR's or liquid-metal cooled test capsules encompasses more than 21,000 pins. Approximately 1% of these pins have failed (Ref. 60). In every case, there was no evidence of local blockage formation resulting from fuel particle release through the cladding breach. With few exceptions, the failures have been small pin-hole failures - intergranular cracks typically several microns in diameter. Because the potential for fuel release is directly related to the defective geometry, fuel particle release has not occurred through such small clad breach sizes. In an experiment with an intentionally defected pin (with a large hole size of 0.005 in.) in an EBR-II test assembly, there was no evidence of fuel release to the coolant upon pin failure (Ref. 61). Even if it is postulated that fuel particle release was to occur, the particle size would be limited by the breach size, which experience has shown to be on the order of several microns. Hence the particulates would be much smaller than the minimum dimension of the fuel pin bundle (0.055 in.) unless an atypical major split or opening is postulated. These particles would be easily swept out of the pin bundle by the coolant and not become trapped in the assembly.

One instance where a gross fuel failure has been observed is in the BR-5 reactor (Ref. 62). In this case, the central pin of a 19 pin assembly was cracked along its entire length (280mm) on opposite sides of the pin. The large failure was attributed to excessive burnup for the design of that particular pin. Even in this exceptional case, although some extrusion of the fuel into the cracks was noted, there was no evidence of blockage formation in the flow channel nor was there any damage to the neighbor pins.

Most of the existing data on fuel pin behavior following failure is for short term consequences (on the order of hours to a few days after the failure). Experience with fuel performance for long periods after failure has occurred is quite limited. Pins containing fuel failures have operated in Dounreay Fast Reactor for over 100 days with little or no deterioration and no evidence of blockage formation (Ref. 63). Failures in Rapsodic and BR-5 driver fuel have also occurred and the fuel has remained in the reactor for periods longer than 100 days with no deleterious effects. 9

Evidence for the pinhole nature of stochastic cladding failures in mixed oxide fuel pins is presented in References 75 and 76. Table 15.4.1.1.5-1 summarizes the fabrication details and operating conditions of the failed pins described in the references, and the conditions at failure. Further discussion of each pin failure is included below. The run-to-cladding-breach tests confirm that:

1) a cladding breach has no deleterious effect on the assembly hardware or on the neighboring fuel pins,

2) the leakage of fission gas through the breach is slow and gradual,

3) the fissures appear to form after considerable plastic deformation of the cladding,

4) there is no evidence that sodium-fuel contact is an immediate consequence of a cladding breach.

Pin PNL 5-1: Three reactor startups were required to confirm removal of the assembly with the breached pin. Disassembly revealed no evidence of failure propagation. Several pins along one side appeared to be twisted and the failed pin was one of these. Even after the failure had been located by internal pressurization it could not be seen on the outer surface at a 10X magnification. Measurements indicate an inelastic strain of 1.1% and a cladding thickness reduction of about 3% at the location of the breach. The nominal cladding temperature was too low to account for the observed magnitude of creep and the observed twisting is believed to have resulted in pin-to-pin contact resulting in a hot spot on the cladding resulting in a local weak spot sufficient to allow localized creep deformation. However, examination of the neighbor pins showed no evidence of deleterious consequences from the failure. Also no evidence of sodium ingress through the failure was found, even though the pin underwent two startups during identification of the assembly with the breached pin.

Pin PNL-10-14: The failed pin underwent 46,500 Mwd/MTM burnup in a 61 pin assembly (X093) before reconstitution into the 37 pin assembly PNL-10 (X193). Failure occured at 63,800 Mwd/MTM. The pins and wire wrap of the original assembly showed considerable wear suggesting a loose fit in the assembly hardware; the least worn pins were used in the reconstituted assembly. After the location of the failure had been identified by internal pressurization, the breach could not be seen on the outer surface. The fissure was in a wear mark. Allowing for wear, severe work hardening and internal intergranular attack, the thickness of unaffected cladding at the break location was reduced to 0.005 inches. The axial length of the fissure was less than 0.008 inches. Since the uniform inelastic strain was essentially zero, and the gas pressure at failure was low, it was concluded that some anomaly existed in the material in order to cause the pin-hole fissure. There was no evidence of axial propagation, sodium ingress or consequence to the neighbor pins or assembly hardware.

Pin HEDL-N-E-122: Subassembly HEDL-N-E (X191) consisted of 37 pins which received pre-irradiation in the 61 pin assembly X096 up to 24,000 Mwd/MTM burnup. Pin breach occurred at 42,300 Mwd/MTM. Internal pressurization located the breach at 0.06 inches below the top of fuel column, but it could not be visually detected from the outside. A region of localized intergranular porosity and cracking was found (20° arc wide and less than 25

0.5 inches long). Examination of this region has not been completed so the extent of this region cannot be clearly defined. The porosity indicates localized creep deformation of unknown cause but perhaps caused by higher-than-expected temperatures. It is noted that the breach had no effect on neighboring pins, neither was there any sodium ingress into the failed pin nor indication of rapid extension of the initial fissure.

Pin P-12A-63K: The 19 pins of subassembly P-12AA (X186) were pre-irradiated in the 37 pin assembly X150 to a burnup of 23,000 Mwd/MTM. The reconstituted assembly included pins with 10, 20 and 30% cold worked 316 stainless steel cladding and plenum volume to fuel volume ratios of 0.62, 0.82 and 0.98. Cladding breach occurred at a calculated burnup of 35,000 Mwd/MTM. The 30% cold worked clad pins showed unusual diameter increases, the increase being proportional to the pressure (i.e. inversely proportional to the plenum volume). The breach (in the 30%C.W., 0.62 plenum/fuel ratio pin) could not be found by internal pressurization and eddy current tests were used as the basis of further examination. Recrystallization of the 30% C.W. in the vicinity of the suspected breach location was noted. Maximum recrystallization is adjacent to the hottest channel in the assembly. The local bulge in the cladding demonstrates a high ductility of the irradiated cladding, and there is no apparent tendency for the fissure to propagate. The failure did not impair performance of the neighbor pins, nor the subassembly.

Pin PNL5-17: The 19 pins in assembly PNL5B (X116B) were previously irradiated in the 37 pin assembly X054 to 50,000 Mwd/MTM. Failure occurred at a calculated final burnup of 140,000 Mwd/MTM after reconstitution. The failed pin was identified by weight loss (that could be attributed to fission gas release) and gamma scan. Detailed examination has not started and the location of failure has not been identified by visual examination. This supports previous observations that breaches are microscopic.

Pin P-12A-11B: The 19 pin assembly P-12AB (X213) was reconstituted from the 37 pin P-12A (X150) assembly at 70,000 Mwd/MTM burnup. Pin failure occurred at 74,000 Mwd/MTM. A delayed neutron signal caused reactor shutdown after a small increase in cover gas activity. Severe distortion of the wire wrap spacers was found permitting contact between the failed pin and an adjacent pin. Detailed examination is just beginning. The failure has not been located but a suspect region has been visually identified in the area of contact with a neighbor pin. There is, nevertheless, no indication of any effect of the failure on the adjacent pin other than a small area of discoloration. The wire wrap loosening was caused by a mismatch in neutron-induced swelling between the annealed 316 stainless steel wire and the cold worked 316 cladding. This is not expected to occur in CRBRP since 20% cold worked 316 is used for both the cladding and the wire wrap spacer.

Pin PNL-11-39: The 37 pin assembly PNL-11 (X194) was reconstituted from the 61 pin assembly X107 at a burnup of 46,000 Mwd/MTM. A delayed neutron signal caused reactor shutdown 50 minutes after the first indication of failure. Three startups were required to identify that the assembly with the failure had been withdrawn. At reconstitution of assembly PNL-11 it was noted that there was severe wear of the pins. Those with the least wear were incorporated into the new assembly. After pin breach it was determined that the wear had continued. The failed pin was identified by weight loss (which can be attributed to fission gas) and gamma scan. Visual examination showed what appeared to be an axial crack in one of the deep wear marks; this was the only sign of cladding breach. Neighboring pins show no effects of the breach. Detailed examination has just started.

Pin X106-018: The seven pins in this assembly were originally in the 19 pin assemblies X040 and X040A. Assembly X040 was reconstituted as assembly X040A at 3% burnup and seven of the pins from the inside rows were formed into X106 at 6% burnup. Examination of the failure suggests that some polyvinyl chloride (PVC) was inadvertently included in the pin during fabrication. This is further supported by the unlikely location of the failure (near the upper tantalum disk) and the fact that many pins have exceeded the failure burnup without a breach occurring. The crack was about 0.0004 inches at the inside surface and about 4.5 microns at the outside as estimated from micrographs.

Conclusions: The range of cladding materials, variation in the mixed oxide pin conditions and the degree of reconstitution in the pins which failed lend strength to the argument that stochastic breaches will be small or microscopic failures. In no case was there evidence that the breach impacted the operation of the adjacent fuel pins or assembly let alone caused pin failure propagation. Even in those instances where pin-to-pin contact occurred, the initial breach did not cause failure of the adjacent fuel pins. 25

Based on this data, our statement that "experience has shown this to be exceedingly low in probability" is considered valid.

9

In Section 15.4.1.3 it is shown that the force of flowing sodium is enough to sweep any reaction products which will fit into a core channel, out of core. Finally, if excessive amounts of fission products were to enter the coolant, their presence would be detected by the Delayed Neutron Monitors and corrective action would be initiated. Therefore, no adverse consequences are expected for fuel particle release.

15.4.1.1.6 <u>Long Term Effects of Operation with Failed Fuel</u>

In preceeding sections, attention was given to the consequences of fission gas release and the intermittent escape of fuel particles through the cladding rupture. In this section, the possibility of long term effects due to pin failure will be examined. It will be shown that there is very little possibility of any long term adverse effects.

Because of failure of the cladding, sodium could enter the pin due to reduction of gas pressure upon cooldown accompanying reactor shutdown. Various adverse effects of sodium absorption (or logging) of the fuel has been postulated as follows:

a. Leaching of fission products from the fuel by the sodium and subsequent deposition in the primary system.

b. Generation of high pressure inside the pin due to sodium vapor, possibly causing further cracking or disintegration of fuel.

c. Fuel pin swelling due to fuel-sodium chemical reaction.

If some sodium were to enter the fuel then some fission product leaching might occur. It would not be expected that the effect on the pin would be significant. The sodium cleanup system could remove these products and no significant adverse effects could occur. If the primary coolant activity became high, the reactor could be shutdown and the fuel assembly with the failed pins would be removed. The generation of sodium vapor pressure inside the pin is very unlikely since the pressure can be relieved by the rupture which allowed the sodium to enter originally. The absence of a problem due to vapor formation is supported by sodium logging experiments (Ref. 23) in which defects were simulated by 0.005 inch diameter holes in the cladding and fuel cladding gaps were of the order of 0.002 inch to 0.003 inch. Fuel rods were thermally cycled employing heating rates 200 times faster than those expected in bringing a reactor up to power. No evidence of fuel pin deterioration caused by the sodium was found. There is no reason to believe that any problem should arise due to sodium vaporization within the failed pin.

The degree of the fuel-sodium chemical reaction and the extent of the resultant fuel pin swelling depend primarily on the quantity of oxygen available for reaction. The secondary effects of temperature and fission product concentration also contribute. The quantity of oxygen available for chemical reaction is dependent on the initial fuel fabrication O/M ratio, the original oxygen impurity level of the coolant, the net oxygen liberated as a result of fissioning and recombination with fission products, the quantity of oxygen leached from the fuel into the coolant, and since the reaction occurs mainly at the fuel surface, the degree of migration of oxygen due to fuel temperature gradients. It is not expected that fuel swelling of a sodium logged pin would be a problem in the CRBRP for the following reasons:

a. The swelling should be localized axially near the cladding hole so any reduction in flow area would be localized and random.

b. Spacing between pins will be maintained by the wrapper wires.

c. The flow reduction due to random swelling would be small.

d. The theoretically predicted uniform linear expansion of CRBR fuel and axial blanket pellets would only be 1.7% and 0.7% $\Delta D/D$, respectively, for extreme reaction conditions after a failure late in life (See Section 4.2.1.1).

However, projected values of oxygen in sodium and fuel or axial blanket materials are higher than experimental equilibrium values so that a stable reaction product might be formed.

With regard to the effect of chemical interaction of sodium and fuel on the fuel structure, the compatibility of liquid sodium with mixed oxide fuel has been studied extensively and found to be a function of fuel stoichiometry, fuel density, sodium purity, and temperature. Hypostoichiometric fuels in contact with clean sodium have been reported to be completely compatible. The initial stoichiometry of the CRBRP mixed oxide fuel is approximately O/M = 1.96. However, the overall ratio may vary along the radius of the fuel pellet and will vary with time due to the burnup of the fuel and the creation of the fission products with different affinities for oxygen. Therefore, it may be concluded that the presence of sodium in contact with irradiated mixed oxide fuel does not appear to result in significant chemical damage to the oxide fuel rod.

While operation with failed fuel could be deleterious, it is expected that with proper control and additional testing and development, satisfactory operation will be confirmed. Finally, it should be noted that many fuel pins have operated in fast reactors for periods up to 2-1/2 years after failure without reported difficulty or adverse effects on reactor operation (Ref. 1 and 25).

15.4.1.2 Overenriched Fuel Rod Failure

An overpower rod is one which contains pellets of enrichment higher than the design value and/or pellets of an enrichment intended for a low flux core region but erroneously loaded in a high flux core region. The possibility that such an overpower rod could actually exist is made extremely small through design features in the core components and through control of the fabrication of the core components and the fuel material. The methods used to prevent core loading errors are described in more detail in Section 4.2.1.2.

15.4.1.2.1 Prevention and Detection

Although detailed specifications for the fuel pellets have not been prepared, it is expected that enrichment specifications and fabrication procedures will be similar to those for FFTF fuel as described below.

The main features which prevent loading of overenriched fuel may be briefly summarized as follows:

Fuel pellets with enrichments differing from normal by more than about 3.5% are extremely unlikely (probability less than 10^{-4}) because of the stringent tests and overchecks required during fabrication (Ref. 26 and 27).

Pellets are loaded at the vendors' sites into cladding with end caps which have mechanical keys to prevent insertion into the wrong fuel assembly. (See Section 4.2.1.2.3.)

Each fuel assembly inlet nozzle has a "discrimination post" which positively prevents it from properly seating in a core region in which it could be undercooled. (See Section 4.2.1.2.3.)

In addition, before a fuel assembly is inserted into the core, the enrichment and flow orifice type are determined by the In Vessel Transfer Machine (IVTM) by interpretation of the identification notches in the handling socket that are arranged in a unique pattern for each fuel assembly.

It is extremely unlikely that several of these measures would fail concurrently thus resulting in an overpower fuel rod. The probability that a defect in the fuel due to improper fabrication could go undetected has been estimated at less than one chance in a thousand (Ref. 28). Since the probability of a rod having an enrichment error of over 3.5% is less than 3 x 10^{-3}, the probability of this occurrence without detection is about 3 x 10^{-6}. However, even if all of the above measures fail to prevent loading of overenriched fuel into the core and rod failure occurred, the failure would be detected and monitored by several methods.

Fission gas released from a failed fuel rod into the cover gas plenum is detected by the Cover Gas Monitoring System.

35 Tag gas released from a failed fuel rod is analyzed by the Cover Gas Tag System Mass Spectrometer.

Primary coolant sodium transported past the delayed neutron monitoring system located in the outlet piping, is continuously monitored to detect delayed neutrons emitted by the decay of radioactive fuel material in the sodium.

15.4.1.2.2 Consequences of Pin Over-Enrichment

An analysis discussed in Section 15.4.1.2.2.1 demonstrates that pins with maximum credible over-enrichment placed into a peak power location may not fail immediately under severe conditions but are marginal for long-term operation and are likely to fail during normally expected transients. The study with over-power as a parameter shows at what level the fuel melting is reached and the extent of the molten zone. Based on the P-19 data correlation, discussed in Section 4.4.2.6, the CRBRP statistical hot pin requires an over-power of greater than 25% to reach the fuel incipient melting point. P-20 data for low burnup pins indicates even greater margins by approximately 20%. Thus, almost 45% margin exists after very low burnup. Figure 15.4.1.2.2-1 illustrates the first small amount of melting. If the power level were held at this level, the molten fuel would drain downward in the central void and resolidify. At higher power levels, a range is reached where some fuel which melted near the midplane will remain molten even after draining toward the bottom of the void. This situation is also illustrated schematically in Figure 15.4.1.2.2-1. Since the neutron flux is much lower at the bottom of the core ($\sim$0.7) compared to that at the midplane ($\sim$1.22), the overpower level required to maintain fuel in the molten condition is much higher than that required to reach incipient melting at the core midplane. Hereafter, this condition will be called "stable molten fuel" to indicate that fuel has relocated and although some has solidified, some is still molten.

15.4.1.2.2.1 Threshold Power for Stable Molten Fuel

Fundamental to the study of an overpower pin is knowledge of the amount of molten fuel which might exist as a function of power. These variables depend upon fuel thermal parameters such as the conductance of the fuel/cladding gap, the thermal conductivity of the fuel, (which depends on its density and microstructure), the melting temperature, and fuel restructuring. Molten fuel is assumed to slump to the bottom of the central void where some of it may again solidify and thus reduce the molten fuel inventory. The volumetric increase which accompanies fuel melting is
44 neglected in the calculation.

There is experimental evidence to support the basic assumption
44 that the molten fuel will undergo slumping. Post-irradiation examination
of the EBR-II P-19 pin showed evidence of slumping after the pin was
taken to about 20 kW/ft. maximum linear power (this is 40% above the max.
pin power). Evidence for the formation of fuel bridges has also been
44 found. These are illustrated in Figure 15.4.1.2.2-1.

To find the power level at which molten fuel could exist in an
equilibrium state in the form of a pool at the bottom of central void
44 (stable molten fuel), a calculation was performed using the following
parameters to ensure that the molten fuel inventory calculation was
pessimistic:

Fuel Melting Temperature at EOL	= 4850^{o}F
Gap Conductance	= 1100 BTU/HR.-Ft.^{2}-^{o}F
Redistributed Fuel-Cladding Volume	= 0

Fuel Thermal Conductivity Correlation is according to the code "LIFE" (see Appendix A)

The model used assumed that fuel slumping could occur when the
solidus temperature is reached. The solidus temperature depends on the
burnup and is about 5000^{o}F at BOL and about 4850^{o}F at EOL. The calcula-
tion used the latter value to maximize the possible amount of molten
fuel. The fuel cladding gap conductance used was 1100 BTU/Hr.-Ft.^{2}-^{o}F
based on P-19 pin data. The fuel-cladding gap will decrease and close
as the fuel undergoes restructuring, swelling, cracking, and other effects
as it is cycled throughout its life, and it will come into contact with 9
the cladding, thus improving the fuel-cladding gap conductivity. In
addition, the volume that existed initially between the fuel pellets and
the cladding reappears as a void in the center of the fuel pellet and/or
as porosity in the outer "unrestructured region". The proportional
division between the additional amount of central void and porosity in
the outer region is not known. However, the larger the amount of void
added to the central cavity the lower the fuel temperatures would be,
thus, the most pessimistic assumption is that none of the fuel-cladding
gap void appears in the central cavity (0% gap redistribution), and
that it is all manifested as porosity (perhaps as cracks) in the unrestruc-
tured region. The fuel thermal conductivity correlation used in these
calculations was a conservative one compared to the one presently
recommended.

Using the conservative parameters discussed above, it was found that the calculated threshold power for stable molten fuel was about 16.8 kW/ft. at the axial midplane (Figure 15.4.1.2.2-2). Since the expected value of maximum power for the peak power pin in the core at 975 MWT is 12.5 kW/ft., this corresponds to an overpower margin of 34%. These results are in agreement with FORE-IIW and NICER code check calculations. These codes use EBR-II P-19 results, and are described in Appendix A.

To illustrate some of the conservatism of the parameters chosen to evaluate the threshold of stable molten fuel, it is noted that, if after restructuring has occurred, the entire fuel-cladding gap has redistributed to the central void (100% gap redistribution), the threshold midplane power level required for stable molten fuel would be about 19.2 kW/ft. or an overpower of nearly 50%. In addition, the P-20 test data indicate that an additional margin to melt of about 20% exists at low burnups. It was calculated that a fuel pin with outer region enrichment placed into the inner region of the core would generate 45% more power than a normally enriched pin, (1.45 x 12.5 kW/ft. = 18.1 kW/ft. peak). Allowing for the usual power margin, a maximum linear power of 20.3 kW/ft. would be estimated for the pin misplaced into the peak location. At this loading Figure 15.4.1.2.2-2 indicates that without gap redistribution over 20% of the fuel mass will be molten, and about 8% with fully redistributed gap. Test data show that under short term, steady state conditions, fuel pins can operate with 25% to 30% of the cross sectional area molten without failure. Thus, a postulated fuel pin loading 13 error will not lead to immediate failure but there is less safety margin to withstand transients. However, the P-20 power to melt data indicates the margin to failure may be higher than assumed in this report.

The implications of long term operation of fuel with melting can be examined in the light of experimental evidence obtained with oxide fuel (References Q001.332-1 and Q001.332-2). Reference Q001.332-1 describes experiments in which as much as 37.5% volume fraction of fuel melted without cladding failure. Although extended operation did not occur in these experiments because their purpose was to measure fission gas release, it was demonstrated that significant melting could occur without cladding failure. Reference Q001.332-2 refers to irradiation tests during which mixed oxide fuel was operated with substantial molten fuel. Volume fractions of molten fuel up to 27% are reported. Irradiation proceeded up to burn-up levels between 3.5 and 3.7 atom percent. Post-irradiation examination showed that two of the elements that operated at ratings over 22 Kw/ft failed. The five others, ranging from 19.6 to 21.4 Kw/ft performed without failure. Post irradiation assessments indicated that the central void in the elements was filled with molten fuel at the end of irradiation. This is due to thermal expansion and volume expansion on melting of the fuel. It is also reported that during irradiation, the temperature of the fuel was cycled through the melting point 57 times. The experience described above, along with other experience with melting such as in Reference Q001.332-3, supports the conclusion that center melting is likely to be without consequence in oxide fuel.

There are no on-going tests which are specifically designed to obtain further data on operation of fuel with center melting. The experimental results quoted, along with the information in Section 15.4.1.2.3 (which indicates that release of molten fuel from a pin would not result in assembly-to-assembly failure propagation), provide confidence that the consequences of an overpower pin are acceptable.

15.4.1.2.3 Thermal Effects of Postulated Molten Fuel Release

Despite the above analysis, it was assumed that cladding failure of an overpower pin could occur with the consequent release of molten fuel. The likelihood that the liquid fuel would be released in the form of a jet which would hit a neighboring pin is again remote, because the fuel would be fragmented when hitting the molten sodium (See Section 15.4.1.2.4). Data compiled from numerous tests (Ref. 29) show resulting particles to be mainly in the range of 0.004 (100 microns) to 0.04 (1000 microns) inches. These can be swept out at sodium velocities of 2 feet per second, since they are smaller than the channel size (~0.07 inch). Nevertheless, analyses were made assuming that fragmentation does not occur and that fuel will solidify on colder metal surfaces.

If molten fuel is postulated to contact a duct wall, the duct surface temperature would increase rapidly. The resulting temperature difference across the duct wall would induce high thermal stresses. Whether this would lead to mechanical failure of the duct depends on a number of parameters, including the duct cold wall temperature (on the opposite surface from that in contact with the fuel), the pressure differential across the duct wall. and the temperature differential across the duct wall. Figure 15.4.1.2.3-1 from Reference 67, shows this relationship. Failure is assumed to occur when the ratio of the moment to the ultimate plastic carrying capacity, ζ equals 0.9. 25

The full pressure developed by the pump at full flow is approximately 155 psi. The pressure differential across the duct wall varies with axial location, but would be less than 155 psi at all locations. It can be seen (by slight extrapolation) that for a pressure differential less than 155 psi, melting is the probable failure mechanism for cold wall temperatures less than 1050°F. For higher cold wall temperatures, mechanical failure could occur at a lower duct ΔT than would cause duct melting. 25

A. Flow Reduction Effects

It was pointed out that molten fuel undergoes fragmentation when it contacts liquid sodium. Once fragmentation occurs, rapid heat transfer from the particles to the sodium becomes possible with rapid heating of the coolant and rapid cooling of the particles. If the coolant saturation temperature is exceeded, sodium vapor will form and the coolant channel pressure may increase. Since the coolant in CRBRP is far below the saturation temperature of 1900°F (based on the pressure at the fuel mid-plane), released molten fuel could be quickly fragmented and solidified without sodium boiling. Since inter-channel mixing is very good in the pin bundle, the coolant temperature in the channels surrounding an overpower pin is increased very little (<50°F) and the coolant temperatures to which molten fuel might be exposed range from 800°F to about 1300°F for rated conditions. Since molten fuel is more likely to exist (if anywhere) below the core midplane, the coolant temperature which it would contact would be near 1000°F.

However, based on TREAT power excursion tests, there is considerable evidence that if molten fuel is released from a pin, the heat exchange between the two liquids is rate-limited, the "molten fuel-coolant interaction" (MFCI) is inefficient and very little energy could be deposited in the sodium (Ref. 30). Thus the effect of molten fuel release on the coolant flow rate is small.

In the E and H series of TREAT tests, fresh and irradiated fuel pins in a flowing sodium Mark-II loop were subjected to power excursions. These tests were intended to simulate conditions far more severe than any that could be associated with a postulated overpower fuel pin and so in most of the tests conducted to date, extensive fuel melting occurred in the pin. In most tests, cladding failure occurred allowing the expulsion of some molten fuel into the sodium. In none of these tests did a sodium vapor explosion occur and maximum pressures were only in the hundreds of psi despite the large amounts of molten fuel in the pins. It is currently believed that the postulated release of a small amount of fuel from a postulated overpower pin in an LMFBR core assembly could not lead to an MFCI pressure pulse with a maximum value of more than a few hundred psi nor of a duration of more than a few milliseconds. The energy in such a pulse would be quite low (a few joules), and the flow in the affected coolant channels would not be reduced significantly.

B. Pin Uncovering Effects of a Postulated MFCI (Molten Fuel Coolant Interaction

Despite the extremely low probability of an MFCI, the discussion of potential consequences is continued assuming that such an event took place. During the postulated MFCI, a sodium vapor bubble would expand to encompass a short axial length of several pins. The thermal interaction observed in the TREAT H-2 test is similar to the behavior to be expected if a molten fuel release is postulated in a full size assembly (Ref. 31). For this test, the duration of the pressure pulse was only about 4 milliseconds and this resulted in a rapid collapse of the vapor bubble. It was estimated that an assumed-spherical bubble would have uncovered fewer than 37 pins. In Section 15.4.1.1.2 it was shown that the fuel pin surface would have to be perfectly insulated for at least 0.12 secs. before the cladding could reach 1600°F. Thus, any vapor bubble which collapsed in less than this time could not cause a neighbor fuel pin to fail. There is, then a very large margin between the MFCI voids created in typical TREAT power excursions and the void duration required to fail a neighbor pin due to thermal insulation of the cladding.

C. Thermal Loading of Adjacent Pins

For most of the ranges of input parameters consisting of molten fuel inventory and cladding rupture size, the molten fuel jet would be deflected by the coolant stream. However, some conditions (e.g., about 0.1 inch axial coverage) resulted in cladding temperatures above 1600°F in the adjacent pin hit by the fuel jet. It was concluded that, while the formation of a molten fuel jet was unlikely, if a jet were to be postulated, then failure of an adjacent pin could not be precluded. However, the spot covered by the postulated jet is small and heat conduction to the coolant should still be good immediately adjacent to the spot. Cladding ductility or internal pressure are too low to allow ballooning of the cladding of this neighbor pin. It has been found that irradiated 20% CW316SS cladding does not deform (or "ballon") when rupture occurs at high temperatures. At the BOL condition internal cladding pressure is too low to cause cladding ballooning, thus, no blockage could occur due to cladding deformation which might accompany rupture. At EOL conditions, the gas pressure is higher; however, the loss in clad ductility precludes ballooning of the clad.

Thus, the postulated failure of the neighbor pin is not a serious consequence since the neighbor is not likely to contain molten fuel. It could not lead to a blockage, and only fission gas would be released. Furthermore, the hole in the neighbor pin would probably be facing the overpower pin so any gas release from the neighbor pin would be directed toward the already-failed overpower pin. Thus, a gas jet which might be postulated would not be directed at a third pin, but only back at the first failed pin. Thus, it appears that even if there were postulated overpower pin failures, they would be self-limiting (see Section 15.4.1.3). 25

D. Thermal Loading of Duct

Impact of an assumed jet of liquid fuel on the assembly duct wall is equally improbable as on adjacent pins because of fuel jet fragmentation and solidification. Assuming, however, such impact to take place, a small wall penetration might be postulated by the mechanism of fuel heating at the wall after sodium expulsion of significant duration. (The latter could only be caused by a simultaneous large assembly blockage, which is most unlikely to occur for the reasons given in Section 15.4.1.3 on flow blockage analysis). The hypothetical situation of existence of molten fuel near the wall in the absence of liquid sodium was analyzed (Ref. 32 and 33) by postulating a 1 inch diameter layer of molten fuel with a 0.1 inch thickness and no solidification. This would cause boiling of the stagnant sodium in the inter-assembly gap in about 4 seconds. The duct would melt through in several more seconds. However, the high heat removal capacity of the sodium flow of the adjacent duct will prevent any damage to it, thus precluding any assembly propagation.

15.4.1.2.4 Mechanical Effects of Postulated Molten Fuel Release

Response of CRBRP Fuel Pin to Side Loadings

The mechanical strength of an empty fuel pin cladding tube at 1600°F was discussed in Subsection 15.4.1.1.4 and shown in Figure 15.4.1.1-2. These results may be used to infer the response of a fuel pin cladding tube to an MFCI (Molten Fuel-Coolant Interaction) pressure pulse typical of those measured in the TREAT E and H series tests as mentioned in Subsection 15.4.1.2.3. For example, in the H2 test about 40% of the fuel in the fuel pin was molten. Of this, apparently only about 3 grams effectively interacted with sodium to generate an MFCI pressure of about 150 psi and a half-height pulse width of about 2 milliseconds. If the time constant α("e-folding time") is substituted for the pulse width at half of maximum amplitude, then even the empty cladding tube could withstand this side loading. As previously noted in Subsection 15.4.1.1.4, the results from the cladding model are very conservative and it is expected that a postulated MFCI of a few hundred psi maximum amplitude and width of a few milliseconds would not fail any neighboring fuel pins.

It should be noted that, even if some neighboring fuel pins were to be failed by a rapid pressure pulse, the only consequence would be a slow release of fission gas which would have no serious effects, as also discussed in Subsection 15.4.1.1.4.

Response of Duct Wall to Side Loadings

The same type of analysis discussed in Subsection 15.4.1.1.4 (Mechanical Effects of Fission Gas Release on Pins and Duct Walls) can be applied to the core region of the duct where a release of molten fuel may be hypothesized to occur. In the above mentioned section it was found that at 1000°F the duct can withstand a steady uniform internal pressure of about 550 psi. Based on the fact that the predicted CRBR duct temperatures in the core region are less than 1000°F, the above mentioned conservative analyses indicate that the CRBR fuel assembly duct could be exposed to pressure pulses even as high as 550 psi with no loss in duct integrity.

For this postulated accident, it is possible that the duct of a neighboring assembly might be deformed due to dynamic coupling of the ducts by intra-fuel assembly sodium or if the ducts were very close to each other due to axial bowing. If the ducts are dynamically coupled by interassembly sodium, then the intra-assembly sodium should also offer dynamic resistance to deformation of the neighbor assembly duct. If the accident assembly (the one in which the pressure pulse is postulated to occur) duct were to rupture, then the pressure within the accident assembly might be relieved. The magnitude of the effect is not known. Considerable energy is required to cause the fully-irradiated duct to rupture. At 1000°F, it has been estimated that a fully irradiated duct could withstand up to 212 in-lb_f/in before failure. This is much larger than the energy available from an MFCI as determined from TREAT Experiments. Realistic estimates of the pressure pulse resulting from a release of molten fuel in the H2 test are on the order of 10 atm. (Ref. 30). To provide a conservative estimate of the duct capability, a hypothetical pulse of 80 atm. was analyzed in (Ref. 32). This resulted in a maximum energy deposition of 91 in-lb_f/in in the duct. Thus even in this overly conservative case, the margin to failure is greater than two.

Furthermore, it would require additional energy over and above this to deform the neighbor assembly duct. It is more probable that any additional energy would be absorbed by the interassembly sodium. Since no initiating pressure pulse from a local fault has been identified which could rupture the accident assembly duct, failure of adjacent assembly ducts is even less likely.

15.4.1.2.5 Consequences of Postulated Duct Deformation of Adjacent Assembly

Although rupture of a neighbor assembly duct would be unlikely even assuming that a large pressure pulse could occur in an assembly, the potential consequences of rupture of a neighbor assembly duct is worthy of discussion to show the depth of protection against a propagation of local failures. The consequences of diversion of flow from a ruptured assembly will be discussed in connection with the postulated duct-crack (Section 15.4.1.2.6). There it is conservatively estimated that even if as much as 33% of the fuel assembly flow could be diverted through a crack, no fuel pin cladding over-heating would occur. This conclusion, with some qualification is also indicative of the margin to pin failure in the postulated event of neighbor duct rupture. If no significant pin or duct distortion occurred, the 33% leakage could occur and there would still be about 150°F margin to the fuel pin integrity limit (1600°F). However, it is conceivable that the pins in the neighbor assembly might be mechanically loaded by: a) axial bending which might lead to a collapse-hinge; b) cross-sectional deformation; or c) concentrated forces at cladding-wrapper wire support points. This, however, will not produce plastic deformation because: a) axial bending would be accommodated over a long length and not enough local bending would occur to result in a collapse-hinge; b) the pins are very resistant to cross-sectional deformation and could also derive support from the fuel inside; and c) the forces at wrapper wire contact points necessary to cause failure are higher than those which would be available. Thus, the area of interest would be whether the neighbor duct could deform plastically in such a manner as to cause flow restriction in the pin bundle. This consideration is addressed below.

An analysis has been performed (Ref. 33) to assess the effect of postulated neighbor assembly duct deformation on coolant channel temperatures within the pin bundle. Since the postulated initiating event is an MFCI pressure pulse, which is not expected to occur, the magnitude of deformation of the neighbor assembly duct and hence, the distortion of the pin bundle, was taken as an input parameter. The analysis was done with the COBRA-III code (see Appendix A). The following model assumptions were used:

a) All six sides of the duct were assumed to be deformed rather than just that side nearest to the ruptured side of the accident assembly.

b) The spacing between pins was allowed to be reduced to much less than the wrapper wire diameter by postulating that severe local cladding deformation could occur allowing the wire to sink into the cladding at contact points.

c) The axial length over which deformation was assumed to occur was 12 inches, of which the full deformation occurred over a length of 8 inches (see Figure 15.4.1.2.5-1). Deformation occurred below and starting at the core midplane.

d) The width of duct which was displaced was assumed constant over the axial length and for all magnitudes of displacement.

e) The sides of the duct were deformed in such a way that each of the six outer corner fuel pins were not displaced (see Figure 15.4.1.2.5-1).

f) Each compressed channel was combined with one adjacent channel and calculated assuming a homogeneous temperature in the channel at each axial increment.

g) The same pressure drop across the bundle was used for each case analyzed.

The spacing between pins was reduced to 10% of nominal spacing to avoid code numerical instability. Also, an effective turbulent mixing factor was used (instead of the wire-wrap model) which gives more conservative results. The results showed that even for four rows of channels constricted, (i.e., three rows of pins compressed) which amounts to a duct deformation of $\sim$0.18 inch, (as shown in Figure 15.4.1.2.5-1) the reduction in total fuel assembly flow was only 10%. The coolant temperature of the peak channel increased by only about 110°F as shown in Figure 15.4.1.2.5-2. There is thus, a large margin to pin failure from purely thermal-hydraulic effects due to crushing of a fuel pin bundle. Therefore, no failure should occur even if a severe amount of pin bundle crushing is postulated. It can be seen in Figure 15.4.1.2.5-2 that when the first or the first and second outer row of channels are constricted, the maximum coolant temperature decreases. This would occur, because for normal geometry the outer three rows of channels are overcooled and the exit coolant temperatures are lower than those in the central six row of channels. Thus, constriction of the outer rows of channels would force more flow into the central area of the bundle reducing the temperatures of coolant there. When a large amount of constriction is assumed, the flow in the outer channels is so severely reduced that the maximum temperature would occur there.

It should be noted that even if some fuel pin cladding failures would occur, the only consequence would be release of fission produce gases which would be insignificant. Pin-to-pin failure propagation would not occur due to gas release as discussed in 15.4.1.1.3.

If duct crushing is postulated to be so severe as to result in some fuel melting in the pins within the distorted section of the pin bundle it may be seen from Figure 15.4.1.2.5-2 that a number of rows of channels much larger than four would have to be compressed before a rise in coolant temperature could occur which would be large enough to lead to fuel melting, i.e., coolant boiling would have to occur and the saturation temperature is $\sim$1900°F. Alternatively, either crushing would have to occur over a length much longer than one foot or the pins would have to be compressed even more tightly than $\sim$0.006 inch spacing for severe coolant temperature rises to occur within the pin bundle. This occurrence must be viewed then as sufficiently improbable as not to provide a realistic path to pin failure propagation in a neighbor assembly, especially since no realistic mechanism of crushing an adjacent assembly has been found. It is thus concluded that assembly-to-assembly failure propagation would not occur.

15.4.1.2.6 Thermal-Hydraulic Consequences of Postulated Fuel Assembly Duct Crack

An accident progression might be postulated to follow an incident stemming from the rupture of an assembly duct. There are, potentially, both thermal and mechanical consequences and these will be discussed below. The thermal effects arise from a loss of flow through the accident assembly due to leakage through the crack. Since the interassembly gap sodium is at a low pressure relative to that internal to the fuel assembly, flow could be diverted from the ruptured fuel assembly. The crack width would be limited by the lateral support of the fuel assemblies adjacent to the faces of the accident assembly crack. Since the fuel assembly duct would absorb energy before rupture would occur, little energy would be available to damage neighbor assemblies and no deformation to neighbor assemblies would occur. Hence, the postulated crack in the fuel assembly duct within which the MFCI was postulated to occur would probably be limited to a small width.

The axial extent of the postulated crack would be limited because of several factors:

a) the pressure decreases rapidly in the axial direction

b) the energy available from the MFCI is small (See Sections 15.4.1.2.3 and 15.4.1.2.4)

c) the initial opening of a crack would begin to relieve the pressure tending to limit propagation of the crack.

d) Toward the inlet, the duct temperature is lower and the strength is greater.

The greatest uncertainty on the duct cracking behavior is the ductility of the irradiated ducts as a function of axial position. However, the following analysis shows that the consequences of even very large duct cracks would be acceptable.

The analysis was performed to determine the consequences of a crack in a duct wall. The postulated crack is assumed to occur at the duct corner with a maximum opening of approximately 0.34 inches, the size being limited by the distance between subassemblies, 0.17 inches. (The flow path from the cracked duct is shown in Figure 15.4.1.2.6-1). The crack is assumed to extend the full length of the core region, 36 inches. Three types of flow resistances exist in the leak path; a contraction loss at the crack, a friction loss as the fluid flows between adjacent duct walls, and a diversion loss at the end of this channel wall. The contraction and diversion losses are obtained from References 65 and 66 and the friction loss coefficient is equal to

$$4fL/D$$

where:

f = friction coefficient obtained from Reference 65

L = length of the channel

D = hydraulic diameter

The preceding three types of resistances are suitable for use in the equation

$$\Delta P = \left(\frac{\rho}{144}\right) \frac{K V^2}{2g_c}$$

where:

ΔP = pressure drop, psi

$\rho/144$ = fluid density, lb/in^2

V = fluid velocity, ft/sec

g_c = gravitational constant, $\frac{lb_m}{lb_f}\frac{ft}{sec^2}$

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For the study of flow diversions, it is more convenient to work in terms of volumetric flow rates rather than velocities as velocities are flow area dependent. The velocity equals $w/\rho A$ where

w = mass flow rate, lb/sec

ρ = density, lb/ft^3

A = flow area, ft^2

Factoring 1/A from this leaves w/ρ, the volumetric flow with units of ft^3/sec. Substituting into the pressure drop equation

$$\Delta P = K\left(\frac{\rho}{144}\right)\left(\frac{1}{2g_c}\right)\frac{1}{A^2}\left(\frac{w}{\rho}\right)^2$$

$$= K'\left(w/\rho\right)^2$$

where $$K' = K\left(\frac{\rho}{144}\right)\left(\frac{1}{2g_c}\right)\left(\frac{1}{A^2}\right)$$

These resistances are then combined with the subassembly resistances (which are calculated in a like manner) and the flows in the system are calculated using the following equation:

$$\left(w/\rho\right)^2 = \frac{\Delta P}{K}$$

The results of these calculations for the worst case assembly (assembly 29) are shown in Table 15.4.1.2.6-1. It can be seen that the flow diversion from the cracked duct results in an increased flow into the assembly inlet with the result that the flow downstream of the crack is reduced to 21% of the normal flow.

This loss of flow is assumed to occur uniformly over the entire length of the crack, and the change in the hot channel outlet temperature is evaluated by comparing the coolant temperature rise in the damaged assembly to that in an undamaged assembly. The coolant temperature rise equals

$$\frac{\overline{Q}}{c_p}\int_a^b \frac{\phi(\ell)}{F(\ell)}\,d\ell$$

where

$\overline{Q}$ = heat generation rate, arbitrary units*

c_p = heat capacity, arbitrary units

$\phi(\ell)$ = normalized flux at location ℓ

$F(\ell)$ = flow at location (ℓ), arbitrary units

* arbitrary units are used because it is the ratio of two integrals that is of interest

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ℓ = distance from bottom of fuel pin, arbitrary units

The coolant temperature rise in the damaged assembly can then be found by

$$\Delta T_2 = \Delta T_1 \frac{\int_a^b \frac{\phi(\ell)}{F(\ell)} d\ell}{\frac{1}{F} \int_a^b \phi(\ell) d\ell}$$

where F is the constant, coolant flow rate in the normal assembly. This factor has been been applied to the hot channel temperature rise. As shown in Table 15.4.1.2.6-II, the hot channel outlet temperature for assembly 29, the worst case assembly, remains below saturation temperature, 1755°F. 20

15.4.1.3 Flow Blockage in a Core Assembly

Introduction

An analysis of local fuel assembly partial blockage for the CRBRP core involves technical analysis and qualitative evaluation. In the first approach which is used in the PSAR, various sizes, geometries, and locations of partial blockages, solid and porous, are analyzed to determine wake fluid and cladding temperatures without regard to a mechanistric route as to how the blockage occurred. It is recognized that such a partial blockage is extremely unlikely because of engineering design features, inspection and operation techniques. The temperatures determined are then used to ascertain whether and where boiling might occur, and if boiling might be detected.

In the second approach, emphasis would be placed on devising a mechanistic chain of events which leads to blockage (if such a chain exists). Thus, the second method involves a probability analysis of blockage and results in a determination of the size, location, and blockage material which are credible.

Postulated Types of Internal Assembly Blockages

Only a few off-normal occurrences can be postulated which could lead to a local blockage within the fuel pin bundle. These are:

1. the presence of foreign material in the primary coolant

2. a wire wrap failure

3. excessive pin bowing

4. excessive clad swelling

Operation of existing fast breeder reactors, as well as water reactors has not demonstrated any traceable history of fuel assembly damage caused by blockages from these sources. There is also an extensive blockage prevention program executed during design, manufacture, inspection and operation which is discussed in Section 15.4.1.3.1. Thus, although the type blockages cited are conceivable, they are very unlikely.

Major emphasis in this section is placed on analyzing a solid non-porous or porous blockage which completely restricts normal flow over a given axial distance in one or more sub-channels. In such a blockage, a wake would probably form downstream (and upstream) of the blockage. The reason considerable attention is given to the "solid" blockage is not because it has been identified by a mechanistic group of events but because it is considered to represent the ultimate worst case for the four types of blockages listed above.

Based on the discussion in Section 15.4.1.3, "Prevention and Detection", it is concluded any small size debris will be swept through the core. No mechanism has been identified which will transport debris preferentially to one or a group of assemblies so only random debris accumulation is possible. Thus, if debris is assumed to cause a blockage, all the assemblies will be affected in a random manner. Moreover, the dispersal of the debris over a large number of assemblies could affect a given assembly only to a small degree.

Blockage caused by wire failure or fuel swelling could not cause total solid blockage of a sub-channel and at worst could cause only local hot spots. As mentioned, the type blockage to which this section primarily addresses itself is the solid blockage in one or more sub-channels which is the improbable ultimate.

The possibility that a postulated broken wrapper wire could be moved by hydraulic forces to form a blockage was assessed. The result concludes that blockage due to a broken wire is unlikely due to the following factors: (1) the normal wire tension is only 8 lbs., whereas a tensile load of about 64 lbs. is required to cause wire breakage; (2) the hydraulic forces are small, (3) the wire is held by the cladding, and (4) other intact wrapper wires would inhibit movement. These conclusions have been verified by irradiation tests of wire-wrapped pin bundles. In these tests. no breakage but considerable stretching occurred of the Type 316 stainless steel (as opposed to stronger cold-worked 316 SS) wire. There were no fretting, galling, or flow blockage problems.

Despite the improbability of a wrapper wire failure causing a blockage, a case has been analyzed assuming that a wire formed a 6 inch long tight helix at the top of the heated zone. Neglecting the wire-cladding contact resistance due to the high conductivity of sodium which is present, the maximum cladding temperature attained was about 1500°F for reference operating conditions. This temperature is well below that required to result in prompt cladding failure. However, some decrease in cladding lifetime might occur due to increased creep rate at the higher temperature.

It could be postulated that mechanisms could come into play which could cause fuel pin bowing and result in local blockage. There are no known forces which could cause significant local fuel pin bowing. Adjacent pin contact is prevented by wire wraps which have a pitch of 12 inches. Even if pin-to-pin contact occurred, the line contact area would be very small and no significant flow perturbation would occur. The result would be better described as a local reduction in cladding surface heat transfer and would only lead to a local cladding temperature increase well below the failure limit.

In the inlet blockage discussion, it was postulated that debris of diameter greater than 0.24 inch could be trapped by the fuel pin lower support plate to form a planar inlet blockage. Debris of diameter less than 0.056 inch could be swept through the pin bundle but particles near that size could be caught in random locations where slightly smaller distances between the pins exist. Such particles of debris could only be randomly distributed within the heated zone and could cause only minute localized flow perturbations. If a single particle were caught in a channel, subsequent particles swept into the same channel would escape to other channels if their diameter were less than that of the wrapper wire. Before initial operation, the simulated core assembly filters will remove particles larger than 100 microns (0.004 inch). However, it may be postulated that several particles were all trapped behind each other in a single channel. The particles could touch the pin cladding in point or line contact over only very small areas. The analysis was performed with the angle of coverage used as a parameter, assuming that the debris was a perfect insulator. By including azimuthal heat conduction, it was found that an angle of coverage of 25° could be tolerated for the entire pin length without cladding failure. Angles of coverage much larger than 25°, which are very improbable, may result in cladding temperatures over 1600°F. An angle of coverage of about 60° would be required before certain damage to a pin cladding would result (2500°F).

The statement that particulates postulated to become trapped in the heated zone of the pin bundle would be randomly distributed is based on the wire wrapped pin bundle design. The helical spacer wire geometry provides no preferential axial or radial location where particulate retention could occur. While it is recognized that there is a variation in the coolant velocity radially across the pin bundle, this effect is not expected to significantly influence the likelihood of particulate trapping. Even if such an effect were determined, the basic conclusion regarding the improbability of local blockage formation would not be impacted since these radial effects would only serve to differentiate debris retention probabilities between two large regions (i.e. the central portion of the bundle and the outer row of pins adjacent to the duct). No specific local region for particulate collection could result from the varying radial velocity profile. Particles which pass through the filtering action of the fuel pin attachment assembly which are larger than 0.056 inches or possess some characteristic, such as shape, which might enhance entrapment in the pin bundle would likely become trapped in the first few inches of the bundle. Such entrapment would result in a very minor local flow disturbance which would not impact fuel pin performance since the occurrence is far below the heated zone. The drag characteristics of any particle smaller than 0.056 inches are such that they will be easily swept along by the coolant. The settling velocity is approximately 2.5fps compared to an average coolant velocity of about 13.8fps in the lowest flow assemblies in the core. Hence any particles small enough to enter the pin bundle will be readily carried through the bundle and no settling of particles into the bundle from regions above it could occur.

Analyses of irradiated swelling and thermal expansion show that these effects would change the channel dimensions only very slightly. At the worst location, between one half and three quarters up the core, the maximum flow channel closure would be only about 0.003 inches at the wire wrap/fuel rod support planes. This small variation is considered to be of negligible consequence. Therefore there is no mechanism by which a significant planar blockage could form at a given elevation. Furthermore there are no reported occurences of particle entrapment within fuel and test assemblies that have been removed from U.S. and foreign liquid metal fast breeder reactors. 25

There is no mechanism known by which a coherent in-core blockage (i.e., a blockage of several connected channels around adjacent pins at the same axial core position) could be formed. Nevertheless, such a blockage was postulated and discussed in Section 15.4.1.3.3.

15.4.1.3.1 Prevention and Detection

Flow blockages of CRBRP fuel, radial blanket, and control assemblies are extremely unlikely due to design features, cleanliness requirements during construction, precautionary operations carried out during initial sodium fill and testing, and sodium purity requirements during reactor operation. Postulated blockage mechanisms of the core assemblies are: (a) large and (b) small non-degradable debris left behind during construction, (c) degradable material left behind during construction, (d) corrosion products, (e) sodium-lubricant reaction products, and (f) failed fuel debris.

During pre-startup acceptance testing, filters in simulated core assemblies remove particles larger than 100 microns from the sodium coolant, eliminating small particles from the system prior to initial startup. The lower reactor internals are designed to prevent a flow blockage to one or more assemblies by providing multiple flow paths to each core assembly. This precaution prevents a "Fermi type" incident i.e., a flat plate completely blocking the flow to one or more core assemblies. During reactor operation the sodium is checked regularly to confirm that it meets the purity requirements of RDT Standard A1-5T, March 1976. Additionall, during normal operation the primary coolant is continuously being circulated through cold traps to remove
27 any foreign particles in the sodium. The following flow blockage mechanisms were considered in the design of the lower internals and reactor assemblies:

A. Large Debris

Large debris consists of non-degradable materials, and would cause a rapid flow blockage if it could occur. The design of lower internals precludes a large object (such as a plate) from blocking a significant amount of flow from entering any core assembly by providing multiple flow paths to each assembly (See Section 15.4.1.5). The largest particles which may reach the fuel, radial blanket or control assemblies are 0.25 inch in diameter because a strainer plate with 0.25 inch diameter holes is located in the lower core modules. The large debris flow blockage of an inlet module is discussed in Section 15.4.1.4. No detection mechanism is available to detect a flow blockage due to large debris. The feature to preclude major flow blockage at the module liner inlet is the auxiliary flow port liner design shown in Section 4.2.

B. Small Debris

Small debris in the reactor can result from material left behind during construction such as nuts and bolts, tools, or non-degradable gloves. Safeguards against such small debris causing a flow blockage include multiple flow paths to each core assembly. The largest particles to reach an assembly are 0.25 inch in diameter. Particles smaller than 0.056 inch will pass through a fuel assembly rod bundle, 0.089 inch will pass through a radial blanket assembly rod bundle, and 0.080 inch will pass through a control assembly rod bundle. The 0.056 inch particles will be swept from the rod bundle in the core fuel assemblies, and 0.089 inch particles will be swept from the radial blanket assemblies in orificing zones 6 and 7, 0.086 inch particles will be swept out in zone 8, and 0.021 inch particles will be swept out in zone 9. Particles larger than 0.056 inch, 0.080 inch, and 0.089 inch, but smaller than 0.25 inch will become trapped either at the entrance to the rod bundle or in the first few inches of the rod bundle.

The mechanism preventing core assembly blockages due to small debris are:

1. Cleanliness requirements during construction of the primary system
2. Cleanliness requirements of the fill sodium
3. Pre-operational cleanup of the primary system with the core special filters assemblies
4. Use of the cold trap during normal operation
5. Normal operating sodium cleanliness requirements.

Detection of a core assembly blockage is difficult. Because the
maximum particle size to reach an assembly is 0.25 inch in diameter, and the
41 potential hangup spot for these particles is a hexagonal shape 4.335 inches
across flats many particles are required to block a small percentage of the
flow area. Figure 15.4.1.3-1 shows that a 50% area flow blockage reduces the
assembly flow by approximately 5%. The 5% reduction in flow is the minimum
detectable if thermocouples were located at the assembly outlets. The possibility of a 50% or larger flow blockage due to small debris occurring is
extremely low because there will not be a substantial amount of small debris
in the primary system. If a detectable flow blockage were to occur, it would
probably occur in a large number of assemblies. The current design calls for
32 coolant outlet temperatures to be monitored for each fuel
assembly. These thermocouples may enhance detection capability of excess
coolant outlet temperatures that could rise from flow blockages (See Section 4.4.5).

C. Corrosion Products

During operation of the reactor mass transfer of corrosion products will occur from hot areas to cold. The mass transfer mechanism will deposit approximately 120 lb/yr of products in the hot leg piping and IHX. Very little will return to the core inlet during normal operation. The deposits consist of a brittle, flaky and fragile material with a maximum deposit thickness of approximately 0.010 inch. The deposits may be released from the IHX during a thermal shock on the IHX or an incident such as a check valve accidently closing causing a "sodium hammer" effect. The corrosion products will then consist of small particles of 0.01 inch or less. These particles will pass through the core assemblies with negligible effects. If debris deposition occurs it will not occur preferentially. Conservative estimates of debris deposition at the pin attachment location have shown that over the 3 year lifetime of the fuel assemblies, a film thickness of 0.6 mils would result. This would cause a 1.7% reduction in flow area which would have no adverse effects on fuel pin performance. A study was performed to determine the physical characteristics of corrosion product deposits (Reference 79). The results of this study are consistent with the above described properties and behavior expected of such deposits.

Additionally, the reactor operator will know when a check valve closes or a thermal shock occurs on an IHX, warning him of the potential release of corrosion products.

D. Products of Degradable Materials

Although highly unlikely, degradable materials such as plastic, rubber, or cotton gloves, may be left behind during construction of the reactor. These materials will thermally decompose on contact with hot sodium and the residue removed by the screens in the simulated core assemblies during pre-startup tests.

1. Quality Control during construction will minimize if not eliminate the amount of material left behind.

2. The lower internals design provides multiple flow paths to each inlet module so that each assembly will receive coolant, even if one flow path is blocked.

3. Most of these materials degrade into ashes on contact with hot sodium.

4. The simulated core assemblies will remove the ashes during pre-startup testing and the cold trap will continue to remove the degraded products from the primary system during operation.

E. Lubricant Reaction Products

Lubricants inadvertently released to the sodium coolant are another potential source of flow blockage. Several potential sources of lubricant in the primary system have been identified, including pump bearing and seal, and the fuel handling machine, among others. The check valves have no lubricant and all auxiliary systems are designed with no hydrocarbons in their cooling systems. The fuel handling machine may emit minute quantities of grease which will have no adverse effects on the sodium purity or flow blockage potential. As discussed in Section 5.3.2.3.1, the pump bearing seal is designed to prevent oil from leaking into the primary coolant. Lubricant leaking past the seal is collected in the leakage reservoir and does not enter the primary sodium. Since the leakage reservoir is sized to hold 150% of the total oil inventory, even if it is postulated that the entire lubricant supply leaks through the seal, oil would still not enter the primary coolant. If such a large leak were postulated to occur, the oil level indicator in the leakage reservoir would immediately notify the operator of significant losses of lubricant. In addition a pump would be activated which would drain the leakage reservoir transporting the excessive oil to a holding ~~tank~~ on the operating floor.

Experiments where DTE-24 turbine oil (a candidate for pump seal lubricant) was mixed with sodium (850 and 1050^{o}F) have shown that oil decomposes rapidly forming volatile hydrocarbons, hydrogen, and carbonaceous particulates. The particulates are friable, low density, and very small (less than 1 mil). Such particles would be swept along with the primary coolant. Because of their small size, these particles would not become trapped within the pin bundle. Furthermore, because of the extremely low concentration (even for large oil leakages), even if the particles are postulated to collect in the pin bundle, they would be spread randomly across the core and blanket and not result in a significant flow blockage. An indication of a very large lubricant leakage could be provided by the hydrogen monitors which continuously monitor the reactor cover gas. In conclusion, even though oil leakages into the primary sodium are improbable and the situation could be detected, the consequences of identified potential sources of lubricant would not impact fuel pin performance.

F. Slow Blockages Caused by Failed Fuel Debris

During normal reactor operation a few fuel and radial blanket rods can be expected to fail. Most of these will be gas leakers, but a small percentage may have fuel directly exposed to the sodium, or may allow sodium to enter the rod, causing sodium logging. The effects of the sodium logging may cause small, localized blockages in the assemblies. The failures will not propagate to other rods and they will not cause any changes in total assembly flow or outlet temperature, thus they are not detectable by instrumentation. Prevention of these failed fuel blockages is accomplished by a conservative design of the assembly fuel rods as indicated in Section 4.2.1. Initial detection of a failed fuel rod is done by the Cover Gas Monitoring System. The Gas Tag System determines which assembly has failed and the Delayed Neutron Detectors determines if fuel is directly exposed to the sodium.

15.4.1.3.2 Consequences of Blockage of a Core Assembly

The result of partial blockage of the fuel rod bundle inlet is a reduction in assembly flow rate as illustrated in Figure 15.4.1.3-2. While existence of extensive inlet blockage is quite unlikely, as discussed earlier, its effect on coolant flow is also relatively small. The inlet flow area has to be more than 50% blocked before a significant reduction in flow takes place. As shown in Figure 15.4.1.3-1, a blockage in excess of 80% is required to raise the coolant outlet temperature by more than 200°F. Blockages of this magnitude are extremely improbable and no mechanism or source can be postulated by which such a blockage could occur.

An analysis was performed to determine the maximum cladding temperature due to complete blockage of a fuel assembly outlet nozzle during refueling. It was found that, even using a conservative analysis, the maximum cladding temperature for this accident (if it occurs after 6 hours from shutdown) would be less than that for operating conditions.

15.4.1.3.3 In-Core Passive Local Blockage (Non-Heat Generating)

The increase in the wake temperature behind a blockage depends on the axial location, radial location, and the blockage size. Flow rate and heat flux are also important parameters. Two basic types of blockages have been identified (Ref. 34); a three-dimensional center or off-center blockage, and a two-dimensional edge blockage. For otherwise equal conditions, a two-dimensional blockage is worse because the wake is larger and the residence time, , is larger. This factor has been verified experimentally (Ref. 35 and 36). The reason a two-dimensional blockage has a larger wake is because the drag coefficient is larger.

Calculations were made to determine the average and maximum fluid temperature in the wake region behind a central six channel blockage. Cladding temperatures were also calculated. The results are shown in Table 15.4.1.3.3-1. Methods used to calculate the results are similar to those discussed in Section 15.4.3.3. Heat transfer coefficients were obtained from the sodium flow blockage data (Ref. 36, and 37).

It can be seen from the table that at either the midplane or the core exit plane, the postulated six channel center blockage would cause temperatures which are considerably lower than the temperature corresponding to prompt cladding failure.

If the six channel blockage occurred on the edge of the fuel assembly, the hydraulics would still be quasi-three dimensional. Although the dimensionless residence time t_R would be larger, the colder edge fluid flowing by the small blockage should partially compensate for the higher t_R. Only if the six-channel blockage occurred in a corner, would there truly be a two-dimensional effect, possibly producing higher wake and cladding temperatures than shown in Table 15.4.1.3.3-1. Because of uncertainties in existing edge blockage data and prediction models, two-dimensional definitive blockage, calculations for edge channels were not made. An appropriate evaluation indicates however that even a true two-dimensional corner blockage of six subchannels does not cause excessive mid-wall cladding temperatures.*

15.4.1.3.4 <u>In-Core Active Blockage (Heat-Generating)</u>

An in-core heat-generating blockage can only arise from gross fuel failure. Delayed neutrons from decaying fission products which are released when the failure occurs will be detectable and will alert the operator within one minute after the event. However, assuming that failure is not detectable, a heat generating blockage would result, which, contrary to the basic assumptions made earlier concerning non-heat-generating blockages, would be porous. A general comparison of porous and nonporous blockages will be discussed first. The discussion is applicable to both core, blanket, and control assemblies.

<u>Porous Blockages</u>

As cited previously, a mechanism which can cause a non-heat-generating complete blockage of one or more flow channels within a core, radial blanket or control rod assembly has not been identified. Nevertheless, if a complete blockage of one or more flow channels does occur by a gradual local buildup of debris, the blockage will be porous. In fact, even a gradually formed heat-generating blockage should be porous. The question to be answered is, given a porous and a nonporous blockage of otherwise similar characteristics, which one causes higher coolant and/or cladding temperatures in the region of the blockage? In the light of existing data this question is answered in the following paragraphs.

Leakage flow through a blockage increases the static base pressure immediately downstream of the blockage (Ref. 35, 38, 39, 40). The result is a decrease in the drag coefficient across the blockage and a decrease in the volume of the near-wake region. As the leakage area ratio β_L increases

*Data (Ref. 36) showed no safety problems for 14 edge channels blocked in a 19-rod fuel assembly.

from zero to unity, the drag coefficient for a three-dimensional blockage (a blockage removed from the assembly can) decreases from 1.0 to zero. Increased leakage flow moves the wake downstream and decreases its volume and the circulation intensity within the wake. Kirsh (Ref. 35) reported experimental data for the rod bundles showing that the wake region essentially disappears when the ratio of the flow rate through the blockage to the steady state flow rate is greater than 0.15. For a two-dimensional blockage, Castro's data indicates that the wake region is very weak when β_L is approximately 0.3. Bearman's (Ref. 39) data are similar. The most important consideration, however, is not whether the wake disappears, but rather what are the temperature effects with leakage flow through blockage. Kirsch (Ref. 35) reported that for a leakage area β_L of the order of 0.05 and a central blockage area-ratio β equal to 0.411, leakage reduced the wake temperature 100°F for the SNR thermal conditions. The basic data reported by Kirsch were obtained in a full size water loop. Recently ORNL obtained basic wake flow entrainment data in a triple size water loop and used these data to compute the coolant temperature increase caused by a 14 sub-channel edge blockage in a 19 rod bundle. A comparison was then made between the calculated temperature increase and the temperature increase obtained from measurements in a full scale 19 rod assembly having sodium flowing at a rated power level of 5 kW/ft and 26 ft/sec. The leakage area ratio was of the order of 0.10. Their data indicates that leakage substantially reduced the fluid temperature in the blockage region (Ref. 36). Interpretation of the ORNL data indicates an approximate 100°F blockage fluid temperature decrease as a result of leakage. The conclusion is that any leakage will reduce the wake fluid and cladding temperatures, thus, the assumption of a non-porous blockage is conservative.

Leakage flow is very effective in reducing the wake temperature and in particular the cladding temperature because the leakage flow acts most effectively where it is needed directly behind the blockage. As the leakage leaves the blockage, the small jets, as they expand, create considerable turbulence.

Both Kirsch and ORNL used a direct leakage path through the blockage, i.e., drilled holes, off-set blockage plate. The hydraulic resistance of such a passage is considerably lower than for the tortous path through a blockage consisting of small particles. The result is that the coolant velocity through a direct leakage path, for the same porosity is than for the type porous blockage envisioned in a reactor (Ref. 41, 42). Thus, porous blockages, even with high porosity, would be expected to be more similar to non-porous blockage. The previous conclusion in regard to non-porous blockages being conservative is nevertheless still valid.

Effects of Heat-Generating Blockage

There are no experimental data describing the effects of heat-generating blockage on fuel rod performance. Such information has not been obtained from in-pile or out-of-pile studies. If the heat generating blockage caused a wake to form downstream of the blockage, much of the hydraulic wake discussion of the effects of non-heat-generating blockage would be expected to be valid. The additional factor and complication is local heat-transfer and temperature distribution within the blockage itself.

A thermal-hydraulic model was devised for a local heat generating blockage; the model was one-dimensional (Ref. 43). The purpose was to determine the heat transfer performance of sodium flowing through a porous medium having internal heat generation. The porous heat thickness which caused coolant boiling was determined as a function of the average effective particle diameter for various bed positions. Over a range of fuel debris sizes of 100 to 1000 μm and bed porosity of 0.25 to 0.50 it was concluded that for an average particle size of 500 to 600 μm and a bed porosity equal to 0.35 to 0.45, a quantity of 2 to 5 grams of fuel per flow channel would be required to produce boiling and that steady state temperatures within the porous medium would be obtained shortly after blockage was initiated. The axial location was midplane of the highest power fuel assembly. The actual blockage height was, therefore, approximately 0.4 to 0.8 inches, a height larger than a fuel pellet. This is a substantial amount of fuel. For these thick blockages, local fuel rod failure might occur releasing fission gas slowly, but no rods outside the blockage zone would be expected to fail. Experimental data is required to substantiate this analysis (See Section 1.5.2.1). It is probable that a three-dimensional analytical model supported by experimental data, would indicate even larger amounts of heat generating debris would be required to cause boiling.

The postulated formation of a heat generating blockage implies the presence of fuel material in the flow channel. Consequently, this implies that a fuel pin has already failed. Any material from that failed fuel pin having blockage potential would be expected to remain adjacent to the failure location. Otherwise it would be swept out of the assembly due to the dispersive nature of flow in a wire wrap assembly (References 73 and 74. The failure location referred to above is presumed to mean the location within a fuel pin adjacent to one that has already failed.

Although it is not possible to predict the exact location of pin failure resulting from severe local overheating, it is expected that any breach would occur somewhere close to the location of the maximum cladding temperature since the clad strength decreases with increasing temperature. For the case of a postulated heat generating blockage, the maximum cladding temperature will occur in sections of cladding adjacent to the exit (top) of the porous blockage. Hence this is a likely location of any pin failure that occurs. However, since pin failure is dependent on several parameters, including local fuel-clad interaction as well as clad temperature, the location of the clad breach cannot be generalized. 25

15.4.1.3.5 Postulated Local Boiling Effects; Core Assembly

There is no basic difference in the method used to calculate the blockage size which might cause local boiling and the method used to calculate the effects of smaller size blockages. Table 15.4.1.3.5-1 shows the calculated sizes of central blockages, fraction area blocked and the boiling temperatures assumed at the midplane and exit plane of the core.

Table 15.4.1.3.5-1 illustrates the significant difference between blockage at the exit and midplane. Recent measurements by ORNL show that the temperature increase caused by a six channel central blockage can be detected at least a foot downstream of the blockage. However, it is unlikely that it could be detected above the fuel assembly.

True edge blockages might be more severe. However, there are not sufficient data to calculate the boiling blockage area with any degree of confidence.

15.4.1.3.6 Long Term Effects of a Hot Spot

In the preceeding sections the effects of inlet flow blockages and non-heat generating and heat generating in-core blockages have been discussed. One effect of these blockages is to increase the local cladding temperature in the vicinity of the blockage. If these blockages go undetected over a long period of time, the burnup capability of the core fuel rods would be reduced.

Figure 15.4.1.3.6-1 shows the effect of a temperature increase over the life of the fuel rod on the lifetime of the statistical hot rod at core outlet. This curve is based upon the time to reach 0.2% ductility limited strain during steady state operation. It is conservative since it is assumed that the design temperature just allows the design life to be achieved. As shown in Figure 15.4.1.3.6-1, a temperature increase of 15°F would have to act over 90% of the design life for a premature failure to occur.

No adverse consequences would be expected from premature failure since it would be of no worse consequence than the stochastic fuel pin failure discussed in Section 15.4.1.1.

15.4.1.3.7 <u>Local Boiling Stability and Dry-Out</u>

Theoretical analysis (Ref. 44) of local boiling in the wake behind a blockage large enough to be detected indicates that boiling will not lead to rapid dry-out, fuel pin failure and ejection of molten fuel, flow instability, bulk-coolant boiling and gross melting of cladding. It should be remembered that even when the hot spot in the wake fluid reaches the local boiling temperature the average wake fluid temperature is considerably lower. (A maximum-to-average wake fluid temperature ratio equal to 1.3 is used). An important basis of this one-dimensional analysis is that because of the short lifetime of bubbles, local dry-out is unlikely to occur during the lifetime of a bubble because a thin liquid layer remains on the fuel rod surface. Moreover, where little or no superheat is required for local boiling, the wake subcooling prevents the steady state vapor velocity from exceeding the liquid film destruction velocity (flooding).

Schleisiek (Ref. 45, 46) conducted tests in which 12 subchannels were blocked. This geometry represented a two-dimensional cut out of a 61-rod bundle with a central blockage of 37 rods. Local boiling conditions were achieved in the wake behind the blockage. In the absence of superheat, boiling started with the formation of small, hardly detectable bubbles. The temperature in the boiling area did not exceed significantly the saturation temperature (1450-1700°F). Nearly all the bubbles collapsed completely and the succeeding bubble was formed without delay. Dry out of the test section wall was observed only when very high superheat caused instability of the total sodium flow. High sodium boiling superheats are not expected under reactor conditions.

<u>Pressure Drop Caused by Blockage</u>

In the ORNL blockage tests using a 19-rod triple scale assembly, the pressure drop was measured for the rod bundle with no internal blockage plates and for the bundle with two different central blockage plates and three edge blockage plates (Ref. 36). Both edge and central blockage plates produced similar pressure drop increase characteristics. The edge blockage plate which obstructed one-third of the flow area caused a 60% increase in pressure drop; the plate which blocked 60% of the area caused a 230% increase in the pressure drop. The pressure drop relation proposed by Novendstern (Ref. 47) gave a good correlation of the data. When applied to CRBRP core which has a length to diameter ratio (L/D) in the order of L/D = 700, a 40% blockage is predicted to cause a 10% increase in pressure drop or a ∿3% reduction in flow. This is in agreement with earlier theoretical predictions. However, as stated earlier for large blockages which might cause local boiling, the temperature difference caused by the blockage might persist far downstream of the blockage. A test is planned to determine whether a temperature abnormality caused by a blockage can be detected by exit thermocouples in a full size CRBRP core assembly (See Section 1.5.2.1). A temperature difference of 5 to 8 degrees can be detected by thermocouples, whereas a local blockage causing boiling produces a local temperature difference of 700 to 900°F.

15.4.1.4 Postulated Module Inlet Blockage

Because of the redundancy of the flow path (discussed in Sec-
41 tion 15.4.1.4.1), the occurrence of a module inlet blockage is very remote.
The presence of any material in the primary circuit which might instantaneously
block the redundant module primary inlets is considered incredible in view
of the procedures and safeguards applied in the design against such an
occurrence. Particles larger than 0.25 inches in diameter entering through
the inlet holes in the module liner will be stopped at the strainer holes in
the stem of the inlet module. Thus, there could not be particles large
enough to block any or part of the six 3/4 x 2.25 inch radial slotted radial
entrances to the assemblies. A blockage of one or two of these holes would not
41 affect the flow to the assembly noticeably, neither would blockage of a few or
even one-half of the strainer holes, since the entrance pressure drop is but
a small fraction of the assembly pressure gradient. Thus, no noticeable
flow redistribution will occur. The same can be shown for blockage of one
or even two of the three module inlet holes.

However, in the unlikely event of complete blockage of the primary
inlet holes by pliant type debris, such as coveralls, the auxiliary flow paths
41 (See Figure 4.2-39) in each module liner and module would provide a secondary
flow path to the module. Thus the redundancy built into the lower internals
design assures that regardless of the postulated blockage mode, sufficient
supplementary flow paths are provided to assure coolant flow to each module.

15.4.1.4.1 Prevention and Detection of Module Inlet Blockage

Complete blockage of an inlet module is considered extremely improbable because of the multiple flow paths and geometric configuration of the inlet modules. The module liner has 6 primary inlet holes with an additional set of inlet ports located just below the CSS lower surface

which supplies flow to the module. An integral hexagonal debris barrier
(See Figure 4.2-39) partitions the safety auxiliary ports from the primary
ports. To restrict flow significantly to the fuel assemblies within the
module some object must close off both the 6 liner inlet holes and the
auxiliary ports spaced around the circumference of the module. An object
41 postulated to block the primary inlet geometry might be a cannister
of diameter large enough to fit around the module liner and sufficient length
41 to cover two tiers of three 3.5 in. dia. holes axially adjacent to each other.
However, there are no known sources for objects of this nature in the inlet
plenum. The hexagonal debris barriers of each module liner mesh together
forming a continuous flat plate above the primary ports with sufficient
apperture to allow coolant entrance and to form a secondary coolant plenum
for auxiliary cooling to the modules. The barrier formed precludes the
entrance of debris large enough to completely block off the auxiliary ports.
In the event the liner and module primary ports become blocked the liner and
module draw high pressure cooling sodium from the secondary plenum via the
auxiliary ports.

One could postulate that an object could enter from the side of this
region and flow upward to the location of the inlet slots. However, because
41 of the geometric configuration of the module liners, no canister shaped object
could enter the region and encapsulate the module liner to the degree
necessary to substantially reduce flow in the module.

Although there is no source for such an object, the geometric configuration is such that an object could be postulated to be transported to the location of the inlet holes and block one of the 6 entrances. However, this would only reduce flow by less than 5%, creating no significant increase in the fuel pin operating temperatures.

The transport of objects in the flow system that could cause blockage is influenced by the change of direction of the flow and also the velocity change which to a degree is a measure of the fluids ability to transport larger solid objects. The reactor inlet flow is through 24 inch diameter inlet pipes which direct the flow downward into the inlet plenum. The ratio of plenum area to inlet pipe area is large so the upward flow velocity toward the module inlets is reduced and the ability to carry sizeable objects is reduced in proportion.

On the assumption that an entire module does become blocked, such a blockage could be detected by observing the outlet temperature rise as indicated by the thermocouples located in the outlet plenum. A discussion of the functional requirements for this instrumentation is contained in Section 4.4.5. It should be emphasized that safety considerations do not require thermocouples to monitor the outlet temperature because of the extreme improbability of a module inlet blockage of sufficient size to cause change in the fuel assemblies.

15.4.1.5 Small Gas Bubble Passage Through Fuel, Radial Blanket and Control Rod Assemblies

15.4.1.5.1 Prevention and Detection of Gas Entrainment

The heat transport system incorporates design features to preclude gas bubbles from entering the core. These include:

a. Vents provided to eliminate possible gas pockets that may form during sodium fill.

b. A low cover gas pressure which reduces gas entrainment.

c. A continuous bleed from the top of the IHX to prevent accumulation of gas during operation is provided.

d. The primary pump is designed to eliminate vortexing and gas entrainment.

e. A high fluid velocity in the piping between the pump discharge and the vessel inlet minimizes the possibility of gas entrainment.

f. A vortex suppressor at the optimum depth to prevent gas entrainment is located in the outlet plenum.

g. One or more holes with special pressure reducers will be provided in the core support cone to vent gas from underneath the cone to the outlet plenum.

Basically, the philosophy is to insure through development testing (See Section 1.5.2.1) that entrainment will not occur; thus explicit detection equipment will not be necessary. From the discussion in Section 15.2.3.2 one can see, however, that gas bubbles of consequential magnitude in the core would produce reactivity insertions (and a resultant power burst) that could be detected by the reactor flux monitors (e.g., a 4 inch high - 8 row bubble would produce over a factor of four increase in reactor power but increase the maximum cladding temperature by only 68°F).

15.4.1.5.2 Dispersion of Large Gas Masses in Lower Plenum

Gas injection tests in an inlet plenum mock-up of the FFTF have demonstrated that even if gas is introduced coherently into the plenum through the inlet nozzles, the gas bubble would be broken up in the plenum. It was found that the bubbles formed after the initial break-up are dispersed across the core before entry. A coherent large bubble that could cause a large reactivity is thus precluded. This same type testing will be performed for the CRBRP inlet plenum which is geometrically similar to that of FFTF. It should be re-emphasized that design features have been included such that there is no source of large bubbles and this testing is being performed to provide added surety of safety.

15.4.1.5.3 <u>Consequences of Small Gas Bubbles</u>

The consequences of large coherent gas bubbles passing through the CRBRP core were discussed in detail in Section 15.2.3.2. For this type bubble, there was a temperature increase to the cladding due to the positive reactivity insertion causing a power burst and due to the insulation effect of the bubble passing along the cladding surface in the heater region. For small bubbles, however, the temperature increase on fuel, radial blanket and control assembly pin cladding would be due only to the insulation effect since the reactor power changes would be insignificant (as discussed in the next section).

15.4.1.5.4 <u>Reactivity Effects of Small Gas Bubbles</u>

In Section 15.2.3.2 it was shown that for a bubble that is 1 inch high - 8 rows in radial extent or one that is 4 inches high - 4 rows in radial extent the reactor power increase due to the positive reactivity insertion would be no more than about 30%; negligible cladding temperature increases were shown to result. The effect was a very quick power burst of a duration less than 0.1 second. It can thus be concluded that the reactivity effects of small gas bubbles should be of negligible consequence.

15.4.1.5.5 <u>Thermal Effects of Gas Bubbles</u>

A conservative analysis was performed to determine the worst case temperature effect of small gas bubbles blanketing fuel, radial blanket and control rod pins. For the duration that a bubble of given length would cover a section of cladding, all the heat lost from the pin was assumed to go into heating of the cladding. The duration of blanketing is the length of the bubble divided by the bubble velocity. The bubbles were assumed to move with the sodium coolant velocity for the hot pin in the particular type assembly being analyzed. This assumption of zero slip between the sodium and gas is conservative since, if the bubble moves faster, the blanketing time would be less. Two locations were considered - the center of the active core and the top of the active core. The hot pin results for the three types of assemblies are shown on Figures 15.4.1.5-1, -2, and -3. As can be seen, the worst location with respect to temperature rise during blanketing is, as would be expected, at the center of core. However, cladding temperatures at this location are about 200°F cooler at steady state than those at the top of the active core. Thus the most limiting position with regard to the maximum temperatures that would be attained (steady state plus transient increase) would be at the top of the active core. To cause a 25°F increase at this position would take a 5.0 inch bubble for a fuel assembly, a 7.5 inch bubble for a radial blanket assembly and over 12.0 inch bubble for a control assembly. These results are conservatively based on the highest power generation condition of each respective type pin considering all core conditions (i.e., BOEC used for hot fuel pin, EOEL used for the hot radial blanket pin and fully inserted primary control assembly considered for control rod hot pin).

The above results are for bubbles that would enter the assemblies via the inlet plenum of the reactor. Gas blanketing due to the internal assembly source of pin cladding failures is discussed in detail in Sections 15.4.1.1, 15.4.2.1 and 15.4.3.1.

15.4.1.5.6 Mechanical Effects of Small Gas Bubbles

As discussed in the previous section, a postulated 5.0 inch long bubble would be required to insulate a fuel pin at the top of the active core sufficient to produce a 25°F cladding temperature rise. A transient temperature increase of this magnitude which has a duration of only a few tenths of a second or less should not produce any significant additional degradation of the cladding lifetime capability.

The passage of the small gas bubble through the core should not be a cyclic process where the same bubble constantly enters the reactor with the coolant after making the primary loop circuit. Once the bubble passes through the core and reaches the fairly low velocity outlet plenum pool, it should rise through the sodium and be deposited in the reactor cover gas. If this bubble does not settle out in this first low velocity region it could do so in the IHX inlet plenum where again there is a low sodium velocity.

Even though the passage of bubbles along the same pin would not be expected to be a cyclic process, an analysis has been performed to determine the effect of 10^6 cycles of a 25°F temperature rise at the same cladding position due to small gas bubble passage. The results showed that no significant cladding damage would be incurred.

15.4.2 Control Assemblies

15.4.2.1 Stochastic Absorber Pin Failures

15.4.2.1.1 Prevention and Detection

Stochastic failure is a postulated random failure due to some extraordinary cause outside the predictable realm. Conservative design philosophy provides margins which minimize stochastic failures. Accordingly, the design of the absorber pins is based on conservative loads and structural criteria, and a high degree of quality control is exercised in the manufacture and assembly of the pins and their assembly in the core.

The design criteria for the absorber pin given in Section 4.2.3.1 include conservatism to ensure safe and reliable performance. Steady state stress limits are held below the proportional limit whereas failure would be expected near the ultimate stress. The cumulative damage function and strain criteria for pin failure includes conservatism by use of worst case loads for all operating modes, upper bounds on plant conditions and 2σ hot channel factors, with 3σ hot channel factors used for the worst case emergency event.

Stringent quality inspections of the pin components and assemblies assures that the occurrence of manufacturing defects are minimized. Assembly of pins of incorrect enrichments in a control assembly is prevented by geometric discrimination features of the structures as is placement of a control rod in an incorrect core position (Section 4.2.3).

Therefore, failure of an absorber pin is unlikely under any predictable conditions. Adequate margins are provided for worst case loads and environments to prevent such failures, and the effects of a postulated failure on operation and control of the reactor can be shown to be minimal.

15.4.2.1.2 Thermal Effects of Gas Release

The potential mechanisms for absorber pin failure propagation resulting from the thermal effects of helium release from a failed absorber pin include gas jet blanketing of an adjacent pin and flow reversal resulting from rapid gas release.

Experimental determination of the effects of gas jet blanketing of adjacent fuel pins in core fuel assemblies was discussed in Section 15.4.1.1.3. It was found that the maximum effect was a temperature increase of about 432°F at a surface heat flux of 7.93×10^5 Btu/hr-ft^2, with the temperature increase proportional to the heat flux. This maximum effect occurs over a narrow range of hole sizes and of pressure ratios across the hole.

These results were applied to the hottest absorber rod including hot channel factors and overpower. It was conservatively assumed that both the maximum cladding midwall temperature and the maximum linear heat rating were at their maximum values of 1225°F and 9.43 kW/ft., respectively, even though these values could not occur on the same pin and at the same elevation. The result was 110°F increase in maximum cladding midwall temperature to 1335°F. This is well below the temperature of 1600°F which may be assumed as the guideline failure limit for short term transients. Therefore, although the geometrical configuration of the experiment differed from that of the control rod, it is considered very unlikely that an absorber pin cladding failure would result in cladding failure of an adjacent pin due to jet blanketing.

Furthermore, if failure of an adjacent pin were assumed to occur, the resulting jet would be directed back to the original failed pin, and therefore, failures would be self-limiting (see Section 15.4.1.1.3).

With regard to the effect of flow reversal resulting from rapid gas release, a conservative analysis was done to ascertain how long an absorber pin would have to be completely insulated before the failure temperature (assumed to be 1600°F) would be reached. The length of time required for the cladding temperature to reach a specified failure point depends on the absorber pin power generation rate in the area covered by the gas, the size of the area covered, and the rate of cooling by the gas and any entrained liquid in the gas. An analysis was performed conservatively assuming that the gas contributed no heating or cooling to the neighbor cladding, the gas surrounded the pin (360° angular coverage) totally eliminating heat removal, the maximum linear power was 9.43 ft. and the initial cladding midwall temperature was 1255°F (even though these two conditions never occur together), and the heat flux into the cladding remained constant (even though it would actually decrease as the cladding temperature increased). For this case, the time required for the cladding midwall temperature to reach 1600°F was 2.3 seconds.

The analytical model which was developed to determine the effects of rapid gas release (Section 15.4.1.1.2) was applied to the Control Rod Assembly. The results for a gas plenum pressure of 4000 psia and a large rupture located at the bottom of the Control Rod are shown in Figure 15.4.2.1-1. The maximum gas blanketing time resulting from an absorber rod failure is about 0.15 seconds, which is less than the conservatively calculated 2.3 seconds required for the insulated cladding temperature to reach 1600^{o}F. Therefore, pin failure propagation as a result of rapid gas release from a failed absorber pin will not occur.

15.4.2.1.3 Mechanical Effects of Gas Release

Absorber pin cladding failure results in the coolant adjacent to the rupture being pressurized. The pressure available may be inferred from out-of-pile cladding burst experiments conducted in connection with EBR-II, which were discussed in Section 15.4.1.1.4. An analytical model was developed in which all of the resistance to gas flow from the pin was assumed to occur at the rupture. The gas bubble was assumed to be spherical and expanding within an infinite sea of incompressible liquid. This simplified approach neglects the effects of the solid surfaces present. The model showed very good agreement with the experimental data in determining the peak bubble pressure although it predicted more rapid pressure decay after the peak was reached. The peak pressure showed little influence of rupture area or of plenum volume. The main effect was that of plenum pressure as is shown in Figure 15.4.1.1-3. Using this figure for the absorber pin, which has a peak plenum pressure of about 4000 psi, the peak bubble pressure is predicted to be about 540 psi.

Because of the high acoustic velocity in sodium of the order of 7000 to 8000 feet per second, a pressure pulse with a rise time of the order of microseconds would be required for a significant pressure differential across an absorber pin. The pressure pulse risetimes found in the EBR-II ducts tests and from the analytical model were several hundred microseconds, and therefore, no appreciable pressure differential across the pin is expected by this mechanism. However, the case was considered of a force acting on an adjacent pin or duct wall as a result of a gas jet from a failed pin impinging on the adjacent pin or duct wall. Assuming the jet to be deflected at right angles, the maximum impulse imparted to the adjacent structure resulting from the gas release from both plenums is 0.597 lb_f-sec.

The maximum impulse attainable from a failed absorber pin is insufficient to fail an adjacent pin. Using a method discussed in Section 15.4.1.1.4 for determining the maximum impulse attainable from a failed pin, it can be shown that this pulse is not enough to cause progressive failure. This conclusion is based upon a study of failure propagation (Ref. 1) in which the critical bending moment--the moment at which a fuel pin buckles to form a plastic hinge--is developed as well as the maximum moment generated by a given impulse on the pin. The failure impulse is defined as that impulse which results in the maximum moment equal to the critical moment.

15.4.2.1.4 Long Term Effects of Absorber Pin Failure

As discussed in 15.4.2.1.3, pin failure occurring near the end of an operating cycle would have minimal effects due to the short time to replacement at the cycle.

Postulated failure of a pin early in life also would have little effect on the mechanical function of the control rod. Erosion tests of B_4C in flowing sodium at 5 fps (Ref. 48) have indicated very small loss rates of approximately 50 mg/cm^2 per 100 hr. Fault sizes in pins have been estimated between 10^{-2} and 10^{-5} cm^2. Because of the small failure dimension, contact between the sodium and the B_4C is unlikely. However, any contact would be at almost zero velocity, therefore, erosion rates would be negligible if any did indeed occur. Thus, there is a very small probability of functional degradation of the control rod due to early pin failure.

Particle size from erosion tests of irradiated B_4C has been shown to have an average diameter of 0.0012 inch (Ref. 48) Particles of this size are not large enough to settle out in flowing sodium and cause channel blockage. The amount of B_4C in the system in terms of parts per million is so small that abrasive wear in normal flow channels within the core would not occur. With an average particle diameter of 0.0012 inch, the number of larger particles is expected to be small. The small number of particles above about 3 to 4 mils that might exist are expected to settle out before reaching the pump bearings. Particles less than $\sim$3 mils are expected to flush through the pump bearings without interaction. The combined low probabilities for release of B_4C and for particle sizes which could be retained in the pump bearings results in an extremely low risk of pump damage.

15.4.2.2 Overpower Control Rod Assembly

15.4.2.2.1 Prevention and Detection of Control Rod Assembly Overpower

The principal overpower considerations for a control assembly are: locating the assembly in the wrong core location; an over-enriched B_4C pin in a lower enrichment assembly; over-enriched pellets and improper orificing. Discriminator posts at the bottom of the inlet nozzle (see requirements in Section 4.2.3.1.5) prevent full insertion of a control assembly in an undercooled control position or into the wrong core assembly location. The increased elevation of the assembly in the wrong location prevents release of the assembly by the refueling mechanisms. Discrimination features are provided on the absorber pin end caps to prevent loading of a highly enriched B_4C pin into a lower enrichment assembly. The potential for over-enriched pellets in an assembly is minimized by quality control in the assembly process.

Manufacturing requirements limit the tolerance on B-10 content in a pellet to ~4% uncertainty from pellet density and B-10 enrichment. This uncertainty is included in overall heat generation uncertainties for which a 15% uncertainty is currently used for design analyses. Improper association of orifices, discriminators and B_4C enrichment of the pin bundle are controlled by quality assurance procedures including visual inspection.

51 The above mentioned discrimination features and quality assurance procedures assure that an overpower condition in a control assembly would be extremely unlikely. In addition, control rods are overcooled (sodium outlet temperature less than fuel assembly outlet) which provides extra
51 margin against an overpower condition by limiting steady state clad temperature to less than 1225^{o}F even for the unexpected, fully inserted primary control rod. The fully withdrawn secondary control rods and primary control rods partially or fully withdrawn have even greater margins against overpower conditions.

51 Even if an overpower situation occurs, it is shown in the following paragraphs that the resulting B_4C and clad temperatures are below melting or short term clad failure limits. The principal effects of locating an enriched assembly in a natural B_4C location would be an increased rate of pin pressure buildup and slightly higher clad temperatures.

15.4.2.2.2 Consequences of Overpower Pin for Steady State and Design Transients

To consider the consequences of an overpower pin in a control rod assembly in the CRBRP, it was postulated than an enriched primary control pin of 55% B-10 is misplaced in the central (or Row 1) control rod location 13
instead of the 20% naturally-enriched B-10 loadings. The CRBRP primary control system utilizes these two types of B-10 enrichments to load the B_4C absorbers, i.e., natural (20%) and enriched (50%) for its fifteen

37-pin control assemblies depending on core cycles and in-core locations. The current design calls for the use of only natural B_4C in the central location throughout the life of the core. Thus, the postulated overpower pin can only occur in the central assembly since all other locations provide adequate cooling for enriched pins.

The peak power thermal performance of the centrally-located control assembly (C/A#1) is reported in Section 4.4. The peak pressure in an enriched B_4C pin in a Row 7 Flat assembly is given in Table 4.2-46. Table
51 15.4.2.2-1 provides peak thermal comparisons between the normal case and 13
the overpower case for the central C/A peak pin at the four key lifetimes of the CRBRP. The results shown in the table are based on the heat generation rates reported in Table 15.4.2.2-2.

The consequences of an overpower pin are higher temperatures in the absorber and clad, increased helium release from the B_4C and higher plenum pressure. It has been estimated that the EOEC plenum pressure in the peak normal pin is 1764 psi and that in the power pin is 3757 psi for the reference rod withdrawal profile used in Table 15.4.2.2-1. The pressure in the plenum is dependent on the rod withdrawal profile. The plenum pressure in the postulated overpower pin at the end of the normal residence time is expected to be very close to a preliminary design guideline maximum pressure of 4000 psi. Therefore failure of the clad may occur.

The consequences of stochastic absorber pin failures were examined in Section 15.4.2.1. Failure was assumed to occur due to a plenum pressure of 4000 psi. Due to the elevated clad temperatures in an overpower pin, clad failures, if they occur, would be expected at pressures of approximately 4000 psi. Thus, the consequences of an overpower pin clad failure are not expected to be more severe than for the stochastic failure. Since the highest clad temperature occurs toward the top of the B_4C pellet stack, the postulated rupture is expected to occur in that region. However, the consequences are not expected to be significantly affected by the axial location of the rupture. Hence, the consequences of the overpower control rod are enveloped by the thermal and mechanical effects of stochastic failure dealt with in Section 15.4.2.1.

