Department of Energy
Washington, D.C. 20545

Docket No. 50-537
HQ:S:82:059

JUN 29 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, 1982

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

Enclosed are responses to Questions CS 760.7, 9, 38, 57, 58, 86, 112, and 136 which will also be incorporated into the PSAR Amendment 69; scheduled for submittal 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

cc: Service List
Standard Distribution
Licensing Distribution

D001

Question CS760.7

Are there any items that should be addressed or have been included in the CRBR design resulting from the lessons learned from the TMI accident?

Response

The lessons learned from TMI have been carefully reviewed and appropriate actions have been initiated. Lessons learned from the TMI accident being addressed for the CP, and their inclusion in the CRBRP design, are discussed in Appendix H to the PSAR.

Question CS760.9

Table 15.1.3-1, page 15.1-94 in the PSAR, lists PPS trip levels and/or equations. Several of the terms are not defined, and no indication of units is given for those that are defined. Please supply definitions for all terms including units. Please also indicate the level for each that leads to the latest trip. Provide the impact of the new settings on the frequency of occurrence of events. Do these changes impact the duty cycles used for plant design?

Response

The variables of Table 15.1.3-1, page 15.1-94, are defined on page 15.1-95. The term "S" represents the Laplace Operator. A variable expressed as a function of S is written in the frequency domain.

The units on single trip level subsystems are defined in the Table. For multiple trip level subsystems whose performance is given by a trip equation, the variables are normalized (i.e., 1.0 = 100%).

For single trip level subsystems, the Table indicates the latest trip value. For multiple trip level subsystems, the latest trip varies as defined by the trip equations. Each accident analyzed in the Section 15 Safety Analysis identifies the latest trip level assumed in the analysis.

The various trip functions with their respective trip settings are provided primarily for the PPS diversity and redundancy such that different types of fault events are adequately protected against. They are not considered to have any impact on the frequency of occurrence of the events specified.

The Plant Duty Cycle is a set of conservative transients which provide an envelope for plant operating conditions. Trip settings assumed in the Duty Cycle accident scenarios account for worst case PPS performance.

In general, trip settings are determined such that changes to the duty cycle are not required. In a few instances, redefinition of duty cycle events may be necessary to accommodate limitations in the PPS trip subsystem such as sensor accuracy or response time.

TABLE 15.1.3-1

PPS SUBSYSTEM TRIP LEVELS OR TRIP EQUATIONS

Primary Shutdown System

High Flux	Trip at 115% power
Flux-Rate	Positive: $\mathcal{L}^{-1}\left[\frac{1.01}{1+28S}\right] -0.99\,(t) + 0.1706Np + 0.0364 \leq 0$
	Negative: $1.01\phi(t) - \mathcal{L}^{-1}\left\{\phi\,(s)\,\middle\vert\,\frac{1.01}{1+28s}\;0.1969Np\right\} + 0.0416 \leq 0$
Flux to Pressure	$1.318\sqrt{P} - \phi + 0.0425 \leq 0$
Primary to Intermediate Speed Ratio	$Np\,(0.147 \pm 0.0022) + 0.0595 \pm 0.0007 - \text{AbsVal}\,[Np\,(1 \pm 0.015) - N_I\,(1 \pm 0.015) + 0.0075 \pm 0.01] \leq 0$
HTS Pump Frequency	Trip at 57 Hertz
Reactor Vessel Level	Trip when level drops 18" from normal operating level
Steam to Feedwater Flow Ratio	Trip at 30% mismatch
IHX Primary Outlet Temperature	Trip at 830°F

Secondary Shutdown System

Flux to Total Flow	$1.2\Sigma Fp - 0.99\,\phi + 0.087 \leq 0$
Startup Flux	Trip before 10% power
Primary to Intermediate Flow Ratio	$Fp\,(0.147 \pm 0.0022) + 0.050 \pm 0.0007 - \text{abs Val}\,[Fp\,(1 \pm 0.015) - F_I\,(1 \pm 0.015) + 0.0075 \pm 0.01] \leq 0$
Steam Drum Level	Trip at 8" drop from full power steady state level

TABLE 15.1.3-1 (Continued)

High Evaporator Outlet Temperature	Trip at 750°F
Sodium Water Reaction	Trip initiated within 3.0 seconds
HTS Pump Voltage	Trip at 75% of rated voltage

Definition of Variables

$\mathcal{L}^{-1}$ = Laplace Operator
O = Reactor Flux
Np = Average Primary Pump Speed
Np = Primary Pump Speed
N_1 = Intermediate Pump Speed
P = Reactor Inlet Plenum Pressure
F = Total Primary Pump Flow
F^p = Primary Pump Flow
$F_1{}^p$ = Intermediate Pump Flow

The above variables are normalized such that their value at 100% conditions = 1.0.

