Department of Energy
Washington, D.C. 20545

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HQ:S:82:058

JUN 25 1982

Mr. Paul S. Check, Director
CRBR Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Check:

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION

Reference: Letter, P. S. Check to J. R. Longenecker, "CRBRP Request for Additional Information," dated April 30 and May 14, 1982

This letter formally responds to your request for additional information contained in the reference letters.

Enclosed are responses to Questions CS760.8, 12, 13, 14, 16, 19, 20, 22, 25, 26, 31, 33, 34, 42, 53, 54, 56, 92, 96, 111, 119, 124, 125, 128, 133, and 156; which will also be incorporated into the PSAR Amendment 69; scheduled for submittal later in July.

Sincerely,

John R. Longenecker

John R. Longenecker
Acting Director, Office of the
Clinch River Breeder Reactor
Plant Project
Office of Nuclear Energy

Enclosures

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Question CS760.8

PSAR page 4.2-15 and Table 15.1.2-2 on PSAR page 15.1-53, list the acceptance criterion for the "Extremely Unlikely Fault" of "Postulated Accident". The aim is to preserve coolable geometry, and the acceptance criterion is listed as coolant saturation; that is, no boiling. This appears to be a criterion that may not be adequate of itself for ensuring coolable geometry in the event of a reactivity insertion accident. While calculated cladding temperatures have a high degree of uncertainty, nevertheless gross fuel expulsion is known to occur in TREAT overpower tests at peak cladding temperatures as much as 200°K or more below coolant saturation, and in many cases 100°K or more below the cladding design temperature guideline. This behavior has now been confirmed in a slow transient (HEDL's W-2 test). The phenomena involved in these overpower events are exceptionally complex, but recent calculations indicate a fuel enthalpy limit with appropriate allowance for uncertainty may be an appropriate acceptance criterion to ensure coolable geometry for overpower conditions. A criterion of that sort is now used for the same purpose in LWR reactors.

The applicant is requested to comment on the adequacy of the no-boiling criterion as an acceptance criterion to ensure coolable geometry in the light of the TREAT test experience.

Response

The areas addressed in this question regarding the adequacy of the no-boiling criterion are similar to those expressed in Question CS490.23, and reference is made to the response already provided. It was shown in that response that there was considerable margin for preserving coolable geometry in the applicable design basis accident, although the event involved a conservatively postulated set of conditions including a large overpower transient, delayed shutdown and release of fission gas from all the pins in the affected assembly. The emphasis of Question CS760.8 is on the revelance of observations from TREAT and other tests to the no-sodium-boiling guideline. In general, the tests impose far more severe conditions on the fuel than would be applicable for the design basis events. Care should be taken in applying results from unterminated transient tests which exceed conditions typical of a design base to the terminated conditions in a design base. Undercooling tests have consistently shown that boiling precedes major damage to the cladding and the fuel. Overpower tests have demonstrated the satisfactory performance of CRBRP fuel within the design envelope. Where pin disruption has occurred in overpower before the coolant reaches saturation, the fuel linear power rating has far exceeded the predicted levels in CRBRP design basis events. For example, the test which is noted in the question, W-2, was an unterminated HCDA oriented test, and the peak linear power rating at the time of major cladding failure was 41.2 kw/ft (1351.5 W/cm). This should be compared with a value of approximately 16 kw/ft (525 W/cm) maximum, corresponding to 15% overpower at which a trip would have occurred in CRBRP for a similar reactivity insertion rate.

Question CS760.12

Since small ramp rates take a long time to develop, justify the assumption that the hot fuel rod is limiting, since the longer thermal inertia of the blanket rods does not apply here.

Response

For small ramp rates that take a long time to develop, the worst overpower condition that can exist would be operating at just under the 15% overpower trip level. If the power exceeds this value, core shutdown would be initiated through the plant protection system. The highest maximum cladding temperature during sustained operation at 15% overpower would be $1496^{\circ}F\ (3\sigma)/<1300^{\circ}F$ (nominal) and $1418^{\circ}F\ (3\sigma)/<1300^{\circ}F$ (nominal) for the fuel and blanket assemblies, respectively. With regard to fuel temperatures, all fuel and blanket rods have been designed to operate at this condition with no molten fuel as described in Section 4.4.3.3.6. After the overpower condition is terminated through scram, higher temperatures can be attained for the blanket rod cladding in the subsequent shutdown heat removal phase due (in part) to the large thermal inertia of the blanket rods alluded to in the question. This mechanism is described in Section 15.1.4.1 where a maximum blanket rod cladding temperature of $1587^{\circ}F\ (3\sigma)/<1300^{\circ}F$ (nominal) is reported (attained $\sim$47 seconds after shutdown).

Question CS760.13

Section 15.2.2.2 analyzes a 60¢ radial movement (stick slip) incident. The analysis does not distinguish between primary or secondary scram. (Only one temperature curve is given). Provide analysis for this transient, listing the appropriate primary and secondary trip functions.

Response

For a 60¢ step reactivity insertion the power increases in almost step fashion from 100% to over 200% as shown by Figure 15.2.3.3-3. Both the primary and secondary high power trip signals are significantly below the increased power level and thus, both trips would occur simultaneously. The table below summarizes results for the highest cladding temperature hot rod in FA-52 considering both primary and secondary scram (each separately).