Analysis of the thermal expansion and swelling characteristics of the B_4C pellets and the cladding indicates that forced pellet-clad contact is unlikely in the overpower pin, just as it is precluded by design in a normally placed pin. Therefore, failure of the clad is expected to be due to gas overpressure. Experimental evidence from the HEDL Neutron Absorber Technology Program indicates that the characteristic times for release of helium from B_4C are long compared to typical times associated with reactor transients. There is also evidence that pellet swelling is insensitive to temperatures above $2000^{\circ}F$ and that at lower temperatures the swelling varies inversely with temperature (Ref. 64). The temperatures of the absorber pellets, as shown in Table 15.4.2.2-1, are lower than the melting point of the boron carbide ($4442^{\circ}F$) by such a wide margin that melting of B_4C was rejected as a possibility under all the design transients in normal pins as well as the overpower pin. The pressure of the plenum gas is expected to rise during a transient due to the increase in temperature. Thus, it is expected that the failure mode of the overpower pin will not be affected

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by reactor transients since failure during steady state operation has already been postulated. Although a transient could cause a failure earlier in life, the mode of failure (i.e. gas overpressure) would be the same.

The failure of the clad is expected to occur as a pinhole due to stochastic failure of intergranular cracking. Such a failure would not result in as high an impulse as a gross rupture. While the mechanism of failure could affect the extent of damage caused by clad failure, the worst consequence would be that the central control assembly becomes inoperative. This could only occur if the inner duct (which is 44 mils thick) is deformed by the gas jet to such an extent that forced contact occurs between the inner and outer duct. This is extremely unlikely, since an irradiated inner duct would be expected to crack at a corner and relieve the internal pressure prior to allowing sufficient deformation to jam the rod in the outer duct. No deformation is expected on the outer duct because of its thickness (120 mils) and support from adjacent assemblies. The safety of the reactor is not jeopardized with one assembly inoperative because the design of the plant assures shutdown capability with the control rod of maximum worth being stuck and assuming that the secondary shutdown system does not operate. Also the central rod is expected to be used for fuel burnup compensation and is not needed to satisfy shutdown system requirements. 13

15.4.2.2.3 Thermal <u>Effects of Postulated B_4C Particle Release and Interaction</u>

For the operating condition of the CRBRP considering 15% overpower and the misplacement of a high enriched absorber pin, the maximum B_4C temperature is expected to be at least 1500°F below the melting temperature. Thus, the existence of any molten B_4C and the postulation of any interaction of 13
the type such as the MFCI (molten fuel - coolant interaction), that is, rapid transfer of heat from some molten B_4C to liquid sodium through a failed pin, must be regarded as highly improbable. Thus, only the question of B_4C erosion in the flowing sodium if failed pin cladding were postulated, must be considered. The maximum erosion rate at the end of a fuel cycle is considered to be much less than the 9% observed and reported (Ref. 48). The 9% erosion is based on an extremely severe case in which approximately 25% of the circumferential area of the B_4C pellets was exposed to flowing sodium. Test results have shown that particles of eroded B_4C are generally less than 0.002 inch in diameter and substantial fractions of them become dissolved in the system sodium. Thus, there is no possibility of flow blockage resulting by the erosion and deposition of B_4C pellets.

The mechanical effects of gas release particularly towards the end-of-cycle due to high gas pressure build up in the control room assembly pins are discussed in Section 15.4.2.1.3. To ensure conservatism, the gas release study postulated a 4000 psi total plenum pressure in an absorber pin before its release. This higher value of pressure was based on the assumption that a control assembly would be left in the core fully inserted due to failure of the CRDM/Driveline system. In the light of this postulation, the peak pin temperatures were examined for the fully inserted condition for the central 13
(Row 1), Row 7 flat and Row 7 corner rods. Table 15.4.2.2-3 lists the peak pin conditions for the three differently located rods under the postulated BOEC condition of fuel rod insertion in core (instead of programmed withdrawal). Again, the B_4C fuel temperature is considerably below the melting temperature. The cladding and coolant temperatures are also considerably below design limits. The conclusions are again that no molten B_4C and coolant interaction will take place, since none of the design limits are exceeded.

15.4.2.2.4 Mechanical Effects of Postulated B_4C Particle Release and Interaction

The maximum particle size of B_4C eroded by sodium flowing at 5 fps has been shown to be 0.002 inch (Ref. 48). The size and velocity of these particles is insufficient to cause damage to adjacent pins or ducts. Additional details on these subjects are given in Section 15.4.2.1.4.

15.4.2.3 Flow Blockage of a Control Assembly

15.4.2.3.1 Prevention and Detection of Assembly Blockage

(This subject is covered in Section 15.4.1.3.1).

15.4.2.3.2 Consequences of Control Assembly Flow Blockage

The result of partial blockage of a control rod bundle inlet is a reduction in flow rate similar to that of a fuel assembly, as discussed in Section 15.4.1.3. Its effect on the flow rate is relatively small though greater than in a fuel assembly because redistribution to the control rod bypass flow is possible. Since temperatures are lower in the control rods, consequences of flow blockage will be less severe than in a fuel assembly.

15.4.2.3.3 In-Control Assembly Passive Local Blockage (Non Heat-Generating)

Using the method discussed in Sections 15.4.1.3.3 and 15.4.3.3.3, In-Radial Blanket Assembly Passive Local Blockage, the value of $t_R = \frac{\tau U}{d_B}$, the dimensionless residence time, was calculated to be 19.6 for a 10% central blockage. However, the value of 22.5 calculated for a core assembly was used. Use of the core value is considered to be conservative. As cited earlier, although the control assembly B_4C rod pitch-to-diameter ratio is 1.05, the number of pins per square inch of assembly cross section is small compared to that in a core assembly. A decrease in the B_4C pin pitch-to-diameter ratio paritally offsets a decrease in the number of B_4C pins per square inch of cross section.

Control Assembly Six-Channel Blockage t_R = 22.5

	Midplane	Exit Plane
Wake Coolant Temperature (°F)	927	956
Local Coolant Temperature Without Blockage (°F)	808	900
Linear Power (kW/ft)	3.6	1.7
Inside Cladding Temperature (°F)	1002	994

These cladding temperatures are considerably below the failure temperature. Although a complete two-dimensional edge blockage is assessed to produce higher cladding temperatures, these temperatures will be somewhat below steady state non-blockage cladding temperatures in a core assembly. Although these calculations considered a control assembly design having 61 rods, the large margins calculated that the 37 rod design would also result in acceptable temperatures.

Refer to Section 15.4.1.3.4 for a discussion of heat generation blockages. The effects are expected to be similar but far less significant due to lower heat generation rates.

15.4.2.3.4 Postulated Local Boiling Effects; Control Assembly

Following the same approach as that used in fuel assembly and radial blanket analysis, the calculated sizes of central blockages required to cause boiling at the midplane and at the exit plane of a control assembly (assuming the control assembly is completely inserted) are found in the table below.

Boiling Temperature	d_B	β Approximately	t_R
°F	Inches	---	--
Midplane			
1800	4.35	0.84	22.5
1700	4.11	0.75	22.5
1700	3.9	0.68	22.5
Exit plane			
1750	7.9	--	22.5
1700	7.59	--	22.5
1650	7.9	--	22.5

Because of the large size central blockages that are calculated to be necessary for boiling to occur, the probability for detecting a true two-dimensional edge blockage is greatly enhanced. Although the calculations are based on a dimensionless residence time t_R extrapolated from central blockage data, they are also judged to be valid for off-central blockage. As discussed earlier, the effect of a porous blockage would be to increase the required blockage size to cause boiling. Moreover a value of $t_R = 22.5$ based on core assembly characteristics is considered to be conservative.

This analysis was based on a design having 61 rods in a control assembly. A reduction in the number of rods from 61 to 37, holding P/D constant, will reduce the severity of blockage and increase the size blockage required to cause boiling based on the current calculational model.

15.4.2.3.5 Long Term Effects of a Hot Spot

The long term effects of a hot spot in the control rod may be to shorten the lifetime. The effects of failure due to a hot spot would be similar to stochastic failure discussed in Section 15.4.2.1.3. Hot spot effects are included directly in the absorber and clad temperature calculation by application of the hot channel factors. Pin lifetimes including stress limits are evaluated using 2σ hot channel factors. Consequently, hot spots are not expected to significantly affect the predicted control assembly lifetimes.

15.4.3 Radial Blanket Assembly

15.4.3.1 Stochastic Radial Blanket Pin Failure

A stochastic pin failure is a random failure that is unpredictable. Such a failure could result from a random cladding defect that goes undetected during manufacturing. Conservative design philosophy provides margins that minimize stochastic failures.

Failure of radial blanket rods under normal operation and transient design conditions is not predicted. The strength of the cladding, the predicted operating temperatures and the design-lifetime of radial blanket assemblies preclude systematic radial blanket pin failures. However, random pin and cladding failures cannot be precluded because of cladding and weld defects which may not be detected during fabrication.

It is expected that radial blanket rod failures can be easily accommodated by the primary coolant system on the same basis as fuel assembly failures.

In general the following sources for random failure of fuel rod should be ruled out:

a. Desposition of corrosion products on the cladding, because mass transport occurs from the cladding heat source area to the heat exchanger heat sink. Deposition of corrosion products on the cladding is limited to the cooler regions, and the amount is insignificant.

b. Cladding wastage due to corrosion and wear because these effects have been considered in design. Cladding wastages which lead to cladding failure can only be caused by changes of the system parameters, for example fuel or sodium oxygen content or other foreign material.

c. Reduction of heat transfer from the fuel to the sodium, because expected changes of the heat transfer have been conservatively considered by design. Other changes can only be caused by deviation of fuel, cladding or coolant from specifications or anticipated conditions.

A break of the cladding is the most probable random failure mode. The consequences of a break of cladding are:

a. A short-term or slow fission gas release which has generally no long-term effect on the rod performance. A short-term release of fission gas can only occur in the unfueled fission gas plenum region of the rod. A rapid short-term gas release in the core region is prevented by a close fuel to cladding contact and high flow resistance for the gas from the plenum to the crack.

b. Entry of the sodium into the fuel pin either by inhalation during power fluctuations, or capillary action through the cladding breach. This could result in a potential for a sodium-fuel chemical reaction.

Experience with failed fuel in EBR-II and other reactors has shown that sodium generally does not leak through small cracks into the cladding.

Random failures are not considered to be a consequence of a particular operational event. It has been shown in Sections 15.2 and 15.3 that the radial blanket rods are designed to withstand the severe design transient events Reactivity Insertion and Undercooling without systematic failures. However, a transient event may initiate or enhance random rod failures. Should a random rod failure occur no molten fuel would be present. It has been shown in Sections 15.2 and 15.3 that fuel melting does not occur during transients. The probability for the presence of molten fuel is discussed in Section 15.4.3.2.

Prior operating experience with failed fuel in sodium cooled reactors indicates that the consequences of random failures are minor and insignificant to the safety of the reactor (Ref. 1). However, to ensure that all possible consequences of a radial blanket rod failure have been considered an analysis of postulated failure modes was performed. Planned verification of the analysis through development testing is identified in Section 1.5.

The following postulated consequences of a rod failure have been investigated in the following sections.

a. Local Flow Reduction due to rapid fission gas release.

b. The potential of gas blanketing of adjacent rods and occurrence of local hot spots.

c. Occurrence of pressure pulses and effect on rods, cladding and ducts.

d. Effect of fuel particle release on the radial blanket assembly and the system.

e. The consequences of long term radial blanket rod failures.

15.4.3.1.1 Prevention and Detection of Radial Blanket Rod Failures

Random failures of radial blanket rods are minimized by virtue of the design, strict quality control and conservative design margins. When failures occur their impact is also contained by design.

The design for steady-state and transient operation has been evaluated in Chapter 4.2 and Chapter 15. In Section 15.1 design margins for transient events have been identified. Additional margins to prevent further propagation of unpredicted failures are a function of the prior stress and irradiation history and radial blanket component.

Several design features of the radial blanket reduce the impact of failures on adjacent rods.

a. Radial blanket rods have a large fission gas plenum and have a small fission gas pressure. This minimizes pressure pulse on adjacent rods and duct.

b. Radial blanket rods, like the fuel rods, have a fuel pellet hold-down tube in the gas plenum, (see Section 4.2). During a postulated large cladding rupture, the flow impedance of the gas escaping from the fuel holddown tube is increased.

c. The material properties of the cladding are such that no large fractures of cladding have been observed in failed fuel rods in EBR-II.

Cladding failures, should they occur, can be detected in three ways.

a. Gas tagging of the blanket fuel pins in each assembly with traceable gas isotopes will allow detection of any significant amount of fission gas leakage even through a small pinhole in the cladding.

b. If it is postulated that the sodium comes in contact with fuel through a cladding crack, the failure will be detected by delayed neutron detection (DND) if sufficient fissionable material has been removed or exposed to flowing sodium. At the start of operation (BOL) for each assembly when the fissionable material content in the fuel is below the detectability limit, failures may not be detected. For these cases, the potential adverse effects due to sodium fuel contact are considered in Section 15.4.3.3.2 in conjunction with flow blockages.

c. The leaching of small fuel particles into the reactor cooling system can also be detected through periodic sampling of the sodium. Furthermore, the coolant cleaning system, the cold traps with filters, will extract small fuel particles together with other particulate substance.

15.4.3.1.2 <u>Local Flow Reduction Due to Fission Gas Release and Pressure Pulse</u>

A potential mechanism for radial blanket pin failure propagation is a flow reversal resulting from rapid gas release. A conservative analysis was done to ascertain how long a radial blanket pin would have to be completely insulated before the failure temperature (assumed to be 1600°F) would be reached. Calculations were made for 3σ plus 115% power EOL conditions for which the cladding temperatures and linear heat rating are highest. The worst location was found to be at an elevation of about 0.6 of the blanket height. The heat flux into the cladding was conservatively kept constant, even though it would actually decrease as the cladding temperature increased. The cladding thickness was reduced to 0.008 inch to allow for fretting and corrosion. For this case, the time required for the cladding midwall temperature to reach 1600°F was 0.096 second.

High internal gas flow resistances will prevent gas ejection rates from a cladding failure in the fuel region from being sufficiently rapid to cause flow reversal. The most severe case is for a large cladding rupture at the top of the blanket. An analytical model for the effects of rapid gas release was described in Section 15.4.1.1.2. This model was applied to the radial blanket assembly geometry and operating conditions. The following parameters were used in the case of interest:

Initial gas pressure	275 psia
Initial coolant velocity	10 ft/sec.
Rupture location	Top of blanket

The results of the cases analyzed are shown in Figure 15.4.3.1.2-1 For the case where 10 pins were postulated to rupture simultaneously, .. was found that a slight flow reversal occurred, with the lower gas-liquid interface penetrating downward approximately 1 inch below the rupture location. Flow of the lower liquid slug resumed in the upward direction after about 0.03 seconds and the lower gas-liquid interface was back to the rupture location after about 0.05 seconds. Thus, the maximum time during which any portion of a radial blanket rod could be blanketed by gas is 0.05 seconds. This is less than the 0.096 seconds required for the cladding temperature to reach 1600°F under worst conditions with complete insulation. Therefore, even if the break were to occur in the worst location in the blanket pellet region and the internal gas flow resistance between the gas plenum and the break were neglected, the cladding would not reach the temperature at which further rod failures would occur.

It should be mentioned that the gas plenum pressure of 275 psia used in the calculations was based upon a preliminary radial blanket assembly shuffling scheme, which has been changed. The present shuffling scheme under consideration results in a maximum plenum pressure of 380 psia (See Section 4.4). However, since the preceding analysis showed that there would be no pin failure propagation if 10 pins with plenum pressure of 275 psia were to fail simultaneously, and since no mechanism has been identified which would cause simultaneous multi-pin failures, it is concluded that if a pin with a plenum pressure of 380 psia were to fail, no further pin failures would occur.

15.4.3.1.3 Gas Blanketing of Adjacent Pins

Experimental determination of the effects of gas jet blanketing of adjacent fuel pins was discussed in Section 15.4.1.1.3. It was found that the maximum effect was a temperature increase of about 432°F at a surface heat flux of 7.93 x 10^5 BTU/hr-ft^2, with the temperature increase proportional to the heat flux. The maximum consequence on adjacent pins occurs over a narrow range of hole sizes and of pressure ratios across the hole.

Applying these results to the hottest radial blanket rod at 3σ plus 115% power conditions, the maximum local cladding temperature resulting from fission gas jet blanketing occurs at an elevation of 0.65 of the blanket height and is 1541°F which is below the limiting value of 1600°F.

Internal resistance to gas flow and a transient, rather than a steady-state, analysis would further reduce the maximum cladding temperature with jet blanketing. Therefore, although the geometrical configuration of the experiment differed from that of the radial blanket, it is considered unlikely that a radial blanket rod cladding failure would result in cladding failure of an adjacent rod due to jet blanketing.

Furthermore, if failure of an adjacent rod were assumed to occur, the resulting jet would be directed back into the original failure subchannel, and therefore, failures would be self-limiting.

15.4.3.1.4 Mechanical Effects of Fission Gas Release on Rods and Duct Walls

When a radial blanket rod fails with rapid failure of the cladding, and fission gas is released,the coolant adjacent to the rupture is pressurized. The pressure available may be inferred from out-of-pile cladding burst experiments conducted in connection with EBR-II, which were discussed in Section 15.4.1.1.4. An analytical model was developed in which all of the resistance to gas flow from the rod was assumed to occur thru the rupture. The gas bubble was assumed to be spherical and expanding within an infinite volume of incompressible liquid. This simplified approach neglects the effects of the solid surfaces present. The model showed good agreement with the experimental data in determining the peak bubble pressure although it predicted more rapid pressure decay after the peak was reached. The peak pressure showed little influence of rupture area or of plenum volume. The main effect was that of plenum pressure as is shown in Figure 15.4.1.1-3. Using this figure for the Radial Blanket rod which has a peak plenum fission gas pressure of about 380 psi, the peak bubble pressure is predicted to be about 110 psi.

Because of the high acoustic velocity in sodium of the order of 7000 to 8000 ft. per second, a pressure pulse with a risetime of microseconds would be required to produce a significant pressure differential across a radial blanket rod. The pressure pulse risetimes found in the EBR-II duct tests and from the analytical model were several hundred microseconds and therefore no appreciable pressure differential across the rod is expected by this mechanism. However, the case was considered of a force acting on an adjacent fuel rod or duct wall as a result of a gas jet from a failed rod impinging on the adjacent rod or duct wall. Assuming the jet to be deflected at right angles, the maximum impulse imparted to the adjacent structure is 0.271 lbm-sec.

The maximum bubble pressure is approximately 60% of that in the fuel assembly because maximum fission gas pressures in the radial blanket rods are smaller. It will be shown in the following that design margins against failure modes are generally larger in the radial blanket than in the core fuel assembly. Differences in design configuration between radial blanket and core fuel rod and rod bundle increase this margin; for example the radial blanket rods have a larger bending stiffness than the core fuel rods because the rod diameter is approximately twice as large and the wire wrap supports the rods every 2 inches instead of every six inches as in the core fuel assembly.

Impingement of a jet of fluid from a failed fuel rod and an adjacent fuel rod was investigated in (Ref.1). A potential failure mechanism which was observed in tests is the local flattening of the cladding and forming of a collapsed hinge in the fuel rod. It is not credible that this failure mechanism can occur in a radial blanket assembly. Figure 15.4.3.1.4-1 shows the critical moment needed to induce a hinge collapse and the moment induced by a high initial pressure pulse on a core fuel rod with 12 inch wire wrap pitch. For a radial blanket rod with 0.253 inch radius the induced moment is below the critical moment even under the severe assumption of Figure 15.4.3.1.4-1. For a rod with 4 inch wire wrap pitch and a pressure pulse of only 90 psi the critical collapse moment is many times larger than the induced moment.

The local buckling stiffness of the cladding is by a factor of $(0.253/0.107)^2 = 5.6$ less than that of the core fuel rod because of the cladding diameter to thickness ratio increase. Local buckling and high local stress could occur due to a pressure pulse primarily at points where reaction forces are transmitted through the wire to an adjacent rod. Application of the formulas of (Ref. 1), Appendix C yield a static local buckling load of 521 lb. The dynamic buckling load is even larger than the static buckling load and considerable time is required to induce buckling. Therefore a pressure pulse of 90 psi cannot induce local buckling of the rod.

The pressure pulse pushes the rod against the adjacent rod. Reaction forces are transmitted through small areas where wire and cladding contact. It is assumed that a 110 psi pressure pulse through a 1 inch long rupture generates a reaction force of 45.5 lb. on an adjacent rod. Based on formulas developed in (Ref. 1), Appendix C the maximum cladding bending stress underneath the wire is 44334 lb/in^2. Conservative assumptions are that the total reaction force is transmitted through one cladding to wire contact point and that load distribution due to rod bending and rod bundle

compressibility is neglected. Should the irradiated cladding burst under this stress, such a failure would not be expected to result in further failures due to the microscopic nature of the breach. A more
realistic and less conservative analytical model would yield lower stresses. 22
It would probably also show that no rod to rod failure propagation can occur, which is also excluded as long as the cladding is so ductile that the dynamic impulse can be absorbed in plastic deformation (see Ref.1, Appendix C).

The mechanical strength of the radial blanket duct was evaluated to withstand fission gas pressure pulse due to rupture of the cladding. The strength of the 0.100 inch thick radial blanket duct wall has only $(0.100/0.120)^2 = 0.69$ times the bending strength of the 0.120 inch thick core assembly duct. The strength to withstand membrane forces is only reduced by 17%. In Section 15.4.1.1.4 it was found that the core assembly duct could withstand a pressure of 300 psi at 1200°F. Therefore, the radial blanket assembly duct strength is sufficient to withstand an internal pressure of at least 207 psi. The maximum pressure pulse exerted on the duct is no larger than 110 psi, and the ducts are safe against rupture by a factor $\sim$2. The evaluation in Section 15.4.1.1.4 is based on very conservative assumptions; for example an axially uniform pressure, a static load and quasi static strength but rupture of the cladding is expected to be local with a slow fission gas release. Hence the safety factor against rupture of the duct is significantly larger than 2.

Even if it is postulated that a radial blanket assembly has ruptured by a pressure pulse, for example at a duct defect; the consequences on the assembly performance would be small. It was shown that the fracture would occur above the fueled region of the radial blanket rod. A crack in the duct wall would open an alternate flow path for the coolant. Therefore, depending on the crack size, the flow through the assembly would decrease above the crack and slightly increase below the crack if the flow resistance through the crack is sufficiently small. In conclusion a crack in the duct above the fueled region does not effect the assembly performance.

15.4.3.1.5 <u>Effects of Fuel Particle Release</u>

Fuel particle release from a radial blanket assembly will:

a. Contaminate the primary system with uranium and slight amounts of plutonium, fission products and

b. Potentially form heat generating flow blockages in the rod bundle.

The release of plutonium to the primary coolant in relatively large amounts can create a contamination problem. Because relatively small amounts of plutonium and fission products are generated in the radial blanket assemblies, released fuel particles should not create a radiation hazard in the primary system. In addition, the amount of uranium released is limited to 10 ppb RDT Standard Al-5T, March 1976. Purity Limits of Operating Sodium Systems. Thus, 7.5 gm of fuel particles released is the maximum allowed by
27 RDT Standard Al-5T, March 1976, before corrective action is taken.

Section 4.2.1.1.3.8 evaluates the reactions between the fuel or blanket materials and the sodium coolant. These reactions form a product which is much less dense than the original fuel or blanket pellets. Fuel-sodium reaction products can cause small, partial flow blockages by either expanding the cladding to a larger diameter or depositing on the outside surface of the fuel rod. The consequences of a partial flow blockage with a heat generating material on radial blanket cladding temperatures is presented in Section 15.4.3.3.3. Effects of a cladding temperature increase on reducing the cladding lifetime is discussed in Section 15.4.3.3.6.

15.4.3.1.6 Long Term Effect of Radial Blanket Rod Failures

The preceding sections discussed the short term effects of fission gas release due to a random cladding failure. This section discusses the possibility of delayed and long term effects of fuel rod failures in the radial blanket. In preceding sections it was shown that random failures can be detected by gas tag sampling, DND and sodium sampling. A signal from each failure detection system indicates a more severe rod damage. It will be shown that operation with failed blanket rods will not have significant adverse long term effects on the reactor safety. The sodium cleanup system is designed to separate fission product and fuel residuals from the coolant even if 1% of the fuel failed.

One postulated long term effect of cladding rupture is an intermittent escape of fuel particles through the cladding rupture. It has been frequently observed in failures of test rods in EBR-II that the rupture size and leak rate for fission gas pressure is so small that it cannot be visually observed. When the crack is large enough to expose fuel to flowing sodium fuel particles may get into the flow channel. In Section 15.4.1.3.1 it was shown that any fuel particles which will fit into a rod bundle flow channel will generally be swept out to the rod bundle of high power assemblies in inner radial blanket positions. In peripheral radial blanket positions, where the assembly power and flow velocities are low, the potential to sweep out fuel particles is marginal. However, at low power, fuel rod damage is less likely to occur than at high power, and it was shown in Section 15.4.3.1 that random rod failures do not involve a safety risk.

Because of failure of the cladding, sodium could enter the rod due to reduction of gas pressure upon cooldown accompanying reactor shutdown. Various adverse effects of sodium absorption (or logging) of the fuel have been postulated as follows:

a. Leaching of fission products from the fuel by the sodium and subsequent deposition in the primary system

b. Generation of high pressure inside the rod due to sodium vapor, possibly causing:

- Internal pressure on the cladding
- Further cracking or disintegration of fuel

c. Fuel pin swelling due to fuel-sodium chemical reaction

If some sodium entered the fuel then some fission product leaching might occur. It would not be expected that the effect on the rod would be significant. The sodium cleanup system could remove these products and no significant adverse effects could occur. Leaching of radial blanket fuel into the sodium and coolant system causes less contamination than leaching of core fuel. The enrichment of radial blanket fuel at end of life is typical for the enrichment of LWR fuel and LWR reactors have operated with failed fuel without significant contamination.

The generation of sodium vapor pressure inside the pin is very unlikely since the pressure can be relieved by the rupture which allowed the sodium to enter originally. The absence of problem due to vapor formation is supported by sodium logging experiments (Ref. 49) in which defects were simulated by 0.005 inch diameter holes in the cladding and fuel cladding gaps were of the order of 0.002 inch to 0.003 inch. Fuel rods were thermally cycled employing heating rates 200 times faster than those expected in bringing a reactor up to power. No evidence of fuel pin deterioration caused by the sodium was found. It appears that there is no reason to believe that any problem should arise due to sodium vaporization within the failed pin. These tests where performed for core fuel rod geometries. The test results will be reevaluated to assess the validity of these test results when applied to radial blanket fuel rods.

Fuel pin swelling due to sodium logging has been observed in predefected pins, failed pins, and sodium-bonded pins (Ref. 50, 51, 52, 53). The potential for this phenomenon to occur in radial blanket rods has been and will be further evaluated considering differences in rod size, power, fuel composition and stoichiometry. Depending on the result of this study planned tests with failed rods may be initiated. The degree of the fuel-sodium chemical reaction and the extent of the resultant fuel pin swelling appeared to depend primarily on the quantity of oxygen available for reaction. The secondary effects of temperature and fission product concentration may have also been important contributors. The quantity of oxygen available for chemical reaction is dependent on the initial fuel fabrication O/M ratio, the original oxygen impurity level of the coolant, the net oxygen liberated as a result of fissioning and recombination with fission products, the quantity of oxygen leached from the fuel into the coolant, and since the reaction occurs mainly at the fuel surface, the degree of migration of oxygen due to fuel temperature gradients.

15.4.3.2 Overpower Radial Blanket

15.4.3.2.1 Prevention and Detection of Erroneous Shuffling or Misorificing

Cooling requirements for the radial blanket assemblies are a function of reactor residence time and the proximity to the reactor core center. These requirements are met by a combination of: 1) coolant flow control and 2) shuffling of the assemblies within the radial blanket zone.

The required coolant flow control for each radial blanket assembly is provided through fixed orifices in the inlet modules (Section 4.4.2.4). These orifices provide a graduated coolant flow with the greatest flow nearest the center of the core. Development testing (Section 1.5), quality control during manufacture and site testing (Section 4.2.1.4) will be utilized to establish the proper orificing for each blanket assembly and assure the coolant passages are free of obstructions.

The required shuffling of blanket assemblies within the radial blanket zone is accomplished through an established schedule (based on residence time) and a preprogrammed shuffling routine carried out by the fuel handling machines (see Section 9.1). Numerous safeguards have been established in the shuffling routine to preclude the possibility of erroneous shuffling, however, due to the complexity of the multi-step shuffling process, misplacement of a blanket assembly and the resulting consequences must be considered.

A worse case postulated shuffle has been analyzed (See Section 15.4.3.2.2) and the results show that there are no severe consequences associated with this event. Some fuel melting was found to occur, however, experimental evidence based on FFTF fuel assemblies showed that as much as 80% molten fuel within a fuel pin does not necessarily lead to cladding failures (see Section 15.4.3.2.2). If cladding failure were postulated to occur the present analysis of molten fuel coolant interaction (MFCI) shows there are no adverse effects, either thermal or mechanical (see the following sections for a detailed discussion).

Even though the consequences of an erroneous shuffle are considered negligible, several design features have been incorporated in the fuel handling process to safeguard against the misplacement of a radial blanket assembly.

- A discriminator built into each of the reactor assemblies permits full insertion of only a radial blanket into a radial blanket location.
- An accounting system provides a record of the location for each specific assembly within the radial blanket zone.
- Before reactor startup, an administrative review of the core loading pattern, residence time, and shuffling history (Section 16.3.1.3) will provide a second check of correct shuffling.
- Each time an assembly is withdrawn from the blanket the remote identification system incorporated into the refueling machine grapple will read out the reactor assembly type, serial number, and angular orientation.

- The start and finish locations of the IVTM are automatically compared with the programmed moves and must be substantiated by the operator.

- If an erroneous shuffle were to end with an empty Core Component Pot (CCP) in the transfer position, the EVTM grapple load weighing system would quickly detect the lack of load and the shuffle routine terminated and corrected.

15.4.3.2.2 <u>Consequences of Overpower Pin for Steady State and Design Transients</u>

The case of a repeated failure to shuffle which causes a radial blanket assembly to remain in the highest flux radial blanket position until, or beyond failure was studied. The thermal consequences of this repeated failure to shuffle are identified in Tables 15.4.3.2-1 and -2. This particular error sequence was selected for analysis for the following reasons.

1. Failure to shuffle any other assembly would be less serious since that assembly would be operating at lower power and lower burnup.

2. Shuffling of an outer row, high burnup radial blanket assembly to an inner position is considered highly unlikely. The present shuffling scheme is consistently "in-out", that is, the assemblies are shuffled only in an outward direction and administrative procedures preclude any inward handling. As discussed in Section 15.4.3.2.1, multiple shuffling errors are required for this failure, in addition to an error in the administrative review of the refueling records.

3. Consecutive failures to shuffle may be the least unlikely error sequence because of the possibility of administrative error (human error) in setting up the original refueling program, rather than multiple mechanical and human errors as involved in shuffling to a wrong position.

The peak operating conditions of the radial blanket pin with the highest fuel temperature are listed in Tables 15.4.3.2-1 and -2 as a function of reactor cycle with a failure to shuffle after two cycles and yearly failure to shuffle each year thereafter. Table 15.4.3.2-1 provides this information for the 2σ hot rod (calculated with a confidence level of 97.72% to be the hottest rod in the blanket) at 100% power and Table 15.4.3.2-2 shows the results for the 3σ rod (99.86% confidence level) at 115% power. The assembly was initially irradiated at the start of cycle 7 (SOC 7) and, for the purpose of this analysis, operating condition predictions were continued until the end of cycle 12 (EOC 12). Included in Tables 15.4.3.2-1 and -2 are the amounts of molten fuel present in the highest power pin with a fuel melting temperature of 5000°F. (5000°F corresponds to a power to melt rating of about 20 kW/ft).

15.4.3.2.3 Thermal Effects of Postulated Molten Fuel-Sodium Interaction

If successive failures to shuffle are postulated, in spite of the precautions designed to prevent errors, molten fuel will eventually occur in the radial blanket fuel elements. However, experimental evidence based upon FFTF type fuel elements irradiated in TREAT (Ref. 54, 55, 56) showed that as much as 80% molten fuel within a fuel pin does not necessarily lead to cladding failure. It is expected that molten fuel within a radial blanket fuel element would behave similarly, and, thus, molten fuel within a radial blanket fuel element is not expected to be hazardous.

Cladding design limits will be exceeded during the first year of irradiation following failure to shuffle (Cycle 9 of Table 15.4.3.2-1) in the highest flux radial blanket position. The postulated failure mode is cladding rupture due to fission gas pressure which will occur at the axial location where the cladding temperature is a maximum. This location is (1) above the molten fuel ejection, and (2) several inches below the top of the pellet stack so the maximum rate of gas release will be reduced by the hydraulic resistance through the pellets or the pellet/cladding gap.

This failure mode is based on analytical evaluations of cladding ductility limited strain versus time at various axial locations along the hot rod of a misshuffled radial blanket assembly. For this evaluation, the cladding strain analysis techniques used in the steady state cladding analysis of Section 4.2.1.3 were utilized. Cladding loads due to high cladding temperatures and internal pressure from the fission gas were considered, with effects of cladding wastages included as outlined in Section 4.2.1.1.3.4. Worst-case cladding temperatures and plenum pressures used for this cladding stress-strain analysis are described in Section 15.4.3.2.2.

The results of this analysis predicted that the cladding hot spot location would experience the greatest cladding ductility limited strain of any point along the rod length, for all times considered. Thus, for the assumed cladding loads, it is expected that cladding failure would occur at the hot spot location. This failure location was also predicted for normal, steady state radial blanket rods, as well as for steady state fuel rods where fuel-cladding mechanical loads occur at the core midplane location.

This analysis predicted that the cladding failure would occur above the region where molten fuel ejection would be expected. However, the uncertainties involved in this calculation are such that the potential for molten fuel ejection must be considered. Therefore, the MFCI is considered in detail in Section 15.4.3.2.4.

Secondary loadings from radial blanket failures have been considered. These loadings include mechanical effects from MFCI (Section 15.4.3.2.4), duct deformation in adjacent assembly (Section 15.4.3.2.5) and duct cracking (section 15.4.3.2.6). 15

Pin to pin failure due to fission gas ejection is discussed in Section 15.4.3.1.3.

By postulating the preceding failure sequence, a molten fuel-coolant interaction due to a slumping molten fuel type failure can be postulated, and hence, a MFCI may then be postulated. This type of accident for the fuel assembly MFCI was discussed in Section 15.4.1.2.3 and the radial blanket assembly MFCI would be expected to behave similarly.

15.4.3.2.4 Mechanical Effects of Postulated Molten Fuel-Sodium Interaction

It is postulated that overpower fuel rod cladding failure does occur so that the consequences can be assessed to determine ultimate safety margins. Section 15.4.3.1.3 demonstrated that fuel rod failure propagation due to fission gas release does not occur. Thus, the only mechanism to cause rod-to-rod failure is molten fuel jet impingement of a neighboring fuel rod.

The likelihood of molten fuel jet formation is very small because the molten fuel would be fragmented when it strikes the relatively cool sodium. This fragmentation process has been observed in many experiments for many different combinations of materials. Since fuel fragmentation will occur, it is important to know the final disposition of the fragments. To be swept from the radial blanket by coolant, the fragments must be smaller than the channel size (0.04 to 0.13 inch) and the coolant velocity must be adequate. Data (Ref. 57, 58) compiled for fuel assemblies show particle sizes range from 0.004 inch to 0.04 inch, which can be swept out at coolant velocities of 2 ft. per second. This required velocity is less than the velocity in the lowest flow assembly, and the particle sizes are smaller than the channel, hence, the available coolant velocities are sufficient to sweep molten fuel fragments out of the radial blanket. The thermal effects on the cladding should fuel particle entrapment occur, are discussed in Section 15.4.3.3.3.

In the event that no fragmentation occurs, the molten fuel jet could impinge on a neighboring fuel rod, resulting in high cladding temperatures. However, this high temperature occurs only at the very small cladding area covered by the fuel jet, since heat conduction to the coolant still exists from the cladding adjacent to the impingement area. Thus, cladding failure over a large area is unlikely in the event a fuel jet does form. The local nature of this overheating precludes coolant blockage due to cladding "ballooning" caused by high cladding temperatures and fission gas pressure. This has been proven by (Ref. 26) experiments in which it was found that 20% CW 316SS cladding did not deform (or "balloon") when rupture occurred at high temperatures. Thus, failure propagation by cladding deformation during high temperature rupture is highly unlikely.

It can be postulated that solidified molten fuel from an improperly shuffled radial blanket assembly may, somehow, accumulate against a neighboring outermost core assembly duct wall, rather than fracturing and being swept away by the coolant.

A scoping analysis using conservative assumptions was performed using the COBRA-III code of 1/12 section of the outermost core assembly. A maximum axial heat flux profile was imposed upon the outside of the core assembly duct wall corresponding to that resulting from the maximum non-boiling fuel thickness at each axial location over the entire length of the assembly. The wall heat flux was determined using the maximum possible heat generation rate (which includes 15% overpower corresponding to 37.6 kW/ft. maximum linear power in the hottest radial blanket pin before failure) of an improperly shuffled radial blanket assembly. Other parameters chosen in the analysis were also very conservative, e.g., K_{fuel} = 1.6 Btu/Hr/Ft/°F, fuel T_{boil} = 6100°F, perfect thermal contact between fuel and wall, etc.

The results of this analysis showed that no local coolant boiling occurred in any of the core fuel assembly subchannels. This was due to the full coolant flow that still existed in the affected core fuel assembly, and the mixing between subchannels. Thus, fuel failure propagation from the radial blanket to the core cannot occur as a result of fuel released from an improperly shuffled radial blanket assembly accumulating upon a core assembly duct wall.

In the case of a molten fuel jet directed at a radial blanket duct wall, a hole similar in size to the breach in the radial blanket pin could possibly be burned through both the radial blanket and neighboring core fuel assembly duct walls if fragmentation does not occur. The breach size, if any, would approach that of a small crack. The discussion of the consequences of a crack discussed in Section 15.4.3.2.6 and 15.4.1.2.6 would also be appropriate in this case. Hence, it is concluded that assembly-to-assembly fuel failure propagation would not be probable by fuel jet impingement.

15.4.3.2.5 Consequences of Postulated Duct Deformation of Adjacent Assembly

As shown in Section 15.4.3.1.4, the radial blanket duct walls are safe against rupture from a fission gas pressure pulse by a factor of safety >2. The rupture of a neighbor assembly duct would therefore be unlikely even assuming that a large pressure pulse occurred in an assembly. However, the potential consequences of rupture of a neighbor assembly duct is worthy of discussion to show the depth of protection against a propagation of local failures. The consequences of diversion of flow from a ruptured assembly will be discussed in Section 15.4.3.2.6. There it is estimated that fuel rod cladding overheating would not occur due to flow diversion through a crack. This conclusion, with some qualification is also indicative of the margin to rod failure in the postulated event of neighbor duct rupture. If no significant rod or duct distortion occurred, then leakage could occur and there would still be a margin to the fuel rod integrity limit. However, it is conceivable that the rods in the neighbor assembly might be mechanically loaded by: a) axial bending which might lead to a collapse-hinge; b) cross-sectional deformation; or c) concentrated forces at cladding-wrapper wire support points. This involves plastic deformation in neighbor rods which probably would not occur due to the effect listed above because: a) the stiffness of the radial blanket rod is sufficient to preclude formation of a collapse hinge due to the anticipated lateral loads; b) the rods are very resistant to cross-sectional deformation and could also derive support from the blanket pellets inside; and c) the forces at wrapper wire contact points necessary to cause failure are higher than those which would be available. Thus, the area of interest would be whether the neighbor duct could deform plastically in such a manner as to cause flow restriction in the rod bundle. This consideration was addressed in Section 15.4.1.2.5 for the fuel assemblies. The conclusions of that section hold true for the radial blanket assemblies.

15.4.3.2.6 Consequences of Postulated Duct Crack

The thermal effects of a postulated duct crack arise from a loss of coolant flow through the accident assembly due to leakage through the crack. Since the interassembly gap sodium is at a low pressure relative to that internal to the radial blanket assembly, flow could be diverted from the ruptured radial blanket assembly. The crack width would be limited by the lateral support of the assemblies adjacent to the faces of the accident assembly crack. Since the radial blanket assembly duct would absorb energy before rupture would occur, little energy would be available to damage neighbor assemblies and no deformation to neighbor assemblies would occur. Hence, the postulated crack in the radial blanket assembly duct within which the MFCI was postulated to occur would be limited to a small width.

The axial extent of the postulated crack would be limited because of several factors:

a. The pressure decreases rapidly in the axial direction.

b. The energy available from the MFCI is small.

c. The initial opening of a crack would begin to relieve the pressure tending to limit propagation of the crack.

d. The duct temperature is lower and the strength is greater at the inlet end and mid core.

The effect of the postulated crack on coolant flow would be small because:

a. The maximum pressure drop across the radial blanket duct wall is low.

b. The close rod spacing and large rod diameter gives the radial blanket assembly a high transverse flow resistance.

It is concluded that a crack longer than a few inches is very unlikely, and therefore, a duct rupture would cause little flow change in the accident assembly. The effects of a large crack were assessed in Section 15.4.1.2.6 for a fuel assembly. By extrapolating those results to a 61-rod radial blanket assembly, the hot channel temperature rise is expected to not exceed a maximum hot channel temperature of 1450°F.

15.4.3.3 Flow Blockage of a Radial Blanket Assembly

Experimental Blockage Data Obtained in Water and Sodium Flow Systems

The only published rod bundle sodium data which show temperatures in the wake region are the results of seven tests reported by ORNL (Ref. 36, 37). A typical set of data, for Test 1, are shown in Figure 15.4.3.3.3-1. Table 15.4.3.3-1 summarizes data for all seven tests. One set of data obtained in a "simulated rod bundle" were partially reported by Kirsch and Schleisiek (Ref. 59).

Whether the wide variation of inside cladding temperatures in the wake region shown in Figure 15.4.3.3.3-1 is caused by a wide difference in heat transfer coefficients, local wake fluid temperature variation, or to other factors is not known. The Figure 15.4.3.3.3-1 test, as do the other six reported tests, shows only the measured inside cladding temperatures and the calculated outside cladding temperatures in the near wake region. (The near wake appears to have a length of approximately 1.75 inches and a L/d_B equal to three). The fluid temperature at the beginning and end of the wake region was assumed to be associated with the grounded junction thermocouple measurements shown. Unfortunately the fluid temperature adjacent to the heaters in the wake region can only be estimated by making an assumption concerning the magnitude of the heat transfer coefficients. In Table 15.4.3.3-1 and Figure 15.4.3.3.3-1 the cladding temperatures are referenced to T_L, the calculated temperature in channel 3 which was normally higher than the average fluid temperature outside the wake region.

Table 15.4.3.3-2 shows experimentally determined values of the dimensionless residence time $t_R = \tau U/d_B$ obtained by ORNL, Winterfied and Karlsruhe (we have calculated the values for the Karlsruhe test based on their data. All the hydraulic data shown were obtained in water systems).

Kirsch and Schleisiek obtained limited residence time data for both water and sodium; temperature measurements were used. They concluded, (1) that molecular heat conduction does not measurably influence energy transport in the recirculation zone (wake), (2) the only factor determining the temperature distribution is the turbulent recirculating flow, (3) experimental results for the temperature level and temperature distribution in the wake obtained from water measurement can be extrapolated to sodium.

Using the dimensionless residence time, $\tau U/d_B$, for the ORNL data from Table 15.4.3.3-1 calculations were made to determine $T_B - T_L$, the fluid temperature increase caused by the blockage. The equations used are:

$$T_B - T_L = 2 \frac{Q\ t}{A_{WF} \rho C p}$$

$$\frac{\tau U}{d_B} = F \text{ (geometry, } \beta)$$

It should be noted that the d_B values of the characteristic blockage dimension obtained in the triple scale model must be reduced by a factor of three when applied to the sodium data obtained in the full size 19 rod bundle.* Moreover the heat input should be that into the fluid within a unit channel. Our calculated results are shown in the column "as calculated $T_B - T_L$" in Table 15.4.3.3-1. The agreement is fair only if it is assumed that the variation in cladding temperatures in Figure 15.4.3.3.3-1 is caused by variations in heat transfer coefficients and not large variations in wake fluid temperatures. This assumption implies that the lower dotted line in Figure 15.4.3.3.3-1 represent approximately the true wake temperature which corresponds to entry $T_B - T_L$ estimated in the Table. Even when this assumption is made the as calculated temperatures $T_B - T_L$ must be multiplied by a factor of 1.5 to 2.0 to get agreement. Kirsch indicated that the as calculated values of $T_B - T_L$ should give high values of the average wake fluid temperature.

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*The need for a scale geometry factor should also be assessed in applying the t_R obtained in one geometry to that of another geometry even when the P/D is identical.

In Tests 6 and 7 the as calculated values overpredicts the "wake temperature" where blockage leakage was suspected, Test 6, and where known leakage was built into Test 7. In Test 5 it is not felt justifiable to make a conclusion because the * value was only estimated. It does appear however, that leakage significantly reduces the wake fluid temperature. Our conclusion is that the simple model used to predict wake fluid temperatures is not completely adequate. However, the approach used in the previous discussion is used tentatively for core, radial blanket and control assembly analysis. The only firm conclusion that can be made at this time is that a complete non-porous 6 channel central blockage or a 14 channel edge blockage with or without leakage does not cause boiling at 10 kW/ft over a range of velocities of 37 to 20 ft/sec for the center blockage, and 5 kW/ft and 26 ft/sec for the edge channels.

Kirsch and Schleisiek (Ref. 59) used their hydraulic data for $t_R = \tau U/d_B$ to calculate blockage conditions in the SNR reactor. The calculations showed that over 40% of the center part of a core assembly could be blocked without boiling occurring based on the average fluid temperature in the wake.

Other water data reported by Kirsch indicated that the maximum temperature in the wake exceeded the average temperature only by approximately a factor of 1.2. In these tests the maximum wake temperature was near the outer edge of the wake.

15.4.3.3.1 Prevention and Detection

Flow blockages of CRBRP radial blanket assemblies are extremely unlikely due to design features of the reactor, cleanliness requirements during construction of the plant, precautionary operations carried out during initial sodium fill and testing, and sodium coolant purity requirements during reactor operation. Postulated blockage mechanisms are: (a) large and (b) small non-degradable debris left behind during plant or radial blanket assembly construction, (c) degradable material left behind during construction, (d) corrosion products, (e) sodium-lubricant reaction products, and (f) failed fuel debris. These failure mechanisms are described in detail in Section 15.4.1.3.1. All of the prevention and detection mechanisms described in Section 15.4.1.3.1 apply to the radial blanket except (f), failed debris. The radial blanket fuel rod failures will be detected by the cover gas monitoring system, and the defective assembly will be located by the tag
35 gas. However, radial blanket fuel-sodium exposure will not be detected by the Delayed Neutron Detectors due to the very low amount of fissile material in the radial blanket fuel, even near end of life. A properly position thermocouple at the outlet of each radial blanket may enhance the detection of excessive coolant outlet temperatures that could arise from flow blockages.

**The need for a scale geometry factor should also be assessed in applying the t_R obtained in one geometry to that of another geometry even when the P/D is identical.

15.4.3.3.2 Consequences of Inlet Blockage of Radial Blanket Assembly

The result of partial rod bundle inlet blockage of a radial blanket assembly are similar to those discussed in Section 15.4.1.3.2 for the fuel assemblies. A large blockage is required to effect a small reduction in flow rate and a significant increase in outlet temperature. However, a major part of the pressure drop through a radial blanket assembly is taken in the orifice plates. These plates have multiple perforations to assure redundancy of flow path. Should a majority of the holes in one plate be blocked, the remaining openings in that plate will control the flow rate and the effect of the other plates will be reduced. Here again, a large blockage fraction is required for a significant reduction in assembly flow. The occurrence of such a simultaneous blockage of several holes is also quite unlikely because of the upstream strainer in the inlet passage.

15.4.3.3.3 In Core Radial Blanket Assembly Passive Local Blockage

In addition to the lower power level and lower axial coolant velocity, the pitch-to-diameter ratio of the rods in a blanket assembly is lower (1.08) than that in a core assembly (1.23). Although the general blockage and wake theory discussed in Section 15.4.1.3 is expected to be valid for a radial blanket assembly, experimental data or theory are not available for predicting the magnitude of t_R, the dimensionless residence time, for fluid in the near wake for low pitch-to-diameter ratios.

Existing data have been reduced and tabulated in Table 15.4.3.3-2.

As shown in Figure 15.4.3.3.3-2, the slope of t_R versus β (the blockage ratio), appears to be constant for the three geometries tested. Although the SNR has a larger pitch-to-diameter ratio than the ORNL test bundle, the parameter t_R is somewhat larger. Thus, in addition to P/D, other factors are involved in determining t_R. (The possible effect of wire wrap versus grid (SNR) cannot be discounted).

The SNR test bundle had approximately 11 rods per square inch of cross section whereas the ORNL test bundle had 1.42 rods per square inch of cross section. In the SNR test bundle, there is a more tortuous path for fluid circulating in the wake. As expected, t_R for the flat plate has the lowest value of t_R because there is no obstruction to circulating fluid in the wake region. But the data for the ORNL 1.23 P/D bundle are very close to those for the Winterfield (Ref. 41) disc test data. Using this inference, the assumption was made

$$t_d = \frac{\tau U}{d_B} = P\left(\frac{P}{D}, \frac{NR}{A_T}, \beta\right)$$

where:

P/D, rod pitch-to-diameter ratio

$\frac{NR}{A_T}$, number of rods per unit cross section area

β, blockage ratio

The final tentative relation obtained is

$$t_R = C_1 \left(\frac{NR}{A_T}\right)^{0.5} \left(\frac{D}{p}\right) \beta^{-0.88}$$

Although the design relation for t_R, which must be substantiated by experiment, does not go to infinity as P/D → 1.0, such a limit may not be required in the P/D range examined. Even though the radial blanket assembly has a P/D = 1.08 number of rods per square inch is relatively small and because of the large size rod, the absolute space between rods is 0.040" (compared to 0.056" in the core).

Although a value equal to t_R = 17 was calculated for a radial blanket assembly at a blockage fraction β = 0.1, the larger value t_R = 22.5 calculated for the core was used. Another conservatism employed is that the blockage factor was not used because of overall uncertainty. For a six-channel blockage, the following values of cladding temperature and wake temperatures were obtained.

Radial Blanket Assembly

Six-Channel Blockage

t_R = 22.5

	Midplane	Exit Plane
Wake Temperature (°F)	1127	1121
Coolant Temperature (°F)	858	1000
Linear Power (kW/ft)	7	3.2
Inside Cladding Temperature (°F)	1203	1156

15.4.3.3.4 In-Core Active Blockage in a Radial Blanket Assembly

The basic analysis for the core assemblies, Section 15.4.1.3.4 applies to radial blanket assemblies; however, the effects of a different flow velocity, heat flux, rod pitch-to-diameter ratio and rod diameter were assessed.

15.4.3.3.5 Postulated Local Boiling Effects; Radial Blanket Assembly

As discussed in Section 15.4.1.3.5, Postulated Local Boiling Effects, Core Assembly, the method used to calculate the size of a blockage which might cause local boiling in the wake behind a blockage is similar to that used to calculate the effects of a six-channel blockage. However, the axial difference in linear power rating and the axial difference in the sodium boiling temperature causes a difference in the size of blockage required to produce local boiling at the midplane and exit plane. The following table shows the calculated sizes of central blockage, fraction area blocked and the boiling temperatures assumed at the mid-plane and exit plane of a radial blanket assembly.

Boiling Temp.	d_B	β Approximately	t_R
°F	inches	---	--
Midplane			
1800	3.64	0.59	22.5
1750	3.44	0.53	22.5
1700	3.25	0.47	22.5
Exit plane			
1750	6.36*	--	22.5
1750	5.93*	--	22.5
1650	5.51*	--	22.5

*The only significance which should be associated with these numbers which are greater than the assembly diameter is that in order for boiling to occur, the size of the blockage could be detected.

It should be noted that in the model used to calculate the size blockage required to produce local boiling, the blockage diameter d_B is inversely proportional to heat flux and directly proportional to the average coolant velocity. In the blanket region directly adjacent to the core, the blanket rod linear heat rating in the row of rods facing the core might have a significantly higher linear power rating than 7 kW/ft used in this analysis. The higher power level would not be expected to influence significantly a central blockage but might affect adversely the size of blockage required to cause boiling for a true two-dimensional edge or corner blockage. However, if the blockage were only a semi-edge blockage the colder fluid around the inside periphery of the assembly would be expected to ameliorate the consequences of the blockage.

15.4.3.3.6 Long Term Effects of a Hot Spot

In the preceding sections the effects of inlet flow blockages and non-heat generating and heat generating in-core blockages have been discussed. One effect of these blockages is to increase the local cladding temperature in the vicinity of the blockage. If these blockages go undetected over a long period of time, the capability of the radial blanket rods to reach their design lifetimes would be reduced. Figure 15.4.3.3.6-1 which shows the effect of a temperature increase on the lifetime of the statistical hot radial blanket rod at core outlet, is based upon the time to reach 0.1% ductility limited strain during steady state operation. The consequences of a premature failure would be no worse than the stochastic radial blanket rod failure discussed in Section 15.4.3.1.

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TABLE 15.4.1.1.5-1

SUMMARY OF FABRICATION DETAILS, OPERATING

CONDITIONS AND CONDITION OF BREACHED FUEL PINS

l assembly identification*		PNL-5A(X114)	PNL-10(X193)	HEDL-N-E(X191)	P-12AA(X186)	PNL-5B(X16B)	P-12AB(X213)	PNL-11(X194)	(X106)°
led pin identification		PNL 5-1	PNL-10-14	HEDL-N-E-122	P-12A-63K	PNL 5-17	P-12A-11B	PNL-11-39	018
dding material, type stainless steel**		304SA	316,20%cw	316,20%cw	316,30%cw	304SA	316,20%cw	316,20%cw	304L annealed
diameter	ins.	0.250	0.230	0.230	0.230	0.230	0.230	0.230	0.290
l thickness	ins.	0.016	0.015	0.015	0.015	0.016	0.015	0.015	0.020
l composition w/oUO_2/w/oPuO_2		75/25	75/25	75/25	75/25	75/25	75/25	75/25	80/20
nium Enrichment	%U235	93	65	77	93	93	93	65	92.7
l Column length	ins.	13.5	13.5	13.5	13.5	13.5	13.5	13.5	10.91
ulator Composition+		Dep UO_2	Nat UO_2	Dep UO_2	Nat UO_2	Dep UO_2	Nat UO_2	Nat UO_2	Dep UO_2
ulator length, top	ins.	0.5	6.63	6.689	0.5	0.5	0.5	6.63	0.47
ulator length, bottom	ins.	0.5	14.00	0.503	0.513	0.5	0.513	14.00	0.47
imum Cladding temperature	°F	920	1050	1142	1200	920	1270	1150	953
imum pin power	kw/ft	12	10	12	11	12	11	14	14.3
dding fluence(E>0.1Mev)	n/cm^2	$8.6X10^{22}$	$5.0X.C^{22}$	$3.2X10^{22}$	$2.3X10^{22}$	$9.6X10^{22}$	$5.2X10^{22}$	$1X10^{23}$	
nup,	Mwd/MTM	128,000	64,000	42,300	$35,000^{++}$	$145,000^{++}$	$79,000^{++}$	$111,000^{++}$	
	a/o	13.1	6.6	4.4	3.6 ++	14.9 ++	8.1 ++	11.4 ++	7.8
volume at STP	c.c.	208	71	42	45	225	103	140	104.9
pressure at failure	psi	1170	570	455	570	1441	870	1100	1160
ation of failure above bottom of fuel column	in.	10.85	13	13.31	12.5	NK$^{\Delta}$	NK	NK	~11.7$^{\Delta\Delta}$

First identifier is the HEDL assembly number, if applicable; the second is the ANL EBR II number

SA-Solution annealed; 20%cw - 20% cold work

Dep UO_2≡Depleted UO_2; Nat UO_2≡Natural UO_2

Calculated

Not known at this time

Above top insulator pellet

Based on ANL-8063

TABLE 15.4.1.2.6-1

Flow Rate Comparison for Assembly 29

Flow Path	Flow Rate, ft^3/sec Normal Duct	Flow Rate, ft^3/sec Duct with Crack
Assembly Inlet to Fuel Pin	.634	.707
Fuel Pin to Assembly Outlet	.634	.133
Flow from Duct Crack	0.	.674

20

TABLE 15.4.1.2.6-2

Effect of Cracked Duct
on Assembly Hot Channel Outlet Temperature
For Assembly 29

	Normal	Cracked
$\int \frac{\phi(\ell)}{F(\ell)} d\ell$	143.2	250.9
Ratio	1.0	1.752
ΔT, °F	558	978
T_{out}, °F	1288	1708

20

TABLE 15.4.1.3.3-1

TEMPERATURES BEHIND A CENTRAL SIX-CHANNEL BLOCKAGE

	Midplane	Exit-Plane
Average Wake Temperature (°F)	1104	1197
Maximum Wake Temperature (°F)	1261	1244
Midwall Cladding Temp. (°F)*	1396	1312
Linear Power (kW/ft.)	11	5
Velocity (ft/sec)	20	20
Dimensionless Residence Time (t_R)**	22.5	22.5

*Based on maximum fluid temperature, a conservative approach.

**$t_R = \frac{\tau U}{d_B}$ where: τ (Tau) is the average residence time of the fluid in the wake region; U is the free stream velocity; and d_B is the characteristic blockage dimension.

TABLE 15.4.1.3.5-1

BLOCKAGE SIZE NECESSARY TO CAUSE LOCAL BOILING

Boiling Temperature	Blockage Diameter*	Fraction of Flow Area Blocked (approx.)	Dimensionless Residence Time (t_R)**
°F	Inches	--	--
Midplane			
1850	2.35	0.25	22.5
1800	2.23	0.22	22.5
1750	2.11	0.20	22.5
Exit-Plane			
1770	3.55	0.56	22.5
1720	3.29	0.48	22.5
1700	3.02	0.40	22.5

*For non-porous blockages.

**$t_r = \frac{\tau U}{d_B}$

where:

τ is the average residence time of the fluid in the near wake region

d_B is the characteristic blockage dimension (e.g., blockage diameter).

U is the free stream velocity.

TABLE 15.4.2.2-1

PEAK PIN THERMAL AND GAS RELEASE COMPARISONS
BETWEEN NORMAL* AND OVERPOWER** CONTROL ROD ASSEMBLIES
IN THE CENTRAL (ROW 1) C/A LOCATION

Lifetime Cases	Gas Release Pressure, psi		Hot Channel Coolant Outlet Temperature		Max. Cladding Midwall Temperature, °F		Max. B_4C Centerline Temp., °F	
(inches withdrawn)	Normal*	Overpower**	Normal*	Overpower**	Normal*	Overpower**	Normal*	Overpower**
BOFC+ (12")	-	-	1076	1162	1090	1184	1886	2296
EOFC+ (36")	818	1185	840	859	851	880	1538	1820
BOECo (11")	-	-	1085	1174	1101	1198	1891	2303
EOECo (36")	965	1614	839	858	847	877	1425	1659

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*Normal = Natural @ 20% B-10 enrichment
**Overpower = Enrichment @ 55% B-10
+BOFC & EOFC = Beginning and end of first cycle
oBOEF & EOEC = Beginning and end of equilibrium cycle

TABLE 15.4.2.2-2

CENTRAL CONTROL ASSEMBLY -
HEAT GENERATION BASES+ FOR TABLE 15.4.2.2-1

Lifetime Cases (inches rod withdrawal)	Assembly Total Power, kw	
	Normal*	Overpower**
BOFC (12")	425	535
EOFC (36")	85	109
BOEC (11")	439	552
EOEC (36")	84	108

* Normal = natural @ 20% B-10 enrichment

** Overpower = Over-enrichment @ 55% B-10

TABLE 15.4.2.2-3

ABNORMAL PEAK CONTROL ROD PIN THERMAL CONDITIONS
(OPERATING UNDER FULL INSERTION AT BOEC)

Rod Type	Hot Channel Coolant Outlet Temp., °F	Maximum Cladding Midwall Temp. °F	Maximum B_4C Centerline Temp., °F
Central	1129	1191	1946
Row 7 flat	1130	1206	2187
Row 7 corner	1140	1206	2037

TABLE 15.4.3.2-1

RADIAL BLANKET PEAK OPERATING CONDITIONS FOR FAILURE TO SHUFFLE (2σ UNCERTAINTIES)

Time of Life	Peak Linear Power (kw/ft)	Peak Coolant Temperature (°F)	Peak Cladding Midwall Temp. (°F)	Molten Cross Section (%)	Gas Pressure (Psi)
SOC7	6.3	927	949	0	186
EOC7	11.6	1068	1115	0	197
SOC8	12.6	1099	1150	0	200
EOC8*	16.1	1180	1249	0	241
SOC9	17.5	1222	1296	0	254
EOC9	19.8	1275	1360	10	344
SOC10	21.6	1325	1419	20	367
EOC10	23.3	1354	1453	27	491
SOC11	25.4	1411	1521	35	522
EOC11	26.4	1420	1532	38	672
SOC12**	28.7	1484	1607	45	713
EOC12**	29.1	1476	1597	46	879

*Shuffling of the radial blanket assembly in the peak flux position normally occurs every two years.

**The percent increase in bred fissile material will be slightly less than the reduction in flux level from control rod withdrawal in the core by cycle 12. Thus both the assembly and peak rod total powers are lower at EOC12 than at SOC12 and consequently the peak coolant and cladding temperatures are lower at EOC12 than at SOC12. At the same time the axial peaking factor increases more than the rod power decreases so the peak linear power is higher at EOC12 than at SOC12.

TABLE 15.4.3.2-2

RADIAL BLANKET PEAK OPERATING CONDITIONS FOR FAILURE TO SHUFFLE (3σ UNCERTAINTIES + 15% OVERPOWER)

Time of Life	Peak Linear Power (kw/ft)	Peak Coolant Temperature (°F)	Peak Cladding Midwall Temp. (°F)	Molten Cross Section (%)	Gas Pressure (Psi)
SOC7	7.3	932	961	0	189
EOC7	13.4	1077	1142	0	202
SOC8	14.5	1110	1175	0	207
EOC8*	18.6	1194	1281	0	258
SOC9	20.2	1236	1333	14	272
EOC9	22.8	1291	1402	27	372
SOC10	24.9	1342	1463	35	396
EOC10	26.9	1372	1502	41	530
SOC11	29.3	1432	1573	48	564
EOC11	30.4	1440	1586	50	725
SOC12**	33.1	1506	1666	56	769
EOC12**	33.5	1497	1657	57	945

*Shuffling of the radial blanket assembly in the peak flux position normally occurs every two years.

**The percent increase in bred fissile material will be slightly less than the reduction in flux level from control rod withdrawal in the core by cycle 12. Thus both the assembly and peak rod total powers are lower at EOC12 than at SOC12 and consequently the peak coolant and cladding temperatures are lower at EOC12 than at SOC12. At the same time the axial peaking factor increases more than the rod power decreases so the peak linear power is higher at EOC12 than at SOC12.

TABLE 15.4.3.3-1

COMPARISON OF EXPERIMENTAL AND CALCULATED EFFECT OF
BLOCKAGE ON LOCAL WAKE FLUID TEMPERATURE

ORNL (Ref. 37) - 19 Rods - 6 Center Channels Blocked (0.250" Thick Plate; P/D = 1.23) Sodium

Estimated ΔT Film	Power	Velocity	Experimental(1) (Maximum) ($T_{BC} - T_L$)	Estimated(1) ($T_B - T_L$)	As Calculated ($T_B - T_L$)	Test No.
	kw/ft	ft/sec	°F	°F	°F	
30-115	10	34	80-180	50	36	1
35-140	10	27.2	100-207	75	45	2
35-120	10	20.4	140-220	105	59	3
20-80	7.5	34	60-120	40	27	4
ORNL (Ref. 33) - 19 Rods - 14 Edge Channels Blocked (0.125" Thick Plate, P/D = 1.23) Sodium						
No Leakage	5	25.8	121*	--	97	5
Leakage ?	5	25.8	69	--	97	6
Leakage (0.014")	5	25.8	76	--	97	7

*This temperature estimated from early Test No. 5 data
$T_B - T_L$ = Wake Fluid Temperature - Adjacent Unblocked Fluid Temp.

(1) These temperatures are based on peak measured cladding temperatures.

TABLE 15.4.3.3-2

DIMENSIONLESS RESIDENCE TIME ($\frac{\tau U}{d_B}$) FOR BLOCKAGE TEST DATA

	Number of Rods	Blockage Fraction (β)	Number Blocked Channels	Bundle Scale	d_B inches	$\frac{\tau U}{d_B}$	$\frac{P}{D}$
Winterfield	-	0.13	-	-	1.97	9	∞
ORNL - Center	19	0.13	6	Triple	1.49	12	1.23
ORNL - Center	19	0.62	24	Triple	3.13	8	1.23
ORNL - Edge	19	0.13	5	Triple	1.22	28	1.23
ORNL - Edge	19	0.60	24	Triple	3.21	19	1.23
Karlsruhe (Center)	169	0.147	54	Full Size	1.71	18**	1.317
Karlsruhe (Center)	169	0.411	150	Full Size	2.88	13**	1.317

*Not a function of Reynolds Number

**Calculated from the Karlsruhe data

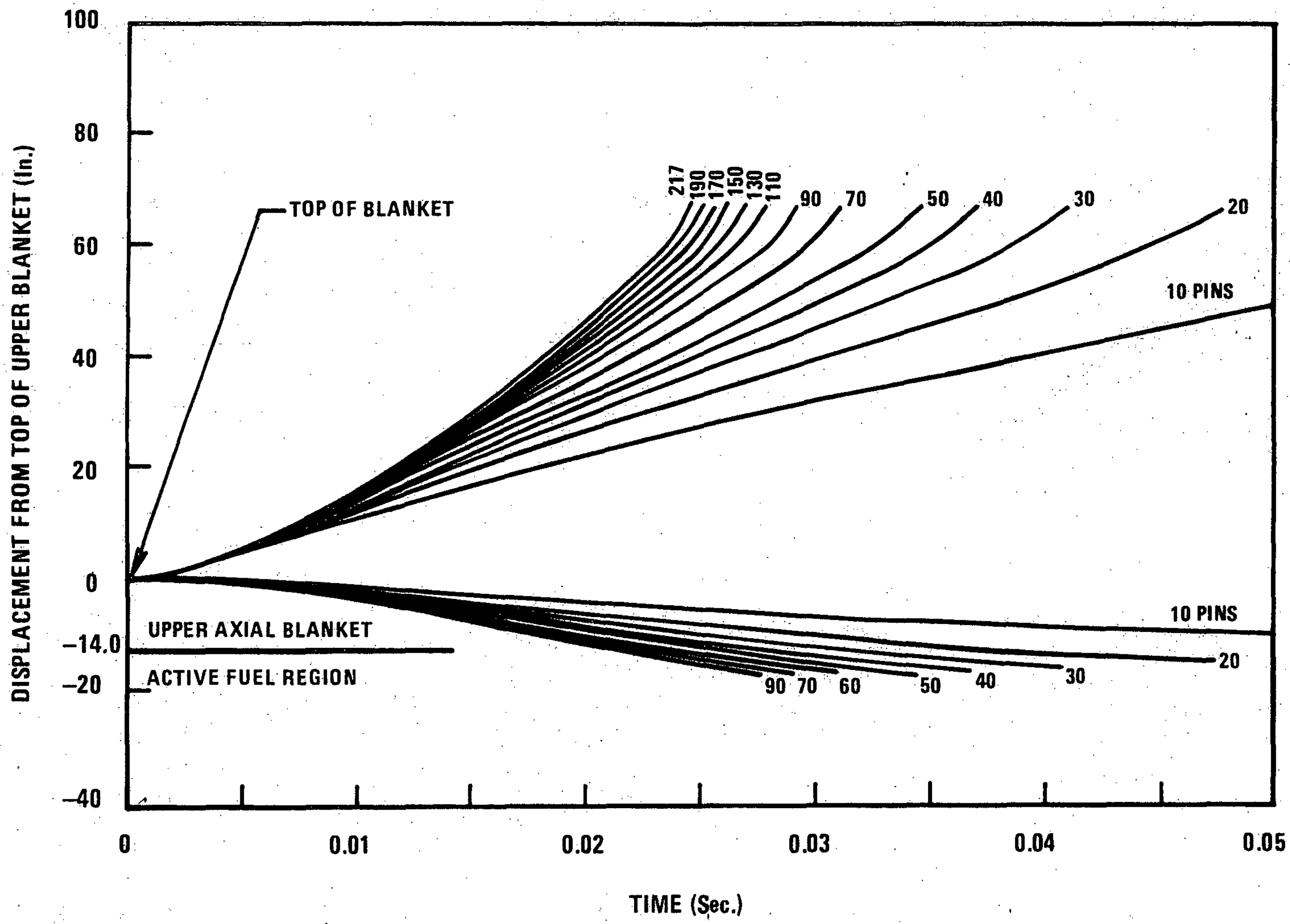


Figure 15.4.1.1-1. Displacements (Upper and Lower Gas-Liquid Interfaces - Inches)– As a Function of Time After Rapid Fission Gas Release for Various Numbers of Pins

6729-7

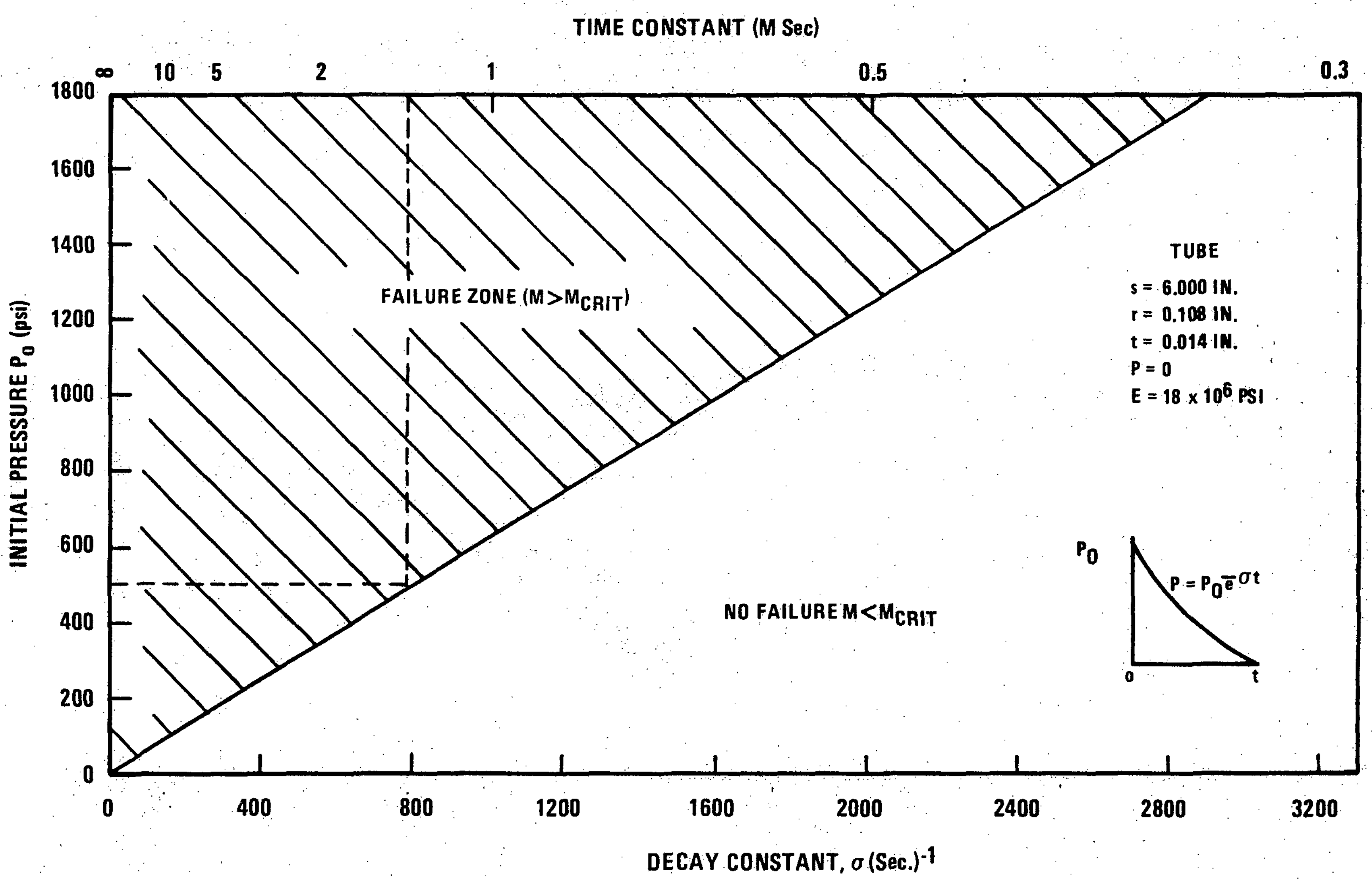


Figure 15.4.1.1-2. Combinations of Initial Pressure and Decay Constant Giving Maximum Moment Less Than Critical Bending Moment

6729-8

6729-9

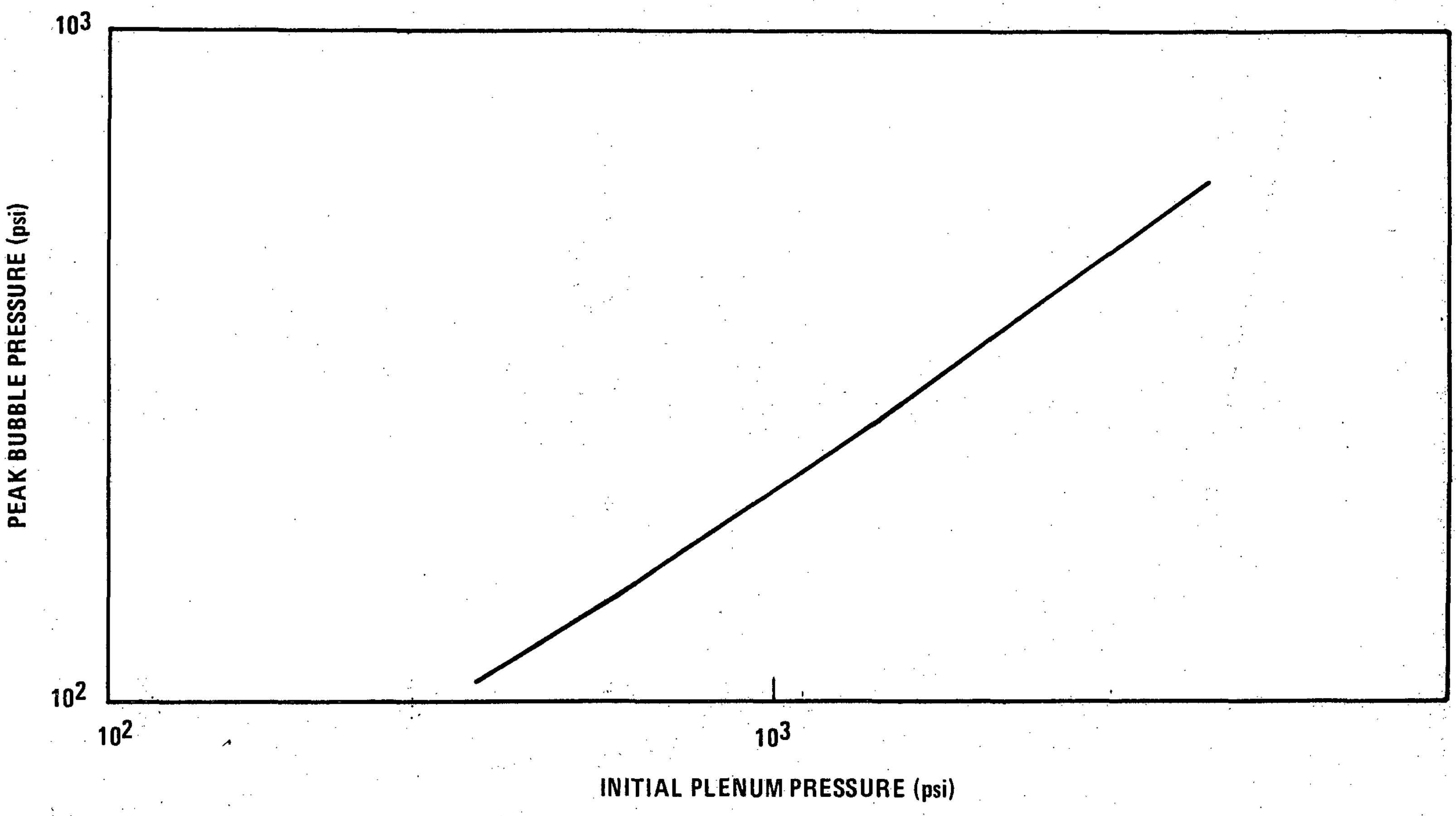


Figure 15.4.1.1-3. Effect of Initial Plenum Pressure on Peak Bubble Pressure

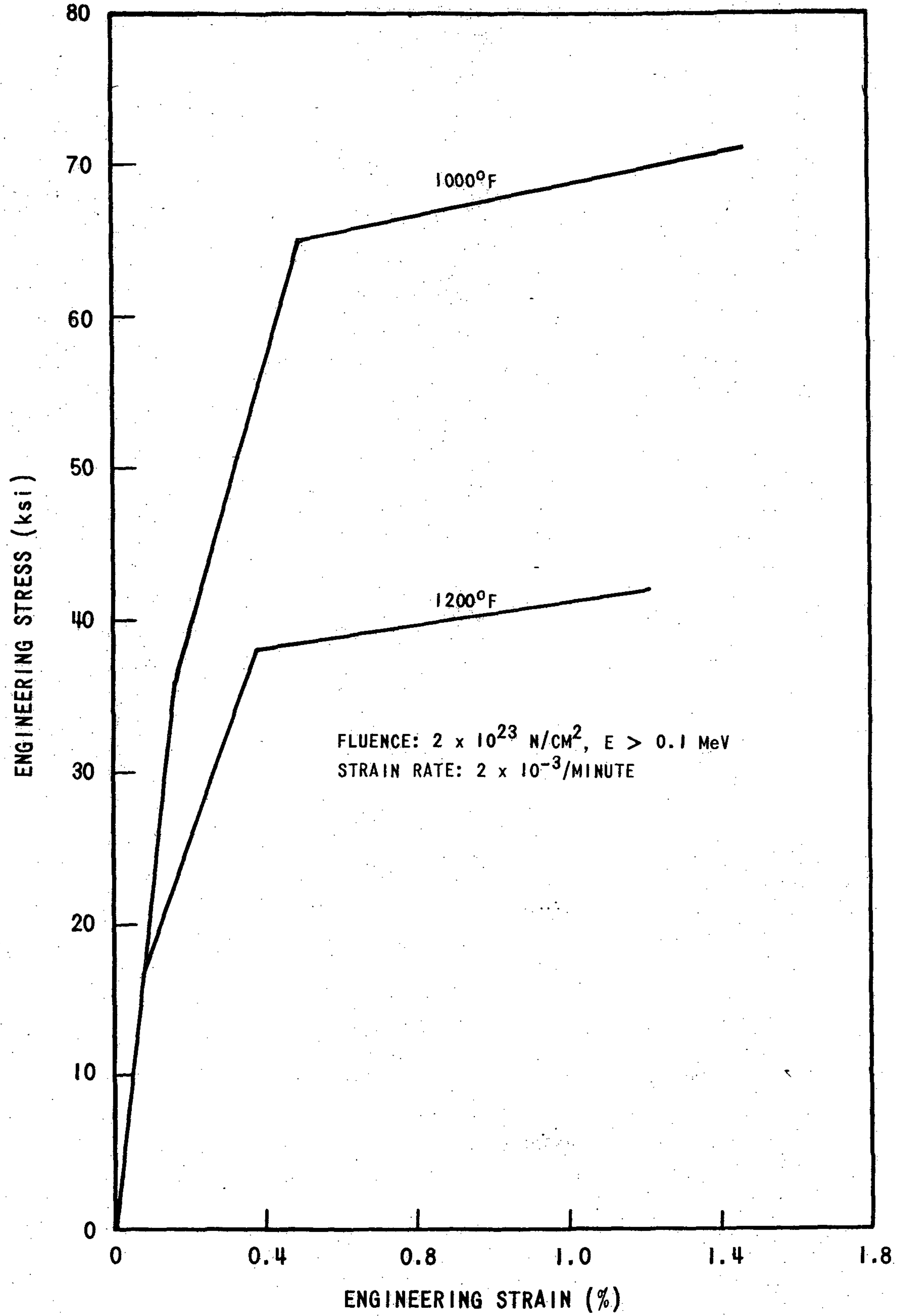
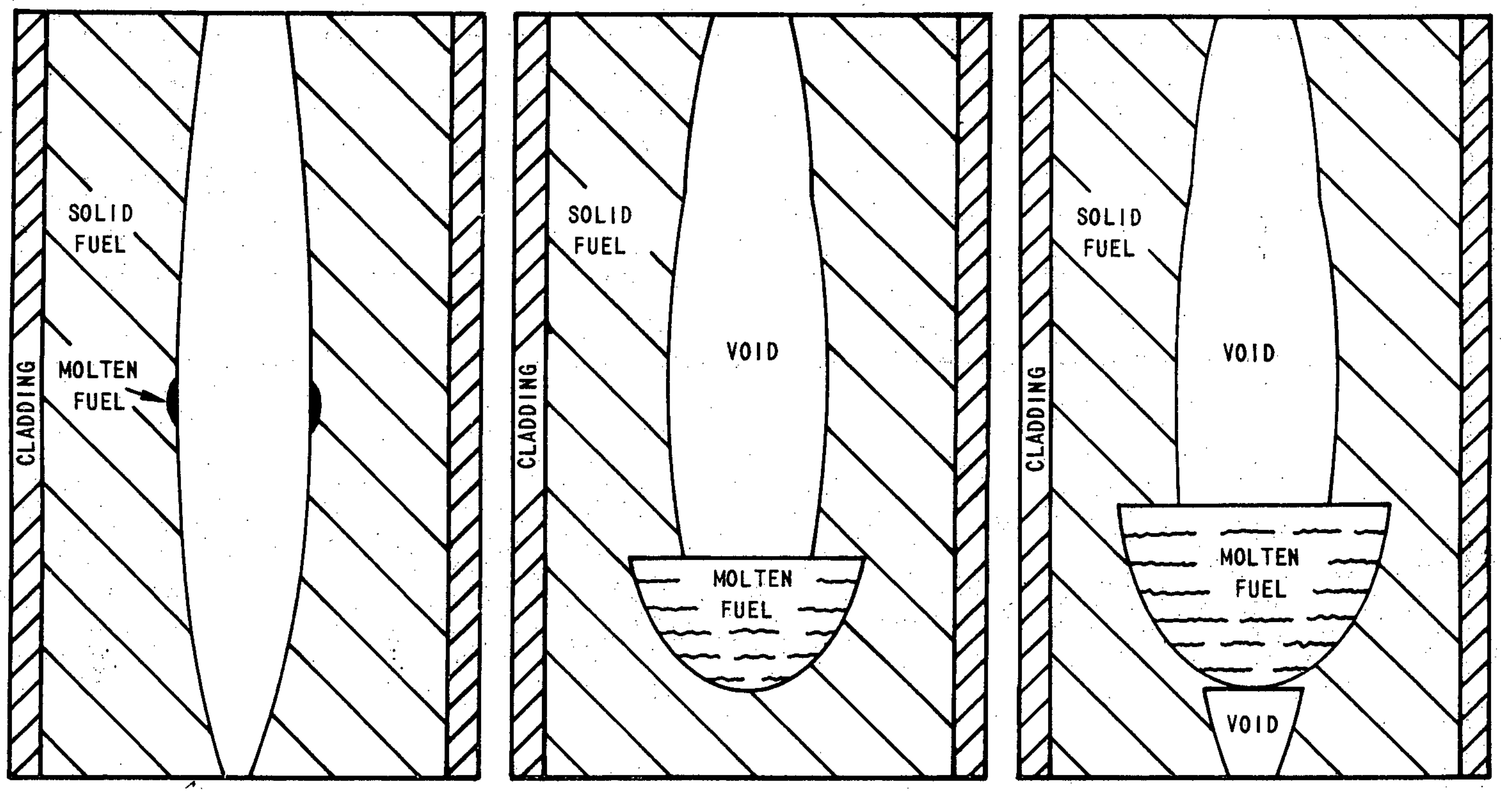


Figure 15.4.1.1-4. Engineering Stress Strain Curves for Irradiated, 20% Cold-Worked 316 Stainless Steel at Goal Fluence (Derived from Samples at Lower Fluences).