Question CS760.38

Provide the rationale for referring to the DHRS as a safety grade system when it also states that it must be "adjusted manually."

Response

DNRS is a safety grade system providing a fourth decay heat removal loop for CRBRP. DHRS is specified as a Safety Class System (see PSAR Section 3.2) with electrical equipment classified as 1E. Manually initiating the DHRS does not degrade its safety grade status.

Manual initiation of the Direct Heat Removal Service (DHRS) is appropriate based on the time period available and the number of operator actions required.

In the worst case transient analysis performed for the DHRS (extremely unlikely event of DHRS initiation Following Reactor Shutdown from 100% Power with Loss of All Heat Transfer through the IHXs at time of reactor trip), it is assumed that no heat is transferred to the DHRS for one-half hour after shutdown. The heat capacity of the primary system is used as the principle heat sink during this period with no operator action required to assure this heat sink. As shown in Section 5.6.2 of the PSAR, the temperatures associated with this event are acceptable.

Furthermore, the manual initiation referred to actually consists of turning six switches in the Control Room on the DHRS panel from the normal to the DHRS position. This is an on-off control rather than an adjustment. These control transfer switches and automatic sequences have been provided to automatically position DHRS valves and control the pumps and ABHXs in order to reduce the operator actions required. A conservative estimate of the time period required for the control transfer switches and the automatic sequences to operate is twelve minutes; therefore, the operator has an adequate time period to determine the need for DHRS use and initiate DHRS.

Based on the time periods noted above and the number of operator actions required, it is judged acceptable for DHRS initiation to be performed remote-manually by the operator in the Control Room.

Question CS760.57

In Sections 15.7.3.1, Leak in a Core Component Pot (CCP), the T across the wall is shown to be about 1500°F. Simple hand calculations show that the stainless steel wall can fail T's ≥500°F.

a. Demonstrate the wall structural integrity for all cases considered.

b. Provided this failure occurs, how does this alter the design and operational procedure?

c. With at least 1 hour per assembly for refueling, how long does it take to refuel (in light of the new core design)?

d. Demonstrate numerical and model accuracies of TAP-4F and DEAP computer codes.

e. Why do clad temperatures keep increasing after reaching the melting temperature? Does a phase change occur and how is it considered?

f. If the clad temperature stays at the melting point during the change of phase (in the case of stacked fuel pellets), there should be significantly less CCP temperature rise as compared to what is calculated. A similar situation applies to the packed bed case, except in this case there is significantly more thermal interaction as compared to the stacked pellet case, and thus, due to contact conduction, CCP temperature must be higher than the previous one. Demonstrate as in item d above for the conditions considered herein.

Response

The 1500°F temperature difference to which the question is addressed is the difference between the CCP and EVTM cold wall temperatures. It is not the temperature gradient across the CCP wall. The question is apparently the result of a lack of clarity in PSAR Figure 15.7.3.1-5, in which the identification of the different regions in the thermal calculation is such that "cold wall" was interpreted as the cold side of the CCP wall rather than as the EVTM cold wall. The approximately 1500°F temperature difference would be across the approximately 7-inch argon gas space between the CCP outer wall and the EVTM cold wall. It is the driving force for radioactive heat transfer of core assembly decay heat to the cold wall limiting the fuel temperature rise. The temperature difference across the CCP wall itself would be only about 10°F.

The following sections of the response discuss the corresponding parts of the question.

a. There is no question of structural integrity with the gradient of 10°F. the maximum radial thermal gradients across the CCP wall occur during its insertion into the EVST sodium pool at the maximum rate of 24 ft/min after transfer from the reactor vessel. This case was analyzed with the CCP assumed to contain a maximum powered fuel assembly and to have been in the EVTM long enough to reach steady-state temperatures. The transient thermal gradients are high, with a peak of approximately 500°F, a few tenths of a second after insertion; however, the duration of the gradient is short. The ASME Code structural analysis accounted for thermal stresses resulting from the design basis number of these immersion cycles and showed that the allowable number of these thermal cycles is greater than the design basis number.

b. No response required.