		MAXIMUM TEMPERATURES (3σ)								
SHUTDOWN SYSTEM	REACTOR POWER AT TRIP	CLADDING			FUEL			COOLANT		
		A	B	C	A	B	C	A	B	C
Primary	115%	1491	0.63	1.6	4576	0.53	1.3	1417	0.63	1.6
Secondary	122%	1544	0.83	2.0	4752	0.63	1.7	1467	0.83	2.1

A - maximum 3σ hot spot temperature attained, $^{\circ}F$.
B - time to reach maximum temperature, sec.
C - length of time temperature is above initial steady state value, sec.

It should be noted that occurrence of a 60¢ step reactivity insertion combined with failure of the primary scram would be less probable than an extremely unlikely category event in which case the primary shutdown of a Safe Shutdown Earthquake (Section 15.2.3.3) would envelope the consequential core damage.

Question CS760.14

Reactivity insertions during startup must be more closely assessed. Under these conditions many of the PPS subsystems are bypassed and the PCS must be relied upon to mitigate the transient. In light of this, discuss the effects of PCS maloperation or operator error under these conditions.

For the cold sodium insertion event:

a. The transient is analysed using instantaneous core inlet temperature and flow rate changes. Shouldn't this be analyzed with more realistic (i.e., ramp type) changes in these conditions?

b. Although the loop transient time is 60s, the actual core inlet temperature will rise slowly. Therefore, shouldn't the transient be analyzed longer than 60s? (Especially for secondary PPS trip).

c. With a minimum Doppler coefficient, can you use BOC1 values for sodium density feedback coefficient?

d. What are the mechanical effects on pins due to cold sodium insertion?

Response

Although specific subsystems may be bypassed, sufficient protection in both the primary and secondary scram systems still exist to ensure that damage limits are not exceeded. Specifically, positive flux to delayed flux, positive modified nuclear rate, flux to pressure (automatically reinstated above 15% power), flux to total flow, and startup flux subsystems would all trip for unacceptable positive reactivity insertions occurring during startup caused by PCS maloperation or operator error.

With regard to the cold sodium insertion event, the following are the responses to the item-by-item questions:

a. The controlling mechanism of this analysis is the positive reactivity from Doppler feedback which occurs when the fuel is initially cooled by the lower sodium inlet temperature. Since it was not known "a priori" that this event would be benign, it was assumed that conditions which would result in the most rapid power increase would be limiting. Thus, a step change in the inlet coolant temperature was conservatively used in the analysis. A slower change of inlet temperature would not increase the transient power level but would decrease the rate at which the power increases.

b. It is estimated that the reactor inlet temperature would increase at a rate of less than 20°F in 300 seconds which would not cause any significant increase in core structural damage. This increase of inlet temperature would attenuate the equilibrium power attained from that shown in the analyses (Figure 15.2.3.1-1). If the analyses were carried out until exact equilibrium is reached, the maximum temperatures would still be less than those shown for full power steady state operation.

c. Yes, but the transient would be less severe because less positive Doppler feedback would result (see Item a).

d. The worst case hot rod cooldown rates shown by Figure 15.2.3.1-3 are very similar to those experienced during a normal scram and insignificant damage is incurred. This is due to cladding being very thin which mitigates thermal shock damage.

Question CS760.16

On page 15.2-2a of the PSAR, the first sentence of the fourth paragraph states, "The first two of the above restrictions are obvious". Subsequently, a third restriction is alluded to . It is not at all clear what the referenced restrictions are. Please clarify.

Response

The treatment of Fuel-Cladding Mechanical Interaction (FCMI) was described in response to Question CS490.10 and at the May 12, 1982 meeting with NRC. Revisions to Section 15.2 of the PSAR will be made to clarify the treatment of FCMI by August 15, 1982.

Question CS760.19

In Section 15.2, present or reference the fuel and cladding temperature histories for the worst fuel and blanket rod during the U2b event. Also, present or reference a synopsis of this event and its consequences, as was done for the other overpower transients in this section.

Response

The U2b event is an enveloping condition for various types of accidents that can be postulated which insert positive reactivity to the core (either in steps or ramps). This event is described in PSAR Appendix B, Section B.1.2.2.2. It should be kept in mind that this event assumes the failure of the rod block at 103% power in either the manual or automatic modes of the PCS. The evaluation results in an analysis of core temperatures at 15% sustained overpower for a period of 300 seconds. If the power increase would be any larger than this magnitude the reactor would automatically be shut down due to scram from an overpower trip signal of the plant protection system. Equilibrium temperatures indicative of steady-state operation at 115% power are reached within the 300 second hold even for the large diameter blanket size rods.

As indicated in response to Question CS760.12, the maximum cladding temperature during the sustained operation at 15% overpower would be 1496°F $(3\sigma)/<1300^{\circ}$F (nominal) and 1418°F $(3\sigma)/<1300^{\circ}$F (nominal) for the fuel and blanket assemblies, respectively. With regard to fuel temperatures, all fuel and blanket rods have been designed to operate at this condition with no molten fuel as described in Section 4.4.3.3.1. After the overpower condition is terminated through scram, higher temperatures can be attained for the blanket rod cladding in the subsequent undercooling phase due to stored heat and decay heat effects as described in Section 15.1.4.1. In this post-shutdown period the maximum blanket hot rod cladding temperature of 1587°F $(3\sigma)/<1300^{\circ}$F (nominal) can be attained after $\sim$47 seconds after shutdown.

Question CS760.20

The design guideline for cladding temperature for anticipated events is listed in Table 15.1.2-2 as 1500^{o}F. However, Table 15.2-2 lists temperature criteria (presumably cladding) of 1450^{o}F and 1400^{o}F (blanket rod reactivity insertion). Please explain the relation between the limits listed in these two tables. Please also provide in detail the basis for these two sets of guidelines.

