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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: Letters, P. S. Check to J. R. Longenecker, "CRBRP Request for Additional Information," dated April 9 and May 14, 1982

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

Enclosed are responses to Questions CS 421.24, 35, and 52 and CS 760.148, 149, and 173 which will also be incorporated into the PSAR Amendment 71 scheduled for submittal later in September.

Sincerely,

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

Enclosures

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Question CS421.24

In the PSAR Section 7.5.6.14, a limited description of the CRBR Sodium Dump System is presented. Provide a detailed discussion of this system and present a single failure analysis for this system.

Response:

Assuming the PSAR reference is Section 7.5.6.1.4. "Sodium Dump", this section outlines the role of the Instrumentation and Monitoring System in support of the Sodium Dump Subsystem described in PSAR Section 5.5.2.7 (the IHTS portion, drain piping and valves, are discussed in PSAR Section 5.4).

The sodium drain operation following reactor shutdown is at the option of the plant operator and is not required to perform an active safety function.

Section 5.5.2.7 has been revised to reflect a more detailed discussion of the Sodium Dump Subsystem design; however, a single failure can be accepted without jeopardizing the heat transport capability of the remaining two loops because of loop redundancy.

A study of the effect of the vent piping diameter upon pressures in the IHTS was made early in the evaluation of SWRPRS. Analyses made with a 24" diameter pipe from the rupture disc assembled to the Reaction Products Separator Tanks showed that the pressures within the IHTS following the postulated design basis sodium-water reaction were within the specified limits.

The Reaction Products Separator Tanks are sized to accommodate the maximum amount of sodium and liquid or solid sodium reaction products which can be ejected or drained from the IHTS into SWRPRS during and following a sodium-water reaction. Regions of the IHTS and components in that system which might drain into SWRPRS during and following a sodium-water reaction were determined by study of the hydraulic profile of the IHTS. It was assumed that all of the rupture discs in the main rupture disc assembled would be broken by the sodium-water reaction. An additional capacity of about 25% above that determined by evaluation of the hydraulic profile of the IHTS was then added to establish the capacity of the Reaction Products Separator Tanks.

SWRPRS piping and equipment internal to the SGB are seismic Category 1. The SWRPRS piping and equipment external to the steam generator building are designed as seismic Category 3. If a major seismic event should damage this portion of the SWRPRS, such that a vent path to the atmosphere is not available, overpressure protection is maintained by the internal SWRPRS volume. The maximum system pressure following a steam generator DBL would be maintained at less than 100 psig by the automatic isolation and venting actions initiated with SWRPRS actuation. The minimum design pressure for SWRPRS is 125 psig.

Tests and Inspections

During plant operation, inert gas pressure will be maintained above atmospheric in the system and observation of the pressure in the system will verify the leak-tightness.

5.5.2.7 Sodium Dump Subsystem

The Steam Generator System (SGS) provides one sodium dump subsystem for each of the three parallel independent Intermediate Heat Transport System (IHTS) circuits: each sodium dump subsystem consists of a sodium dump tank located at the lowest building level beneath the evaporator and superheater modules.

A sodium dump tank having a useable capacity of approximately 9480 cubic feet at 930°F is provided within each sodium dump subsystem which is large enough to store all of the sodium from the IHTS and auxiliary systems to serve as a:

a. Sodium dump for sodium at operating temperatures

b. Intermediate sodium storage

c. Sodium fill

d. Removal or clean-up of Na-H_2O reaction products.

The Sodium Dump Subsystem will be used for normal drainage or for drainage of sodium from each IHTS circuit after the sodium has been contaminated by sodium-water reaction products from a large sodium-water reaction occurring within an evaporator or superheater module. Upon manual initiation, rapid drainage of IHTS sodium and $Na-H_2O$ reaction products will be accomplished by lines which are connected at five different locations within the IHTS sodium circuit. These piping runs provide gravity drains from the sodium expansion tank, the superheater, the evaporators, and the low points in the hot and cold legs of the IHTS main piping to the dump tank, as shown in Figure 5.1-2a. Each piping run contains a pair of isolation valves in series, with independent controls. To prevent inadvertent draining of an IHTS loop, interlock features are incorporated into the operation of the valves.

The IHTS sodium will be cooled to a bulk average temperature of less than 800°F prior to opening the sodium dump valves following duty cycle events which increase the IHTS average bulk sodium temperature above that associated with full load steady state operating conditions. The sodium dump tank will accommodate the average bulk sodium temperature associated with the IHTS at full load steady state operating conditions.

In the event that a sodium-water reaction occurs, sodium contaminated with sodium-water reaction products would be dumped into the dump tank and maintained in a molten state by trace heaters on the tank. Later, the contaminated sodium in the dump tank will be cleaned by circulation through the Intermediate Sodium Processing System (see Section 9.3) and then transferred to the IHTS sodium loop. The dump tank could then be cleaned and the Sodium Dump Subsystem be made ready for use again.

Question 421.35

Provide a more detailed discussion (PSAR Section 7.5.4.1.1) on the argon cover gas monitoring system and indicate the design criteria for this system. Also, Section 7.5.4.1.1 indicates that a minicomputer will be used for cover gas analyses. Discuss the use of this minicomputer and how its failure relates to system operation.