6729-6



MOLTEN FUEL BEFORE RELOCATION
(Slightly Beyond The Fuel
Incipient Melting Point)

MOLTEN FUEL RELOCATION
WITHOUT FUEL BRIDGE
(Stable Molten Fuel)

MOLTEN FUEL RELOCATION
WITH FUEL BRIDGE
(Stable Molten Fuel)

Figure 15.4.1.2.2-1 Fuel Bridge Model in Analysis of Overpower Pin

6729-1

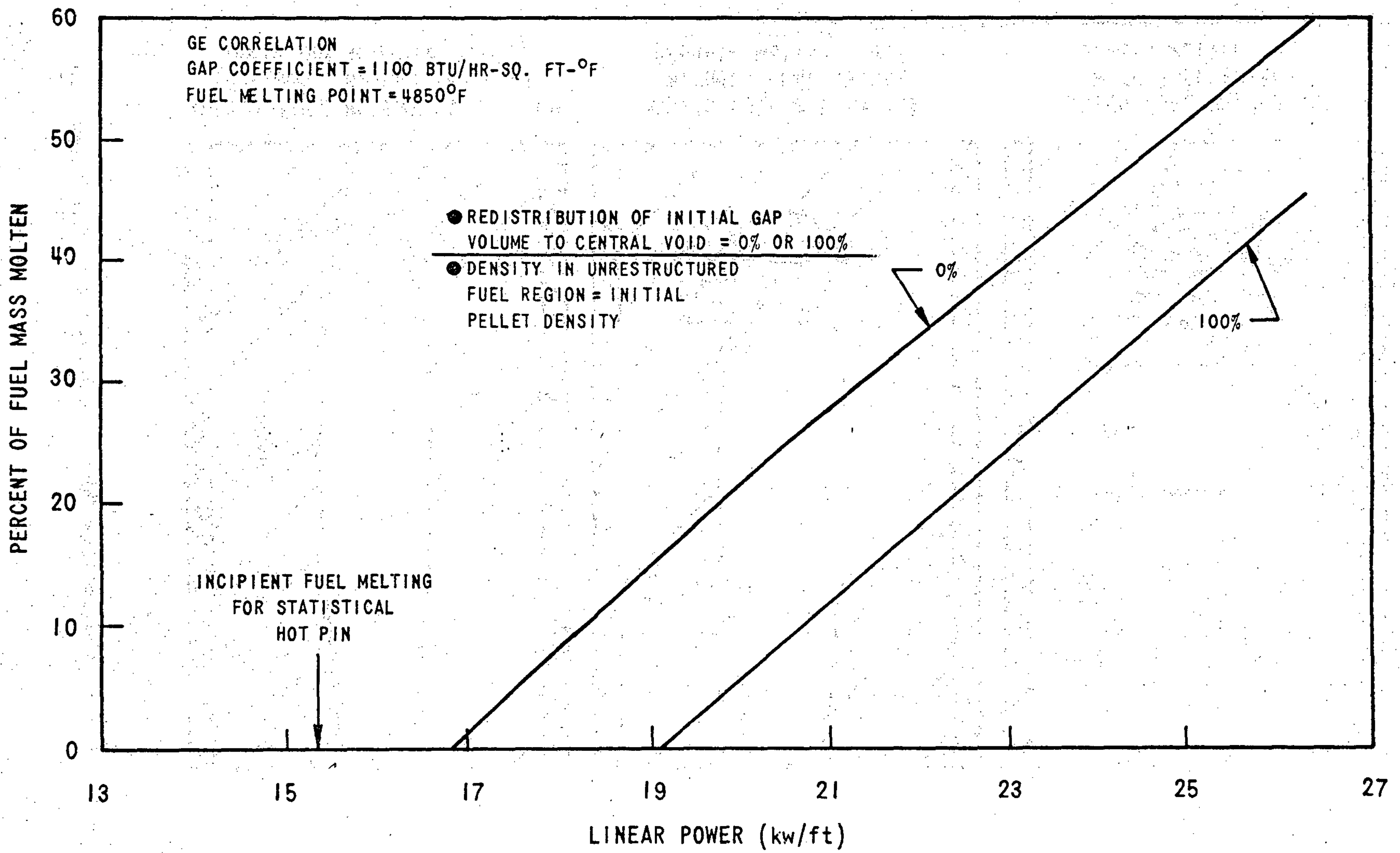


Figure 15.4.1.2.2-2. Effect of Overpower on Mass of Molten Fuel

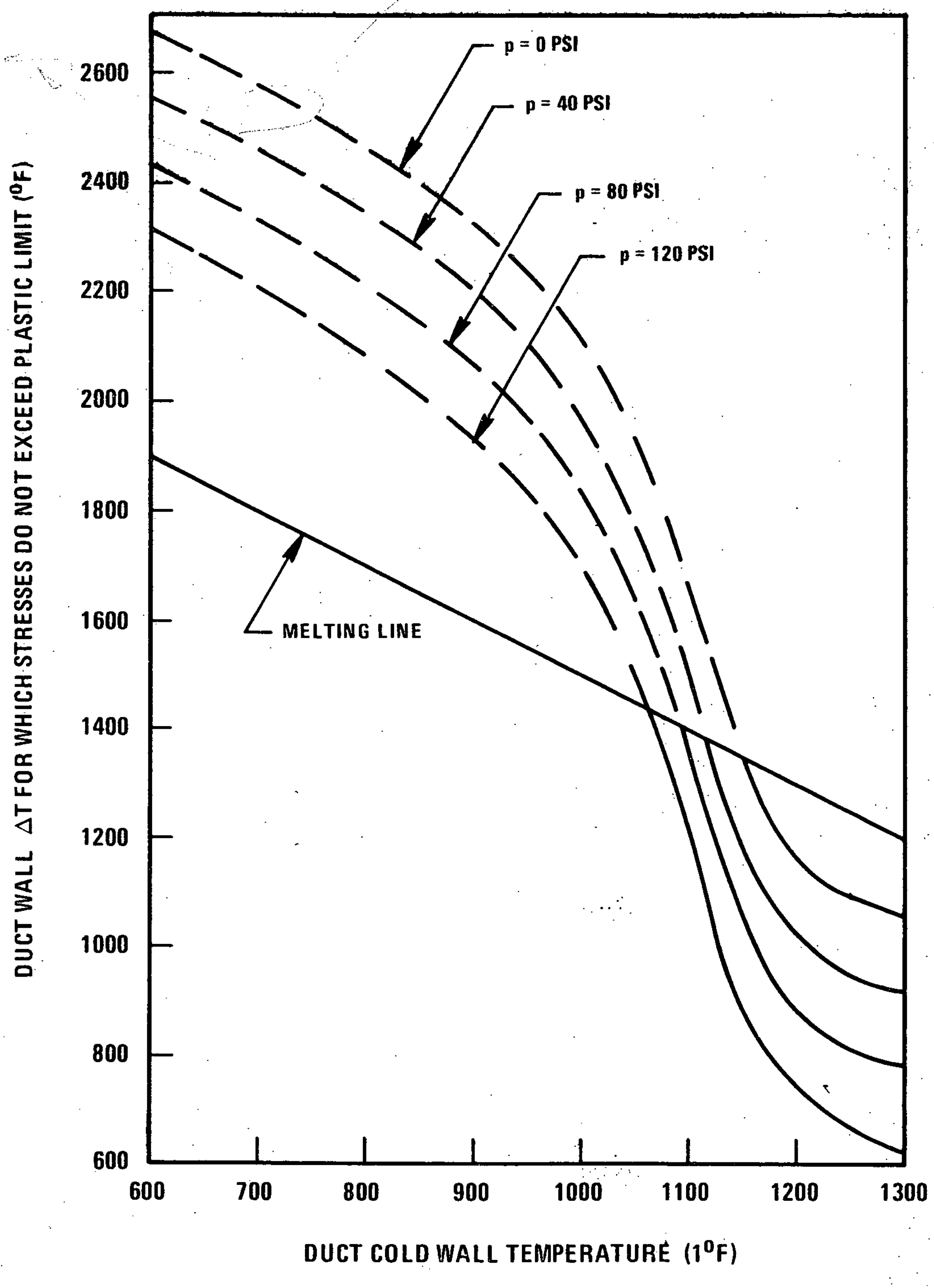


Figure 15.4.1.2.3-1 Temperature-gradient Limits For Plastic Behavior Of A Type 316 Stainless Steel Duct Wall Of Irradiated Subassembly For $\zeta = 0.90$. (Ref. Q001.334-1)

7683-111

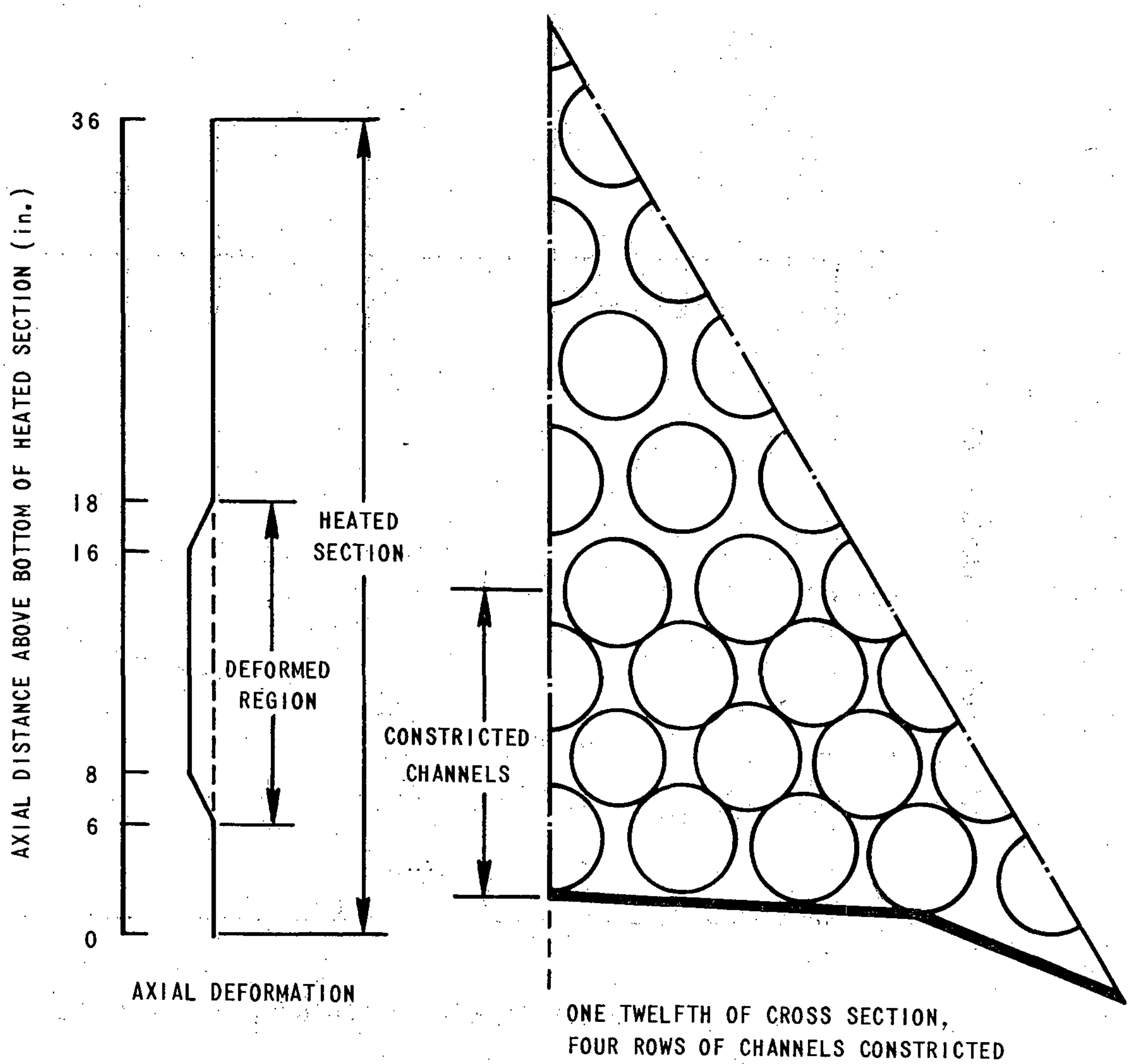


Figure 15.4.1.2.5-1. Axial and Cross-Sectional Deformations Used for Deformed Fuel Assembly Calculations

6729-5

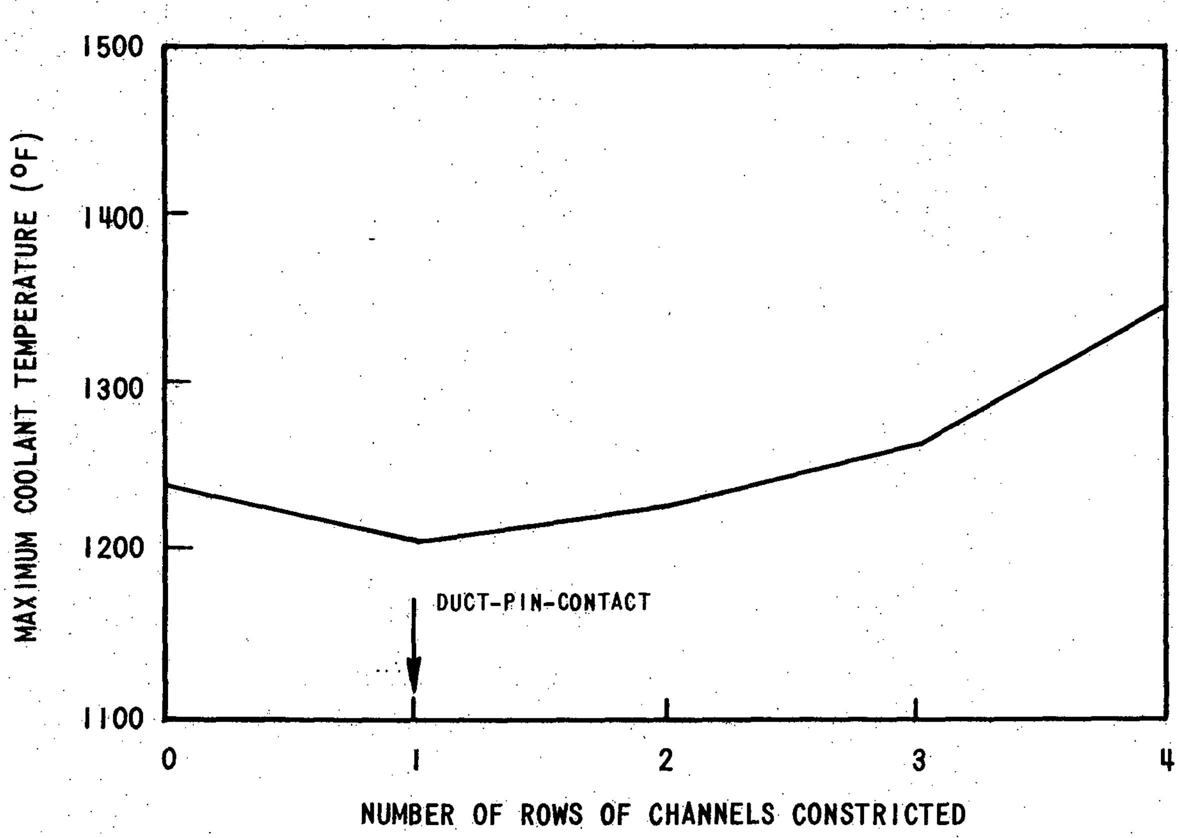
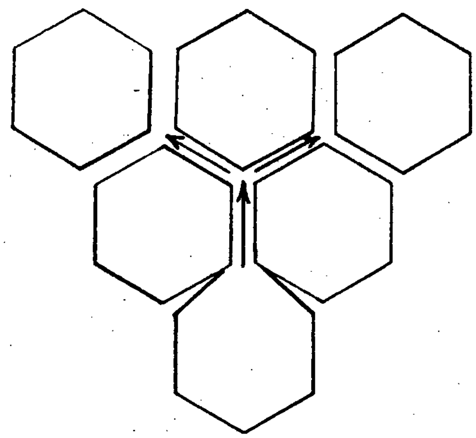


Figure 15.4.1.2.5-2. Maximum Coolant Temperature in Fuel Assembly Versus Number of Constricted Outer Coolant Channels

6729-4



20

Figure 15.4.1.2.6-1

Flow Path From A Cracked Duct

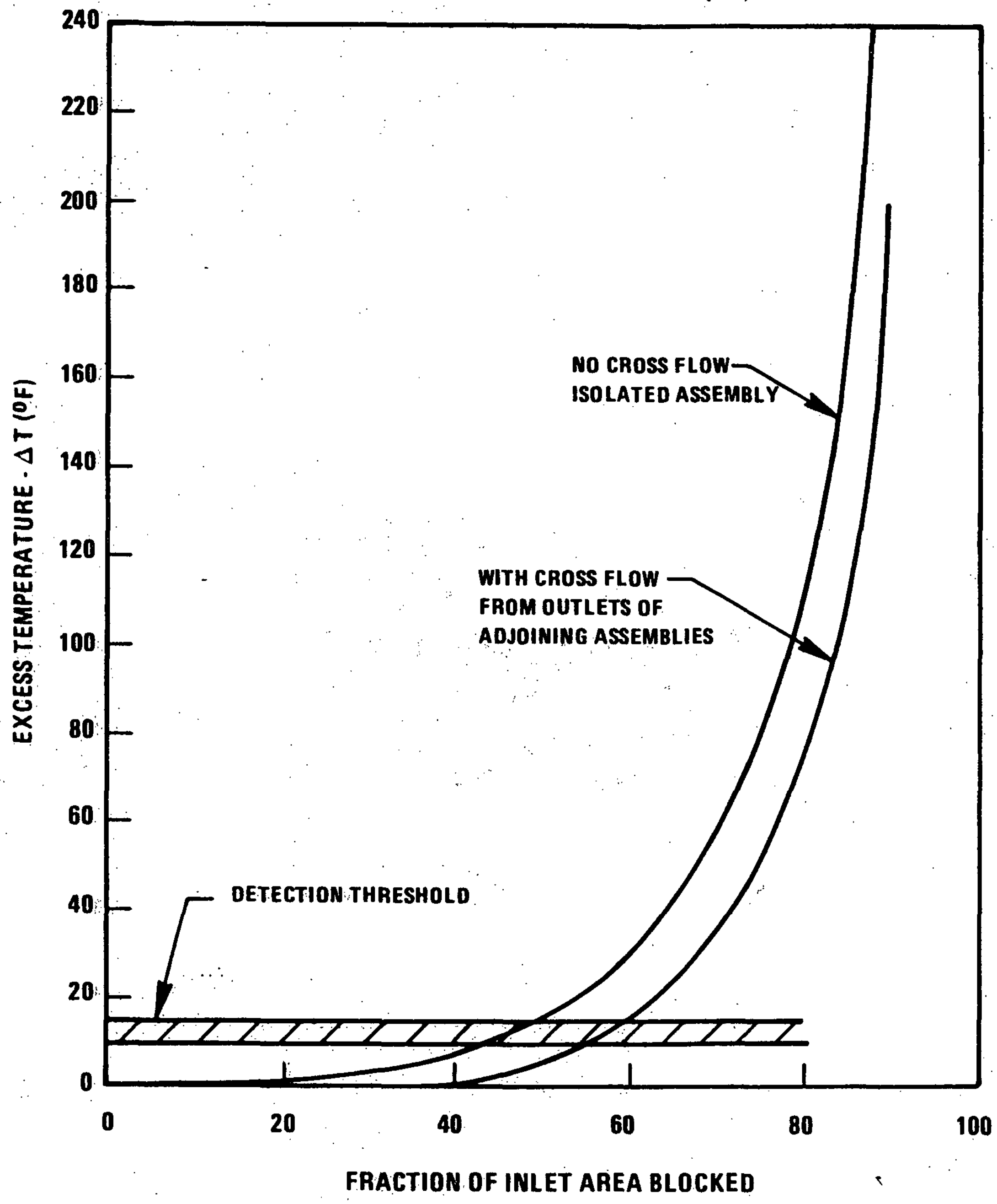


Figure 15.4.1.3-1. Effect of Blockage in Core Assembly Inlet on Mixed Mean Temperature Measurement in Outlet Passage

6729-10

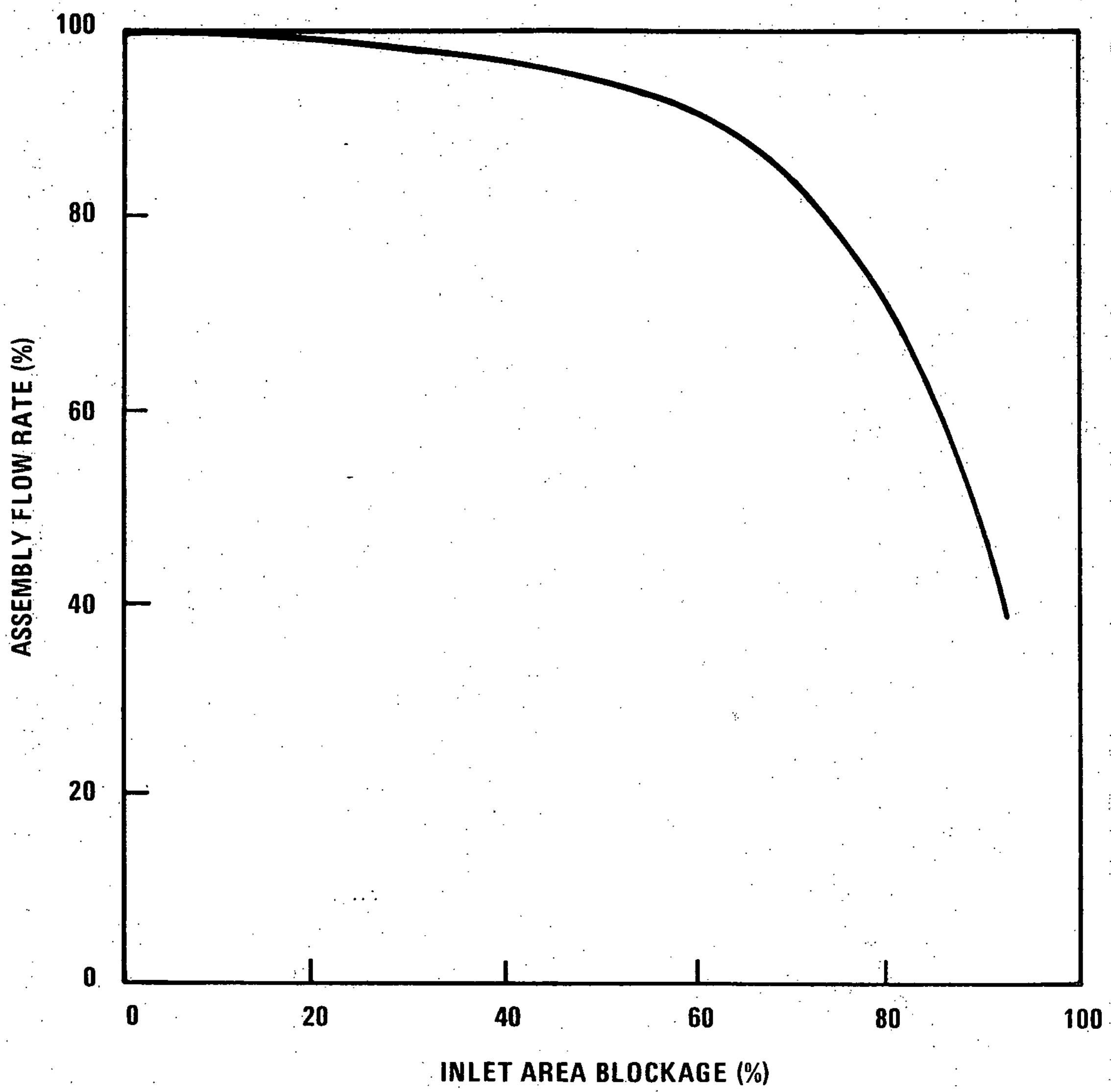


Figure 15.4.1.3-2. Prediction of Assembly Flow Reduction by Rod Bundle Inlet Blockage

6729-11

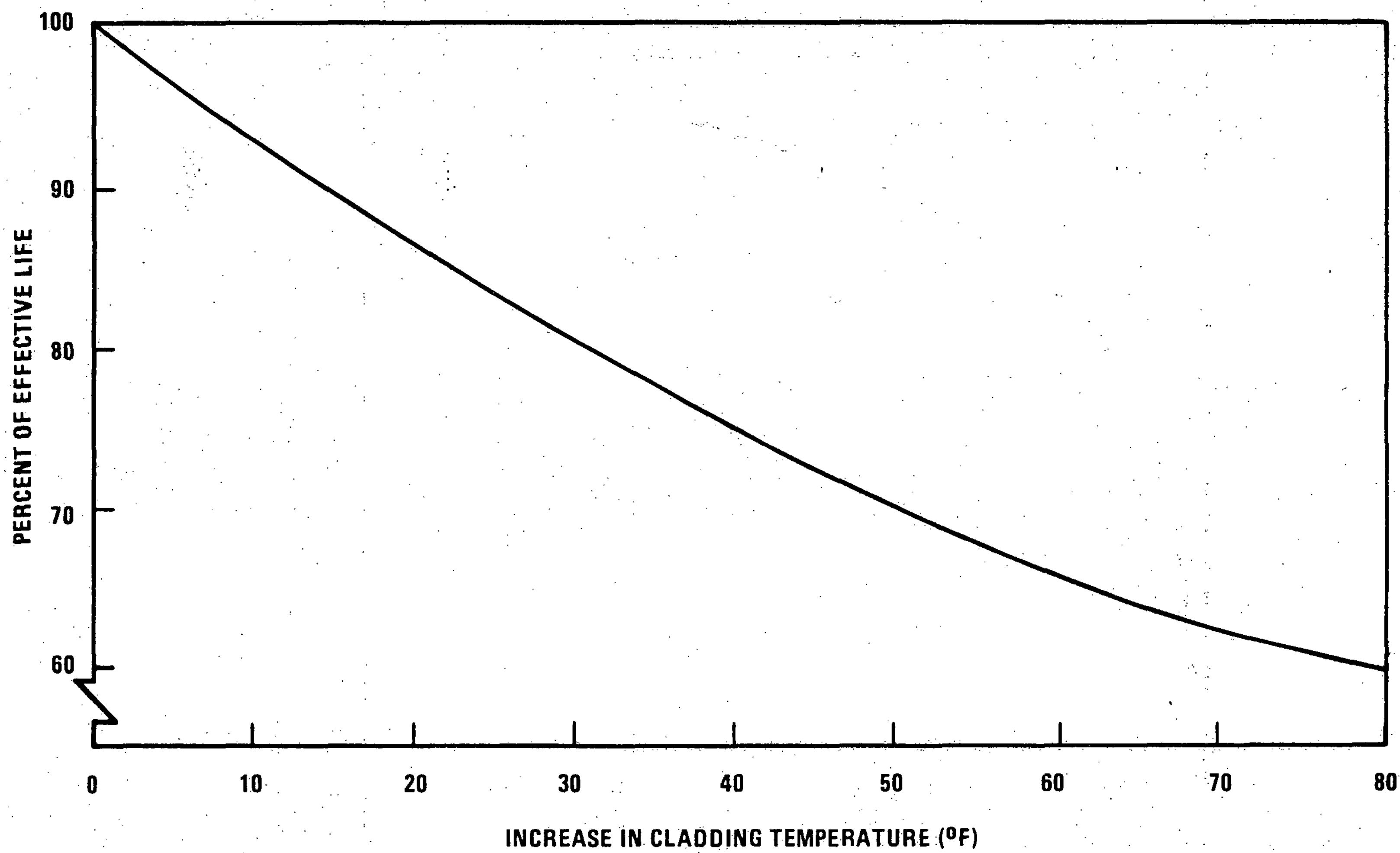


Figure 15.4.1.3.6-1. Effect of Hotspot Temperature on the Life of the Statistical Hot Fuel Rod

6729-12

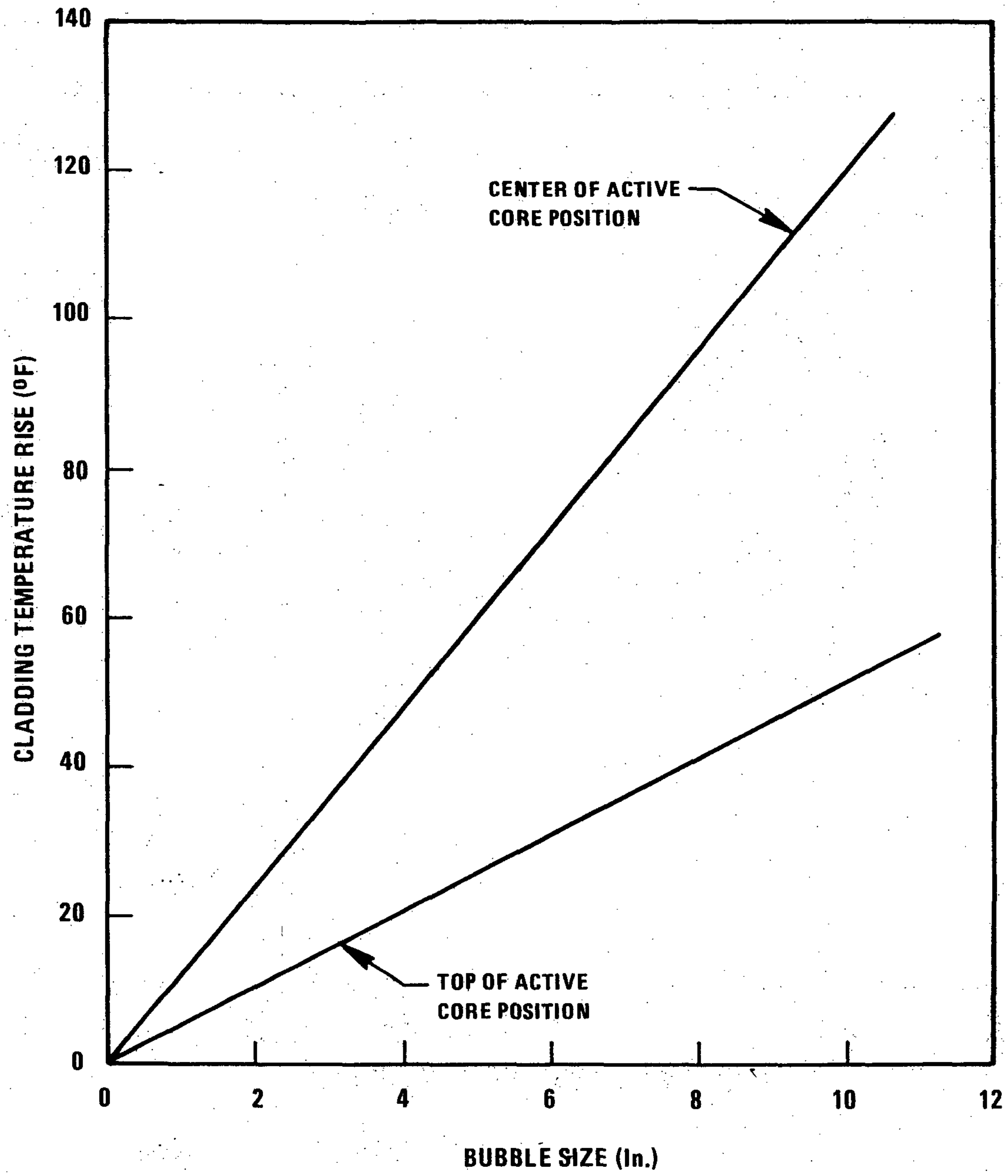


Figure 15.4.1.5-1. Fuel Pin Maximum Cladding Temperature Rise Due to Insulating Effect of Various Size Bubbles

6729-13

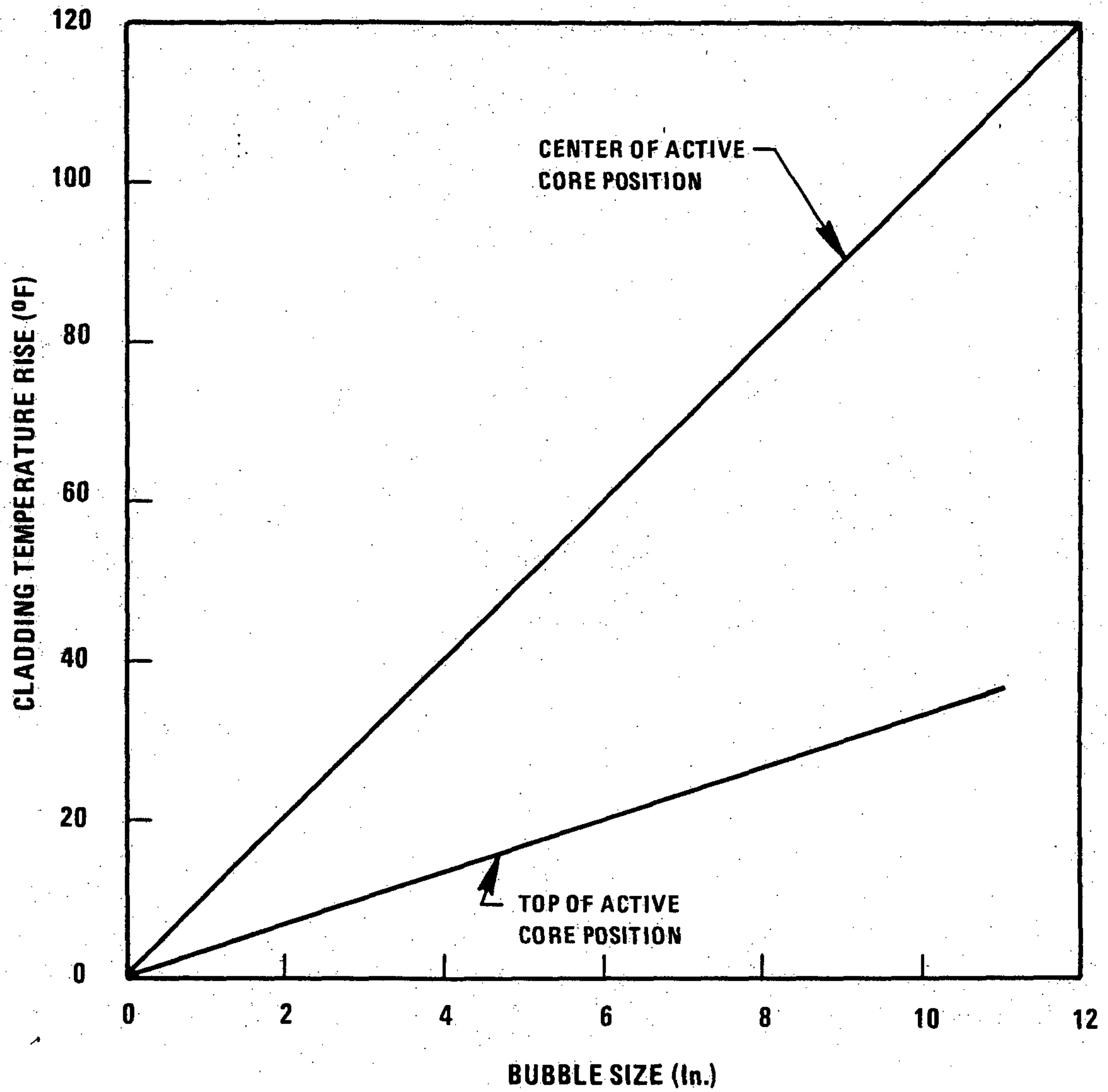


Figure 15.4.1.5-2. Radial Blanket Pin Maximum Cladding Temperature Rise Due To Insulating Effect of Various Size Bubbles

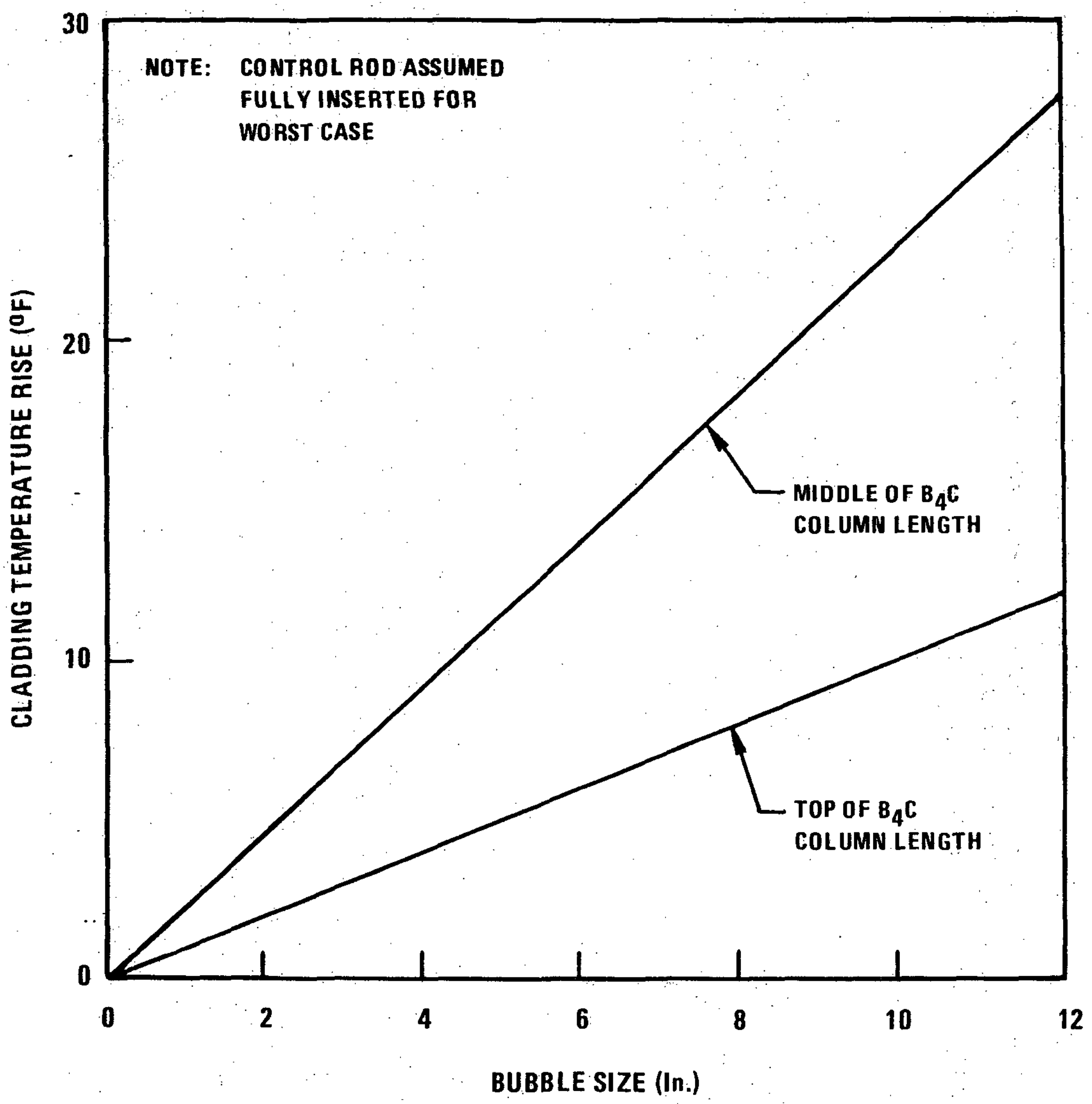


Figure 15.4.1.5-3. Control Rod Maximum Cladding Temperature Rise Due to Insulating Effect of Various Size Bubbles

6729-15

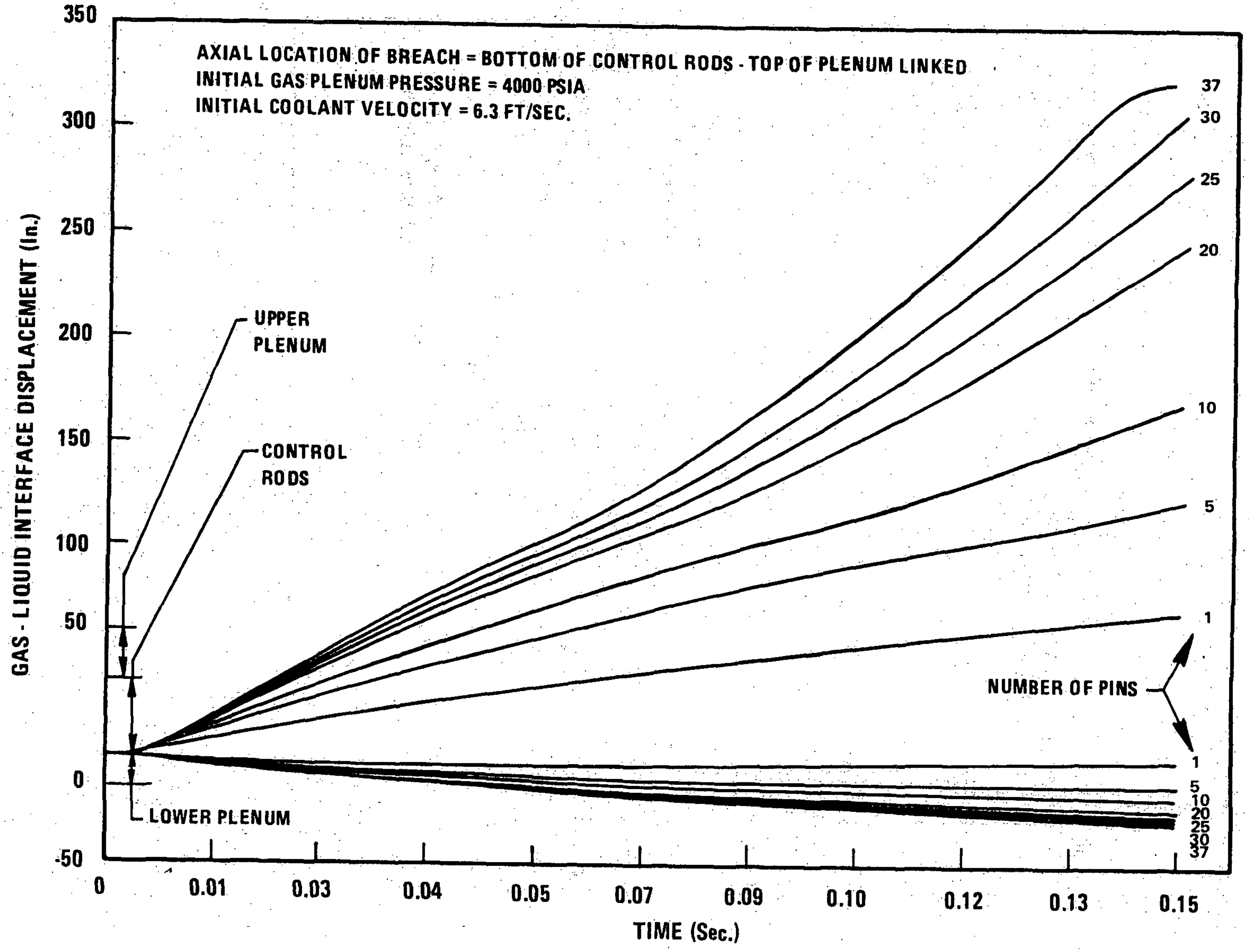


Figure 15.4.2.1-1. Rapid Gas Release In CRBR Control Rod Subassembly

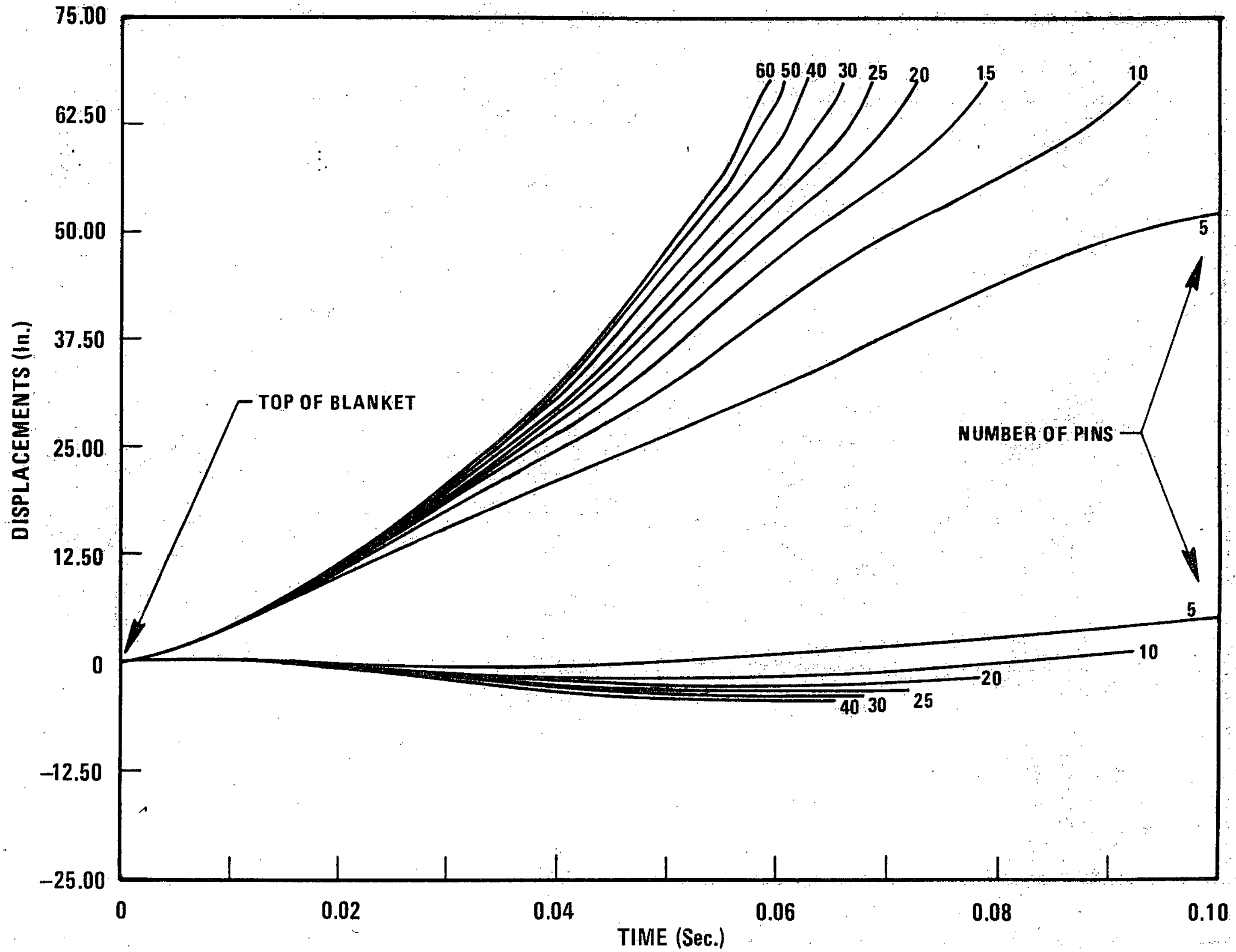


Figure 15.4.3.1.2-1. Displacements (Upper and Lower Gas-Liquid Interfaces - Inches) As a Function of Time After Rapid Fission Gas Release for Various Numbers of Pins

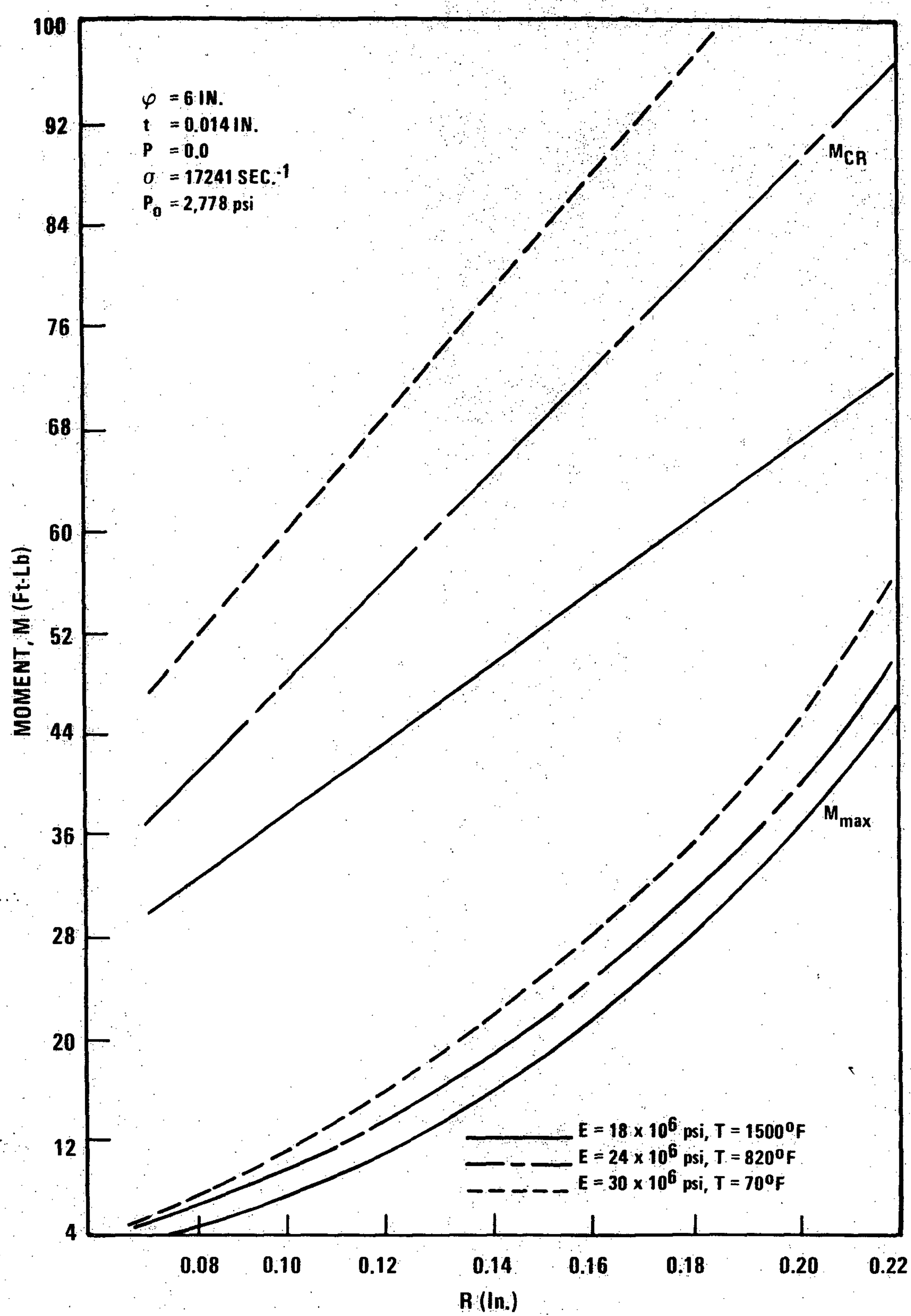


Figure 15.4.3.1.4-1. The Effects of Fuel-Element Radius and Temperature on Moment, M Collapsed Hinge Failure

6729-18

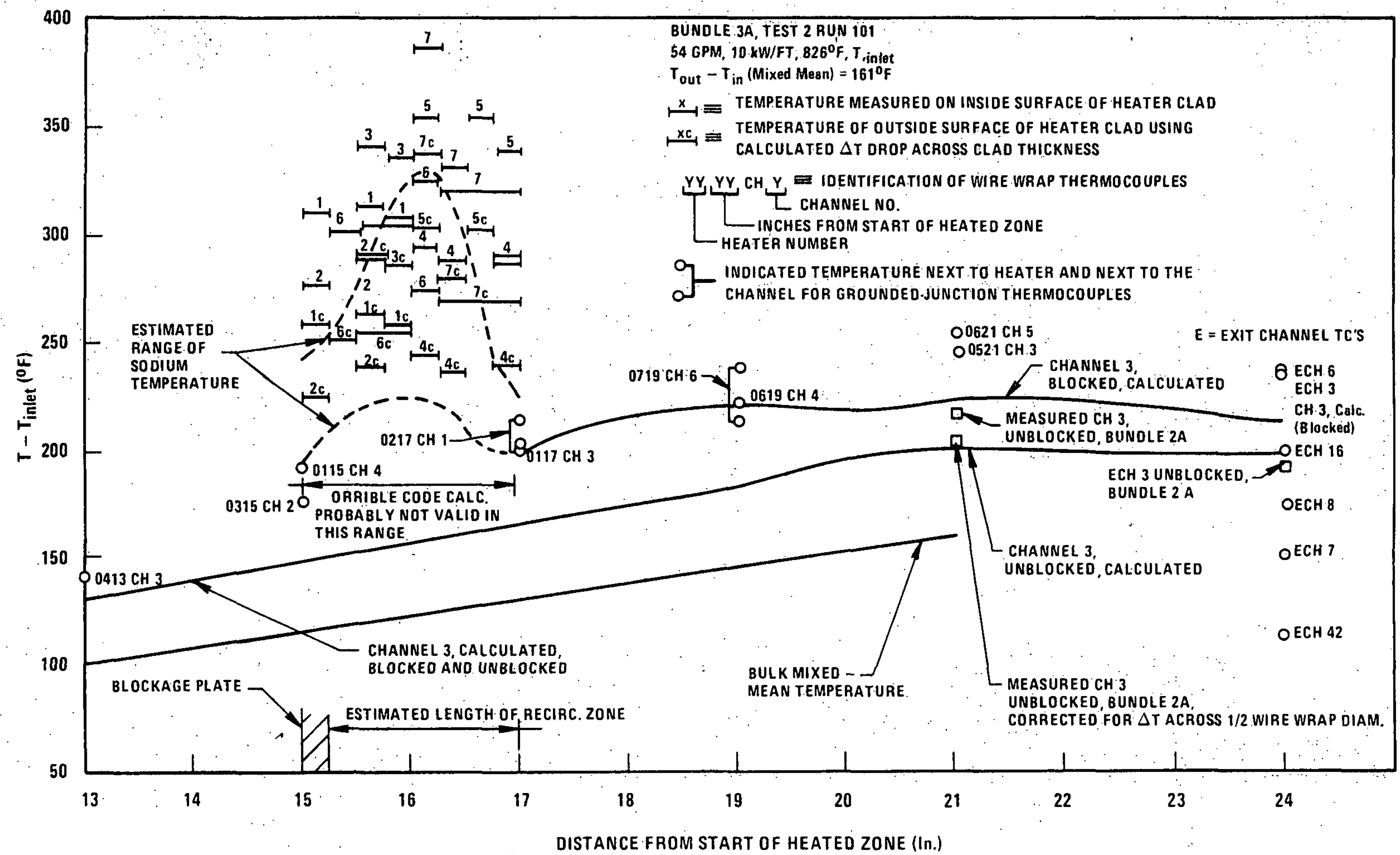


Figure 15.4.3.3.3-1. Experimental and Calculated Temperatures for the Six Channel Central Blockage Test Performed in the FFM Facility

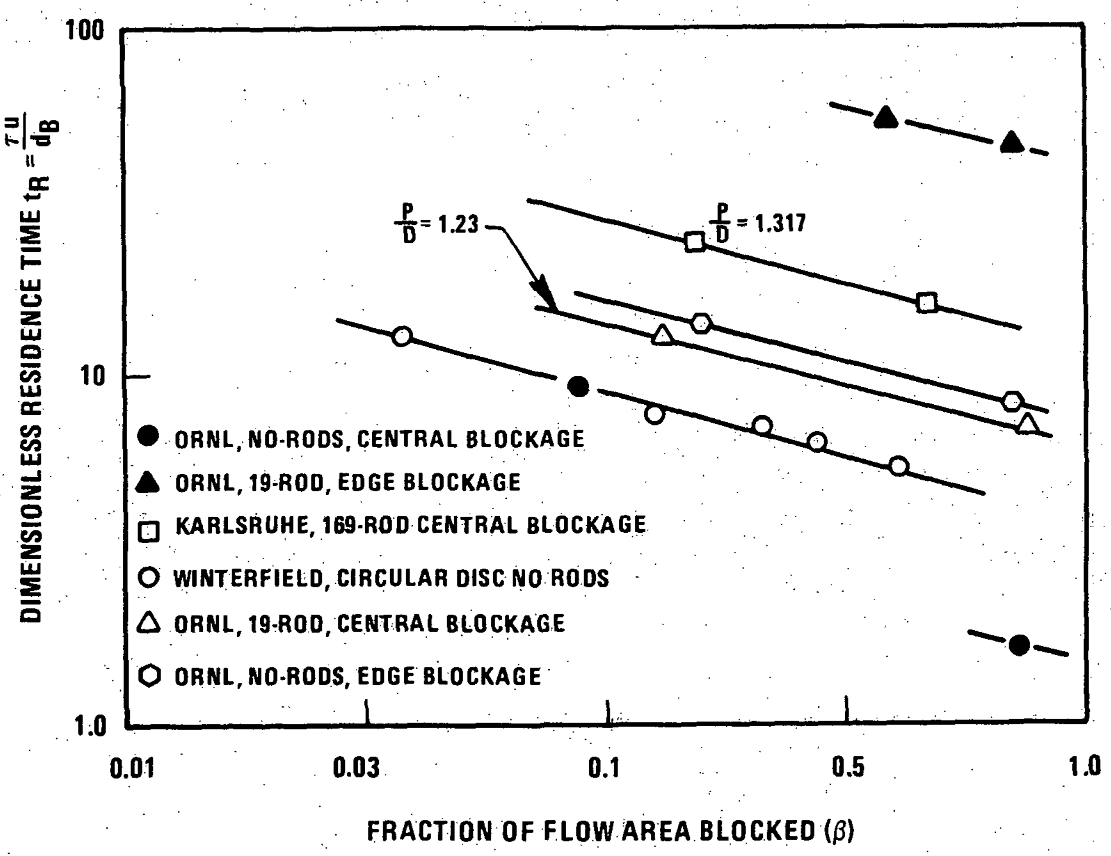


Figure 15.4.3.3.3-2. Dimensionless Residence Time as a Function of Fractional Blockage Area

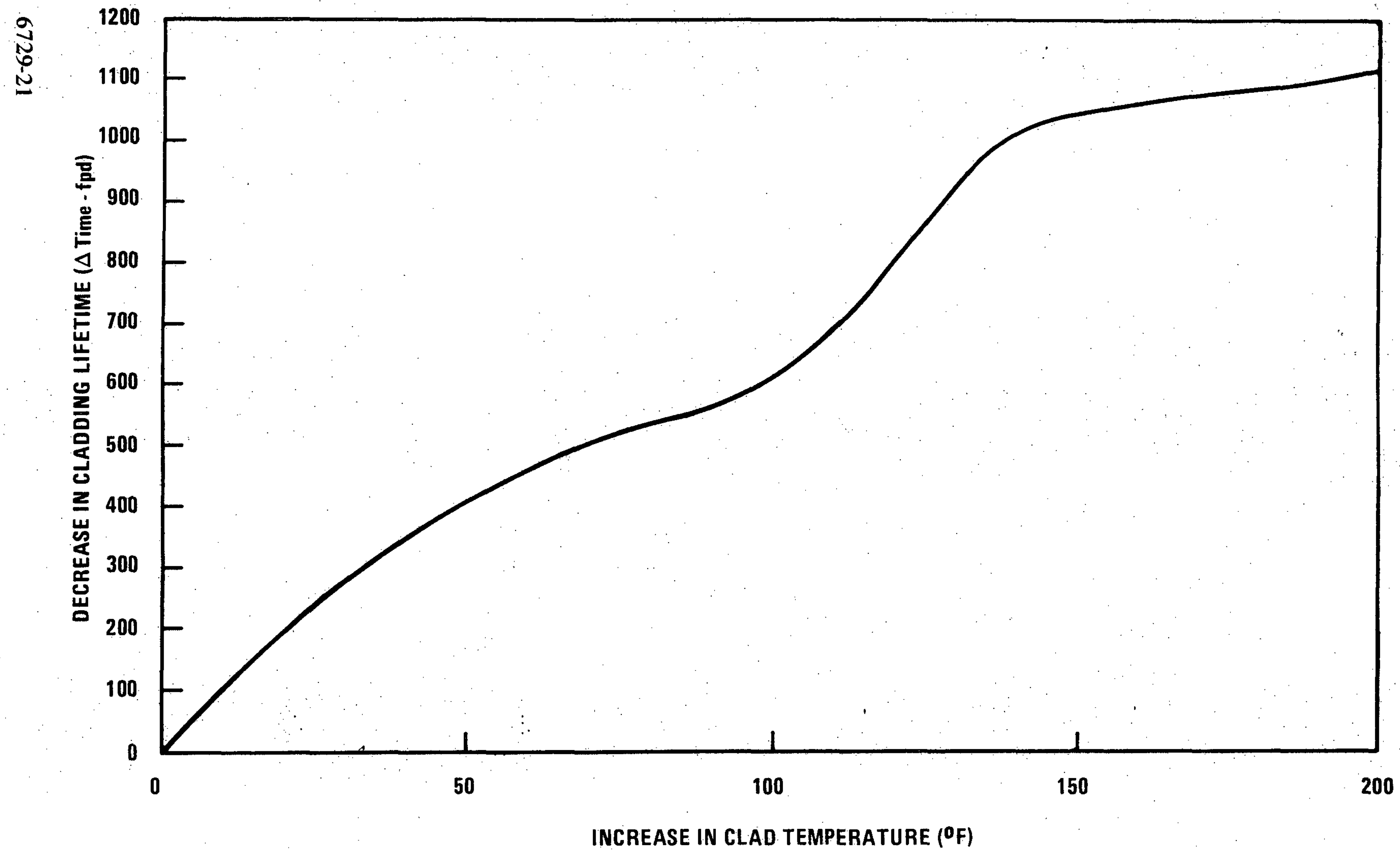


Figure 15.4.3.3.6-1. Effect of a Cladding Temperature Increase on Radial Blanket Rod Lifetimes

15.5 FUEL HANDLING AND STORAGE EVENTS - INTRODUCTION

Of considerable importance to the safe operation of the CRBRP is the determination of the consequences associated with a group of postulated fuel handling and storage events. The events presented in this Section (see Table 15.5-1) are considered to be representative of the type of fuel handling and storage events that could be expected to be encountered during the lifetime of the plant. For these accident events either 1) the whole body dose at the Site Boundary and also at the Low Population Zone will be presented, or 2) reference will be made to an enveloping analysis which has previously been considered and known to be within the suggested guideline dose limits. The suggested guideline limits for whole body dose used for these events are 10CFR20 for the anticipated and unlikely events and one-tenth of 10CFR100 for the extremely unlikely events.

The conservative assumptions and conditions used for these analyses are:

1.) The fuel assemblies are considered to be at end-of-equilibrium-cycle thus providing maximum fission product inventory.

2.) The site-boundary and low-population zone doses were calculated based on a worst short term atmospheric dispersion as identified from historical site data (see Section 2.3.4.2 for more details). The atmospheric dilution factors (X/Q) used for the accident conditions are consistent with Chapter 2 and shown here in Table 15.5-2.

3.) Reactor cover-gas radionuclide inventory is based on continuous plant operation with 1% failed fuel.

The following is a summary table of the events considered in this Section. Table 15.5-1 identifies, 1.) the event, 2.) the site boundary dose and the low-population dose, and 3.) comments on the severity of the event.

The requirement from the SFAC that a loss of site power during a fuel handling event be presented in the PSAR has not been addressed on the basis that off-site and on-site power are independent sources. Loss of either of these independent power sources during fuel handling would not prevent a safe continuation of the fuel handling process.

TABLE 15.5-1

FUEL HANDLING AND STORAGE EVENTS

Section No.	Event	@ Site Boundary (2-hr)	@ Low Population Zone (30-day)	Comments
15.5	Fuel handling & storage events			
15.5.1	Anticipated events (None)			
15.5.2	Unlikely events			
15.5.2.1	Fuel assembly dropped within reactor vessel during refueling	$<3.0 \times 10^{-4}$ REM	$<2.42 \times 10^{-5}$ REM	Consequences of this event are within the umbrella of Section 15.5.2.4
15.5.2.2	Damage of fuel assembly due to attempt to insert a fuel assembly into an occupied position.	$<2.77 \times 10^{-3}$ REM	$<1.0 \times 10^{-3}$ REM	This event is well within the suggested guideline dose rate.
15.5.2.3	Single fuel assembly cladding failure and subsequent fission gas release during refueling	2.77×10^{-3} REM	1.0×10^{-3} REM	This event is well within the suggested guideline dose limits
15.5.2.4	Cover gas release during refueling	3.0×10^{-4} REM	2.42×10^{-5} REM	This event is well within the suggested guideline dose limits
15.5.2.5	Heaviest crane load impacts reactor closure head	$<3.0 \times 10^{-4}$ REM	$<2.42 \times 10^{-5}$ REM	Consequences of this event are within the umbrella of Section 15.5.2.4
15.5.3	Extremely unlikely event			
15.5.3.1	Collision of EVTM with control rod drive mechanism	$<3.0 \times 10^{-4}$ REM	$<2.42 \times 10^{-5}$ REM	Consequences of this event are within the umbrella of Section 15.5.2.4

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TABLE 15.5-2

ATMOSPHERIC DILUTION FACTORS
X/Q FOR ACCIDENT CONDITIONS
(sec/m^3)

Distance (miles)	Hours		Days	
	0-8**	8-24	1-4	4-30
0.1	2.88E-2*	1.21E-2	3.13E-3	1.29E-3
0.2	8.57E-3	2.18E-3	8.71E-4	3.61E-4
0.3	4.03E-3	1.11E-3	4.18E-4	1.73E-4
0.4+	2.30E-3	6.09E-4	2.35E-4	9.12E-5
0.5	1.69E-3	4.47E-4	1.70E-4	7.03E-5
0.7	1.05E-3	2.69E-4	1.01E-4	4.18E-5
1.0	7.97E-4	1.51E-4	5.81E-5	2.41E-5
2.0	3.99E-4	6.75E-5	2.51E-5	1.04E-5
5.0	1.44E-4	2.36E-5	7.75E-6	3.21E-6
10.0	6.49E-5	2.87E-6	2.15E-6	8.90E-7
20.0	3.20E-5	1.99E-6	8.81E-7	3.65E-7

Minimum exclusion distance (2200 ft)

*2.88E-2 = 2.88 x 10^{-2}
**Includes building wake factors given in Table 2.6-23 of Chapter 2.0
+Exclusion distance - 2200 ft

15.5.1 Anticipated Events (None)

15.5.2 Unlikely Events

15.5.2.1 Fuel Assembly Dropped Within Reactor Vessel During Refueling

15.5.2.1.1 Identification of Causes and Accident Description

Dropping a fuel assembly into the core during refueling can be caused by either of two circumstances. These are: (1) that the fuel assembly is initially grappled improperly, or (2) that the In-Vessel Transer Machine (IVTM) grappler fingers are released erroneously. A long series of IVTM design feature failures and interlock failures, plus operator errors, are required to cause an erroneous IVTM grapple release or improper initial grappling. Figure 15.5.2.1.1-1 shows in diagramatic form the various sequences required in order to drop a fuel assembly during refueling. Since the IVTM draws a fuel assembly out of the core into a tubular housing, the fuel assembly can only be dropped in the vertical position, consequently the assembly can be dropped into an open lattice position such as a core or storage location or onto an occupied location which already contains another assembly. Dropping of the assembly
44 would be detected by the grapple load cells registering a loss of weight.

15.5.2.1.2 Analysis of Effects and Consequences

The maximum safety consequences of dropping a spent fuel assembly in the reactor is the release of fission gas from the failed cladding to the interior of the reactor vessel, and the possible dispersal of fuel particles due to mechanical damage to the assembly. This location within the reactor vessel has a sealed atmosphere with a radioactive cover gas processing (clean-up) system capable of limiting the release of radioactivity to the environment. Any radioactive materials dispersed in the sodium coolant are confined within
44 the primary heat transport system which is housed completely within the reactor containment system. Even if a damaged, dropped assembly is left in the core, the radioactive fuel material will only be dispersed within the primary heat transport system by the reactor coolant. So the consequences of damaging of a fuel assembly are confined by the boundaries of the primary heat transport system.

The particular damage to the fuel assemblies in the core depends on the circumstances of the particular "dropped assembly" incident. If the assembly is dropped freely and fully into an open lattice position, the only damage expected will be due to the high impact "g" load exerted on the "dropped" assembly. If the assembly doesn't drop freely and fully into an open lattice position, then additional severe damage can be inflicted to the assembly, to the adjacent assemblies, to the IVTM, and to the reactor upper internals, if the triple rotating plugs in the reactor head are operated. If the assembly is dropped on an occupied position, the consequences are the same as attempting to insert a fuel assembly into an occupied position (15.5.2.2.) with additional severe damage being inflicted if the triple rotating plugs are operated.

If an assembly were to be dropped during refueling, there would be no possibility of criticality occurring. During the refueling process, the reactor is shutdown with all control rods fully inserted (control rod drivelines are disconnected). With the reactor in this configuration, the core is shutdown by as much as 17$ to 31$. (31$ @ end of 1st cycle and 17$ @ loading of last fuel assembly prior to 2nd cycle operation.) At the end of a typical equilibrium cycle the reactor is shutdown by as much as 42$. The highest worth fuel assembly is less than 2$. Therefore, even for this worst case, the drop of a single fuel assembly prior to start up of the second cycle, the net shutdown margin remains considerable (greater than 15$).

15.5.2.1.3 Conclusions

Dropping of a fuel assembly into the core during refueling can be accommodated from both the standpoint of a potential rupture of the clad and subsequent release of fission gas and from the resulting reactivity insertion associated with the high worth fuel assembly. Based on the discussion of analysis of effects for this event (Section 15.5.2.1.2) the radioactivity released is less than that calculated for the umbrella event of Section 15.5.2.4. The net shutdown margin following insertion of the highest worth fuel assembly is greater than 15$.

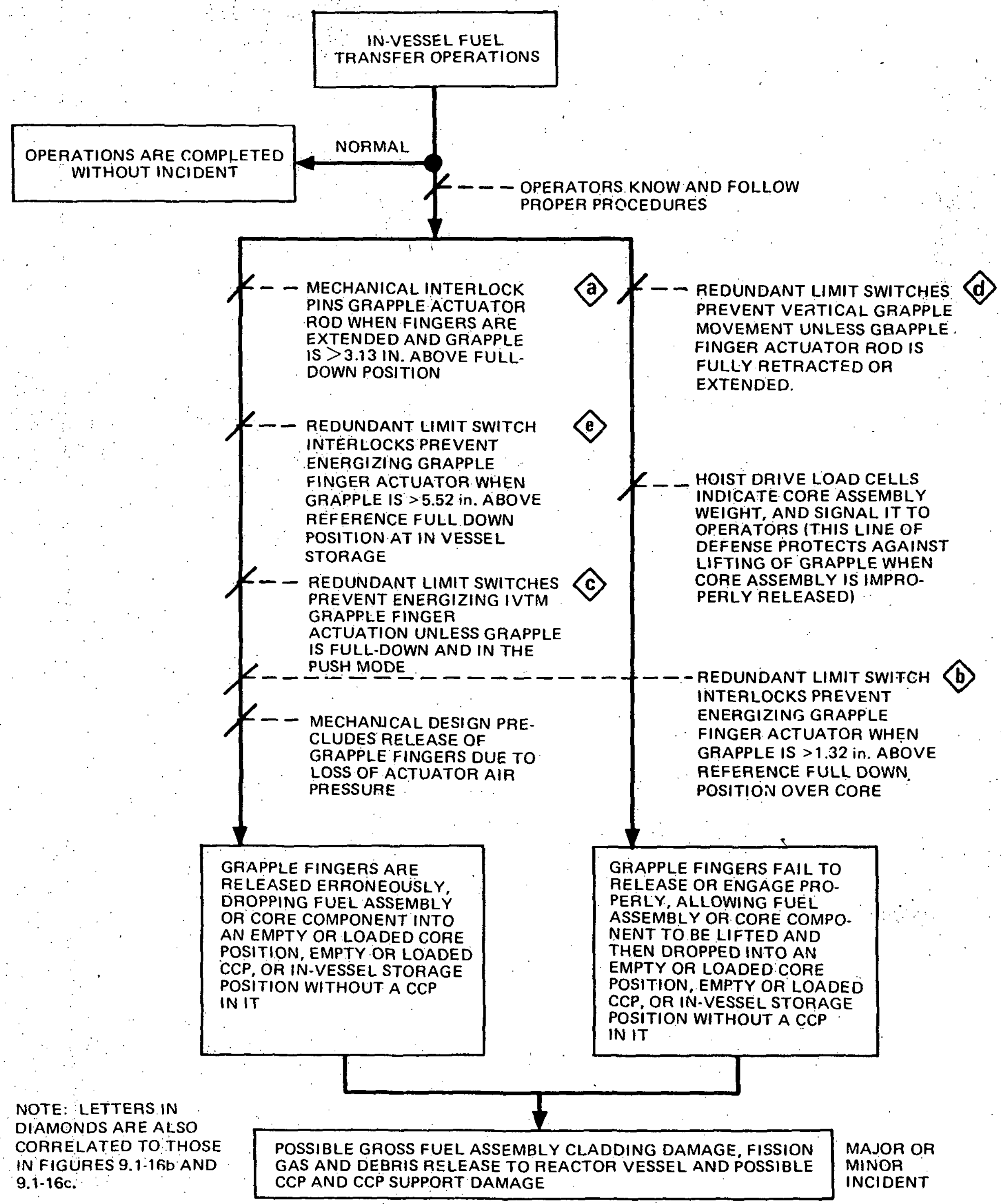

FIGURE 15.5.2.1.1-1. DROPPING A FUEL ASSEMBLY INTO CORE POSITION, CCP, OR IN-VESSEL STORAGE POSITION

15.5.2.2 Damage of Fuel Assembly Due to Attempt to Insert a Fuel Assembly Into An Occupied Position

15.5.2.21 Identification of Causes and Accident Description

A fuel assembly can be inserted into an occupied position in the
core lattice only as a result of a sequence of independent operator errors.
The refueling procedures require that detailed records of the location of core
assemblies are maintained. Core assemblies are identified by coded notches
in the core assembly handling socket before insertion, and the data system
44 cross checks the locations of assemblies for compatibility. Figure 15.5.2.2-1
shows in diagrammatic form the sequence required in order to damage a fuel
44 assembly by inserting a fuel assembly into an occupied position.

In addition, design features are provided for a load limiting system for the IVTM grapple, and a fuel assembly outlet nozzle configuration prevents the fuel pins within from being damaged by a second assembly being inserted into the outlet of the first.

15.5.2.2.2 Analysis of Effects and Consequences

Grapple drive systems are controlled through setpoints to maintain
low drive speeds, insertion and withdrawal forces, impact loads, and accelera-
tions (decelerations) within allowable limits. Loading a core assembly into
an occupied position can cause either fission gas release and/or core assem-
44 bly damage. The sudden release of fission gas from a fuel assembly during
refueling and its consequences are discussed in Section 15.5.2.3.

The core assembly outlet nozzle is designed to prevent insertion
of an inlet nozzle or a second assembly into an occupied core position during
refueling. Damage to the fuel pins within the fuel assemblies is mitigated
by the design of a restriction within the outlet nozzle. The restriction is
designed to prevent the lower end of a second assembly from entering the first
and causing mechanical damage to the internals. The restriction is designed
44 to take a bearing load of 3000 pounds with the normal load limit of the IVTM
set at 1000 pounds. Electrical interlocks are provided to halt the event when
the 1000 pound load limit is reached. Indication of this event would occur
on the downstroke, 13 ft higher than the normal expected overload trip point.

The core assembly outlet nozzle is also designed to prevent the inlet nozzle or a second assembly from seizing when inserted into the first thus assuring that the second assembly can be withdrawn out of the core once the interaction has been detected.

15.5.2.2.3 Conclusions

Interlocks, design features and procedures inherent in the refueling operation preclude the insertion of a fuel assembly into an occupied position. The analysis shows that the consequences of this event, fission gas release (Section 15.5.2.3) and/or core assembly damage, are acceptable.

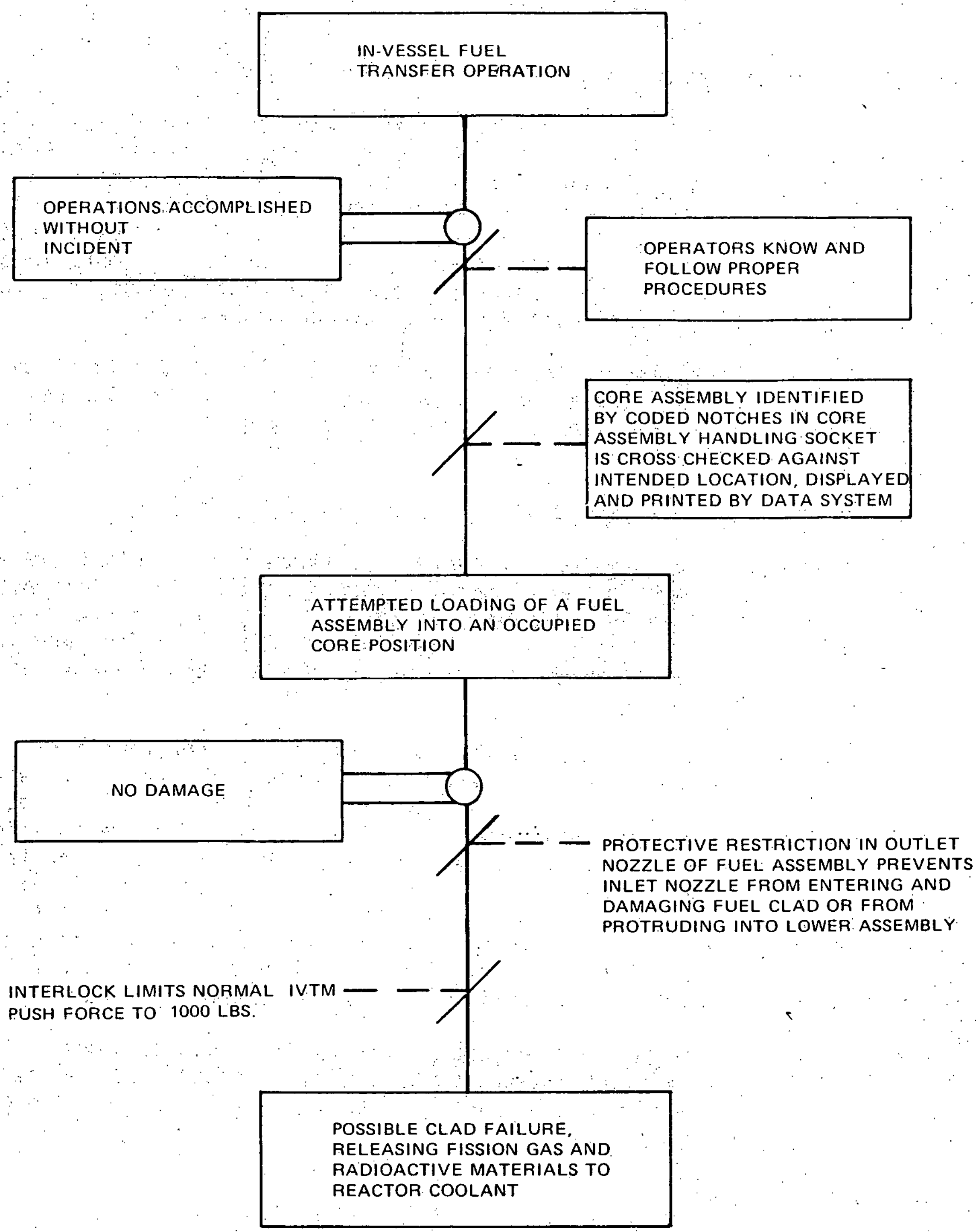

FIGURE 15.5.2.2-1. PROTECTION AGAINST INSERTION OF FUEL ASSEMBLY INTO OCCUPIED POSITION

15.5.2.3 Single Fuel Assembly Cladding Failure and Subsequent Fission-Gas Release During Refueling

15.5.2.3.1 Identification of Causes and Accident Description

Spent fuel assemblies are handled as single entities by the Ex-Vessel Transfer Machine (EVTM), the Fuel Handling Cell (FHC) gas cooling
44 grapples, and IVTM.

The basic potential causes for fuel assembly cladding failure during refueling are mechanical damage and inadequate cooling. Conceivably, mechanical damage could be caused by dropping the assembly, improper loading (attempted loading into an occupied location), or faulty sequencing of refueling motions (moving or closing different components with the fuel assembly in the wrong place). Inadequate cooling could conceivably be caused by loss of power during refueling, or cooling system failure.

The Refueling System is designed to reduce the probability of the occurrence of mechanical damage to fuel assemblies and other components to a level as low as possible, approaching the extremely unlikely classification. This is accomplished by selecting design features that are inherently less likely to fail, by incorporating redundant features and controls where needed, by use of interlocks and controls that prevent maloperation, and by use of operator training and approved written operating procedures. Specifically, the fuel assembly is of a "ducted" design where the fuel pins are "shrouded" to protect them from contact with other objects. Also, the fuel handling grapples within the EVTM and FHC are provided with positional and load limiting interlocks to preclude mechanical damage to the fuel assembly being handled. The Refueling System design bases and safety evaluations are discussed in Section 9.1.

The cooling systems designs for the EVTM and FHC gas cooling grapple are directed toward features that reduce the likelihood of system capability falling below requirements. This is accomplished with redundant cooling sys-
44 tems and control interlocks that prevent misoperation, and through the use of operator training and approved written operating procedures.

The consequences of fission-gas release from a fuel assembly are dependent on the location of the fuel assembly at the time of the event. In particular, the potential for release of radioactivity is reduced where facilities have gas cleanup systems, is increased in locations containing a large number of seals, and is also increased in locations where the gas volume is low, creating a high concentration of activity. Of the locations for possible cladding failure, the reactor vessel, ex-vessel storage tank (EVST), and the fuel handling cell (FHC) have gas cleanup systems, and fairly large gas volumes which result in low concentrations of activity. The ex-vessel transfer machine (EVTM) has no gas cleanup system, a fairly small gas volume, and a large number of seals. Therefore, the location for fission-gas release from a spent fuel assembly most likely to result in the largest release of radioactivity is the EVTM when it is not connected to a location with gas cleanup. This is the accident that was analyzed.

Amend. 44
April 1978

15.5.2.3.2 Analysis of Effects and Consequences

The earliest possible time, based on a refueling operations time study, for handling of any core component with the EVTM is 36 hr after reactor shutdown (from long-term full-power operation); the earliest scheduled time for handling any fuel assembly is 87 hr after shutdown; and the earliest scheduled time for handling the maximum powered fuel assembly is ~320 hr after shutdown. These times assume 100% handling efficiency. Expected efficiencies are in the range of 75 to 50% which further lengthens the normal decay time by a factor of 1.3 to 2 (no credit is taken for the additional decay). Based on the fission product inventory of the maximum powered fuel assembly after 3 years of full-power operation, the noble gas and iodine activity of the fuel assembly for 0, 36, and 87 hr of decay are shown in Table 15.5.2.3-1.

TABLE 15.5.2.3-1

MAXIMUM FUEL ASSEMBLY FISSION-GAS INVENTORY

Isotope	Half Life	Ci 0-hr Decay* Time	Ci 36-hr Decay Time	Ci 87-hr Decay Time
Kr^{83m}	1.86 hours	1.61×10^{4}	2.36×10^{-2}	~0**
Kr^{85m}	4.4 hours	3.23×10^{4}	109	3.46×10^{-2}
Kr^{85}	10.8 years	516	516	516
Kr^{87}	76 minutes	5.44×10^{4}	1.52×10^{-4}	~0**
Kr^{88}	2.80 hours	6.61×10^{4}	8.77	2.82×10^{-5}
Xe^{131m}	11.8 days	832	827	809
Xe^{133m}	2.26 days	7020	5600	3310
Xe^{133}	5.27 days	2.35×10^{5}	2.18×10^{5}	1.74×10^{5}
Xe^{135m}	16 minutes	7.94×10^{4}	1970	10.3
Xe^{135}	9.2 hours	2.75×10^{5}	4.78×10^{4}	1310
Xe^{138}	17.0 minutes	2.14×10^{5}	~0	~0
I^{129}	1.6×10^{7} years	2.77×10^{-3}	2.77×10^{-3}	2.77×10^{-3}
I^{130}	12.6 hours	1660	229	13.8

TABLE 15.5.2.3-1 (Continued)

	Isotope	Half Life	Ci 0-hr Decay* Time	36-hr Decay Time	87-hr Decay Time
	I^{131}	8.1 days	1.45×10^5	1.31×10^5	1.10×10^5
	I^{132}	2.4 hrs	1.90×10^5	1.34×10^5	8.56×10^4
	I^{133}	20.3 hrs	2.34×10^5	7.13×10^4	1.32×10^4
20	I^{134}	53 min	2.64×10^5	∿0**	∿0**
	I^{135}	6.68 hrs	2.57×10^5	6.30×10^3	33.0
20	Total		2.07×10^6	6.18×10^5	3.89×10^5

*Average Fuel Assembly Fission Gas activities at time of shutdown were in-
44 creased by 20% to approximate the values for the maximum fuel assembly.
**Less than 10^{-6}.

Of the two cases considered, 36 and 87-hr decay, the first represents a maximum upper limit release that is in the Extremely Unlikely category, and the second represents a maximum release for events in the Unlikely class. In both cases, the releases would be from the fuel assembly to the interior of the EVTM. For the purpose of this evaluation, immediate release of 100% of the noble gas and halogen inventories of the fuel assembly to the interior of the EVTM was assumed. This is an extremely conservative assumption, since clad failure of all pins in the assembly would be required to effect this type of release. The radioactive gases inside the EVTM can then slowly diffuse through the seals of the EVTM and be released inside the Reactor Containment Building and the Reactor Services Building. (The refueling hatch connecting the two buildings will be open at this time.) Based on the 70 ft^3 of EVTM gas space being filled with reactor cover-gas prior to the fuel cladding failures (the machine is normally purged to prevent reactor cover gas from entering the EVTM), the gas activities within the EVTM for the two cases are shown in Table 15.5.2.3-2.

All seals in the EVTM are double seals. In addition, all dynamic seals are supplied with a pressurized buffer gas between the seals, and the buffer gas pressure is monitored. All static seals will be double with installed capability for periodic testing. Thus, leakage of EVTM gases due to physical defects in the seals is unlikely. The only realistic mechanism for leakage through these seals is by diffusion of the material (radioactive gases in particular) through the elastomer. Based on the EVTM seal materials, dimensions, operating temperatures, and the experimentally determined permeabilities for Fast Flux Test Facility (FFTF) (Ref. 1) seals, the diffusion rates for the isotopes in the concentrations listed in Table 15.5.2.3-2 were evaluated. Due to the uncertainty in seal materials, movable versus static seals, etc., the permeabilities used were conservatively assumed to be twice the maximum measured permeabilities. The specific rates used in the analysis, which are two times the measured rates, are as follows: 4.7×10^{-3}, 3.8×10^{-3}, 2.9×10^{-3}, and 1.4×10^{-2} cc/sec, for Xe, Kr, Ar, and H-3, respectively. This small leakage will be assured by a technical specification limit (section 16.3.10) and appropriate leak testing requirement. 1

Permeabilities are based on the maximum measured permeabilities for Buna-N seals. Test data for this type of seals are reported in Reference 2 and shown in Figure 15.5.2.3-1.

The diffusion rate for each element was calculated from its permeability and the proper seal dimensions according to the following formula:

$$D_i = A \sum_{j=1}^{n} (\overline{P}_i \times L)_j$$

where D_i = diffusion rate of element i (cm^3/sec)

A = conversion factor = 193.04 (cm Hg/atm) (cm/inch)

$\overline{P}_i$ = permeability of element i at operating temperature through seal j (std cm^3 . cm/sec . cm^2.cm Hg)

L = seal perimeter of seal j (inch)

n = number of seals

The diffusion rates are considered conservative because of the following reasons:

(1) The permeability test data was used were maximum measured values which are higher than the averaged ones reported in Figure 19 of Reference 1.

25

15.5.2.3.3 Conclusion

The earliest scheduled time for handling a fuel assembly for refuel-
ing is 87 hr after reactor shutdown. In the unlikely event that complete re-
lease of the fuel assembly noble gases and halogens to the interior of the
EVTM occurs at this time, the potential offsite exposure will not exceed the
44 limits of 10 CFR 100.

The possibility of the postulated fission products release occurring
36 hr after shutdown is extremely unlikely. The refueling procedures call for
44 handling only control, blanket, and radial shield assemblies during the initial
stages of refueling. Proper fuel handling sequence is assured by administra-
tive procedures. Nevertheless, if the fuel assembly failure in the EVTM is
arbitrarily postulated 36 hr after shutdown (Case 1, Table 15.2.3-4), the re-
sultant offsite exposure is well within the guideline limits of 10 CFR 100.

REFERENCES

1.) AI-AEC-13113, "Quarterly Technical Progress Report, Sodium Technology and Cover Gas Seal Development Programs," July-September, 1973, issued November 15, 1973.

2.) AI-AEC-13113, "Quarterly Technical Progress Report, Sodium Technology and Cover Gas Seal Development Programs, July - September 1973", issued November 15, 1973. 25

TABLE 15.5.2.3-2

EVTM GAS ACTIVITY

Isotope	μCi/cc 36-hr Decay Time	μCi/cc 87-hr Decay Time
H^3	1.39×10^{-4}	1.39×10^{-4}
Ar^{39}	0.145	0.145
Kr^{85m}	54.9	1.74×10^{-2}
Kr^{85}	260	260
Kr^{88}	4.42	1.42×10^{-5}
Xe^{131m}	416	407
Xe^{133m}	2820	1670
Xe^{133}	1.10×10^{5}	8.76×10^{4}
Xe^{135m}	992	5.19
Xe^{135}	2.41×10^{4}	659
I^{130}	115	6.95
I^{131}	6.59×10^{4}	5.54×10^{4}
I^{132}	6.75×10^{4}	4.31×10^{4}
I^{133}	3.59×10^{4}	6640
I^{135}	3170	16.6
Total	3.11×10^{5}	1.96×10^{5}

TABLE 15.5.2.3-3

INITIAL RELEASE RATE THROUGH EVTM
TO RCB-RSB HIGH BAY

Isotope	µCi/sec 36-hr Decayed Fuel Assembly	µCi/sec 87-hr Decayed Fuel Assembly
H^{3}	1.90×10^{-6}	1.90×10^{-6}
Ar^{39}	4.26×10^{-4}	4.26×10^{-4}
Kr^{85m}	0.206	6.54×10^{-5}
Kr^{85}	0.978	9.978
Kr^{88}	1.66×10^{-2}	5.34×10^{-8}
Xe^{131m}	1.94	1.90
Xe^{133m}	13.1	7.78
Xe^{133}	514	408
Xe^{135m}	4.62	2.42×10^{-2}
Xe^{135}	112	3.07
I^{130}	0.536	3.24×10^{-2}
I^{131}	307	258
I^{132}	315	201
I^{133}	167	30.9
I^{135}	14.8	7.74×10^{-2}
Total	1450	912

TABLE 15.5.2.3-4

OFF-SITE DOSES DUE TO FUEL FAILURE IN EVTM

	10CFR100	Dose (REM) SB (2 hr) (0.417 mi)	Dose (REM) LPZ (30 days) (5.0 mi)
CASE 1 - 36 hours Decay Time (Extremely Unlikely)			
Cloud			
D_β (Skin)		5.87-03*	2.07-04
D_γ (Whole Body)	25	1.54-03	4.25-04
Inhalation			
Lung	75	3.7-02	1.5-02
Thyroid	300	1.89	0.121
Whole Body Inhalation	25	7.47-03	2.90-03
CASE 2 - 87 hours Decay Time (Unlikely)			
Cloud			
D_β (Skin)		3.23-04	1.15-04
D_γ (Whole Body)	25	8.98-04	2.40-04
Inhalation			
Lung	75	2.40-02	1.02-04
Thyroid	300	1.44	0.628
Whole Body	25	2.77-03	1.00-03

*5.87-03 = 5.87×10^{-3}

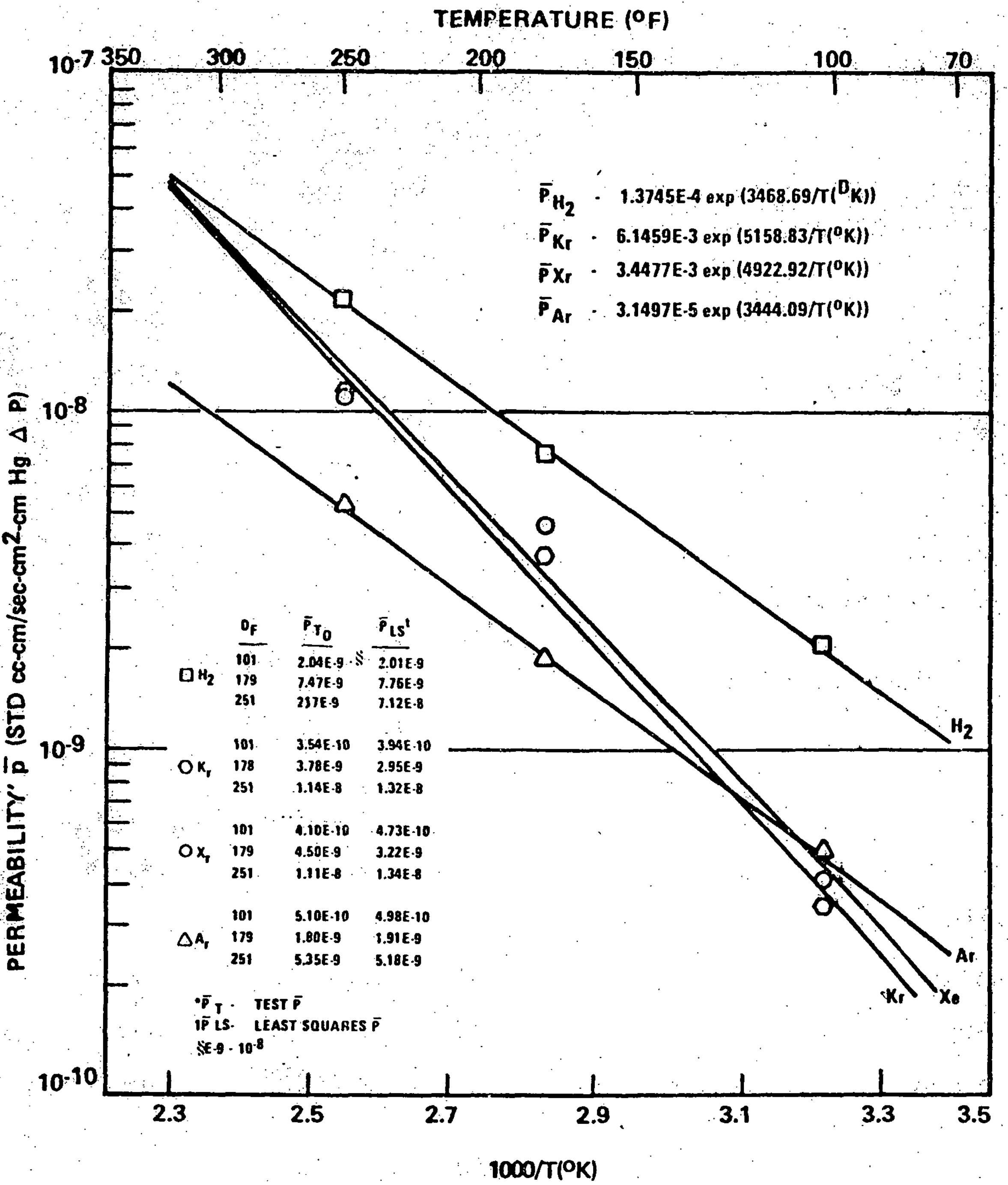

Figure 15.5.2.3-1 Log $\bar{P}$ vs 1/T(°K) for H_2, Ar, Kr, and Xe for Minnesota Rubber Co., Buna - N, Compound 366Y

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15.5.2.4 Cover-Gas Release During Refueling

15.5.2.4.1 Identification of Causes and Accident Description

The potential causes for cover gas release during refueling are: (a)
mechanical damage, (b) improper sequencing of refueling motions, and (c) sepa-
44 ration of the AHM from an open floor valve during a seismic event. Mechanical
damage is assumed to include seal, system, and component damage resulting from
misoperation, and collisions of the refueling and other equipment that leads to
cover gas release. Failure of inner redundant seals on equipment mounted on the
reactor closure, if undetected during seal leak testing or surveillance, con-
ducted before outer seal opening, could lead to release of cover gas when the
outer seal is opened to accommodate refueling operations. The improper sequenc-
ing of refueling motions could lead to cover gas release by failure to close
floor or closure valves prior to moving of equipment, or by opening these valves
at an inappropriate time. The design basis of the AHM requires "breakaway" of
the AHM from the floor valve on the small rotating plug (SRP) of the reactor
during a seismic event (see 9.1.4.5). Though unlikely, the "breakaway" could
potentially happen at the very moment when the floor valve is in the open posi-
44 tion, resulting in reactor cover gas release through the SRP port into the RCB.

The Refueling System is designed to reduce the probability of occurence of the accidents to as low a level as possible. This is accomplished by selecting design features that have an inherently low likelihood of failure by use of interlocks and controls that prevent misoperation (these are described in Sections 7.6.2 and 7.7.1), and by use of operator training and administrative procedures that add knowledge and caution during potentially sensitive procedures. The Refueling System safety features are discussed in Section 9.1.