Response

Table 15.1.2-2 does provide design guidelines which includes the 1500^{o}F cladding temperature for both fuel and blanket for anticipated events. The basis for the guidelines was explained in response to Question CS490.21 and addressed at the February 25, 1982 meeting with NRC.

There is no Table 15.2-2 in the PSAR. Originally, in earlier revisions to the PSAR, there were such conservative cladding temperature limits as those described. However, analyses consistent with those now appearing in Section 4.2 of the present PSAR showed the limits to be overly conservative and thus, is the basis for the 1500^{o}F guideline now appearing in Table 15.1.2-2.

Question CS760.22

Discuss how the changes to a heterogeneous core affect reactivity effects for the operating basis (OBE) and safe shutdown (SSE) earthquakes.

Response

The change to a heterogeneous core has no significant effect on reactivity changes for the operating basis (OBE) and safe shutdown (SSE) earthquakes. This is because parameters which control the response, the radial reactivity worths and mechanical dimensions, are very similar. Uncertainties in the core mechanical response to the earthquake and core motion reactivity worth factors will mask any reactivity differences between the homogeneous and heterogeneous core designs.

Question CS760.25

Provide the basis for the assembly power distribution numbers.

Response

The CRBRP power distribution calculations and uncertainties are discussed in depth in Section 4.3.2.2 of the CRBRP PSAR.

Question CS760.26

What is the effective bypass inertia (L/A) eff.?

Response

The total reactor flow during a flow coastdown from full flow is governed by the inertia in the pumps (the principal contributor) as well as the fluid inertia. The contribution of the fluid inertia in the reactor is small. For example, the L/A for the flow path between the pump and the reactor vessel inlet is 94.4 $ft.^{-1}$ while that for the fuel region is <3 $ft.^{-1}$. The effective L/A for the "bypass" region would be even smaller because of the lower L/A value and the small fraction of total flow associated with the bypass region. Flow redistribution between the four parallel flow paths in the DEMO model (fuel, inner blanket, radial blanket and "bypass") occurs when the total flow derivative is very small and inertial effects are negligible. Thus, the effective bypass inertia is ignored (i.e., set to zero) and the computation of the redistribution of flow between the four parallel flow paths within the reactor is accomplished by balancing the flows between the four paths such that each path has an identical pressure drop. The pressure drop for the bypass is the sum of gravitational and form pressure losses.

Question CS760.31

The most notable change in the new natural circulation analysis (CRBRP-ARD-0308, Feb. 1982) is the change in pump coastdown time from 55 to 120 seconds. This change is attributed to a substantial decrease in low speed frictional torque seen in prototype tests. Since the previous estimates based on design requirements have been replaced by the new best estimate torque, will all future pumps be required to conform to this behavior (i.e., will any pump with a higher frictional torque be rejected)?

Response

The coastdown requirements specified for the CRBRP primary sodium pump/pump drive motors were stated in Reference CS760.31-1. These requirements were established in 1975 and were based on the results of analyses which examined the effects of various coastdowns on the plant's natural circulation capability as well as transient effects on structures such as the UIS. These requirements provided a basis for the design of the pump and pump drive system. Since the design of the pumps as well as the drive motors (and associated bearing and seal assemblies) is now completed, analyses are based on the actual coastdown characteristics shown by tests of the pump/pump drive prototype.

The availability of prototype pump water test data has made it possible to develop and refine predicted pump characteristics for opertion in sodium. Correlations for pump head-flow and pumping torque characteristics were developed from water test head-flow and efficiency data. A value for pump inertia was obtained by using water test coastdown speed vs. time data, and approximation for loss torque at high pump speeds and the equation of motion for the pump. A trial-and-error procedure was then used to select a set of coefficients for frictional torque correlations at low pump speeds that most closely matched the measured results of water test coastdown runs. This method ensures that inaccuracies in the pumping torque correlation due to departures from similarity at low speeds will be absorbed in the frictional torque correlation.

The prototype pump coastdown characteristics developed from water test results represent an accurate model of sodium pump performance in CRBRP. Coastdown tests conducted during the Maximum Isothermal System Tests (MIST) for FFTF primary and secondary main coolant pumps showed only slight variations in pump-to-pump coastdowns. For example, at 30 seconds after trip, the respective pump speeds in primary loops 1, 2 and 3 were 155 RPM, 156 RPM and 155 RPM. The greatest differences between measured pump speeds occurred below 4% speed immediately prior to the pumps stopping. The variation in coastdown time for the primary pumps was approximately 15%, while for the secondary pumps it was 12%. Since a similar correspondence in coastdown performance can be expected for CRBRP sodium pumps, variations in frictional torque large enough to warrant rejection are unlikely to occur. If, however, pump coastdown characteristics in the plant pumps result in pump coastdowns that occur earlier than used in the analyses, the Project will demonstrate adequate natural circulation conditions still exist or provide modifications to the pump coastdown characteristics to achieve acceptable natural circulation conditions.

Reference

CS760.31-1 R. R. Lowrie and W. J. Severson, "A Preliminary Evaluation of the CRBRP Natural Circulation Decay Heat Removal Capability", CRBRP-ARD-0132, November, 1977.

Question CS760.33

In the revised natural circulation report (CRBRP-ARD-0308, February, 82) by Severson, et al., it is stated that no credit is "taken for inter- and intra-assembly flow and heat redistribution." This is consistent with the DEMO-REV4 which has a fixed flow fraction to each group of assemblies. However, it is also stated that "the code calculates flow redistribution between the four regions." This appears to be a major change from the previous conservative approach.