Response:

The response to this question is provided in revised PSAR Section 7.5.4.1.1.

7.5.4.1 Design Description

The following subsystems make up the FFM system.

1. Cover Gas Monitoring

   This subsystem continuously samples the cover gas and determines, through gamma analysis:

   - the concentration of selected radioactive fission gases to inform the plant operations staff upon each instance of fuel or blanket pin cladding failure.

   - the concentration of radioactive fission gases to characterize the failed pins as to burnup and other information.

2. Reactor Delayed Neutron Monitoring

   This subsystem continuously monitors for the presence of fission products in the sodium coolant which decay with the emission of neutrons. A predetermined increase in the neutron signal from the Primary Heat Transport System sodium, above the normal background level, is taken as an indication of fuel contact with sodium.

   The Impurity Monitoring and Analysis System provides verification of fuel exposure to the sodium by removing sodium with a grab sampler and by subsequent laboratory analysis for fuel and fission product material.

3. Failed Fuel Location

   Stable (non-radioactive) xenon and krypton isotopes (that are not fission products) are placed in each fuel and blanket assembly pin. Each assembly has a unique ratio of isotopes which will be released to the cover gas upon failure of a pin in the assembly. Analysis of a processed sample of the cover gas, using a mass spectrometer, is used to identify the assembly containing the failed pin.

The FFM subsystems are described in greater detail in the following sections. A block diagram of the FFM system is provided in Figure 7.5-3.

7.5.4.1.1 Cover Gas Monitoring Subsystem

The Reactor Cover Gas Monitoring Subsystem (CGMS) detects a fuel or blanket failure in the presence of up to four existing failures. It continuously monitors the reactor cover gas for radioactive gases which are present as background (Ne-23 and Ar-41) and gaseous fission products which escape from failed fuel and radial blanket pins. The cover gas first passes through a sodium vapor trap, which is located in the Reactor Containment Building, then passes through a charcoal delay bed which is located in a shielded cell in the Reactor Service Building. The purpose of the delay bed is to concentrate the radioactive xenon and krypton fission gases relative to Ne-23 and Ar-41 in

order to increase the signal to background ratio. Monitoring is done with a planar germanium (Ge) gamma detector which continuously monitors specific fission gas radioisotopes. An alarm in the main control room (plant annunciator) will be activated in the event of an abnormal activity level increase for any of these radioisotopes. A multichannel analyzer, including a minicomputer, analyzes the signal from the detector to display the entire gamma ray spectrum. The minicomputer with additional input of the reactor power provides basic characteristics of the failure, i.e., magnitude and burnup, which may be used to supplement the Failed Fuel Location Subsystem through correlation with core and blanket history.

Failure of the minicomputer does not affect the failure detection capability of the CGMS. In this case, the multichannel analyzer memory will still be functional; however, the characterization capability will be lost. During normal operation when there are no failures, the minicomputer is not used. However, if there is a fuel failure, which requires characterization, data from the multichannel analyzer can be recorded and analyzed manually. In addition, a gas grab sample could be obtained, and then analyzed in the Plant Service building Laboratory which has equipment equivalent to the CGMS multichannel analyzer and minicomputer. By using this equipment one can characterize the fuel failure and provide the desired information.

7.5.4.1.2 Reactor Delayed Neutron Monitoring Subsystem

The Reactor Delayed Neutron Monitoring Subsystem includes a Delayed Neutron Monitor consisting of an assembly of three BF_3-filled gas proportional neutron detectors, mounted in a shielded moderator assembly adjacent to each of the three Primary Heat Transport System hot leg pipes.

Coolant sodium transported past the detector assembly, is continuously monitored for delayed neutrons emitted by decay of radioactive precursors in the sodium. The system sensitivity is dependent on the signal-to-background ratio of the system. Signal is defined as detected delayed neutrons produced by recoil of precursors from fuel exposed by cladding failure, or from fission of fuel washed out into the sodium through a failure. Background is defined as detected neutrons from known sources which are not initially related to failed fuel (fuel pin contamination, fissionable impurities in core structural materials, fissionable materials in the sodium, and neutrons from the reactor).

The shielding and moderator assembly provides 1) reduction of gamma interference from Na-24, 2) moderation of neutrons, 3) capability for remote insertion of a calibrated neutron source, 4) capability for insertion and removal of the detector assemblies from the reactor containment building operating floor without deinerting the PHTS cells.

Question CS421.52

Describe how the effects of high temperatures in reference legs of steam drum water level measuring instruments subsequent to high energy breaks are evaluated and compensated for in determining setpoints. Identify and describe any modifications planned or taken in response to IEB 79-21. Also, describe the level measurement errors due to environmental temperature effects on other level instruments using reference legs.

Response:

The steam drum water level instrument uses a reference leg which operates at ambient temperature. The maximum cell temperature following a high energy pipe break would be 215°F. This results in the indicated water level being apprximately 1.4 inches above actual water level.