Also, the sequence of operations that must be performed before postulated failures could permit the release of the cover gas tend to mitigate the consequences of the incident by permitting the radioactivity to decay and to be removed by operation of the Radioactive Argon Processing System (RAPS).

Estimates of the time required to perform the operations identified above indicate that a minimum of 30 hours will elapse before the first redundant seal is removed. Therefore, the earliest time that any potential release of reactor cover gas could occur during this sequence is 30 hours after reactor shutdown.

To obtain a conservative estimate of the consequences of cover gas
release, it has been assumed that the release occurs at 30 hours following shut-
44 down, the earliest one could anticipate the occurence of this event.

15.5.2.4.2 Analysis of Effects and Consequences

The consequences of cover gas release are dependent on the location of the release, the quantity of activity, and the time of interval over which the release occurs. The reactor cover gas is the largest potential source of radioactive gas. The core is assumed to contain a small fraction (1%) of fuel pin with small cladding defects. The EVST, the EVTM, and the fuel handling cell (FHC) atmospheres are also potential sources of radioactive gas. The EVST and FHC have gas cleanup systems that maintain their cover gas activity below that of the reactor cover gas. The EVTM is normally expected to have little gas activity. Therefore, the analysis
49 of cover gas release was restricted to the reactor.

The cover gas fission product inventories at the time of this postulated release are based on the assumption that the Radioactive Argon Processing Systems (RAPS) has been in normal operation for an extended period of time before shutdown and has processed reactor cover gas for at least 30 hours after shutdown. The design basis (1% failed fuel) cover gas inventories are shown in Table 11.3-2. For this analysis, these inventories were
49 adjusted to account for radioactive decay and processing through RAPS during the 30 hour interval between shutdown and the postulated release.

The majority of this activity can be released to the Reactor Containment Building (RCB) and Reactor Service Building (RSB) in a fairly short period of time; and for evaluation purposes, all of it was assumed to be released instantaneously to these buildings and released as a puff to the outside environment. The refueling hatch in the RCB is assumed to be open at this time. It should be noted that at no time during the refueling operations can both the refueling hatch and the railroad door in the RSB be open at the same time. Therefore, even though no credit has been takne for it in the analysis, the reactor cover gas is always isolated from the environment by the confinement provided within the RCB/RSB. The resultant doses are given in Table 15.5.2.4-1

15.5.2.4.3 Conclusions

From the discussion presented in Section 15.5.2.4.1, the earliest one could anticipate the occurrence of this event is 30 hours into the refueling operation. From the data shown in Table 15.5.2.4-1, the whole body dose at the site boundary (4.4×10^{-3} REM)is well below the 10 CFR 100
49 guideline limits.

TABLE 15.5.2.4-1

OFF-SITE DOSES FROM COVER GAS RELEASE DURING REFUELING

	Dose (REM)*	
	Site Boundary (2 hr. -0.42 mi.)	LPZ (30 days-2.5 mi.)
D_β (Skin)	4.0×10^{-3}	1.1×10^{-3}
49 D_γ (Whole Body)	4.4×10^{-3}	1.2×10^{-3}

49 * Integrated exposure based on puff release.

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15.5.2.5 The Heaviest Crane Load Impacts the Reactor Closure Head

15.5.2.5.1 Identification of Causes and Accident Description

The CRBRP polar crane will service the head access area with refueling and maintenance equipment. The heaviest load identified to be handled over the reactor vessel head is the AHM. The polar crane is a double reeved design with velocity limiting drum brakes which limit the lowering speed to 5 fpm. If the AHM load of ~100 tons is accidentally lowered onto the head at the crane velocity limit, an impact of ~100 tons on the reactor enclosure head assembly is imposed. Normally only the AHM extender weight rests on the head
44 assembly.

Collision of the crane-handled AHM with the head and/or head-mounted equipment such as CRDM's has been classified as unlikely due to restrictions on crane travel, and as a result of operating personnel knowing and carrying out the written and approved procedures. The polar crane lowering speed restrictions mitigate possible damage to the head and head-mounted equipment.

15.5.2.5.2 Analysis of Effects and Consequences

The AHM being lowered at the crane velocity limit of 5 fpm onto the AHM floor valve and port adapter, would result in two overload considerations for the reactor head assembly: (1) supporting the AHM static load of ~100 tons, and (2) absorbing the impact energy developed by a weight of ~100 tons at 5 fpm.

An analysis was performed for this case and results showed that the head assembly can and will be designed to support the static load (~100 tons) of the AHM. In addition, the design will also be capable of absorbing, without any detrimental structural affect, the ~100-ton load at an impact velocity of
44 5 fpm.

At the extreme limit of damage to the head, the leakage of fission gas from the reactor in this event will not exceed that which was analyzed in Section 15.5.2.4.

15.5.2.5.3 Conclusions

Based on the data currently available, it appears that the head assembly can withstand without any detrimental structural effects, the lowering of
44 the ~100-ton Auxiliary Handling Machine at a velocity of 5 fpm. However, if damage to the CRDM's or other head-mounted equipment should occur, analysis of the consequences shows that the release of reactor cover gas is within the design limits discussed in Section 15.5.2.4.

15.5.3 Extremely Unlikely Events

15.5.3.1 Collision of EVTM with Control Rod Drive Mechanisms

15.5.3.1.1 Identification of Causes and Accident Descriptions

The EVTM is a massive, railway gantry-mounted, shielded cask type fuel transfer machine. It moves, during refueling, on its gantry to within ∿1 ft of the CRDM's. Note that these operations only occur when the reactor is shut down. At that time, the absorber assemblies are fully inserted and disconnected from the CRDM's.

During refueling, the EVTM is moved on its gantry between EVST, fuel handling cell, and reactor on rail tracks. These rail tracks are mounted in a floor trench. The gantry wheel track structure incorporates anti-liftoff and over-turning restraints. At both rail ends, fixed mechanical rail stops are mounted to limit the EVTM gantry travel in the event of a travel limit switch failure, and in addition, an operator error or braking failure. These features, combined with the written and approved operating procedures for the operators, reduce the likelihood of an EVTM collision with the CRDM's to the extremely unlikely level.

15.5.3.1.2 Analysis of Effects and Consequences

Because the reactor is shut down at the time, and the absorber assemblies are fully inserted into the core and disconnected from the drive lines, the collision cannot cause an increase in reactivity. Therefore, no reactivity event can occur as a result of this incident. The collision can cause damage to the CRDM's which would result in delaying startup because of repair time. The collision can also cause failure of the CRDM cover gas seals, which would result in the release of cover gas to the Reactor Containment Building and the Reactor Service Building. This event is not as severe as the reactor cover gas release event, because the seal failure would result in a more gradual leakage of cover gas. At the extreme limit, the leakage of fission gas from the reactor in this event will not exceed that which was analyzed in Section 15.5.2.4.

15.5.3.1.3 Conclusions

Because of the inherent design features and operating procedures for the Ex-Vessel Transfer Machine, the likelihood of a collision of this machine with the CRDM is extremely unlikely. However, analysis of the consequences, should this event occur, show that release of reactor cover gas is within the design limits discussed in Section 15.5.2.4.

15.6 SODIUM SPILLS - INTRODUCTION

Postulated sodium fires could possibly result in the dispersion of some radioactive material to the atmosphere. Fires involving primary sodium coolant are of most concern since this sodium circulates through the reactor core and accumulates radioactivity due to neutron activation and entrainment of fission products leaking from defective fuel. Postulated fires involving sodium used in the Ex-Vessel Storage Tank (EVST) cooling system could also result in radiological releases. The EVST sodium is essentially non-radioactive at the beginning of plant life. However, during refueling a small quantity of primary sodium is tranferred to the EVST along with each irradiated assembly, resulting in a slow buildup of radioactivity in the EVST sodium.

Besides the potential radiological impact of postulated sodium fires, these fires can result in pressure/temperature transients. Therefore, for each fire the consequences are evaluated in terms of: 1) the potential individual whole body and organ doses at the site boundary and low population zone and 2) the pressure/temperature transient in the affected cell/building. The possibility of occurrence of any of the fires considered in this section is extremely unlikely. As such, it will be shown: 1) that the potential off-site doses are well within the guideline limits of 10CFR100, and 2) that the pressure/temperature transient does not exceed the design capability of the affected cell/building.

These fires can also result in pressure, temperature and aerosol challenge to equipment contained in the cell where the fire occurs and any connected cells. These challenges are generally mitigated by providing redundant equipment in a cell which is separate and isolated from the cell where the fire is postulated to occur. For those cases where such separation of redundant equipment is not possible, the environments resulting from sodium fires have been explicitly identified as challenges to be considered in the environmental qualification program. This includes both (1) the environment inside the cell or building in which the fire occurs and (2) the environment resulting from ingestion of the combustion products into other buildings after initial release from the plant.

The computer codes utilized in the analysis of sodium spills and fires are SPRAY-3B, GESOFIRE, SOFIRE-II, SPCA, and HAA-3B. These codes are described in Appendix A with identification of supporting references.

Sodium spills at potential locations other than those discussed in this section have been examined. However the results of these spills were considered to be less severe in terms of radiological consequences and cell temperature/pressure transients and for this reason are not presented.

Since cells containing either primary or EVST sodium are normally closed and inerted, the potential for large postulated radioactive sodium fires exists only during maintenance, when these cells are opened and deinerted, and sufficient oxygen is available to sustain combustion. A spectrum of fires, both in inerted and de-inerted atmospheres, is investigated in this section.

The consistent application of conservative assumptions throughout the analyses presented in this section provides confidence that the consequences of the fires are within the predicted results. A number of these assumptions are generic to all the fires evaluated in this section, and are summarized below:

1. The radioactive content of the sodium is based on continuous plant operation for 30 years. The design basis radioisotope concentrations were assumed present in the sodium for the accident analyses. Included in the basis and discussed in PSAR Section 11.1.5 is a design limit of 100 ppb (parts per billion) for plutonium content of the primary coolant.

2. Retention, fallout, plateout, and agglomeration of sodium aerosol in cells of buildings, whose design does not include specific safety features to accomplish that function are not accounted for in the analysis. Neglecting these factors (an assumption that all of the aerosol is available for release to the atmosphere) leads to over-prediction of potential off-site exposure.

3. No credit for non-safety related fire protection systems is taken.

4. Dispersion of aerosol released to the atmosphere was calculated utilizing the conservative atmosphere dilution factors (X/Q) applicable to discrete time intervals provided in Table 2.3-38 (the 95th Percentile Values). Guidance provided in NRC Regulatory Guide 1.145 was followed in calculating the X/Q values. Detailed descriptions of the atmospheric dilution factors estimates are provided in Section 2.3.4.

5. Fallout of the aerosol during transit downwind was neglected.

6. The cells will be structurally designed to maintain their integrity under the accident temperatures and pressures and the weight of the spilled sodium. For radiological calculations, no credit is taken for cell atmosphere leak tightness.

7. The cell liners, catch pans, and catch pan fire suppression decks are designated as Engineered Safety Features and will have design temperatures equal to or greater than the sodium spill temperature, thus confining the sodium spill.

8. Both inerted and air-filled cells will be designed to accommodate liquid metal spills resulting from a leak in a sodium or NaK pipe/component in the cell producing the worst case spill/temperature condition. The leak is based on a Moderate Energy Fluid System break (1/4 x pipe diameter x pipe thickness) as defined in branch technical positionn MEB3-1 with the sodium or NaK system operating at its maximum normal operating temperature and pressure.

9. The only credit for operator action in mitigation of postulated sodium spills is shutdown of the Na overflow system makeup pumps 30 minutes after plant scram for a postulated leak in the Primary Heat Transport System (see Section 15.6.1.4).

10. Analyses of liquid metal burning in inerted cells assumes burning of all oxygen in the cell in which the liquid metal is postulated to leak and burning of all the oxygen contained in cells which are environmentally connected to the cell with the liquid metal leak.

11. The analysis of postulated liquid metal fires in air-filled cells does not include reaction of the liquid metal with postulated water released from concrete. The validity of this approach is presently being verified in conjunction with the large scale sodium fires test program discussed in Section 1.5.2.8 of the PSAR. If the test program does not support the present analysis approach, the appropriate effects of water release from concrete will be included in subsequent analyses.

12. The Moderate Energy Fluid System (MEFS) leak is used in this section to conservatively establish the CRBRP cell structural design performance. For purposes of assessing the functional performance of the heat transport systems, a leak rate corresponding to a 4-inch crack (8 gal/min) was selected based on a detailed evaluation of the PHTS piping integrity (see Section 15.3.3.4).

Table 15.6-1 provides a summary of the initial conditions for each fire considered and the maximum off-site dose as a percentage of the 10CFR100 guideline limits. As the table indicates, a large margin exists between the potential off-site doses and 10CFR100. A discussion of the pressure/temperature transient for each event is provided in the following sections; in no case do the fires result in conditions beyond the design capability of the cell/building.

The Project is assessing the impacts of NaK spills in the Reactor Service Building and will provide the results of aerosol released from the Reactor Service Building when the assessments are completed. The aerosols released from the RSB as a result of NaK spill will be controlled so as not to affect safety-related equipment.

TABLE 15.6-1

SODIUM SPILL EVENTS

Section No.	Events	Sodium Spill Gallons	Sodium Spill Temp (F)	Atmosphere	Location* Bldg.	Cell	Max. Off-Site Dose % of 10CFR100	Max. Cell Gas Press/Temp
15.6	Sodium Spills							
15.6.1	Extremely Unlikely							
15.6.1.1	Primary sodium in containment storage tank failure during maintenance	35,000	400	Normal Air	RCB	Overflow Tank Cell	0.19	0.8 psig $138^{\circ}F$**
					Design Press 10 psig			
15.6.1.2	Failure of ex-vessel sodium cooling system during operation	15,250	600	Inerted	RSB	Ex-Vessel Sodium Tank Cell	0.48	3.8 psig $254^{\circ}F$***
					Design Press 12 psig			
15.6.1.3	Failure of ex-containment primary sodium storage tank	50,000	450	Inerted	SGB/IB	Storage Tank Cell	2.13	3.5 psig $260^{\circ}F$***
					Design Press 4 psig			
15.6.1.4	Primary Heat Transport System piping leak****	35,100 (PHTS Cell)	1015	Inerted	RCB	PHTS Cell	$<10^{-4}$	14.4 psig $680^{\circ}F$***
					Design Press 30 psig			
		29,200 (Reactor Cavity)	750	Inerted	RCB	Reactor Cavity		10.3 psig $650^{\circ}F$***
					Design Press 35 psig			
15.6.1.5	Intermediate Heat Transport System piping leak	39,000	$800^{\circ}F$	Normal Air	SGB/IB	IB	3	0.4 psig $630^{\circ}F$***
					Design Press 3 psig			

*RCB - Reactor Containment Building
RSB - Reactor Service Building
SGB/IB - Steam Generator Bldg/Intermediate Bay
PHTS - Primary Heat Transport System

** In Containment
*** In Affected Cell
****Although considered to be beyond the PHTS design basis, the MEFS leak is included in the extremely unlikely category for cell structural evaluations only. See Section 15.3.3.4 for systems effects from leaks.

15.6.1 Extremely Unlikely Events

15.6.1.1 Primary Sodium In-Containment Storage Tank Failure During Maintenance

15.6.1.1.1 Identification of Causes and Accident Description

A detailed description of this postulated event is provided in Section 6.2. Section 6.2 includes a complete discussion of the analysis methods and the calculated consequences for this event.

15.6.1.2 Failure of the Ex-Vessel Storage Tank Sodium Cooling System During Operation

15.6.1.2.1 Identification of Causes and Accident Description

There are three Ex-Vessel Storage Tank (EVST) sodium cooling circuits, two forced convection circuits normally used (alternately) to cool sodium circulated to and from the EVST, and one backup natural convection circuit used in the event the normal circuits are unavailable. Each normal circuit is located below grade in the Reactor Service Building (RSB); the backup loop is located above grade. Each cooling circuit is located in separate cells. The pump suction line for each circuit exits from the EVST at an elevation above the normal sodium level in the tank. There are internal downcomers within the EVST which extend down below the sodium level. A remotely operated isolation valve in the pump suction line for the normal cooling circuits is located slightly above the tank outlet elevation.

During operation, all the sodium cooling circuit cells are closed and inerted. The interior surfaces of the cells are protected with a steel liner, 3/8-in. thick. The cell walls are nominally 4-ft. thick concrete. The free volume of the cell is approximately 14,950 ft^3 and the cell floor area is 680 ft^2. The postulated accident is a leak in a loop pump return line in the operating normal cooling circuit in cell 337. In the event of this postulated accident, the other normal or backup cooling circuit would be brought on line to permit continued EVST cooling. The spill volume in cell 337 assumes a leak in the 4-in. EVST return line from sodium cooling Loop 1, with Loop 1 in operation. The spill volume is based on the loss of the loop inventory and pump out of EVST sodium down to the inlet of the suction piping within the EVST. The leak is essentially constant at an MEFS rate of 6 gpm; sodium temperature is assumed to be 600 degrees F.

The maximum spill postulated would require a simultaneous major piping failure plus failure of the remotely operated isolation valve (which is located in a separate environment from the spill). As such, the accident is extremely unlikely and is not expected to occur over the life of the plant.

The EVST sodium is essentially non-radioactive at the beginning of plant life. However, during refueling a small quantity of primary sodium is transferred to the EVST along with each irradiated assembly, resulting in a slow buildup of the radioactivity in the EVST sodium. For conservatism, it is assumed that the accident occurs at the end of plant life (30 years) and immediately following a refueling operation when the EVST sodium activity has reached its peak. The design basis radioisotope concentrations in the EVST sodium under these conditions are summarized in Table 15.6.1.2-1. Only those isotopes which make a significant contribution to the radiological content of the EVST sodium are included in the Table. The models and assumptions used in computing the radionuclide concentrations are included in Sections 11.1 and 12.1.3.

The Design Basis spill temperature is 600°F. The potential radiological consequences of this event are controlled by the extent of radioactive sodium aerosol formation. The aerosol formation is controlled by the limited amount of oxygen available in the inerted (2% O_2) EVST cooling equipment cell. Thus, the radiological consequences are rather insensitive to a wide range of initial sodium release (spray or pool) conditions. This is especially true because no credit was taken for retention, plate-out, or settling of the aerosol in either the EVST cooling equipment cell or the Reactor Service Building. It was conservatively assumed that all the aerosol generated during combustion was released directly to the environment.

TABLE 15.6.1.2-1

DESIGN BASIS RADIOACTIVE CONTENT OF EVST SODIUM
30 YEARS REACTOR OPERATION

Isotope	uCi/gm Sodium	Isotope	uCi/gm Sodium
Na-24	1.47E+1*	I-132	1.50E-1
Na-22	5.80E-1	Sb-125	8.04E-3
Cs-137	7.10E+0	Sr-90	2.87E-3
Cs-136	4.39E-1	Am-241	6.39E-4
Cs-134	7.10E-1	Am-242m	2.60E-5
I-131	8.90E-1	Cm-244	1.22E-4
Pu-238	6.90E-3		
Pu-239	1.86E-3		
Pu-240	2.42E-3		
Pu-241	1.63E-1		
Pu-242	5.18E-6		
H-3	1.40E-2		

*Peak activity during the fuel handling cycle.

15.6.1.2.2 Analysis of Effects and Consequences

The consequences of this postulated event were determined as follows:

a. The sodium reacts with all the available oxygen in the inerted cell (~2% O_2). The burning releases Na_2O as aerosol.

b. The radioisotope concentrations in the aerosol are the same as the initial concentrations in the sodium.

c. Radioactive decay during the accident is neglected.

d. No credit for retention, plate-out, or settling of the Na_2O aerosol in either the EVST cooling equipment cell or the RSB was taken. It was conservatively assumed that all the aerosol generated during combustion was released directly to the atmosphere.

e. Fallout of the aerosol during transit downwind was neglected.

SPRAY-3B analysis of the fire in the inerted cell indicates that combustion is completed (O_2 depleted) in less than 2 hours. A total of 45.4 kg of Na_2O, containing 33.8 kg of Na, is released to the atmosphere as a result of the postulated accident. Release during specific time intervals is as follows:

Time (hr)	Mass Na Released (kg)
0-2	15.8
2-8	17.8
>8	0.2

Even though no credit for aerosol retention in the Ex-Vessel Sodium Tank Cell was taken in the analysis, the cell pressure/temperature history was computed for an evaluation of cell integrity. The results of the analysis indicate a peak cell pressure of only 5.8 psig. This peak occurs 42 hours following the postulated spill. The cell gas pressure decreases to less than 4.3 psig after 90 hours. The cell temperature increases from nominally 135°F to 310°F in 42 hours and then decreases gradually to approximately 250°F 90 hours after the postulated spill.

The radiological assessment was performed utilizing a total sodium oxide aerosol release approximately 10% lower then that indicated above, and a 0-2 hour release approximately 20% lower than that indicated above. The results of the radiological assessment are provided in Table 15.6.1. 2-2. The radiological assessment was performed utilizing atmospheric dispersion factors (X/Q) in Chapter 2 of the PSAR.

15.6.1.2.3 Conclusions

The calculated transient cell pressures and temperatures are within the design pressure and temperatures. The offsite radiological consequences are small fractions of the 10 CFR 100 guidelines.

TABLE 15.6.1.2-2

POTENTIAL OFF-SITE DOSES FOLLOWING FAILURE OF THE
EVST COOLING SYSTEM

Organ	10CFR100	Dose (Rem) SB (0.42 mi) 2-hours	LPZ (2.5 mi) 30-days
Whole Body**	25	2.59 E-2	5.31 E-3
Thyroid	300	2.20 E-2	4.52 E-3
Bone	150+	7.13 E-1	1.46 E-1
Lung	75+	3.51 E-2	7.20 E-3

*2.59 E-2 = 2.59 x 10^{-2}
+Not covered in 10CFR100; used as guideline values.
**Includes both inhalation and external gamma cloud exposure.

15.6.1.3 Failure of an Ex-Containment Primary Sodium Storage Tank

15.6.1.3.1 Identification of Causes and Accident Description

The two ex-containment primary sodium storage tanks are located in a cell (cell 211) on the lowest level of the Intermediate Bay of the Steam Generator Building. These tanks will be used to store primary sodium only in the event maintenance requires the complete drainage of more than 1 PHTS loop or the EVST or maintenance is required in cell 102A. The postulated accident is the complete failure of one of the tanks, when full, which results in the complete spill of the contained sodium to the cell floor. This postulated accident is extremely unlikely.

When the ex-containment sodium tanks are full of liquid sodium, access to the tank cell is prohibited due to the sodium activity and the cell is closed and inerted (~2% O_2). The cell floor area is approximately 2400 ft^2 and the free volume of the cell is 55,700 ft^3. The floor of the cell is protected with a Engineered Safety Feature steel catch pan, 3/8 inch thick. The sides of the catch pan extend vertically upward to a height such that the maximum potential spill volume can be safely contained within the catch pan.

For conservatism, the postulated accident is assumed to occur near the end of plant life (30 years) when the radioactive content of the primary sodium has potentially reached its peak. A minimum of 10 days decay time, in-containment, is required prior to charging an ex-containment storage tank, to insure substantial decay of Na-24.

The postulated accident results in the spill of 50,000 gallons (333,000 lbs.) of 450°F sodium to the cell floor. This spill represents 100% of the contained volume of one of the two tanks and is an extremely conservative upper bound. The total postulated spill is contained by the catch pan.

15.6.1.3.2 Analysis of Effects and Consequences

The consequences of this postulated event were determined as follows:

a. The spilled sodium reacts with the available oxygen (2%) in the cell, burns and releases 27% of the Na_2O formed as airborne particles (Reference 3).

b. The radioisotope concentrations in the aerosol are the same as the initial concentrations in the sodium.

c. Radioactive decay during the accident is neglected.

d. No credit for retention, plate-out, or settling of the aerosol in either the ex-containment storage tank or the Steam Generator Building was taken. It was conservatively assumed for radiological evaluations that all the aerosol generated during combustion was released directly to the environment.

e. Fallout of the aerosol during transit downwind was neglected.

SOFIRE-II analysis of the fire in the inerted cell indicates that combustion is completed (less than 0.1 gram of oxygen remaining in cell) in approximately 8 hours. A total of 37.6 Kg of Na_2O, containing 27.7 Kg of Na, is released to the atmosphere as a result of the postulated accident. Release during specific time intervals is as follows:

Time (hr)	Mass Na Released (kg)
0-2	27.4
2-8	0.3
>8	0

Even though no credit for aerosol retention in the ex-containment storage tank cell was taken in the analysis, the cell pressure/temperature history was computed for an evaluation of cell integrity. The results of the analysis indicate a peak cell pressure of approximately 3.5 psig. This peak occurs 1.2 hours following the postulated spill. The cell pressure then decreases to about 1.3 psig 2 days after the spill. The cell temperature increases from nominally 100°F to 260°F at 1.3 hours and then decreases gradually to 160°F after 2 days. There are no safety related systems or components identified within or with an environment common to the accident cell.

The potential radiological consequences of this failure are conservatively assessed assuming instantaneous release of 90,000 gallons of sodium. This is approximately twice the amount of sodium which could spill, even in the event of total failure of one of the storage tanks. The following provides the rationale for using SOFIRE-II for the initial stage of this event.

The sodium stored in the tank will be essentially a stagnant pool under very low cover gas pressure and thus any foreseeable failure, such as the cracking of a fill or drain line, would result in only a slow leakage of sodium and a spill orders of magnitude less than the assumed 90,000 gallons. Further, the tank is mounted near the floor level of the tank cell, with a floor clearance of approximately 3 feet. The proximity of the tank to the cell floor coupled with the low operating pressure of the tank precludes releases, characteristic of sodium sprays. Because of the design, location, and operating pressure of the tank, there is no mechanism leading to the pressurized discharge of sodium from the tank. However, the potential consequences of a postulated tank failure were assessed assuming the immediate release of the total tank inventory and thus the SOFIRE-II pool fire analysis presented in this Section is judged to conservatively bound the consequences of postulated tank failures.

15.6.1.3.3 Conclusions

The potential 2-hour, site boundary and 30-day, LPZ whole body and organ doses resulting from this postulated accident are summarized in Table 15.6.1.3-1.

The most limiting doses in this case result from 2-hour exposure at the site boundary. Even with the extremely conservative assumptions used in the analysis, large margins exist between the potential doses and the applicable guideline limits (factor of 45 for bone and 100 for all other doses). The nominal catch pan design temperature of 1000°F and cell design pressure of 4 psig are not exceeded for this event.

TABLE 15.6.1.3-1

POTENTIAL OFF-SITE DOSES FOLLOWING FAILURE OF EX-CONTAINMENT NA STORAGE TANK

		Dose (Rem)	
Organ	10CFR100	SB(0.42 mi) 2-hour	LPZ (2.5 mi) 30 days
Whole Body**	25	2.38E-1	3.83E-2
Thyroid	300	8.85E-1	1.42E-1
Bone	150+	3.19E+0	5.11E-1
Lung	75+	1.77E-1	2.84E-2

*2.38E-1 = 2.38 x 10^{-1}
+Not covered in 10CFR100; used as guideline values.
**Includes both inhalation and external gamma cloud exposure.

15.6.1.4 Primary Heat Transport System Piping Leaks

15.6.1.4.1 Identification of Causes and Accident Description

Small sodium leaks have occurred a number of times in sodium testing facilities and in operating reactors. Consequently, PHTS leaks are considered in the design and evaluation of the plant to assure that the design has adequate capability from the standpoints of integrity of the cell liners and long term integrity of the cell structures.

A large leak in the primary piping could only occur if a large through-wall defect in the primary piping were to occur. Reference 2 of Section 1.6 provides details demonstrating that the possibility of such a large leak is extremely small. The low probability of such a large pipe leak is based on the following lines of defense:

1. Stringent Quality Assurance measures applied during design, fabrication and construction will minimize the possibility that significant defects will exist in the primary heat transport system boundary. A defect in piping larger than about 1 inch long and about 10 mils deep will be detected with subsequent rejection of the subject piping.

2. Fracture mechanics analyses supported by tests show that even if a defect existed which was orders of magnitude larger than the allowable size, it would not grow significantly over the life of the plant. These analyses consider all stresses associated with the duty cycle.

3. Analyses and tests show that even if a very large defect in a pipe is postulated, and then subjected to much more extreme conditions than those imposed by the duty cycle, the defect would grow and penetrate the pipe wall before the length would reach the critical size. A through-wall crack provides a sodium leakage path and leak detection systems are provided to monitor sodium leakage. Corrective action will be taken upon detection of any primary sodium leak.

The CRBRP will have several diverse leak detection methods which provide adequate and diverse detection capability. The performance of these systems is shown in a preliminary quantitative way and is summarized in Section 7.5.5. At least two leak detection methods will be available to detect leaks. The leak detection methods include a wide range of diversity and sensitivity. They assure that very small leaks (as low as 100 gm/hr) would be detected in less than 250 hours or, for larger leaks, before a total spill volume of 150 gal. of sodium occurred.

A Cell Design Basis Leak was chosen to conservatively envelop the maximum undetectable leak (100 gm/hr) and the maximum leak believed possible (that from a 4 inch crack-see Reference 2 of Section 1.6). The leak chosen is that with flow equivalent to the flow from a sharp edged circular orifice with area equal to one half the pipe diameter times one-half the pipe wall thickness. This leakage rate is that for piping with low internal pressure specified by NRC Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment." The leak rates for the PHTS cells and the Reactor Cavity (RC) range from an initial flow rate of 947 gpm and 847 gpm to a low of 58 gpm respectively. The details of these leaks are presented in Table 15.6.1.4-1. The Design Basis Leaks chosen provide a large margin between detectability and the leak rates selected for evaluation purposes.

The design requirement for the cell thermal transient is that there should be no damage to the cell that could impair the safety functions of other equipment (e.g., other HTS loops or containments). Reactor decay heat would be removed by the 2 unaffected PHTS loops and transported to the environment via the IHTS, SGS and either the turbine bypass or SGAHRS.

15.6.1.4.2 Analysis of Effects and Consequences

Analyses have been performed to determine the pressure and thermal transients imposed on the PHTS cells and the Reactor Cavity as a result of the Design Basis Leaks. The leak rates as a function of time for each cell are presented in Table 15.6.1.4-1.

The computer codes SPRAY and SOFIRE-II were used in combination to evaluate the cell transients. Resulting temperatures and pressures during the leakage discharge phase of the accident were obtained with the SPRAY code. Temperature and pressure boundary conditions determined by the SPRAY code at the termination of leakage are used as initial conditions in the SOFIRE-II code in determining long term accident conditions. The material properties used in the SPRAY and SOFIRE-II analyses are itemized in Table 15.6.1.4-2.

Summary of Results

Table 15.6.1.4-3 and Figures 15.6.1.4-1 through -12 present the results of the analyses. For each cell the following peak transient values are itemized in Table 15.6.1.4-3: gas pressure, gas temperature, floor structural concrete temperature, and the wetted wall structural concrete temperature. The concrete temperatures provided in the table represent the temperatures in the first one and one-half inch of structural concrete behind the floor and the wall insulating concrete.

Two general observations based on the data summarized in Table 15.6.1.4-3 and Figures 15.6.1.4-1 through -12 follow:

- Cell atmosphere pressures and temperatures peak during the period of high sodium leakage and decrease at the time of pump shutdown, when the sodium leak flow rate decreases.

- The peak structural concrete temperatures in the floor and the wall occur much later in time, during the pool phase of the analysis.

Leaks Within the Primary Heat Transport System (PHTS) Cell

The design basis sodium leak for evaluating the structural capabilities of the PHTS cell is identified in Table 15.6.1.4-1. The location of the leak is from the point in the hot leg outside any guard vessel resulting in the maximum spill to the cell floor. The spill is contained by the (ESF) Engineered Safety Feature cell liner. The cell has a free cell volume of 95720 ft^3 and a floor area of 1660 ft^2. The cell wall thickness is approximately 5.1 ft. and the floor thickness is 5.5 ft.

The total duration of the leak is 502 minutes and the total sodium injected into the cell is approximately 35,000 gallons; 12% of this sodium is injected in the first 4.5 minutes.

The initial flow rate of 947 gpm is maintained until the Plant Protection System (PPS) reacts on a reactor low level sodium trip in 4.5 minutes which shuts down the primary pumps. Pump coast down lasts for one minute followed by pony motor system flow for approximately 500 minutes. Reactor vessel makeup flow is assumed shut off by operator action within 30 minutes after the low level scram. This reduces the total spill volume by limiting the amount of sodium transferred from the overflow vessel to the reactor vessel. The complete time history of the leak is presented in Table 15.6.1.4-1. The sodium temperature is assumed to be constant at 1015°F. The cell pressure was assumed to be atmospheric in calculating the leak rates. This is conservative since the positive cell pressure generated by the initial leakage will reduce the leakage.

An analysis of the consequences of this leak during the time of sodium flow was conducted with the SPRAY Code (Appendix A.85). Over this period, all the sodium injected was assumed to be in the form of a spray (0.18" dia. droplets) occupying one-third of the cell volume. The peak cell temperature (680°F) and the peak cell pressure (14.4 psig) occur at 4.5 minutes, the end of the maximum discharge rate. Figures 15.6.1.4-1 and -2 present the results of the spray phase transient analysis.

The long term transients following the termination of the sodium flow are evaluated with the SOFIRE-II code. The initial conditions for this SOFIRE-II analysis are taken from the SPRAY analysis at the end of the MEFS leak.

Figures 15.6.1.4-3 through -6 present the long-term cell transients based on the SOFIRE-II analysis.

Leaks Within the Reactor Cavity

The design basis sodium leak for evaluating the structural capabilities of the Reactor Cavity is defined to have a total duration of 400 minutes and the total sodium injected into the cavity is approximately 29,000 gallons; 16% of this sodium is injected in the first 5.9 minutes. The Reactor Cavity leak originates from a cold-leg piping fault and the temperature of the sodium injected is 750°F, the peak cold-leg sodium temperature. The same assumptions and analysis procedure as described for the PHTS Cell leak were used to evaluate this leak. The complete time history of the leak is presented in Table 15.6.1.4-1.

The leak is assumed to be located in the cold leg outside the guard vessel resulting in the maximum spill to the cavity floor. The sodium spill volume is contained by the ESF cell-liner. The Reactor Cavity has a free cell volume of 55600 ft^3 and a floor area of 1260 ft^2. The cell wall thickness is approximately 7 ft. and the floor thickness is 25.7 ft.

Figures 15.6.1.4-7 and -8 present the results of the spray-phase transient analysis. The peak cavity atmosphere pressure and temperature, 10.3 psig and 650°F, occur at approximately 2.2 minutes. The pressure and temperature decline gradually out to 4.9 minutes, the end of the maximum leak rate (847 gpm), and begin to decrease sharply at 4.9 minutes when the leak rate decreases first from 491 gpm to 128 gpm and finally to 58 gpm. Figures 15.6.1.4-9 through -12 present the long-term cavity transients based on the SOFIRE-II analysis. The peak structural concrete temperatures are presented in Table 15.6.1.4-3.

Radiological Analysis

The radiological analysis is based on the PHTS cell sodium leak rate time history discussed in Table 15.6.1.4-1. The potential off-site doses from a leak in the Reactor Cavity are estimated to be 25% (based on quantity of sodium aerosols produced) less than those associated with a postulated leak in the PHTS cell.

The potential radiological consequences of a leak in a PHTS were assessed using the following approach:

1. The leak history was analyzed as a spray fire using the SPRAY code for the complete duration of the spill (502 minutes). Long term effects were then determined with the SOFIRE-II code using as input boundary conditions the results from the SPRAY code analysis. A total of 423 pounds of sodium burn in reacting with essentially 100% of the available oxygen in the PHTS cell. Approximately 570 pounds of sodium aerosol (Na_2O) result from the burning of the sodium.

2. No credit for retention, plate-out, or settling of the sodium aerosol, generated during combustion, in the PHTS cell was taken. It was conservatively assumed that all the aerosol generated during combustion was released directly to the upper containment atmosphere.

3. Release of airborne aerosol from the containment building to the atmosphere is prevented by automatic containment isolation following detection of high radioactivity in the RCB HVAC exhaust.

4. Aerosol leakage from containment was computed based on the design leak rate of the RCB with an assumed pressure of 0.5 psig. Prior to containment isolation, the RCB is assumed to be vented at 14,000 CFM. Since the postulated accident does not produce an appreciable pressure in the RCB, the 0.5 psig pressure was conservatively assumed after containment isolation for the duration of the accident and containment leakage was computed applying a square-root pressure relationship for leakage. A leak rate of 0.22% Vol/day was calculated based on the design leak rate for containment of 0.1% Vol/day at 10 psig.

5. Aerosol leakage from containment after containment isolation will be into the annulus which is vented via a filtration and recirculation system. 1% of the containment leakage was assumed to bypass the filters and leak directly to the environment.

6. The radioisotope concentrations in the aerosol are the same as the initial concentrations in the sodium. For conservatism, the radioactive content of the sodium was based on continuous plant operation for 30 years. The design basis radioisotope concentrations, at zero (0) days decay, were assumed present in the sodium. The radioisotope concentrations of the sodium under these conditions are provided in Table 15.6.1.4-4.

7. Radioactive decay during the accident was conservatively neglected.

8. Fallout of the aerosol during transit downwind was conservatively neglected.

Based on the above assumptions, HAA-3 (Reference Appendix A) was used to determine the time behavior of the aerosol in the Reactor Containment Building including leakage to the atmosphere.

A total of 3.6 grams of Na is released to the atmosphere over a 30-day period. Release during specific time intervals is as follows:

Time (hrs)	Mass Na Released (gm)
0-2	1.0
2-8	0.8
8-24	1.2
24-96	0.5
>96	0.1

Aerosol leakage beyond 4 days (96 hours) is insignificant. The suspended aerosol concentration in the RCB at 4 days is 2.4×10^{-2} ugm/cc. At the conservatively assumed containment leak rate of 0.22% Vol/day after containment isolation, the total quantity of sodium leaked per day is less than 0.2 grams. In addition the aerosol concentration, and consequently the potential leakage to the atmosphere, continues to decrease with time beyond 4 days.

15.6.1.4.3 Conclusions

The Reactor Cavity and PHTS Cell pressure and temperature transients resulting from their respective design basis leaks are within the design capabilities for those structures.

The radiological consequences of primary pipe leak events are small fractions of 10 CFR 100 guideline values as shown in Table 15.6.1.4-5.

Amend. 64
Jan. 1982

TABLE 15.6.1.4-1

SUMMARY DESCRIPTION OF PIPING LEAK EVALUATED

Parameter	Fluid System Leak Data
PHTS Cell	
Discharge	4.5 Min. @ 947 GPM 1.0 Min. @ 542 GPM 10.0 Min. @ 129 GPM 19.0 Min. @ 119 GPM 467.5 Min. @ 58 GPM
Duration	502 Minutes
Total Spill	35100 Gallons
Na Temperature	1015°F
Reactor Cavity	
Discharge	4.9 Min. @ 847 GPM 1.0 Min. @ 491 GPM 10.0 Min. @ 128 GPM 19.0 Min. @ 119 GPM 361.1 Min. @ 58 GPM
Duration	396 Minutes
Total Spill	29200 Gallons
Na Temperature	750°F

Amend. 64
Jan. 1982

TABLE 15.6.1.4-2

MATERIAL PROPERTIES USED FOR THE SPRAY-SOFIRE ANALYSES

Material	Density lb./ft.^{3}	Thermal Conductivity Btu/hr.ft.^{3} ^oF/ft.	Specific Heat Btu/lb. ^oF
Steel	490	25	0.12
Liner Air Gap*	0.037	0.0135	0.6
Light Weight Insulating Concrete	65	0.4 at 100^oF to 0.21 at 400^{oo}	0.4 at 100^oF to 0.26 at 400^oF
Structural Concrete	147	1.72 at 100^oF to 1.06 at 300^oF	0.27 at 100^oF to 0.24 at 300^oF
Ambient Air	0.0709	0.0135	0.17**

*Air Gap properties based on saturated steam at 1 atm and 212^oF.
**Specific Heat at Constant Volume.

TABLE 15.6.1.4-3

SUMMARY RESULTS FOR SODIUM LEAKS INTO THE
PHTS CELLS AND THE REACTOR CAVITY

Peak Transient Values

	Gas Pressure psig	Gas Temperature °F	Floor Structural Concrete Temperature °F	Wetted Wall Structural Concrete Temperature °F
PHTS Cells	14.4	680	190	200
Reactor Cavity	10.3	650	200	210

Amend. 64
Jan. 1982

TABLE 15.6.1.4-4

DESIGN BASIS RADIOACTIVE CONTENT
OF PRIMARY SODIUM COOLANT
30 YEARS REACTOR OPERATION

	μCi/gm Sodium Days After Shutdown			μCi/gm Sodium Days After Shutdown	
ISOTOPE	0	10	ISOTOPE	0	10
Na 24	2.94E+4*	4.32E-1	Te 127	2.48E-1	1.95E-1
Na 22	3.49E+0	3.46E+0	Te 127m	2.08E-1	1.95E-1
Rb 86	2.00E+0	1.38E+0	La 140	6.54E-2	3.80E-2
Cs 137	8.42E+1	8.42E+1	Ce 141	7.75E-2	6.26E-2
Cs 136	1.74E+1	1.05E+1	Ce 144	4.59E-2	4.48E-2
Cs 134	1.07E+1	1.06E+1	Pr 144	4.59E-2	4.48E-2
Sb 125	4.83E-1	4.80E-1	Pr 143	5.49E-2	3.30E-2
I 131	4.97E+1	2.10E+1	Nd 147	2.57E-2	1.38E-2
Te 132	3.53E+0	4.16E-1	Pm 147	2.57E-2	2.55E-2
I 132	3.35E+1	3.95E+0	Pu 238	1.60E-2	1.60E-2
Te 129m	7.18E-1	5.86E-1	Pu 239	4.24E-3	4.24E-3
Te 129	7.18E-1	5.86E-1	Pu 240	5.54E-3	5.54E-3
Sr 89	1.10E-1	9.60E-2	Pu 241	4.60E-1	4.59E-1
Sr 90	6.80E-2	6.80E-2	Pu 242	1.18E-5	1.18E-5
Y 90	6.80E-2	6.80E-2	Np 238	4.91E-6	1.80E-7
Y 91	3.13E-2	2.78E-2	Np 239	1.58E-2	8.23E-4
Zr 95	5.83E-2	5.24E-2	Am 241	1.64E-3	1.64E-3
Nb 95	5.83E-2	5.24E-2	Am 242m	6.46E-5	6.46E-5
Ru 103	8.31E-2	6.98E-2	Am 242	7.39E-5	6.46E-5
Ru 106	5.75E-2	5.64E-2	Am 243	2.64E-5	2.64E-5
Rh 106	5.75E-2	5.64E-2	Cm 242	1.20E-3	1.15E-3
Sb 127	3.65E+0	5.85E-1	Cm 243	1.59E-5	1.59E-5
Ba 140	6.54E-2	3.80E-2	Cm 244	3.32E-4	3.32E-4
			H3	2.34E+0	2.34E+0

*2.94E+4 = 2.94×10^4

TABLE 15.6.1.4-5

POTENTIAL OFF-SITE DOSES

Organ	10CFR100	Dose (rem) SB (0.2 mi) 2-hour	LPZ (2.5 mi) 30 days
Whole Body**	25	9.89 E-5	1.97 E-5
Thyroid	300	8.30 E-5	1.64 E-5
Bone	150+	1.12 E-4	2.20 E-5
Lung	75+	3.27E-5	6.44 E-6

*9.89 E-5 = 9.89 x 10^5
+Not covered in 10CFR100; used as guideline values.
**Includes both inhalation and external gamma exposure.

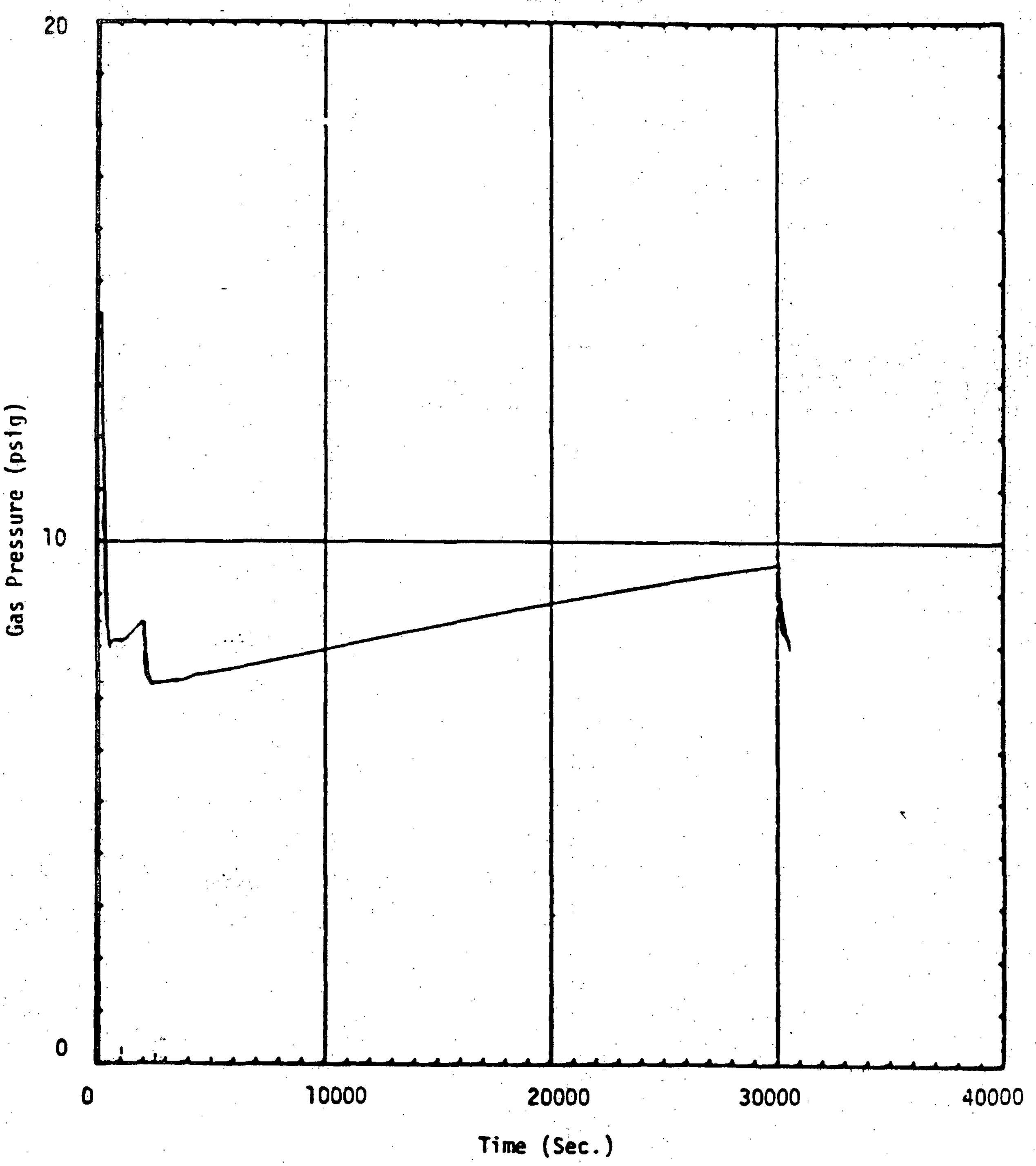

Figure 15.6.1.4-1. PHTS Cell Gas Pressure (SPRAY Phase)

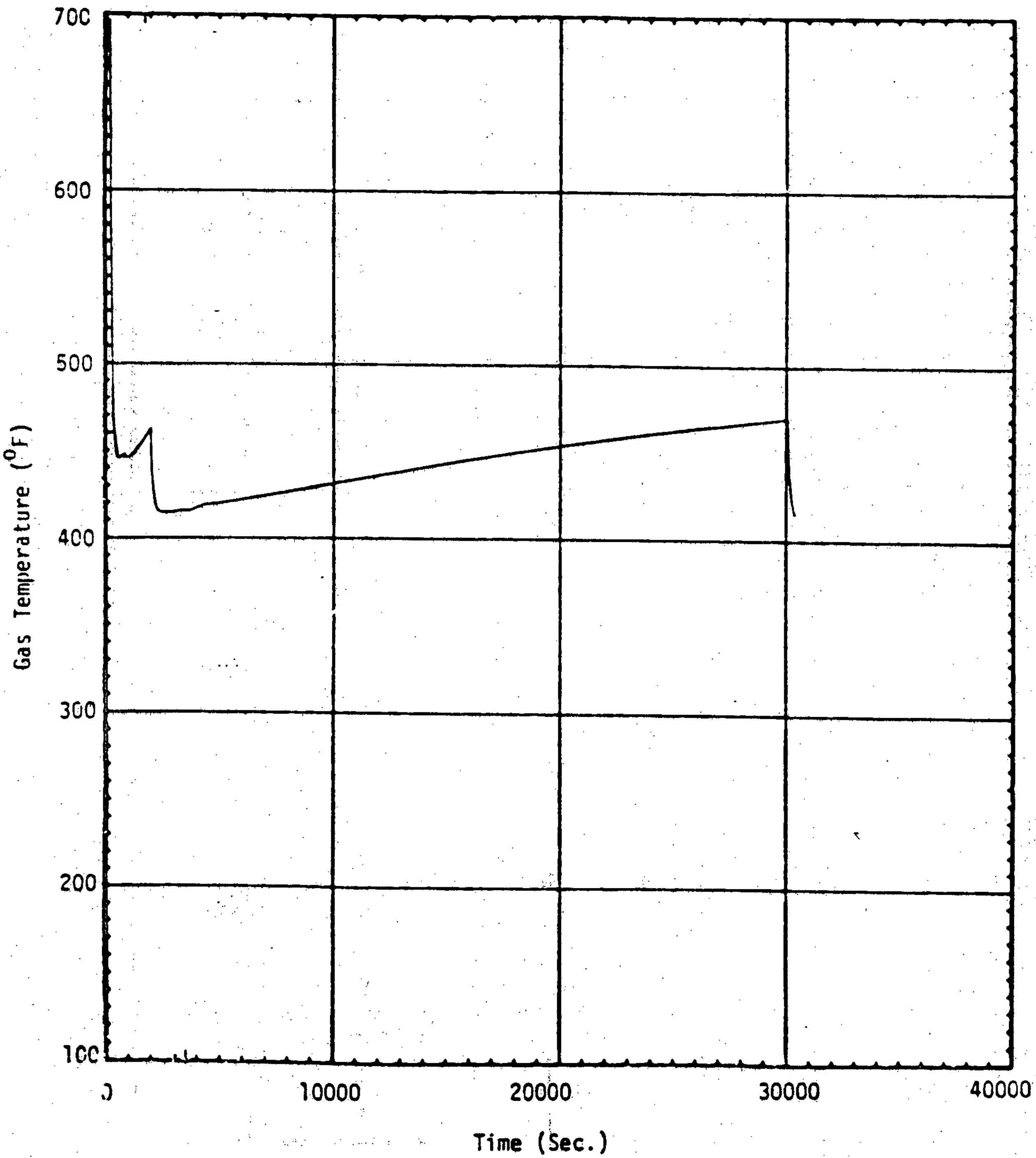

Figure 15.6.1.4-2. PHTS Cell Gas Temperature (SPRAY Phase)

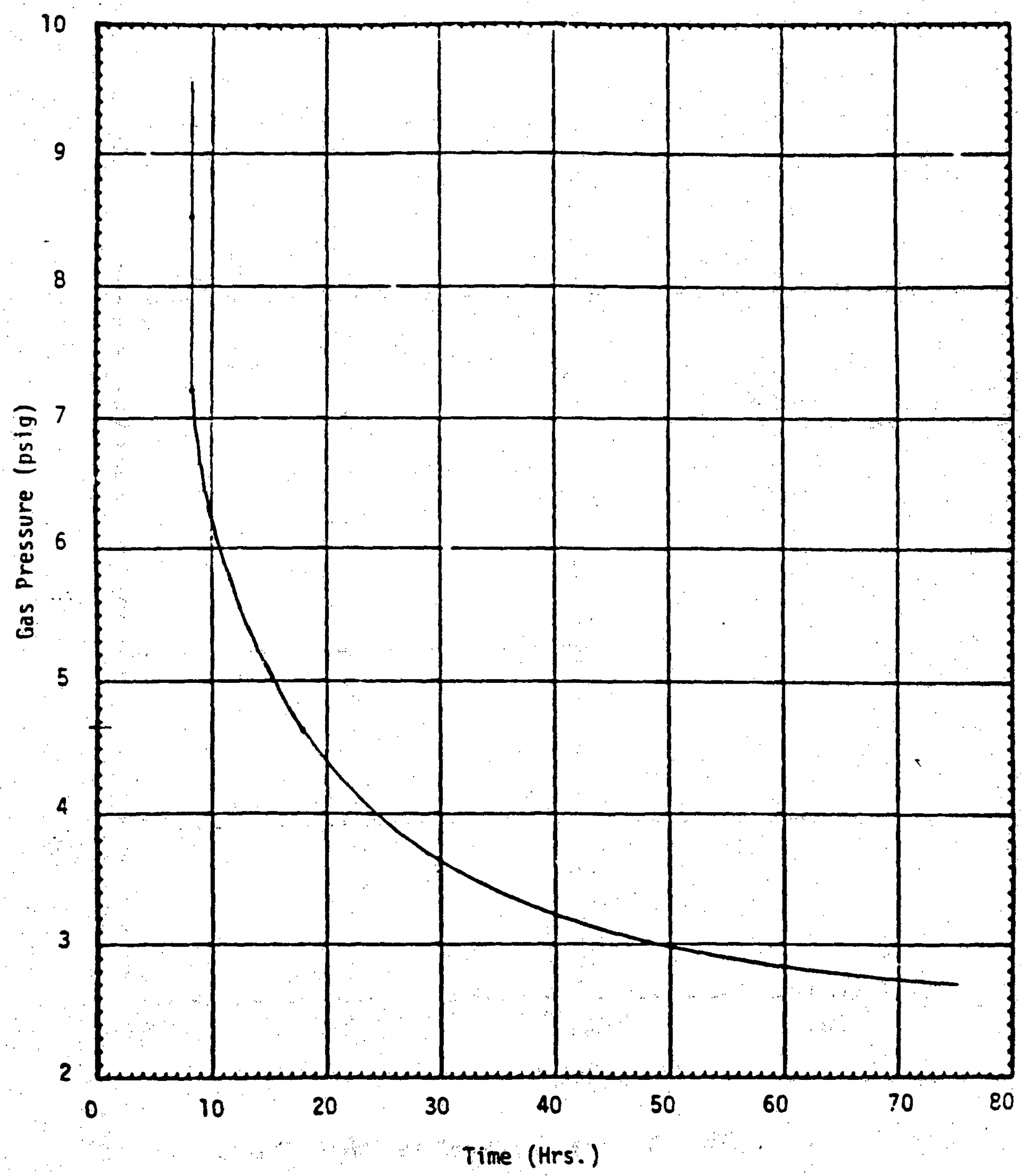

Figure 15.6.1.4-3. PHTS Cell Gas Pressure (SOFIRE Phase)

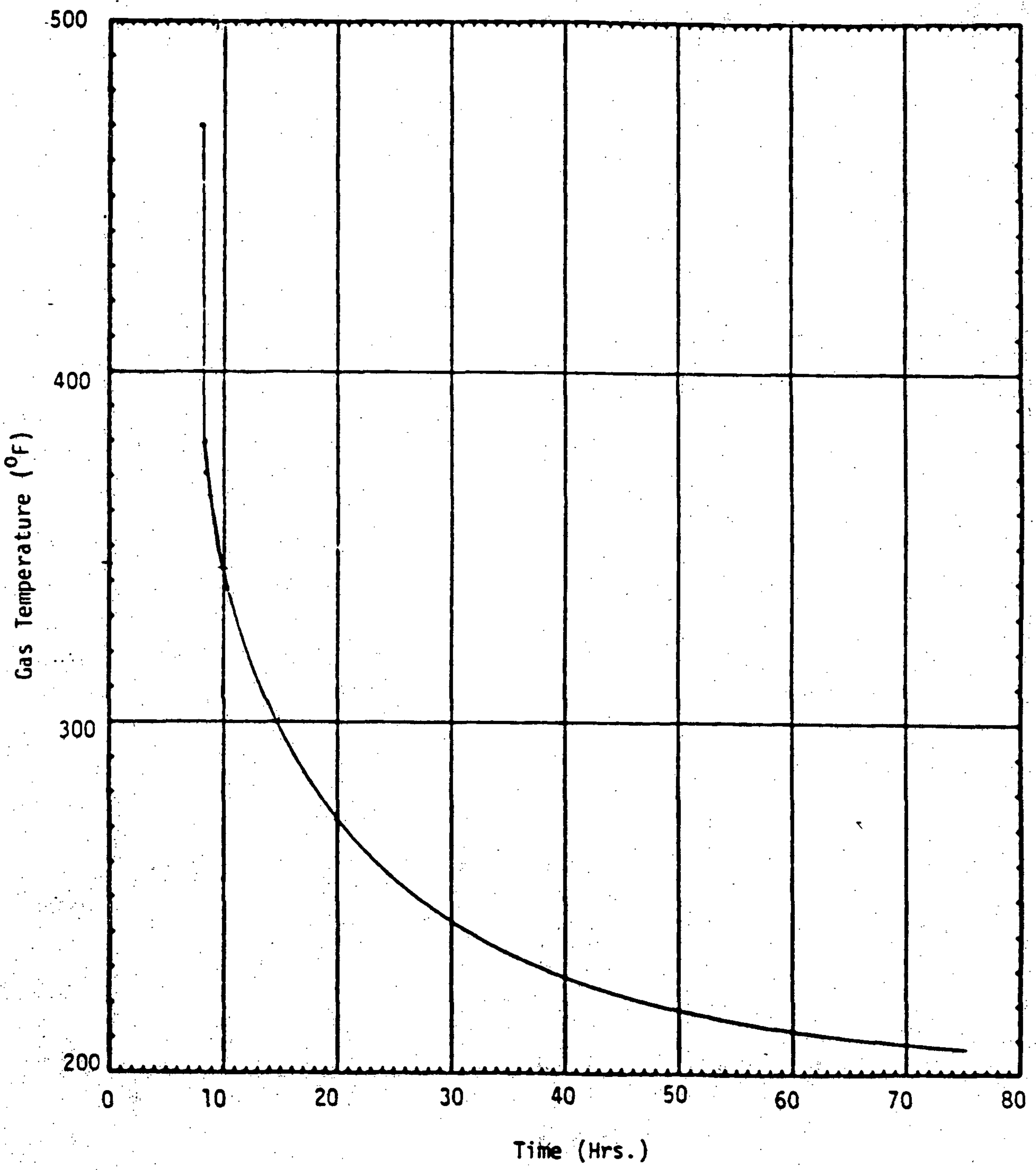

Figure 15.6.1.4-4. PHTS Cell Gas Temperature (SOFIRE Phase)

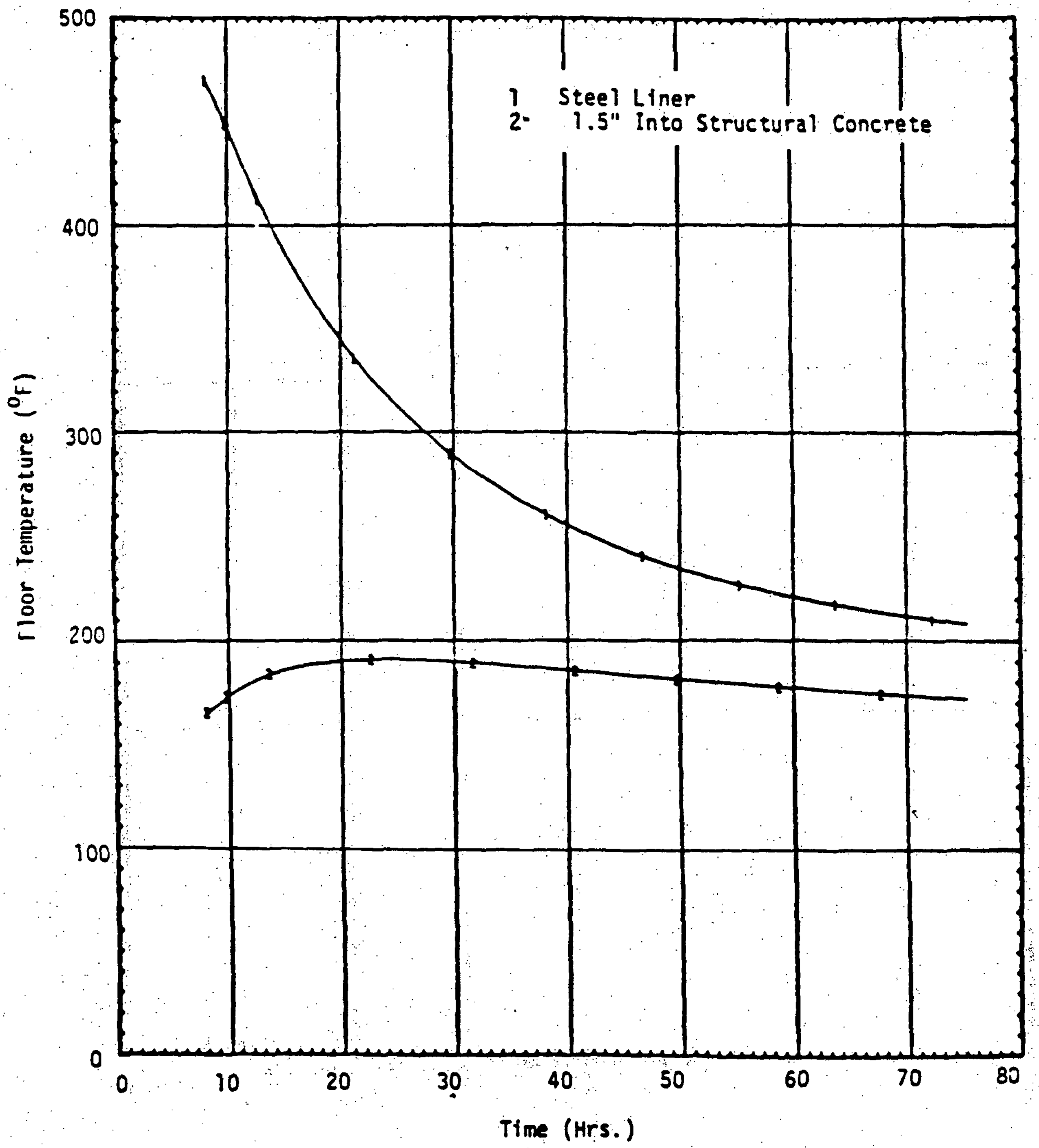

Figure 15.6.1.4-5. PHTS Cell Floor Temperature

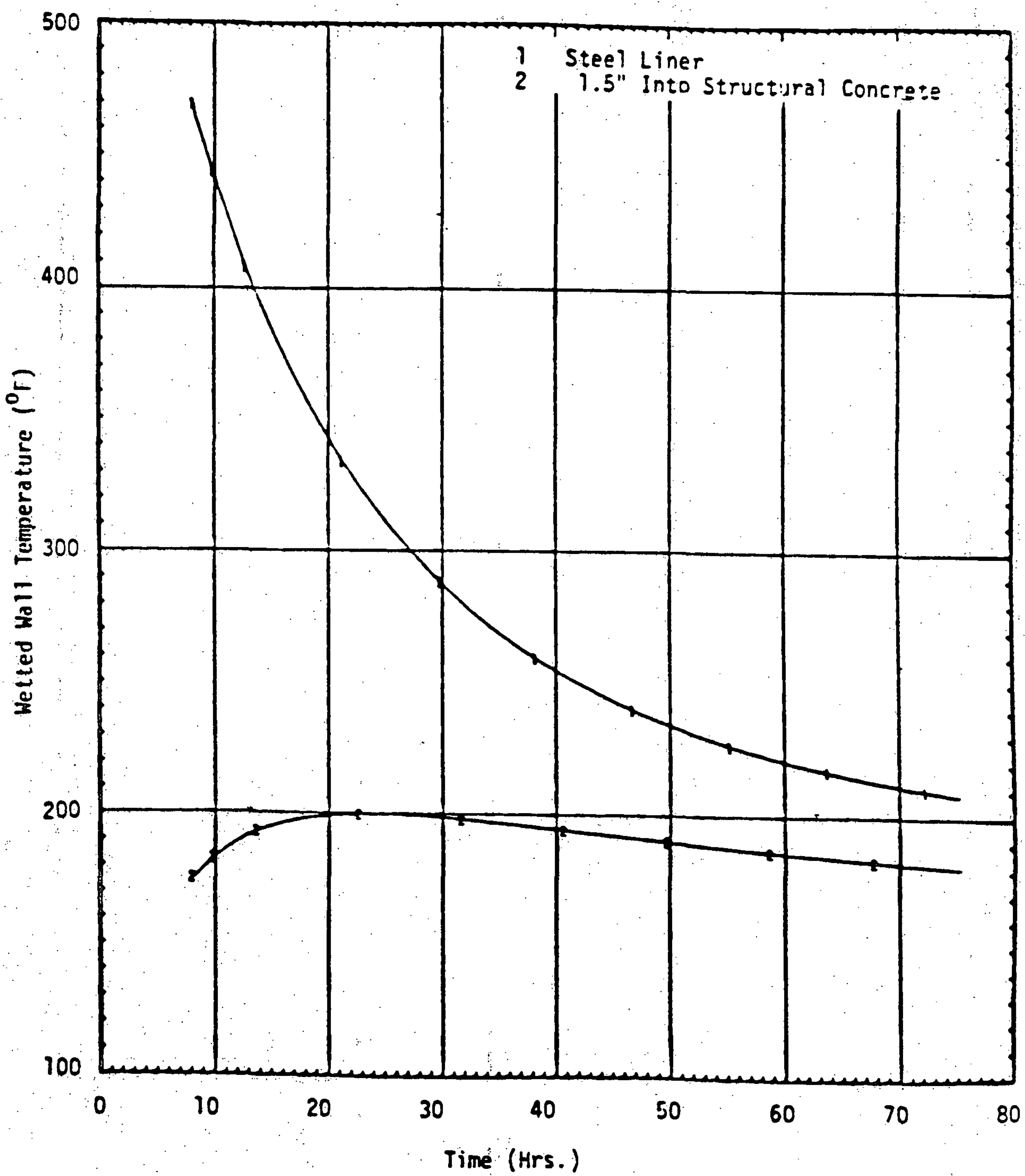

Figure 15.6.1.4-6. PHTS Cell Wall Temperature

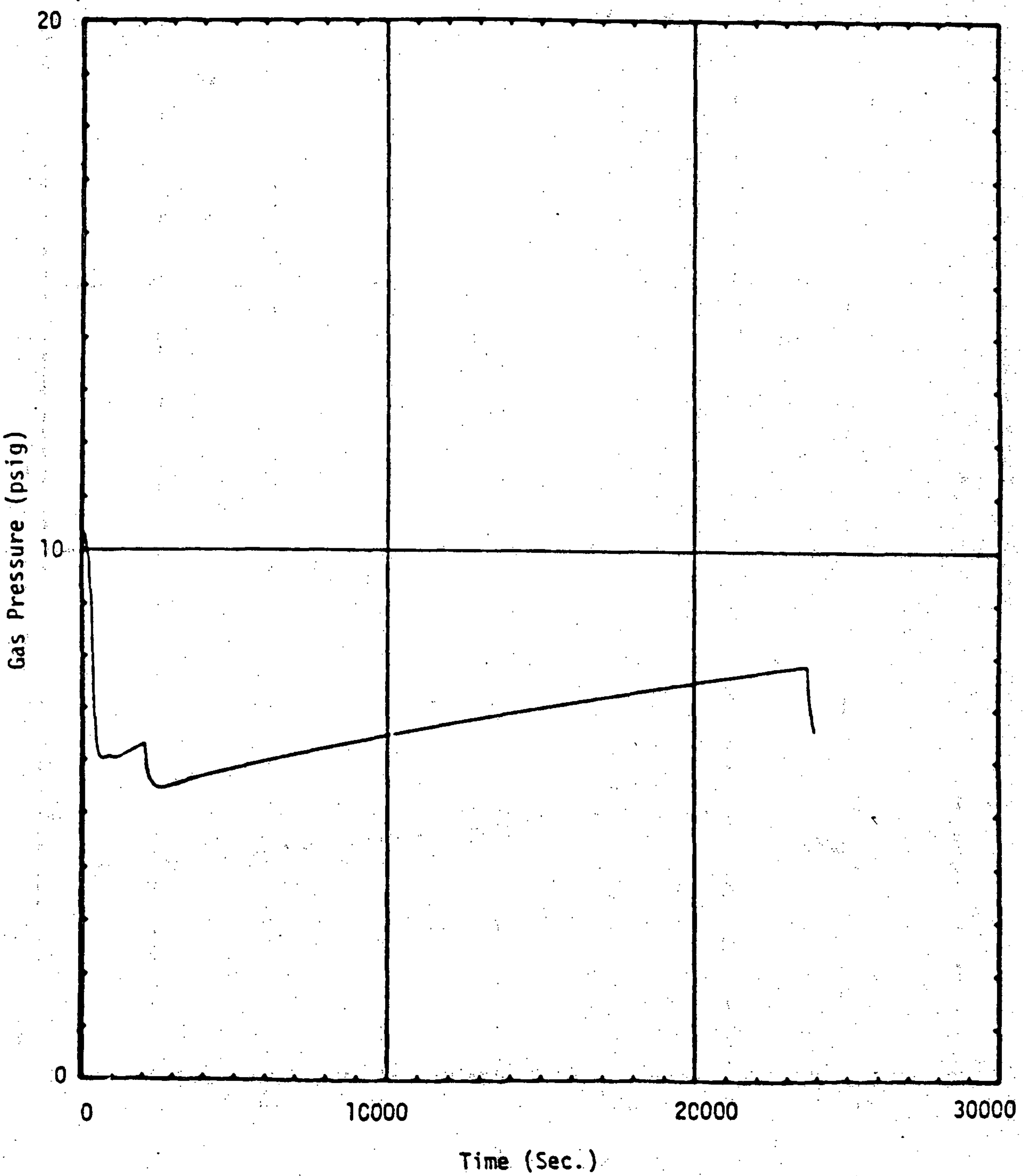

Figure 15.6.1.4-7. Reactor Cavity - Gas Pressure (SPRAY Phase)

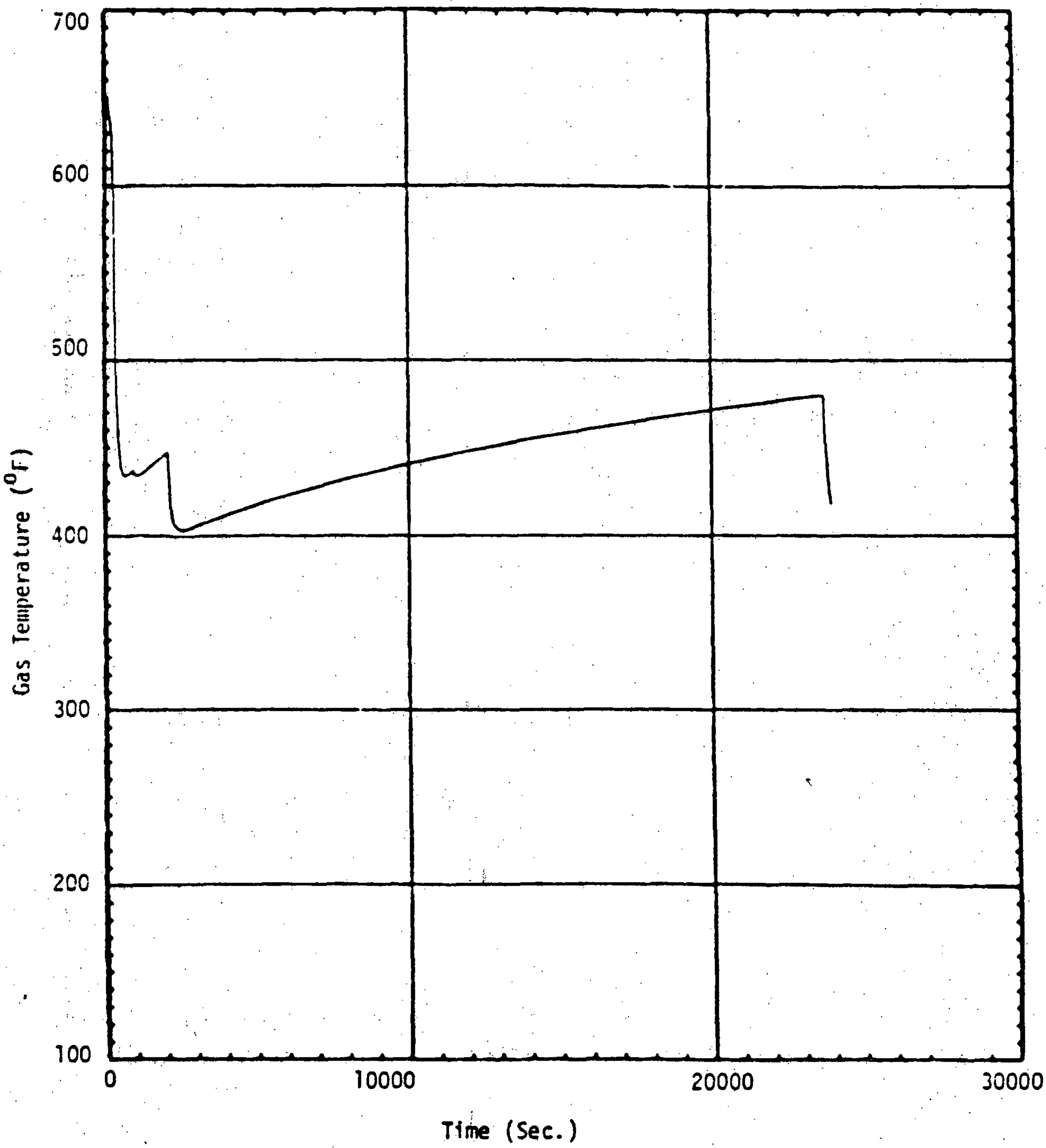

Figure 15.6.1.4-8. Reactor Cavity - Gas Temperature (SPRAY Phase)

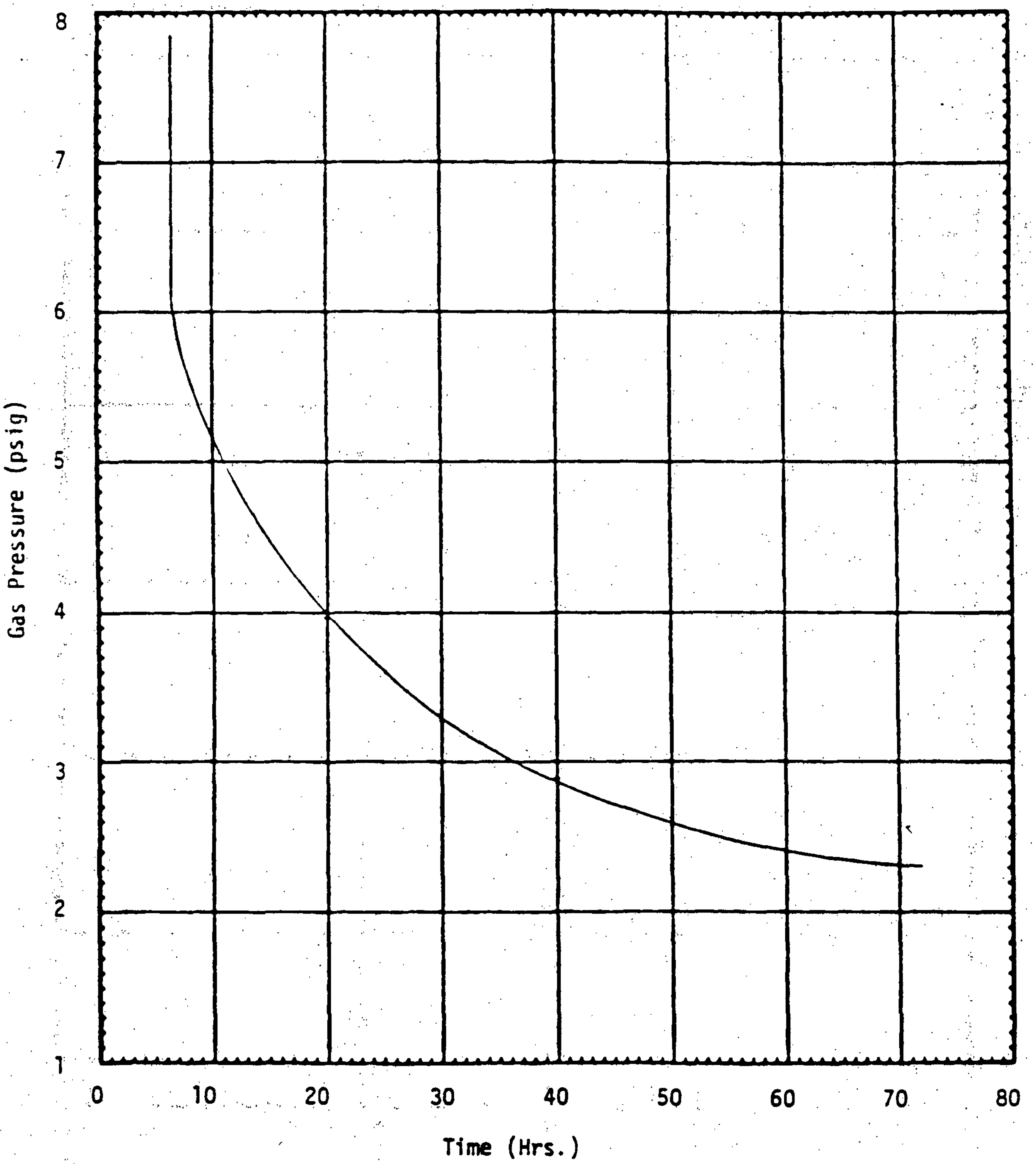

Figure 15.6.1.4-9. Reactor Cavity - Gas Pressure (SOFIRE Phase)

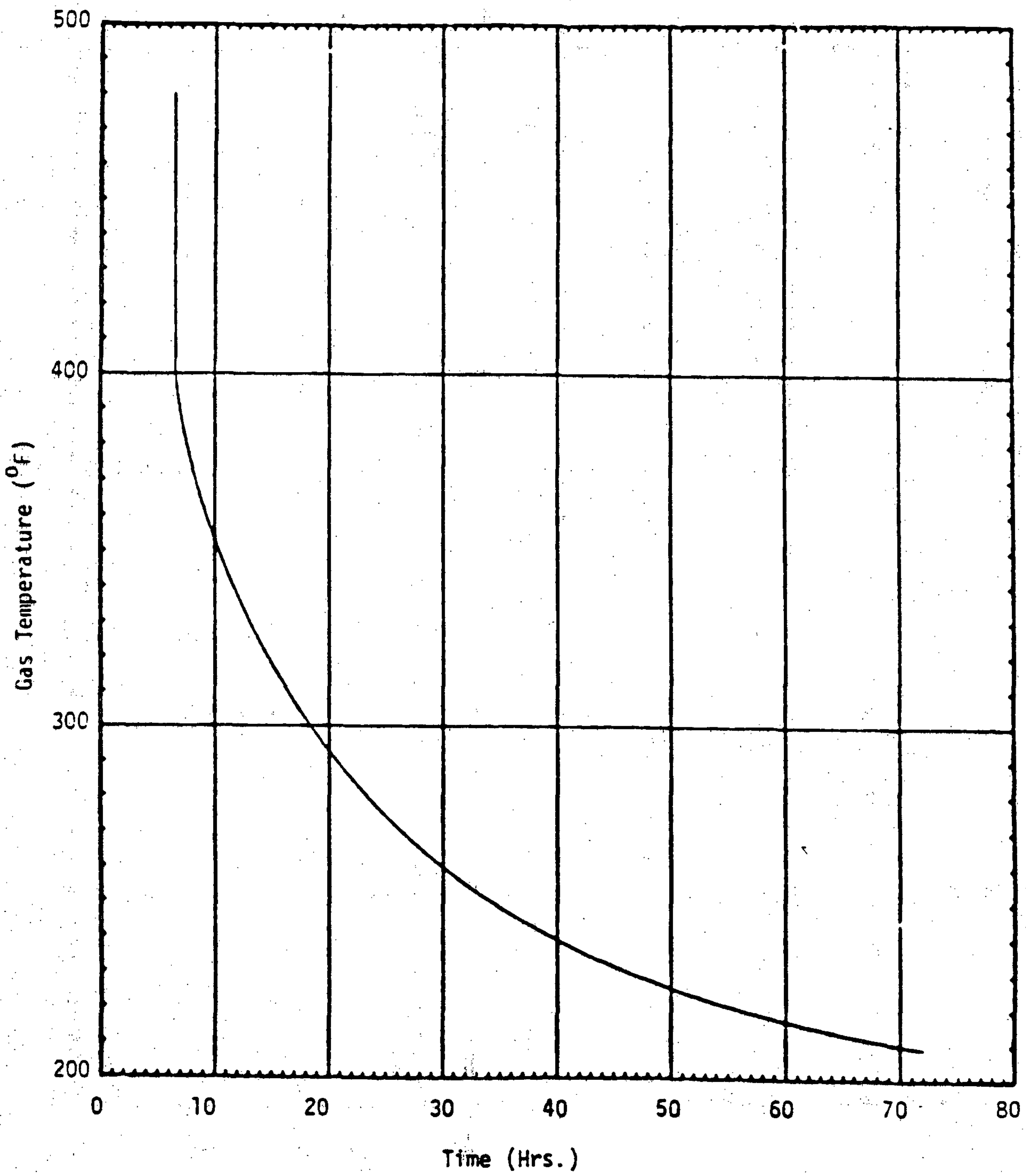

Figure 15.6.1.4-10. Reactor Cavity - Gas Temperature (SOFIRE Phase)

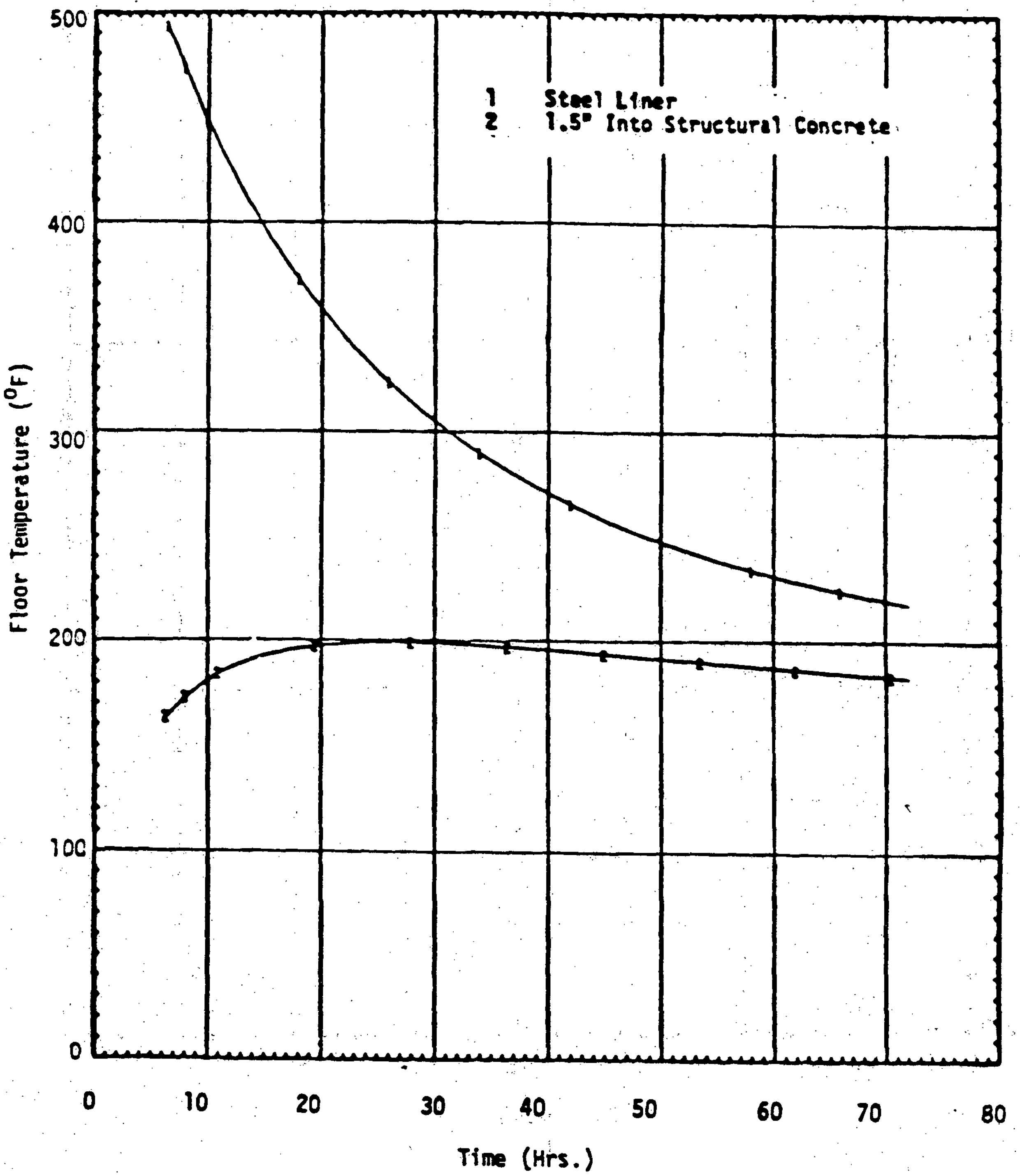

Figure 15.6.1.4-11. Reactor Cavity - Floor Temperature

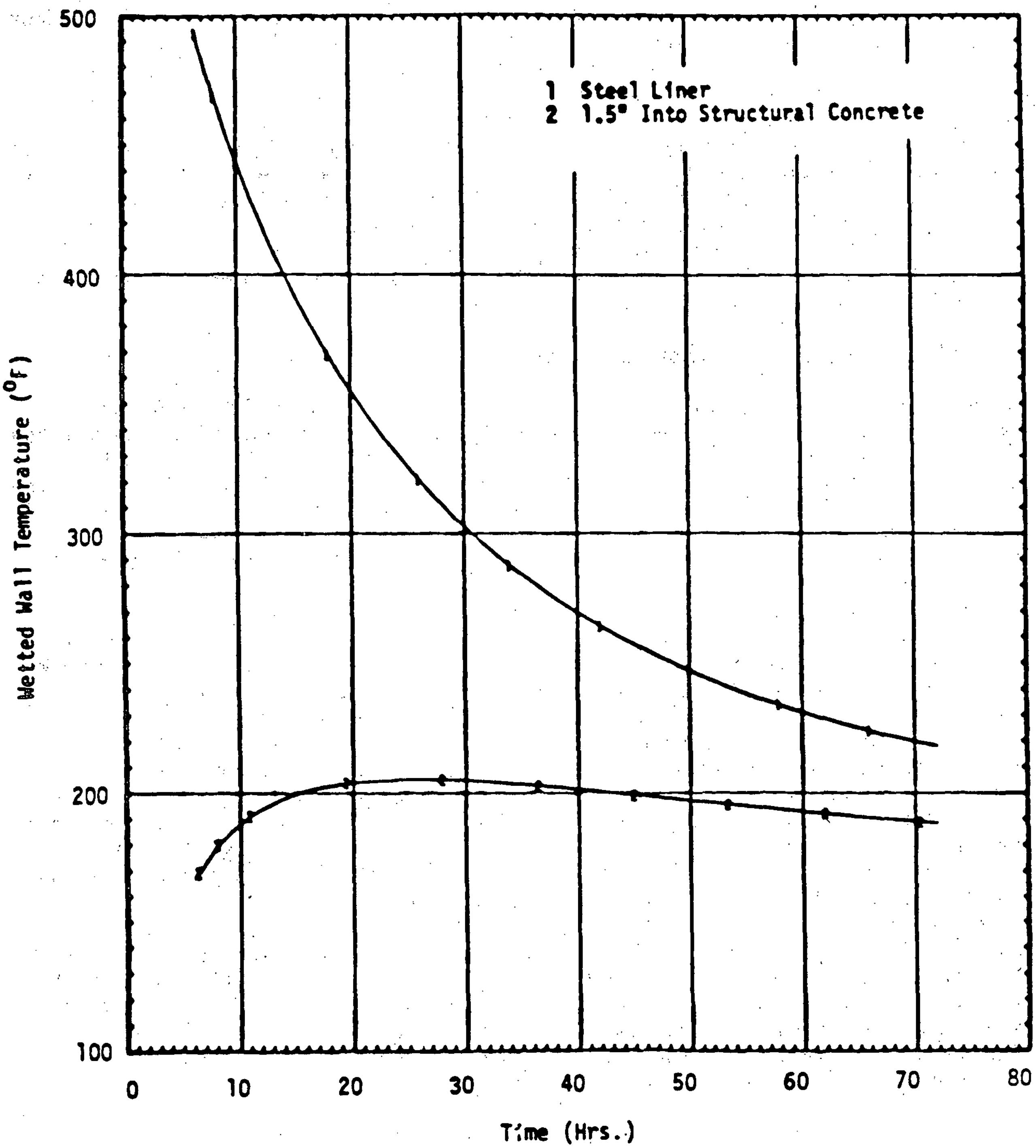

Figure 15.6.1.4-12. Reactor Cavity - Wall Temperature

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15.6.1.5 Intermediate Heat Transport System Pipe Leak

15.6.1.5.1 Identification of Causes and Accident Description

It is expected that results of inservice inspection, pipe fabrication and installation quality assurance measures, fracture mechanics analyses and tests, and leak detection provisions will lead to the conclusion that a sudden large failure approaching the complete severence of an IHTS pipe is not credible. In particular, data from tests of leak detectability indicate that the selected methods of leak detection ensure early detection of small IHTS leaks. The Design Basis IHTS leak selected on the basis of the existing information is that equivalent to the flow from a sharp edged circular orifice whose area is equal to one-half the pipe diameter times one-half the pipe wall thickness. (For the 24 inch IHTS piping the orifice area is 2.85 square inches.) This pipe break is consistent with the Moderate Energy Fluid System (MEFS) leak for piping with low stored energy identified in NRC Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment."

Thermal and Aerosol Consequence Assessment

A sodium leak in the 24-in.-OD main loop hot leg piping in Cell 226 was selected as the limiting case for the design of the SGB; leaks in branch lines or thermowells would fall within the magnitude of this limiting analysis. Leaks in the main loop piping in other cells have been evaluated; however, the leak in Cell 226 represents the limiting case for design since the potential cell pressure and the potential combustion product aerosol release to the outside atmosphere are maximized. The leak is assumed to occur while the IHTS is operating at maximum normal operating temperature and pressure. The pipe break location was chosen to be at the low point of the main loop hot leg piping. This location maximizes the spill volume. The spill parameters were generated by considering the system hydraulic behavior during the pipe break. A conservative assumption is made that no operator action is taken to trip the pump in the leaking loop or to drain the loop to the dump tank. This assumption disregards the probable alarm of any leak by the extensive detection provisions of the Sodium-to-Gas Leak Detection System which are discussed in Section 7.5.5. A reactor trip is caused by a Plant Protection System signal from a Primary-Secondary flow mismatch. Loop flow is assumed to continue under pump head until the pump tank is emptied through the leak. The leak continues at a decreasing rate determined by the cover gas pressure and gravity head. The initial sodium discharge flow rate is 129 lb/sec, and the total spill quantity is approximately 300,000 lb of sodium. The spill duration is approximately 5.5 hours. The leak rate time history is depicted in Figure 15.6.1.5-1. The temperature of the initial sodium discharge is 936^{o}F, and the average bulk temperature of the sodium is 800^{o}F. The reactor decay heat is removed through the two remaining loops via the condenser by-pass or via steam venting and the protected air-cooled condensers. This accident is classified extremely unlikely.