What would the hot-spot temperature be if the fixed flow fraction were maintained throughout the transient?

Response

A four region reactor model which provides for flow redistribution between the fuel assemblies, inner-blanket assemblies, outer-blanket assemblies and bypass channel has been added to DEMO since the publication of DEMO-REV4 (WARD-D-0005, REV 4, January, 1976). Flow distribution between the four flow paths is computed by equating the pressure drop for each path. Flow redistribution is a more physically accurate approximation to a two dimensional reactor flow model than the fixed flow fraction model. This flow redistribution model was used in the DEMO analysis (which generated the total reactor flow rate vs. time) reported in CRBRP-ARD-0308.

The DEMO analysis employing the four region reactor model which accounts for redistribution between the above mentioned regions results in slightly lower reactor flows than that which would be computed by the earlier fixed flow fraction model. The reason is that the fuel assembly flow which establishes the plenum to plenum ΔP in the fixed flow fraction version of DEMO will be higher in the redistribution model thus increasing the dynamic pressure losses and at the same time reducing the thermal head. Thus, the redistribution model produces a conservatively low total reactor flow.

The highest core temperatures at the hottest locations of the hottest rods for the fuel, inner-blanket and radial blanket assemblies were then computed using FORE-2M based on fixed fractions (equal to their initial fractions) of the total reactor flow. In this hot channel analysis, no credit was taken for inter- and intra-assembly flow and heat redistribution.

No analysis of hot-spot temperatures using a forced flow fraction has been performed which could be directly compared with the valves reported in CRBRP-ARD-0308, however, because of the increased flow associated with using a fixed flow fraction the hot-spot temperatures would be lower for such a case and therefore the valves reported in CRBRP-ARD-0308 can be considered bounding for this aspect.

Question CS760.34

There is a built-in time delay in reverting from perfect mixing in the upper plenum to stratified flow. What is the basis for the specific delay and what is the effect if no delay is added?

Response

The DEMO upper plenum uses two distinct modes of mixing, the fully mixed and stratified mode, to simulate the mixing in the reactor upper plenum. At the initiation of a transient, perfect mixing was assumed in the outlet plenum. The transition from perfect mixing to stratified flow mode depends on jet height. A jet height of 20 ft. was assumed in the analysis reported in CRBRP-ARD-0308. The stratified flow portioin of the model begins with the filling of the lower region of the outlet plenum by the cold sodium existing from the top of the chimney. During this time and until the time the hot/cold sodium interface reaches the bottom of the outlet nozzle, the reactor vessel outlet temperature is set equal to the hot outlet plenum temperature at the start of the stratification. This time period is the delay referred to in the above question. When the hot/cold sodium interface rises above the bottom of the outlet nozzle, the outlet nozzle temperature starts to decrease as a function of the cold flow area in the outlet nozzle. The cold flow area is that portion of the outlet nozzle covered by the cold fluid. The remaining outlet nozzle area is assumed to be covered by the hot plenum fluid. The outlet nozzle temperature is calculated by assuming perfect mixing of the hot and cold sodium. The resulting temperature is, therefore, given by:

$$T_{VO} = T_{CH} + (T_{HL} - T_{CH})\ \mathrm{EXP}\ (-(t - t_f)/\tau) \qquad (1)$$

where T_{VO} is the vessel outlet temperature

T_{CH} is the mixed mean temperature of fluid entering the upper plenum from the UIS chimney

T_{HL} is the hot sodium temperature in the upper plenum

t is the transient time

t_f is the time when the stratified cold sodium rises to the bottom of the nozzle

τ is a convective time constant relating to the rate of change of the hot/cold sodium interface from the bottom to the top of the outlet nozzle.

When the cold fluid rises to the top of the outlet nozzle, the flow through the nozzle and its temperature will be the same as the cold sodium. The level of the hot/cold interface is assumed to be constant and the temperature above the interface is hot and below the interface is cold.

It is clear from this model description that a delay between switching from fully mixed to stratified mode of calculation is required by virtue of the physical dimension of the plenum and the fluid transport time. It should be pointed out that the model assumptions used are quite conservative. Credit for heat transfer to the cold fluid from both the hot plenum metal and the plenum sodium was not accounted for in this stratified mode plenum calculation. In addition, the values of t_f and τ used in the analysis reported in the CRBRP-ARD-0308 report were also more conservative than the actual data. This was done deliberately to allow the cold sodium entering the plenum to leave the outlet nozzle earlier. Since the sodium exiting the reactor plenum rises abruptly into the vertical run of the primary hot leg pipe, this would result in a lower overall primary loop thermal head. This lower thermal head in turn, would result in a more conservative estimate of the primary loop flow.

The effect of neglecting the delay totally would mean that the cold sodium entering the plenum will appear instantaneously at the reactor vessel outlet nozzle. This is not only overly conservative, it is physically impossible. The two mode model described above is both adequate and conservative for the analysis of a natural circulation transient event.

Question CS760.42

In Section 15.4.1.1.5 (page 15.4-12), it is stated that "...cladding defects (in the fuel zone region) of 0.1% of the fuel rods...would result in an end-of-life plutonium concentration of 0.1 ppm in the primary system sodium." Please provide the reference from which these data were taken.

Response

The design basis limit for plutonium release to the primary coolant is 100 ppb (0.1 ppm). This has been related to a continuous 0.1% failure rate of the fuel rods for 30 years based on a Pu escape rate coefficient of 9.71×10^{14} atoms/cm^2 sec if each of the failures correspond to a 0.03 inch hole.