c. There are between 3 and 379 core assemblies exchanged per refueling, with an average of 159. At 1 hour per transfer (2 hours per exchange) between the EVST and the reactor vessel as assumed in the event analysis in PSAR Section 15.7.3.1, the time for an average refueling would be 13-1/4 days plus time for setting down and picking up assemblies. The transit time used for the event analysis is a conservative estimate which is longer than the expected refueling times. (These times cover only core assembly transfers; preparation of the reactor for refueling and termination from refueling are not included.)

d. The TAP-4F and DEAP computer codes are described in Appendix A of the PSAR. Verification of the codes is covered in these descriptions.

e. Cladding temperatures keep increasing after reaching the melting point because the thermal analysis does not model the stainless steel phase change. This is conservative since it results in reaching steady-state temperatures more quickly. It does not affect prediction of steady-state temperatures.

The effect of cladding melting was considered by running two additional cases which assume no cladding at all is present. The accident analysis in PSAR Section 15.7.3.1 covers the CCP loss of sodium event to its expected conclusion and two extensions of this event (first, to collection of fuel fragments in the fuel assembly duct, then to a hypothesized relocation of the fuel fragments to the bottom of the CCP). The temperature distribution and fuel assembly configurations for these cases are shown in PSAR Figures 15.7.3.1-4, 15.7.3.1-6, and 15.7.3.1-8, respectively.

f. The question of cladding melting does not arise in the second case of a packed bed since this case assumes no cladding is present and fuel pellets have collapsed into a packed bed following loss of cladding. For conservatism, all cladding of the entire 3-ft high fuel region was assumed to be absent.

CCP temperature is determined, not by fuel temperature, but by the heat transfer resistance betwaen the CCP wall and the heat sink, which is the EVTM cold wall. This resistance does not change among the three cases except for a minor reduction in heat transfer area as the fuel region gets shorter as the event progresses from one case to the next. This reduction is reflected by the small increase in CCP temperature (100°F) from one case to the next.

Thermal interaction within the CCP is modeled to be less (i.e., thermal resistance is greater) in the packed bed case because of high contact resistance between particles and higher resistance to radiant heat transfer than when the fuel is in well-ordered stacks. This was done purposely in order to obtain conservative estimates of fuel temperature to assure that fuel melting would not occur. As noted above, CCP temperature is not affected by increases in heat transfer resistance within the CCP; these changes affect ony fuel temperatures. This can be seen by noting the substantial fuel temperature increases from one case to the next. This conservative sequence of cases demonstrates that fuel melting will not occur.

Question CS760.58

In 15.7.2.5, page 15.7-16a, please elaborate on the portion of the statement that reads, "Loss of fluid due to solidification of the concentrated waste." This statement refers to what process.

Response

As discussed in Sections 11-2 and 11-5, liquid radioactive waste is concentrated in an evaporator system to an approximate concentration of 24% by weight. The evaporator bottoms (concentrated liquid radioactive waste) are then transferred to the Radioactive Waste Solidification System for immobilization in cement. Since the evaporator bottoms retain a portion of the influent water (and therefore some tritium), the tritium activity in the decontaminated water is somewhat reduced. The subject statement in PSAR Section 15.7.2.5.2 has been reworded for clarification.

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 Analysis of Effects and Consequences

Gaseous Release

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.

Liquid Release

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

Question CS760.86

Core Structure Cooling - 6% of the total coolant flow is bypassed to cool control assemblies, radial shield assemblies, the reactor thermal liner and to account for bypass and leakage flow. In optimizing the reactor performance, the bypass flow allocation has only a minimal impact on core performance. The bypass flow allocation can be increased by 20% and the coolant flow to core and blanket assemblies is only reduced by 1%. However, after the design is "frozen" these flow allocations must be such that they provide sufficient cooling to the structures under full flow as well as natural convection conditions.

While the bypass flow can be fully sufficient under full power operating conditions, there is concern that there is also sufficient flow under low power - low flow conditions, especially during natural circulation transients. There is always a problem to control low flow rates. For example, under full power operating conditions, the radial shield receives a total of 1.35% of total flow, which amounts to approximately 0.5 lb/sec of sodium flow per assembly. Under natural convection conditions, any flow through these assemblies will be invisible. Even at reduced power operation, it is not clear how much flow these shield assemblies will see. The flow path to get to the shield assemblies, is rather complicated. First, the flowing sodium enters the lower inlet flow module. The leakage flow goes up the hydraulic balance bleed hole and enters the lower piece of the bypass flow module. From there it goes up and goes through the socket for the RRS finally into the RRS. For this flow to proceed this far, the LIM has to fit into the core support plate. Inside the core support structure are holes which allow the sodium to flow from the LIM to the BPFM. In the BPFM are holes which have to line up with the sockets for the RRS. The tolerances in all of those complex fittings will affect the flow which goes into the RRS.