This assessment has not included potential sodium jet impingement on SGB concrete walls. The Project is investigating techniques to mitigate the effects of sodium jet impingement on SGB concrete walls and will incorporate discussions of mitigation features into the PSAR as they are developed.

Radiological Consequence Assessment

An even more conservative assessment was made to demonstrate the potential radiological consequences of an IHTS pipe leak do not pose an undue hazard to public health and safety. The IHTS Design Basis Leak was combined with the maximum undetected IHX leak (i.e. a break in the IHX tube bundle), loss of all off-site power, and no credit was taken for any active components or systems to mitigate the consequences of the accident. The maximum undetected IHX leak is that leak which would cause a one (1) foot drop in the IHTS sodium level in 24 hours given normal pressures in the IHTS and PHTS. This leak rate is 0.78 gpm. Due to sodium loss via this leak, the IHTS pressure at the IHX would equalize with the PHTS several minutes after reactor trip and flow through the defective IHX would eventually reverse, allowing radioactive primary sodium to enter the IHTS and mix with the IHTS sodium. It was assumed that primary sodium leaked into the IHTS at 0.78 gpm for two (2) hours following reactor trip. It was further assumed that operator action is taken within 24 hours to break siphon of primary sodium to the IHTS.

15.6.1.5.2 Analysis of Effects and Consequences

Thermal and Aerosol Consequence Assessment

For calculational purposes, the accident was modeled as a high-velocity sodium jet which impacts an obstructionand is converted into a spray. It was conservatively assumed that impingement occurred on the highest possible elevation. This maximizes both the fraction of sodium converted to spray (spray conversion factor = 1) and the spray height. The spray height determines the burning time of the droplets. The spray volume was determined based on the assumed spray height and the cell dimensions. The sodium that reaches the floor of the cell at the 765-ft elevation is drained to a lower cell equipped with a catch pan with a fire suppression deck. Hot gases generated by the fire are vented to the atmosphere in a controlled manner.

The SPCA (Spray-Pool Combustion Analysis) code was used to determine the thermal conditions in the SGB during and after the leak (fire) accident (See Appendix A). The model accounts for the heat generated by spray burning and by pool burning on the wetted surfaces of the walking grating and the catch pan. Due to the relatively long duration of the spray fire, both the spray and pool burning thermal effects are combined into a single analysis.

The spray burning rate is computed by using the initial burning rate and cell oxygen concentration (ambient 0.23 weight fraction) from a separate SPRAY (See Appendix A) code calculation of the same accident, as an input parameter. This initial burning rate is then varied, in relation to the sodium mass leak rate and the cell oxygen mass fraction, to yield the spray burning rate as a function of time. The pool burning rates are computed using the free convection mass transfer correlation equation used in the GESOFIRE code (See Appendix A). In this correlation, the burning rate is determined by the rate of diffusion of oxygen to the burning surface.

The heating of the cell gas, as a result of sodium burning, causes an expansion of the cell atmosphere. The model includes a provision for venting hot gases to the outside atmosphere or to an adjacent cell. Both natural convection and forced convection gas flows are considered. The mass fraction of oxygen in the cell changes during the course of the accident due to the net effect of the venting out of cell gas, the venting in of outside air, and the consumption of oxygen by combustion. The aerosol generation rate and the gas venting rate computed by SPCA are used along with the cell geometric data as input to the HAA-3B code (See Appendix A). HAA-3 computes the aerosol concentration in the cell and the rate of discharge of aerosols to the outside atmosphere.

Radiological Consequence Assessment

For calculating the 2-hour Exclusion Area boundary doses, it was assumed that the 94 gallons of primary sodium leaked into the IHTS is mixed evenly with the 39,000 gallons of IHTS sodium spilled. Accounting for the catch pan/fire suppression deck features in the SGS (see Section 6.6), approximately 10% (4000 gallons) of sodium is burned including 9.5 gallons of primary sodium.

For the LPZ dose, it was assumed that the flow of primary sodium into the IHTS is mitigated within 24 hours via operator action with approximately 600 gallons of primary sodium being spilled into the SGB.

Twenty seven (27) percent of the sodium burned is assumed to be released as sodium oxide aerosol. The entire mass of aerosol formed in the SGB is assumed to be released at ground level, disregarding aerosol mitigation features that are expected to limit aerosol release to a very small fraction of that formed inside the SGB.

Meterological dispersion was calculated using the 95th percentile X/Q identified in Section 2.3.4. Fallout of the aerosol during transit down wind was neglected. The isotopes considered are listed in Table 15.6.1.5-2. Other isotopes in the primary sodium are not significant countributors to the off-site dose.

15.6.1.5.3 Analysis Results

Thermal and Aerosol Consequence Assessment

The analysis results show the following effects:

1) The cell gas temperature is heated to 630 ^{o}F two minutes after leak initiation;

2) during the first several minutes of the event, hot gases containing combustion product aerosols are exhausted to the outside atmosphere at a rate of 10^5 cfm; and

3) after the initial atmosphere heatup, the building temperature decreases as the leak rate decreases, and heat is transferred to the building structure and to the outside atmosphere.

To quantify the above considerations, a determination was made of the maximum quantity of combustion product aerosols that could be released to the atmosphere without impairing the operation of the plant safety-related equipment. The dispersion factors used to determine the aerosol concentrations at the equipment intakes were derived using formula (6) of Murphy and Campe (Ref. 15.6.1.5-1). The impact of ingested aerosols on the performance of the safety-related equipment was evaluated considering both the effects of plugging of the flow passages and the degradation of heat transfer surfaces by deposited aerosols. It was concluded that in order to assure continued operation of the plant safety-related equipment it was necessary to limit the total aerosol release to 630 lbs. with no more than 100 lbs. being released after the first 5000 seconds of the accident. (See Section 6.2.7)

The following analysis assumptions were made, consistent with the design features discussed in Sections 9.13.2.

1) At the time aerosols are detected in the SGB HVAC exhausts, all SGB HVAC inlet and exhaust dampers are automatically closed except for one 40-ft^2 opening. Detection capability is provided by engineered safety feature aerosol detectors (Section 9.13.2).

2) As the SGB atmosphere expands, the cell gases and combustion product aerosols are discharged through the 40-ft^2 opening.

3) Approximately 5 minutes later, the 40-ft^2 opening is automatically closed by a timing circuit. This prevents the subsequent release of aerosols by natural convection.

4) Protected Air Cooled Condenser fan operation, if initiated, is automatically interrupted, for a period of up to 5000 seconds, upon the detection of aerosols in the air uptake. This prevents the PACC's from ingesting aerosols during the first 5000 seconds of the accident when aerosol release may be significant.

Additionally, the following actions are taken to mitigate the consequences of the limited aerosol release:

> Upon detection of aerosols in the SGB exhaust air, all the HVAC air intake openings throughout the plant, with the exception of the Emergency Diesel Generator Rooms (which rely on outside air for cooling), are automatically closed within 1 minute by the outside air dampers from a central signal. This limits the quantity of aerosols that can be ingested into other areas of the plant and prevents plugging of the HVAC unit air intake filters. The air intakes on those HVAC units serving the adjacent IHTS loops may be reopened, if such action is required to provide cooling to the adjacent loop cells.

The peak concrete cell wall and floor temperature transients are shown in Figures 15.6.1.5-2 and -3, respectively. The cell gas temperature transient is shown in Figure 15.6.1.5-4, and the integrated combustion product aerosol release to the atmosphere is shown in Figure 15.6.1.5-5. It is seen that the maximum concrete temperatures remain below 300^oF, and the long-duration (greater than 24 hours) concrete temperatures are below 200^oF. These temperatures are acceptable for building structural design. Approximately 440 lbs. of aerosols are discharged to the atmosphere during the initial pressure pulse, and an additional release of 90 lbs. occurs after closure of the 40-ft^2 vent. This later release is associated with an increase in pressure accompanying the increase in sodium leak rate at 3000 seconds depicted in Figure 15.6.1.5-1. It is estimated that an additional 100 lbs. of aerosols could diffuse through building cracks/penetrations during the course of the accident. The total release is within the design limit. The peak cell pressure during the initial heatup is estimated to be approximately 0.4 psig.

Radiological Consequence Assessment

The radiological consequences of a large IHTS sodium spill and fire would normally be insignificant since the only radioactive material in the IHTS sodium will be a low level of tritium. However, an extremely conservative assessment of potential off-site doses was performed assuming the plant had been operating with an undetected IHX leak prior to the accident.

The radiological analyses results provided in Table 15.6.1.5-3 show that the off-site doses resulting from the IHTS Design Basis Leak are below the 10CFR100 guideline limits.

15.6.1.5.4 Summary of Conclusions

A large sodium leak in the 24-in.-OD main loop hot leg piping of the IHTS was analyzed to evaluate the impact on the building structure and on the operation of the plant safety-related equipment. An analysis of the consequences of the postulated accident, shows that the sodium combustion product aerosol release is acceptable. The structural concrete temperatures and building pressures are within allowable limits.

References

15.6.1.5-1, K. B. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19", Proceedings of The 13th AEC Air Cleaning Conference, Vol., pp 401-428, CONF-740807.

TABLE 15.6.1.5-1

PHYSICAL PARAMETERS IHTS PIPE LEAK CELL 226

Spray Droplet Diameter (in.)	0.18
Cell Wall Area (ft^2)	35,725
Initial Gas Temperature (^oF)	100
Initial Gas Pressure (psia)	14.7
Initial Sodium Temperature (^oF)	936
Cell Free Volume (ft^3)	525,220
Spray Volume (% Vol.)	25
Spray Height (ft)	105.5
Oxygen Concentration (% Vol.)	21
Floor Area (ft^2)	4,726
Ceiling Area (ft^2)	5,476
Floor Thickness (ft)	2
Wall Thickness (ft)	3
Insulation Thickness (in.) (Catch Pan, El. 765 ft)	4
Vent Area (ft^2)	40

TABLE 15.6.1.5-2

Isotopes Considered In the
Radiological Consequence Assessment

Isotope
Na-24
Cs-137
Cs-136
Cs-134
I-131
I-132
Pu-238
Pu-239
Pu-240
Pu-241
Sr-90

TABLE 15.6.1.5-3

POTENTIAL OFF-SITE DOSES

Organ	10CFR100	Dose (rem) SB(0.42 mi) 2-hour	LPZ (2.5 mi) 38 days
Whole Body**	25	0.65	2.0
Thyroid	300	0.55	1.61
Bone	150+	0.77	2.14
Lung	75+	0.21	0.62

+Not covered in 10CFR100; used as guideline values.
**Includes both inhalation and external gamma exposure.

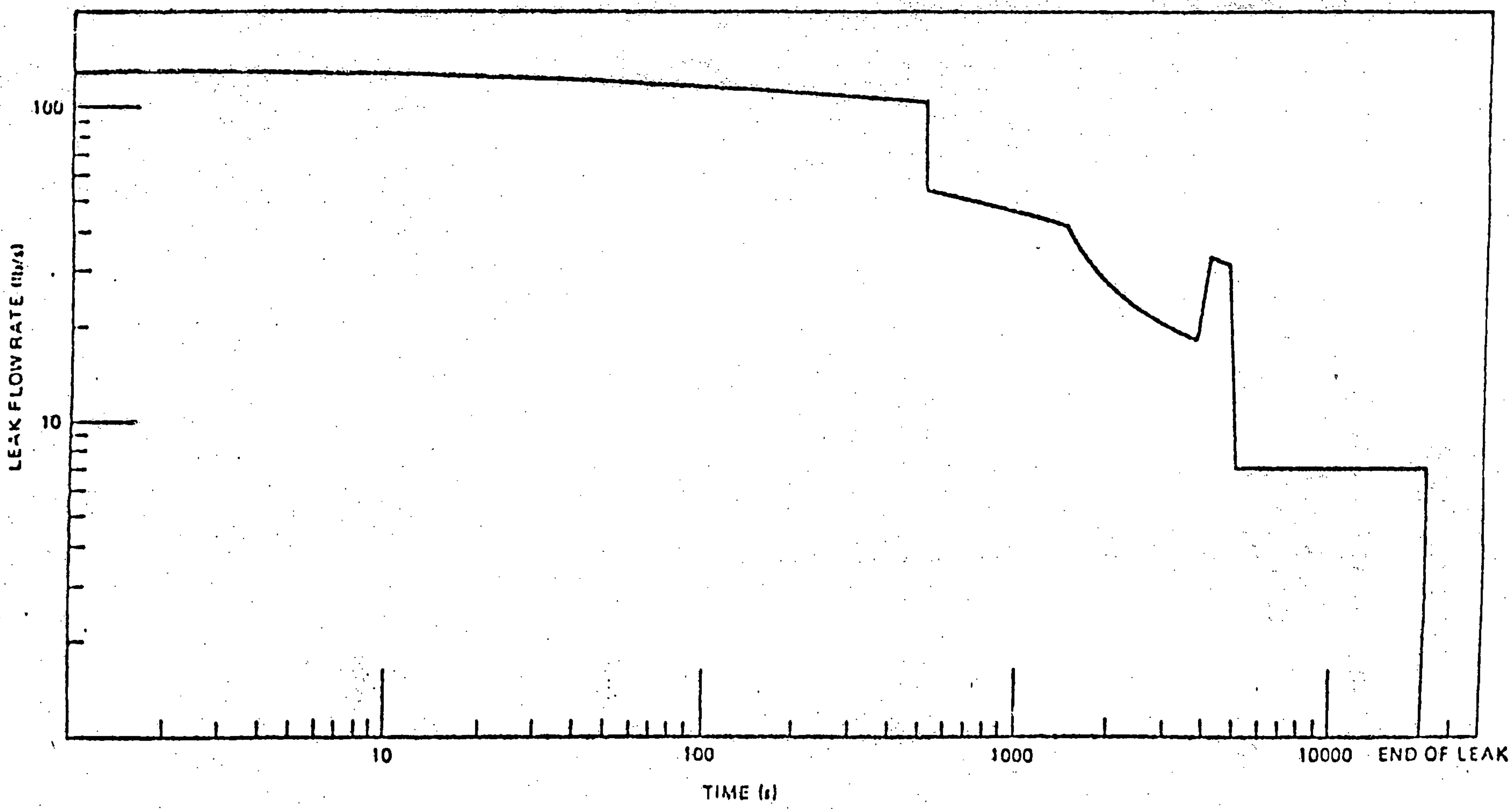

Figure 15.6.1.5-1. Leak Rate Time History IHTS Pipe Leak Cell 226

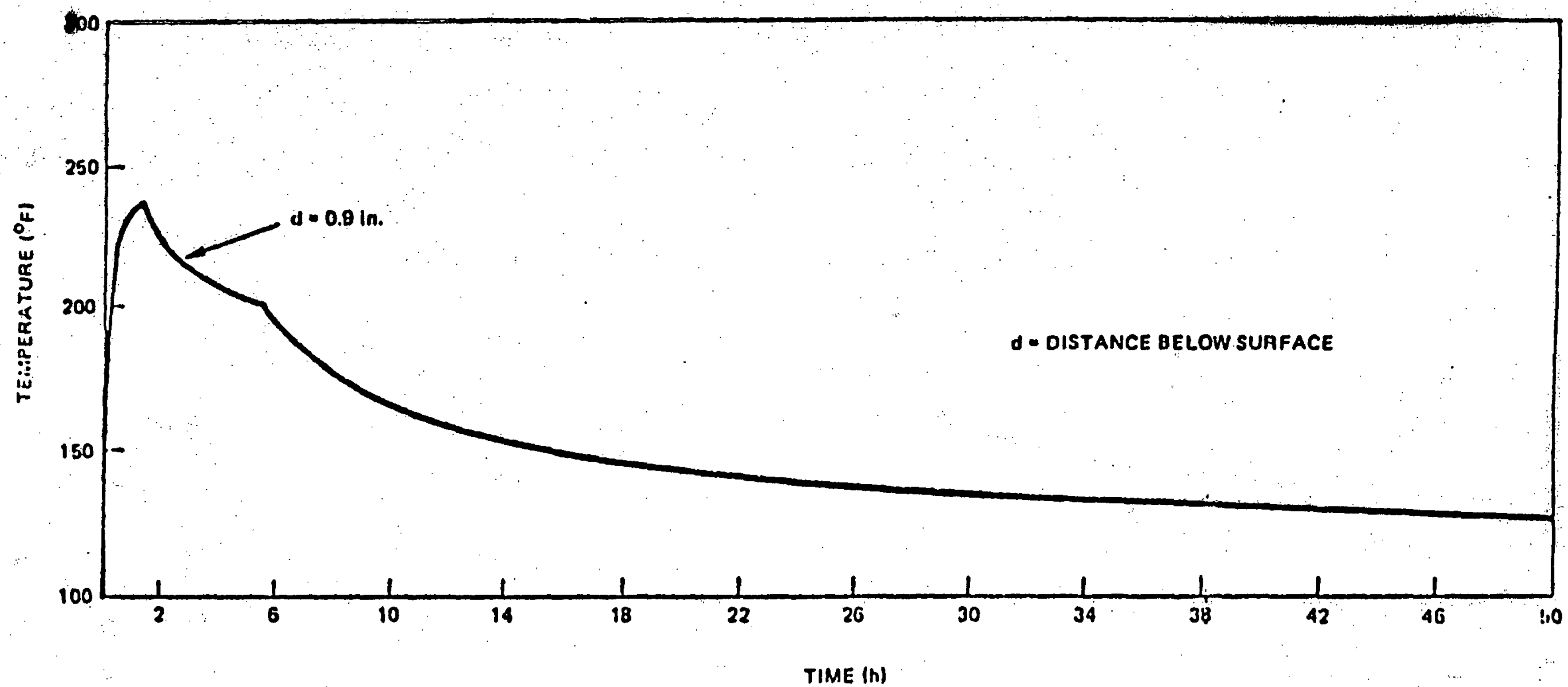

Figure 15.6.1.5-2. Peak Wall Temperature IHTS Pipe Leak Cell 226

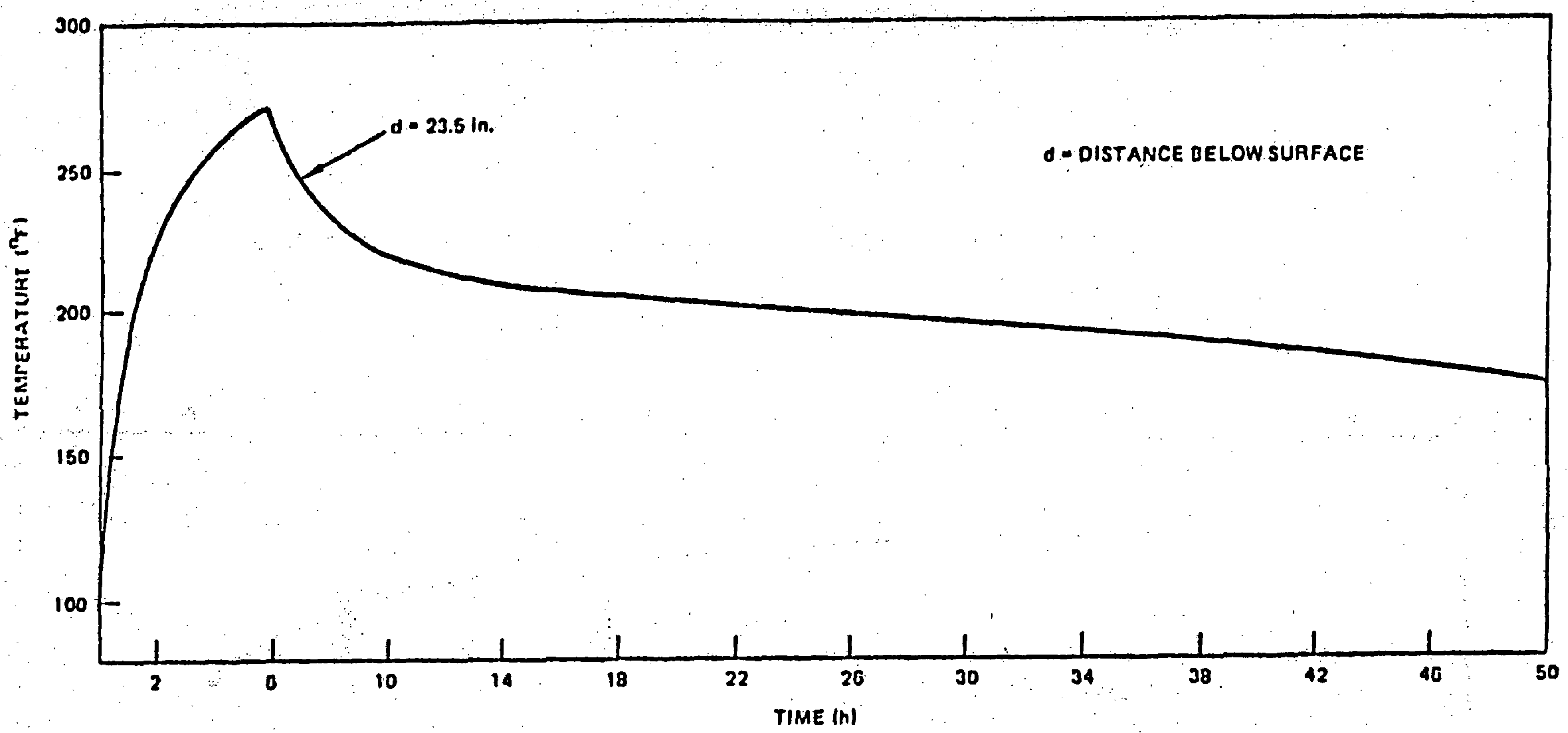

Figure 15.6.1.5-3. Peak Wall Temperature IHTS Pipe Leak Cell 226

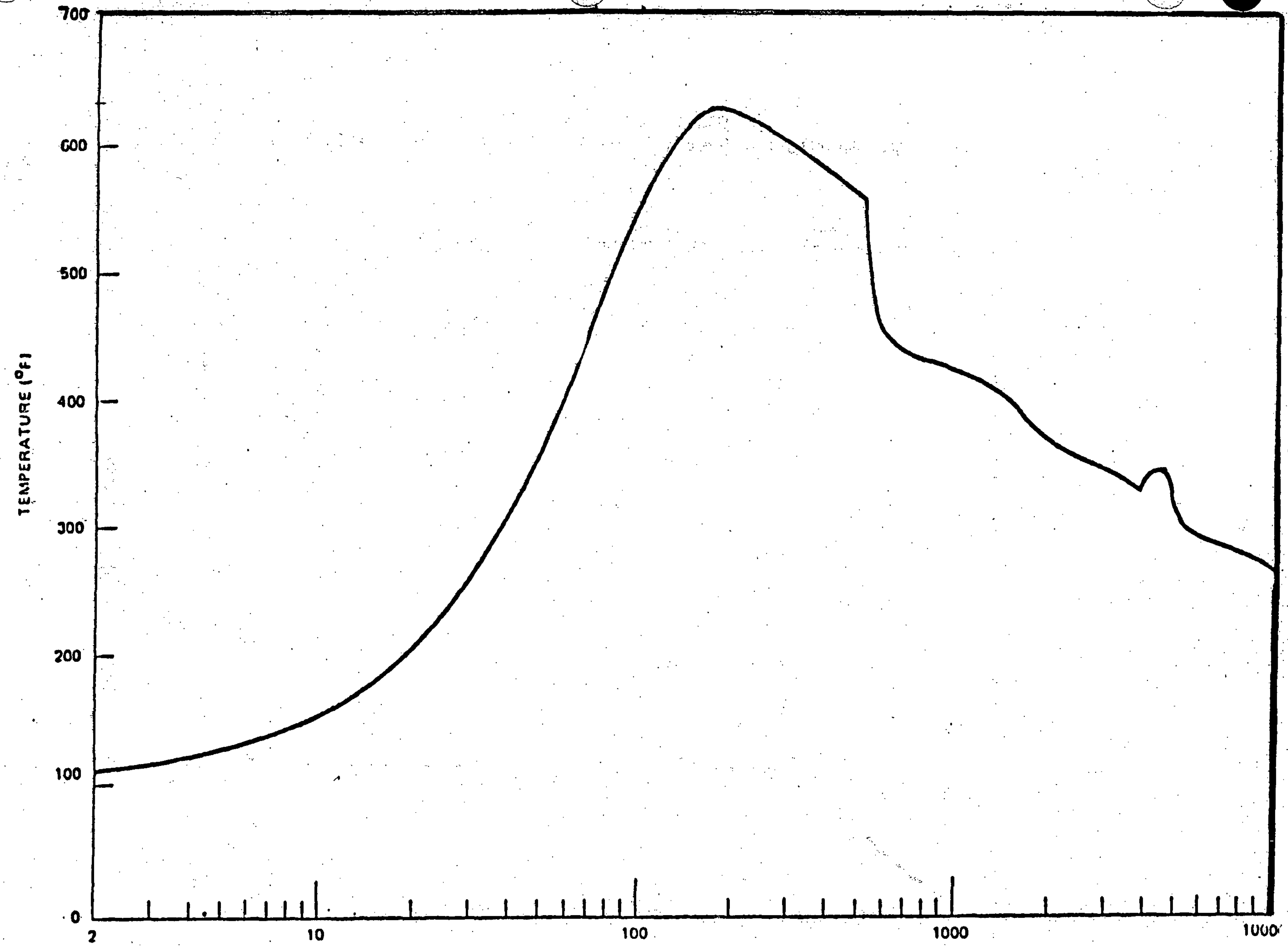

Figure 15.6.1.5-4. Cell Gas Temperature IHTS Pipe Leak Cell 226

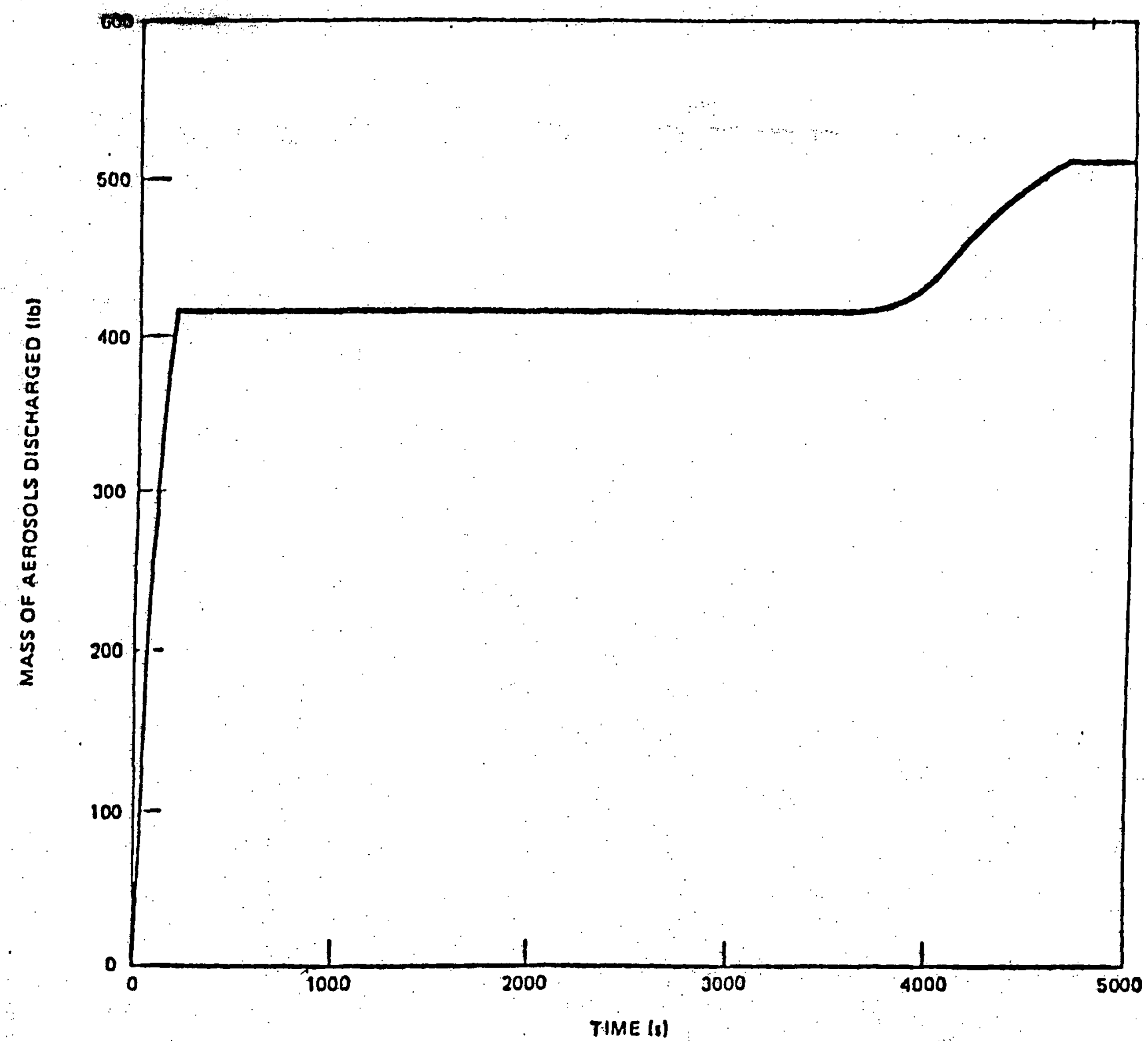

Figure 15.6.1.5-5. Combustion Product Aerosols Discharged
IHTS Pipe Leak Cell 226

15.7 OTHER EVENTS - INTRODUCTION

This section of the Chapter 15 accident events groups together with all those events of significance that do not appear to fall under any of the preceding categories. For these accident events either: 1) the potential limiting accident parameter will be presented, or 2) reference will be made to an umbrella event that has been previously established as a safe event.

A summary of the events discussed in this section are shown in Table 15.7-1. Table 15.7-1 also identifies potential limiting parameters and provides comments on each of the events.

For those events in which a radiological release accompanies the accident event, the resultant doses at the site boundary are compared to 10% of 10CFR100 guideline value. This value is used only as a suggested guideline value in order to establish a benchmark for determination of the severity of the event.

TABLE 15.7-1

OTHER EVENTS

Section No.	Events	Potential Limiting Parameters	Comments
15.7	Other Events		
15.7.1	Anticipated Events		
15.7.1.1	Loss of One D C System	None	No adverse operating conditions have been identified with this event.
15.7.1.2	Loss of instrument or valve air system	None	Detailed description of failure effects or safety-related instrument air supplies, if any, will be provided in the FSAR.
15.7.1.3	IHX Leak	None	Core sees normal shutdown.
15.7.1.4	Off-normal cover gas pressure in the reactor primary coolant boundary	None	No adverse operating conditions associated with this event.
15.7.1.5	Off-normal cover gas pressure in IHTS	None	No adverse operating conditions associated with this event.
15.7.2	Unlikely events		
15.7.2.1	Inadvertent release of oil through the pump seal (PHTS)	None	No adverse consequence identified at this time.
15.7.2.2	Inadvertent release of oil through the pump seal (IHTS)	None	No adverse consequence identified at this time.
15.7.2.3	Generator breaker failure to open at turbine trip	None	Core sees only normal shutdown.
15.7.2.4	Rupture of RAPS Cryostill	<3 REM (integrated 2-hr dose at the site boundary)	Consequences would be within the suggested guideline doses.
15.7.2.5	Liquid rad-waste system failure	3.7×10^{-6} REM @ site boundary 3.05×10^{-7} REM @ LPZ	Consequences would be within the suggested guideline doses.
15.7.2.6	Failure in the EVST NaK System	None	No adverse consequences associated with these events.
15.7.2.7	Leakage from sodium cold traps	7.8×10^{-5} REM @ site boundary 2.3×10^{-5} REM @ LPZ	Consequences would be within the suggested guidelines doses.

TABLE 15.7-1

OTHER EVENTS (Cont'd.)

Section No.	Events	Potential Limiting Parameters	Comments
15.7.2.8	Rupture in RAPS Noble Gas Storage Vessel Cell	<3 REM (integrated 2-hr dose of the site boundary	Consequences would be within the suggested guideline doses.
15.7.2.9	Rupture in the CAPS cold box	0.14 REM @ site boundary	Consequences would be within the suggested guideline doses.
15.7.3	Extremely unlikely events		
15.7.3.1	Leak in a core component pot	~3200°F Center Fuel Pin	Only slight cladding melting. Fission gas release within umbrella of Section 15.5.2.3.
15.7.3.2	Spent fuel shipping cask dropped from maximum possible height	8.89×10^{-7} REM Whole Body @ SB (2-hr) 1.13×10^{-6} REM Whole Body @ LPZ (30-day)	Doses are well within the suggested guidelines.
15.7.3.3	Maximum possible conventional fires, flood, and storms	None	None
15.7.3.4	Failure of plug seals and annuli	None	No adverse consequences associated with this event.
15.7.3.5	Fuel rod leakage combined with IHX and steam generator leakage	None	No adverse consequences associated with this event.
15.7.3.6	Sodium Interaction with Chilled Water	None	None
15.7.3.7	Sodium-Water reaction in large component cleaning vessel	0.01 REM @ site boundary	Consequences would be within the suggested guideline doses.

15.7.1 Anticipated Events

15.7.1.1 Loss of D. C. System

15.7.1.1.1 Identification of Causes and Accident Description

Among the credible events which can cause the loss of one D.C. system, in whole or in part, are a faulted D.C. system main bus, a faulted branch circuit, or an open connection. Regardless of the cause, the loss of voltage on the main bus, or on safety-related branch circuits, will be annunciated in the control room, and action will be taken to return the system to normal operation.

15.7.1.1.2 Analysis of Effects and Consequences

The plant design includes three independent battery supported Class 1E D.C. systems which are fully described in Section 8.3.2.1. The loss of one D.C. system will not prevent the operation of Class 1E D.C. loads as these systems will be designed with sufficient physical separation, electrical isolation, and redundancy to prevent occurrence of common failure modes.

It should be noted that the loss of one D.C. system will not result in a loss of the associated vital bus as this bus will be automatically transferred to the Class 1E A.C. Distribution System. Since the transfer will be accomplished with static transfer switching circuitry synchronized with the aforementioned system voltage, the performance of loads fed from the vital bus will not be degraded by the transfer.

15.7.1.1.3 Conclusions

The loss of one D.C. system will not prevent the functioning of Class 1E D.C. loads since these loads will be separated into redundant groups, each group being powered by a separate D.C. system. Further, since the D.C. systems are designed to preclude common failure modes, the redundant counterpart of the system that is lost will remain operational to provide the required safety actions.

15.7.1.2 Loss of Instrumentation or Valve Air

15.7.1.2.1 Identification of Causes and Accident Description

The system design precludes the loss of air supply to safety-related valves or instruments due to a single credible event. However, multiple failures, or a single failure occurring at the time of a design basis event, could cause loss of instrumentation or valve air. Among such single failures are check valve malfunction caused by valve seal failure. Table 15.7.1 provides a listing of safety-related valves which requires a compressed air supply and their preferred operating directions.

15.7.1.2.2 Analysis of Effects and Consequences

The systems supplying compressed air to safety-related valves or instruments will be designed such that a single credible failure will not cause interruption of the air supply. The instrument air system is designed to supply clean, dry, and oil-free air for plant instrumentation and control. The air receiver tanks are designed to the ASME Boiler and Pressure Vessel Code, Section VIII, Division 1. Piping is designed to ANSI B31.1.0. Piping which penetrates the reactor containment walls, and the containment isolation valves are ASME Section III, (Sections 3.9.2 and 6.2.4). Intercooler and after-coolers are designed to TEMA Class R.

All active safety-related, air operated valves will be designed to move in a preferred direction with the loss of air supply. Table 15.7.1.2-1 identifies the safety-related valves requiring compressed air and the normal and failed positions and function performed. Valves required to be operable for a safe shutdown are equipped with safety-related accumulators. Each safety-related system is redundant.

There is no compressed air supplied to safety-related instrumentation such that the loss of compressed air would result in a loss of the instrumentation safety-related function.

15.7.1.2.3 Conclusions

Based on the preceeding discussion, the compressed air system will be designed to prevent any adverse effects on the safe operation of the plant due to loss of instrument of valve air.

TABLE 15.7.1.2-1

ACTIVE SAFETY-RELATED VALVES OPERATED BY COMPRESSED AIR

System	Valve Number	Normal Operating Position	Failed Position After Loss of Compressed Air	Function
Primary Sodium Removal and Decontamination System (Nuclear Island General Purpose Maintenance System)	HV001A	Opened	Closed	Containment Isolation
	HV044A	Opened	Closed	Containment Isolation
	HV004B	Opened	Closed	Containment Isolation
	HV085A	Opened	Closed	Containment Isolation
	HV085B	Opened	Closed	Containment Isolation
	HV086B	Opened	Closed	Containment Isolation
Emergency Chilled Water	NV353	Opened	Failed Open	System Isolation
	NV354	Opened	Failed Open	System Isolation
	NV400	Opened	Failed Open	System Isolation
	NV401	Opened	Failed Open	System Isolation
	NV403	Opened	Failed Open	System Isolation
	NV404	Opened	Failed Open	System Isolation
	NV409	Opened	Failed Open	System Isolation
	NV410	Opened	Failed Open	System Isolation
	NV141AC	Opened	Failed Open	System Isolation
	NV141AD	Opened	Failed Open	System Isolation
	NV141BC	Opened	Failed Open	System Isolation
	NV141BD	Opened	Failed Open	System Isolation

TABLE 15.7.1.2-1 (Continued)

ACTIVE SAFETY-RELATED VALVES OPERATED BY COMPRESSED AIR

System	Valve Number	Normal Operating Position	Failed Position After Loss of Compressed Air	Function
Emergency Chilled Water (cont'd.)				
	AOV165	Opened	Closed	Containment Isolation
	AOV166	Opened	Closed	Containment Isolation
	AOV167	Opened	Closed	Containment Isolation
	AOV168	Opened	Closed	Containment Isolation
	AOV211	Opened	Closed	Containment Isolation
	AOV212	Opened	Closed	Containment Isolation
	AOV79	Opened	Closed	Containment Isolation
	AOV80	Opened	Closed	Containment Isolation
	AOV415	Opened	Closed	Containment Isolation
	AOV418	Opened	Closed	Containment Isolation
Auxiliary Liquid Metal System				
EVST Na Cooler Outlet NaK				
Loop 1	HV359*	Open	Fail as Is	System Isolation
Loop 2	HV420*	Open	Fail as Is	System Isolation
EVST NaK Loop 1 Isolation	HV357*	Closed	Fail as Is	Containment Isolation
EVST NaK Loop 1 Isolation	HV358*	Closed	Fail as Is	Containment Isolation
EVST NaK Loop 2 Isolation	HV415*	Closed	Fail as Is	Containment Isolation
EVST NaK Loop 2 Isolation	HV416*	Closed	Fail as Is	Containment Isolation

*Air stored in an accumulator for emergency operation of the valve.

TABLE 15.7.1.2-1 (Continued)

ACTIVE SAFETY-RELATED VALVES OPERATED BY COMPRESSED AIR

System	Valve Number	Normal Operating Position	Failed Position After Loss of Compressed Air	Function
Inert Gas Receiving and Processing System	RPHV001(1)	Opened	To RAPS	Process Effluent
	RPHV002(1)	Opened	Closed	Containment Isolation
	RPUV015A(1)	Opened	Closed	System Isolation
	RPUV015B(1)	Opened	Closed	System Isolation
	RPUV018(1)	Opened	Closed	System Isolation
	RPUV019(1)	Opened	Closed	System Isolation
	APHV001(2)	Opened	Closed	Containment Isolation
	APHV002(2)	Opened	Closed	Containment Isolation
	NGHV351A(3)	Opened	Closed	Containment Isolation
	NGHV351B(3)	Opened	Closed	Containment Isolation
	CGHV501(4)	Opened	Closed	Containment Isolation
	CGHV301(4)	Opened	Closed	Containment Isolation

(1) See Figure 11.3-4; (2) See Figure 11.3-6; (3) See Figure 9.5-8; (4) See Figure 9.5-2

Evaporator Water Dump	53WDV001-004	Closed	Closed	System Isolation
Superheater Outlet	53SGV106-108	Closed	Closed	Relief (Power Operation)
Evaporator Outlet	53SGV100-103	Closed	Closed	Relief (Power Operation)
Steam Drum Outlet	53SGV104-105	Closed	Closed	Relief (Power Operation)

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TABLE 15.7.1.2-1 (Continued)

ACTIVE SAFETY-RELATED VALVES OPERATED BY COMPRESSED AIR

System	Valve Number	Normal Operating Position	Failed Position After Loss of Compressed Air	Function
Heating Ventilation and Air Conditioning System	ARAOV046A	Opened	Closed	Containment Isolation
	ARAOV046B	Opened	Closed	Containment Isolation
	ARAOV046C	Opened	Closed	Containment Isolation
	ARAOV047A	Opened	Closed	Containment Isolation
	ARAOV047B	Opened	Closed	Containment Isolation
	ARAOV047C	Opened	Closed	Containment Isolation
	ACAOV064A	Opened	Closed	System Isolation
	ACACOV064B	Opened	Closed	System Isolation
	ACAOV122A	Opened	Closed	System Isolation
	ACAOV122B	Opened	Closed	System Isolation
	ACAOV123A	Closed	Open	System Isolation
	ACAOV123B	Closed	Open	System Isolation
Floor Drain System	AOV34	Opened	Closed	Containment Isolation
	AOV67	Opened	Closed	Containment Isolation

15.7.1.3 IHX Leak

15.7.1.3.1 Identification of Causes and Accident Descriptions

The leakage of sodium from the IHTS into the primary system is monitored by temperature-compensated sodium level sensing devices. In the event such a leak in the IHX were to occur, both the reactor overflow tank level indicators and the intermediate system expansion tank temperature-compensated sodium level comparators would monitor any sodium level change, other than those generated by temperature changes; thereby, indicating an IHX sodium leak. The operators would then take appropriate corrective action.

15.7.1.3.2 Analysis of Effects and Consequences

Discussed in Sections 5.3.2, 5.3.3.1.5, and 5.3.3.5 of this PSAR are the IHX design descriptions, safety factors, allowances, and analytical methods applied in order to minimize the probability of an IHX leak. Also, discussed in Section 5.3.2.5 is the analysis of the effect and consequences produced by an IHX tube leak or other IHX failure where nonradioactive sodium leaks into the primary system. Section 15.7.3.5 treats the consequences of an IHX tube leak combined with a steam generator tube leak at a time when failed fuel is present in the reactor.

15.7.1.3.3 Conclusions

The sodium comparator leak detection system detects an IHX leak by the change in sodium inventory, before any adverse effect can take place in either the IHTS or the PHTS. Following shutdown, corrective maintenance will be performed on the faulted component.

The leakage rate of primary sodium to the intermediate system is zero. This is discussed in Section 5.3.2.5 of this PSAR. The prevention of primary flow into the intermediate system is provided by an overpressurization of the intermediate system relative to primary system at all steady state conditions of operation.

15.7.1.4 Off-Normal Cover Gas Pressure In the Reactor Coolant Boundary

15.7.1.4.1 Identification of Causes and Accident Description

As described in Section 9.5.1, the cover gas system serving the Reactor and Primary Heat Transport System maintains a pressure in the gas space of the Reactor Coolant Boundary of 6" ± 2" of H_2O. There is a constant sweep flow into the cover gas spaces and through the shaft seals of the primary pumps. This in-leakage is accommodated by two parallel pressure regulators in the line between the RAPS and the primary system overflow tank, which is maintained at the same pressure by a gas pressure equalization line connecting the pumps, reactor vessel, and overflow tank. The makeup regulators and the regulators controlling the bleed from the overflow tank to RAPS are both controlled from the same pressure signal. Failures of the pressure regulators (primary and redundant) or operator error, could cause deviation from the normal operating pressure of 6" ± 2" W.G.

15.7.1.4.2 Analysis of Effects and Consequences

a. Under pressure: If the pressure regulators (including redundant regulators) between the overflow tank and the RAPS system fail open, the pressure in the cover gas spaces within the Reactor Coolant Boundary will go sub-atmospheric since gas from the overflow tank will flow into the RAPS vacuum vessel. Since the vacuum vessel volume is approximately 300 ft^3 (at 8 psia minimum) and the combined gas volume of the reactor vessel, three primary pumps and the overflow tank is about 4500 ft^3, the reduction in pressure is modest: about 1 psi. Such a reduction in pressure would have no adverse affect on the primary system. The change in NPSH available to the pump would not be significant, and the seals in the reactor and pump closures would not be materially affected.

b. Over pressure: If the regulators between the overflow tank and the RAPS should close and the regulators controlling flow to the reactor vessel should, at the same time, fail open, the cover gas pressure in the reactor coolant boundary would increase. Any potential problem is mitigated however by: 1) the time required to establish any significant overpressure, and 2) pressure relief devices on the overflow tank. As mentioned above, the volume of the cover gas space within the reactor coolant boundary is about 4500 ft^3 at normal operating conditions. Since the gas makeup system will be designed to limit the makeup rate to about 50 SCFM, it would take at least an hour to double the cover gas pressure in the reactor coolant boundary. An annunciator in the control room will alert operators to take appropriate action (such as isolation of the makeup gas regulators) long before any appreciable overpressure will be realized. In addition, relief valves set to relief at 15 psig, will limit the pressure even if no operator action is taken prior to reaching this pressure. The discharge of cover gas from the relief device will be modest and CAPS action will preclude any hazard to the public. Even if the pressure does increase to 15 psig, there will be no affect on reactor vessel level or pump tank level performance.

The pressure boundary margin seals will resist pressure in excess of 300 psid without failure. If the pressure in the reactor vessel should increase to 15 psig, these seals would remain intact. Cover gas would bubble through the dip seal and be trapped in the riser annulus between the dip seal and inflatable elastomer seals.

The primary system (and reactor vessel) design pressures have been established on the basis of a 15 psig cover gas pressure, and therefore the system, from a structural standpoint, is unaffected by any overpressure which could occur. If the primary system gas pressure should drift up due to one of the postulated failures, the 10 psi minimum P between the intermediate and primary sodium in the IHX would decrease; however, this loss of P would be monitored and annunciated and appropriate action would be taken.

15.7.1.4.3 Conclusion

Off-normal cover gas pressures in the Reactor Coolant Boundary will not cause a safety problem. Underpressure would be limited to approximately 1 psi below the normal operating pressure of 6 inches W.G. Overpressure conditions would be limited to 15 psig by relief actions and would take about an hour to achieve. Even if such an overpressure condition were to exist, there will be no deleterious effect on the integrity of the Reactor Coolant Boundary. Since radiation dose rate builds up slowly and adequate radiation monitoring is provided, the radiation consequences would be small to the operating staff and are trivial to the public.

15.7.1.5 Off-Normal Cover Gas Pressure in IHTS

15.7.1.5.1 Identification of Causes and Accident Description

The IHTS argon cover gas is obtained from the steam generator building argon supply. The pressure in the pump and expansion tank cover gas is controlled to 100 psig during normal operation by a common feed and bleed control system. Off-normal cover gas pressure could result from a malfunction of the cover gas pressure control system or a loss of integrity of the cover gas pressure retaining boundary.

15.7.1.5.2 Analysis of Effects and Consequences

Lower cover gas pressure will lower pressures in the IHTS and will be detected by a differential pressure measurement between the primary and intermediate sides of the IHX. An annuciatory will sound when the intermediate system pressure is less than 10 psi above the primary system pressure. In addition, the cover gas pressure monitoring system will indicate lower cover gas pressure. An undetected low cover gas pressure could result in the introduction of radioactive material into the IHTS through potential leaks in the IHX tubes. This would be detected by a radiation monitor located outside containment on the intermediate hot leg. A complete loss of cover gas pressure would not reduce heat transport capability since the pumps will operate with atmospheric cover gas pressure without cavitation.

High cover gas pressure will be detected by the cover gas pressure indicator. The maximum cover gas pressure possible is equal to the steam generator building argon supply pressure which is 175 psig (see Section 9.5.1.2). Attainment of this pressure would require a failure of the cover gas control system, failure of pressure reducers located between the argon supply and the control system, failure of the cover gas pressure indicator, and/or failure of the operator to detect the off-normal condition. However, even if the cover gas pressure did reach 175 psig, there would be no deleterious effect on the IHTS. In this event, the maximum local pressure in the IHTS would be at the low point in the cold leg piping, which occurs at the first dump valve in the cold leg dump line. The pressure at this point is a combination of:

. IHTS cover gas pressure	∿175 psig
. IHTS pump tank static head	∿ 5 psig
. IHTS pump discharg head (max.)	∿125 psig
. Static head from pump discharge to the dump valve	∿ 10 psig
. Friction pressure drop from pump discharge to the dump line	∿ -2 psig
	Σ + 313 psig

The pressure of 313 psig is below the IHTS design pressure of 325 psig. 8

15.7.1.5.3 Conclusions

High cover gas pressure (to the maximum pressure of the argon supply) has no deleterious effect on IHTS performance. Lower cover gas pressure (to atmospheric pressure) will have no effect on IHTS heat transport capability. Low pressure could result in the introduction of radioactive material into the IHTS through the IHX only after failure of the IHX intermediate-primary ΔP monitor, failure of the cover gas pressure indicator, and leaks in the IHX tubes. This would be detected by a radiation monitor in IHX intermediate side outlet.

15.7.1.6 Small NaK Spills in the EVST Nak Systems

15.7.1.6.1 Identification of Causes

Of the potential for leakage in the NaK system, the only anticipated leakage is considered to occur at NaK valves. The leak is assumed to result from a failure of the bellows which is provided to prevent leakage at the valve shaft.

15.7.1.6.2 Analysis of Effects and Consequences

All NaK valves will include a backup seal (packing) to prevent a significant NaK leak in the event of failure of the bellows. The bellow failures will be detected by the leak detector but, prior to shutdown, may result in a small leak of NaK which will collect in the catch pan provided by the fire protection system. A small fire, localized at the leaking component, may result. The fire protection system described in Section 9.3.1 provides leak detection, catch pans with fire suppression features, and nitrogen flooding capability, thus alerting operating personnel and permitting shutdown, drainage, and replacement of the faulted component and, if required, startup of the backup EVST cooling circuit. The NaK is nonradioactive. The small spill volume and low temperature (approximately 350^{o}F) preclude significant temperature buildup. The component arrangement, with each of the cooling circuits physically separated from the other, insure against loss of both EVST circuits.

15.7.1.6.3 Conclusions

The absence of radioactivity, coupled with the design of the NaK system and components, as well as the available fire protection system, provides assurance that any anticipated leakage from the EVST NaK system will not adversely affect EVST cooling. There is no potential radiological release from the plant.

15.7.2 Unlikely Events

15.7.2.1 Inadvertent Release of Oil Through Pump Seal (PHTS)

15.7.2.1.1 Identification of Causes

The primary sodium pump has oil-lubricated bearings and/or seals above the pump tank which contains sodium. The seals will be designed to prevent oil leaking into the pump tank for all modes of operation.

The primary pump concept incorporates a seal lubrication system with a fixed total oil inventory (see Figure 5.3-14a). Oil that leaks through the lower seal will be collected in a lower seal leakage tank and pumped to waste during servicing. Abnormal leakage must be made up by deliberate manual action to open the system and add oil. The lower seal leakage tank has the capacity to hold the total seal oil inventory and thereby precludes any seal leakage from entering the pump tank in the event of an abnormal leak rate. An additional and last barrier preventing seal leakage from entering the pump tank sodium is provided in the pump design by a shaft oil slinger and reservoir located below the normal seal rubbing faces.

Any oil overflowing the lower seal leakage collection tank or running down the pump shaft is collected in a reservoir which has a capacity in excess of the total oil inventory. The primary pump concept, therefore, would require a combination of independent failures to occur coupled with a deliberate manual addition of oil to the system before oil could enter the pump tank.

Although the release of oil from the primary pump seal to the primary sodium is considered an extremely low probability event, the results of such an event have been evaluated. Two potential effects have been identified:

1. Plugging Effects
2. Reactivity Effects

15.7.2.1.2 Analysis of Effects and Consequences

If it is postulated that the oil were to be released to react with the primary sodium, the following analysis is presented.

The oil above the seal would flow down the pump shaft and vaporize, or react with the sodium in the pump tank. The reaction of oil and sodium will result in the release of hydrogen and carbon. The carbon compounds will either float on the sodium, dissolve (on the order of one ppb), or sink to the bottom of the pump tank. These are small particles which are easily fractured.

The release of these particles from the pump tank to the primary loop will depend upon the manner in which the pump is operating. If the pump is shutdown, the solids will stay in the pump tank.

If the pump continues to operate after a seal failure, the reaction products would eventually go into the primary loop. In the present pump concept the pump tank will contain approximately 800 gal. of sodium, and will be changing at 700 gpm due to flow from the IHX return (200 gpm) and bearing return flow (500 gpm).

Plugging Effect

Three different conditions were evaluated as follows:

A. To calculate the maximum plugging temperature in the pump discharge, the following conservative assumptions were made:

1. Pump tank temperature is 1000°F.

2. The pump tank vents to cover gas system through the pump standpipe bubbler. Maximum gas pressure is 12 in. W.G. plus equivalent static head of sodium @1000°F for elevation between normal RV sodium level and normal level in the pump tank. This assumes no pump draw down. Pressure is 97 in. W.G.

3. Oil leaks into the pump tank at a rate just sufficient to saturate the pump tank sodium volume of 800 gallons with H_2 at the temperature and pressure above. This results in a concentration of 121 ppm of H_2 in the pump tank sodium.

4. The pump tank mixture is drawn into the pump and mixed with primary sodium at the ratio of 700 gpm/34000 gpm (IHX and bearing return flow vs pump discharge flow).

5. The resultant pump discharge contains 2.5 ppm of disolved H_2 and the plugging temperature is 460°F.

B. To calculate the maximum plugging temperature in the core and the remainder of the system the following conservative assumptions were made:

1. Assume that leakage continues as in the previous condition until the entire 6 gallon inventory of oil in the seal system has leaked into the tank which is at 1000°F. 19

2. Assume that no oil is volatized to the cover gas and that no H_2 is drawn off the surface of the sodium into the pump tank cover gas. Note the standpipe bubbler draws off cover gas at a rate of 0.25 SCF per minute. Under these conditions approximately 5% by weight of the reaction products convert to H_2 dissolved in the sodium. Due to mixing in the reactor plenum and other loops this weight of H_2 can be assumed to mix with the entire primary sodium volume of 172,000 gal. (with an average temperature of 860°F).

3. Assume no cold trapping during this event.

The resulting 2.0 PPM of dissolved H_2 will give a sodium plugging temperature of approximately 440°F.

C. For operating conditions where the maximum primary sodium temperature in the pump tank is less than 1000°F, the weight of the reaction products of oil converting to dissolved hydrogen in the sodium will be less than the 5% used above. Test data indicate that the percentage by weight of oil converting to dissolved hydrogen drops off rapidly and is less than 2% at 850°F the lowest temperature for which data is available. Under these conditions (850°F) the 6.0 gallons of oil reacting with the sodium in the systems will result in a hydrogen concentration of less than 0.74 ppm and a plugging temperature of less than 377°F. With Hot Standby and Refueling Conditions where the primary sodium temperature in the pump tank is 600°F and 400°F respectively, the plugging temperatures will be well below the 377°F.

Reactivity Effects 19

Since the only known source of oil which might be postulated to be admitted to the primary system is in the pump seal area, this discussion will be limited to pump oil. Any leakage of this oil will normally not enter the primary system but will be handled in the seal oil leak off system. However, should there be an inadvertent bypassing of the seal oil leak off system and should the oil by some means enter the area around the pump shaft above the sodium, it would drain down the pump shaft to the sodium surface where (a) if it is a normal hydrogenous carbonaceous oil it would react with the sodium forming hydrogen gas and carbon particulates or (b) if it is a mineral oil it would react with the sodium and form hydrogen gas, but no particulates would form. In either case, most of the hydrogen would immediately rise to the pump tank free sodium surface, enter the cover gas system and eventually be processed through the RAPS. The remaining hydrogen would become dissolved in the sodium. The carbonaceous particulate would be very finely divided and although it may stay in the sodium system, it and the dissolved hydrogen would enter the main primary sodium stream very gradually through the controlled leakage areas of the pump. Therefore, although the vendor has not finalized the design of the seal system nor the volume of oil that is available for a postulated leak nor the selection of oil, it is highly improbable that any gaseous hydrogen from this oil could be postulated to enter the core. Calculations indicate that the reactivity effects associated with this scenario is of little consequence to the core (<<1¢). 25

Conclusions

It is highly improbable that any quantity of oil could be released through the pump seals in such a manner as to interact with the primary sodium coolant. Nevertheless, analyses, based on conservative assumptions have been carried out to determine the resultant; (1) plugging effects, and (2) reactivity effects associated with this postulated occurrence. The conclusions from these analyses indicate that; (1) during normal operation with mixing in the total primary sodium inventory the 19

plugging temperature was found to be on the order of 440°F well below the minimum operating temperature of approximately 640°F, (2) during Refueling or Hot Standby the plugging temperature was found to be well below 377°F which is below the 400°F temperature for Refueling conditions, and therefore presenting no safety problems, and (3) the potential reactivity effect associated with this event is of such a small nature
that the consequences to the core are considered insignificant. 19

15.7.2.2 <u>Inadvertent Release of Oil Through the Pump Seal into Sodium (IHTS)</u>

The release of oil in the PHTS has been discussed in Section 15.7.2.1. The release of oil to the IHTS from the pump oil bearing requires the failure of multiple barriers designed to prevent such a release. If oil contamination of the IHTS sodium did occur, it could be detected by monitoring the seal oil inventories or from a chemical analysis of sodium samples. An undetected
59 loss of the entire seal oil supply to the IHTS sodium would have consequences for the IHTS heat transport capability no more severe than those evaluated in Section 15.3.2.2 (Single Intermediate Loop Pump Seizure), Section 15.3.3.5 (Intermediate Loop Pipe Break), or Section 15.7.2.1 (Inadvertent Release of
59 Oil through Pump Seal (PHTS)).

15.7.2.3 Generator Breaker Failure to Open at Turbine Trip

15.7.2.3.1 Identification of Causes and Accident Description

In the event of a turbine trip, the generator load break switch is automatically opened by a signal from the turbine trip logic. The turbine trip logic simultaneously causes the generator field breaker to open regardless of whether or not the generator load break switch opens. A generator load break switch failure can occur from electrical or mechanical failure of the tripping mechanism.

15.7.2.3.2 Analysis of Effects and Consequences

If the generator load break switch fails to open after a turbine trip, a Plant Power Supply lockout is initiated. The lockout initiates the disconnection of the Plant Power Supply by tripping the appropriate 161 KV circuit breaker in the Generating Yard. This causes loss of the Preferred AC Power Supply as described in Section 8.2.1.1. Upon loss of power from the Preferred AC Power Supply, the Normal AC Distribution System and the Safety-Related AC Distribution System, are automatically transferred to one of the Reserve Transformers as described in Section 8.3.1.1.4. The reactor can be shut down with no adverse consequence, as described in Section 15.3.1.5, which evaluated the effects of a turbine trip.

15.7.2.3.3 Conclusions

The consequences of a turbine trip with subsequent failure of the generator load break switch to open is negligible, since one offsite power supply is still available to the AC Power Distribution System.

15.7.2.4 Rupture In the RAPS Cold Box

15.7.2.4.1 Accident Description

The RAPS cold box contains the cryogenic still in which krypton and xenon are extracted from the reactor argon cover gas stream. During normal operation, this stream is collected in the RAPs surge vessel and flows at a controlled rate of 10.0 scfm into the cold box and then through the cryostill. The argon is condensed to a liquid as it passes through the coiled tubing in the cryostill condenser, which is surrounded by liquid nitrogen. For the purpose of the accident analysis, it is conservatively assumed that the reactor has been operating sufficiently long, with gaseous fission products from 1% failed fuel, for steady-state isotopic composition to exist in the cover gas system. It is assumed, also conservatively, that the cryostill has not been off-loaded to the noble gas storage vessel for 1 year (maximum period), and therefore, contains a maximum inventory of radioactivity.

A major rupture of the cryostill could release the liquid argon in the cryostill (including the radioactivity it contains) and liquid nitrogen to the cold box cell atmosphere. Although such a major rupture is not expected, it is assumed to occur.

15.7.2.4.2 Analysis of Effects and Consequences

Following the cryostill rupture, the cold box cell H&V radiation monitor will sense the presence of radioactivity, sound an alarm, initiate a signal which will automatically close the cell H&V vent, close the cold box cell influent and effluent process lines, and open the cold box bypass line. However, the signal does not close the LN_2 supply line.

The volume of nitrogen released to the cell corresponds to 1.5 cf of LN_2 released from the cryostill reboiler, plus 72 lbm (12 gallons) of LN^2 in-flow; after this time, the nitrogen in-flow is automatically valved off by a high cell pressure signal. Thus, LN_2 equivalent to 1935 scf of nitrogen is estimated to be released into the cell at the initiation of the incident. Also released at this time is the liquid argon still bottoms, 1.5 cubic ft, which corresponds to 1190 scf of argon.

The assumed initial condition, then, is that the cryogenic liquids released come instantly to standard temperature but elevated pressure. No allowance is taken for radioactive decay during the pressure-rise period. The total amount of gas released into the cell (whose net volume is 6500 cf) is then 3125 scf. The initial radioactivity inventory is shown on Table 15.7.2.4-1.

No credit is taken for the leak tightness of the RAPS cold box cell. This is an extremely conservative assumption. Additionally, it is assumed that the RCB refueling door is open at the time of the accident. No overpressure of the RCB is assumed since the released gas will now vent out of the H&V exhausts. This scenario is never expected to occur.

15.7.2.4.3 Conclusions

As shown in Table 15.7.2.4-1, the 0-2 hour site boundary whole body doses for the postulated scenario are well below the 10 CFR 100 limit. Therefore, it is concluded that the postulated accident will not result in unacceptable environmental consequences.

TABLE 15.7.2.4.-1

RUPTURE OF THE RAPS CRYOSTILL

Refueling Door Open - No Cell Leak Tightness Assumed

Isotope	Initial Inventory In the Cryostill (Ci)	Radioactivity Released From the Plant In 2 Hours (Ci)	0 to 2 Hours Whole Body Site Boundary Dose (Rem)
Xe133	4.67×10^5	3.92×10^4	1.38
Xe135	8.79×10^4	6.89×10^3	1.33
Kr88	1.66×10^3	1.11×10^2	0.169
Total	5.57×10^5	4.62×10^4	2.88

*There is an additional contribution of 0.09 rem from the daughter product of Kr88, which is Rb88.

TABLE 15.7.2.4-2

DELETED

15.7.2.5 Liquid Radwaste System Failure (Leak or Rupture)

15.7.2.5.1 Identification of Causes and Accident Description

The rupture of one of the liquid waste collection or storage tanks, or evaporators is defined as an "unlikely fault". A postulated failure can be used to evaluate the design bases for the cells and the ventilation system for the building which houses the cells.

The liquid radwaste collection and processing system are located in a non-hardened building which is connected to the Reactor Service Building (see Section 3.A.4, Reactor Service Building).

The intermediate activity level process streams which process the highest activity are located below grade in concrete cells. The floors and walls of the cells are painted with a flexible epoxy coating to prevent leakage of contaminated water to the outside ground water and to facilitate decontamination. The floors of all cells in the basement are protected with a pliable undercoat to prevent in-leakage of ground water to the cell. Each cell is provided with a sump drain and a sump pump which is used to transfer spilled fluids to another tank. The activity of fission and corrosive products is such that the cell is closed during operation.

The postulated tank failure, malfunction, or operator error which results in a spill, is assumed to occur when one of two 20,000 gallon tanks is full. The tank failure was selected as the largest inventory of radioactivity in the liquid waste system. If a failure or malfunction occurs in an evaporator, it will not result in flashing of any liquid which contains radioactivity. The Liquid Radwaste System evaporators are designed to operate at sub-atmospheric pressures, at temperatures of 160°F. At no time is there a higher quantity of radioactivity in an evaporator bottom than there is originally in the collection tanks due to limits on salt concentration in the evaporator bottoms. The evaporator is designed to cause an alarm and to trip appropriate valves to shut off the steam supply in the event of pump, coolant, or heater failure.

Following the assumed failure, the fluid spreads over the surface area of the floor and drains into the sump. The operator is alerted by the low level liquid indicator in the tank and by the level indicator associated with the sump. After the operator is alerted, the liquid in the sump is pumped into another tank thereby eliminating any long duration exposure or release of tritium to the environment.

The floor surface area is 1000 ft^2, the cell volume is 3.8×10^4 ft^3, the radwaste building volume is 7.4×10^5 ft, and the exhaust rate to the atmosphere at a rate of 1.6 air changes per hour.

The activity levels in the liquid are given in Table 11.2-4 of Section 11.2 of the PSAR. There are no gaseous radioactive iodine species which can be released because the fluids used to remove contaminated sodium from components form salts which are stable. Any radioactive inert gas which may have been trapped in the sodium that is eventually reacted with water and processed by the Radwaste System is negligible. This is true because the quantity of these gases dissolved in sodium is small. The spilled fluid contains fission and corrosion products which are not evaporated. Thus, only water vapor containing tritiated water (HTO) can be released in the event that a failure occurs.

15.7.2.5.2 <u>Analysis of Effects and Consequences</u>

<u>Gaseous Release</u>

The highest activity resulting from a radwaste system failure involves collection tank leakage or rupture. 100% of the average annual collection tank inventory of 20,000 gallons of water contains 1.44×10^5 Ci of tritium as HTO. The build-up of tritium in the recycle liquid over the 30 year life of the plant is a function of: (1) input from the primary sodium removal system, (2) radioactive decay, (3) retention of a portion of the influent in the evaporation bottoms which are transferred to the solid waste system for immobilization, and (4) the release of a fraction of the storage tank inventory to the cooling tower water blowdown. The value of 1.44×10^5 was conservatively estimated by using a loss of only 4700 gallons per year out of the 40,000 gallons of storage capacity.

A conservative analysis was made to calculate the off-site doses if 10% of the tritium contained in the spilled liquid radwaste was released to the atmosphere in two hours following the spill. This highly conservative assumption resulted in a Beta Skin Dose of 4.47×10^{-8} REM and a whole body inhalation dose of 3.7×10^{-6} REM, at the site boundary. The potential beta skin and whole body doses at the LPZ are 0.68×10^{-9} REM and 3.05×10^{-7} REM, respectively.

<u>Liquid Release</u>

For conservatism, the event has been analyzed assuming no credit for the floor, drains or operator actions.

As pointed out in Section 2.4.13, accidental liquid spills are not seen as posing a danger to present or future groundwater users in that the ultimate destination of contaminants in the groundwater would be the Clinch River. Movement of groundwater is from groundwater ridges to adjacent groundwater lows. Review of Figures 2.4-68 and 2.4-69 lends support to the assumption made of the cooling tower blowdown discharge point as a conservative assumption (in terms of location to nearest intake) for entry of spillage into the Clinch River. The analysis has assumed 80% of the tank activity is discharged and that this release occurs over a two hour period. For conservatism, no decay of activity levels between the time of tank failure and entrance into the Clinch River is assumed in the analysis although cation exchange characteristics are known to exist in the earth at the site. In addition, no credit was assumed for building retention, plateout or condensation on walls. As discussed in section 15.7.2.5, the fluids used to clean components such as the IHX will form salts which are non-volatile in nature. Accordingly, the analysis assumes all activity released is retained in the fluid spilled from the tank.

The CRBRP Environmental Report (Appendix A to section 10.3) has developed dilution factors for discharge of cooling tower blowdown into the Clinch River. Compared to the blowdown discharge, the spillage of radwaste contents into the Clinch River can be expected to exhibit (a) a decreased jet velocity and (b) a higher temperature (assuming no cooling while in the groundwater). ER analysis estimates a dilution factor of 0.05 during typical stet river flow conditions for an area within 60 feet of the blowdown discharge point. As discussed in Sections 2.4 and 11.2, the intake for potable water nearest to the discharge point of the CRBRP is for the K-25 facility (Clinch River Mile 14.4) more than one mile and one half from the discharge point. Assigning a dilution factor of 0.05 to the K-25 intake vicinity (greater than two orders of magnitude further downstream than the 60 foot area) as a result of spillage entry into the river is expected to be an extremely conservative approach even allowing for the differing characteristics of blowdown and tank spillage.

15.7.2.5.3 Conclusions

Gaseous Release

Based on the analysis described in the preceding section (15.7.2.5.2), the Beta Skin Dose (4.47×10^{-8} REM) and the Whole Body Inhalation Dose (3.7×10^{-6} REM), are decades below the 10CFR20 guidelines. The conclusion is that there are no adverse consequences to the health and safety of the public resulting from this accident.

Liquid Release

Table 15.7.2.5-1 provides estimated activity concentrations for the postulated event. The first column provides a maximum environmental impact (dilution factor=1.0) and would be characteristic only of the very immediate area of discharge. The second column of the table provides conservative estimates of activity concentrations in the vicinity of the intake for the K-25 facility (Dilution factor=0.05).

As described in Section 11.2, the radwaste tanks will be classified as quality group D. In accordance with Reg. Guide 1.26, current operating plant technical specifications are established on the basis of meeting 10CFR20 site boundary exposure limits of 0.5 rem to the whole body or its equivalent to any part of the body as a result of a postulated failure. Table 15.7.2.5-2 provides a comparison between the maximum (DF=1) resulting doses and appropriate federal regulations. The results show the conservative estimates to be well within 10CFR20 limits.

Table 15.7.2.5-1

Clinch River Water Concentration Estimates
Following Failure of Liquid Radwaste
Collection Tank

Isotope	Entry Point+ of Clinch River (uC_i/ml)	Clinch River++,* Mile 14.4 (uC_i/ml)
H-3	4.34×10^{-8}	2.17×10^{-9}
Na-22	2.48×10^{-7}	1.24×10^{-8}
Na-24	6.94×10^{-8}	3.47×10^{-9}
Cr-51	2.11×10^{-5}	1.06×10^{-6}
Mn-54	1.45×10^{-4}	7.24×10^{-6}
Co-58	9.28×10^{-5}	4.64×10^{-6}
Fe-59	7.07×10^{-7}	3.54×10^{-8}
Co-60	1.42×10^{-4}	7.10×10^{-6}
Sr-89	8.08×10^{-5}	4.04×10^{-6}
Y-89M	8.08×10^{-5}	4.04×10^{-6}
Sr-90	5.82×10^{-5}	2.91×10^{-6}
Y-90	5.82×10^{-5}	2.91×10^{-6}
Y-91	2.41×10^{-5}	1.20×10^{-6}
Zr-95	4.51×10^{-5}	2.26×10^{-6}
Nb-95	4.51×10^{-5}	2.26×10^{-6}
Mo-99	5.08×10^{-6}	2.54×10^{-7}
Ru-103	6.26×10^{-6}	3.13×10^{-7}

+Assumed DF=1.00
++Assumed DF=0.05
*K-25 intake vicinity
Typical Summer Flow Conditions (4777 CFS)

Table 15.7.2.5-1(continued)

Isotope	Entry Point of Clinch River	Clinch River Mile 14.4
Ru-106	4.86×10^{-5}	2.43×10^{-6}
Rh-106	4.86×10^{-5}	2.43×10^{-6}
Ag-111	1.61×10^{-5}	8.05×10^{-7}
Sb-125	6.45×10^{-8}	3.22×10^{-9}
Te-129M	4.98×10^{-4}	2.49×10^{-5}
Te-129	4.98×10^{-4}	2.49×10^{-5}
I-131	2.98×10^{-6}	1.49×10^{-7}
Te-132	3.54×10^{-4}	1.77×10^{-5}
I-132	3.54×10^{-4}	1.77×10^{-5}
CS-134	2.98×10^{-7}	1.49×10^{-8}
CS-136	1.37×10^{-6}	6.84×10^{-8}
CS-137	1.10×10^{-5}	5.50×10^{-7}
BA-140	3.27×10^{-5}	1.64×10^{-6}
LA-140	3.27×10^{-5}	1.64×10^{-6}
Ce-141	5.39×10^{-5}	2.70×10^{-6}
Pr-143	2.84×10^{-5}	1.42×10^{-6}
Ce-144	3.85×10^{-5}	1.92×10^{-6}
Pr-144	3.85×10^{-5}	1.92×10^{-6}
N_d-144	1.19×10^{-5}	5.96×10^{-7}
Pm-147	2.19×10^{-5}	1.10×10^{-6}
Eu-155	2.11×10^{-6}	1.06×10^{-7}
Ta-182	1.74×10^{-5}	8.68×10^{-7}
Pu-238	1.30×10^{-7}	6.50×10^{-9}
Pu-239	2.36×10^{-8}	1.18×10^{-9}

Table 15.7.2.5-1(continued)

Isotope	Entry Point of Clinch River	Clinch River Mile 14.4
Pu-240	3.10×10^{-8}	1.55×10^{-9}
Pu-241	1.76×10^{-6}	8.80×10^{-8}
Pu-242	6.70×10^{-11}	3.35×10^{-12}

Table 15.7.2.5-2

Maximum Exposure For an Individual Following Postulated Failure of the Liquid Radwaste Tank

Organ	Dose (Rem)+	Permissible 10CFR20 Exposure
Whole Body	0.039	0.500
Thyroid	0.078	1.500
Bone	0.054	3.000
G.I. Tract	0.039	1.500

+Individual assumed to remain at site boundary and exposed to liquid (including drinking) environments.

15.7.2.6 Failure (Leak or Rupture) in the EVST NaK System

15.7.2.6.1 Identification of Causes and Accident Description

44 The EVST NaK systems operate at low pressure (~100 psig) and temperature (approximately 350°F) and utilize all welded construction. With the exception of valve bellow failures discussed in Section 15.7.1.6, leaks or ruptures are not expected to occur. The accidents discussed below are leaks occurring in various locations from unidentified causes.

15.7.2.6.2 Analysis of Effects and Consequences

44 Analysis of specific leaks in the EVST NaK systems will be performed when leak volumes and system design and arrangement are finalized. There are three general locations in which NaK leaks can be postulated, in the inerted cells containing the EVS sodium components, in the air-atmosphere cells containing the EVST airblast exchangers (ABHX's) and the EVST National Draft Heat Exchanger (NDHX), and in the ABHX's, and NDHX themselves. Leaks occurring in the inerted cells will result in consequences less severe than the sodium leaks
44 postulated in Section 15.6.1.2 due to the smaller NaK inventory (smaller potential spill), the absence of radioactivity in the NaK, and the lower NaK temperature (350°F NaK vs 500°F sodium). Leaks in the air atmosphere cells will result in a localized fire which will be extinguished by the fire protection system (Section 9.3.1). Leaks within the EVST, ABHX's or NDHX are contained within, and mitigated by, the units themselves, which include a self-contained catch pan under the tube bundle and leak detectors. Indication of a leak results in loop shutdown, isolation of the air side of the heat exchanger
44 by closure of inlet and outlet dampers, and nitrogen flooding of the unit.

15.7.2.6.3 Conclusions

Since the NaK is nonradioactive, no potential accident can result in activity release from the plant. Structural design and component arrangement insure that a failure in one of the EVST NaK circuits does not affect the
44 operability of the other circuits. The conclusion is that there are no adverse consequences to public safety or to EVST cooling from potential leaks in the EVST NaK System. 2

15.7.2.7 Leakage From Sodium Cold Traps

15.7.2.7.1 Identification of Causes

Sodium cold traps are used in three locations in the CRBRP auxiliary systems: 1) the primary sodium cold traps, which are liquid cooled by circulation of NaK through a welded jacket on the outside of the trap; 2) the EVS sodium cold trap, which is nitrogen cooled; and 3) the intermediate sodium cold traps, which are air cooled. All traps collect and contain radioactive material; the primary traps contain corrosion and fission products and tritium, the EVS trap contains the same but in much smaller quantity, and the intermediate traps contain tritium. The quantities of radioactive material are listed in Table 11.1-9 and in Section 11.5.3. All traps are of basically similar construction, consisting of an economizer, which lowers the temperature of incoming sodium, and a crystallizer, in which the impurities (including the radioactive material) are precipitated and collected. Potential release of significant amounts of activity (larger than the normal concentration in the sodium being purified) is possible only due to a leak in the crystallizer.

15.7.2.7.2 Analysis of Effects and Consequences

The operating principle of all the sodium cold traps is the same. The sodium flows down around the periphery of the crystallizer, and is cooled until the impurities precipitate as a solid. The tritium, as sodium tritide, is collected as a solid. The precipitate is caught and collects on wire mesh in the central portion of the crystallizer. The crystallizer operates at low temperature, with the sodium cooled to a normal range of $250-300^{\circ}F$. The result of a leak in the shell is basically a leak of sodium with its inlet concentration of activity; the bulk of activity, accumulated during operation, remains as a solid distributed throughout the mesh in the central portion of the trap.

During operation of the primary cold traps, a failure of the crystallizer will not result in a sodium leak due to the enclosing NaK jacket. The NaK pressure is maintained higher than that of the sodium to insure NaK in-leakage into the sodium in the event of a leak. Loss of inventory in the NaK system will signal the failure, and the cold trap will be isolated, removed, and replaced.

During operation, the EVS and intermediate sodium traps are cooled by gas, and air, both at $>100^{\circ}$ lower than the freezing point of sodium. Should a leak in the crystallizer occur, the leaking sodium will oxidize and solidify. Sodium vapor carried by the cooling stream will activate leak detectors located in the gas/air outlet of each trap. The trap will then be isolated and removed and replaced. Postulated loss of the inventory in the cold traps during plant operation will result in off-site doses well within 10CFR100 guidelines as shown by the analysis of the hypothetical event in Chapter 15 Appendix A.

When a cold trap is to be removed, it is isolated from the processing system and the sodium allowed to freeze. The trap is then cut out and the openings capped and seal-welded. Once removed, all traps will remain frozen without supplementary cooling, thus leakage and radiological release after removal is not a practical concern. Prior to final disposal, the 8 primary and EVS sodium cold traps will be stored in a shielded cask. Clean-up and solid waste disposal of the spilled sodium from a cold trap is typical of spills from other sodium components. The method of disposal of the traps and their contents has not yet been determined. 8

An analysis of a design basis cold trap fire has been performed in response to NRC ER Question 000.28. The analysis of this fire for PSAR considerations using conservative rather than realistic (ER) assumptions would be identical except that, (a) the cold trap inventory is increased based on the assumption of operation with 1% failed fuel instead of the 0.5% used in the ER, and (b) dispersion based on PSAR accidental release meterology. The leak rate of aerosol from the RCB, as stated in the response to NRC ER Question 000.28, is based on a 1 psig containment overpressure and was calculated to be 0.032% vol/day. Table 15.7.2.7-1 presents the boundary and Low Population Zone (LPZ) doses following an assumed cold trap fire. All of these values are orders of magnitude below the limits of 10CFR100. 25

15.7.2.7.3 Conclusion

The consequences of leakage from any of the three sodium cold trap locations in the CRBRP auxiliary systems has no effect on either reactor safety or the safety of the public. 8

TABLE 15.7.2.7-1

OFF-SITE DOSE RESULTING FROM A POSTULATED COLD TRAP FIRE

Organ	2 Hour Dose (Rem) At Site Boundary (0.42 Mile)	30 Day Dose (Rem) LPZ (5.0 Miles)
Bone	1.02×10^{-3}	3.03×10^{-4}
Lung	7.51×10^{-4}	2.22×10^{-4}
Thyroid	4.17×10^{-5}	1.23×10^{-5}
Whole Body	7.81×10^{-5}	2.31×10^{-5}
Skin	5.13×10^{-7}	1.51×10^{-7}

15.7.2.8 Rupture in RAPS Noble Gas Storage Vessel Cell

15.7.2.8.1 Accident Description

The RAPS noble gas storage vessel (NGSV) normally contains radioactive gas which is off-loaded annually from the RAPS cryostill. It contains mainly argon (including argon-39) but krypton and xenon isotopes, both stable and radioactive, are also present. The gas is bled slowly from the vessel into CAPS so that its pressure normally decreases over the annual period. A rupture of this vessel or of associated piping and components could release radioactive gas at above-ambient pressure into the noble gas storage vessel cell. Although such a rupture is not expected, it is assumed to occur. For the purpose of the accident analysis, it is conservatively assumed that the reactor has been operating sufficiently long with gaseous fission products from 1% failed fuel for steady-state isotopic composition to exist in the cover gas system. One years' accumulation of noble gas isotopes, under that condition, that had accumulated in the cryostill has been off-loaded to the noble gas storage vessel. Furthermore, it is assumed that some unspecified maintenance operation has required that the new fresh cryostill charge also be off-loaded to the storage vessel, this in quick sequence, so that the storage vessel contains two charges and is approximately at maximum pressure.

Assuming the vessel (260 actual cubic feet volume) is at 1 atmosphere pressure absolute before the two cryostill off-loadings (1.5 cubic feet of liquid argon each), it will contain 2640 scf of gas prior to the accident.

15.7.2.8.2 Analysis of Effects and Consequences

Following the storage vessel rupture, the NGSV cell H&V radiation monitor will sense the presence of radioactivity, sound an alarm, and initiate a signal which will cause the cell vent line to CAPS to close. The cell (whose net volume is 3560 actual cubic feet including the vessel volume) pressure will then increase to 9.8 psig, assuming instant temperature equilibration to ambient. The initial radioactivity inventory is shown in Table 15.7.2.8-1.

The consequences are calculated assuming the NGSV cell is not a leakage barrier which is an extremely conservative assumption. For this assumption, the radioactivity is assumed to be released directly to the RCB. Additionally, it is assumed that the RCB refueling door is open at the time of the accident. No overpressure of the RCB is assumed since the released gas will now vent out of the H&V exhausts. This scenario is never expected to occur.

15.7.2.8.3 Conclusions

As shown in Table 15.7.2.8-1, the 0 to 2 hour site boundary whole body doses for the postulated scenario are well below the 10 CFR 100 limit. Therefore, it is concluded that the postulated accident will not result in unacceptable environmental consequences.

TABLE 15.7.2.8.-1

RUPTURE OF THE NOBLE GAS STORAGE VESSEL

Refueling Door Open - No Cell Leak Tightness Assumed

Isotope	Initial Inventory in the NGSV (Ci)	Radioactivity Released From the Plant in 2 Hours (Ci)	0 to 2 Hours Whole Body Site Boundary Dose (Rem)
Xe133	4.67×10^5	3.92×10^4	1.38
Xe135	8.79×10^4	6.89×10^3	1.33
Kr88	1.66×10^3	1.11×10^2	0.169
Total	5.57×10^5	4.62×10^4	2.88

*There is an additional contribution of 0.09 rem from the daughter product of Kr88, which is Rb88.

15.7.2.9 Rupture in the CAPS Cold Box

15.7.2.9.1 Accident Description

The CAPS cold box contains two charcoal delay beds in series, which adsorb xenon and krypton from the process gas stream before it is discharged to H&V. During normal operation, this stream is collected in the CAPS surge vessel and flows at a controlled rate of approximately 38 scfm into the cold box and then through the delay beds. The krypton and xenon isotopes, both stable and radioactive, are adsorbed on the beds which are cryogenically cooled by injecting LN_2 into the influent stream of each one. A rupture of the charcoal delay beds is assumed to occur, which results in radioactivity release.

15.7.2.9.2 Analysis of Effects and Consequences

Following the rupture of the delay beds, the radioactive and nonradioactive gases adsorbed on the beds are conservatively assumed to immediately desorb. This releases approximately 3300 standard cubic feet of gas to the cell and the initial inventory of radioisotopes shown in Table 15.7.2.9.-1. The initial inventory is not calculated based on the CAPS design or expected inputs discussed in Section 11.3 since this would give a CAPS annual average inventory which is not conservative for accident analysis. Instead, the inventory was calculated assuming the plant is refueling, which maximizes the inventory for normal operation.

After activity is released to the cell, the cell H&V radiation monitor detects the activity, sounds an alarm, and automatically closes the cell H&V vent line, the tritium-water removal unit drain line, and stops process gas flow to the cold box cell components. The LN_2 line supplying liquid nitrogen to the delay beds is also assumed to rupture and flow 72 lbm (12 gallons) of LN_2 into the cell before it is automatically shut off due to high cell pressure. (This is conservative since the LN^2 supply would already be isolated due to the cell pressure caused by the 3300 scf of gas previously released to the cell). The 72 lbm of LN_2 will vaporize to 1000 scf of GN_2.

The initial condition, then, is that the gases released to the cell come instantly to standard temperature but elevated pressure. No allowance is taken for radioactive decay during the pressure-rise period. The total amount of gas released to the cell (whose net volume is 9500 cf) is 4300 scf.

All of the radioactivity on the delay beds is assumed to be immediately released from the RSB following the accident. The scenario is extremely conservative and is never expected to occur.

15.7.2.9.3 <u>Conclusions</u>

As shown in Table 15.7.2.9-1, the 0 to 2 hour site boundary whole body doses for the postulated scenario are well below the 10 CFR 100 limit. Therefore, it is concluded that the postulated accident will not result in unacceptable environmental consequences.

TABLE 15.7.2.9-1

RUPTURE OF THE CAPS DELAY BEDS

Delay Beds Inventory Immediately Released from the RSB

Isotope	Initial Inventory on the Delay Beds (Ci)	Radioactivity Released From the Plant In 2 Hours (Ci)	0 to 2 Hours Whole Body Site Boundary Dose (Rem)
Xe133m	33	33	1.1×10^{-3}
Xe133	4.1×10^{3}	4.1×10^{3}	0.14
Total	4.1×10^{3}	4.1×10^{3}	0.14

15.7.3 Extremely Unlikely Events

15.7.3.1 Leak in a Core Component Pot

15.7.3.1.1 Identification of Causes and Accident Description

A leak in the core component pot (CCP) can be caused by a number of conditions that are all of low probability. These include, defects that were not found during inspection following manufacture, failure subsequent to damage caused by an accident in which pot damage was not suspected or uncovered, accidents resulting directly in failure of the pot (none have thus far been identified) or failure of the pot due to corrosion. In addition, the leak would have to be undetected to cause an accident.

Operations with high-powered spent fuel assemblies (20 kW maximum,
44 i.e., design decay heat load of the EVTM) are in a sequence that requires a CCP 29
to be used to transport a new fuel assembly to the reactor vessel, just in advance of its being used to transport a spent fuel assembly from the reactor vessel to the EVST. In the process of transferring the new fuel assembly, the operator would be made aware of any significant pot leakage by observing dripping sodium through the EVTM view port, and by the fact that the grapple load cells would indicate a significant reduction in load between the time the pot was picked up at the EVST and when it was lowered into the reactor.

44 Operations with lower-powered spent fuel assemblies (6 kW) are con- 29
ducted between the EVST and fuel handling cell without first using the CCP for transporting a new fuel assembly immediately before. In this case, the CCP would have been used for transfer of the same spent fuel assembly from the reactor to the EVST between 3 and 9 months earlier. However, the time required for transfer between the EVST and FHC is shorter and the decay power is much lower, so only the limiting case of a 20 kW assembly transferred between the reactor and EVST is analyzed here.

Periodic inspection of CCP's and features provided to observe CCP's in the FHC and near the bottom of the EVTM enable the detection of a leaking CCP before a gross failure could occur.

The main barrier inhibiting this accident is the high quality of the CCP's constructed to the requirements of ASME Code Section III, Class 2. The lines of defense against the undetected occurrence of a major sodium leak in the CCP are shown in the Safety Assurance Diagram Figure 15.7.3.1-1.

In order to evaluate the full consequences of this event, the potential accident sequence of a major CCP leak has been extended up to its termination in a safe condition. The sequence of events following a major CCP leak and the lines of defense between them are schematically shown in Figure 15.7.3.1-2, which is a continuation of the Safety Assurance Diagram given in Figure 15.7.3.1-1. 29

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The postulated sequence starts with a major CCP leak, leading to a complete loss of CCP sodium while the EVTM is at the reactor and is in the process of hoisting a 20 kW spent fuel assembly in a CCP into the EVTM cask. The leak in the CCP is assumed to occur instantaneously at the moment when the CCP is being raised above the liquid sodium level in the reactor vessel. From this moment on, the CCP will normally not be under sodium again until it is deposited by the EVTM in the EVST. The time span between a CCP surfacing from the reactor sodium and submerging under EVST sodium is 56 minutes, assuming normal refueling operations.

The time-phased sequence of events following this initial condition are listed in Table 15.7.3.1-1 together with the normal refueling operation steps. This sequence assumes that the operators have no knowledge of the CCP leak, do not take corrective or accelerating actions, and perform normal refueling operations.

47 The warning the refueling operators would receive of an off-
normal condition would consist of a grapple load cell signal indicating a
load reduction. Administrative procedures based on this signal would
direct the EVTM operator to return the EVTM to the nearest fuel transfer
port and immerse the CCP under sodium as quickly as possible. Sufficient
time (about 30 minutes) would be available, after the low load cell sig-
nal indication, before the cladding of all fuel rods in the mid-plane of
the fueled region could melt. If the EVTM operator does not respond to
47 this signal, the EVTM would continue its travel through RCB and RSB to
the EVST, mate with the floor valve at a fuel transfer port, and discharge
the CCP into an EVST storage tube. At approximately 56 minutes after
initiation of the event, the fuel assembly would be cooled by immersion
of the CCP in the EVST sodium. Extensive clad melting in the fueled
region may have occurred at that time, but the CCP will retain its struc-
tural integrity and can be handled by the EVTM grapple.

A continuation of the event sequence beyond the normal time
required for submersion of the CCP under sodium has been hypothesized.
For this purpose, all administrative controls and operator actions follow-
47 ing the load reduction indications were ignored. The physical lines of
defense against fuel breakup, loss of structural integrity of the CCP,
and loss of EVTM containment are shown in Figure 15.7.3.1-2 and are
discussed in the next section. 29

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15.7.3.1.2 Analysis of Effects and Consequences

Thermal Consequence Analysis

59 A. Thermal Model

The thermal calculations for this accident were performed using the computer codes TAP-4F (Thermal Analyzer Program) and DEAP (Differential Equation Analyzer Program) which are listed in Appendix A of the PSAR.

The thermal analysis network modeling a spent fuel assembly in a core component pot surrounded by the EVTM cold wall is shown in Figure 15.7.3.1-3.