Question CS760.53

This section on extremely unlikely events treats only the first case on the core component pot. Have all of the other cases been treated and what are the results?

Response

PSAR Section 15.7.3.1 discusses the event of leakage of the sodium from a core component pot (CCP) suspended in the EVTM, resulting in overheating of a contained fuel assembly. A single event is considered, however, extensions of it beyond its expected termination are considered. The event would result in partial melting of the cladding and fuel assembly ducts but the columns of fuel pellets would stay intact. Hypothesized extensions of this event discussed in this section of the PSAR are the collapse of the pellets to the bottom of the core assembly and the redistribution of the pellets outside the fuel assembly duct in the bottom of the CCP. In all cases, the core assembly materials would be contained within the CCP. The only material which would escape from the CCP would be fission products which are volatile before the maximum temperature is reached by the fuel. While this temperature is high and the release would include many elements, most would be deposited on the colder interior walls of the EVTM.

The release of materials outside the EVTM pressure boundary is through EVTM elastomer seals which are more than 6 ft below the bottom of the suspended CCP. The highest seal temperature for the event (including the hypothesized extensions) would be 260°F. The worst case release would be of elements which are volatile above this temperature and this is the enveloping release for all cases described in PSAR Section 15.7.3.1. Is it the same release as for the unlikely event described in PSAR Section 15.5.2.3, single fuel assembly cladding failure and subsequent fission gas release during refueling (in the EVTM). That event is referenced in PSAR Section 15.7.3.1.3 for the offsite exposure from the extremely unlikely CCP leak event. (It should be noted that the treatment of the isotopes which are volatile above 200°F is described in the response to NRC Question 001.212 (15.5.2.3.2). The 200°F temperature considered there and in PSAR Section 15.5.2.3 is equivalent to the 260°F real temperature in PSAR Section 15.7.3.1, since the only isotope with a melting point between the two temperatures, iodine (see PSAR Table 15.7.3.1-3), is included among the elements considered).

Question CS760.54

Regarding Section 15.7.1.3 on sodium leaks. Sometimes (as in the Phenix Reactor), the IHS springs a leak as a result of the strains which occur with shutdown and startup. If the primary loop pumps come up before the secondary pump loops, then it may be possible for contaminated primary sodium to be driven into the secondary loop. Please discuss this possibility with an undetected IHX leak.

Response

The IHTS is designed to insure the pressure in the IHTS is always higher than the PHTS by at least 10 psid. IHTS design also includes a low IHTS/PHTS Δp alarm on the Main Control Panel (MCP) to alert the operator of a problem.

It is possible to bring up the Primary Heat Transport System (PHTS) pumps before the Intermediate Heat Transport System (IHTS) pumps and achieve a pressure in the PHTS higher than the IHTS, however, CRBRP operating procedures would have to be ignored or violated and the low IHTS/PHTS Δp alarm would have to fail to alarm or be ignored when it is received.

Assuming normal IHTS pressure when the PHTS pumps are started, the PHTS flow would have to be increased to >85% flow before the PHTS pressure would exceed IHTS pressure.

Therefore, if one assumes an undetected leak in the IHX combined with several operator errors and alarm failures, it is possible to get primary sodium into IHTS system, although it is considered a very unlikely event.

Assuming the PHTS to IHTS leak did occur, the IHTS boundary would prevent any release of radioactivity to the atmosphere and the health and safety of the public would not be endangered from the event.

Question CS760.56

The PSAR claims, in Section 15.7.1.6 regarding NaK spills in the EVST system, that the NaK will be non-radioactive. It is possible, however, that some radioactivity could get into this NaK by such sources as the 1% failed fuel or carryover sodium from the fuel transfer. The cover gas could also become contaminated by leakage of fission gas from failed spent fuel rods.

a. Does CRBR have instrumentation in the EVST to detect radioactivity in the cover gas and in the NaK?

b. Does the EVST have any instrumentation for local detection of activity, temperature, or local boiling within a possibly partially blocked subassembly?

Response

The NaK will be non-radioactive because it is kept separate from the EVST sodium which is the primary coolant. EVST heat is transferred from sodium to NaK in a heat exchanger (see PSAR Section 9.1.3.1). The EVST sodium will contain radioactivity from carryover of primary sodium during refueling and perhaps from fuel assembly fission gas releases in the EVST. The activity will not be transferred to the NaK because of the sodium and NaK separation. The NaK is kept at a higher pressure than the sodium at the heat exchanger to prevent contamination of the NaK in the event of a leak (see PSAR Section 9.1.3.1.3). The plant design provides the capability to detect NaK leaks into the EVST sodium so that NaK levels and pressures could not decrease to the point where radioactivity could leak from the EVST sodium into the NaK coolant.

The plant design provides the capability to detect radioactivity in the EVST cover gas and sodium coolant. The cover gas activity is continuously monitored by maintaining a flow of cover gas through a radiation detector. The sodium activity is monitored by sampling the sodium periodically for laboratory analysis of its radioactivity concentration.

Individual core assemblies in the EVST are not instrumented because the conditions are not severe enough to require it. Core assembly power levels (20 kW maximum power per assembly versus several MW in the reactor) and cladding temperatures (~600°F versus ~1100°F in the reactor) are both low compared to reactor operating conditions, and the cooling method is such that the effect of partial blockage of a subassembly is minimized. The flow of EVST sodium coolant is outside the core component pots (CCPs) in which core assemblies are stored. Heat removal by this coolant is from the CCP walls. There is no significant mixing between this sodium and the sodium in a CCP. Core assembly decay heat is transferred to the CCP walls from the fuel rods by conduction and by convection of the sodium in the CCP. Since the driving force for the convection flow is the fuel temperature, any partial blockage causing a temperature rise would be self-correcting by increasing the flow through the assembly rather than diverting flow through a lower resistance assembly as would occur with forced flow cooling.