This example illustrates the problem in core structure cooling at both full power and convection flow conditions. The following questions need to be answered:

1. What is the accuracy in the prediction of the bypass flow distribution at full power and low power conditions?

2. What are the temperature limits for the core structures which determine the required coolant flows?

3. Is there a possibility to control the flow to the various core structures?

4. What margins are built into the bypass coolant flows?

5. What happens if there is no flow through the RRS?

6. Can the bypass flows be measured directly?

7. Since there is no gamma heating at BOL in the RRS, what are the assurances that they will receive sufficient cooling?

8. Is it possible that there is a local overheating of the vessel because of lack of cooling of the RRS?

Response

Before addressing the specific questions, some discussion of general conditions appears necessary to clarify the Project approach to assuring adequate bypass flow over the entire operating range including natural circulation.

- The flow to the RRS, reactor vessel/liner and other peripheral components is controlled through orifices located either in the LIM's or in the Bypass Flow Modules. Some test data have been obtained for the flow control orifices to characterize the bypass flow.

- To date, bypass flow elevations have been done at low steady-state forced flow conditions as part of the flow management calculations.

- Regarding transients, the verified CØBRA-WC code has the capability to dynamically model RRS and other low flow paths simultaneously with the high flow fuel and blanket assemblies over the entire operating range as shown during simulation of the FFTF natural circulation tests. For example, at ~3% total reactor flow, CØBRA correctly predicted the flow split between the Row 8 reflector and driver assemblies.

- From the design standpoint, at low flow conditions the components cooled by bypass flow have very little heat generation, and any increase in temperature following flow reduction enhances the natural circulation phenomenon.

Following are responses to the specific questions asked:

1. The flow accuracy requirements have been established in the individual component design specifications; these translate directly into the uncertainties required for the flow control devices. This approach can be taken because the general orifice characteristics over the range of flow conditions are well known, and the desired specific characteristics can be designed into each device. It is recognized that the pressure drop uncertainties are greater at low flows, and this is factored into the design. The flow control mechanisms will be tested to verify that the requirements over the flow operating range have been met. Final uncertainty levels will be assessed following evaluation of experimental data (as discussed in the response to Question CS760.77). The results of these evaluations, together with the entire CRBRP reactor flow distribution network as-finalized, will be reported in the FSAR.

2. The steady-state and transient thermal constraints of RRSAs are based on the cross-duct temperature gradients at the axial load pads as well as their absolute temperatures which are well below any structural limits of the 316 stainless steel and the boiling temperature of sodium. These analyses took into account the worst combination of uncertainties at the $+2\sigma/-2\sigma$ confidence level.

   The vessel flow was based on the requirement that the vessel temperature remains under 900°F for nominal conditions and meets structural requirements during transients.

3. Flow to the various core structures is controlled by flow tested orifices in the Lower Inlet Modules and the Bypass Flow Modules. The leakage flow in the flow path, particularly the leakage across the piston rings, is taken into account. The Bypass Flow Module is fed by orifices which are flow tested to accurately control the flow, not by leakage flow up the hydraulic balance bleed holes.

4. 3σ uncertainties are considered in the CATFISH representation of the reactor flow network.

5. The RRSAs are cooled by two different methods: directly by orificed coolant flowing through the assemblies and indirectly by interstitial flow through the core. Therefore, the case of "no flow" is not expected.

6. The bypass flow, as well as the flow to any other core component, cannot be measured directly in the reactor. However, out-of-pile flow testing of orifices has been or will be conducted to characterize the flows.

7. If there is no heating, there should be no concern about sufficient cooling. It is not true that there is no gamma heating at BOL in the RRS. Neutron radiation capture and inelastic scatter is practically constant throughout life. The only difference between BOL and EOL conditions is in the additional heating due to fission product gamma from the core and neutron activation decay power, which is minor at full power conditions.

8. Vessel temperatures are insensitive to RRS temperatures since the vessel is cooled via a separate, parallel, controlled flow path. Actually, if less flow than designed went through the RRS, this extra flow would be redistributed to the other flow paths in the flow network, including that of the vessel liner.

Finally, it should be reiterated that a detailed description of the final evaluation of the core structures thermal-hydraulic design will be reported in the FSAR.

Question CS760.112

What are the various setpoints for the Turbine Bypass system?