The analysis used the following assumptions as input:

	Fuel assembly decay power	20 kW
44	Heat generation within fuel assembly	86%
44	Heat generation outside fuel assembly (by gamma heating)	14%
	Air flow for coldwall cooling	4,600 lb/hr
	Emissivity for fuel and CCP	0.4
	Emissivity for EVTM coldwall	0.2*

59 B. Thermal Analysis - CCP Submersed After Normal Transfer Time

The analysis results are shown in Figures 15.7.3.1-4 and 15.7.3.1-5. Figure 15.7.3.1-4 is a plot of the maximum transient temperatures of the center fuel rod cladding (hottest rod), cladding of a fuel rod in the outer row, the fuel assembly duct, the core component pot, and the coldwall. The low decay heat flux in spent fuel, as compared to the much larger heat flux during reactor operation, produces temperature differentials of less than 20^{o}F between fuel rod center and cladding. As soon as the sodium has drained below the level of the fueled region, the temperatures of the fuel assembly and CCP rise, and reach steady-state values after about one hour. Clad melting in the center fuel rod starts after about 17 minutes. The clad melting zone progresses to fuel rods in the outer row in about 30 minutes. After about one hour, 90% of the clad in the fueled region has melted, and the fuel duct reaches the melting point in a localized circular zone.

*Coldwall emissivities are normally expected to be 0.7 or larger. A degradation of coldwall emissivity to 0.2 was postulated since the coldwall was assumed to be covered with a film of recondensed sodium due to the accident. This is a conservative assumption resulting in higher fuel and CCP temperatures. 29

It should be noted that not the fuel assembly but the core component pot is attached to the EVTM grapple. The fuel assembly, standing unrestrained in the pot, experiences only stresses due to its own weight, and due to thermal gradients. The radial clearance between fuel assembly hexagonal duct corners and CCP is about 0.8 in. The CCP temperature after one-hour is about 2050^{o}F, well below the melting point of stainless steel, and has almost reached the steady-state value.

Figure 15.7.3.1-5 gives the axial and radial steady-state temperature distribution along a vertical cut through a part of the EVTM. All maximum temperatures appear at the midplane of the fueled region.

The analysis indicates the steady-state temperatures (see Summary Table 15.7.3.1-2) for the fuel material in a spent fuel assembly in a CCP without sodium in the EVTM are well below the melting point for the mixed oxide fuel.

The isothermal lines near the coldwall show that the nearest seals (lower coldwall seals) will stay below 100^{o}F. All volatile fission products released from the fuel rods into the EVTM will therefore be contained in the EVTM. The maximum pressure in the EVTM due to argon gas heating by the dry fuel assembly, and due to release of fission gas and helium from the fuel
44 rods, could amount to about 26 psia, well within the design pressure of 30 psia.

The results of this analysis indicate the following consequences for a "dry" CCP in the EVTM, if the CCP is not submerged under sodium within the normal CCP transfer time (56 minutes):

(1) release of volatile fission products from all fuel rods to the EVTM containment

(2) extensive fuel rod cladding melting

(3) localized fuel assembly duct melting

(4) no fuel melting

(5) no CCP melting; the CCP can support the fuel assembly

(6) no seal overheating; the EVTM can contain the fission products with only limited diffusion of activity resulting.

59 C. Thermal Analysis - CCP Submersion Delayed

From these consequences it was concluded that the event sequence could be safely terminated and no public safety hazard would ensue, even if all lines of defense preventing this event (see Figure 15.7.3.1-1) were rendered ineffective.

An additional investigation was carried out to examine the consequences of this event if the normally expected CCP transfer time from reactor sodium to EVST sodium (56 minutes) were to be prolonged. Two potentially worse cases were postulated and analyzed, again to explore the worst potential consequences of this event. The two cases were based on 29
the following assumptions:

(1) fuel break-up and collapse in a packed-bed configuration

(2) relocation of fragmented fuel outside of the fuel assembly in the CCP bottom.

A scenario has been postulated in which fuel pellets, stripped of their cladding, could break-up in smaller pieces, collapse, and form a packed-bed type structure. The restructured fuel in its new configuration could have a higher energy density than in its original geometry as rods, depending on the size and packing of the fuel particles. This, in turn, could cause the temperature of the fuel assembly duct to exceed the melting point in a localized zone, and could result in a loss of structural integrity of the fuel assembly.

59 1. Improbability of Fuel Collapse

Results of in-pile and out-of-pile experiments with LMFBR fuel assemblies subjected to high temperatures support the above-described scenario as being conservative. Fuel behavior tests performed in the transient reactor test (TREAT) facility at ANL, in support of the analyses for the hypothetical loss-of-core-coolant accident, indicated that fuel pellets did not fall apart once the cladding had melted and gave no further support. These tests (Reference 1) were performed with pre-irradiated fuel and showed that the fuel pellets sintered together with a strong, dense column formed by the equiaxed region. The fuel rods retained their identify as columns and bowed, rather than crumbled as individual pellets or pieces. Similar test
59 results were obtained when new fuel was subjected to loss-of-coolant experiments in the TREAT reactor. Although fuel cladding had melted off in these experiments, the fuel rod pellets remained stacked at termination of the transient (Reference 2). Intact fuel columns were also observed in several loss-of-coolant experiments performed as in-pile transients on new fuel in the Reactor Centrum Nederland (RCN) (References 3, 4, and 5).

Out-of-pile tests ("dry capsule" experiments), reported in Reference 6, also showed that the fuel column of an irradiated pin heated to its solidus held together after the cladding melted, and remained essentially intact even after considerable bowing and buckling.

During reactor operation, a break-up of the solid fuel occurs due to high thermal gradients in the fuel during reactor power transients, and due to changes in the grain structure of mixed oxide fuel. It is well established that initial fuel break-up is followed or accompanied by a crack healing process whose effectiveness is a function of fuel temperature and reactor operating time. Research at ANL has shown that uranium oxide, for example, exhibits crack healing when exposed to temperatures above 2900^{o}F for a period of 48 hours, and recovers its as-fabricated strength. This crack healing does not occur as a consequence of solidification of molten fuel, but proceeds by a mass-transport mechanism involving grain growth and diffusion (Reference 7).

The maximum rate of temperature increase in a fuel rod during the postulated accident was calculated to be about 2^{o}F/sec. This is less than 1/30 of the temperature rates representative of in-core, loss-of-flow accidents and their experimental simulation (References 8 and 9). The lower heating rate of the fuel rods has the following consequences:

1. The fuel pellets experience less severe temperature gradients, reducing the potential for thermal shock induced cracks.

2. The less rapid heating rate allows for redistribution of fission gas trapped in grain boundaries and for gas pressure equalization within the entire fuel rod. (Reference 10)

From the heat transfer analysis of the postulated accident and from the above considerations, the following observations are derived:

1. The maximum steady-state fuel temperature in the center fuel rod and in all other fuel rods is well below the fuel melting point.

2. Clad melting could occur over approximately 36 in. length of the fuel rods. The molten cladding will solidify near the lower axial blanket where the temperatures are below the melting point of stainless steel, as evident in Figure 15.7.3.1-5.

3. A collapse of the fueled region with resulting dispersal and relocation of fuel fragments will not occur
59 after the cladding of the fueled region has melted.

Despite these considerations, the accident sequence has been extended and fuel fragmentation, followed by fuel collapse into a packed-bed structure, has been hypothesized. The packed bed was postulated to be supported by the lower axial blanket, since cladding temperatures in the lower axial blanket are substantially below the melting point, see Figure 15.7.3.1-5. This region will therefore retain its structural integrity and is the most likely place for fuel to collect if the pellets do fragment and collapse.

The fragments were postulated to be all of equal size, with a representative diameter of 0.1 in. This implies the break-up of each fuel pellet into 12 spherical pieces with equivalent mass. The formerly 36 in. long fueled region consisting of stacked fuel pellets (encased in a cladding tube) could thus be compacted to a length of 28 in., consisting of a "pebble bed" of fuel particles. The packed-bed fuel configuration could lead to a temperature increase due to the higher energy density and reduced effective conductivity. The latter would cause the fuel to retain more heat and transmit less to the fuel assembly duct.

The transient temperature distribution for this hypothetical fuel configuration is shown in Figure 15.7.3.1-6. It can be noted that, due to the reduced effective fuel conductivity, the fuel duct temperature near the midplane of the fueled region is actually lower than in the case when fuel pellets remain stacked. After one hour, this effect is counter-balanced by the higher temperature of the compacted fuel fragments. If the event were not terminated at the normally expected time (56 minutes) by submersion of the CCP under EVST sodium (see Table 15.7.3.1-1) the fuel assembly duct would start to melt in a circumferential zone after about 1.05 hours (63 min.). The transient axial temperature distribution plotted in Figure 15.7.3.1-7, shows that the high temperatures
are axially confined to the fueled region and extend only partially into 29
the axial blankets.

59 3. Thermal Analysis - Fuel Redistribution Outside Duct

The accident sequence has been further extended to investigate the consequences of a loss of fuel assembly integrity. It was hypothesized that fuel particles might leave the fuel assembly duct, fall down in the annular space between hexagonal housing and circular CCP, and accumulate at the CCP bottom. Due to considerable geometrical distortion of the fuel assembly near the fueled region (from overtemperature during this event) and the presence of solidified, previously molten, material from the fuel assembly duct and cladding near the (colder) CCP wall, only a restricted passage for fuel particles will exist. Only a small amount of fuel material would therefore be expected to fall to the bottom of the CCP. 25% of the fuel material was judged to be the upper limit of this amount. However, the value was varied up to 100%, to show the effect of this parameter.

The calculated peak transient temperatures in the fuel, CCP, and nearest seal are plotted in Figure 15.7.3.1-8 for these amounts of fuel present at the CCP bottom.

After an initial drop of the fuel temperature due to the fuel relocation in a cold area, the fuel temperature rises slowly. The peak CCP temperature at the CCP side and bottom, and the temperature of the nearest seal (lower cold wall) also rise slowly. The calculations show that at about 1.5 hours later initiation of the event, i.e., after loss of sodium from the CCP, the transient temperatures in the fuel and CCP reach steady-state conditions if 25% of the fuel fragments are accumulated at the CCP bottom. The steady-state temperatures are as follows:

Center of Fuel	$3140^{o}F$
CCP, Bottom	$1890^{o}F$
CCP, Side	$1865^{o}F$
Lower Cold Wall Seal	$260^{o}F$

The movement of fuel particles from the original fuel region within the fuel assembly to the CCP bottom has the beneficial effect of lowering the energy density of the heat source and thereby lowering the temperatures of the fuel and its surrounding. This explains the lower temperatures when 25% of the fuel has accumulated in the CCP bottom. A stress analysis indicated that the stresses in the CCP, due to support of its own weight and that of the fuel assembly, are very low. The tensile stress in the tubular part of the CCP is 230 psi, the compressive stress at the CCP bottom is 700 psi. This compares to an ultimate strength of about 6000 psi for the CCP material (SS 304) at $1900^{o}F$.

Fission Product Release Analysis

The analysis and the supporting temperature data presented above show that the postulated accident will not lead to any fuel melting, but 29 could lead to extensive clad melting. It can be conservatively estimated that most of the fission products which are volatile in the temperature range of about 2800° to 3500°F are released into the EVTM. This temperature range corresponds to the maximum axial steady-state temperature which the fuel rods in the assemblies reach, dependent on their radial location (see Figure 15.7.3.1-5).

Table 15.7.3.1-3 lists those fission product elements contained in a fuel assembly of the equilibrium core at the end of cycle which are in the molten or vapor phase below 3500°F. The entire isotopic content of fission products is given in Table 12.1-35. The fission products of Table 15.7.3.1-3 are assumed to be released into the EVTM either partially or completely, depending on their melting points and partial pressures.

The maximum cold wall temperature of the EVTM was calculated to be 435°F (see Figure 15.2.3.1-5). This "hot spot" is at an axial location coresponding to the midplane of the fueled region in the fuel assembly. The nearest seals are 6.3 ft downwards at the lower end of the cold wall near the air inlet module. These seals will not reach temperatures higher than 260°F during this accident. The elastomer seals will contain the radioactive fission products in the EVTM. Permeabilities of elastomeric seals have been experimentally determined up to 300°F (see Reference 1 of Section 15.5.2.3).

It was therefore concluded that all fission products which are in the liquid or gaseous phase above 260°F are plated out on the cold surfaces in the EVTM, specifically at the cold wall and/or near the seals. Only fission products which are in the liquid or gaseous phase at or below 260°F were considered to leave the double seals by diffusion.

The diffusion rates of fission products from the EVTM to the RSB/RCB are given in Table 15.5.2.3-3. In determining these diffusion rates, it was assumed that about 15% of all EVTM seals are at a temperature of 300°F and 85% at 150°F. This assumption is conservative with respect to the postulated accident, since only one set of seals, representing about 1% of all EVTM seals, could exceed a temperature of 150°F. The diffusion rates of Table 15.5.2.3-3 are therefore higher than those which would be expected as a result of the accident discussed here. Fission products other than those listed in Table 15.5.2.3-3, but which are volatile at EVTM seal temperatures, were discussed in the response to Question 001.212. As shown there, only Cs and Rb need to be considered, yet the radioactivity contribution of all Cs and Rb isotopes combined, passing through the hottest EVTM seals, is smaller than that of all other volatile fission product isotopes (i.e. mainly of Xe 133, I131, and I132) by a factor of approximately 10^5 at 36 hr. after reactor shutdown, and by a factor of more
59 than 10^3 at 80 days after reactor shutdown.

Based on the above considerations, the radioactivity leakage from the EVTM to the RSB/RCB due to the postulated accident will be less 29
than, or is enveloped by the leakage presented in Section 15.5.2.3.

15.7.3.1.3 <u>Conclusions</u>

Based on the analysis shown by the steady-state temperatures in Table 15.7.3.1-2 for a postulated spent fuel assembly in a CCP without sodium in the EVTM, no fuel melting, but extensive clad melting of the fueled zone, is expected. Though no rearrangement of the fuel pellets is anticipated, a hypothetical redistribution of fuel fragments was found not to raise the temperature of the nearest EVTM seals beyond 260°F.

This accident could lead to release of fission products which are volatile at temperatures up to 3500°F into the EVTM, but only fission products which are volatile at 260°F could diffuse through the double EVTM seals. This fission product release from the EVTM is discussed in Section 15.5.2.3, and represents the limiting release case. The off-site
59 exposures reported in Section 15.5.2.3 are well within the dose limits. 29

References to Section 15.7.3.1:

1. "Fuel Dynamics Experiments Supporting FTR Loss-of-Flow Analyses," L. W. Deitrich, et al., Proc. Fast Reactor Safety Meeting, CONF-740401-P1, Page 239, April 1974

2. "Fuel Movement in R3, R5, and R6 Loss-of-Coolant Simulations in TREAT," A. De Volpi, et al., Trans. Am. Nucl. Soc., Vol. 21, Page 288, June 1975

3. "Loss-of-Cooling Experiments," Fast Reactor Program Combined Second and Third Quarters 1971 Progress Report, E. K. Hoekstra (Comp.), RCN-164, Page 95, January 1972

4. "Loss-of-Cooling Experiments," Fast Reactor Program Second Quarter 1973 Progress Report, E. K. Hoekstra (Comp.) RCN-190, Page 39, August 1973

5. "Loss-of-Cooling Experiments," Fast Reactor Program First Quarter 1975 Progress Report, E. K. Hoekstra (Comp.), RCN-228, Page 17, July 1975

6. "Studies of Fast Reactor Fuel Element Behavior under Transient Heating to Failure," R. R. Steward, et al., Argonne National Laboratory Report ANL-7552, 1969

7. "Crack Healing in UO_2," J. T. A. Roberts and B. J. Wrona, Journal of the Am. Ceramic Soc., Vol. 56, No. 6, Page 297, June 1973

8. "Laboratory Studies on Melting and Gas Release Behavior of Irradiated Fuel," E. T. Weber, et al., Proc. Fast Reactor Safety Meeting, CONF-740401-P2, Page 641, April 1974

9. "Thermal-Shock Cracking in UO_2 During Power Transients," B. J. Wrona, et al., Trans. Am. Nucl. Soc., Vol. 22, Page 419, November 1975

10. "Internal Pressurization in Solid Fuel Due to Transient Fission-Gas Release," C. C. Meek, et al., Trans. Am. Nucl. Soc., Vo 22, Page 418, November 1975 29

TABLE 15.7.3.1-1

EVENT SEQUENCE FOLLOWING A POSTULATED GROSS LEAK IN CCP

Time (min)	Event	Normal Refueling Operation	
		Initial Condition: 20-kw fuel assembly in CCP has been fully raised above liquid sodium surface in reactor.	
0	A gross leak, corresponding to a 1-in. diameter or larger hole develops at the bottom of the CCP.		
1.0	All sodium has left the CCP.		
7.5	Center fuel rod reaches 1500°F. Fission gas release into EVTM commences.		
11.0		EVTM starts to move away from reactor fuel transfer port (FTP).	
12.0	All fuel rods reach 1500°F. All fission gas released into EVTM.		
17.0	Cladding of center fuel rod starts melting.		
30.0	Cladding of all fuel rods in mid-plane of fueled region has melted.		
32.0		EVTM is mated to EVST fuel transfer port, and all service ties between EVTM and floor valve are connected.	
50	Fuel pellets at center line have reached 3500°F, at periphery 2800°F.		
56.0	All temperatures start to drop.	Fuel assembly and CCP are fully submerged under EVST sodium.	29

SUMMARY TABLE 15.7.3.1-2

PEAK STEADY-STATE TEMPERATURES RESULTING FROM COMPLETE LOSS OF SODIUM FROM A CCP CONTAINING A 20 kw SPENT FUEL ASSEMBLY IN THE EVTM

Location	Maximum Temperature (°F)	Melting Point (°F)
Center Fuel Rod (Hottest Rod) Fuel Temperature	3500	5000
Cladding of Center Fuel Rod*	3500	2525
Cladding on Outer Fuel Rod	2845	2525
Fuel Assembly Housing	2515	2525
Core Component Pot	2060	2600
EVTM Cold Wall	435	2600

*Melting begins 17 minutes after sudden loss of sodium

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TABLE 15.7.3.1-3

FISSION PRODUCT ELEMENTS RELEASED INTO EVTM AND THEIR MELTING POINTS

Element	Melting Point (°F)	Element	Melting Point (°F)
Ga	86	Sb	1167
Ge	1719	Te	841
As	1503	I	236
Se	423	Xe	-169
Br	19	Cs	83
Kr	-250	Ba	1337
Rb	102	La	1690
Sr	1416	Pr	1708
Y	2772	Ce	1470
Zr	3366	Nd	1870
Po	489	Pm	1976
Ag	1763	Sm	1971
Pd	2826	Eu	1512
Cd	610	Gd	2395
In	314	Tb	2473
Sn	450	Dy	2574
		Ho	2685
		Er	2784

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15.7.3.2 Spent Fuel Shipping Cask Drop from Maximum Possible Height

15.7.3.2.1 Identification of Causes and Accident Description

The maximum height for a potential Spent Fuel Shipping Cask (SFSC) drop in the CRBRP is the 72-ft. vertical distance from the operating floor of 12
the RSB to the bottom of the SFSC handling shaft.

Section 9.1.4.8 discusses the design features preventing an SFSC drop in the CRBRP. These consist mainly of handling the SFSC above the operating floor of the RSB and within the cask handling shaft only with the double reeved RSB bridge crane (125 ton capacity) using rigging specially designed and tested
59 for the SFSC. The operational requirements of RDT Standard F8-6T applying to critical items will cover all moves of the SFSC when handled by the RSB bridge crane. Due to these design features and operational precautions, dropping of an SFSC within the RSB is considered a hypothetical event. 12

As identified in Section 9.1.2, the SFSC will be licensed separately. The cask is designed to withstand a hypothetical accident condition of a 30-ft. free drop as specified in 10CFR71. 12 Under these conditions, the cask is designed to maintain its structural integrity with zero leakage of its radioactive content. This design condition satisfies the requirements of 10CFR71 which specify radioactivity release limits for a cask under hypothetical accident 12 conditions.

15.7.3.2.2 Analysis of Effects and Consequences

The free fall impact energy of an SFSC dropped to the bottom of the handling shaft is smaller than that for which the cask is designed, as discussed in Section 9.1.4.8.

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Though a 72-ft. drop to the bottom of the cask handling shaft is not expected to occur and would not result in a break of the SFSC containment, a
59 break of the outer cask containment and release of radioactivity through the seals of the inner cask containment has been postulated and analyzed. The purpose of the analysis is to demonstrate the inherent safety margins available, even under the following conservative assumptions:

1) The SFSC is loaded with core assemblies of the highest fission gas inventory and a short decay time. This assumption contains two design margins with respect to radioactivity:

   59 a. The spent fuel assemblies are assumed to be of the highest power and shipped at 80 days after reactor shutdown. This exceeds the design requirements that spent fuel shipment commence no sooner than 100 days after reactor shutdown. Administrative procedures will actually require that the highest powered fuel assemblies will be the last ones of one refueling batch to be shipped i.e., at a decay time substantially greater than 100 days. 12

b. The fission gas inventory of six highest powered spent fuel assemblies and three blanket assemblies is considered. This inventory
59 corresponds to the maximum total core assembly decay heat load of the SFSC (26 kw).

2) All fuel rods in the six fuel assemblies are assumed to fail, releasing the entire fission gas inventory instantaneously into a helium gas
59 space in the SFSC canister. The canister forms the inner containment of the SFSC. The long-lived, volatile radionuclides of this inventory with
59 significant activities at the time of shipping are shown in Table 15.7.3.2-1.

3) The fission gas is assumed to leak through the inner and outer containments of the SFSC at the maximum allowable (see SFSC SAR) inner containment seal leak rate for helium of 6 x 10^{-5} Std cm^3/sec, adjusted for the higher canister pressure after the drop. This assumption does not take credit for the outer containment seals which have a maximum allowable leak rate of 4.3 x10^{-6} Std cm^3/sec at 10 psi differential. It also does not take account of the fact that the outer containment is at a pressure lower than ambient (nominal 10 psia), which only allows for leakage of gas out of the outer cask containment during the early days of the SFSC shipping. All SFSC seals consist of stainless steel O-rings. Low leakage rates will be assured by appropriate leak testing.

4) The maximum steady temperature near the canister seals was calculated to be 350°F. All fission products which are volatile at this temperature were considered to leak through the seals.

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The potential off-site doses calculated and presented in Table 15.7.3.2-2 assumed that the gases that leaked from the SFSC were exhausted directly to the atmosphere via the RCB/RSB ventilation system. No credit for holdup in the RCB or RSB was taken. The LPZ dose was calculated using the time integrated
59 radioactivity release assuming continuous leakage for 30 days. 12

15.7.3.2.3 Conclusions 12

This accident would not present any hazard to the public, the doses being well below the 10CFR100 guideline values.

TABLE 15.7.3.2-1

FUEL ASSEMBLY INVENTORY AND RELEASE RATES OF LONG-LIVED, VOLATILE FISSION-GAS ISOTOPES WITH SIGNIFICANT ACTIVITIES FOR SFSC DROP FROM MAXIMUM POSSIBLE HEIGHT

Isotope	Total Activity in One F/A at 80-Day Decay Time (Ci)	Specific Activity in Cask Gas at 80-Day Decay Time (Ci/scc)	Leak Rate from Dropped Cask (Ci/sec)
Kr^{85}	616	1.10×10^{-3}	1.24×10^{-7}
Xe^{131m}	34.0	6.11×10^{-5}	6.97×10^{-9}
Xe^{133}	11.4	2.05×10^{-5}	2.34×10^{-9}
I^{131}	185	3.32×10^{-4}	3.78×10^{-8}
Cs^{134}	3600	1.9×10^{-7}*	2.2×10^{-11}
Cs^{136}	219	0.9×10^{-8}*	1.0×10^{-12}
Cs^{137}	9930	5.2×10^{-7}*	5.9×10^{-11}
59 Rb^{86}	41.5	2.1×10^{-8}*	2.4×10^{-12}

* Based on vapor pressure of Cs and Rb at the maximum SFSC seal temperature of 350°F.

Table 15.7.3.2-2

Off-Site Doses (REM) Due to Fuel Failure and SFSC Leakage

ORGAN	10CFR100 GUIDELINE	2 HOURS SB (0.42 MILES)	30 DAYS LPZ (5.0 MILES)
Cloud			
D (Whole Body)	25	9.64-7*	1.19-6
Inhalation			
Lung	75	1.29-8	1.59-8
Thyroid	300	4.39-4	5.41-4
Whole Body Inhalation	25	8.89-7	1.13-6

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15.7.3.3 Maximum Possible Conventional Fires, Flood, Storms or Minimum River Level

15.7.3.3.1 Identification of Causes and Accident Description

The causes postulated for the maximum conventional fires cover a broad spectrum of initiating events, from acts of nature to outright sabotage. The maximum floods experienced at the site is generally considered to be initiated by an act of nature, however, it is conceivable that an act (or acts) of sabotage could be committed on the upstream dams. The maximum possible storms confronting the site must be considered as caused by nature. Similarly, a minimum river level must be assumed as the result of an act of nature.

15.7.3.3.2 Analysis of Effects and Consequences

Maximum Conventional Fires

The plant fire protection system is designed to provide adequate and reliable fire protection, detection, and signal capability in those areas of the power station where a maximum conventional fire hazard might exist. The fire protection system is designed to supply water, carbon dioxide, and other fire extinguishing agents to various locations throughout the plant site (Section 9.13.1). To combat forest fires, the plant building complex is located a minimum of 300 feet from the nearest tree line in any direction. In the event of local forest fire, the yard protection loop will be employed to assist in extinguishing the blaze.

Maximum Conventional Flood

The maximum flood elevation at the CRBRP site is estimated at Elevation 809.2 feet. This is based upon the combined effect of 1/2 PMF, seismic event (OBE) concurrent with dam failures, and wave runup. With the plant grade established at Elevation 815.0 feet, there will be no major flooding on the safety-related structures at the plant site except the consideration of the potential hydrostatic pressure and buoyancy on some of the deeply embedded structures. For more information, see Section 3.4 (Water Level and Flood Design) and Section 2.4.2 (Floods).

Maximum Conventional Storms

Considering the maximum rainfall, severest snow and glaze storms, thunderstorms and hail, tornadoes, strong winds and hurricanes discussed in Section 2.3.1.3, the site (on a geographical basis) is situated in a region where these conditions are not a significant factor.

The probable maximum precipitation (PMP) in the Clinch River Watershed is discussed in Section 2.4.3.1. The description of runoff models is given in Section 2.4.3.3 and the resulting water levels of the Clinch River are summarized in Table 2.4-8.

The overall site drainage facilities will be designed for 3.5 inches of rainfall in one hour once in 100 years, with a 50 percent runoff coefficient. The drainage facilities for safety-related structures will be designed for the PMP of an 8-hour storm depth of 29.5 inches with a maximum 1-hour depth of 14 inches, as specified by the hydrometerological branch of the National Weather Service. The maximum recorded rainfall for a 24-hour period for this site was 7.75 inches.

As discussed in Section 2.3.1.3, the highest average monthly total of snowfall in the site area is 3.1 inches. Accordingly, significant amounts of ice or snow forming or accumulating on the roofs of safety-related structures and on exposed safety-related equipment will be infrequent. However, for design purposes, a live load of 20 pounds per square foot will be imposed on all roofs - this amount of load intensity would be equivalent to a weight of 4 inch deep water uniformly applied to a level roof (equivalent to a 40-inch snowfall). Local meteorlogical data reports a maximum 24 hour snowfall of 12 inches and a maximum monthly snowfall of 21 inches.

On a geographical basis, the site is situated in a region where hail is not a significant factor, therefore, no adverse effects from hail are considered plausible.

The principal structures on the site, those housing safety-related system and components, are Category I, tornado-hardened reinforced concrete structures. These structures are designed for a 90 mile-per-hour basic wind 30 feet above grade with a 100-year period of recurrence. The peak gust recorded by the Oak Ridge City Office is about 59 miles-per-hour for a 16 year record. These same structures are designed to withstand tornadoes with maximum wind velocity of 360 MPH (290 MPH rotational velocity and 70 MPH maximum translational velocity).

Hurricanes are rare as far inland as the site because hurricanes lose force rapidly when cut off from their source of moisture. Consequently, these storms are in the post-hurricane stage with diminished winds by the time they reach the site region, and therefore covered by the strong wind loadings discussed in Section 3.3.

Wind waves during a postulated major storm were computed by using the Corps of Engineers' procedures. For a 40 mile-per-hour overland wind, 99.6 percent of the waves would be less than 2.4 feet high from crest to trough, and runup above still reservoir levels would be 2.8 feet on a smooth 3:1 slope and 3.8 feet on a vertical wall.

Minimum River Level

For design purposes, the minimum river level shall be taken at an elevation of 735 feet in accordance with controlled pool elevations set by the TVA regulations of upstream and downstream dams. Since the intake structure is located 5.5 feet below the minimum water level, the condition of minimum water level will have no effect on plant operation or assumed plant accidents.

15.7.3.3.3 Conclusions

Based on the information presented in Section 15.7.3.3.2, the maximum conventional fires, floods, storms or minimum river level presents no deliterious effects on the plant.

15.7.3.4 Failure of Plug Seals and Annuli

15.7.3.4.1 Identification of Causes and Accident Description

Malfunction of the reactor cover gas system in such a way to produce an excessively low transient cover gas pressure followed by recovery to normal operating pressure, would cause enough displacement of the liquid in the reactor vessel head plug annulus dip seals to allow reactor vessel cover gas to bubble under the seal blade and up into the inner buffer annulus between the dip seal and the inflatable seals immediately under the rotating plug bearings. Classification of this event as extremely unlikely is predicated on the massive failure of the seals.

This cover gas can be contaminated with fission products and other radioisotopes. Abnormally high concentrations of radioisotopes in the inner buffer annulus would increase the rate of leakage of these isotopes to the reactor vessel head access area.

15.7.3.4.2 Analysis of Effects and Consequences

Quantitative analyses of the abnormal rates of radioisotope release to the head access area in this event have not yet been done. There are a number of factors which qualitatively lead one to believe that the abnormal leakage rates are very unlikely to constitute a serious hazard to operating personnel.

a. There are two intact outer seals with an outer buffer space between them. The pressure resisting capability of these seals will be larger than the pressure difference which can be created by cover gas system failure. Tests described in Section 1.5 will verify the capability of the seals.

b. Normally expected leak rates through these seals cause a very small fraction of the allowable radioisotope concentration in the head access area.

c. A radiation monitor in the head access area will warn of abnormally high concentrations so the area can be evacuated if required. In addition, it can be assumed that the indication of abnormally high concentrations of radioactivity would result in the isolation of containment.

15.7.3.4.3 Conclusions

Based on the discussion of the preceding section (Section 15.7.3.4.2), there would appear to be no adverse consequences associated with failure of the cover gas system. Because of the pressure-resisting capability of the seals, the expected low leak rates and the capability to monitor the radiation level of head access area and isolate containment, the radiation dose quidelines of 10CFR100 will not be violated.

15.7.3.5 Fuel Rod Leakage Combined with IHX and Steam Generator Leakage

15.7.3.5.1 Identification of Causes

This event is proposed to determine the potential for fission gases from leaking fuel rods to pass from the primary sodium system to the steam generator system if leaks were to occur in both the IHX and steam generator tubes.

To eliminate the possibility of radioactive products reaching the steam generator system via the intermediate system, the intermediate system pressure at the IHX is maintained at least 10 psi higher than the primary system during all normal operational modes. This pressure differential as well as the IHX leakage, is discussed in Section 15.7.1.3. In addition, the steam generator system is operated at much higher pressures than the intermediate system. Therefore, during normal operations, leakage would be in the direction towards the intermediate system in the unlikely eventuality of steam generator tube leakage in conjunction with the IHX tube leakage. The probability of steam generator tube leaks along with the follow-up action and consequences, are discussed in Sections 15.3.2.3 and 15.3.3.5.

15.7.3.5.2 Analysis Effects and Consequences

As discussed in Section 5.3.3.5, the IHX is designed to minimize the possibility of tube leakage. In the event of a detectable leak in the IHX, the sodium leak detection system will sense the change of sodium inventory and provide the required information for the operators to take the necessary action with regard to plant shutdown and follow-up maintenance procedures (see Section 15.7.1.3). If, however, the leak is not detected due to the leak being very small or a leak detection system malfunction, the plant could be operating with an unknown breach of the primary to intermediate system barrier. As discussed above, no problems regarding the release of radioactive products will exist inasmuch as the leakage will be from the non-radioactive intermediate system into the primary system. If, in addition to the IHX tube leak, a steam generator leak were to occur in the same loop, appropriate plant shutdown procedures will commence ensuring that no additional potential for the release of radioactive products from the primary to steam generator system will occur. These procedures include maintaining a positive differential pressure from the steam generator system to the intermediate system and to the primary system.

15.7.3.5.3 Conclusions

It is considered that it is extremely unlikely that this event will occur based on the very low probability of the steam generator leak occuring in the same loop as an undetected IHX leak. If, however, the event were to occur, the potential for radioactive products passing from the primary system to the steam generator system is virtually eliminated due to the plant pressure differentials and the operating procedures that will be employed.

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15.7.3.6 Sodium Interaction With Chilled Water

15.7.3.6.1 Identification of Causes

The design of the Chilled Water Systems (Section 9.7) incorporates features to maintain three barriers between water and sodium. For an interaction between sodium and water to occur, two pipe failures and a third boundary must fail simultaneously. The third boundary failure will be either a structural failure or the result of either a valve failure or a failure of redundant leak detectors.

15.7.3.6.2 Analysis of Effects and Consequences

The description of Normal and Emergency Chilled Water System design features intended to prevent sodium/water contact is given in Section 9.7.3. The following points address several accident events during which one or more of these design features may fail:

a. Leakage from a chilled water cooling coil inside a recirculating gas fan cooler unit will result in automatic closure of the chilled water and Nitrogen valves thus isolating the affected unit upon a signal from the redundant water moisture or leak detectors. If any isolation valve fails to close, as evidenced by a failure of the valve status lights to indicate closure, the nearest upstream valves will automatically close. Automatic and redundant fan-cooler unit drain valves will open upon a leak detection signal to prevent water build up in the unit. If both drain valves fail to open, and the chilled water and nitrogen valves fail to close, calculations indicate that under a postulated leak condition per USNRC-SRP-3.6.1 and 3.6.2, nearly two hours may pass before water will overflow the gas cooler unit and enter the cell containing sodium piping. This time period is sufficient to both correct a drain valve problem and, if necessary, manually isolate the affected unit.

b. Leakage from chilled water cooling coils inside HVAC coolers will be controlled by leak detection and drainage features similar to those listed above for recirculating gas coolers. These features will be provided only for HVAC coolers serving areas containing sodium piping or equipment.

c. Leakage from chilled water piping located in the reactor containment and service buildings will actuate floor drain system area leak detectors. Under these conditions, the portion of chilled water line affected will be either remote manually or automatically isolated.

d. Sodium leakage from piping or equipment in areas served by the recirculating gas cooling system will result in a sodium leak detection signal. Upon confirmation of a sodium leak, the water and gas isolation valves to the recirculating gas cooler serving the affected area will be remotely closed.

15.7.3.6.3 Conclusions

Based on the extremely unlikely combination of events required to cause an interaction of sodium with chilled water, such an interaction is considered to have a very low probability of occuring. The three-barrier design features described in Section 9.7, together with the automatic leak detection and drainage features described above, provide assurance that a sodium and chilled water interaction may be considered incredible.

15

15.7.3.7 Sodium-Water Reaction in Large Component Cleaning Vessel

15.7.3.7.1 Identification of Causes and Accident Description

A sodium-water reaction accident in the Large Component Cleaning Vessel (LCCV) would be caused by unplanned introduction of liquid water which would react with bulk sodium prior to completion of the WVN phase of the sodium removal process. The consequences of the accident would depend on the amount of sodium on the component in the LCCV and the geometry of the component. This analysis assumes that the component having the largest initial sodium inventory is being cleaned. The frequency of sodium removal from components having enough sodium to make possible a serious sodium-water reaction is very low. This, together with the design features which prevent such a reaction, makes this an extremely unlikely event.

In the normal sodium removal process, all sodium except small amounts isolated in crevices is removed during the WVN cycle by the following reaction:

$$Na + H_2O = NaOH + 1/2\ H_2$$

Monitoring to determine completion of the sodium-water vapor reaction in the WVN cycle is accomplished by measuring hydrogen concentration in the gas leaving the LCCV. The reaction rate is controlled by establishing a water vapor concentration in the WVN entering the LCCV to limit the exhaust gas hyrdorgen concentration to less than 4%. As sodium is removed, the reaction rate and resulting hydrogen concentration decrease for a fixed inlet water vapor concentration. To maintain the reaction rate, the water vapor concentration is gradually raised to a maximum of 15%. The reaction is considered complete when the hydrogen concentration in the exit gas then falls below 100 ppm.

The rinse cycle in the normal sodium removal process removes the inert reaction products of the WVN cycle. This will not normally involve significatnt chemical reactions. Presence of the amounts of sodium necessary for a significant chemical reaction could occur only as a result of initiating the rinse cycle without performing the WVN cycle. The design includes an interlock to prevent this error by preventing opening of the water supply valve until 24 hr after opening of the steam supply valve for the WVN cycle. The interlock can be bypassed by use of a key switch whose key is kept under supervisory control. The accidental addition of water while all sodium remains on the component is the worst possible case and is analyzed for the sodium-water reaction accident.

Sodium in the LCCV prior to the WVN cycle would react with water during the accidental rinse cycle by the same reaction as in the WVN cycle. The hydrogen and heat generated would result in high pressure and temperature in the vessel. This would promote the additional reactions listed below; however, the reaction of the above equation would be predominant and is used in the analysis of this event.

$$Na + H_2O = NaOH + \frac{1}{2}H_2 \qquad (1)$$

The heat of formation for the above reaction is:

$$\Delta H_{Na} = -44.19 \text{ Kcal/g-mol}_{Na}$$

$$\Delta H_{NaK} = -158.8 \text{ Kcal/g-mol}_{NaK}$$ 23

$$Na + NaOH = 2NaO + 1/2H_2$$

$$2Na + NaOH = NaO + NaH$$

$$NaH + H_2O = NaOH + H_2$$

$$NaO + H_2O = NaOH$$

The sodium of interest for this analysis is in the form of frost deposited on parts which have been in the cover gas space above the reactor sodium pool. Since this form of sodium deposit presents a high-surface area for reaction, it was assumed that the reaction is instantaneous when water reaches sodium.

Many components will use the LCCV for sodium removal; however, all except two components, the intermediate rotating plug (IRP) and the small rotating plus (SRP), contain a quantity of sodium for which complete instantaneous reaction with water would result in an LCCV internal pressure less than the 15 psig design pressure. The design of the SRP is similar to that of the IRP described in the next paragraph. The event for the SRP would be the same as for the IRP, but the amount of sodium involved would be less by a factor of about six. Also, the SRP is expected to be cleaned only only per 30 yr. the same frequency as for the IRP. Therefore, the sodium-water reaction with the IRP is an enveloping event and was the case analyzed.

The IRP consists of a series of horizontal plates supported by four columns supported by the rotatable plug which is part of the reactor vessel closure head. The suppressor plate is the lowest plate. It and the lower 36 in. of its support columns are immersed in the reactor sodium pool during operation and will have a 0.003-in.-thick film of sodium when removed for cleaning. The area of the plate is about 47,000 in.^{2}, giving a sodium content of 4.5 lb. The lower 36 in. of the support columns will contain another 2 lb. of sodium film. The next plate, 48.7 in. above the suppressor plate, is the lowest of the reflector plates. There are 20 reflector plates, each separated by 1/2 in. and having a surface area of about 36,000 in.^{2}. Each has a coating of frost deposits consisting principally of sodium but also containing Na_2O and NaH. The thicknesses of these coatings range from 0.0445 in. for the bottom plate and the upper section of the support columns to 0.0005 in. for the top reflector plate. In this analysis, it is assumed that these are the thicknesses of solid sodium film. The lowest reflector plate contains about 65 lb. of sodium. The next two higher plates contain 51 and 40 lb. The total sodium on the IRP is about 350 lb.

The sodium-water reaction event would begin with the addition of water to the LCCV at a rate of 125 gpm. A flow of nitrogen at 50 cfm would be maintained through the water into the LCCV and out through the vent to maintain a purge of the system. The nitrogen would carry over water droplets which, together with the water vapor above the water surface, would react with the sodium at a rate comparable to that in the WVN cycle. It is assumed, however, that no sodium is removed by this reaction and that it all remains until the water reaches it. When the water reaches the suppressor plate, the 4.5 lb. of sodium on it will react instantaneously. The resulting pressure in the LCCV will be less than the LCCV design pressure of 15 psig. The hydrogen concentration in the LCCV nitrogen will be 2.5%, which is less than the 4% annunciator and interlock setpoint. It is assumed that water addition will continue at 125 gpm. The water level will rise at about 1-1/2 in. per min so that about 30 min will be required to reach the lower reflector plate. During this time, the hydrogen from the suppressor plate reaction and the slow reaction with support column sodium will be purged from the LCCV.

Water and the 65 lb. of sodium on the lower reflector plate will react when the water level has risen to the plate elevation. The pressure and hydrogen concentration in the LCCV gas space will increase. At a pressure of 8 psig, an interlock is activated to close the rinse-water inlet valve. At a pressure of 16.5 psig the LCCV pressure relief valve will open to vent the gas into the Large Component Cleaning Cell. The maximum pressure which would be reached without venting would be 89 psig. This is lower than the burst pressure of all components of the system, so that the hydrogen-nitrogen mixture will be contained except for venting through the pressure relief valve and the normal system vent to the H&V System. The hydrogen concentration in the LCCV gas will be 22%. The increase will be detected within a few seconds by the hydrogen analyzer in the LCCV vent line. When the detected level exceeds 4% an interlock will be activated to close the valve in the water inlet line. This interlock provides backup for the high pressure interlock which closes the same valve.

The hydrogen-nitrogen mixture which is vented through the LCCV pressure relief valve is mixed with the air at atmospheric pressure in the 67,000 ft^3 cell. The pressure resulting from adiabatic expansion of the mixture into the cell is about 2 psig. The hydrogen concentration in the cell after mixing with the air is about 2.5%.

15.7.3.7.2 <u>Analysis of Effects and Consequences</u>

The sodium-water reaction described in the above section is an extremely unlikely event because the two components with which it could occur are each cleaned only once in 30 years, and because of the number of failures which must occur to permit the event. The principal failure would be in not completing the WVN cycle before adding water in the rinse cycle. An interlock requires that the WVN cycle must be started by opening the steam valve and must proceed for 24 hrs before the water inlet valve may be opened without using the key switch interlock bypass. Control of the key by supervisory personnel will avoid improper use of the bypass. Once the WVN cycle is begun, failure of a second interlock would be required to terminate it before the hydrogen concentration in the exhaust was less than 100 ppm. This low hydrogen concentration ensures that much of the sodium is reacted even if the inlet water vapor concentration is not raised to the normal 15%.

Analysis of the event hypothesized the instantaneous reaction of the 65 lb. of sodium on the lowest reflector plate. The reaction releases hydrogen into the 2,100-ft^3 nitrogen gas space above the water level and releases heat. It is assumed that all heat from the reaction goes to heating the nitrogen and the reaction products (NaOH and H_2). Due to this heating, the gas space above the water would be pressurized to a maximum of 89 psig, which is less than the static rupture pressure of all components in the system. It is assumed for the analysis that all of the gas is released adiabatically into the LCCV Cell.

The resulting cell pressure of about 2 psig is less than the cell design pressure of 10 psig. It is also assumed that there is complete mixing of the vented LCCV gas and the LCCV cells air atmosphere. The resulting hydrogen concentration of 2.5% is less than the 4% explosive limit of hydrogen in air.

Since there is no designed vent between Cell 125 and the RCB atmosphere, the aerosol would be confined in the cell. Since the pressure in the cell is ony 2 psig, there would be minimal leakage past the cell penetrations seals and by the time this leakage works its way up to the RCB atmosphere and through the system RCB filtration system, the impact on the site boundary would be negligible.

15.7.3.7.3 Conclusion

Based on tne analysis described in the preceding sections, it is concluded that the vessel and system design is adequate to protect the plant and the public, and that there are no adverse consequences to the health and safety of the public which would resuit from this accident. Specifically:

- o An uncontrolled sodium-water reaction in the LCCV is an extremely unlikely event.
- o The LCCV pressure relief valve is set to vent the gas to the cell at a pressure 10% above the design pressure of the vessel.
- o Failure of the relief valve to open will result in a maximum pressure of 89 psig. This is less than the calculated burst pressure for the LCCV and connected process equipment.
- o Release to the LCCV cell of all reaction products will pressurize the cell to only 20% of the cell design pressure of 10 psig.
- o There will be an inperceptible impact on the site boundary dose.

15.7.3.7.4 Enveloping Other Sodium-Water Reactions

To envelope the site boundary dose of all other sodium-water reactions in any of the cleaning vessels, calculations were made assuming 100% of the radioactivity deposited on the IRP to be released via a hypothetical vent from the LCCC to the RCB HVAC and thus to the environment. Activity content of the assumed release was derived from information in PSAR Table 11.1-7 decayed for 10 days. Such a release would isolate the RCB and the postulated effluent will pass through the filter system before release to the outside environment. A decontamination factor of 20 for iodine and 100 for particulates was assumed in the analysis. The activity is conservatively assumed to be in the form of a "puff".

Table 15.7.3.7-1 provides the resultant doses from this set of conservative assumptions and event. All doses are well within the appropriate requirements and guideline values of 10CFR100.

<u>References</u>:

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10. "Studies of Fast Reactor Fuel Element Behavior under Transient Heating to Failure," R. R. Steward, et al., Argonne National Laboratory Report ANL-7552, 1969

11. "Crack Healing in UO_2," J. T. A. Roberts and B. J. Wrona, Journal of the Am. Ceramic Soc., Vol. 56, No. 6, Page 297, June 1973

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13. "Thermal-Shock Cracking in UO_2 During Power Transients," B. J. Wrona, et al., Trans. Am. Nucl. Soc., Vol. 22, Page 419, November 1975

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Table 15.7.3.7-1

Release From LCCV - Potential Site Boundary Doses

Organ	Dose at Site Boundary (0.4 mi) (Rem/2 hrs.)	Dose at Low Population Zone (2.5 mi) (Rem/30 days)
Whole Body	2.60×10^{-2}	7.14×10^{-3}
Thyroid	2.70×10^{-1}	7.39×10^{-2}
Bone	1.93×10^{-1}	5.29×10^{-2}
Lung	1.07×10^{-2}	2.92×10^{-3}

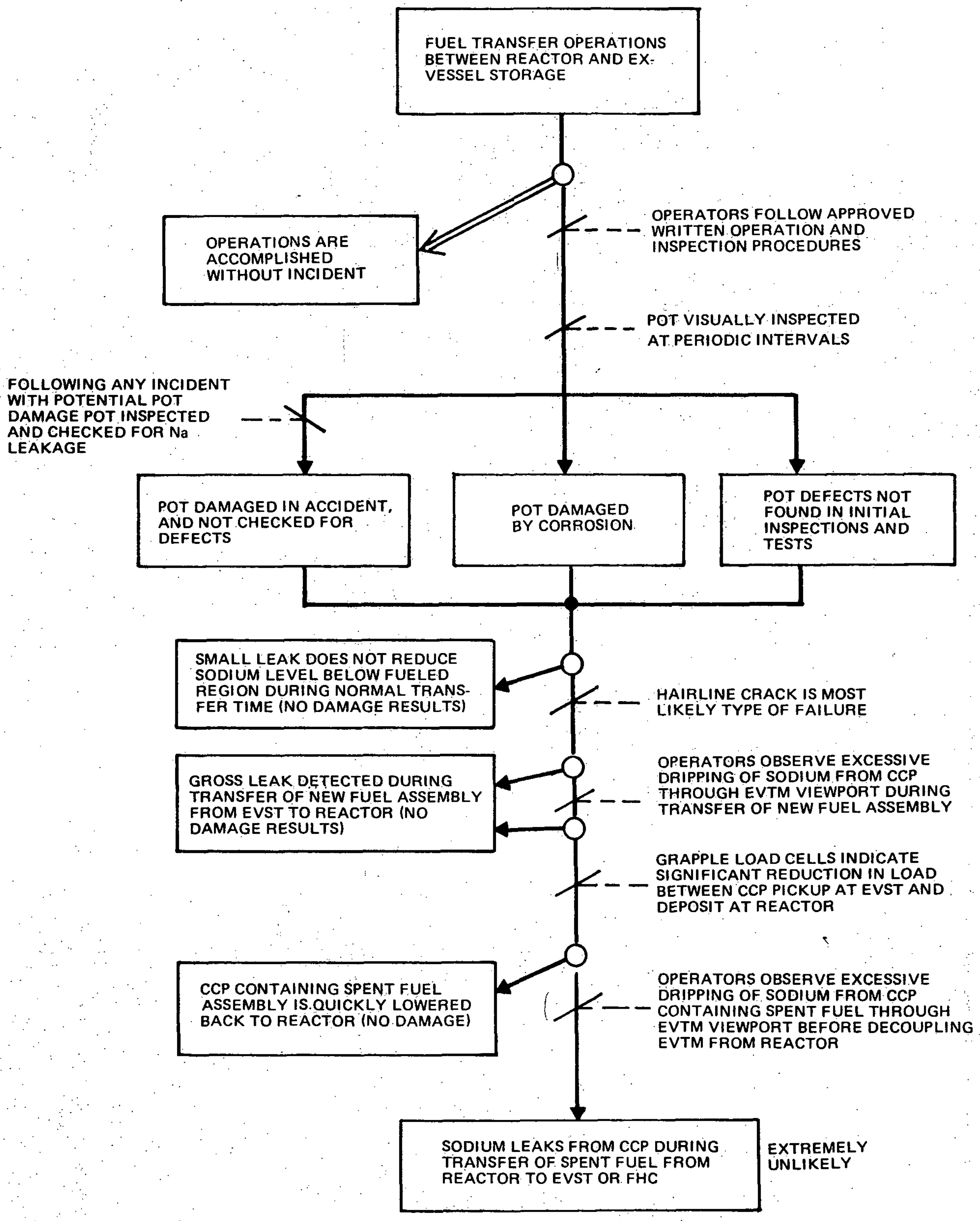

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FIGURE 15.7.3.1-1 PROTECTION AGAINST LEAK IN CORE COMPONENT POT

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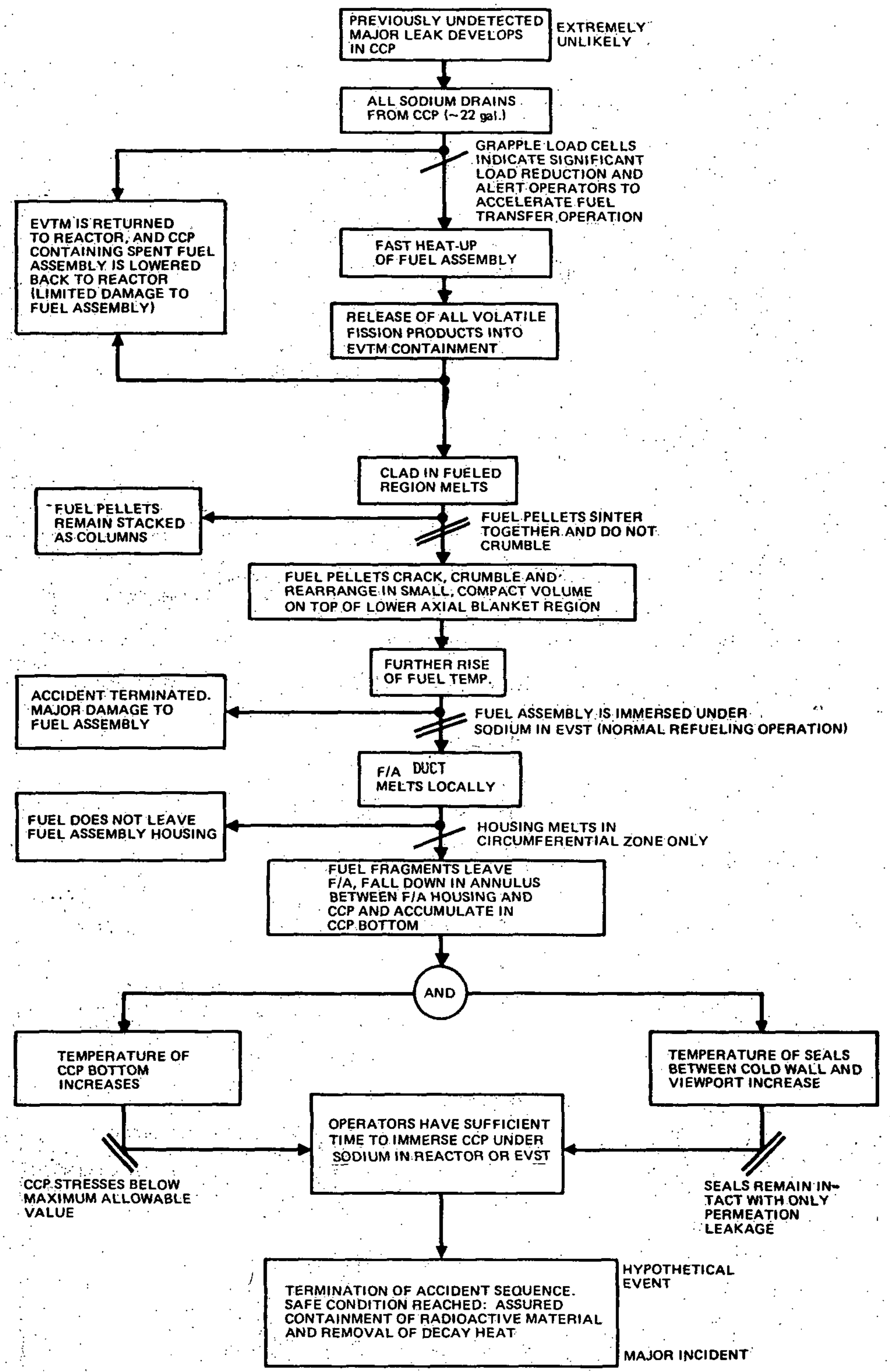

FIGURE 15.7.3.1-2 CONTINUATION OF SAFETY ASSURANCE DIAGRAM FOR LEAK IN CORE COMPONENT POT

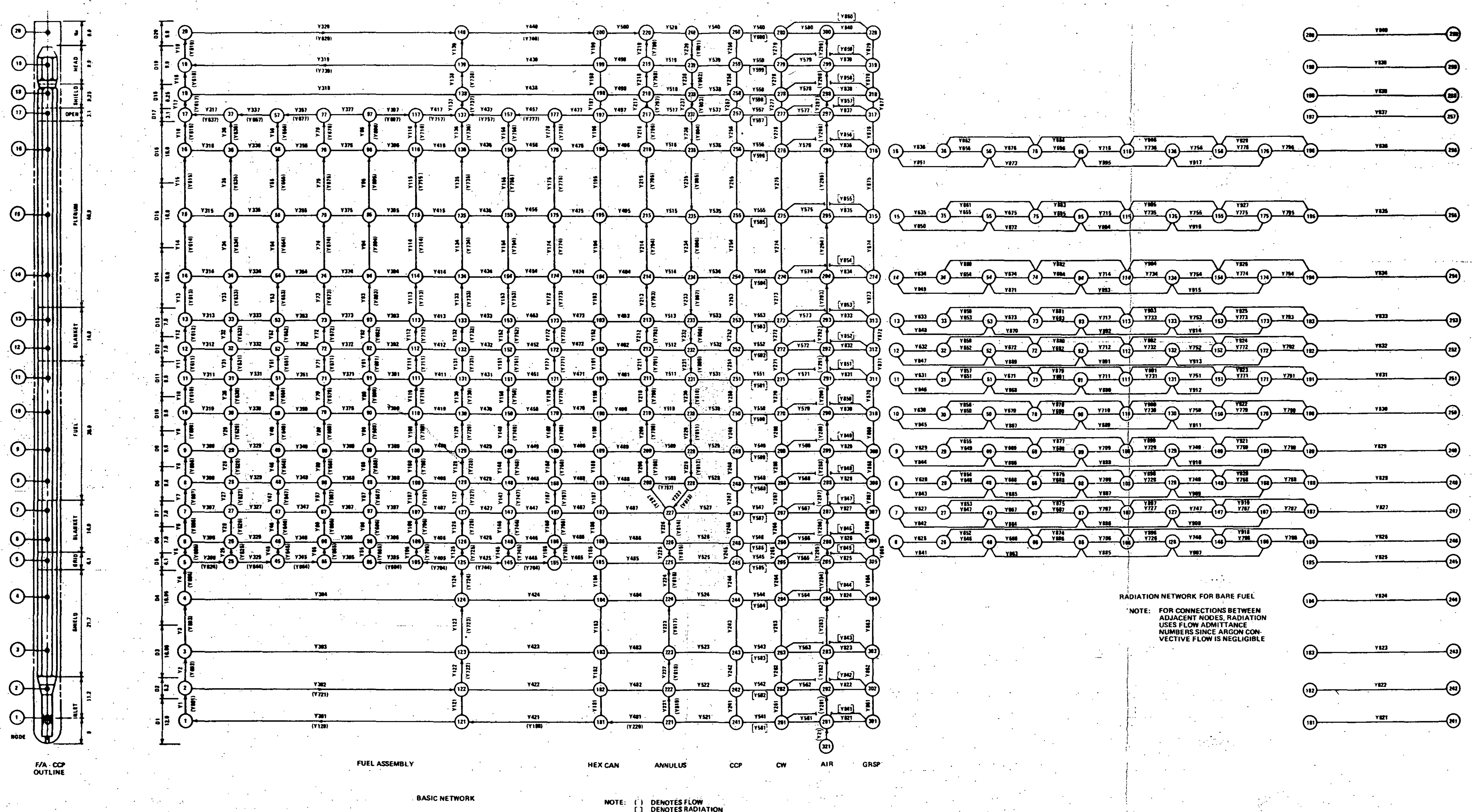

Figure 15.7.3.1-3 Thermal Analysis Network Model for Fuel Assembly in CCP Surrounded by EVTM Cold Wall

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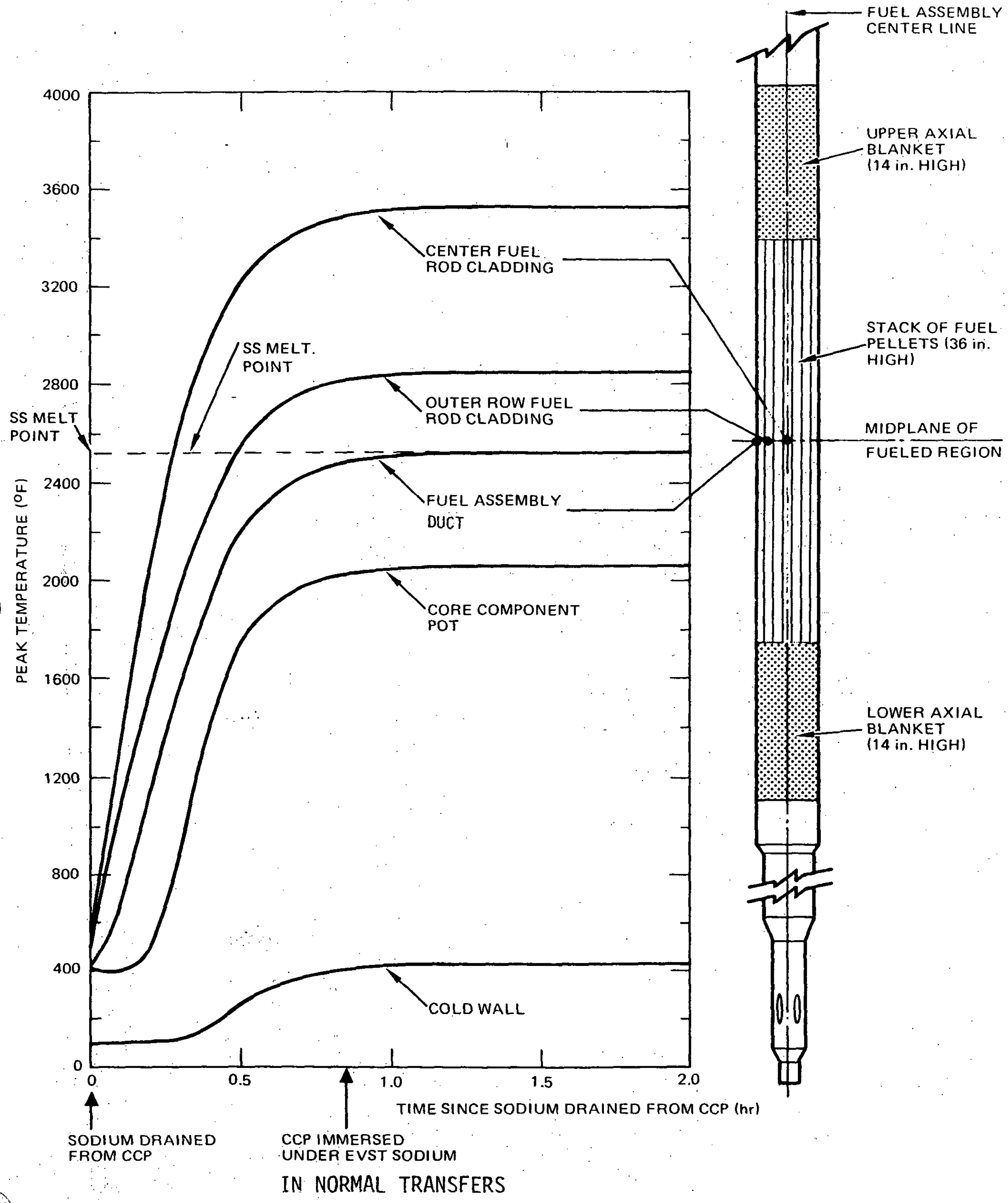

FIGURE 15.7.3.1-4 Peak Transient Temperatures in EVTM with Dry 20-kW Spent Fuel Assembly in CCP. - Stacked Fuel Pellets CCP Not Reimmersed in Sodium After Draining

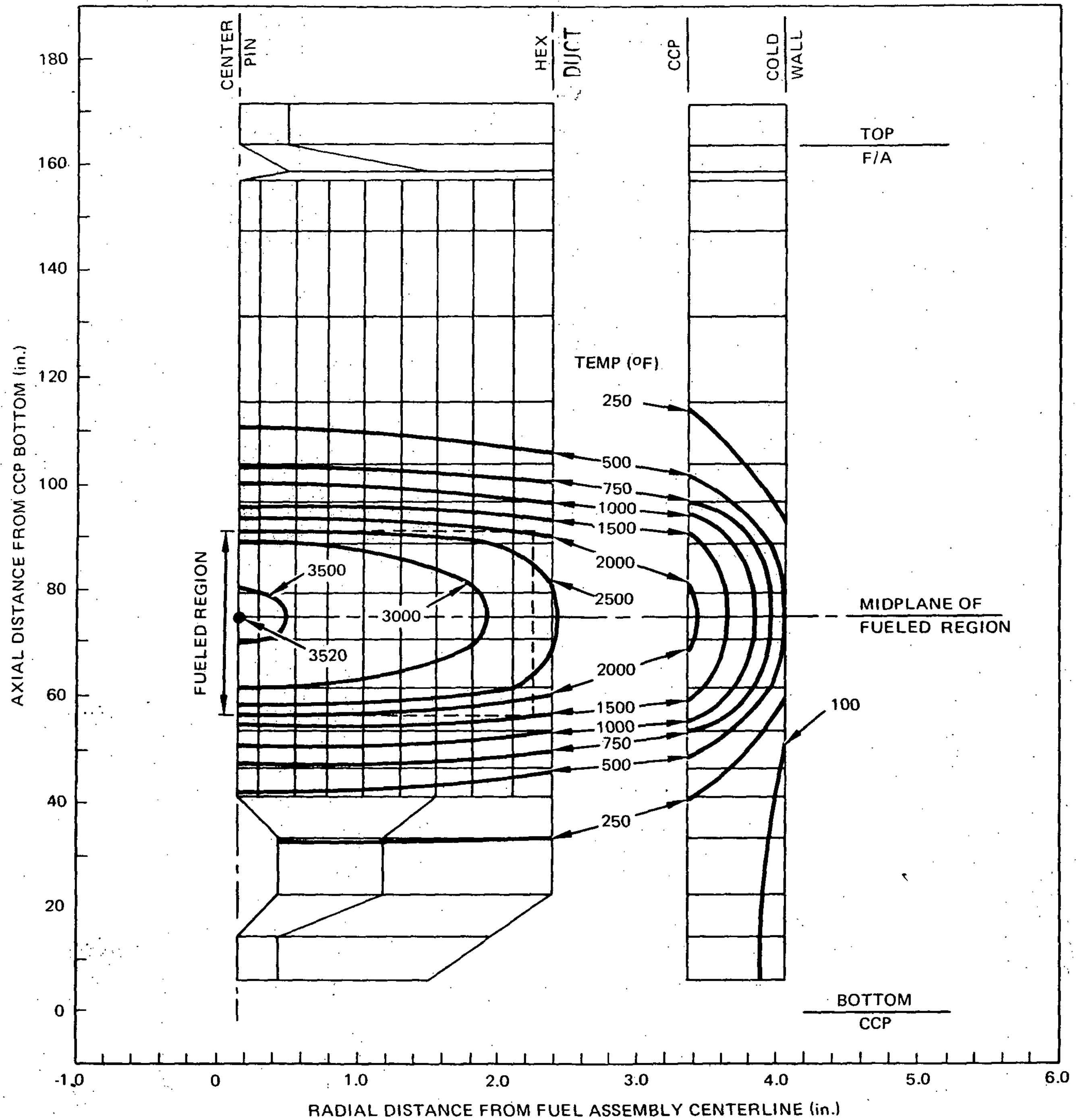

FIGURE 15.7.3.1-5. STEADY STATE TEMPERATURE DISTRIBUTION IN EVTM WITH DRY 20-kW SPENT FUEL ASSEMBLY IN CCP. — STACKED FUEL PELLETS

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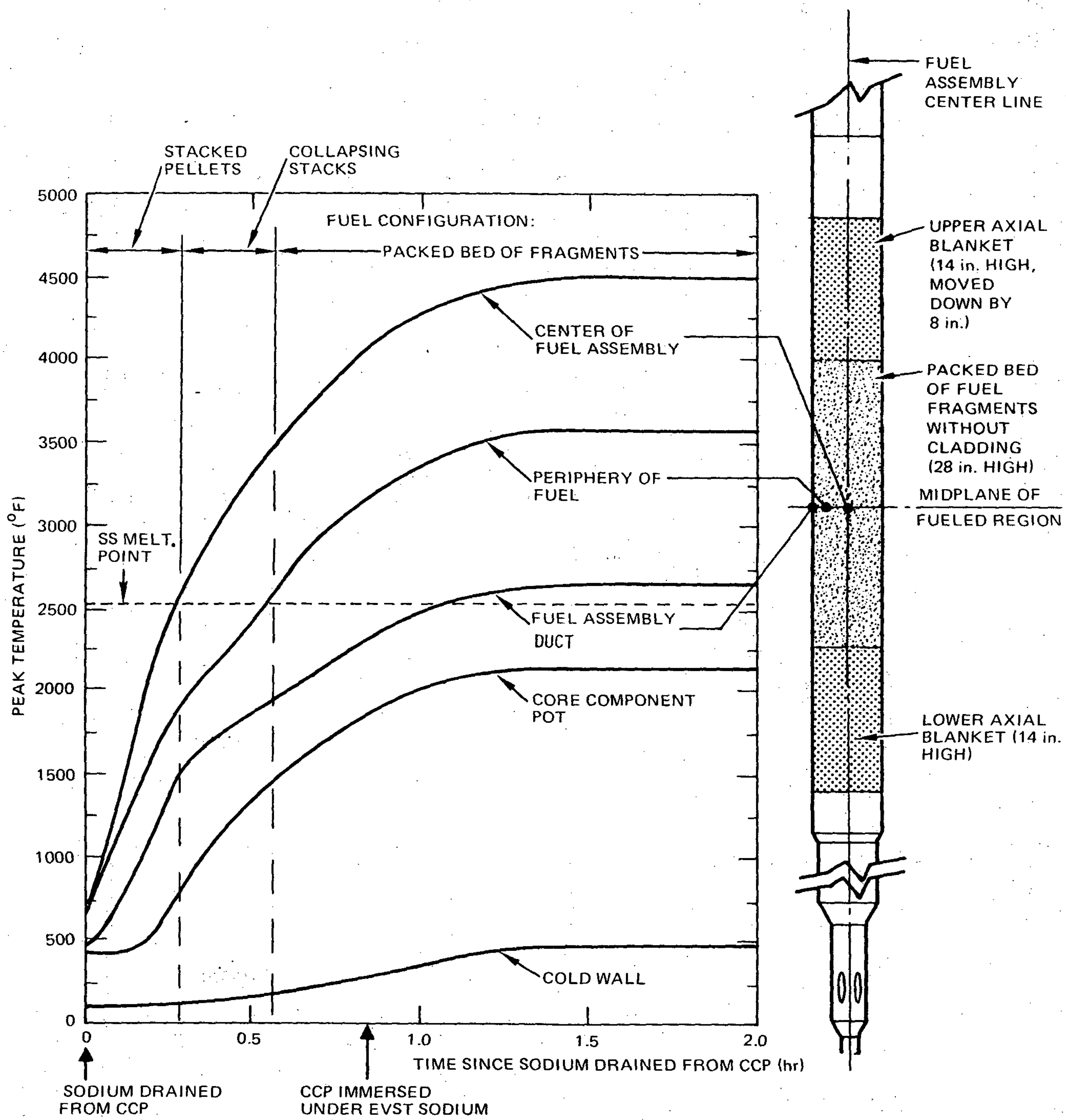

FIGURE 15.7.3.1-6 Peak Transient Temperatures in EVTM with Dry 20-kW Spent Fuel Assembly in CCP. - Packed Bed of Fuel Fragments

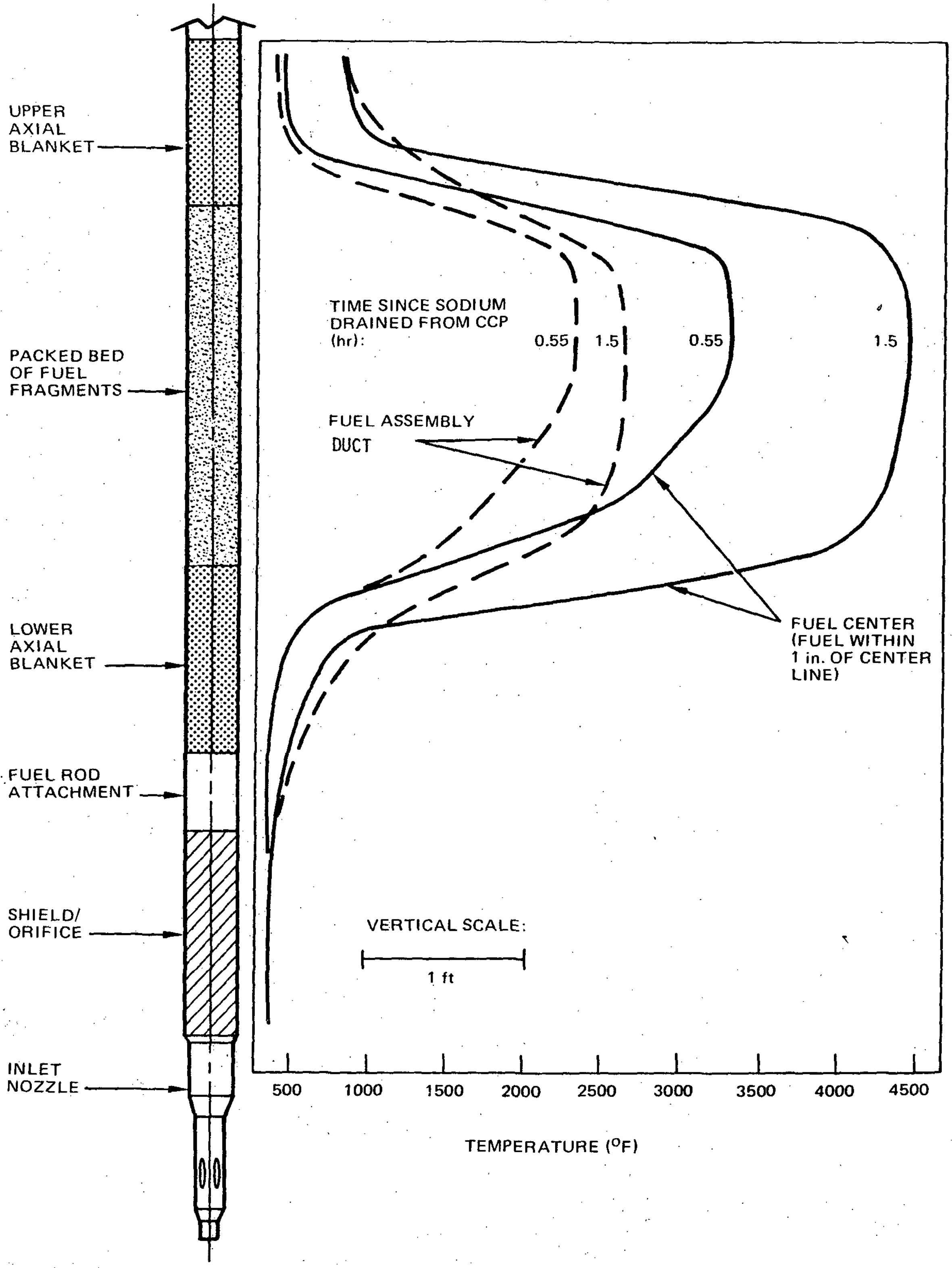

FIGURE 15.7.3.1-7. TRANSIENT AXIAL TEMPERATURE DISTRIBUTION IN DRY 20-kW SPENT FUEL ASSEMBLY WITH PACKED BED OF FUEL FRAGMENTS

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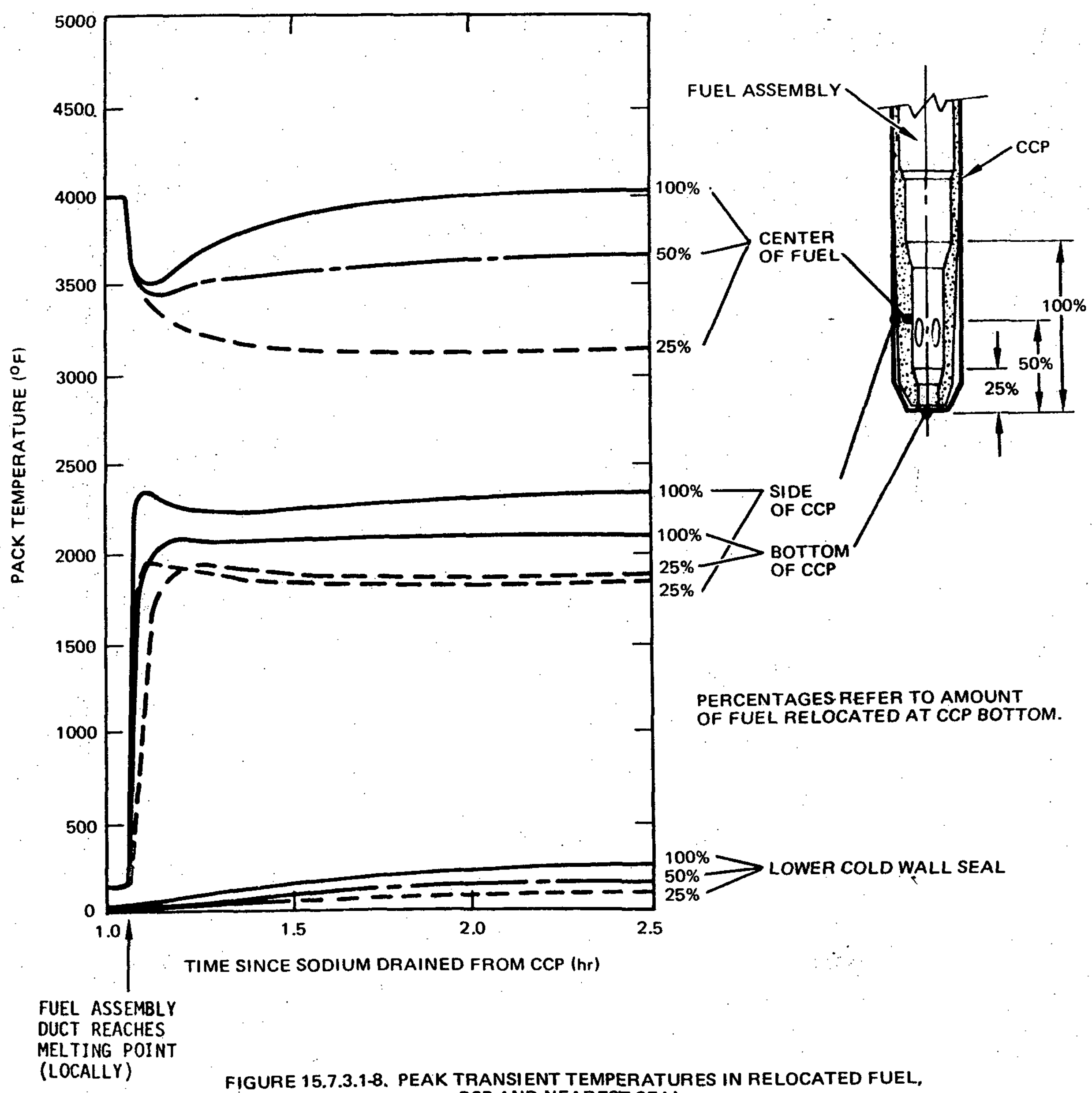

FIGURE 15.7.3.1-8. PEAK TRANSIENT TEMPERATURES IN RELOCATED FUEL, CCP AND NEAREST SEAL

APPENDIX 15.A

RADIOLOGICAL SOURCE TERM FOR ASSESSMENT OF SITE SUITABILITY

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40

15.A.1 INTRODUCTION

In accordance with Title 10 Code of Federal Regulations Part 50 (10CFR50), the CRBRP Project has submitted an Environmental Report (ER) and a Preliminary Safety Analysis Report (PSAR) to support an application
40 for a license to construct the CRBRP. These reports include an evaluation of a spectrum of postulated accidents. For each accident an analysis of the potential consequences to the health and safety of the public is presented. Consistent with the intent of Regulatory Guide 4.2, "Preparation of Environmental Reports for Nuclear Power Plants", the accident evaluations presented in the ER are based on realistic accident analyses and analytical assumptions. The evaluations presented in the PSAR are based on conservative accident analyses and analytical assumptions. The spectrum of accidents considered in Chapter 15 of the PSAR and Chapter 7 of the ER encompass Class 1 through Class 8 events. This spectrum constitutes the accidents included in the design base for the plant. Class 9 events are of such low probability that they can be excluded from the design bases.

29

In accordance with Title 10 Code of Federal Regulations Part 100 (10CFR100), a major fission product release from the core has been hypothesized for the purpose of determining the suitability of the selected site for the construction and operation of the CRBRP. In compliance with 10CFR100, the potential hazards resulting from this hypothesized release are not exceeded by those from any design basis accident analyzed in Chapter 15 of the PSAR. The radiological source term associated with this hypothetical release is specified in terms of percentages of fission products and fuel material released from the core to the Reactor Containment Building. The source term used for site suitability assessment is as follows:

- 100% Noble Gas Inventory
- 50% Halogen Inventory (25% Airborne)
- 1% Solid Fission Product Inventory
- 1% Plutonium

The applicant has utilized this source term in compliance with specific direction from the Nuclear Regulatory Commission (Ref. 1). However, while accepting this source term and committing to design features to assure acceptable consequences as a result of it, the applicant considers
40 this source term to be overly conservative.

The source term specified by NRC not only envelopes all design basis accidents considered in Chapter 15, but further envelopes a wide range of conservatively hypothesized core-related events. Evidence, both analytical and experimental, supports the Applicant's position that compliance with the requirements of 10CFR100 could be demonstrated with a less stringent source term.

40 The potential radiological consequences of the above source term are conservatively calculated and compared to the guideline values of 10CFR100, thus providing the basis for conducting an assessment of the 29 site suitability.

15.A.2 SITE SUITABILITY SOURCE TERM

15.A.2.1 Source Term

The source term is identified in terms of percentages of fission products and fuel material released from the core to the Reactor Containment Building. The source term is itemized in Table 15.A-1. The indicated percentages of these materials are assumed instantly released to and uniformly distributed in the RCB. For the halogens, 50% of the halogens released to the RCB are assumed to immediately plateout on surfaces (consistent with LWR practice), thus being removed from the airborne source term available for leakage from the RCB, with the net result that 25% of the initial halogen inventory is assumed airborne in the RCB.

The initial core fission product inventories are based on end-of-cycle equilibrium core conditions for power operation at 975 megawatts-
54 thermal.

The specific isotopes included in each fission product category, as identified in Table 15.A-1, are as follows:

Noble Gases:	Xe, Kr
Halogens:	Br, I
40 Solids:	All remaining fission products

The plutonium and transuranic element heavy metal inventories are itemized in Tables 15.A-2 and 15.A-3.

Plutonium inventories (Table 15.A-2) are provided for core loadings fueled with low Pu-240 fuel. For low Pu-240 fuel the isotope Pu-238 is a trace element, and was consequently not included in the discussion of this fuel type in Section 4.3. The low Pu-240 fuel specifications provide that the maximum amount of trace Pu-238 will not exceed 0.15% of the total Pu. To insure a conservative source term assessment, this maximum trace amount of Pu-238 is assumed present since this isotope is a significant dose contributor. Low Pu-240 fuel can be obtained from available sources,
51 and will be used for the initial fueling of CRBRP.

40 15.A.2.2 Source Term Attenuation Within Containment

The principal exposure pathway associated with a major radioactivity release in containment is leakage of airborne material to the environment. Potential off-site exposure is dependent on the leakage rate from containment and the concentration of airborne radioactive material within the containment as a function of time. With the exception of gaseous radionuclides, considerable reduction in the airborne concentration of radioactive material is expected as a result of natural deposition processes. For characteristic LMFBR postulated accident source terms, and for the CRBRP site suitability source term in particular, a large portion of the airborne radioactivity is associated with non-gaseous species. The suspended concentration-time behavior of such source terms is predictable with current aerosol behavior models. Characteristic depletion mechanisms for suspended aerosols include: (1) settling due to gravity, (2) wall plating, and (3) agglomeration due to Brownian motion and/or agglomeration due to gravity.

The concentration-time behavior of the airborne radioactive source term in the RCB was computed with the HAA-3 computer code.

A code description and code references are provided in PSAR Appendix A.

40

Previous parametric studies, conducted to confirm the theoretical
40 basis of the HAA-3 code (see Appendix A) indicate that the predicted aerosol
behavior is rather insensitive to the selection of the Stokes' correction
factor (α) and the gravitational collision effeciency (ε). The actual values
of α and ε (0.1 and 1.0, respectively) used for the source term analysis are
judged to provide a conservative assessment of aerosol behavior.

HAA-3 is used to compute depletion factors as a function of time;
40 these factors are used as input to COMRADEX-II which computes leakage of
the airborne radioactivity from the RCB. Considerations pertinent to
40 leakage from the RCB are discussed in Section 15.A.2.3. Note that depletion
is considered only for non-gaseous species; no credit for depletion via
plate-out or settling is taken for gaseous species.

40 The aerosol depletion factors describe the rate (fraction/sec) at
which the suspended aerosol concentration is reduced via natural aerosol
deposition processes: plateout and settling. Mathematically, this factor
is analogous to a radioactive decay constant, which describes the rate of
decay (reduction) of the activity of a particular isotope. However, whereas
the rate of radioactive decay is constant with time, aerosol depletion is
time-dependent due to the changing suspended aerosol concentration, and
the fact that the aerosol deposition processes are dependent on the concen-
tration of the suspended aerosol. Physically, for an initially high con-
40 centration of suspended aerosol, the rate of depletion is relatively rapid;
however, as the aerosol concentration decreases, due to depletion, the rate
of depletion decreases.

40 The time-dependent depletion factors are computed in HAA-3 by
considering the change in the suspended concentration over a series of
many small time increments, the sum of which equal the total time interval
being evaluated.

Consider the time interval $t_2 - t_1 = \Delta t$, where $t_2 > t_1$, and let

A_i = suspended aerosol mass at t_i

P_i = plated aerosol mass at t_i

S_i = settled aerosol mass at t_i

A_i, P_i and S_i are determined by HAA-3. The aerosol depletion
factor, defined at time $(t_1+t_2)/2$, is then computed as follows:

$$\frac{\text{Mass Plateout Rate} + \text{Mass Settling Rate}}{\text{Average Mass Suspended}}$$

40 $$= \frac{(P_2-P_1)/\Delta t + (S_2-S_1)/\Delta t}{(A_2 + A_1)/2} \quad \text{time}^{-1}$$ 25

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The mass associated with the non-gaseous portion of the source term, initially airborne in the RCB, is shown in Table 15.A-4. The resultant initial airborne concentration is also provided.

57 The quantity of fuel (62.4 kg) included in the source term aerosol analysis was selected to represent 1% of the total (core plus blanket)
51 plutonium-oxide mass plus 1% of the core uranium-oxide mass. The mass of the uranium-oxide in the blanket was not included in the aerosol analysis. This approach is conservative since including the uranium blanket mass would result in a much higher initial airborne concentration and subsequently more rapid aerosol depletion. Even though the uranium blanket mass has been excluded from the aerosol analysis, it has been conservatively assumed that the radioactive inventory of the blanket is included in the source term.

Table 15.A-5 presents the important input parameters to the HAA-3 code, used to compute the concentration-time behavior of the source term aerosol and resultant aerosol depletion factors. The time-dependent depletion factors computed by HAA-3 for the source term aerosol and used in the COMRADEX radiological analysis are itemized in Table 15.A-6.

15.A.2.3 Containment Modeling

A complete description of the reactor containment/confinement system and the engineered safeguards associated with it is presented in Chapter 6 of the PSAR.

For the radiological analysis, it is conservatively assumed that all leakage (except bypass) from the RCB to the annulus is directly to the intake of the filter system. This assumption neglects any credit for delay time in the annulus. The recirculation flow was assumed to mix in 50% of the annulus volume. Only one-half of the annulus volume is used to be consistent with the 50% mixing assumption specified in
40 Standard Review Plan Section 6.5.3.

Leakage of airborne radioactivity from the RCB was assumed to occur at the containment design leak rate, 0.1% Vol/Day for the duration of the evaluation. The RCB is designed to limit leakage to 0.1% Vol/Day at a containment overpressure of 10 psig. The use of the containment design leak rate (0.1% Vol/Day) for the duration of the site suitability source term evaluation is conservative, since assuming a constant 10 psig containment overpressure for the duration of the site suitability source term evaluation is conservative.

A portion of the leakage from the RCB may bypass the confinement annulus. Chapter 6 of the PSAR identifies the individual containment penetrations contributing to bypass leakage; the majority of the bypass leakage is associated with the containment airlocks. The containment/confinement system is being designed to achieve a bypass leakage value of less than 1% of the RCB design leak rate, i.e., 1% x 0.1% Vol/Day = 0.001% Vol/Day. Sixty percent of this bypass leakage escapes directly to the outside atmosphere and the remaining forty percent escapes to the Reactor Service Building (RSB). The treatment of leakage to the RSB depends upon the status of the railroad door in the RSB. When the railroad door is closed, the RSB atmosphere is maintained at a negative pressure with respect to the outside atmosphere. When the railroad door is open, maintenance of a negative pressure in the RSB is not assured.

If the RSB railroad door is open, both doors of the equipment hatch airlock are secured and the airlock atmosphere is vented to the containment/confinement Annulus Filtration System. In this mode, essentially all (96.4%) the bypass leakage from the RCB to the RSB (40% of total bypass) is vented from the equipment hatch airlock directly to the Annulus Filtration System, where it is subject to filtration and recirculation prior to release to the environment. The remainder of leakage into the RSB (3.6%) escapes directly to the atmosphere. When the railroad door is closed, the airlock vent to the Annulus Filtration System is closed and the airlock atmosphere is isolated from the containment/confinement annulus. In this mode, all bypass leakage from the RCB to the RSB (40% of total bypass) escapes directly to the RSB where it is subject to recirculation and filtration prior to release to the environment.

Airlock operation with the railroad door open (i.e., with the airlock atmosphere vented to the annulus) results in larger potential off-site exposures for the site suitability source term analysis than operation with the railroad door closed and the radiological consequences are therefore presented when the railroad door is assumed open. Confirmation that this does result in more limiting exposures is given below.

When the railroad door is closed and all bypass leakage to the airlock escapes to the RSB, this leakage is filtered prior to ultimate release to the environment. Considering the efficiencies (99% particulate and 95% iodine) of the RSB filters and the recirculation flow pattern (1700 cfm exhausted per 14300 cfm recirculated), the net filtration efficiency of the RSB system is greater than 99% for both particulates and halogens. Conse-
40 quently, non-gaseous releases (which are controlling with respect to off-site

exposures for the source term) to the RSB are attenuated by at least two orders of magnitude when the railroad door is closed. When the railroad door is open and RSB filtration unavailable, 96.4% of the bypass leakage to the RSB is vented from the equipment airlock to the annulus, but 3.6% of the bypass leakage to the RSB leaks directly to the atmosphere. This provides an attenuation factor of about 28 for bypass leakage to the RSB. Therefore, the contribution of bypass leakage to off-site exposure is more limiting when the railroad door is open and the airlock is fully secured and vented to the Annulus Filtration System than when the railroad door is closed and RSB filtration is available.

The final containment related parameter pertinent to the source term radiological analysis is the shielding attenuation provided by the containment structures. The analysis considers the shielding provided by both the steel RCB and the concrete Confinement Building.

A summary of the containment/confinement related parameters used in evaluating the source term is provided in Table 15.A-7.

40 15.A.2.4 Environmental Dispersion

Atmospheric dilution factors (χ/Q's) applicable to discrete time intervals following postulated accidental releases have been established as a function of downwind distance from the CRBRP site. A detailed discussion of the development of these χ/Q's is provided in Chapter 2 of the
40 PSAR.

The specific χ/Q's used for the analyses are itemized in Table 15.A-8. These χ/Q's are the "95th percentile" values (atmospheric dilution is more favorable 95% of the time). Consistent with the Standard Review Plan, Section 2.3.4, the 0-2 hour exposure intervals at both the exclusion boundary and low-population zone were evaluated based on the <u>single-hour</u>, 95% χ/Q value.

15.A.2.5 Radiological Parameters

The major parameters relating the COMRADEX dose calculations are as follows:

The standard-man time-dependent breathing rates are identical to the values recommended in Regulatory Guide 1.4, and are as follows:

Time	Breathing Rate, m^3/sec
0-8 hrs.	3.47×10^{-4}
8-24 hrs.	1.75×10^{-4}
> 24 hrs.	2.32×10^{-4}

40

The inhalation dose factors (Rem/Ci inhaled) for internal exposure to the bone, lung, thyroid, and whole body are identical to the values recommended in Reference 2 for a standard adult. A number of isotopes included in the COMRADEX source term analysis are not included in Reference
1 2. For these isotopes, the inhalation dose factors are consistent with those provided in References 3 and 4.

External gamma whole body exposure is based on a semi-infinite cloud model per Regulatory Guide 1.4 for released material and includes direct exposure from material within the Reactor Containment Building.

40 15.A.2.6 Off-Site Exposure

The potential whole body and organ doses resulting from the site suitability source term are itemized in Table 15.A-9. In accordance with 10CFR100, off-site doses resulting from 2 hour exposure at the exclusion boundary (0.42 miles) and accident duration exposure at the low-
40 population zone (2.5 miles) have been determined.