Question CS760.92

Frequent changes in power level (e.g., following plant trip) can entail swings of up to several hundred degrees in the coolant temperatures. Thus, large thermal stresses may appear in the reactor vessel, coolant piping, or other components, which may eventually threaten the system integrity. What are the methods and models presently used to determine these temperature swings? How are they factored in to provide assurance that they conservatively cover the entire duty cycle of the plant?

Response

The basis for NSSS component structural evaluations is the plant Design Duty Cycle. The transients specified for the structural evaluation of plant components are generally the results of the DEMO code output. The DEMO code is described in Appendix A.21 of the PSAR.

The plant Design Duty Cycle transient events were selected to be representative of operating conditions, which are considered to occur during plant operation and which are sufficiently severe or frequent to be of possible significance to components. These transients are based on a conservative envelope of plant operation and were developed primarily for use in component stress analyses. The events, as well as their associated frequencies, are based on LWR, FFTF, and fossil plant experience; system and component reliability estimates; and engineering judgment. The description of the transient events that are used in CRBRP component analyses, and their assigned frequencies are presented in Appendix B of the PSAR.

The analysis of each Design Duty Cycle event is based on conservatively biased parameters for each system and/or component using the DEMO computer code. The analysis of each event was performed such that the rates of change and total range of temperature change were conservatively computed for each run of piping or component. The rated power initial conditions included hot and cold leg temperatures biased upward 20°F to account for temperature measurement and cointrol uncertainties. For some transients an alternate set of initial conditions were used which employed hot to cold leg temperature differences of 300°F for the primary system and 340°F for the intermediate systems (with a total temperature difference of 390°F between the PHTS hot leg and IHTS cold leg). In addition, some of the plants sensible heat (piping heat capacity, for example) was neglected, energy delivery rates from reactor to SGS were maximized by conservatively high pony motor flows and reactor upper plenum stratification was included or not included to assure conservatism. Other plant parameters such as system pressure drop, pump inertia, pump loss torques, decay heat, pony motor speed, PPS actuation time, rod reactivity and delay and valve stroke time and capacity are individually biased in the most conservative direction. These conservatively developed duty cycle transients are then included in all NSSS component/piping histograms as discussed below.

All normal events (and frequencies) are applied to each component in the system at their specified frequency. Upset events are grouped into a smaller set of umbrella events (typically 10 to 13). Less severe transients are combined with more severe transients by increasing the event frequency of the umbrella event, such that the frequency of the umbrella event equals the sum of the frequency of that event and the frequency of each event umbrellaed under it. Emergency events are incorporated by determining the most significant event and applying it five times (evenly spaced in time) plus two consecutive occurrences of the most severe event or combination of events. All events that are defined as a faulted event for a component are included in that component evaluation.

These duty cycle transients are in general applied approximately evenly in time over the thirty year life of the CRBRP, divided into ten three year periods. Worst case sequencing is assumed within these periods consistent with physical possibility. This combination of events results in a conservative histogram for component evaluation.

In conclusion, due to the combined use of:

- o design thermal transients based on worst case plant conditions for the component under evaluation for that event;
- o conservative estimates of events and their associated frequencies;
- o conservative umbrellaing techniques; and
- o applying a worst case histogram.

the components have been evaluated against temperature swings that will conservatively cover the transients that are expected to occur in the plant.

Question CS760.96

The recirculation pump is described as single speed, yet will experience varying mass flow rates at different power levels and will go through varying speeds while coasting down after a trip. How does the pump head vary with flow rate and speed, i.e., what does the homologeous pump curve look like?

Response

The pump curves are attached as Figures QCS760.96-1, 2. The curves were established by actual vendor pump testing. The recirculation pump is a single speed unit with a Design Speed of 1794 rpm, a Design Flow of 5920 GPM, and a Design Dynamic Head of 397 ft. This design point is marked on the attached curve. The pump conditions given in the "Coast Down Curve" are measured on the suction side of the pump.