Response

See PSAR Section 10.3.2 for the requested Turbine Bypass System setpoints. See PSAR Section 10.4.4 and revised Section 7.7.1.8 for description of the steam dump and bypass control system.

Control of the argon supply and vent values is accomplished by an "on-off" type pressure controller which cycles the supply and vent values to maintain the cover gas pressure between the lower and upper limits. Sufficient dead band is provided between lower and upper limit operation to prevent undue cycling of the supply and vent valves.

Operational Considerations

The pressure controllers for the sodium dump tanks are located at the local control panel in the Steam Generator Building. However, manual overrides for the supply and vent valves are provided in the main control room and may be utilized at the plant operator's discretion.

High and low pressure alarms alert the operator to off-normal conditions which may result from a malfunction of the pressure control system. Pressure data is provided to the Data Handling and Display System and is available for display upon call by the operator.

7.7.1.8. Steam Dump and Bypass Control System

The Steam Dump and Bypass Control System provides the necessary control and instrumentation hardware to operate the Turbine Bypass System as described in Section 10.4.4 and shown in Figure 10.3-1.

Redundant interlocks are provided to prevent bypass operation in the event the condenser is unable to accept steam flow (e.g., high condenser back pressure or loss of circulating water flow).

At reactor poer levels above 40% rated power, the Turbine Bypass Control System is operated in a load error mode where bypass valves are opened proportional to the difference between turbine demanded load and generated power. A valve position signal is provided to the Turbine Bypass Control System by the Supervisory Control System which makes the comparison. The circuitry includes a dead band with a 10% load setpoint.

A pressure control channel is provided for the regulation of main steam pressure following reactor trip, during decay heat removal operation and during turbine standby, loading and unloading operations. The pressure control mode is manually selected for reactor power levels below 40%.

At reactor power levels above 40%, the Steam Dump and Bypass Control System automatically positions bypass valve(s) to regulate bypass steam flow approximately proportional to reactor power, however; the pressure control mode may be manually selected by operating personnel at any power level. The pressure control mode is automatically selected following a reactor scram and a turbine trip condition.

Question CS760.136

The list of design transients (Table 5.7-1) includes two-loop operational events yet two-loop conditions are not included in the heat transport system design. We feel that the Project's present position with regard to redundant heat removal capability is not consistent with two-loop operation. Any future two-loop operation may entail considerable design changes.

Response

Revised PSAR Section 1.1.1 describes the Project's approach to two-loop operation.

The plant is designed with three main coolant loops and the intended mode of operation is that all three loops should be continuously in service.

While the system design is intended to be capable of allowing for operation at power on two loops, the applicant is not requesting NRC review of this operational mode. If, at some time in the future, the applicant considers that all safety requirements can be met under two-loop operatioin without significant additional design features, the applicant may elect to apply to NRC for a two-loop operation capability. This should not constrain the NRC review of the construction permit application.

The construction completion data for the plant was originally scheduled for September, 1981. Until current Congressional and Administration actions are completed, the reactor criticality data cannot be re-scheduled.

1.1.2 Overview of Safety Design Approach

The design of the CRBRP is based on the defense-in-depth safety philosophy, commonly known as the Three Levels of Safety design approach. A summary of the design safety approach for the CRBRP is provided in Tables 1.1-1 and -2.

Level 1 Design

The first level of safety provides reliable plant operation and prevention of accidents during normal operating conditions through the intrinsic features of the design, such as quality assurance, redundancy, maintainability, testability, inspectibility, and fail-safe characteristics. The plant is designed not only to accommodate steady-state power conditions, but also to have adequate tolerance for normal operating transients, such as start-up, shutdown, and load-following. As a basic part of the CRBRP development program, a number of large-scale engineering proof tests are being performed to verify the design concepts. This testing process provides predictability of performance and, hence, safety through assurance of the use of proven methods, materials, and technology.

Extensive pre-operational test programs will be conducted in the plant to assure conformance of components and systems to the established performance requirements. Key parameters will be monitored continuously or routinely and well-define surveillance, in-service inspection, and preventive maintenance programs will be carried out by a trained operating and maintenance staff to provide assurance that as-built quality is maintained through the life of the plant.

Level 2 Design

The second level of safety provides protection against Anticipated and Unlikely Faults (such as partial loss of flow, reactivity insertions, failure of parts of the control system, or fuel handling errors - Faults are defined in Table 1.1-1A) which might occur in spite of the care taken in design, construction, and operation of the plant. This level of safety