For the actual radiological dose analysis, accident duration is defined as 30 days, i.e., dose calculations are continued out to 30 days. Based on the leakage characteristics assumed for the RCB (leak rate constant at design value) and the aerosol depletion effects within containment, incremental exposure beyond 30 days is insignificant. More
40 than 98% of the accident duration doses result from the first week of exposure at the LPZ.

For both the whole body and all organs considered, the most
40 limiting doses result from 2 hour exposure at the exclusion boundary. The doses at the low population zone in all cases are less than the exclusion boundary doses. Even with the conservative assumptions applied throughout the analysis, the doses calculated are in agreement with the applicable 10 CFR 100 guideline values.

15.A.2.7 Conclusions

NRC has specified a site suitability source term for CRBRP which
40 is conservative. The potential radiological consequences associated with this site suitability source term have been determined with the consistent
40 application of conservative analytical assumptions. The 2-hour exclusion boundary and accident duration low population zone exposure doses have been calculated. These doses were compared to the guideline values of 10CFR100 and the guidance provided by NRC in Reference 1. This comparison,
40 summarized in Table 15.A-9 indicates that the potential radiological consequences of this assumed source term are well within the guideline values of 10CFR100 and in agreement with the values established by NRC for use during the construction permit review, and thus provides the basis for
40 establishing suitability of the Clinch River site with respect to the criteria set forth in 10CFR100.

15A.3.0 REFERENCES

26 1. Letter, R. P. Denise (USNRC) to L. W. Caffey (CRBRP), May 6, 1976.

2. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I", Revision 1, October 1977.

3. NUREG - 0172, "Age Specific Radiation Dose Commitment Factors for a One-Year Chronic Intake", November 1977.

4. TI-001-130-051, "Internal Dose Factors for COMRADEX-II", Specht, E. R., February 25, 1975.

51 15 5. "Site Suitability Report In the Matter of CRBRP", Office of Nuclear Reactor Regulation, USNRC.

25

26 40

TABLE 15.A-1

SITE SUITABILITY SOURCE TERM

Isotope Class	% Inventory
Noble Gases	100%
Halogens	50% (25% Airborne)
Solid Fission Product	1%
40 Fuel (Including Plutonium)	1%

TABLE 15.A-2

HEAVY METAL* MASS (KG) INVENTORY IN THE CRBRP (EOEC)

	Fuel	Inner Blanket[a]	Radial Blanket[a]	Lower Axial Blanket	Upper Axial Blanket
End-of-Fourth-Cycle					
Pu-239	1216.	206.8	285.6	34.9	21.2
Pu-240	273.5	8.0	11.3	0.9	0.3
Pu-241	32.7	---	---	---	---
Pu-242	5.2	---	---	---	---
U-235	5.4	11.6	21.3	3.8	4.0
U-238	3421.	7381.	12936.	2149.	2165.
Fission Products	414.2	55.2	55.7	4.4	2.4
Total Heavy Metal	5368.0	7662.6	13309.9	2193.0	2192.9

* Heavy metal excludes oxygen.

51 57 (a) Including axial extensions

TABLE 15.A-3

CRBRP TRANSURANIC INVENTORY (EOEC)

	Isotope	Half-Life	Mass (gms)	Curies
	Np237	2.14×10^{6} Y	3.38×10^{3}	2.38×10^{0}
	Np238	2.1 D	1.50×10^{0}	3.93×10^{5}
	Np239	2.35 D	4.08×10^{3}	9.48×10^{8}
	Am241	458 Y	7.33×10^{3}	2.51×10^{4}
	$Am242^{m}$	152 Y	1.62×10^{2}*	1.57×10^{3}*
	Am242	16 H	3.81×10^{0}	3.08×10^{6}
	Am243	7650 Y	2.17×10^{2}	4.18×10^{1}
	Am244	10 H	8.69×10^{-4}	2.58×10^{4}
	Cm242	163 D	6.23×10^{2}	2.06×10^{6}
	Cm243	32 Y	2.27×10^{1}	1.04×10^{3}
	Cm244	18.1 Y	8.70×10^{0}	7.05×10^{2}
	Cm245	9320 Y	1.45×10^{-1}	2.57×10^{-2}
	Cm246	5480 Y	2.19×10^{-3}	6.77×10^{-4}
	Cm247	1.67×10^{7} Y	1.92×10^{-5}	1.69×10^{-9}
	Cm248	4.7×10^{5} Y	1.78×10^{-7}	7.29×10^{-10}
57 40	Cf252	2.55 Y	4.15×10^{-16}	2.22×10^{-13}

51 *Estimated Value

TABLE 15.A-4

MASS OF SOURCE TERMS INITIALLY AIRBORNE IN RCB

	Isotope Class	Mass (kg)
	Noble Gases*	74.34
	Halogens**	1.59
	Solid Fission Product	5.55
	Fuel	62.45
51	Total Non-Gaseous	69.59
57	Initial RCB Concentration (μgm/cc)	0.68

*Mass of Noble Gases excluded from aerosol analysis.

**25% of EOEC Inventory.

40

TABLE 15.A-5

HAA-3 INPUT PARAMETERS USED FOR SOURCE TERM ANALYSIS

Parameter	
Initial Concentration, Particles/cc	1.34×10^{8}
Count Mean Particle Radius, μm	0.1
Geometric Mean Deviation, μm	2.0
Aerosol Material Density, gm/cc	10.55
Stokes Correction Factor, α	0.1
Gravitational Collision Efficiency, ε	1.0
RCB Volume, cm^3	1.02×10^{11}
RCB Leak Rate, fraction/sec	1.16×10^{-8}
Plating Constant, Δ	4×10^{-5}

57
40

TABLE 15.A-6

AEROSOL DEPLETION FACTORS USED FOR SOURCE TERM

Time (Sec)	Depletion Factor (Fraction/Sec)
6.50-3	9.55-5
4.60-2	9.49-5
2.63-1	9.17-5
1.25+0	8.21-5
3.84+0	6.92-5
9.22+0	5.79-5
1.95+1	4.87-5
3.94+1	4.10-5
7.73+1	3.45-5
1.47+2	2.91-5
5.20+2	2.04-5
1.90+3	1.37-5
6.42+3	9.43-6
1.98+4	7.06-6
1.09+5	6.33-6
4.21+5	4.56-6
1.09+6	2.24-6
2.60+6	1.40-6

57

40

TABLE 15.A-7

CONTAINMENT/CONFINEMENT PARAMETERS USED FOR SOURCE TERM ANALYSIS

RCB Leakage to Annulus (Direct to Annulus Filter Intake)	0.1% Volume/Day
Annulus Flow Rates Filtered Exhaust Filtered Recirculation	 3000 CFM 3500 CFM per 1000 CFM Exhausted
Time Delay from Source Term Release to Initiation of Annulus Filtration	No Delay
Time Delay from Source Term Release to Initiation of Annulus Recirculation	<10 Seconds
Total Bypass Leakage (1% of RCB Leakage)	0.001% Volume/Day
Bypass Leakage Direct to Environment (60% of Total Bypass)	0.0006% Volume/Day
Bypass Leakage to the RSB (40% of Total Bypass)	0.0004% Volume/Day
Sources of Bypass Leakage to the RSB	96.4% Personnel and Airlock Equipment 3.6% All other sources
Gamma Shielding	1.5" Steel (RCB) Plus 4' Concrete
Filter Efficiencies Iodine Particulate 40 Noble Gases	 95% 99% 0

TABLE 15.A-8

METEOROLOGICAL PARAMETERS USED FOR SITE SUITABILITY ASSESSMENT

Exclusion Boundary (0.42 Miles)	X/Q (sec/m^3)
0-2 Hours	3.12×10^{-3}*
Low Population Zone (2.5 Miles)	
0-2 Hours	8.55×10^{-4}*
2-8 Hours	2.85×10^{-4}
8-24 Hours	2.60×10^{-5}
1-4 Days	1.30×10^{-5}
4-30 Days	8.70×10^{-6}

---

40 * 0-2 Hour X/Q's based on single-hour 95% X/Q Values.

TABLE 15.A-9

OFF-SITE EXPOSURE SUMMARY

(Power Level = 975 Megawatts-Thermal)

Dose (Rem)

	Organ	10CFR100	2-Hour Site Boundary (0.42 Miles)	30-Day Low Population Zone (2.5 Miles)
	Bone	150*	7.2	4.1
	Lung	75*	1.6	.9
	Thyroid	300	23.1	12.6
57 51	Whole Body**	25	3.7	1.7

*Equivalent to 10CFR100 guideline values; see Reference 4.

**Includes inhalation, external gamma cloud, and direct gamma shine exposures.

40

TABLE 15.A.3-5

REFERENCE DESIGN SITE SUITABILITY SOURCE TERM DOSE SUMMARY

Organ	10CFR100 Rem	Site Boundary (0.42 Mi.) 2 Hours Dose Rem	LPZ (5.0 Mi.) Accident Duration Dose Rem
Bone	150*	14	2.7
Lung	75*	6	1.1
Thyroid	300	11	1.9
Whole Body**	25	3.6	0.13
Beta Skin	-	1.2	0.11

* Not covered in 10CFR100; used as guideline values.

** Includes inhalation, external gamma cloud, and direct gamma shine exposures.

ATTACHMENT 1

Analyses of Three Extreme Conditions
Considered in Developing Source Term

I. Purpose

The purpose of this attachment is to present the principal parameters, assumptions, and analytical methods used in the analysis of the three extreme hypothesized conditions discussed in the main text of this Addendum. The three conditions considered are as follows:

1. Complete reaction of all RCB oxygen with radioactive primary sodium.

2. Instantaneous release, during reactor operation, of the radioactive cover gas; with the cover gas burdened with the equivalent of the total noble gas inventory of an end-of-cycle equilibrium core fuel assembly.

3. Complete and instantaneous release of all radioactivity in a Primary Sodium Cold Trap at the end of its design life.

Section II presents an outline of the analyses leading to the specification of an initial RCB radioactive source term for each of the hypothesized conditions. Considerations pertinent to (1) the attenuation of the non-gaseous portion of these source terms via natural aerosol depletion mechanisms, and (2) releases to the environment via RCB leakage are discussed in Sections III and IV, respectively.

II. Specification of Initial RCB Source Terms

As discussed in the main text of this Addendum, the three extreme hypothesized conditions do not represent mechanistic sequences. Rather, each condition was arbitrarily hypothesized to result in an instantaneous radioactive source term release to the RCB. Considerations pertinent to the definitiion of this instantaneous source for each event considered are as follows:

1. Complete RCB Oxygen and Primary Sodium Reaction

During reactor operation, all primary sodium is contained within inerted cells. Therefore, during operation, even if primary system sodium leaks are postulated, reaction of this leaked sodium with RCB oxygen could not occur. Communication between primary heat transport system cells and the RCB could only occur during maintenance or refueling operations. Consequently, for the evaluation of this extreme condition, it was assumed that the primary sodium has decayed for 10 days after reactor shutdown. This 10 day period is typical of decay times required for maintenance operations involving de-inerting and opening primary sodium containing cells. During such maintenance operations, reaction of primary sodium (10 days decayed) with the RCB oxygen is possible although highly improbable, since the failure of a primary system sodium containing component must be postulated.

The radioactive content of the sodium used for this evaluation is based on continuous plant operation for 30 years. The design basis radioisotope concentrations (fission and corrosion product concentrations resulting from 30 years operation with 1% failed fuel and a plutonium concentration of 100 ppb) at 10 days after shutdown, were assumed present in the sodium. The radioisotope concentrations in the sodium under these conditions are summarized in Table 15.1.1.1-1 of the PSAR. For convenience, the radioisotope concentrations per Table 15.6.1.1-1 have been grouped by isotope class as follows:

Isotope Class	μCi/gm Sodium
Na	1.94
I	21.68
Pu	0.415
All Others	84.90

Based on the RCB free volume 4.1×10^6 ft^3), a normal air RCB atmosphere (21 vol% O_2), and the formation of sodium monoxide (Na_2O), a total of approximately 200,000 lbs. of sodium can be reacted with the RCB oxygen. Arbitrarily assuming that this quantity of sodium is reacted, and assuming that the radioisotope concentrations in the reacted sodium are the same as the initial concentrations, in the primary sodium, the initial RCB radioactive source term for this condition is calculated as follows:

$(2 \times 10^5 \text{ lbs}) (453.6 \text{ gm/lb}) (10^{-6} \text{ Ci}/\mu\text{Ci}) (X\mu\text{Ci/gm})$,

where the above expression is computed for each isotope class. The resultant RCB initial radioactive source term is as follows:

Isotope Class	Source Term, Curies
Na	180
I	2,000
Pu	38
All Others	7,700

It is recognized that long burning times (days) would be required to react a substantial fraction of the RCB oxygen. However, the total reaction products are assumed to be instantly dispersed throughout the RCB atmosphere.

2. Instantaneous Reactor Cover Gas and One Fuel Assembly Noble Gas Inventory Release to the RCB

Based on the design basis cover gas radioactive inventory per Table 11.3-2 of the PSAR, and the noble gas inventory of one end-of-cycle equilibrium core fuel assembly, this hypothesized condition results in an initial RCB source term of 2.56×10^6 curies of noble gas.

3. Complete and Instantaneous Release of Cold Trap Activity

For this evaluation, it is assumed that the total, end-of-life, design basis radioactive inventory of a Primary Sodium Cold Trap is instantly released to the RCB. Based on the design basis Cold Trap Inventory per Table 12.1-10 of the PSAR, and the assumed instantaneous release, the initial RCB source term, per isotope class, is as follows:

Isotope Class	Initial RCB Source Term, Curies
Na	6.7×10^4
I	2.8×10^3
Pu	2.3×10^2
All Others	2.7×10^5

III. Source Term Attenuation Within Containment

As discussed in detail in the main text of this Addendum, (see Section 15.A.3.2.1), with the exception of gaseous species, considerable reduction in the RCB airborne concentration of radioactive material is expected as a result of natural deposition processes. For the three extreme hypothesized conditions investigated, a conservative estimate of the attenuation afforded by characteristic aerosol depletion mechanisms has been made. The estimate is based on a number of parametric aerosol case studies, to be discussed shortly. The concentration-time behavior of the aerosol for the cases considered was computed with the HAA-3 computer code. The following discussion outlines the methodology used to provide a conservative aerosol attenuation factor for the three extreme conditions evaluated.

Because the three extreme conditions have been derived non-mechanistically, a precise statement of the mechanism or nature of release to the containment is not possible. However, a spectrum of potential release mechanisms can be examined. A range of potential release conditions has been selected to parametrically evaluate the behavior of the airborne radioactive concentration in containment. This range encompasses hypothetical releases as well as mechanistic, although highly improbable, releases. The purpose of this parametric investigation is to demonstrate that over the entire range of potential release mechanisms - hypothetical to mechanistic - considerable depletion of the airborne containment source term can be expected.

Four parametric cases based on the Cold Trap release, have been examined. As will be discussed later, these parametric cases also encompass aerosol concentrations associated with the other extreme conditions (except the instantaneous cover gas release plus fuel assembly noble gas release, which does not have an aerosol characteristic). The parametric cases are:

Case 1 Hypothetical instantaneous release of the entire sodium and radionuclide inventory. The total sodium inventory is approximately 5000 lbs. and the end-of-life radionuclide mass is approximately 30 lbs. For this evaluation, the entire inventory (sodium and radionuclides) is assumed dispersed instantly in the RCB; an initial containment aerosol concentration, characterized by Na_2O properties, is specified. No mechanism can be identified that could lead to such a release, but the case is included to maximize the suspended sodium mass in containment.

Case 2 Hypothetical instantaneous release of the entire radionuclide inventory with no sodium. For this evaluation, an initial containment aerosol concentration, characterized by Cesium properties (more than 99% of the radionuclide mass consists of Cs), is specified. No mechanism can be identified that could lead to such a release, but the case is included to minimize the suspended sodium mass in containment.

Case 3 Postulated Cold Trap rupture and resultant sodium pool fire in the Cold Trap cell. The Cold Trap cell is inerted (2% O_2) during normal operation, maintenance and replacement operations will be conducted in an inerted atmosphere or when the sodium in the trap, to be replaced, is frozen, thus eliminating the potential for a sodium fire. A complete failure was assumed to result in a 5000 lb. sodium fire in the inerted cell. A SOFIRE II analysis was conducted to determine the resulting burning rate and aerosol generation rate. No credit for retention, plate out, or settling of the aerosol in the cell was taken. It was conservatively assumed that all the aerosol generated during combustion was released directly to the upper containment volume. The pool fire analysis indicates that combustion is completed (O_2 depleted) in approximately 5 hours. A time-dependent aerosol source term, 5 hours in duration, characterized by Na_2O properties, is thus defined. The total available oxygen in the cell can react approximately 30 lbs. of the 5000 lb. Na spill. Therefore, for this more mechanistic, yet still highly improbable release, only a small fraction of the radioactive inventory could be released to containment. However, for conservatism and comparison purposes with the other cases, the entire radioactive inventory is assumed evenly distributed in the aerosol released to the RCB.

Case 4 Postulated Cold Trap inlet piping rupture and resultant sodium spray fire. A rupture of the inlet pipe of the Cold Trap regenerative heat exchanger was evaluated because this location is subject to the highest system pressure and temperature. The entire sodium discharge from the rupture was assumed to react as a spray. A SPRAY I analysis was conducted to determine the sodium burning rate and aerosol generation rate. No credit for retention, plate-out, or settling of the aerosol in the cell was taken; 100% of the sodium reacted during the spray was assumed to be released as aerosol directly to the upper containment volume. The spray fire analysis indicates that combustion is complete (O_2 depleted) in approximately 200 seconds. A time-dependent aerosol source term, 200 seconds in duration, characterized by Na_2O properties is thus defined. For conservatism and comparison purposes with the other cases, the entire radioactive inventory is assumed evenly distributed in the aerosol released to the RCB.

For each case, the suspended aerosol concentration-time behavior in the RCB was calculated with the HAA-3 computer code. The results of these analyses are shown in Figure I, and summarized in Table I. Recall that for each of these parametric cases, the total radioactivity released to the RCB is identical. The differences in the magnitude of the suspended aerosol result from the varying quantity of sodium assumed released for each case. The slopes of the curves in Figure I are indicative of the rate at which the containment radioactive source term for leakage is reduced by aerosol depletion. The significant parameter with respect to source term reduction is the rate of suspended aerosol depletion. For each case, three characteristic depletion times, T_x, have been computed and itemized in Table I. As used here, depletion time is defined as the time, measured from the start of the assumed event, at which the suspended aerosol concentration has been reduced to some fraction of its peak value. As the data in Table I indicate, the initial depletion times decrease (more rapid depletion) as the initial mass of sodium released increases. For example, the peak aerosol concentration for Case 1, maximum sodium, is reduced to one-half its value in approximately 1 hour; whereas, for Case 2, minimum sodium, $T_{1/2}$ is approximately 20 hours. At very long times, the airborne concentration in containment is relatively insensitive to the initial aerosol conditions. Regardless of the mechanism assumed for aerosol release to the containment, the airborne radioactive source has been reduced by a factor of 10 within 4 days and by a factor of 100 within 20 days. (Note: This reduction is due to aerosol depletion mechanisms and not to source depletion via containment leakage. At the RCB design leak rate, 0.1%/day, only 2% of the initial source is removed via leakage within 20 days.)

It is clear from this parametric investigation that a significant aerosol reduction factor can reasonably be applied. Based on the data presented in Table I, an aerosol reduction factor of 2 could reasonably be applied within 1 day after the postulated event. A reduction factor greater than 10 would be reasonable for times greater than 4 days. However, to insure the conservatism of the evaluation, it was assumed that (1) no depletion occurs for the first 24 hours following the postulated release, (2) depletion between 1 day and 30 days results in a factor of 10 reduction and (3) long term depletion effects (from 30 days onward) results in an additional factor of 10 reduction in the RCB source term.

In addition to being applied to the instantaneous release of activity from a cold trap failure, these depletion factors are also conservatively applied to the extreme condition of the complete RCB oxygen reaction with primary sodium. As noted previously, the depletion rate increases as the initial aerosol concentration increases. The four cases presented in Figure I and Table I have peak concentrations ranging from 0.04 to 26 μgm/cc. The aerosol reduction factors were selected to be conservative over this entire range of concentrations, and would be even more conservative for higher concentrations. The complete RCB oxygen reaction with primary sodium would result in an initial aerosol concentration of 10^3 μgm/cc if the aerosol concentration were (unrealistically) assumed to be formed and released instantaneously. More realistic rates of aerosol formation, based on pool burning phenomena, would reduce the peak concentrations by about two orders of magnitude. Nevertheless, over this whole range, the aerosol reduction factors applied to the source term are quite conservative. The other extreme condition evaluated is the instantaneous release of all reactor cover gas burned with the noble gas from one fuel assembly. This condition does not involve aerosols; consequently, no aerosol reduction factors are applied in that case.

IV. Environmental Release Via RCB Leakage

Leakage of airborne radioactivity from the RCB to the environment was assumed to occur at the containment design leak rate. The RCB is designed to limit leakage to 0.1% Vol/Day at a containment overpressure of 10 psig. Use of the containment design leak rate is conservative, as discussed in the main text of this Addendum (see Section 15.A.3.2.2).

The fractional containment leakage, $\lambda \ell$, corresponding to 0.1% Vol/Day is 1.16×10^{-8}/sec. Since the volumetric rate of containment leakage is constant, but the mass leakage rate is decreasing because the contained mass within the RCB is decreasing as a result of leakage and aerosol depletion, the containment leakage as a function of time is computed exponentially, where λa is the mass leakage decay constant.

Based on the conservative assessment of aerosol depletion effects, discussed in Section III, a time-dependent aerosol depletion factor, λa, has been defined. This depletion factor is applied only to non-gaseous species; no credit for depletion of gaseous species is taken. A non-depleting in-containment source is assumed to persist for 24 hours ($\lambda a = 0$). Between one day and 30 days, the in-containment source is attenuated exponentially ($\lambda a = 9.19 \times 10^{-7}$/sec) such that at 30 days, a factor of 10 reduction is effected in the in-containment source, which conservatively accounts for the cumulative effects of aerosol depletion during the 30 day period. At the end of 30 days, the remaining, non-gaseous, in-containment source is further reduced by a factor of 10,and conservatively assumed released to the environment as a puff release.

Based on continuous containment leakage at its design rate, and the aerosol depletion effects discussed above, the initial RCB source terms (where the specific isotopes comprising each isotope class are treated separately to account for radioactive decay burning holdup within containment), the activity released to the environment per isotope class and time increment (2 hours and ∞) was determined for each of the three extreme conditions considered. The results of these analyses have been previously summarized in Tables 15.A.3-1 and -2.

TABLE I

PARAMETRIC AEROSOL DEPLETION STUDIES

Parameter	Case 1 Instantaneous Source 5000 lbs. Na	Case 2 Instantaneous Source, 30 lbs. Radionuclides	Case 3 Time Dependent Source 5 hour Na Pool Fire	Case 4 Time Dependent Source 200 seconds Na Spray Fire
Peak Aerosol Concentration, μgm/cc	26.40	0.109	0.044	0.133
Time to Reach Peak, sec.	0	0	6052	202
T_x - Depletion Time, sec.				
$T_{1/2}$	4.0×10^3 (1.1 hrs.)	7.0×10^4 (19.4 hrs.)	5.9×10^4 (16.4 hrs.)	5.9×10^4 (16.4 hrs.)
$T_{1/10}$	7.5×10^3 (2.1 hrs.)	3.0×10^5 (3.5 days)	3.0×10^5 (3.5 days)	2.5×10^5 (2.9 days)
$T_{1/100}$	2.7×10^4 (7.5 hrs.)	1.2×10^6 (13.8 days)	1.6×10^6 (18.5 days)	1.3×10^6 (15 days)

T_x - Time, measured from t = 0, at which Concentration = x(Peak Concentration).

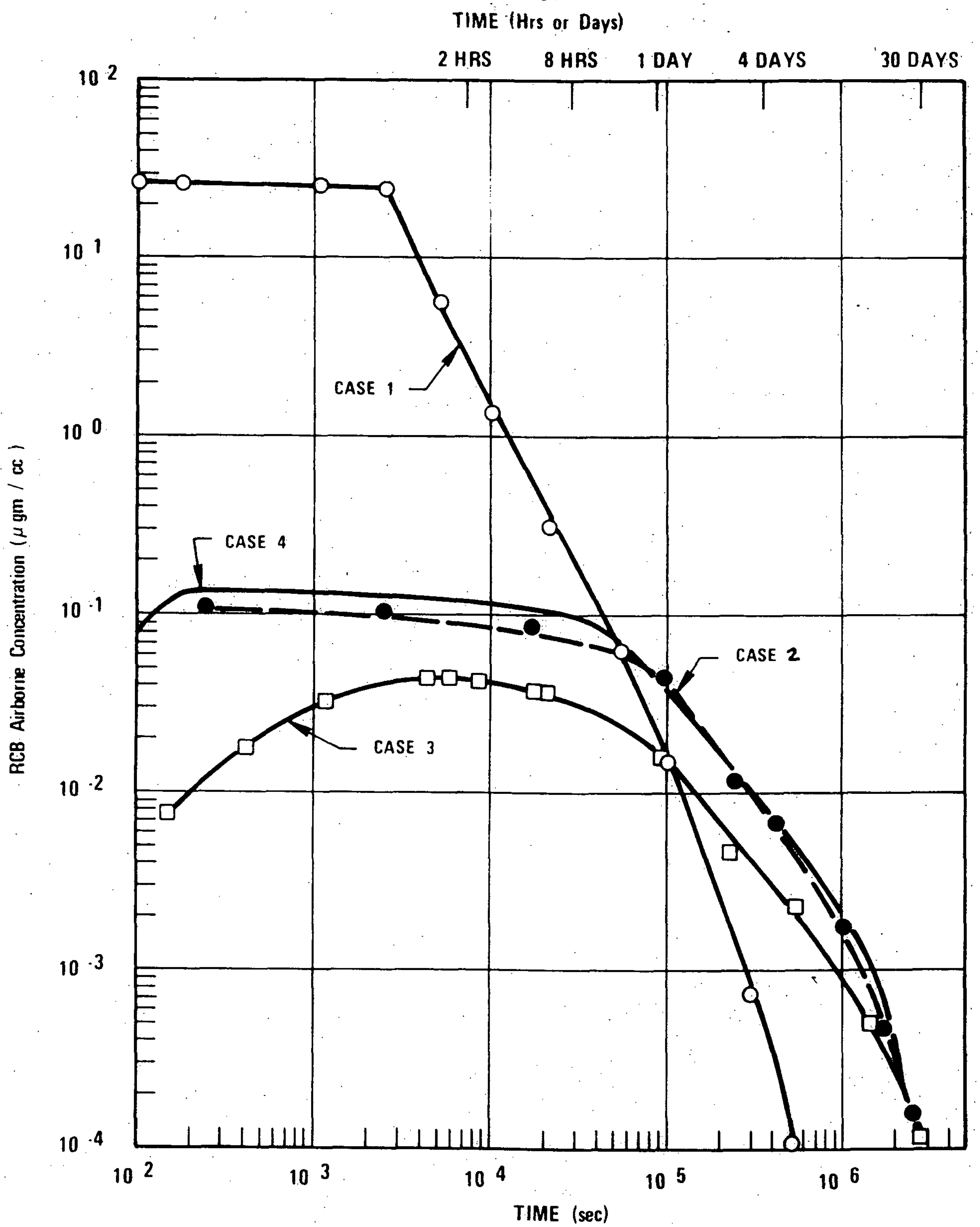

Figure I Parametric Aerosol Depletion Studies
RCB Aerosol Concentration-Time Behavior

-744-1

Amend. 4
Sept. 1975

CLINCH RIVER

BREEDER REACTOR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 16

TECHNICAL SPECIFICATIONS

PROJECT MANAGEMENT CORPORATION

CHAPTER 16 TECHNICAL SPECIFICATIONS

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16.1 DEFINITIONS

16.1.1 Reactor Operating Condition

16.1.1.1 Rated Power

Rated power is defined as a steady state thermal power output of 975 MWt.

16.1.1.2 Thermal Power

Thermal power is the total rate of thermal energy input to the primary coolant from components inside the reactor vessel.

16.1.1.3 Normal Reactor Power Operation

The reactor is operating between and including the state points of
59 40% rated power and rated power.

16.1.1.4 Two Loop Reactor Power Operation

The reactor is critical, two loops are in operation, and the neutron flux power range instrumentation indicates not more than TBD reactor power.

16.1.1.5 Transitory Operation

The reactor is operating between the state points of refueling, hot shutdown, hot standby, and 40% rated power, exclusive of these.

16.1.1.6 Hot Standby

See Table 16.1-1.

16.1.1.7 Hot Shutdown

See Table 16.1-1.

16.1.1.8 Refueling

59 See Table 16.1-1.

16.1.1.9 Reactor Startup

Reactor Startup is the sequence of operations in which the reactor is
59 brought from hot shutdown to Normal Reactor Power Operation of 2 loop Reactor Power.

Amend. 59
Dec. 1980

16.1.1.10 Operating Cycle

The interval between the end of one refueling outage to the end of the next subsequent refueling outage is one operating cycle.

16.1.1.11 Refueling Outage

Refueling Outage is that period of time between the shutdown of the unit prior to a refueling and the startup of the unit after that refueling. When refueling outage is used to designate a surveillance interval, the surveillance will be performed during the refueling outage or up to six months before the refueling outage. When a refueling outage occurs within eight months of the previous refueling outage, the surveillance testing need not be performed. The maximum interval between surveillance tests is 20 months.

16.1.1.12 Changes in Core Geometry

The addition, removal, relocation, or other movement of any material above the core support plate, below the upper internals or within the core barrel except for functions normally performed during reactor operation in accordance with intended design of equipment such as control rod movement shall constitute a change in core geometry.

16.1.1.13 Reactor Critical

The neutron chain reaction is self-sustaining and keff = 1.0.

16.1.1.14 Reactivity Units

Reactivity units expressed as dollars, multiplied by the effective de-
51 layed neutron fraction of 0.0034 gives reactivity units expressed as $\Delta k/k$.

16.1.2 Reactor Core

16.1.2.1 Fuel Assembly

A Fuel Assembly is an arrangement of 217 fuel rods, containing pellets of (Pu,U) O_2 and axial blanket pellets of UO_2, held in a triangular array by a spiral wire wrap spacing inside a hexagonal duct.

16.1.2.2 Blanket Assembly

1 A Blanket Assembly is an arrangement of 61 rods containing only UO_2 pellets in a triangular array.

16.1.2.3 Control Assembly

A Control Assembly is an assembly of clad boron carbide pins in a hexagonal lower guide assembly which has the same outside geometry as the fuel assembly.

16.1.2.4 Failed Fuel Pin

A fuel pin has failed when the cladding cannot contain the radioactive fission products generated within the fuel, or the fill gas for new fuel.

16.1.3 Plant Protection System Instrumentation

16.1.3.1 Plant Protection System

The Plant Protection System consists of both the Reactor Shutdown System and the Engineered Safety Features System. The protection system encompasses all electric and mechanical devices and circuitry (from sensors through actuated devices) which are required to operate in order to place or restore a Nuclear Steam Supply System to a design safe state.

16.1.3.2 Instrument Channel

An Instrument Channel is the combination of sensor(s), signal processing elements, output devices and other components and circuitry as required to measure and evaluate a process variable for the purpose of observation, control and/or protection of the process system. A channel may produce both analog signal outputs and discrete (signal or electromechanical component operation) outputs. The channel terminates and loses its identity where individual channel outputs are combined.

16.1.3.3 Protective Logic Channel

Protective Logic Channel is defined as an arrangement of relays, contacts, or other components which operate in response to instrument channel outputs to produce a decision output. At the channel level, the decision output is the initiation of a protective action signal or the operation of an actuation device (relay, breaker or other). At the system level, the decision output is the operation of a sufficient number of actuation devices and the associated actuated equipment as required to place or restore a Nuclear Steam Supply System to a design safe state. The channel is deemed to include the actuation devices.

16.1.3.4 Actuation Device

An Actuation Device is a component or assembly of components that directly controls the motive power (electricity, compressed air, etc.) for actuated equipment. The following are examples of an actuation device: a circuit breaker, a relay, and a valve (and its operator) used to control compressed air to the operator of a containment isolation valve.

16.1.3.5 Operable

For a component or system to be operable, it shall be properly installed in the system and capable of performing the intended functions in the intended manner as verified by testing and tested at the frequency required by the Technical Specifications.

16.1.3.6 Operating

A component or system is operating when it is performing the intended functions in the intended manner.

16.1.3.7 Actuated Equipment

Actuated Equipment is defined as a component or assembly of components that performs or directly contributes to the performance of a protective function such as reactor trip, containment isolation, decay heat removal. The following are examples of actuated equipment: an entire control rod and its release mechanism, a containment isolation valve and its operator, and an auxiliary feed pump and its prime mover.

16.1.3.8 Instrument Channel Calibration

Instrument Channel Calibration shall consist of the adjustment of channel output(s) such that it responds, with acceptable range and accuracy, to known values of the parameter(s) which the channel monitors. Calibration shall encompass the entire channel, including all channel outputs and shall be deemed to include the channel functional test.

16.1.3.9 Instrument Channel Functional Test

The injection of simulated signal(s) into the channel as close to the primary sensor(s) as practicable to verify operability, including all channel outputs, as appropriate, shall constitute an Instrument Channel Functional Test.

16.1.3.10 Instrument Channel Check

An Instrument Channel Check is defined as the qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument outputs with the outputs of other independent instruments measuring the same variable.

16.1.3.11 Logic Channel Functional Test

A Logic Channel Functional Test consists of the application of input signals, or the operation relays or switch contacts, in all the combinations required to produce the required decision outputs including the operation of all actuation devices. Where practicable, the test shall include the operation of the actuated equipment as well; i.e., pumps will be started, valves operated, etc.

16.1.3.12 Degree of Redundancy

With reference to redundant instrument or logic channels, the Degree of Redundancy is the difference between the number of operable channels and the minimum number of these channels which, when tripped, will cause an automatic system actuation.

16.1.3.13 Protective Function

A Protective Function is the monitoring of one or more plant variables associated with a particular plant condition, and the initiation and completion of a particular Protective Action, at values of the variables established in the Design Basis. Protective Action is considered complete when the condition initiating the action is brought to a status at which the consequences of terminating the Protective Action are considered to be acceptable.

16.1.3.14 Engineered Safety Features

Engineered Safety Features are all Protective Subsystems which function:

to mitigate the consequences of an incident, and to provide for decay heat removal, for example:

Containment Systems
Reactor Guard Vessel
PHTS Major Components Guard Vessel
Residual Heat Removal System
Habitability Systems

The Reactor Shutdown System is excluded.

16.1.3.15 Class 1E System (Electrical)

The systems that provide electric power used to shutdown the reactor and limit the release of radioactive material following a design basis event constitute a Class 1E System.

16.1.4 Safety Limit

Safety limits for nuclear reactors are limits upon important process variables which are found to be necessary to reasonably protect the integrity of certain of physical barriers which guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor shall be shut down. The licensee shall notify the Commission, review the matter and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude reoccurrence. Operation shall not be resumed until authorized by the Commission.

16.1.5 Limiting Safety System Setting (LSSS)

Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting shall be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor. He shall notify the Commission, review the matter and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude
59 reoccurrence.

16.1.6 Limiting Conditions for Operation (LCO)

Limiting conditions for operation are the lowest functional capability
or performance levels of equipment required for safe operation of the facility.
When a limiting condition for operation of a nuclear reactor is not met, the
licensee shall shut down the reactor or follow any remedial action permitted by
the technical specification until the condition can be met. The licensee shall
notify the Commission, review the matter, and record the results of the review,
including the cause of the condition and the basis for corrective action taken
59 to preclude reoccurrence.

16.1.7 Surveillance Requirements

Surveillance Requirements are requirements relating to tests, calibrations, or inspections to assure that the necessary quality of a system and its components is maintained; that the facility operations will be within the safety limits; or that the limiting conditions for operation will be met.

16.1.8 Containment Integrity

Conformance with all the following conditions:

1. All automatic containment isolation valves are operable, or secured in the closed position or isolated by closed manual valves or flanges.

2. All nonautomatic containment isolation valves which are not required to be open during accident conditions are closed and blind flanges are installed where required.

3. Refueling Hatch is closed.

4. At least one door in each air lock is closed and sealed.

16.1.9 Abnormal Occurrence

An abnormal occurrence means the occurrence of a plant condition that results in any of the following conditions:

1. A safety system setting less conservative than the limiting setting established in the Technical Specifications.

2. Violation of limiting condition for operation established in the Technical Specifications.

3. An uncontrolled or unplanned release of radioactive material from any plant system designed to act as a boundary for such material in an amount in excess of the limits prescribed in Technical Specifications.

4. Failure of a component of a Plant Protection System that causes the feature or system to be incapable of performing its intended function as defined in these Technical Specifications or in the Safety Analysis Report.

5. Abnormal degradation of one of the several boundaries designed to contain the radioactive materials resulting from the fission process.

6. Uncontrolled or unanticipated changes in reactivity greater than 1% $\Delta k/k$.

7. Observed inadequacies in the implementation of administrative or procedural controls such that the inadequacy causes the existence or development of an unsafe condition in connection with the operation of the plant.

8. Conditions arising from natural or manmade events that, as a direct result of the event, require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

TABLE 16.1-1. OPERATIONAL STATE POINTS (1)

	Refueling Conditions	Hot Shutdown	Hot Standby	40% Rated Power	Rated Power
Reactor	Subcritical K_{eff} <.95	Subcritical	Critical with the power level not to exceed 5% of rated power	Critical @40% Power	Critical @100% Power
Control Rods	Primary & Secondary Rods fully inserted and disconnected	Primary & Secondary Rods fully inserted and unlatched	Secondary-Parked Primary - Critical elevation in banked configuration with Row 4 fully with-drawn	Secondary-Parked Primary - Critical elevation in banked configuration with Row 4 fully with-drawn	Secondary-Parked Primary - Critical elevation in banked configuration with Row 4 fully with-drawn
Scram Breakers	Open	Open	Closed	Closed	Closed
Turbine Generator Output Breakers	Open	Open	Open	Closed	Closed
PHTS	Pony Motor Flow, Na Temp @ $400^{o}F \pm 25^{o}F$	TBD Na Temp @ $600^{o}F + 50^{o}F - 10^{o}F$	Main Motor Flow (40% nominal) Na Temp @ $600^{o}F + 50^{o}F - 10^{o}F$	Main Motor Flow (40% nominal)	Main Motor Flow Power/Flow = 1
IHTS	Pony Motor Flow	Pony Motor Flow	Main Motor Flow (40% nominal) IHTS temp consistent with PHTS	Main Motor Flow (40% nominal)	Main Motor Flow Power/Flow $\approx$ 1
SGS	Recirc Pump & Motor cooling & seal cooling/injection systems operating	Recirc Pump & Motor cooling & seal cooling/injection systems operating	Recirc Pump & Motor cooling & seal cooling/injection systems operating	In Operation	In Operation
SGAHRS	PACCs operating as required (see 1.3.B)	PACCs operating as required	PACCs operating as required	On Standby	On Standby
BOP	TBD	Supply feedwater to SGS to support main steam system heating	Steam dump operat-ing as necessary. Feedwater system operating and main-taining proper feedwater temp.	Operating @ 40% Power Conditions	Operating @ Rated Power Conditions. Main steam flow 3.32×10^{6} lbs/hr

(1) All plant operations involve operating at, or about, a set of steady state conditions or making a transition from one set of conditions to another. These sets of conditions defined as state points are characterized in this table.

16.2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

16.2.1 Safety Limit, Reactor Core

16.2.1.1 Applicability

Applies to the limiting combination of coolant temperature change, thermal power, and primary coolant flow.

16.2.1.2 Objective

To prevent clad melting and thereby maintain a coolable core geometry.

16.2.1.3 Specification

For inlet temperature $\leq$ TBD°F and outlet temperatures $\leq$ TBD°F the combination of thermal power and primary coolant flow shall not exceed the coolant boiling curve (to be shown in the FSAR).

16.2.1.4 Basis

By requiring that the coolant boiling curve not be exceeded, a coolable core geometry can be guaranteed.

16.2.2 Limiting Safety System Settings

16.2.2.1 Applicability

Applies to trip settings for protective subsystems which monitor reactor and plant process variables.

16.2.2.2 Objective

To provide for automatic protection action such that the principal process variables do not exceed a safety limit.

16.2.2.3 Specification

The limiting safety system settings for the subsystems listed below will be supplied in the FSAR.

16.2.2.4 Basis

(To be supplied in the FSAR)

TABLE 16.2-1

PLANT PROTECTION SYSTEM PROTECTIVE FUNCTIONS

Primary Shutdown System

- Flux-Delayed Flux
- Flux-$\sqrt{\text{Pressure}}$
- High Flux
- Primary to Intermediate Speed Ratio
- Primary Pump Electrics
- Reactor Vessel Level
- 50 Steam-Feedwater Flow Mismatch
- IHX Primary Outlet Temperature

Secondary Shutdown System

- Modified Nuclear Rate
- Flux-Total Flow
- Startup Nuclear
- Primary to Intermediate Flow Ratio
- Steam Drum Level
- Evaporator Outlet Sodium Temperature
- Sodium Water Reaction

Amend. 50
June 1979

16.3 LIMITING CONDITIONS FOR OPERATION

16.3.1 Reactor Operating Conditions

16.3.1.1 Applicability

Applies to the reactor core and upper internal structures.

16.3.1.2 Objective

To assure that core parameters remain within the acceptable range.

16.3.1.3 Specification

a. The reactor power shall not be allowed to exceed the limiting curve of Figure 16.3.1 for three loop operation or Figure 16.3.2 for two loop operation. (Figures to be supplied in FSAR.)

b. The initial core of the CRBRP shall not be operated with fuel assemblies whose peak burn-up exceeds 80,000 MWD/MT.

c. The reactor shall not be made critical unless each core assembly position is occupied with an assembly which has been tested and approved for proper flow characteristics.

d. The average power coefficient, as measured by a change in control bank height between 40% and full power, shall be negative.

The net feedback reactivity in the startup range (0 + to
40% of full reactor power) may be slightly positive but
shall be consistent with the reactivity requirements of
42 Section 3.1.3.2 (Criterion 9 and 10).

e. 1. The Upper Internal Structure (UIS) shall not be rotated until it has been determined that the main primary pump motors have been shutdown.

2. Prior to startup of the main primary pump motors, the (UIS) shall be verified to be in its normal position for reactor operation.

If the reactor is critical and any of the above specifications are not met, it shall be placed in hot shutdown in an orderly fashion. The reactor shall not be taken critical again until a review has determined that continued operation shall represent no danger to the health and safety of the public.

16.3.1.4 Basis

By restricting the maximum combination of power and flow, the plant protection system will be able to mitigate the effects of the normal, upset, and emergency transients described in Appendix B of this report.

The peak burnup limit is specified to ensure fuel cladding integrity. As described in Chapter 15.1.2, the fuel cladding integrity is affected by the peak burnup.

Amend. 42
Nov. 1977

The specification on the loading of flow tested assemblies is intended to ensure that core coolant flow is not bypassed through an empty grid location or that an assembly with improper flow characteristics is not loaded into the core.

The dynamic characteristics of the reactor have been analyzed for a variety of power excursion events. These analyses used, as a minimum negative power coefficient, a value of TBD. The sum of the individual components, which make up the power coefficient, is TBD. By using a value of TBD the analysis includes a -20% uncertainty in the Doppler calculations. In power excursions studies, the minimum power coefficient results in maximum full assembly temperatures.

The final set of specifications are provided to ensure that the backup holddown functions of the UIS are operative whenever the main primary coolant pump motors are energized.

16.3.2 Primary Heat Transport System (PHTS)

16.3.2.1 System Components

16.3.2.1.1 Applicability

This specification applies to the operational limits of the PHTS components.

16.3.2.1.2 Objective

To specify the operational limitation of the PHTS components to assure continued power operation of the PHTS over the service life of the plant. The PHTS components consist of the primary pump, check valve, intermediate heat exchanger and connecting piping.

16.3.2.1.3 Specification

A. The PHTS components shall not be operated at temperatures and pressures exceeding those specified in Table 16.3.2.1.1.

B. In the event of a component boundary failure or if the sodium system is open for maintenance, the pumps shall not be operated at a speed greater than pony motor speed.

If any of the above specifications are not met, or cannot be complied with by the corrective action delineated in the appropriate operating manuals, an orderly shutdown of the plant shall be initiated. Follow-up action such as, system/component check-out, inspection and incident evaluation shall be performed in accordance with the approved procedures.

C. Upon indication of sodium leakage from any point in the reactor or HTS, the following action is required.

1. Preparation for a reactor shutdown shall be made and an investigation shall be initiated to establish the cause of the alarm.

2. If the existance of a sodium leak has been confirmed, the reactor shall be shutdown.

16.3.2.1.4 Basis

A. Specification 16.3.2.1.3A defines the structural design parameters of the PHTS components and piping.

B. Specification 16.3.2.1.3B insures that the sodium inventory above the minimum safe level of the reactor vessel is maintained to provide safe reactor shutdown and decay heat removal capability following a PHTS boundary failure.

C. Specification 16.3.2.1.3C defines the precautions to be taken when a possibility of a sodium leak is suspected. For such events, reactor scram is not immediately necessary since a combination of sodium make-up control and normal reactor shutdown would suffice to ensure adequate core cooling.

Confirmation of sodium leakage is defined by response from a leak detection method with a high reliability or redundant and/or diverse leak detection methods.

16.3.2.2 Startup and Shutdown

16.3.2.2.1 Applicability

Applies to the operating status of the primary sodium system during startup and shutdown operations.

16.3.2.2.2 Objective

To specify those limiting conditions to ensure continued reliable cooling of the reactor core and to limit potential radioactivity releases from the primary sodium system during plant startup and shutdown operations.

16.3.2.2.3 Specification

A. The reactor shall not be taken critical or operated with less than two heat transport system loops operating.

B. The reactor shall not be made critical unless each of the three primary sodium loops has been filled with sodium coolant to the normal level.

C. The reactor cover gas pressure shall be at a positive differential pressure not greater than 15 inches of water with respect to the reactor cavity and heat transport system cell pressures, whenever the reactor is loaded with fuel.

D. The maximum heat transport system heatup/cooldown rate between refueling and 600°F shall not exceed an average of 50°F/hr.

E. The maximum rate of change of the temperature of the primary heat transport system hot leg shall not exceed an average of 150°F/hr between standby and 40% thermal power operating conditions.

16.3.2.2.4 Basis

A. Specification 16.3.2.2.3A prohibits operation with only one loop in service when the reactor is critical to ensure sufficient redundancy for decay heat removal.

B. The precautions listed in Specification 16.3.2.2.3B ensure adequate sodium inventory for reactor core cooling and to eliminate the possibility of anomalous reactivity fluctuations due to gas entrainment from operating or idle primary loops.

C. A positive differential cover gas pressure is required at all times to prevent sodium oxidation due to in leakage of air. The upper limit on the cover gas pressure given in Specification 16.3.2.2.3C is set at a value that ensures that no excessive loss of primary sodium inventory occurs following a sodium boundary leak. Since the top of any of the guard vessels is 7 ft. above the minimum safe level in the vessel, and the pony motor shut-off head is limited to 5 ft., the specified limit of 15 inches of water (corresponding to 1.5 ft. of sodium at 1050°F) provides approximately a 1/2 foot of sodium margin, sufficient to preclude the possibility of excessive sodium inventory loss.

D. Specifications 16.3.2.2.3D and E ensure that the heat transport system piping and component structural design heatup/cooldown rate limits are not exceeded.

16.3.2.3 Maximum Cover Gas Activity

16.3.2.3.1 Applicability

Applies to the maximum concentration of radioisotopes in the reactor cover gas.

16.3.2.3.2 Objective

To specify the limiting concentration of radioisotopes in the cover gas for continued reactor operation.

16.3.2.3.3 Specification

(To be provided in FSAR if required.)

16.3.2.4 Maximum Impurities in Reactor Coolant

16.3.2.4.1 Applicability

Applies to the sodium purity requirement for the Primary Heat Transport System.

16.3.2.4.2 Objective

To specify the sodium purity requirement for operating the Primary Heat Transport System.

16.3.2.4.3 Specification

1. The Primary Heat Transport System shall be normally operated with the plugging temperature at least 50°F below the temperature of coldest part of the sodium system.

2. The plugging temperature shall not exceed 300°F when any part of the Heat Transport System is above 800°F.

If the above specifications are not met, or cannot be complied with by the corrective action delineated in the appropriate operating manuals in TBD hours, an orderly shutdown of the plant shall be initiated.

16.3.2.4.4 Basis

To ensure reliable operation of the PHTS to prevent the plugging of system components, and to minimize corrosion.

16.3.3 Intermediate Heat Transport Coolant System

16.3.3.1 Intermediate System Components

16.3.3.1.1 Applicability

Applies to the Intermediate Heat Transport System (IHTS) which connects the Steam Generator System (SGS) to the Intermediate Heat Exchanger (IHX).

16.3.3.1.2 Objective

To specify the operational limitation of the IHTS components to assure continued power operation of the IHTS over the service life of the plant.

16.3.3.1.3 Specification

1. During plant operation, at least two IHTS cooling circuits (associated with the operable primary loops) shall be operable.

2. The argon cover gas pressure in the intermediate sodium pump and in the intermediate sodium expansion tank shall not exceed 150 psig.

3. The IHTS temperatures and pressures, as determined at the various instrumented locations, shall be maintained at or below the values shown in Table 16.3.3.1.1.

4. The intermediate heat exchanger must be maintained with a positive intermediate-to-primary pressure differential.

If any of the above specifications are not met, or cannot be complied with by the corrective action delineated in the appropriate operating manuals, an orderly shutdown of the plant shall be initiated. Follow-up action such as, system/component check-out, inspection and incident evaluation shall be performed in accordance with the approved procedures.

16.3.3.1.4 Basis

During plant operation, at least two of the three Intermediate Heat Transport (IHTS) loops will be operational to assure an adequate heat removal capability if one of these loops should suffer a fault which would interfere with its heat removal function.

The maximum argon cover gas pressure combined with the pump head shall not be allowed to exceed the structural design limit of 325 psig. Limiting the cover gas pressure to 150 psig provides this assurance with a
29 suitable margin.

The values in Table 16.3.3.1.1 represent the structural design parameters of the Intermediate Heat Transport System.

The intermediate system is maintained at a higher pressure than the primary to insure that radioactive sodium does not enter the intermediate system from the primary system.

16.3.3.2 Maximum Impurities in Intermediate Coolant

16.3.3.2.1 Applicability

Applies to sodium purity requirements for the Intermediate Heat Transport System (IHTS).

16.3.3.2.2 Objective

To specify the sodium purity requirements for operating the Intermediate Heat Transport System.

16.3.3.2.3 Specification

1. The Intermediate Heat Transport System shall be normally operated with the plugging temperature at least 50°F below the temperature of the coldest part of the sodium system.

2. The plugging temperature shall not exceed 300°F when the temperature of any part of the Heat Transport System is above 800°F.

If the above specifications are not met, or cannot be complied with by corrective action delineated in the appropriate operating manuals in TBD hours, an orderly shutdown of the plant shall be initiated.

16.3.3.2.4 Basis

To ensure reliable operation of the IHTS during high temperature operation, and prevent the plugging of system compounds.

16.3.4 Steam Generation System (SGS)

16.3.4.1 Steam Generators

16.3.4.1.1 Applicability

Applies to the Steam Generator System (SGS) which provides independent steam generation capability for each of the three Reactor Heat Transport Systems.

16.3.4.1.2 Objective

To assure reliable and adequate cooling to maintain the IHTS sodium cold leg temperature at a value which will assure proper core cooling.

16.3.4.1.3 Specification

1. During operation, at least two SGS cooling circuits shall be operable, corresponding to the two operable PHTS and IHTS loops.

2. During reactor power operation, the water level in at least 2 steam drums shall not be below TBD inches. These shall correspond to the two operable PHTS and IHTS loops.

3. During reactor power operation, all power/safety relief valves on the steam generator system shall be operable for the SGS circuits operating in conjunction with PHTS and IHTS loops.

   If the above specifications are not met, or cannot be complied with by corrective action within TBD hours, an orderly shutdown of the plant shall be initiated, and the plant, or affected loop, shall be placed on shutdown condition in TBD hours.

4. During plant shutdown, the recirculation water temperature in at least one SGS loop shall not drop below TBD°F.

If the specification of item No. 4 above is not met, immediate corrective action shall be taken to bring the plant within this specification within 24 hours.

16.3.4.1.4 Basis

During plant operation, at least two of the three Steam Generator loops will be operational to assure an adequate heat removal capability should one of these loops suffer a fault which would interfere with its heat removal function.

To assure adequate operation, the water level in the steam drum will not be below TBD inches. If the water level drops below the low limit, there is a possibility that the steam drum may dry out. This event could result in loss of the Steam Generator heat removal capability and must be avoided. In order to assure that at least two independent flow paths are available for decay heat removal, the water level in at least two steam drums shall not be allowed to drop below the lower limit.

All power/safety relief valves included in the steam generator system must be operable to provide adequate relief during overpressurization. In the present design, there is no redundant capacity.

During plant shutdowns for periods longer than about two hours, the reactor decay heat is transferred to the atmosphere by the SGAHRS protected air cooled condenser. The condensate returns to the steam drum and is recirculated through the evaporator. The process must be controlled to prevent cooling IHTS sodium below the plugging temperature.

16.3.4.2 Steam Generator Auxiliary Heat Removal System

16.3.4.2.1 Applicability

Applies to the Steam Generator Auxiliary Heat Removal System (SGAHRS) and related systems.

16.3.4.2.2 Objective

To assure adequate post shutdown core cooling capability if the normal nuclear steam supply heat sink is unavailable.

16.3.4.2.3 Specification

1. During normal reactor power operation (three loops) the SGAHRS short-term and two SGAHRS long-term removal subsystems shall be operable. The operable short-term heat removal subsystems must correspond to operable long-term subsystems.

2. During two loop reactor power operation the SGAHRS short-term and two SGAHRS long-term heat removal subsystems shall be operable. The operable long-term heat removal subsystem must correspond to two operating heat transport loops.

3. During two loop and three loop reactor power operation, the turbine driven auxiliary feedwater pump may be taken out of service for maintenance provided the emergency diesels are activated for the two electrically driven auxiliary feedwater pumps. The two feedwater pumps shall be tested for operability on both the normal and emergency power supplies.

4. During two loop and three loop reactor power operation, one of the electrically driven auxiliary feedwater pumps may be taken out of service for maintenance provided the turbine driven auxiliary feedwater pump has been tested and shown to be operable and the other electrically driven auxiliary feedwater pump has been tested and shown to be operable with normal and emergency power supplies. The emergency diesel corresponding to the operable pump must be operable.

5. During normal and two loop reactor power operation and during standby operation, the protected water supply system shall be operational, and the volume of water in the protected water storage tank shall be greater than TBD gallons.

   If the above specifications are not met, an orderly transition to hot shutdown conditions shall be initiated.

6. During periods of hot shutdown or refueling shutdown, PHTS and IHTS temperatures shall not subsequently rise above the steady state design temperature limit (See Sections 16.3.2 and 16.3.3, respectively). This is assured by the following considerations:

   a. At all times during plant shutdown periods, a minimum of 2 decay heat removal loops individually capable of removing the full decay heat load shall be available. The two loops may consist of two main HTS loops and their associated heat dumps or 1 HTS loop and/or DHRS.

   b. At least one auxiliary feedwater pump in the short term heat removal subsystem must be operable. The short term heat removal subsystem must be available to supply makeup water to the operable long term heat removal subsystem as required.

If the specification of Item Number 6 above is not met, the components subjected to high temperatures shall be examined and evaluated for suitability for return to power operation.

16.3.4.2.4 Bases

During plant operation at least two of the three SGAHRS long-term cooling circuits shall be operable to assure heat removal capability if one of these loops suffers a fault which would interfere with its heat removal function. The remaining loop(s) will be capable of removing the required heat load without exceeding plant safety limits.

The Protected Air Cooled Condensers (PACC) and piping connecting them to the steam drum constitute the long term heat removal circuit. Each circuit is uniquely associated with a steam drum. To assure redundant long-term cooling, the two circuits associated with the operating heat transport loops must be available during two loop operation. Then, if one of the heat transfer loops loses its ability to remove heat, the plant can be shutdown and decay heat is removed by the remaining heat transfer loop with redundancy being provided by the Direct Heat Removal Service (DHRS)

The Auxiliary Feedwater System which supplies feedwater from the Protected Water Supply (PWS) to the steam drum, in combination with the power relief valves in the Steam Generator System, constitutes the short-term heat removal circuit. The Auxiliary Feedwater System consists of three pumps and associated piping and valves which permit delivering water to all or any combination of steam drums. During plant operation, this system shall be capable of supplying feedwater to at least two steam drums to assure adequate removal of decay heat and to avoid steam drum dry out if the normal feedwater supply is interrupted.

During plant operation on two loops, only two short-term cooling circuits are available since the third heat transport loop is not operating. To assure redundant short term cooling, the two short-term circuits associated with the operating heat transport loops must be available.

During periods of reactor shutdown, three independent decay heat removal means must be available to provide redundancy in the unlikely event one of them becomes inoperable. Therefore, when the Direct Heat Removal Service (DHRS) is not available, three main heat transport loops must be available for
26 long term decay heat removal. If DHRS is available then only two main heat transport loops must be available. When decay heat is sufficiently low that the long-term heat removal circuit(s) and/or DHRS can maintain plant temperatures steady or decreasing, then it is not necessary to maintain active short-term circuit(s). Short-term circuit(s) must be available to provide makeup
26 water if needed by active long-term circuit(s). This is done to assure the availability of cooling water to keep the long-term circuit(s) operating as long as required.

During plant operation the protected source of water shall be available to the auxiliary feedwater pumps to guard against a loss of normal feedwater accident. The protected water volume shall be greater than (TBD) in order to provide an adequate supply for the short-term heat removal circuit venting and to assure that the steam drums have adequate water inventory for long term heat removal.

16.3.4.3 Sodium-Water Reaction Pressure Relief Subsystem (SWRPRS)

16.3.4.3.1 Applicability

Applies to the Sodium Water Reaction Pressure Relief Subsystem of the Steam Generator System.

16.3.4.3.2 Objective

To assure overpressure protection for the Intermediate Heat Transport System, Intermediate Heat Exchanger, and sodium side of the Steam Generator System, and to limit the consequences of a sodium-water reaction by removing the sodium reaction products, water, and steam from the effected components.

16.3.4.3.3 Specification

Any time there is water/steam on the tube side of the evaporator/superheater and sodium on the shell side of the evaporator/superheater, the Sodium-Water Reaction Pressure Relief System (SWRPRS) shall be operational.

If the above specification is not met, or cannot be complied with by corrective action within TBD hours, an orderly shutdown of the plant shall be initiated, and the effected loop shall be placed in a condition such as to prevent a sodium-water reaction. The reactor may then be operated with two HTS loops in service at a power level commensurate with heat transport capabilities.

16.3.4.3.4 Basis

During all modes of plant operation, the sodium side pressure relief systems must be fully operable for the operating heat transport loops. These systems are required to limit the consequences of a water-to-sodium leak in the Steam Generator System and to protect the Intermediate Heat Exchanger, Intermediate Heat Transport Systems, and steam generator components from the resulting sodium water reactions.

16.3.5 Auxiliary Liquid Metal System

16.3.5.1 Applicability

This specification applies to radioactivity limits for operation of the Auxiliary Liquid Metal System.

16.3.5.2 Objective

To define radioactivity limits for normal operation and maintenance of the Ex-vessel Storage (EVS) Cooling Subsystem and the ex-containment primary sodium storage system.

16.3.5.3 Specification

1. The activity of the EVS sodium shall not exceed the following limits:

   A. Inerted cell

      1. Total plutonium activity - TBD curies

      2. Total gross β-γ activity - TBD curies

   B. Deinerted cell

      1. Total plutonium activity - TBD curies

      2. Total gross β-γ activity - TBD curies

2. The activity of the sodium transferred to ex-containment storage shall not exceed the following limits:

A. Deinerted cell

1. Total plutonium activity - TBD curies

2. Total gross β-γ activity - TBD curies

16.3.5.4 Bases

The bases for these specifications are the accidents analyzed and reported in Chapter 15.6. In all cases, the limits assure compliance with 10CFR100.

16.3.6 Inert Gas System Cover Gas Purification System

16.3.6.1 Purity of Gas

16.3.6.1.1 Applicability

Applies to the concentration of gaseous impurities in the recycle argon (argon after processing by the RAPS).

16.3.6.1.2 Objective

To define the maximum allowable concentration of impurities in the argon to be supplied to the reactor and PHTS pump cover gas spaces.

16.3.6.1.3 Specification

(To be specified in the FSAR.)

16.3.6.2 Limiting Activity in the Radioactive Argon Processing System (RAPS)

16.3.6.2.1 Applicability

Applies to the inventory of the Radioactive Argon Processing System.

16.3.6.2.2 Objective

To define the limiting activity in RAPS.

16.3.6.2.3 Specification

59 |49| 1. The radioactive inventory in the RAPS cryostill shall not exceed TBD Ci.

59 | 2. If the above limit is exceeded an orderly shutdown of the plant shall be initiated within TBD hours after this has been determined.

16.3.6.2.4 Basis

59 |49| The specification is designed to limit the site boundary dose to conform to 10CFR100, in the event of a RAPS cryostill rupture as described in Chapter 15.7.2.

16.3.6.3 Cell Atmosphere-Oxygen Control

16.3.6.3.1 Applicability

This specification applies to the primary heat transfer cells and the EVS cooling system cells during normal operation.

16.3.6.3.2 Objective

To assure that accident design limits in inerted cells are not exceeded in the event of a large sodium spill because of a high oxygen concentration in the cell atmosphere.

16.3.6.3.3 Specifications

1. If the oxygen level in the inerted cell atmosphere is greater than 2% or less than 0.5%, corrective action shall be implemented to bring the level to within the specifications.

2. If, after TBD hours of corrective action, the oxygen level in the inerted cells is not within specification, an orderly isolation, drain,
59 or cooldown of alkali metal inventory in the cell shall be initiated.

16.3.6.3.4 Basis

The upper limit of 2% oxygen is based on the allowable level developed in the accidents analyzed in Chapter 15.6.1.1 and 15.6.1.5. The lower level of 0.5% is established to prevent nitriding.

16.3.7 Auxiliary Cooling System

16.3.7.1 Fuel Storage Heat Removal

16.3.7.1.1 Applicability

Applies to the limiting conditions for operation of the spent fuel storage facilities.

16.3.7.1.2 Objective

To ensure that no incident could occur during spent fuel storage that would adversely affect the public health and safety.

16.3.7.1.3 Specifications

Items a and c through f shall be continuously satisfied.

a. Two independent power supplies shall be available for spent fuel storage facilities and their cooling systems when spent fuel decay heat removal is required.

b. All three (3) EVST heat removal systems shall be operable (capable of removing up to TBD power) at the beginning of a refueling which will increase the decay power in the EVST to TBD. If one of the heat removal systems becomes inoperative during refueling, operations will be suspended if the power level exceeds TBD or the inoperative cooling system is not restored in TBD hours.

c. The EVST shall have at least two heat removal systems operable. Each of the two systems shall be capable of handling the maximum design heat load of 1800 kW. Prior to scheduled inspection or routine maintenance of any heat removal system the two remaining heat removal systems shall be in operable condition. If during the inspection or maintenance period one of the two remaining heat removal systems fails, the heat removal system undergoing scheduled inspection or routine maintenance shall be returned to service within the time which would be required for the EVST sodium to reach 775° with no cooling.

d. The two forced convection, normal, independent EVST sodium cooling loops shall not be operated simultaneously, except when switching trains. When the loop is in operation, the other loop shall be kept on standby with its outlet valve closed.

e. Except when required for EVST cooling, the isolation valve in the lower EVST outlet line of loop 2 shall be locked closed.

f. Before an inerted cell containing one of the EVST sodium cooling loops is to be exposed to the RSB atmosphere, the enclosed sodium cooling loop shall be isolated from the EVST and, if the sodium radioactivity concentration exceeds TBD µCi/CC, the loop will be drained.

g. Coolant levels in the EVST shall not be less than TBD.

If any of the above limiting conditions are not met, corrective action must be initiated to resolve the deficiency.

16.3.7.1.4 <u>Bases</u>

The first five specifications in Section 16.3.7.1.3 ensure equipment redundancy for cooling spent fuel so that a single failure or an initiating event following a single failure cannot cause overheating of fuel.

Specification f. is required on the basis that a potential sodium spill in the EVST sodium cooling loop cell might result in radioactivity release with a site boundary dose exceeding one tenth of the 10CFR100 limits.

Specification g. ensures the the EVST storage vessel sodium level is maintained above the inlet lines to each of the three cooling systems even in the event of a storage vessel rupture.

16.3.7.2 Fuel Handling Heat Removal

16.3.7.2.1 Applicability

Applies to the limiting conditions for operation of the ex-vessel transfer machine (EVTM).

16.3.7.2.2 Objective

To ensure that no incident could occur during spent fuel handling operations that would adversely affect the public health and safety.

16.3.7.2.3 Specifications

The following conditions shall be continuously satisfied while the
20 EVTM is transferring spent fuel or blanket assemblies.

a. Two independent power supplies shall be available for the EVTM.

b. The EVTM shall have two heat removal systems operable, each capable of removing 20 Kw of heat, before being used to handle spent fuel.

c. The EVTM shall not be used to handle fuel assemblies from the
59 reactor until their calculated decay heat is less than 20 kw.

If any of the above limiting conditions are not met, corrective action must be initiated to resolve the deficiency. No spent fuel or blanket assemblies shall be handled by the EVTM before the above conditions are restored. However, any fuel or blanket assembly in the EVTM may be trans-
20 ferred to the EVST or the reactor fuel transfer position, whichever is closer.

16.3.7.2.4 Bases

The first two specifications in Section 16.3.7.2.3 establish equipment redundancy for cooling spent fuel so that a single failure cannot cause overheating of fuel. The last specification ensures that no fuel assembly is handled with a decay heat exceeding the cooling capacity of the EVTM.

26 16.3.7.3 Direct Heat Removal Service (DHRS)

16.3.7.3.1 Applicability

Applies to the Auxiliary Liquid Metal Subsystem as related to the Direct Heat Removal Service function.
26

16.3.7.3.2 Objective

To provide adequate long term post shutdown core cooling capability for the remote case in which the normal nuclear steam supply heat sink and the steam generator auxiliary heat removal systems cannot provide adequate heat removal with sufficient redundancy.

16.3.7.3.3 Specification

1. During two loop reactor power operation, with the out of service
41 HTS loop unavailable for decay heat removal, the DHRS shall be 39
capable of being brought into service within TBD hours after shutdown. If this specification cannot be met, an orderly shutdown of the core shall be initiated.

2. **During periods of hot shutdown or refueling shutdown, reactor vessel overflow temperatures shall not exceed the steady state design temperature limit (Section 9.3.2.2.1). This is assured by the operating considerations defined in Section
16.3.4.2.3. If this specification is not met, DHRS compo- 39
41 nents subjected to the high temperatures shall be examined and evaluated for suitability for return to power operation.**

16.3.7.3.4 Basis

During plant operation on two loops, the DHRS shall be capable of 39
41 being brought into service within TBD hours to assure continuous removal of
core decay heat in the event that one of the two operating heat transport loops are lost to decay heat removal. This is intended to assure that at least two independent flow paths are available following the loss of a flow path. This condition exists if the out-of-service loop cannot be brought into service in time to satisfy the safety function.

The DHRS utilizes the normally operating components of the Auxiliary 39
41 Liquid Metal Systems which includes two air blast heat exchangers associated
with the EVST cooling subsystem. This circuit is sized to provide long term
41 cooling capability for the DHRS. 39

If the plant is shutdown with two main HTS loops available for decay
41 heat removal, the DHRS must be operable to guarantee an alternate heat removal 39
path if one of these loops is lost.

If the plant is shutdown with one main HTS loop available for decay
41 heat removal, DHRS must be operated continuously to guarantee an alternate 39
heat removal path if this single HTS loop should become inoperable.

Amend. 41
Oct. 1977

16.3.7.4 Compartment and Structural Cooling Systems

16.3.7.4.1 Applicability

This specification applies to the temperature limits of the gas in inerted cells cooled by the Recirculating Gas Cooling System (RGCS) in relation to cell liner and concrete temperatures.

16.3.7.4.2 Objective

To define the gas temperature limits for cells cooled by RGCS.

16.3.7.4.3 Specification

1. The cell gas temperature shall be maintained below 250°F.

2. If the cell gas temperature exceeds this temperature, the reactor shall be shutdown and the primary coolant temperature will be reduced to TBD°F.

16.3.7.4.4 Basis

The maximum cell temperature of 250°F is set 20°F below the value of 270°F from ASME "Proposed Standard Code for Concrete Reactor Vessels and Containments" for abnormal loads.

16.3.7.5 Ultimate Heat Sink Train

16.3.7.5.1 Applicability

33 15 Applies to the operation of the Emergency Cooling Tower Structure, the Emergency Plant Service Water System, and the Emergency Chilled Water System.

16.3.7.5.2 Objective

To define the conditions necessary to assure immediate availability of the Emergency Cooling Tower structure, the Emergency Plant Service Water System, and the Emergency Chilled Water System. 15

16.3.7.5.3 Specification

The reactor shall not be made critical unless the following conditions are satisfied.

A. Emergency Cooling Tower

1. The Emergency Cooling Tower Basin shall contain the required water quantity for 30 days uninterrupted operation of the Emergency Plant Service Water System. 33

43 2. The two Emergency Cooling Tower fans are operable.

3. The water temperature in the Emergency Cooling Tower Basin is not higher than TBD.

B. Emergency Plant Service Water System (EPSW)

43 1. The EPSW System pumps are operable.

2. The manual isolation valves provided, at all equipment served by the EPSW system, are in "locked open" position.

33 C. Emergency Chilled Water System (ECHW)

1. The ECHW pumps are operable.

2. The ECHW system water chillers are operable.

3. The manual isolation valves at the ECHW system chillers are in "locked open" position.

4. The manual isolation valves at the Air Conditioning System's cooling coils and Unit Coolers cooling coils are in "locked open" position.

5. The automatic isolation valves separating the ECHW system from the Normal Chilled Water System are tested and proved remotely operable. Normally, the valves shall be in the open position.

15

D. Conditions During Maintenance

Maintenance is allowed, during power operation of the plant, on any component(s) in the Emergency Plant Service Water and Safety Related Chilled Water Systems which will not remove more than one interrelated train of all systems from service. Components shall not be removed from service so that the affected system train is inoperable for more than a specified period. If the system is not restored to meet the requirements of 16.3.7.5.3 within a specified time, the plant shall be placed in a standby condition. If the requirements of Specification 16.3.7.5.3 are not met within an additional time period, the plant shall be placed in cold shutdown condition.

The time period indicated above, will be supplied in the FSAR.

The above conditions are subject to the following limitations:

33

(a) A certain percentage of the Unit Coolers connected to the Emergency Chilled Water System can be out-of-service for an extended period of time. The permissible number of unit coolers and the extended time period will be presented in the FSAR.

(b) Prior to initiating maintenance on any of the components, the duplicate (redundant) component shall be tested to assure operability.

16.3.7.5.4 Bases

The requirements of Specification 16.3.7.5.3 assure that before the reactor can be made critical, adequate Safety Related features are operable. Redundant equipment and piping are specified for the entire interrelated Ultimate Heat Sink Train. However, only one train is required to supply ECHW to the safety related equipment in the event of an accident.

The EPSW system is designed to remove the heat from the ECHW system chiller-condensers, and from the standby Diesel Generator coolant heat exchanger.

The ECHW system is designed to remove heat from the control and battery rooms, the Diesel Generator Building, emergency chiller and switchgear rooms, Steam Generator Building, Reactor Service Building, and the Reactor Containment Building.

16.3.7.3.2 Objective

To provide adequate long term post shutdown core cooling capability for the remote case in which the normal nuclear steam supply heat sink and the steam generator auxiliary heat removal systems cannot provide adequate heat removal with sufficient redundancy.

16.3.7.3.3 Specification

1. During two loop reactor power operation, with the out of service
41 HTS loop unavailable for decay heat removal, the DHRS shall be 39
capable of being brought into service within TBD hours after shutdown. If this specification cannot be met, an orderly shutdown of the core shall be initiated.

2. **During periods of hot shutdown or refueling shutdown, reactor vessel overflow temperatures shall not exceed the steady state design temperature limit (Section 9.3.2.2.1).** This is assured by the operating considerations defined in Section
41 16.3.4.2.3. If this specification is not met, DHRS compo- 39
nents subjected to the high temperatures shall be examined **and evaluated for suitability for return to power operation.**

16.3.7.3.4 Basis

During plant operation on two loops, the DHRS shall be capable of 39
41 being brought into service within TBD hours to assure continuous removal of core decay heat in the event that one of the two operating heat transport loops are lost to decay heat removal. This is intended to assure that at least two independent flow paths are available following the loss of a flow path. This condition exists if the out-of-service loop cannot be brought into service in time to satisfy the safety function.

41 The DHRS utilizes the normally operating components of the Auxiliary 39
Liquid Metal Systems which includes two air blast heat exchangers associated with the EVST cooling subsystem. This circuit is sized to provide long term
41 cooling capability for the DHRS. 39

If the plant is shutdown with two main HTS loops available for decay
41 heat removal, the DHRS must be operable to guarantee an alternate heat removal 39
path if one of these loops is lost.

If the plant is shutdown with one main HTS loop available for decay 39
41 heat removal, DHRS must be operated continuously to guarantee an alternate heat removal path if this single HTS loop should become inoperable.

16.3.7.4 Compartment and Structural Cooling Systems

16.3.7.4.1 Applicability

This specification applies to the temperature limits of the gas in inerted cells cooled by the Recirculating Gas Cooling System (RGCS) in relation to cell liner and concrete temperatures.

16.3.7.4.2 Objective

To define the gas temperature limits for cells cooled by RGCS.

16.3.7.4.3 Specification

1. The cell gas temperature shall be maintained below 250°F.

2. If the cell gas temperature exceeds this temperature, the reactor shall be shutdown and the primary coolant temperature will be reduced to TBD°F.

16.3.7.4.4 Basis

The maximum cell temperature of 250°F is set 20°F below the value of 270°F from ASME "Proposed Standard Code for Concrete Reactor Vessels and Containments" for abnormal loads.

16.3.7.5 Ultimate Heat Sink Train

16.3.7.5.1 Applicability

33 15 Applies to the operation of the Emergency Plant Service Water System, and the Emergency Chilled Water System.

16.3.7.5.2 Objective

To define the conditions necessary to assure immediate availability of the Emergency Cooling Tower, the Emergency Plant Service Water System, and the
15 Emergency Chilled Water System.

16.3.7.5.3 Specification

The reactor shall not be made critical unless the following conditions are satisfied.

33

A. Emergency Plant Service Water System (EPSW)

1. The EPSW System pumps are operable.

2. All EPSW air blast heat exchangers required for safe shutdown are available.

3. The manual isolation valves provided, at all equipment served by the EPSW system, are in "locked open" position.

33 B. Emergency Chilled Water System (ECHW)

1. The ECHW pumps are operable.

2. The ECHW system water chillers are operable.

3. The manual isolation valves at the ECHW system chillers are in "locked open" position.

4. The manual isolation valves at the Air Conditioning System's cooling coils and Unit Coolers cooling coils are in "locked open" position.

5. The automatic isolation valves separating the ECHW system from the Normal Chilled Water System are tested and proved remotely operable. Normally, the valves shall be in the open position.

15

D. Conditions During Maintenance

Maintenance is allowed, during power operation of the plant, on any
3 component(s) in the Emergency Plant Service Water and Safety Related Chilled Water Systems which will not remove more than one interrelated train of all systems from service. Components shall not be removed from service so that the affected system train is inoperable for more than a specified period. If the system is not restored to meet the requirements of 16.3.7.5.3 within a specified time, the plant shall be placed in a standby condition. If the requirements of Specification 16.3.7.5.3 are not met within an additional time period, the plant shall be placed in cold shutdown condition.

The time period indicated above, will be supplied in the FSAR.

The above conditions are subject to the following limitations:

33

(a) A certain percentage of the Unit Coolers connected to the Emergency Chilled Water System can be out-of-service for an extended period of time. The permissible number of unit coolers and the extended time period will be presented in the FSAR.

(b) Prior to initiating maintenance on any of the components, the duplicate (redundant) component shall be tested to assure operability.

16.3.7.5.4 <u>Bases</u>

The requirements of Specification 16.3.7.5.3 assure that before the reactor can be made critical, adequate Safety Related features are operable. Redundant equipment and piping are specified for the entire interrelated Ultimate Heat Sink Train. However, only one train is required to supply ECHW to the safety related equipment in the event of an accident.

3 The EPSW system is designed to remove the heat from the ECHW system chiller-condensers, and from the standby Diesel Generator coolant heat exchanger.

The ECHW system is designed to remove heat from the control and battery rooms, the Diesel Generator Building, emergency chiller and switchgear rooms, Steam Generator Building, Reactor Service Building, and the Reactor Containment Building.

15 The Chilled Water System is designed to remove heat from the EVST E. M. pumps and components, Fuel Handling Cell and reactor makeup pumps.

When the plant is operating, maintenance is allowed per Sections B, C, D, and E which assure operability of the duplicate components. The specified maintenance down time is the maximum time period that can be allowed, considering the absence of redundancy in one of the Ultimate Heat Sink Trains. An allowable maintenance period may be utilized if the operability of equipment redundant to that removed from service is demonstrated immediately prior to removal. Operability of the specified components shall be based on the result of testing.

16.3.8 Containment Integrity

16.3.8.1 Applicability

16.3.8.2 Objective

To define the status of the containment required to ensure no undue risk to the health and safety of the public.

16.3.8.3 Specification

Containment integrity (as defined in 16.1.8) shall not be violated unless the reactor is sub-critical by at least 4% $\Delta k/k$, and the inlet coolant temperature is less than TBD°F.

16.3.8.4 Basis

The circumstances under which a violation of containment is permissible are chosen such that the remaining provisions available to prevent a release of radioactivity can be relied upon to perform their function. Thus, by maintaining the reactor in a shutdown condition, the control system will provide sufficient assurance that excessive radioactivity releases can be prevented. The value of 4% $\Delta k/k$ is consistent with the discussion in 16.3.10.4 ($11.0 subcritical).

16.3.9 Auxiliary Electrical System

16.3.9.1 Applicability

Applies to the availability of electric power for operation of plant auxiliaries and to the availability of electric power required for safe shutdown of the plant.

16.3.9.2 Objective

To define those conditions of electric power availability necessary to provide for safe power operation and to provide for the continuing availability of Safety Related Class 1E equipment.

16.3.9.3 Specification

1. The reactor shall not be made critical unless all of the following conditions are satisified:

   a. Two independent offsite power sources shall be available from the reserve station service switchyard and capable of providing power supplies to the Class 1E buses.

   b. Both diesel generators are operable, with sufficient on-site fuel supply for seven days of continuous full load operation of both diesel generators.

   c. All Class 1E 4.16-KV buses and 480-V buses are energized. 37

   d. All 125 Volt and 250 Volt DC Station Batteries are operable and at full charge, each with an operable charger.

   e. All 120 V vital AC instrumentation and control power supply buses are energized.

   f. Power to the Reactor Shutdown System, the Containment Isolation System, SGAHRS, and diesel loading system are energized.

2. During reactor power operation or the return to power from standby conditions, the availability of the auxiliary electric system shall be as specified above except as amended by one of the following:

   a. Operation of the plant may be permitted for up to seven days with only one source of offsite power available from the reserve-station switchyard, provided that both diesel generators are demonstrated to be operable daily.

   b. Operation of the plant may be permitted for up to seven days with one diesel generator out of service provided that:

      1. Two independent sources of offsite power are available from the reserve station-service switchyard.

      2. The redundant diesel generator is determined to be operable daily.

      3. All Class 1E equipment of the redundant load group is operable.

   c. Operation of the plant may be permitted for up to seven days
37 with one Class 1E 4.16-KV bus or one Class 1E 480-V bus out of service provided that the redundant diesel generator is determined to be operable daily and redundant equipment is available to assure safe shutdown of the reactor under postulated accident conditions.

d. Operation of the plant may be permitted for up to seven days with one Station battery out of service provided the battery chargers and the other batteries remain operable with the battery charger, which is associated with the failed battery, carrying the DC load in its subsystem. However, if the loss also results in the loss of DC power for controlling the
37| Class 1E 4.16-KV and 480-V buses or for diesel generator field, the requirement of (c) above shall apply.

e. In the event two diesel generators are inoperable, a plant shutdown shall be initiated within two hours.

16.3.9.4 Basis

The electrical system is designed so that no single failure can impair the ability of the system to supply sufficient power to the Engineered Safety Features equipment required for plant safety under all conditions of operation or postulated accidents. The Engineered Safety Feature equipment is divided into redundant load groups, either of which is capable of safely shutting down the plant.

The offsite power system provides a reliable source of AC power to the plant. The system consists of the preferred AC power supply and the reserve AC power supply. The preferred AC power supply provides two connections to the TVA 161-kV grid. The reserve AC power supply provides two physically separate connections to the TVA 161-kV grid. All four of these grid sources are continuously energized and any one of them can supply the plant auxiliary distribution system to facilitate and maintain a safe plant shutdown.

37| Power for each of the Class 1E load groups is distributed by a 4.16-KV switchgear, 480-V load centers and 480-V motor control centers.

Two diesel generators are provided as standby power supplies for the
37| two 4.16-KV Class 1E buses. They are automatically started by a bus undervoltage condition as described in Section 8.3. Each generator is capable of supplying all the loads of one Class 1E load group. Both diesel generators have sufficient onsite fuel supply for seven days continuous full load operation. Sufficient maintenance and test procedures ensure that power for the Class 1E loads is always available during and after any design basis event.

Control power for each of the redundant Class 1E load groups and associated standby power supplies is fed from separate Class 1E DC power supplies.

One redundant Class 1E, DC power supply or standby AC power supply may be taken out of service for TBD hours to permit maintenance, repair and testing.

16.3.10 Refueling

16.3.10.1 Applicability

Applies to the limiting conditions for operation of the Reactor Refueling System (RRS) equipment and facilities, and to refueling operations.

16.3.10.2 Objective

To ensure that during refueling operations, core reactivity is within controlled limits and to ensure that the release of radioactivity from the containment or RSB in the event of a fuel handling accident is within the limits of 10CFR20 and 10CFR100.

16.3.10.3 Specification

16.3.10.3.1 The following conditions shall be continually satisfied while the Reactor Refueling System (RRS) equipment and facilities are operating.

44 a. The EVST and FHC gas activities shall be less than TBD μCi/cc, and TBD μCi/cc, respectively.

59 b. The railroad doors into the hardened portion of the RSB shall be closed and shall remain closed during the following conditions:

(1) When the EVTM is transferring irradiated core fuel assemblies;

44 (2) When irradiated core fuel assemblies are handled in the FHC or are being inserted into the spent fuel shipping cask.

If the above limiting conditions are not met, corrective action shall be taken to resolve the deficiency. No EVTM mating operation shall be initiated to the EVST or FHC if the respective gas activity is higher than specified.

16.3.10.3.2 The following conditions shall be met before initiating refueling operations involving the reactor.

a. The reactor shall be maintained in the Refueling Shutdown Condition as defined in Section 16.1.1.

b. The primary pump main circuit breakers shall be racked out and tagged.

c. During any movement of fuel within the core, a licensed operator shall be present in the Refueling Communication Center or the IVTM mezzanine.

59 d. All refueling system equipment required for the refueling operations shall be checked out and verified to be operational.

59 e. The primary and secondary control rod drive mechanisms shall be disconnected from the control assemblies and the UIS raised and
51 pinned. Prior to movement of the large rotating plug, a verification shall be made that all control rods are disconnected from their drive line assemblies.

59 f. The reactor cover gas activity shall be less than 2.2 μCi/cc.

44 g. The IVTM limit switch which precludes premature release of fuel
59 and blanket assemblies shall be set less than 1.38 inches above the fully seated position as indicated in Figure 9.1-16B. 1 16

If any of the above specified limiting conditions are not met, the refueling shall not be initiated.

16.3.10.3.3 The following conditions shall be met during refueling operations involving the reactor.

59 a. Direct communications among personnel in the plant control room at the IVTM control console, and in the refueling communications center shall exist whenever changes in core geometry or fuel transfers are taking place.

b. All three source range flux monitor (SRFM) channels shall be operating with any fuel assemblies in the core. If any one of the channels fails, operations in progress to transfer fuel into or out of the reactor core shall be stopped or reversed to place the reactor in a safe hold point configuration until the defective channel is restored to operation.

The Source Range Flux Monitoring System (SRFM) trip
51 points will be set at signal levels equivalent to a subcriticality of TBD for the first core and TBD for the equilibrium core. If the trip points are exceeded, the refueling operation must be stopped immediately and a determination made as to the cause of the reactivity anomaly.

c. During refueling operations, not more than two vacant positions in the core may exist at any one time. These vacant positions may not be adjacent to each other.

If any of the specified limiting conditions for refueling are not met, refueling shall cease until the specified limits are met, and no operations will be initiated which may increase the reactivity of the core beyond the reactivity resulting from normal temperature fluctuations within the refueling temperature dead band.

16.3.10.3.4 Following refueling operations involving the reactor, the following conditions shall be met prior to reactor startup.

a. The reactor rotating plugs shall be secured and their drive power sources physically disconnected.

b. The refueling hatch between the RSB and the RCB shall be closed and leak tested.

16.3.10.3.5 The following conditions shall be met before initiating fuel handling or shipping operations in the FHC.

Both FHC cooling grapple blowers, both argon cooling system trains, the dynamic seals, and the FHC radiation monitors, shall be checked and verified to be operational.

If the above specified limiting condition is not met, FHC fuel handling or shipping operations shall not be initiated.

16.3.10.3.6 The following leak rate tests shall be performed at periodic intervals.

a. The EVTM shall be leak rate tested at 11 psig. The leak rate
59 shall not exceed 1 vol. % per day.

b. The FHC shall be leak rate tested at -3 inches water gauge. The
59 leak rate shall not exceed 0.14 vol. % per day.

If the above limiting conditions are not met, correction action shall be taken to resolve the deficiency. No EVTM or FHC operations involving irradiated core assemblies shall be initiated if the respective leak rates are higher
44 than specified.

16.3.10.4 <u>Bases</u>

The respective limits in Section 16.3.10.3.1 are established on the basis that if either amount of activity was all released instantaneously into the RSB operating area, the radiation dose at the site boundary would be less than the limits of 10CFR20 (Annual).

Immediately prior to refueling, Section 16.3.10.3.2 lists the conditions which must be satisfied. Item a is based on permissible core
shutdown levels. Item b is written to prevent the operation of
the primary pumps during refueling and Item c is intended to assure that proper
supervision will exist during movement of fuel within the core. Items d and e
59 are written to prevent unexpected movement of core components during refueling
which could affect core reactivity. Item f is intended to control the release
44 of radioactivity to the atmosphere. The level specified in Item f is based on
the premise that if this amount of activity was all released instantaneously
into the RCB operating area, the radiation dose at the site boundary would not
exceed the limits of 10CFR20 (Annual) and the airborne radiation dose in the
RCB would be below the quarterly 10CFR20 limits for restricted areas. Item g 1
44 is intended to prevent dropping of a core assembly or insertion of a core
9 assembly into an incorrect position.

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59 The specifications of Section 16.3.10.3.3 during refueling establish control of the operation. During any subcritical operation other than the intentional approach to critical, the SRFM must provide a warning to the operator and thereby assure that the reactor does not approach criticality any closer than that level from which criticality could be attained by the worst single refueling error with adequate margin for the associated uncertainties. The minimum shutdown reactivity requirement during refueling is based on this criterion. An alarm will sound in the control room if the
51 minimum shutdown requirement, described above, is violated.

Shuffling of blanket assemblies cannot be done without temporarily leaving open two core positions. If two adjacent core assemblies are removed, the resulting misalignment could exceed the design value, so that a new core assembly or an assembly to be reinserted could either not be inserted or be inserted in the wrong position. Item c of Section 16.3.10.3.3 is written to prevent this event. Note, however, that shuffling is not part of the current fuel management scheme, but is only a capability provided for any future fuel
59 management scheme.

59 The specifications in Section 16.3.10.3.4 are written to assure that modifications made to accommodate the refueling are corrected before reactor startup.

The specification of Section 16.3.10.3.5 is mainly intended to ensure spent fuel cooling capability of the FHC to prevent potential fission gas activity release resulting from overheating of fuel pins. In addition, proper performance of inflatable and dynamic seals will be checked as a further backup of 16.3.10.3.6 b for maintaining a low leakage cell. Operational checkout of FHC radiation monitors is required to ensure that the limits of 16.3.10.1.a will not be exceeded.

The specifications of Section 16.3.10.3.6 are intended to control the release of radioactivity to the atmosphere as a consequence of the respective design basis accidents.

The maximum leakages specified in Section 16.3.10.3.6 Items a and b are determined by the activities of the highest power fuel assemblies handled by the EVTM and in the FHC which, if released to the RSB operating area and subsequently to the site boundary, at the specified leak rate, would be less
44 than the limits of 10CFR20 (Annual).

16.3.11 Effluent Release

16.3.11.1 Liquid Waste

16.3.11.1.1 Applicability

Applies to the liquid radioactive effluents from the radioactive waste system to the environment.

16.3.11.1.2 Objective

To assure that liquid radioactive material released to the environment is kept as low as practicable and, in any event, is within the limits of 10CFR20.

16.3.11.1.3 Specification

1. If the experienced release of radioactive materials in the liquid wastes, within a calendar quarter period, is such that these quantities, if continued for a year, would exceed twice the design objectives, the following actions will be taken:

   a) An investigation shall be made to identify the causes for such releases.

   b) A program shall be defined and initiated to reduce such releases to within the design values.

2. The release rate of radioactive materials in liquid waste from the plant shall be controlled, by in-line monitoring, such that the concentration in the cooling tower blowdown will not exceed the concentrations specified in 10CFR20.106.

3. All radioactivity liquid effluents released from the plant shall be reported in accordance with 16.6.7.B.

16.3.11.1.4 Basis

Liquid effluent release rate will be controlled in terms of the concentration in the discharge tunnel containing cooling tower blowdown. This basis assures that even if a person obtained all of his daily water from such a source, the resultant dose would not exceed that specified in 10CFR20. Since no such use of the discharge tunnel is made and considerable natural dilution occurs prior to any location where such water usage could occur, this assures that offsite doses from this source will be far less than the limits specified in 10CFR20.

In addition to the sampling and analysis of each batch prior to discharge, a radiation monitor on the radioactive waste discharge line and a sampler in the discharge tunnel give further assurance that annual average discharge concentration is kept within the specified limits.

16.3.11.2 Gaseous Waste

16.3.11.2.1 Applicability

Applies to the release of radioactive gaseous effluents from design release points.

16.3.11.2.2 Objective

To assure that the amount of radioactivity released as low as is reasonably achievable and will result in site boundary doses which are below 10CFR50,
59 Appendix I limits.