Figure QCS760.96-1

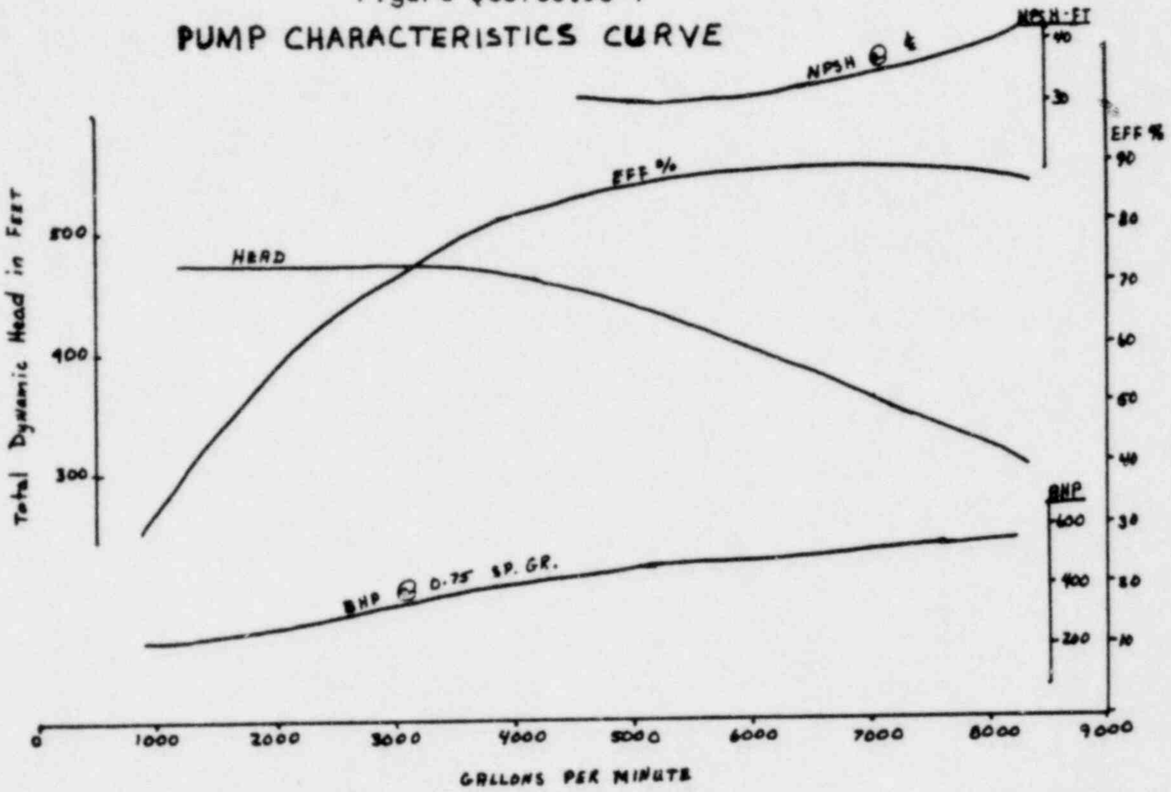
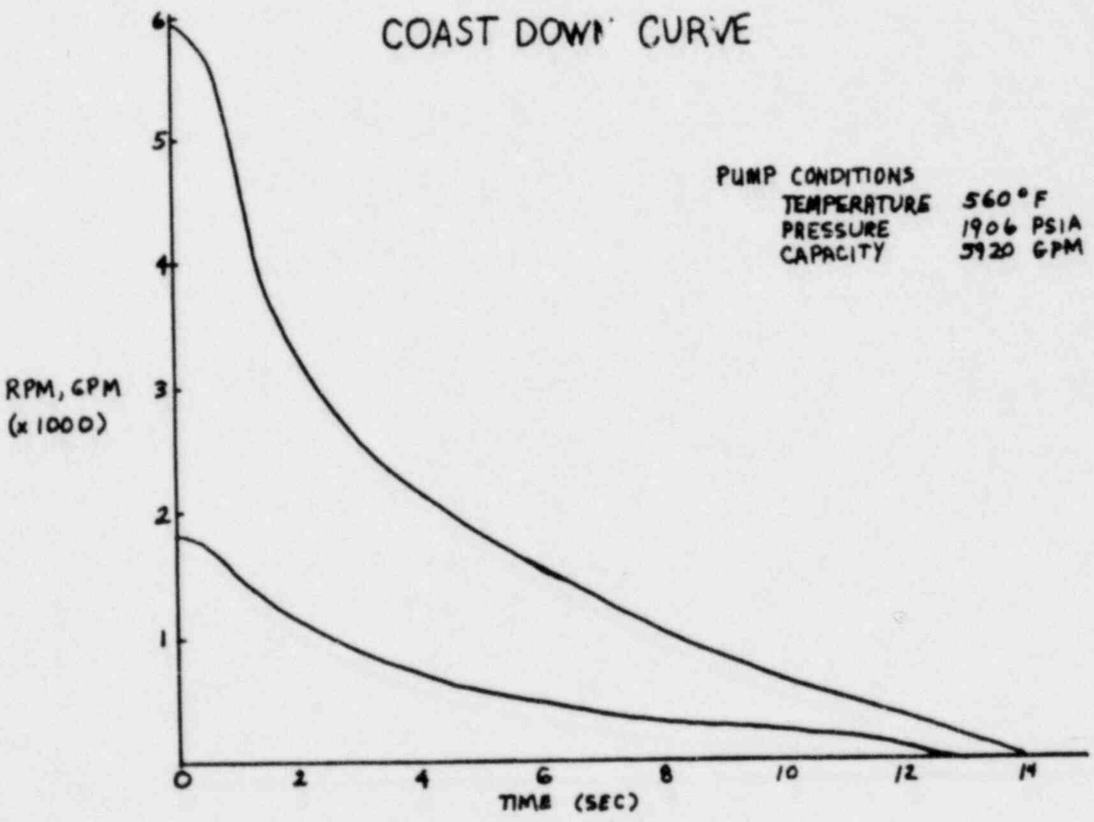


Figure QCS760.96-2



Question CS760.111

What is the flow area through the SGAHRS vent valves when they are fully open?

Response

The flow area of a fully open SGAHRS vent valve is 9.01 square inches.

Question CS760.119

In Section 5.6.2.3.2 of the PSAR, it is stated that the DHRS is not designed to provide heat removal by natural circulation. Since the overflow concept requires pumping in order to function within its design objectives please provide a discussion of the following:

a. How is the DHRS diverse for electrical power (onsite and offsite failure)?

b. Other potential common mode failures.

Response

DHRS is not designed, nor intended to be, diverse for electrical onsite and offsite failures. The diversity provided in the plant for onsite and offsite electrical failure is the natural circulation capability through the the PHTS/IHTS/SGS/SGAHRS. DHRS provides diversity for those failures which could disrupt heat removal through the IHTS and steam generator system.