16.3.11.2.3 Specification

1. Radioactive gases released from design release points shall be continuously monitored and/or sampled such that the total release can be quantified.

59

2. The effluent monitor for CAPS shall be operable and capable of alarming when radioactivity is detected at a maximum pre-set
59 concentration of TBD μCi/cc.

3. The effluent monitor for undefined mixtures from the exhaust of radwaste area of the Reactor Service Building shall be operable and capable of alarming when radioactivity is detected at a level corresponding to (TBD) percent of the maximum permissible radionuclide concentrations given in 10CFR20.

4. The effluent monitor for undefined mixtures from the reactor service area (RSB) exhaust shall be operable and capable of alarming when radioactivity is detected at a level corresponding to (TBD) percent of the maximum permissible radionuclide concentrations given in 10CFR20 for unrestricted areas.

5. The effluent monitor for undefined mixtures from the Intermediate Bay exhuast shall be operable and capable of alarming when radioactivity is detected at a level corresponding to (TBD) percent of the maximum permissible radionuclide concentrations given in 10CFR20 for unrestricted areas.

6. The effluent monitor for undefined mixtures from the Turbine Generator Building exhaust shall be operable and capable of alarming when tritium activity is detected at a level corresponding to (TBD) percent of the maximum permissible concentration given in 10CFR20 for unrestricted areas.

7. In the event of an alarm due to high radioactivity in the effluent of a design discharge point, appropriate action will be taken as defined (to be supplied in FSAR).

8. If an effluent monitor is inoperable, appropriate action will be initiated and be in effect until the monitor is restored to operational status (action to be defined in FSAR).

9. If the quantities of radioactive material released during any semi-annual period are significantly above design objectives, the CRBRP shall:

   1) Make an investigation to identify the causes of such releases.
   2) Define and initiate a program of corrective action.

59

16.3.11.2.4 Basis

Dose rate estimates have been made for the CRBRP design release points for off-normal occurences. Based on these calculations, release of activity at the alarm limits will result in an off-site annual dose rate which will not exceed (TBD) mr/yr, well below 10CFR20 limits. Estimates of the activity inventory assume failed fuel conditions described in Section 11.3.

16.3.11.3 HVAC and Radioactive Effluents

16.3.11.3.1 Applicability

Applies to the release of radioactive effluents through the HVAC exhausts.

16.3.11.3.2 Objective

To assure that radioactivity released to the environment is kept as low as practicable and, in any event, is within the limits of 10CFR20 guidelines.

To assure that the release of radioactivity to unrestricted areas meet the "as low as practicable" concept, the following design objective applies:

a) The release rate of radioactive isotopes, averaged over a yearly interval except for halogens and particulate radioisotopes with half-lifes greater than 8 days, discharged from the plant, should not exceed:

$$\sum_i \frac{Q_i}{(MPC)_i} \leq 800 \text{ m}^3/\text{sec}$$

where Q_i is the annual average release rate (Ci/sec) of radioisotope i and $(MPC)_i$ in Ci/cc is defined for isotope i in column 1, Table II of Appendix B to 10CFR20.

16.3.11.3.3 Specification

1) The instantaneous release rate of radioactive isotopes, discharged from the plant, shall not exceed:

$$\sum_i \frac{Q_i}{(MPC)_i} \leq 40{,}000 \text{ m}^3/\text{sec}$$

where Q_i and $(MPC)_i$ are as defined above.

2a) The gaseous and particulate activity of the potentially contaminated HVAC discharge paths shall be monitored and recorded along with the corresponding effluent flow rates.

2b) Radiation monitors as required in 16.3.11.3.3-2a above shall be operable and capable of detecting a composite radioactivity release rate less than the design objective rate.

2c) Whenever any of the radiation monitors are inoperable, grab samples shall be taken in the affected discharge path and analyzed.

3) When the annual projected release rate of radioactivity, averaged over a calendar quarter, exceeds the annual objective, corrective action shall be taken to reduce such release rates to below the objective rate and/or orderly shutdown of the reactor shall be initiated.

4) When the instantaneous release rate or radioactivity exceeds twice the design objective rate, the licensee shall identify the cause of such release rates, initiate action to reduce such release rates to below the objective rate.

16.3.11.3.4 Basis

The specifications provide reasonable assurance that the resulting annual exposure rate from noble gases at any location at the site boundary will not exceed 10 millirems per year. At the same time, these specifications permit the flexibility of operation, under unusual operating conditions, which may temporarily result in releases higher than the design levels but well below the concentration limits of 10CFR20.

The release rate stated in the objective sets the concentration of radioisotopes, except for halogens and particulate radioisotopes with half-lives greater than eight days, at less than 2% of 10CFR, Part 20 requirements at the site boundary (<10 mrem per year).

Specification (1) requires the licensee to limit the release of all radioisotopes such that concentrations at the site boundary are less than the levels specified in 10CFR20.

Specification (2) requires that suitable equipment to monitor radioactive releases are operating during any period these releases are taking place.

Specification (3) establishes an upper limit for the quarterly average release rate for noble gases equal to the annual design rate. The intent of this specification is to permit the licensee the flexibility of operation under unusual operating conditions which may result in short-term release higher than the annual objective rate.

Specification (4) requires the licensee to initiate action to reduce instantaneous release rates to the annual design level whenever the measured release rate exceeds twice the annual design rate. The intent of this specification is to require the licensee to control and report short-term releases that exceed the annual design rate.

16.3.12 Reactivity and Control Rod Limits

16.3.12.1 Shutdown Reactivity

16.3.12.1.1 Applicability

Applies to the minimum control rod reactivity worth of the primary and secondary control rod systems.

16.3.12.1.2 Objective

To ensure reactor shutdown from any operating power condition to zero power following reactor scram using either the primary or secondary control rod system.

16.3.12.1.3 Specification

1. Control rod bank insertion limits for the primary and secondary control rod systems are TBD.

2. If any of the above conditions are not met, an orderly shutdown of the plant shall be initiated.

16.3.12.1.4 Basis

The primary control rod bank limits assure sufficient worth at all times in the reactor cycle, assuming the failure of any single active component (i.e., a stuck rod), to shutdown the reactor from any operating condition to zero power and to maintain shutdown over the full range of design coolant temperatures. Allowance has been made for the maximum reactivity fault associated with any anticipated occurrence.

The secondary control rod limits assure sufficient worth at all times in the reactor cycle, assuming the failure of any single active component (i.e., a stuck rod), to shutdown the reactor from any operating condition to 600°F. Allowance has been made for the maximum reactivity fault associated with any anticipated occurrence.

The reactivity fault allowance is included in the requirements on both control systems in place of a specific subcritical shutdown margin. The maximum reactivity fault is postulated to occur upon the accidental uncontrolled withdrawal (not ejection or drop-out) of the highest worth control rod in the reactor from its fully inserted position. Thus, if the faulted rod withdrawal is initiated from a partially withdrawn position, or the fault does not exist, substantial subcritical shutdown margin exists.

16.3.12.2 <u>Rod Axial Misalignment Limitations</u>

16.3.12.2.1 <u>Applicability</u>

Applies to the limits on the deviation of an individual control rod in a bank from the average bank position.

16.3.12.2.2 <u>Objective</u>

To ensure that the minimum scram performance requirements are met and to prevent distortions in the core power distributions due to the axial misalignment of control rods in a bank, in the power range exceeding 5 percent of full power.

16.3.12.2.3 <u>Specification</u>

If an operable primary control rod is axially misaligned from its bank, as indicated by the Rod Position Indicators, by more than TBD inches, it will be realigned within TBD minutes. If the realignment cannot be accomplished within the specified time, the rod shall be declared as inoperable (16.3.12.4).

16.3.12.2.4 <u>Basis</u>

The rod axial misalignment specification is intended to preclude excessive distortions in the radial power distributions and to assure that the minimum scram performance requirements are met.

16.3.12.3 <u>Inoperable Rod Position Indicator</u>

16.3.12.3.1 <u>Applicability</u>

Applies to the rod position indicating systems.

16.3.12.3.2 <u>Objective</u>

To provide indication of rod position to the operator during plant operations.

16.3.12.3.3 <u>Specification</u>

During operation of the reactor, either the absolute or the relative rod position indication system for each rod that is manuevered during operation must be operational. Failure of both systems requires reactor shutdown (not scram). Restart can be undertaken only after the absolute Rod Position Indication system is restored to operational status.

16.3.12.3.4 <u>Basis</u>

Rod position indication is required to provide information on correct banking of the control rods. Correct banking assures that the appropriate scram reactivity characteristics are met. The rod position indication systems provide the basic input to this banking determination. Sustained operation with both relative and absolute position indication systems inoperable for any rod that is to be maneuvered is not permissible.

16.3.12.4 Inoperable Rod Limitations

16.3.12.4.1 Applicability

Applies to the control rods of the primary and secondary system.

16.3.12.4.2 Objective

To assure safe shutdown and control capability at all times for the reactor.

16.3.12.4.3 Specification

1. A rod is defined to be inoperable if, in the course of normal operations, the rod fails to respond normally to a design command.

2. If an operable rod, with the exception of the four secondary rods, is within TBD inches of fully inserted, the reactor may be operated at a power level not greater than TBD percent of full rated power for not more than TBD hours.

3. If the inoperable rod is located at a position other than within TBD inches of fully inserted, corrective action shall be taken to determine the cause of the malfunction and correct it. If after TBD hours, the inoperable rod has not been restored to an operating status, an orderly shutdown of the reactor shall be initiated.

16.3.12.4.4 Basis

Operation of the reactor with a rod within TBD inches of fully inserted does not compromise the shutdown capability or rate of reactivity insertion during a scram. However its affect on local and general power distribution requires a limit on the power level and time at power.

For a rod inoperable at some other position, the primary and secondary control systems each have the capability to safely shutdown the reactor with a single stuck rod in each system. However, this capability is provided to accomodate the unexpected event, and is not intended as an operating condition. Time is provided to repair an inoperable rod condition to avoid unnecessary plant shutdown. However, if the condition cannot be relieved promptly, the plant must be shutdown.

16.3.12.5 Rod Drop Time

16.3.12.5.1 Applicability

Applies to all control rods at all operating temperatures.

16.3.12.5.2 Objective

To assure prompt operation of all control rods.

16.3.12.5.3 Specification

For all operating temperatures and flow rates, the drop time of each control rod shall be less than TBD seconds from tripping of the Plant Protection System Logic to dashpot or damper entry.

16.3.12.5.4 Basis

The allowable control rod system insertion times from start of rod motion for all operating conditions are presented in Section 4.2.3.1.3 and are consistent with safe operation of the plant. The delay between tripping of Plant Protection System logic and start of rod motion is required to be less than 0.1 seconds, consistent with plant safety. The maximum time specified is the sum of the delay time and insertion time.

This requirement is preliminary and represents the maximum insertion time allowed to assure that the allowable damage severity limits are not exceeded for all design basis transients. Iterative transient evaluations such as in Chapter 15 of this PSAR have led to the specified minimum insertion rates. Anticipated transients such as the loss of off-site power have been particularly influential in establishing this requirement.

This requirement is to be satisfied under all potential control rod positions within the design limits established and within worst case positional uncertainties for banked primary system control rods. The delay time of 0.1 sec. is specified for consistency with the insertion speeds. Potential tradeoffs between the delay time and insertion speed requirements may be made while assuring that the overall insertion speed requirements are met. This specification is not intended to require rod drop testing during power operation.

16.3.13 Plant Protection System

16.3.13.1 Applicability

Applies to the equipment included as part of the Plant Protection System.

16.3.13.2 Objective

To assure operability of the Plant Protection System.

16.3.13.3 Specification

During all operations requiring PPS action, the following conditions for operability of the PPS shall be met:

1. At least two instrument channels of each subsystem shall be operational. If one channel is inoperative, the comparator output of that channel shall be in the tripped state.

2. At least two logic trains shall be operational. For the primary and secondary shutdown systems where 3 logic trains are provided, the logic output shall be in the tripped state if that logic train is inoperable.

3. Where on-line testing disrupts the capability of the element to initiate channel trip, the output of the channel shall be placed in the tripped mode when that channel is placed in the test mode and the detector is disconnected.

4. The manually instated bypasses for infrequent operating modes shall be properly configured.

16.3.13.4 Basis

For all operating conditions, the PPS provides sufficient redundancy to tolerate a single failure without affecting the ability of the PPS to initiate appropriate protective action. Specifications 1-3 assure that suitable redundancy is preserved even if single element failures occur during test operations. Since certain bypasses are provided for infrequent operations, such as two loop operation, which are not automatically taken out, it is necessary to assure that these bypasses are configured properly for the current operations. Verification of trip settings prior to startup after refueling provides assurance that PPS performance meets the specifications of 16.2.2.

TABLE 16.3.2.1.1

PRIMARY HEAT TRANSPORT SYSTEM LIMITING TEMPERATURES AND PRESSURES

System Section/Component	Pressure (psig)	Temperature (°F)
Reactor Outlet to Pump Inlet	30	1015
Pump Tank and Suction	30	1015
Pump Discharge to IHX	200	1015
IHX Shell Side	200	1015
IHX to Reactor Inlet	200	775
Check Valve	200	775

TABLE 16.3.3.1.1

INTERMEDIATE HEAT TRANSPORT SYSTEM LIMITING TEMPERATURES AND PRESSURES

System Section/Component	Pressure (psig)	Temperature (°F)
Hot Leg Piping	325	965
Cold Leg Piping	325	775
IHX Tubes	325	1015
Flowmeter	325	775
Pump	325	775
Expansion Tank	325	775

16.4 SURVEILLANCE REQUIREMENTS

16.4.1 Operational Safety Review

16.4.1.1 Applicability

Applies to equipment directly related to Safety Limits and Limting Conditions for Operations.

16.4.1.2 Objective

To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

16.4.1.3 Specification

(To be supplied in the FSAR.)

16.4.2 Reactor Coolant System Surveillance

16.4.2.1 In-Service Inspection

16.4.2.1.1 Applicability

Applies to in-service inspection of the HTS and SGS system.

16.4.2.1.2 Objective

To insure the integrity of the HTS over the life of the plant.

16.4.2.1.3 Specifications

(To be supplied in the FSAR.)

16.4.2.2 Steam Generator Auxiliary Heat Removal System

16.4.2.2.1 Applicability

Applies to the surveillance of the Steam Generator Auxiliary Heat Removal System (SGAHRS) and related systems.

16.4.2.2.2 Objective

To establish the surveillance requirements necessary to assure SGAHRS operability and early detection of system degradation.

16.4.2.2.3 Specification

1. The active components of the SGAHRS shall be operationally tested 4 times per year.

2. The components of the SGAHRS with the exception of the protected air cooled condenser shall be visually inspected 4 times per year. Additional details of the required surveillance will be provided in the FSAR.

16.4.2.2.4 Bases

The primary purpose of the SGAHRS is to provide an alternate heat removal path for reactor decay heat and primary and intermediate system stored heat in the event that the normal nuclear steam supply and condensate system or normal feedwater system not be available. The SGAHRS is a safety system which must operate when called upon. Accordingly, the SGAHRS will be tested and inspected periodically to assure:

1. The structural and leaktight integrity of the components,

2. The operability and the performance of the active components, and

3. To the extent possible, the operability of each complete system, and under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation for reactor shutdown and following postulated accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

16.4.2.3 Sodium Water Reaction Pressure Relief Subsystem (SWRPRS)

16.4.2.3.1 Applicability

Applies to the surveillance of the Sodium Water Reaction Pressure Relief Subsystem.

16.4.2.3.2 Objective

To establish the surveillance requirements necessary to assure operability of SWRPRS.

16.4.2.3.3 Specification

1. The Sodium Water Reaction Pressure Relief System (SWRPRS) activation sensor will be functionally tested at (TBD) intervals. TBD sensors must be available to initiate plant shutdown.

2. The atmosphere within the SWRPRS piping will be monitored at (TBD) intervales to verify that the oxygen content is less than (TBD) percent.

3. The nitrogen pressure within SWRPRS piping and components will be monitored at TBD intervals to verify that it is between TBD psi and TBD psi. Additional details of the required surveillance will be provided in the FSAR.

4. The space between the two rupture discs is continuously monitored by sodium leak detection sensors and instrumentation. If the inboard disc is leaking, this disc must be replaced at the next shutdown provided TBD days are not exceeded. If another disc leak is detected in any of the remaining rupture discs in the plant, an immediate shutdown is initiated.

If the specification of items 1, 2, or 3 above are not met, an orderly transition to hot shutdown conditions shall be initiated.

If the specification of item no. 4 is not met, the affected rupture disc assembly shall be replaced at the next shutdown provided TBD days are not exceeded. Otherwise, shutdown and correction must be initiated after TBD days.

16.4.2.3.4 Bases

The components of SWRPRS are required to protect against the consequences of sodium water reactions resulting from water to sodium leaks in the Steam Generator System. Since this system is maintained in a standby condition during plant operation, it requires periodical function testing to assure its operability when required.

There are three activation sensors located in each inerted relief line just downstream of the rupture discs. It is important that they be functional because coincident signals from two out of three of these sensors indicate the rupture discs have burst. The signals automatically initiate reactor scram, sodium pump shutdown, and water/steam side dump, depressurization and isolation.

This system (SWRPRS) will receive and store sodium and reaction products in the event of a large sodium water reaction. The system is inerted and separated from air by a check valve and atmosphere seal (low pressure rupture disc.). Should air leak into the system and the specified oxygen limit be exceeded, a sodium fire could result following actuation of the SWRPRS.

16.4.3 Containment Tests

16.4.3.1 Integrated Leakage Rate Test (Type A)

16.4.3.1.1 Applicability

Applies to leakage rate tests of the primary containment structure.

16.4.3.1.2 Objective

To define the integrated leakage rate tests for the primary containment structure.

16.4.3.1.3 Specification

A. Frequency: three tests, at approximately equally spaced intervals, during each ten year service period.

B. Test Pressure: Integrated leakage rate tests shall be performed at reduced pressure in accordance with 10CFR50 Appendix J.

C. Test Duration: Integrated leakage rate tests shall be performed for the test period of 24 hours.

D. Acceptance Criteria:

1. The integrated leakage rate test shall be preceded by containment inspection and maintenance as specified in Section 16.4.3.3.

2. The containment leakage rate shall not exceed that allowable with B above.

3. If two consecutive periodic integrated leakage rate tests fail to meet this acceptance criteria an integrated leakage rate test will be conducted during each yearly plant shutdown for refueling until two consecutive integrated leakage rate tests meet the acceptance criterion, at which time the test frequency reverts to that specified in A above.

16.4.3.2 Local Leak Rate Tests (Type B & C)

16.4.3.2.1 Applicability

Applies to leakage rate tests of primary containment penetrations and containment isolation valves. All containment penetrations and isolation valves will be leak tested. 20

16.4.3.2.2 Objective

To define the local leak rate tests for the primary containment penetrations and the containment isolation valves.

16.4.3.2.3 Specification

Local leakage rate tests shall be performed with a test pressure of 10 psig and with the containment at atmospheric pressure.

A local leak rate test shall be considered satisfactory if the total combined leakage rate is equal to or less than 0.06 percent by volume of the air contained by the containment at 10 psig per 24 hours.

A leak rate of 0.1% weight* per 24 hour intergrated leak is specified for the containment vessel. In order to conform to Section III, C, 3 of 10CFR50 Appendix J, the leakage rate allotted to type B & C tests is specified as .06% wt. per 24 hrs. with 0.04% wt. per day allotted to the Containment Vessel itself. 20

*It should be noted that weight percent leak rate is the same as a volume percent leak rate. 30

All chilled water lines penetrating the containment will be provided with vents and drains to permit their being drained as follows:

1. Normal chilled water supply and return headers immediately upstream and dowstream of the containment isolation valves will be drainable.

2. Emergency chilled water supply and return lines immediately upstream and downstream of the containment isolation valves will be drainable.

Vents and drains will be opened to permit drainage and to permit communication of the containment test pressure to the closed isolation valves. 20

16.4.3.3 Containment Inspection

16.4.3.3.1 Applicability

Applies to the primary containment structure.

16.4.3.3.2 Objective

To define the extent of the required annual inspection.

16.4.3.3.3 Specification

A visual examination of the accessible interior and exterior surfaces of the containment structure and its components shall be performed prior to any integrated leak test, to uncover any evidence of deterioration which may affect either the containment's structural integrity or leak-tightness. The discovery of any significant deterioration shall be accompanied by corrective actions in accord with acceptable procedures, non-destructive tests and inspections, and local testing where practical, prior to the conduct of any integrated leak test. Such repairs shall be reported as part of the test results.

16.4.3.4 Containment Modification

16.4.3.4.1 Applicability

Applies to modifications to the primary containment structure.

16.4.3.4.2 Objective

To define the conditions and tests required after containment modification.

16.4.3.4.3 Specification

Any major modification or replacement of a component(s) of the containment performed after the initial leakage rate test shall be followed by either an integrated leak rate test, or a local leak rate test, and shall meet the acceptance criteria of 16.4.3.1.

Modifications or replacements performed directly prior to the conduct of an integrated leakage rate test shall not require a separate test.

16.4.3.4.4 Basis

Regular testing and inspection of the primary containment structure and its penetrations is necessary to guarantee the design leak rate, and, in the event of an accident, acceptable dose rates at the site boundary.

16.4.4 HVAC and Radioactive Effluents

16.4.4.1 HVAC Monitoring

16.4.4.1.1 Applicability

Applies to the monitoring and recording of radioactive effluents through the HVAC exhausts.

16.4.4.1.2 Objective

To assure that radioactive releases from the plant are maintained as low as practicable and within the limits of Specification 16.3.11.3.

16.4.4.1.3 Specification

1. Airborne Effluents

   a) All potentially contaminated HVAC discharge paths shall be monitored and records retained showing the identify and quantity of radioactive isotopes released to the environs. The station records shall indicate the existing meteorological conditions for the period of release. For abnormal releases, hourly meteorological data shall be recorded for the periods of actual release.

   b) Monitoring systems shall be functionally tested and calibrated every 3 months.

   c) The monitors shall be capable of measuring release rates of noble gases of 500 μCi/sec, particulates of 0.5 μCi/sec, and iodine of 0.5 μCi/sec. Whenever any of the monitors is inoperable, grab samples will be taken in the affected discharge line.

2. Noble Gas and Tritium Release to the Atmosphere

   Gases continuously released to the atmosphere shall be sampled and analyzed for the isotopic activity:

   a) Within one month after the initial criticality of the reactor and at least monthly thereafter, and

   b) Following each refueling, process change, or other occurrence which could alter the mixture of radionuclides.

   This analysis shall provide the identity and quantity of the principal radionuclides, except tritium, released each month. The sensitivity of the analysis should be such that at least 150 μCi/sec of each nuclide released continuously to the atmosphere is measurable and,

c) the release rate of tritium shall be determined at least quarterly. The sensitivity of the analysis shall be such that at least 150 μCi/sec released continuously to the atmosphere is measurable.

3. Iodine Releases to the Atmosphere

For discharge paths which contain, or potentially contain iodines, a representative sample shall be drawn continuously through an iodine sampling device. The sample collected shall be analyzed at least weekly for I-131. An analysis shall also be performed of a weekly sample at least quarterly for the radionuclides I-133 and I-135. The sensitivity of the analysis for radionuclides shall be such that at least 4×10^{-2} μCi/sec released continuously to the atmosphere is measurable. The results of these analyses shall be used as the basis for evaluating, and reporting the quantities of radioiodines released during the sampling period.

4. Particulate Release to the Atmosphere

For discharge paths that contain or potentially contain radioactive material in particulate form, a sample shall be drawn continuously through a particulate filter. Measurements shall be made on these filters to determine the quantities of radionuclides with half-lives greater than 8 days that are released in particulate form. The particulate filters shall be analyzed,

a) At least weekly for gross radioactivity β, γ, and analyzed for the principal gamma-emitting nuclides (at least for the radionuclides barium-lanthanum-140 and iodine-131), and

b) at least quarterly, on a composite of all filters, for strontium-89 and strontium-90, and

c) at least monthly, on a composite of all filters, for gross alpha radioactivity.

The sensitivities of the analysis shall be such that at least 1×10^{-2} μCi/sec of each gamma-emitting nuclide, 1×10^{-1} μCi/sec for Sr-89 and Sr-90, and 2×10^{-3} μCi/sec for gross alpha radioactivity and 4×10^{-3} μCi/sec for gross β, γ radioactivity released continuously to the atmosphere is measurable. The results of these analyses shall be used as the basis for evaluation and reporting the quantities of radioactive material in particulate form released during the sampling period.

16.4.4.1.4 <u>Basis</u>

The surveillance given under Technical Specification (1) and (2) provide assurance that radioactive effluents released from the plant through the HVAC exhausts are properly monitored and recorded. These surveillance requirements provide the data for the licensee and the Commission to evaluate the plant performance relative to radioactivity released to the environment.

The sensitivities for monitoring and analysis are based on technical feasibility, taking into account the ventilation parameters, and on the potential significance in the environment of the quantities released. For some radionuclides, lower detection limits than those given herein may be achievable and when measurements below the stated sensitivities are attained, the results should be recorded and reported.

For certain mixtures of gamma-emitting nuclides, it may not be possible to measure specific radionuclides at the stated sensitivity limits when other radionuclides are present in the sample in much greater concentrations. Under these circumstances, it will be more appropriate to calculate releases of such radionuclides using observed ratios of these radionuclides, in other processes, to those radionuclides which are routinely identified and measured.

16.4.4.2 Control Room Ventilation System

16.4.4.2.1 Applicability

Applies to components of the Main Control Room Emergency Supply Filter Units.

16.4.4.2.2 Objective

To verify that these components will be able to perform their design function.

16.4.4.2.3 Specification

In-place tests of each HEPA filter section and each charcoal absorber unit will be performed following installation and at refueling intervals thereafter. In addition, in-place test of individual entire filter trains will be performed after each filter or absorber unit change and after any maintenance that may affect the structural integrity of either the filtration or absorber components or the unit housing.

The in-place HEPA filter section DOP test shall conform to ANSI N 101. 1-1972 - "Efficiency Testing of Air Cleaning Systems Containing Devices for Removal of Particles." HEPA filter sections shall be tested to confirm a penetration of less than 0.05% at rated flow. HEPA filter sections that fail to satisfy this requirement shall be replaced or otherwise repaired to meet the test requirement; visual inspection of the filter section will be made after each unsuccessful test to locate any repairable leakage paths that could result in inability to meet the test requirements.

The charcoal absorber section shall be leak-tested with a gaseous halogenated hydrocarbon refrigerant (Freon F-112 or its equivalent) in accordance with USAEC Report DP-1082, "Standardized Nondestructive Test of Carbon Beds for Reactor Confinement Application," to ensure that leakage through the absorber section is less than 0.1%. Absorber sections that fail to satisfy this requirement shall be replaced or otherwise repaired to meet the test requirement.

16.4.4.2.4 Basis

The design of the Control Room HVAC System provides for emergency
filtration of airborne particulates and removal of iodine by means of HEPA fil-
ters and charcoal absorbers. Two emergency air supply filter units are provided.
In the event of high radionuclide concentrations in the outside environment, as
detected by the Radiation Monitoring System, outside makeup air is routed
through one of the emergency filter units in compliance with the requirements
49 | of CRBRP General Design Criterion 17 (defined in Section 3.1).

Since these filter units are not normally in operation, periodic operation is required to ensure their operability when needed. Weekly operation of these units will show that they are available for their safety function. Periodic tests of the filter and absorber sections are required to verify system efficiencies.

This technical specification has been written under the assumption
that emergency filtration of airborne radioactive particulates and absorption
of radioactive elemental iodine will be required to meet the requirements of
49 | CRBRP General Design Criterion 17. In the event that analysis of all postulated
accidents, including extremely unlikely events, shows that there is no hazard
due to the presence of iodine the HVAC system design may be modified to elimi-
nate the charcoal absorber sections. Under these conditions, this technical
specification will be modified accordingly.

16.4.4.3 HVAC System Isolating Valves

16.4.4.3.1 Applicability

Applies to testing of the RCB HVAC system isolating valves.

16.4.4.3.2 Objective

To assure the continual operability of the HVAC valves.

16.4.4.3.3 Specifications

1. At specified intervals, the HVAC isolating valves shall be cycled starting with normal flow conditions and verified by valve position indicator.

2. At specified intervals, isolating valves shall be fully closed and reopened by power operation.

3. At specified intervals, automatic operation shall be verified by simulating the initiating signal.

4. At specified intervals, isolating valves shall be tested for leak
49 | tightness and closure time. Closure time of less than four (4)
seconds shall be verified. Leakage rate shall satisfy the require-
ments of 16.4.3.2.

Amend. 49
April 1979

Note:
Test frequency to be determined following valve manufacturer's recommendations.

49

If the above specifications cannot be complied with in TBD hours, the reactor shall be shutdown and not be taken critical until the deficiencies are remedied.

16.4.4.3.4 Basis

The RCB HVAC system is described in Section 9.6.2. The system is provided with double isolation valves where the air supply and exhaust ducts penetrate the containment. These valves permit isolation of the RCB air atmosphere under accident conditions precluding the release of radioactivity to the environs. Monitors are provided in the exhaust duct work, to continuously monitor the containment exhaust for air-borne particulates and radioactive gases. Signals from these monitors are sent to the plant protection system to automatically actuate closure of the valves in the event of an accidental release of radioactivity into the containment air atmosphere.

16.4.5 Emergency Power System Periodic Tests

16.4.5.1 Applicability

Applies to the periodic testing and surveillance requirements of the emergency power system.

16.4.5.2 Objective

To verify that the standby diesel generators, the Class 1E DC power system, and the standby power and power transfer control systems and the vital AC transfer control system will respond promptly and are capable of performing their intended functions when required.

16.4.5.3 Specification

The following tests and surveillance shall be performed. In the event of failure to pass in-service or shutdown tests, continued reactor operation or reactor restart shall be only in accordance with Section 16.3.9.3.

1) Standby Diesel Generators

   a) Each diesel generator shall be manually started, synchronized with the auxiliary AC power distribution system and at least 75% loaded at intervals of at least once a month. The set shall be run for a period necessary to normalize all operating temperatures. The time to reach operating voltage and speed shall be measured. Only one diesl generator shall be tested at a time.

b) During the monthly test, diesel generator starting air compressors shall be checked for operation. Their ability to recharge the air tanks is checked by reducing the air pressure in each set of tanks below the point at which the corresponding compressor automatically starts. One set of tanks must be full while the other set is being tested.

c) During the monthly test, ability of the diesel fuel oil transfer pumps to refill the day tank shall be demonstrated.

d) During each refueling outage, the condition of a loss of all offsite power signal shall be simulated and the functional operation of the following subsystems shall be checked:

1) Diesel generator start and time to reach operating voltage and speed.

2) Tripping of offsite supply breakers and subsequent closing of diesel generator breaker.

3) Shedding of loads and starting of required engineered safety feature loads in the proper sequence wherever possible within specified time limits.

e) Each diesel generator will be inspected at intervals based on the criteria recommended by the manufacturer. If the recommended interval is expected to occur during an operation cycle, the inspection will be performed during the preceding refueling outage.

f) All test and inspection results shall be recorded.

2) Class 1E DC Batteries

a) Once a week all battery cells shall be visually inspected for cracks, electrolyte leakage damage or discoloring of plates and general cleanliness. The terminals and connectors shall be inspected for evidence of corrosion.

b) Once a week, the electrolyte temperature of pilot cells shall be measured and recorded.

c) Every month, electrolyte level of all cells shall be checked and any water addition shall be recorded.

d) Every three months, cell voltage and specific gravity of all the cells shall be measured and recorded. Total battery voltage shall be measured and recorded.

e) During refueling outage, intercell connections shall be checked for tightness, and an anti-corrosion coating shall be applied to intercell connections and cell terminals.

f) Once a year, a detailed visual inspection of all cells shall be performed and recorded.

g) Once a year, the integrity of the battery racks shall be checked and recorded.

h) During refueling outage, cell to cell terminal connection resistance shall be checked and recorded. The data thus recorded shall be compared with the previous record.

i) A performance test of battery capacity shall be made within the first two years and thereafter at intervals and in accordance with procedures.

j) Spare battery chargers shall be tested to confirm operability during each refueling period.

3) Standby Power Control System

Every six months all diesel generator start, load sequencing control systems, and load shedding shall be tested to demonstrate that the components are operable.

4) Power Transfer Control System

Every refueling period the sequencing control systems, which transfer the two AC distribution systems from the plant AC power supply to the reserve (off-site) AC power supply shall be tested to demonstrate that the components are operable.

5) Vital AC Transfer Control System

Every refueling period, the control systems which transfer the vital buses from the inverter system to its Class 1E backup power source, shall be tested to demonstrate that the components are operable.

16.4.5.4 Basis

The purpose of the standby diesel generators, the Class 1E DC batteries, the standby power control system, and the vital AC transfer control system is to provide a Class 1E source of electrical power to Class 1E loads. Accordingly, these Class 1E systems and equipments will be tested and inspected periodically to confirm their intended performance and to detect any degradation.

The testing of the power transfer control system provides confidence in the availability of the Reserve (off-site) AC Power Supply to provide power to Class 1E loads. (The Reserve AC Power Supply and the power transfer control system are not Class 1E systems.)

16.4.6 Inert Gas System

16.4.6.1 Applicability

Applies to the cover gas purification system.

16.4.6.2 Objective

To define the surveillance requirements on the reactor cover gas.

16.4.6.3 Specification

1. The recycle argon from RAPS processing shall be analyzed for gaseous impurities on the schedule indicated in Table 16.4.6-1.

2. If concentration limits in excess of those listed in Table 16.4.6-2 are confirmed, the source of the contamination shall be determined and corrective action initiated. If the specifications of Table 16.4.6-2 cannot be met in TBD hours, an orderly shutdown of the plant shall be initiated.

16.4.6.4 Basis

The proper operation of RAPS is essential to ensure the necessary cover gas purity.

16.4.7 Reactivity Anomalies

16.4.7.1 Applicability

Applies to potential reactivity anomalies.

16.4.7.2 Objective

To define the requirement for detection of gross reactivity anomalies during reactor startup and operation.

16.4.7.3 Specification

A. Before the reactor is taken critical, a calculation of the Estimated Critical Position (ECP) of the controlling bank shall be completed and recorded.

B. If criticality is achieved with the controlling bank at a height which is above or below the ECP by an amount consistent with TBD dollars, the core power level will not be increased above TBD until the discrepancy has been resolved.

C. After initial normalization of predicted control rod bank height according to measured initial positions, the measured bank height will be compared to the predicted bank height during power operation (accounting fully for the expected power defect, fuel burnup and previous operating history) every full power week and following startup after a shutdown of 72 hours or longer duration.

If the deviation in rod bank height (measured from the bottom of the core) about the expected height corresponds to a reactivity of ≥40¢ per full-power-week or following startup after a shutdown of 72 hours or longer, the reactor will be shutdown for an evaluation of the cause of the reactivity anomaly, which should include an evaluation of the rate of reactivity deviation.

If the deviation from expected bank height corresponds to a reactivity of less than 40¢ per full-power-week, normal operation may continue as long as the total cumulative reactivity anomaly does not exceed $1.50 during an operating cycle. If the cumulative reactivity discrepancy exceeds $1.50, the reactor will be shutdown for an evaluation of the anomaly.

16.4.7.4 Basis

During any approach to criticality, the ECP will be calculated on the basis of expected core loadings and control bank worths. Any significant deviation from the expected critical position may indicate a core misloading, control rod inoperability or other unexpected reactivity change. The maximum allowable deviation from the ECP is based on consideration of the uncertainties in the predictions of criticality and control rod worth, and uncertainties in the fissile fuel loadings, fuel replacement worth (refueling), and the control rod position indicators.

Small, progressive and cumulative reactivity anomalies may be detected by observing the deviations from the expected control rod bank height. The initial uncertainties in bank height are nulled early in life by normalizing the expected height to the observed height.

An increase in bank height above that expected would indicate such occurrences as loss of fuel from the rods, more rapid fuel depletion or less rapid buildup of fission products. In general, less excess reactivity would need to be controlled. The 40¢ limit (60¢ less uncertainties) is specified to account for mechanical motion of core components. This limit is consistent with with the plant safety analysis assumptions (Chapter 15).

The $1.50 long term limit is specified to account for the largest expected calculational uncertainty associated with the prediction of the rate of reactivity change during one operating cycle.

An anomaly analysis of the rate of deviation can be used to detect small, progressive and cumulative reactivity anomalies.

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16.4.8 Pressure and Leakage Rate Test of RAPS Cold Box Cell

14.4.8.1 Applicability

This specification applies to the pressure retaining capability and leak-tightness of the cell containing RAPS cold box.

16.4.8.2 Objective

To assure than an undue risk to the public health and safety does
49 will not exist because of a failure of the RAPS pressure boundary within
the RAPS cold box cell. 36

16.4.8.3. Specification

49 1. The RAPS cold box cell will be leak-rate tested prior to initial plant startup at TBD psig.

49 2. The RAPS cold box cell will be leak-rate tested prior to initial plant startup and thereafter, only if the cell is accessed or as may be dictated by inservice inspection requirement, in accordance with the requirements of Table 16.4.8-1.

16.4.8.4 Basis

The basis for the pressure test is the analysis of the maximum cell pressure as a result of worst-postulated accidents that may occur within the cell. The test for cell leakage rate is based on accepted extrapolations from design leakage rates or leakage rates assumed in accident analysis, whichever is less. 36

16.4.9 Pressure and Leakage Rate Test of RAPS Noble Gas Storage Vessel Cell

16.4.9.1 Applicability

This specification applies to the pressure-retaining capability and leak-tightness of the cell containing the RAPS noble gas storage vessel.

16.4.9.2 Objective

To assure that an undue risk to the public health and safety will not exist because of a failure of the RAPS pressure boundary within the RAPS noble gas storage vessel cell.

16.4.9.3 Specification

1. The RAPS noble gas storage vessel cell will be leak-rate tested prior to initial plant startup at TBD psig.

2. The RAPS noble gas storage vessel cell will be leak-rate tested prior to initial plant startup and thereafter, only if the cell is accessed or as may be dictated by inservice inspection requirement, in accordance with the requirements of Table 16.4.9-1.

16.4.9.4 Basis

The basis for the pressure test is the analysis of the maximum cell overpressure as a result of worst-postulated accidents that may occur within the cell. The test for cell leakage rate is based on accepted extrapolations from design leakage rates or leakage rates assumed in accident analysis, whichever is less.

TABLE 16.4.6-1

SCHEDULE FOR THE ANALYSIS OF RECYCLE ARGON

TBD

TABLE 16.4.6-2

MAXIMUM CONCENTRATION OF IMPURITIES IN RECYCLE ARGON

Impurity	Concentration, ppm by Volume
Oxygen	10
Nitrogen	TBD
Total Hydrogen	8
Total Carbon	25

TABLE 16.4.8-1

INSPECTION REQUIREMENTS ON RAPS COLD BOX CELL

	Design Leakage Rate	12 %/day at	16 psid	
49	Acceptable Accident Leakage Rate	29 %/day at	15.1 psid	36
	Test Leakage Rate	TBD		

TABLE 16.4.9-1

INSPECTION REQUIREMENTS ON RAPS NOBLE GAS STORAGE VESSEL CELL

	Design Leakage Rate	10%/day at 10 psid
49	Acceptable Accident Leakage Rate	28%/day at 7.3 psid
	Test Leakage Rate	TBD

16.5 DESIGN FEATURES

16.5.1 Site

16.5.1.1 Applicability

Applies to the location and extent of the reactor site.

16.5.1.2 Objective

To define those aspects of the site which affect the overall safety of the installation.

16.5.1.3 Specification

The Clinch River Site is in east central Tennessee, in the eastern part of Roane County approximately 25 miles west of Knoxville and consists of 1364 acres. The site is on a peninsula bounded on the south by the Clinch River from approximately Clinch River Mile (CRM) 15 to CRM 18 and on the north by AEC's Oak Ridge Reservation.

The point of minimum exclusion distance from the center of the containment will be the opposite river bank and will be approximately 2200 feet.

16.5.2 Confinement/Containment

16.5.2.1 Applicability

Applies to those design features of the reactor containment
18 structure.

16.5.2.3 Specifications

Containment for the reactor consists of a steel containment vessel surrounded by a reinforced concrete, confinement building. The annular space between the two buildings is maintained at a negative pressure with respect to the atmosphere in order to achieve as close to zero leakage out
18 of the confinement building as possible.

Structure

The containment vessel is a low leakage, steel structure consisting of a vertical cylinder, a hemispherical dome, and a bottom line plate encased in concrete. The interior is divided into two volumes: a lower volume and an upper volume. The reinforced concrete confinement building surrounds the containment vessel, with an annular space between the containment and confinement. During an accident, the annulus ventilation exhaust is discharged through a high efficiency filter with a 99% particulate and 95% absorbant efficiency.

The reactor confinement/containment system and penetrations are designed to limit the leakage of radioactive fission products to less than 10CFR100 values for the largest credible mass and energy releases following a design basis accident.

The containment vessel is designed for an internal pressure of 10 psig. The confinement/containment system is designed for an earthquake with simultaneously acting maximum horizontal and vertical ground accelerations.

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16.5.3 Reactor

16.5.3.1 Applicability

51 Applies to the CRBRP nuclear fuel and inner blanket region and axial
and radial blanket regions.

16.5.3.2 Objective

To define those system features which are essential in providing for safe system operations.

16.5.3.3 Specification

The CRBRP features a mixed plutonium/uranium dioxide fueled, sodium
cooled fast breeder reactor design. The initial fuel loading contains approxi-
51 mately 5189 kg of heavy metal (Pu + U) of which 1502 kg is fissile plutonium
(Pu - 239 + Pu - 241). A single fuel enrichment is used in all fuel assemblies.
The fuel is in the form of sintered powder pellets, encapsulated in nonvented 20% cold worked austenitic stainless steel tubing to form fuel rods. Blanket pellets made of depleted uranium dioxide are placed in each fuel rod above and below the fuel pellet stack to form upper and lower axial blanket regions. A space is provided in each rod above the upper axial blanket for collecting the fission gases released from the pellets during power operation. The fuel rods are held in a triangular array by spiral wire wrap spacing inside a hexagonal duct to form a fuel assembly. Each fuel assembly contains 217 fuel rods.

The initial core contains 156 fuel assemblies, 82 inner blanket
assemblies, 9 primary system control assemblies and 6 secondary system control
assemblies. The fuel and blanket assemblies are arranged in alternating rows
51 near the center of the core (a so-called radial parfait arrangement). Fuel
assemblies are selectively clustered around the control rods.

Surrounding the fuel and inner blanket assemblies are 132 radial
blanket assemblies which are arranged in two rows and are of construction
51 similar to the fuel assemblies. Each radial (and inner) blanket assembly
contains 61 rods that are fueled with depleted uranium dioxide pellets which have an overall stack height that matches that of the core fuel plus the two axial blankets.

The outer four rows that complete the reactor assembly lattice group-
51 ing are made up of 306 radial shield assemblies. These assemblies protect the

core restraint former rings and the core barrel structure, which supports the
rings, from excessive structural damage by neutron fluence. The core barrel
is a thick wall upright circular cylinder that surrounds the reactor assembly
group and is welded to the core support plate. It extends upward to the top
of the reactor assemblies. The annular space between the core barrel and the
reactor vessel is closed at the top of the barrel by a horizontal baffle. This
5 ring-like structure is thick to provide insulation and is flex-ring seated at
the barrel and vessel interfaces so that it forms a barrier between the hot
sodium above in the outlet plenum and the cooler sodium below.

Reactor core cooling is provided by the upward flow of the liquid sodium. The sodium enters the inlet plenum of the reactor vessel and flows upward to the 61 inlet modules inserted in the core support plate. After entering each module the flow is distributed to the seven reactor assemblies it holds. The coolant then flows upward and through each reactor assembly.

The reactor core is arranged in five flow zones and the radial blanket in four. Mechanical devices are provided to prevent fuel assemblies from being inserted into positions where they would be undercooled. Radial blanket assemblies also are provided with mechanical devices which preclude their insertion into positions other than radial blanket positions. Flow orifices at the bottom of each core and radial blanket duct and in the inlet modules establish reactor coolant flow control. Shield pieces in the bottom ends of the core and radial blanket assemblies attenuate the neutron fluence to the core support plate thus protecting it from excessive structural damage and internal heating. The core and radial blanket assemblies have identification notches and a conical surface on their outlet nozzles for remote identification and ease of engagement by the fuel handling machines. For the radial blanket assemblies, this provides means for avoiding shuffling of radial blanket assemblies, in the fuel management plan, into flow zone positions in the blanket region where they would be undercooled.

The core assemblies are held down against their respective inlet modules by a hydraulic balance arrangement and their own weight with mechanical backup provided by the upper internal structure. A passive radial restraint system is provided in the design which involves all of the reactor assemblies, core support and upper internal structures, and core restraint former rings. The two core restraint former rings girth the outside contours that are formed by the upper and lower load pads on the outer row radial shield assemblies.

Lateral actions by the reactor assemblies are limited by reactions at the former rings which are placed at the same elevations as the hard faced load pads on the outsides of all reactor assemblies. Interactions between reactor assemblies and the former rings are limited to these pads since they are raised above the outer surfaces of the ducts.

The reactor assemblies are controlled by the core restraint system so that they bow under thermal gradients in a way that the fuel rods spread a controlled amount as power is increased. This action contributes to the negative temperature and power coefficients of reactivity which the reactor has. The ratio of fissile to fertile nuclei in the fuel is such that Doppler broadening also slightly decreases reactivity as power is increased. These negative feedback coefficients provide inherent dynamic power stability of the reactor.

Net reactivity operational control in the reactor core is accomplished by the two control rod systems which include two independent shutdown systems to enhance the overall CRBR shutdown reliability. Reactor shutdown can be achieved by either system with the other system completely inoperable and with the control rod of highest worth stuck in the operable system. Both systems use boron carbide pellets that are sealed in tubes to form neutron absorber pins.

The isotope B-10 in the B_4C has a significant absorption cross section for fast neutrons and therefore the pins of the control rod assemblies act as a poison in controlling reactivity in the core by depth of insertion into the 15 strategically located lattice positions occupied by the primary and secondary system control assemblies. The secondary control rods are parked in above-core positions and are used only for shutdown. The absorber pins of all 15 control
51 rods are 92% enriched in B-10.

The additional features provided in the reactor design include the
51 following:

a. Redundant flow paths in the inlet modules, support plate and reactor assembly nozzles to preclude flow blockage.

b. Strainers in the inlet modules to prevent particular matter that has a major dimension greater than 1/4 inch from entering the core and radial blanket assemblies.

c. Spiral wire wraps on the outsides of the fuel and blanket rods and the neutron absorber pins to promote sodium coolant mixing and hence more uniform cladding and duct wall temperatures.

d. Ex-vessel flux detectors for continuous monitoring of the neutron flux level of the reactor core.

51 e. Thermocouples at fuel, inner and radial blanket assembly outlets and outlet plenum positions for measuring and monitoring reactor thermal performance.

51 f. Discrete duct enclosed fuel, inner and radial blanket assemblies to limit the potential for propagation of local fuel failures and promote safety in fuel handling operations.

g. Radial key extensions on the upper internal structure which engages the upper core barrel former ring to assure proper alignment with the core and provide seismic support.

h. Vortex suppressor plate to minimize entrainment of argon cover gas in the liquid sodium reactor coolant.

i. Tag gases in the fuel rods, that differ for each assembly, to permit identification and location of leaker fuel assemblies by means of mass spectrometer examination of the cover gas.

j. Measurements of Reactor Vessel sodium level to ensure adequate sodium for core cooling.

16.5.4 Heat Transport System

16.5.4.1 Applicability

Applies to the primary and intermediate heat transport systems boundary and cooling capability.

16.5.4.2 Objective

To define the design features essential to provide continued reactor core cooling and to assure the integrity, of the primary and intermediate heat transport systems boundary.

16.5.4.3 Specification

1. The heat transport system (HTS) consists of the piping and components required to transport reactor heat to the steam generators. The HTS is comprised of three independent cooling circuits each of which includes a primary sodium loop and an intermediate sodium loop thermally coupled by an intermediate heat exchanger (IHX).

2. The primary heat transport system (PHTS) piping and components are located in nitrogen inerted cells within the containment building. The cells are separated from each other by concrete shielding and flexible seals. The physical layout of the primary and intermediate heat transport systems shall preclude the propagation of a failure in one loop to the remaining loops.

3. Each PHTS loop is arranged in an "elevated loop" concept (see Figure 5.1-3) to provide protection against loss of coolant in the unaffected loops in the event of a sodium boundary failure in one of the loops.

4. The intermediate heat transport system (IHTS) loops circulate non-radioactive sodium coolant from the tube side of the intermediate heat exchangers (located in the containment building) to the steam generators (located in the steam generator building). A rigid seal is provided at each piping penetration through the containment wall.

5. Design, fabrication, erection, and testing of the HTS piping and components which comprise the sodium boundary shall be in accordance with the ASME Boiler and Pressure Vessel Code, Section III for Class 1 components and the applicable code cases and RDT Standards given in Sections 5.3, 5.4, and 5.5.

6. The HTS shall be designed to accommodate the thermal transients resulting from the normal, upset, emergency and faulted conditions described in the CRBRP Design Duty Cycle given in Appendix B. Specifically the system shall be designed such that:

   a. Normal or upset event does not adversely affect the useful life of any component;

   b. Following an emergency condition, resumption of operation must be possible following repair and reinspection of the components; and

   c. Following a faulted condition, the system must remain sufficiently intact to be capable of performing its decay heat removal function.

7. The HTS shall be designed to accommodate the pressure transients without loss of decay heat removal capability, resulting from the following conditions:

   a. The pressures imposed on the IHTS by a major sodium-water reaction.

   b. The pressures resulting from a PHTS check valve closure caused by the most severe flow degradation, such as a primary pump seizure.

8. All HTS piping and components are Category 1 and shall be designed and analyzed in accordance with the environmental design criteria as given in Sections 5.3, 5.4, and 5.5.

9. One sodium pump is provided in the hot leg of each of the three PHTS loops. The pump is a centrifugal unit equipped with a variable speed drive. An auxiliary pony motor on each pump units provides low flow capability for decay heat removal and other low power standby conditions.

10. Each IHTS loop contains a centrifugal pump with similar hydraulic characteristics to the primary pump. The pump is located in the IHTS cold leg and is provided with the same basic speed control system and auxiliary pony motor as the primary unit.

11. The shell and tube, vertically mounted, IHX transfers the reactor heat from the PHTS to the IHTS and acts as the barrier between the primary radioactive sodium and the secondary non-radioactive sodium. There is one IHX in each of the three HTS circuits on the primary side, the IHX is located down stream of the primary pump just ahead of the check valve; on the intermediate side, the IHX is located between the intermediate pump and the superheater.

12. The HTS shall be designed such that decay heat removal can be effected by utilizing the normal heat removal train. This capability must be assured for both three and two loop operation for all upset, emergency and faulted events. For these events sufficient coolant flow shall be provided to ensure that corresponding fuel design limits defined in Chapter 4 are not exceeded. The relative elevations of the reactor core, IHX tube bundle and the steam generator modules are arranged to promote natural circulation of sodium in the PHTS and IHTS loops in the event of loss of all electrical power to the pumps.

16.5.5 Fuel Storage

16.5.5.1 Applicability

Applies to the storage of new and spent fuel assemblies.

16.5.5.2 Objective

To define those system features which are essential in providing for safe fuel storage.

16.5.5.3 Specification

The fuel storage facilities consist of new fuel storage and spent fuel storage.

A. New Fuel Storage

New fuel is stored in the RSB in the EVST (see Item B). In addition to the EVST, new fuel is temporarily retained in shipping containers after a truck with the Safe Secure Trailer arrives in the hardened part of the RSB and in two new fuel unloading stations below the RSB operating floor. Each fuel unloading station consists of a pit which can contain one shipping container with a single new fuel assembly. New fuel assemblies are unloaded from
14 20 the shipping containers in the two unloading stations using the EVTM.

New fuel is also stored under sodium in the EVST, described below.

B. Spent Fuel Storage

Spent fuel is stored in the RSB in two locations: in the EVST and, on a temporary basis, in the spent fuel transfer station of the fuel handling
4 cell (FHC).

The EVST is a single vessel, sodium-filled storage facility with a two-tier rotatable turntable. It is located between the EVTM gantry rails in
20 the RSB. It can store approximately 650 new and/or spent fuel assemblies, each in a core component pot (CCP). The primary vessel is surrounded by a

guard tank as a safety measure against any sodium leaks. The guard tank is
situated in a nitrogen gas-filled concrete vault. The space between primary
vessel and guard tank is sized to maintain a minimum safe sodium level above
59 20 the fuel assemblies (i.e., 31 inches above the upper edge of the lower axial blanket)
in the extremely unlikely event of a gross primary vessel failure. The primary vessel is supported from its upper flange, suspended into the guard tank. The
turntable is supported through a bearing and seal configuration above the
20 primary vessel flange. The guard tank is bottom supported from the vault floor.
The fuel assembly storage positions are cylindrical tubes, arranged in concentric rows, restrained and supported by a stainless steel gridwork in the
rotatable storage rack. Each storage tube holds two CCP's one above the other.
Sodium coolant flow enters each tube at the bottom and leaves at the top, as
well as circulating around the outside of the tube. Heat is removed by two
independent, redundant sodium cooling loops. The primary vessel is sealed
and shielded from the RSB operating floor by a heavy closure head. The closure
44 20 head and a striker plate above it also prevent internal EVST damage from external drop loads.

The FHC spent fuel transfer station is located directly below the
FHC fuel transfer port. It is a temporary storage facility cooled by natural
argon convection with a rotatable basket holding up to 3 fuel assemblies in
20 core component pots in a triangular array. The transfer station is supported
at its upper flange. The rotatable basket consists of a stainless steel web
structure and cylindrical sockets for support of the CCP's. Each storage
location holds one fuel assembly. The method of heat removal is by natural
convection to the FHC argon atmosphere. Sealing and shielding at the RSB
operating floor is provided by the heavy FHC steel roof plug structure. It
44 also provides protection of the FHC interior against external drop loads.

The safety features provided in the EVST and FHC spent fuel trans-
44 fer station design include the following:

59 a. Physical separation of fuel assemblies with structural support to prevent changes in separation distance or displacement due to combined normal and SSE or other abnormal loads.

b. A heavy roof structure and steel-lined concrete vault walls protect the RSB operating floor, FHC operating gallery and
20 neighboring cells from radiation.

c. Double seals around the fuel transfer port plugs, FHC viewing windows and manipulator penetrations, between the EVST primary vessel and head, and between EVST cover plate and vault lining prevent radioactivity release from the EVST and FHC.

d. The location of sodium inlet and outlet pipes, provisions of antisyphon devices, and the presence of a guard vessel prevent any loss of sodium coolant from the EVST that could prevent
4 cooling of spent fuel.

e. Temperature instrumentation, sodium level sensors, and sodium leak detectors monitor the thermal performance and the coolant inventory of the spent fuel storage facilities. Local area radiation monitors at the RSB operating floor and in the FHC operating gallery warn of any potentially hazardous radioactive release from the storage facilities.

16.6 ADMINISTRATIVE CONTROLS

16.6.1 Organization

1. The plant manager has onsite responsibility for the safe operation of the facility and shall report to the Assistant Director of Nuclear Power (Operations). In the absence of
61 the plant manager, the assistant plant manager will assume his responsibilities.

2. The portion of TVA management line of responsibility which relates to the operation of the plant is shown in Figure 16.6-1.

3. The functional organization for the operation of the plant shall be shown in Figure 16.6-2.

4. Shift manning requirements shall, as a minimum, be as delineated in Section 16.6.8.

5. Qualifications of the CRBRP management and operating staff shall meet the minimum acceptable levels as described in ANSI/ANS-3.1-1978, Selection and Training of Nuclear Power
61 Plant Personnel, dated January 17, 1978.

6. Retraining and replacement training of plant personnel shall be in accordance with ANSI/ANS-3.1-1978, Selection and
61 Training of Nuclear Power Plant Personnel, dated January 17, 1978. The minimum frequency of the retraining program shall be every two years.

16.6.2 Review and Audit

The Manager of Power has delegated responsibility to the Nuclear Safety Review Board (NSRB) to monitor the Plant Operations Review Committee activities and to ensure the proper operational safety review by off-site personnel who have no direct responsibility for plant operations. The Office of Power Quality Assurance and Audit Staff has the responsibility for the plant audit functions.

16.6.2.1 Nuclear Safety Review Board

The Nuclear Safety Review Board (NSRB) advises the Manager of Power on the adequacy and implementation of TVA nuclear safety policies and programs and assures that these policies and programs are in compliance with NRC regulatory requirements. In general, the review and investigation
61 functions are performed independently of NSRB. The NSRB is responsible for evaluating the results of such activities to determine that all nuclear safety-related aspects are being adequately considered. In addition, the NSRB may conduct reviews or investigations of any nuclear safety-related activity in order to evaluate the TVA nuclear safety program.

The NSRB is comprised of a chairman and at least five other members appointed or approved by the Manager of Power. Members of the NSRB may be from the Office of Power, or other TVA organization or external to TVA. The NSRB meets on a periodic and as-required basis to perform those
61 functions identified above.

Membership requirements and responsibilities of the NSRB will be defined in the Nuclear Safety Review Board Charter and the Nuclear Safety Review Board Guidelines, both of which are formally approved by the Manager of Power.

61 16.6.2.2 Office of Power Quality Assurance and Audit Staff

The Office of Power Quality Assurance and Audit (QA&A) Staff
61 Organization will be responsible for assuring the implementation and maintenance of an effective quality assurance program, including the auditing of all safety-related activities of the CRBRP. Through the audit program, existing and potential deficiencies are identified and appropriate corrective actions are assigned. Through formal audit reports, the Nuclear Safety Review Board and Manager of Power are advised of any identified deviations from procedural requirements and licensing commitments.

The Quality Assurance and Audit Staff Organization is comprised
61 of two sections plus a number of Quality Assurance Representatives who are resident in the operating nuclear plants and report directly to the QA&A Manager on the status of in-plant quality assurance. Its functional arrangement is shown in Figure 16.6-4.

16.6.2.3 Plant Operations Review Committee (PORC)

1. Membership

   The PORC shall consist of the plant manager, assistant plant manager, maintenance supervisors, health physics supervisor, operations supervisor, plant engineering supervisor, and Supervisor, Quality Assurance Staff. An assistant plant supervisor may serve as an alternate committee member when his supervisor is absent; however, no more than two
   61 alternates shall participate as voting members in PORC activities at any one time.

   The plant manager will serve as chairman of the PORC. The assistant plant manager will serve as chairman in the absence of the plant manager.

2. Meeting Frequency

   The PORC shall meet at regular monthly intervals and for special meetings as called by the chairman or as requested by individual members.

3. Quorum

The chairman or his designated alternate, plus four of the other members, or their alternates, will constitute a quorum. A member will be considered present if he is in
61 telephone communication with the committee. The member who is absent and whose alternate has not been provided must not be that member having principal responsibility or expertise in the area being reviewed.

4. Duties and Responsibilities

The PORC serves in an advisory capacity to the plant manager and as an investigation and reporting body to the Nuclear Safety Review Board in matters related to safety in plant operations. The plant manager has the final responsibility in determining the matters to be implemented and/or referred to the Nuclear Safety Review Board.

The responsibilities of the committee will include:

a. Review all standard and emergency operating and maintenance instructions and any proposed revisions thereto, with principal attention to provisions for safe operation.

b. Review proposed changes to the license and Technical
61 Specifications.

c. Review proposed changes to equipment or systems having safety significance, or which may constitute "an unreviewed safety question," pursuant to 10CFR50.59.

d. Investigate reported or suspected incidents involving safety questions, violations of the Technical Specifications, and violations of plant instructions pertinent to nuclear safety.

61 e. Review reportable occurrences, unusual events, operating anomalies, and abnormal performance of plant equipment.

f. Maintain a general surveillance of plant activities to identify possible safety hazards.

g. Review plans for special fuel handling, plant maintenance, operations, and tests or experiments which may involve special safety considerations, and the results thereof, where applicable.

h. Review adequacy of quality assurance program and recommend any appropriate changes.

i. Review adequacy of Technical Specifications and recommend any appropriate changes.

61 j. Review unit operations to detect potential nuclear safety hazards.

k. Review all proposed tests and experiments that affect
61 nuclear safety.

l. Review the site Radiological Emergency Plan and the Plant
61 Physical Security Plan.

m. Review adequacy of employee training programs and recommended changes.

n. Review every unplanned onsite release of radioactive material to the environs.

o. Review changes to the Radwaste Treatment System.

p. Review meeting minutes of the Radiological Assessment Review Committee (RARC).

61 q. Performance of reviews as requested by plant manager.

5. Authority

The PORC shall be advisory to the plant manager.

6. Records

Minutes shall be kept for all PORC meetings with copies sent to Director, Division of Nuclear Power, Assistant Director of Nuclear Power (Operations), and Chairman of the Nuclear
61 Safety Review Board.

7. Procedures

Written administrative instructions including applicable check-off lists prepared and maintained describing the method for submission and content of presentations to the committee, review and approval by members of committee actions, dissemination of minutes, agenda and scheduling of meetings.

16.6.3 Instructions

A. Detailed written instructions, including applicable check-off lists covering items listed below shall be prepared, approved and adhered to.

1. Normal startup, operation, and shutdown of the reactor and of all systems and components involving nuclear safety of the facility.

2. Refueling operations.

3. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms and abnormal reactivity changes.

4. Emergency conditions involving potential or actual release of radioactivity.

5. Preventive or corrective maintenance operations which could have an effect on the safety of the reactor.

6. Surveillance and testing requirements.

7. Radiation control procedures.

8. Radiological Emergency Plan implementing procedures.

9. Plant security program implementing procedures.

B. Written procedures pertaining to those items listed above shall be reviewed by PORC and approved by the plant manager prior to implementation except that temporary changes to procedures which do not change the intent of the original procedure may be made with the concurrence of two persons holding senior reactor operator licenses. Such temporary changes shall be documented and subsequently reviewed by PORC and approved by the plant manager.

C. Drills on actions to be taken under emergency conditions involving release of radioactivity are specified in the radiological emergency plan and shall be conducted annually. Annual drills shall also be conducted on the actions to be taken following failures of safety-related systems or components.

D. Radiation Control Procedures

Radiation control procedures shall be maintained and made available to all plant personnel. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10CFR20. This radiation protection program shall be organized to meet the requirements of 10CFR20. The project is proposing, however, that the provisions of Section 20.203(c) subparagraphs (2), (3), and (4) apply only to areas where the radiation levels are continuously 1,000 mRem/hr or greater.

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61 16.6.4 <u>Actions to be Taken in the Event of Reportable Occurrence in Plant Operation</u>

A. Any reportable occurence shall be promptly reported to the Director, Division of Nuclear Power, and the NSRB, and shall be promptly reviewed by PORC. This committee shall prepare a separate report for each reportable occurrence. This report shall include an evaluation of the cause of the occurrence and recommendations for appropriate action to prevent or reduce the probability of a repetition of the
61 occurrence.

B. Copies of all such reports shall be submitted to the Assistant Director of Nuclear Power (Operations); Manager, Nuclear Regulations & Safety; Chief, Radiological Hygiene Branch; Supervisor, Nuclear Safety Review Staff; and the
61 Chairman of the NSRB for their review.

C. The plant manager shall notify the NRC within 24 hours, as specified in Specification 16.6.7 of the circumstances of any abnormal occurrence. A written report shall follow within 10 days.

16.6.5 <u>Action to be Taken in the Event a Safety Limit is Exceeded</u>

If a safety limit as defined in 10CFR50.36(c) (1) (i) is exceeded, the reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC. A prompt report shall be made to the Director, Division of Nuclear Power and the Chairman of the NSRB. A complete analysis of the circumstances leading up to and resulting from the situation, together with recommendations to prevent a recurrence, shall be prepared by the PORC. This report shall be submitted to the Assistant Director of Nuclear Power (Operations); Chief, Radiological Hygiene Branch; Manager, Nuclear Regulation & Safety; Supervisor, Nuclear Safety Review Staff; and the NSRB. Notification of such occurrences will be made to the NRC by the plant manager within 24 hours as specified in Specification 16.6.7 followed by a written report within 10 days to the Director, Office
61 of Management Information and Program Analysis US NRC.

16.6.6 <u>Station Operating Records</u>

A. Records and/or logs shall be kept in a manner convenient for review as indicated below:

1. All normal plant operations including such items as power level, fuel exposure, and shutdowns.

2. Principal maintenance activities.

3. Abnormal occurrences.

4. Checks, inspections, tests, and calibrations of components and systems, including such diverse items as source leakage.

5. Reviews of changes made to the procedures or equipment or reviews of tests and experiments, to comply with 10CFR50.59.

6. Radioactive shipments.

7. Record of annual physical inventory verifying accountability of sources on record.

8. Gaseous and liquid radioactive waste release to the environs.

9. Off-site environmental monitoring surveys.

10. Fuel inventories and transfers.

11. Plant radiation and contamination surveys.

12. Radiation exposures for all plant personnel.

13. Updated, corrected, and as-built drawings of the plant.

14. Minutes of meetings of the Nuclear Safety Review Board.

B. Except where covered by applicable regulations, items 1 through 6 above shall be retained for a period of at least 5 years and items 7 through 14 shall be retained for the life of the plant. A complete inventory of radioactive materials in possession shall be maintained current at all times.

16.6.7 Reporting Requirements

A. Routine and Reportable Occurrence Reports to NRC

Information under this category to be reported to the NRC includes the following:

1. Reports required by Title 10, Code of Federal Regulations.

2. Reports of radioactive discharges and radiological monitoring which have been transferred, per directive from NRC, from Appendix B to Appendix A of Regulatory Guide 1.16.

3. Reports required by the current revision of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A, Technical Specifications", as applicable to liquid metal fast breeder reactors with exceptions to be determined.

61

B. Special Reports to NRC

Special reports to the NRC are required covering inspections, tests, and maintenance activities, and nonroutine activities which are specified in parts of Title 10, Chapter 1, Code of Federal Regulations.

Specific requirements will be included in the FSAR.

C. Environmental Monitoring Reports to Environmental Protection Agency (EPA)

Reporting information to the EPA concerning non-radiological environmental surveillance and environmental impact will
61 follow the guidelines and requirements of the NPDS permits issued to the CRBRP.

16.6.8 Minimum Staffing

A. Table 16.6-1 shows the number of shift personnel whenever the plant is not at shutdown or refueling shutdown conditions.

B. A licensed senior operator shall be present at the site at all times when there is fuel in the reactor.

C. A licensed operator shall be in the control room when the reactor contains a potential critical mass.

D. A licensed senior operator shall be in direct charge of a reactor refueling operation; i.e., able to devote full time to the refueling operation.

E. A Shift Technical Advisor shall be onsite at all times except when the reactor is at refueling temperature.

F. A health physics technician shall be present at the facility at all times when there is fuel in the reactor.

G. Two licensed operators shall be in the control room during startups, while shutting down the reactor, and during recovery from any plant trip.

H. Either the plant manager or the assistant plant manager shall have acquired the experience and training normally required for examination by the NRC for a Senior Reactor Operator's License, whether or not the examination is taken. In addition, the operations supervisor or assistant
61 operations supervisor shall have an SRO license.

61 TABLE 16.6-1

MINIMUM SHIFT CREW REQUIREMENTS

	Shift Position	Members	Type of License
	Shift Engineers (SE)	1	SRO[(a)]
61	Shift Technical Advisor(STA)	1	None
	Assistant Shift Engineers (ASE)	1	SRO[(a)]
61	Unit Operators (UO)	2	RO[(b)]
	Assistant Unit Operators (AUO)	2	None
	Health Physics Technician	1	None
61	Minimum Shift Crew	8	None

(a) SRO - Senior Reactor Operator

(b) RO - Reactor Operator

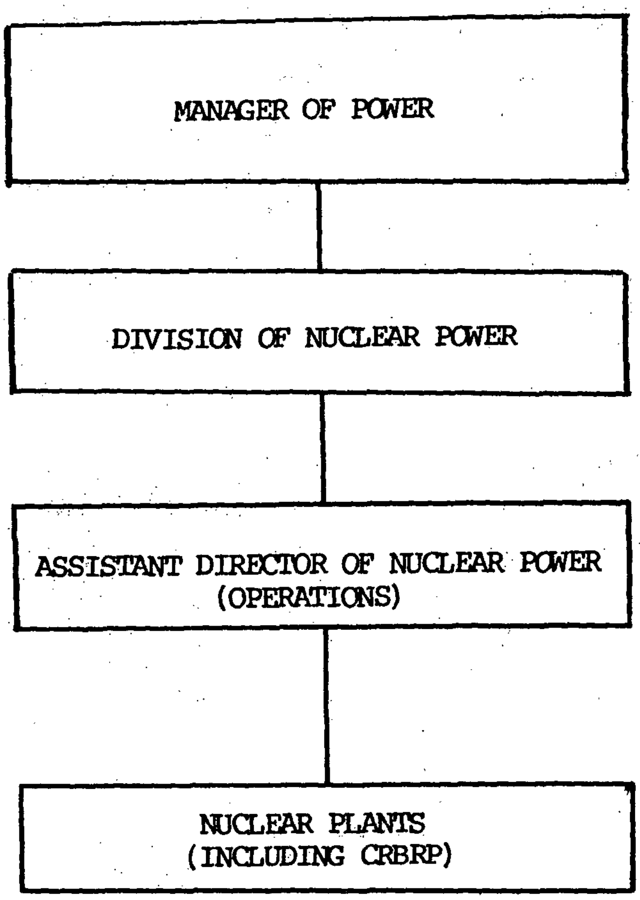

Figure 16.6-1. TVA Office of Power Organization Line of Responsibility for the CRBRP

6738-1

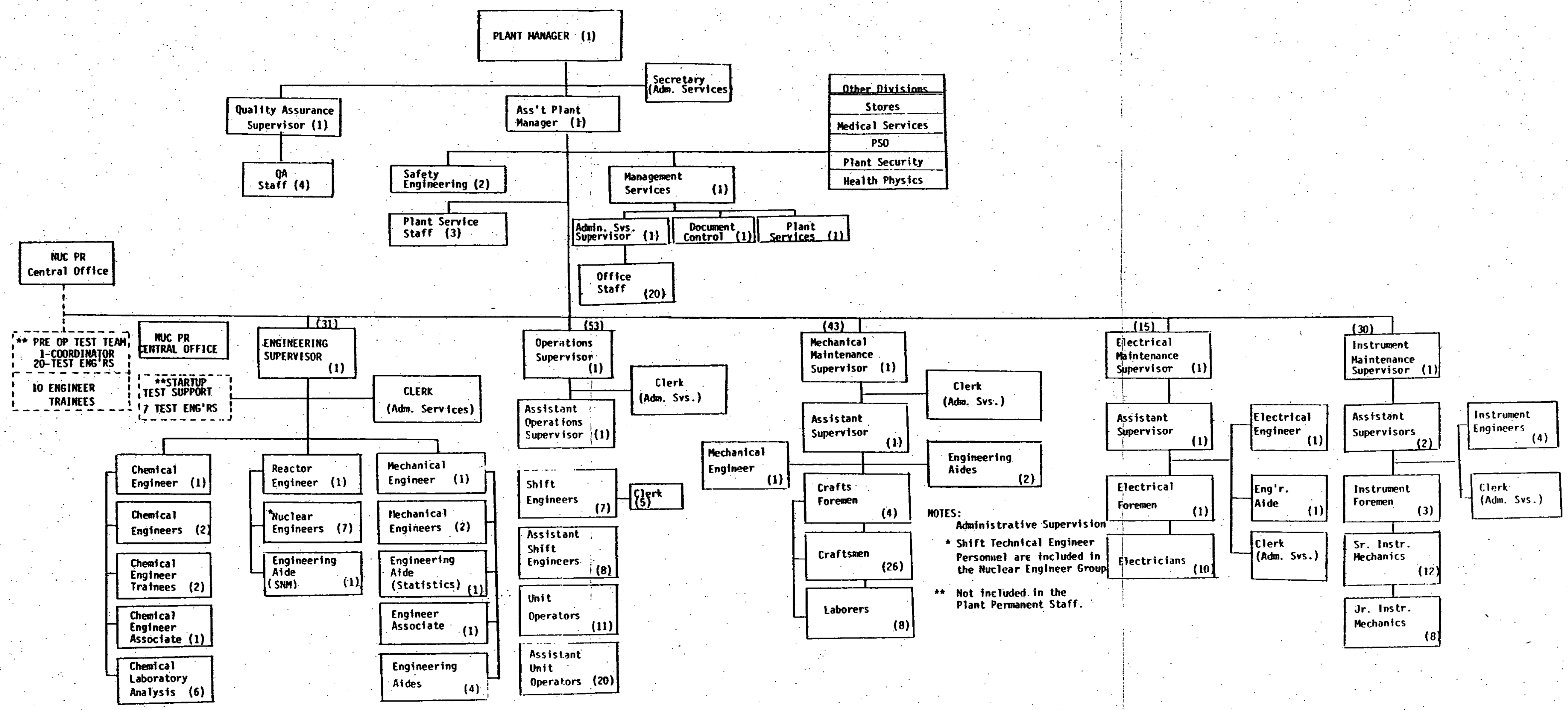

FIGURE 16.6-2 CRBRP Organization Chart (1 year after power operations) 6717-1

Amend. 61
Sept. 1981

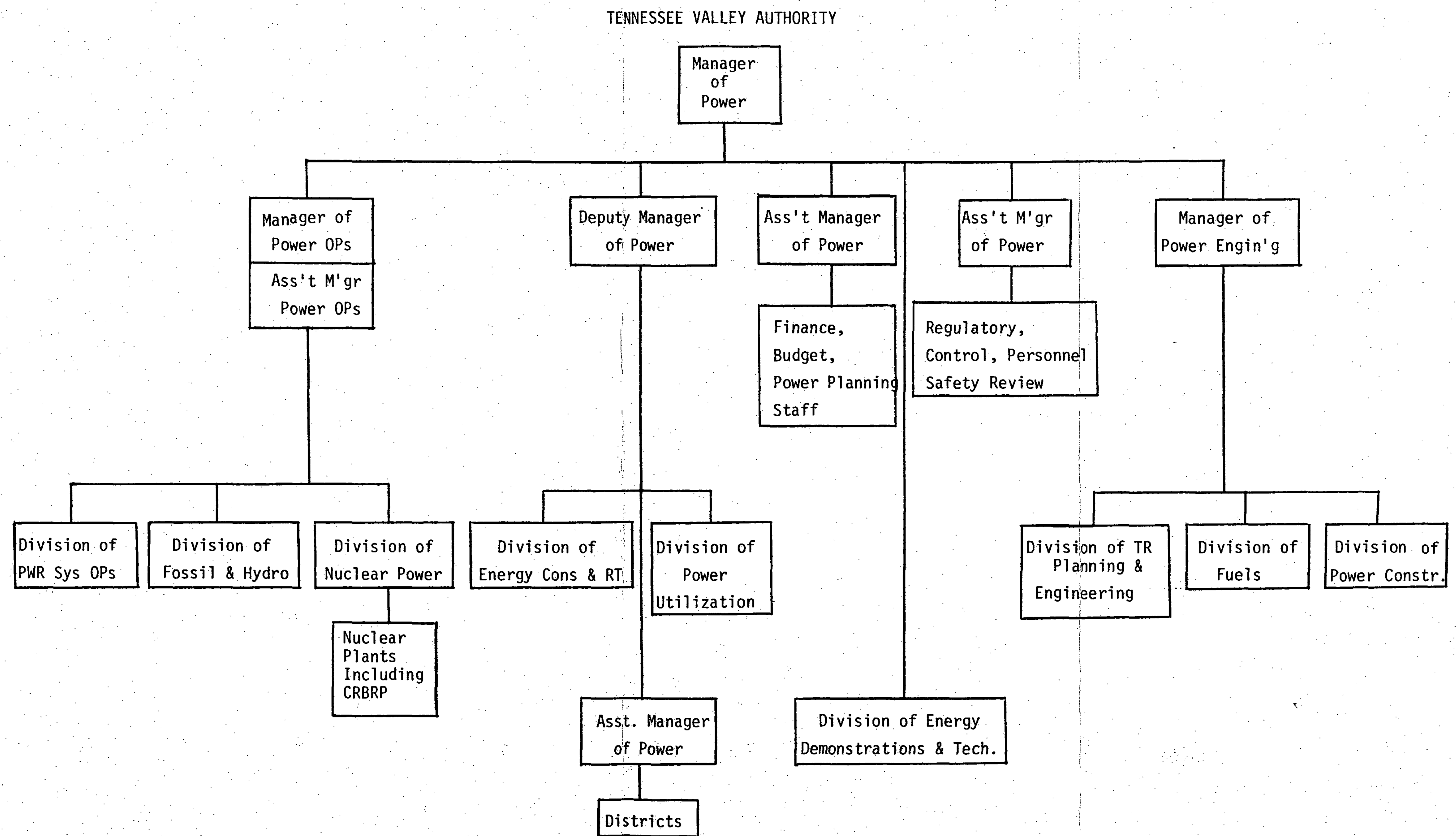

FIGURE 16.6-3 Office of Power Organization (TVA)

6738-4

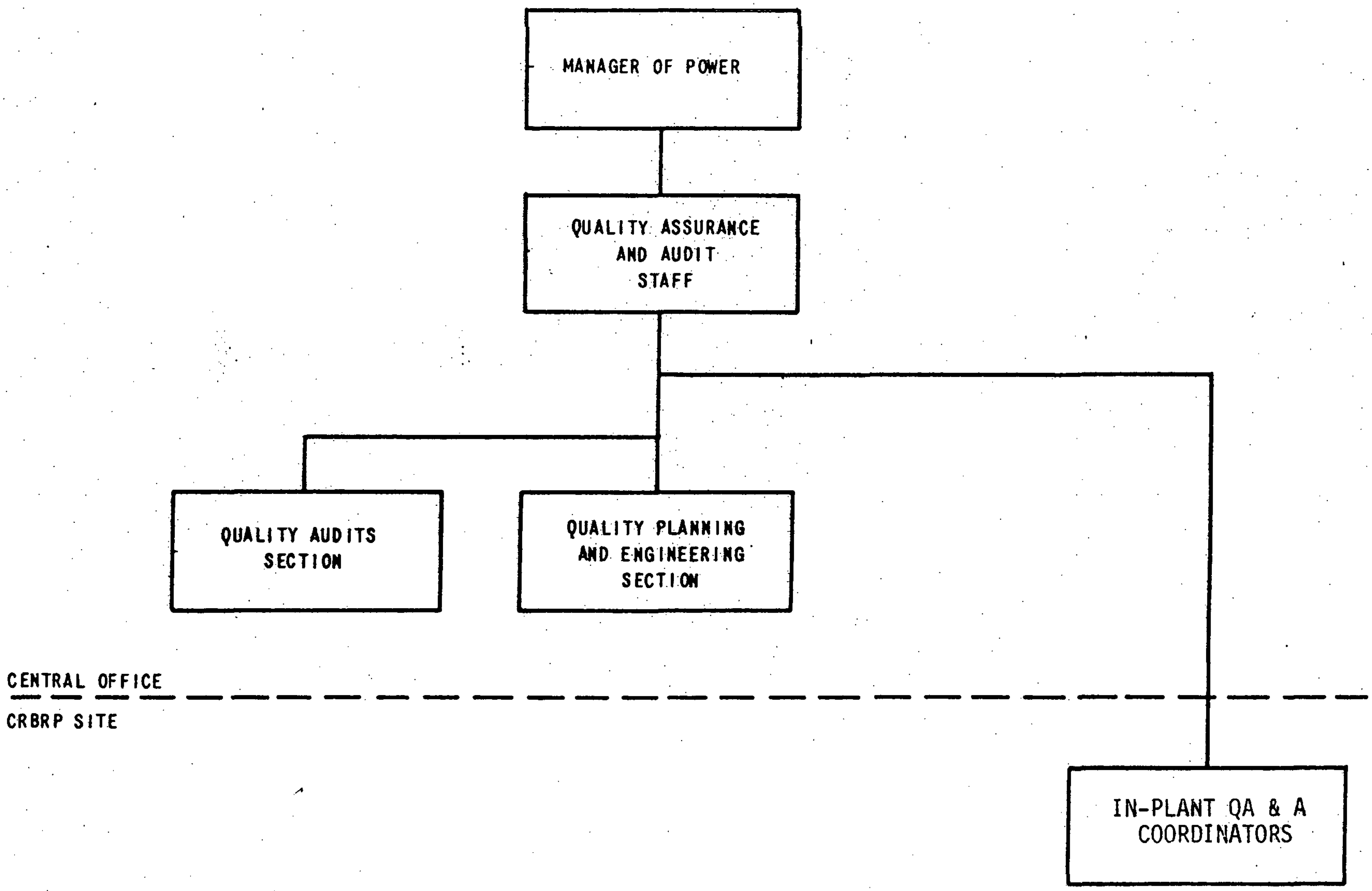

Figure 16.6-4. Quality Assurance and Audit Staff Organization

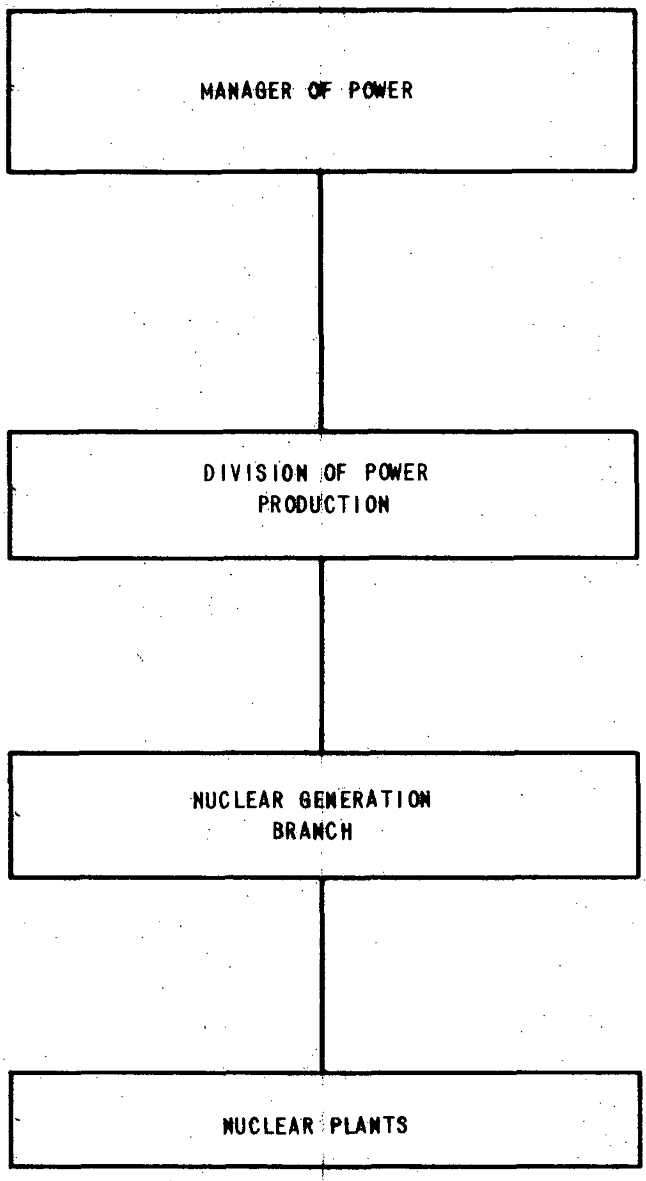

Figure 16.6-1. TVA Office of Power Organization Line of Responsibility for the CRBRP

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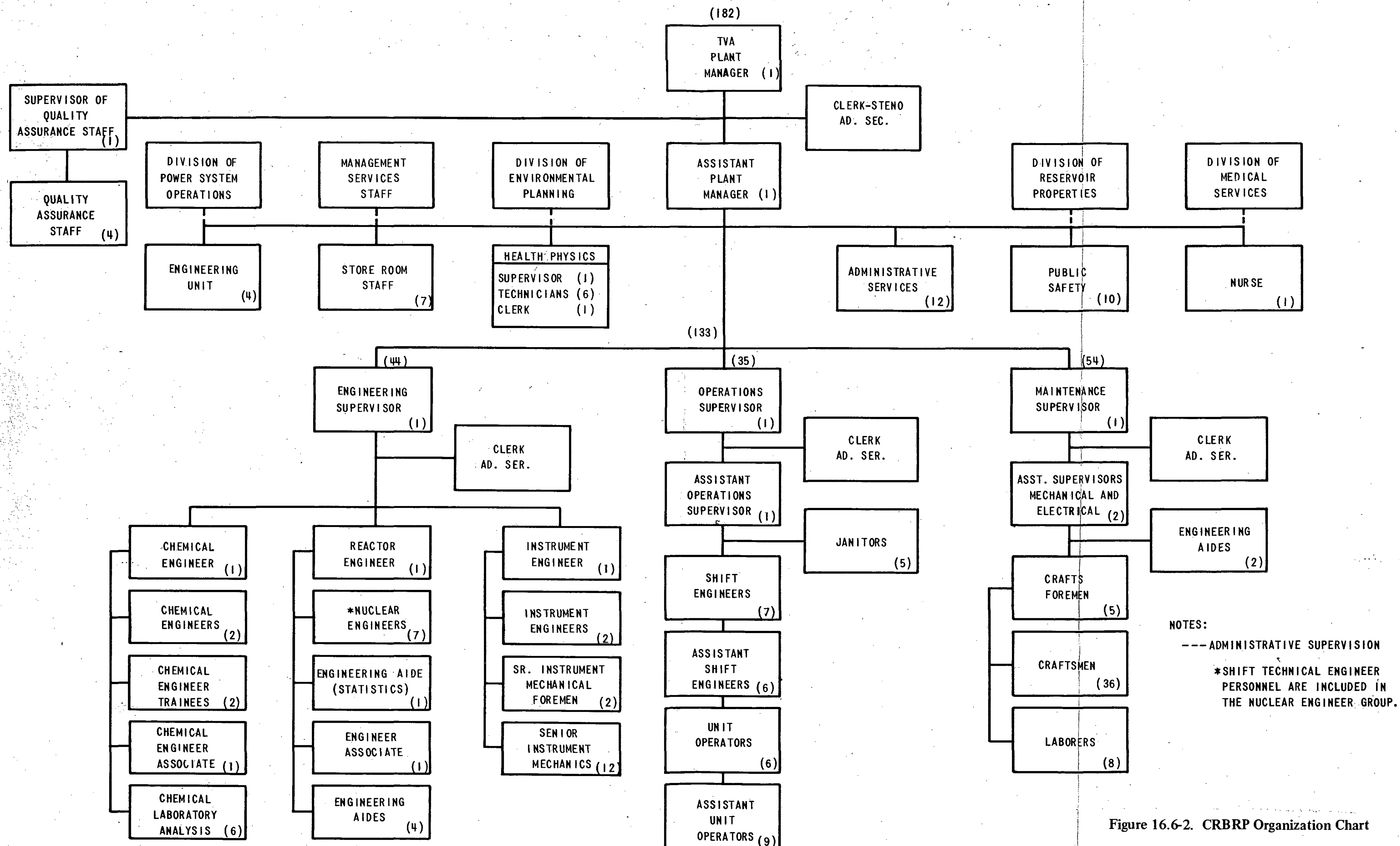

Figure 16.6-2. CRBRP Organization Chart

6738-2

6738-3

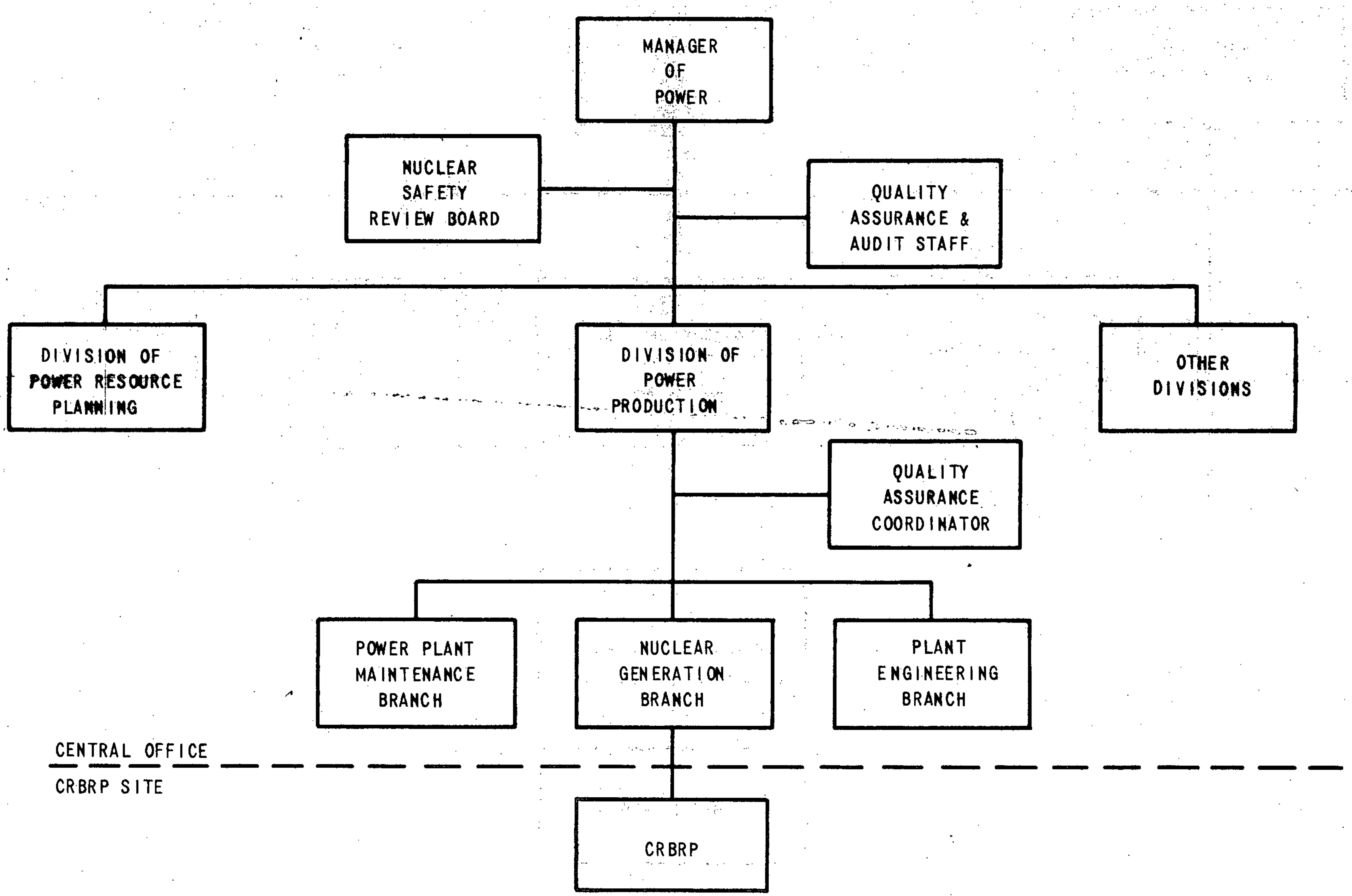

Figure 16.6-3. TVA Office of Power Organization