Question CS760.124

In reviewing pump coastdowns how were effects of extended coastdown considered? How were differences between "identical" pumps considered in your analysis?

Response

It is assumed that the above question relates to a natural circulation event.

Extended coastdowns for main coolant pumps beyond those presently used in the DEMO plant simulation code enhance the natural circulation decay heat removal capability of CRBRP. The critical period for the natural circulation decay heat removal mode occurs shortly after the primary and intermediate pumps have stopped. At that time, thermal driving heads necessary to promote adequate flows are required to prevent resulting core temperatures from exceeding acceptable limits. Maximum core temperatures reached during the natural circulation transient are largely dependent upon the decay power. Extended pump coastdowns allow time for reductions in both decay heat and reactor sensible heat and consequently provide greater margins to boiling in the core. The analysis of the natural circulation event used pump coastdown characteristics developed from prototype pump water tests.

Differences between "identical" pumps were not considered in analysis of the natural circulation event. Differences in pump-to-pump performance during plant operation are not expected to be significant enough to justify inclusion of separate models for individual pumps. Further discussion of this point is provided in the response to Question CS760.31.

Question CS760.125

In reviewing the progression of uncertainties how were the following items considered:

- o Pressure drop
  - core
  - pump
  - piping
  - IHX
  - valves
- o Flow Coastdown
  - pump inertia
  - pump friction
  - differences between "identical" pumps
- o Stratification
  - upper plenum
  - piping
- o Intra-assembly heat and flow redistribution
- o Inter-assembly flow redistribution
- o Heat losses to outside
- o Bypass flow
- o Decay heat.

Response

The individual data sources for the current natural circulation assessment are discussed in the response to Question CS760.28.

In developing the transient response of the CRBRP, each of the Design Duty Cycle events, has a set of parameters individually chosen at their limits and a series of models individually incorporated or deleted to the DEMO computer code that are appropriate for that duty cycle event. The individual uncertainties requested are discussed in the following table.

Parameter	Consideration
o Pressure drop	
- core	Maximum or minimum used as required to assure conservatism.
- piping	Included with IHX in analysis.
- pump	Head flow characteristic assumed at the minimum for all analyses and locked rotor resistance assumed to be at the maximum.
- IHX	Piping and IHX pressure drops combined and chosen as maximum or minimum as required to assure conservatism.
- Valves	Maximum or minimum used as required to assure conservatism.
o Flow coastdown	
- pump inertia	Chosen consistent with the maximum or minimum specified requirement for the pumps.
- pump friction	Chosen consistent with the maximum or minimum specified for the pumps.
- differences between "identical pumps"	None - See response to Question CS760.124.
o Stratification	
- upper plenum	Fully mixed or stratified model used to provide the most severe transient.
- piping	See response to Question CS760.28.
o Intra-assembly heat and flow redistribution	No credit taken.
o Inter-assembly flow redistribution	No credit taken.
o Heat losses to outside	No credit taken.
o Bypass flow	No uncertainty applied.
o Decay heat	Maximum or minimum chosen to provide most severe transient.

Question CS760.128

During the descent from 10% power, what are the safeguards to prevent unacceptably high usage of feedwater from the protected water storage tank?

Response

The Auxiliary Feedwater System (AFWS) is not used during normal descent from 10% power; therefore, no water is drawn from the protected water storage tank (PWST). The AFWS is only operated when the Steam Generator Auxiliary Heat Removal System (SGAHRS) is initiated. The PWST water use is discussed in PSAR Section 5.6.1.3.9.

Question CS760.133

Very little basis is given for the assumed frequency of events. Please categorize the frequency as to source (in order of preference).

a. Commercial reactor experience

b. Test reactor experience

c. Other data

d. Engineering judgment

Response

This response is prepared assuming the question refers to Table 5.7-1, "Preliminary Summary at Heat Transport System Design Transient".

The frequency for the overall plant duty cycle events was initially determined from a review of available commercial reactor experience and specific meetings with commerical reactor vendors. The selection of specific duty cycle events and the allocation of frequencies to the specific events was developed based on engineering judgment and an understanding of the design differences between an LWR and an LMFBR. The structural evaluation of the effects of each individual duty cycle event on each reactor plant component was analyzed by grouping the duty cycle events for each component into a single transient event (umbrella) which is conservatively representative of the group with the frequency of the entire group. Since the individual transients have different effects on different components, the umbrella transients and the transients grouped under that umbrella are developed differently for each component. Different frequencies are therefore assigned to each umbrella transient for each component. The selection of umbrella transients, and the groupings under each umbrella transient, was based on preliminary analysis of the effects (temperature, pressure, and resultant stresses) of each duty cycle transient on each component. This engineering effort resulted in the frequencies shown in Table 5.7-1 of the PSAR for each major component of the Heat Transport System.

Question CS760.156

Discuss the leak test method used following replacement of the equipment hatch. How were the permissible leak rates determined?

Response

The leak test method to be used for periodic testing of the equipment hatch after completion of each refueling will be local pneumatic pressurization of the dual compressible hatch seals utilizing the in-place test connection. Determination of the actual leak rate will be by measuring the pressure decay for a prescribed time duration.

In the case of anticipated actual replacement of the equipment hatch, special installation checks such as dye and chalk tests for alignment verification coupled with pneumatic pressurization of the dual seals will be performed.

Permissible leak rates to be finalized in early 1983 will be consistent with the acceptance criteria for type B tests as delineated in 10CFR50, Appendix J.

It should be noted that the equipment hatch is always closed during all Reactor Plant operations and is only opened for refueling and/or maintenance activities.