The setpoint which is affected by this event is the 8-inch below normal water level which results in a protection system reactor trip and Steam Generator Auxiliary Heat Removal (SGAHRS) initiation. The 8-inch below normal water level setpoint is such that the trip signal will occur with sufficient water inventory to prevent the water level from going out of range or Steam Drum dryout prior to a reactor trip and SGAHRS initiation.

It should be noted that the three loops in CRBRP are in separate cells, and the environment from a high-energy pipe break will only affect one loop. The other two loops will provide decay heat removal. In addition, the 1.4 inch higher than actual indication does not affect the ability to monitor the water level in the affected loop during post accident monitoring.

IEB 79-21 identifies the need for bias circuits to correct large increases in indicated versus actual water level indication as a result of increased ambient temperature. This is due to the long steam generator reference legs in pressurized water reactors. In CRBRP the steam drum has a short reference leg (38 inches). Therefore, CRBRP level instruments do not have this large magnitude of level inaccuracy as a result of high ambient temperature. The small increase of 1.4 inches in indicated level above the actual level is well within the setpoint margin, to ensure that water inventory is maintained within the level instrument range during the event and post accident monitoring. Therefore, there is no need for compensation circuits or channel modification.

Level instruments in the Reactor Primary Heat Transport System and Intermediate Heat Transport System do not need nor use reference legs.

Question CS760.148

During a startup testing of FFTF, a non-linearity of the ex-vessel neutron detector response, as a function of reactor power level was observed. This non-linearity was due to temperature changes during the power ascent affecting the leakage of neutrons from the core to the detectors. The observed non-linearity caused indicated power to be different from actual power at operating points other than full power and caused an extensive revision of the FFTF PSAR Chapter 15 safety analysis to account for this affect. In consideration of the above, please provide the following information:

a) The predicted non-linearity affect on the CRBR ex-vessel neutron detectors in going from zero to 100% power.

b) A description of how this affect will be accommodated in the plant operating plans/procedures and in the Chapter 15 Safety Analysis.

Response:

Ex-vessel neutron detector response (wide range and power range) will be affected by temperature changes during the power ascent. The leakage of neutrons from the reactor system to the ex-vessel detectors is dependent upon the temperature changes in the reactor system and has a time dependency due to the different heat-up rates for reactor system components and the various flow paths of the primary coolant. The effect of primary sodium coolant temperature on the leakage of neutrons from the reactor vessel was predicted for FFTF, however, the magnitude of the effect and the non-linearity with respect to sensor data was not predicted. Reviews of the FFTF experience and studies of the CRBRP ex-vessel detector response and effects of non-linearity of detector response on CRBRP systems are ongoing.

In support of the FSAR, studies will be carried out (including review of the safety analyses in Chapter 15) to confirm that available margins and equipment design features will accommodate the non-linearity predicted for CRBRP. As was the case for FFTF, appropriate adjustments to the control and protection system settings will be made following startup tests so as to meet the specified operating and safety criteria.

Question CS760.149

Power operation over an operating cycle may cause changes in the ex-vessel neutron detection readings due to flux profile changes caused by burnup and control rod withdrawals. Please provide:

(a) The predicted change in ex-vessel neutron detector readings over an operating cycle due to flux profile changes.

(b) A description of how this affect will be accommodated in the plant operating procedures and in the Chapter 15 Safety analysis.

Response:

(a) Ex-vessel neutron detector readings (wide range and power range) will be affected by changes in the flux distribution during an operating cycle caused by control rod withdrawal and fuel burnup effects. Studies have been performed to determine the effect of anticipated changes in control rod configuration and burnup on the ex-vessel detector reading. This effect is 10 percent or less for changes in control rod configuration and burnup that occur during two consecutive equilibrium operating cycles (550 effective full power days).

(b) The wide range and power range flux monitors are periodically recalibrated throughout the operating cycle to eliminate the effects of this flux distribution change. Manual calorimetric calibrations are to be performed at least once per shift. It is required that the wide range and power range flux monitors be recalibrated against such a manual calorimetric calibration at least once per day with a maximum allowable difference (tolerance) of 1 percent. Thus, any significant change in the flux distribution will be reflected in the recalibration of the wide range and power range monitors on a timely basis. In addition, a calorimetric calibration can be performed by the Plant Data Handling and Display System computer at any time the operator requests.

Given the above recalibration requirement, any effects due to flux profile change are zeroed out during plant operation. Therefore, this will have no affect on the Chapter 15 analyses.

Question CS760.173

Section 15.6.1.2 presented the analysis of a 7,500 gallon sodium spill in the RSB. What RSB leakrate was assumed in this analysis? Is this leakrate a design requirement on the RSB?

Response

As discussed in Section 15.6.1.2 of the PSAR, the postulated Ex-Vessel Storage Tank accident occurs in an inerted cell. All aerosols resulting from this postulated accident would be retained in the Ex-Vessel Storage Tank cell with no release to the air-environment of the RSB. However, for radiological evaluation conservatism, it is assumed that accummulated aerosols are released directly to the external atmosphere at the end of 2 hours, 8 hours and 24 hours. These releases are then applied to atmospheric dispersion factors (χ/Q) in Chapter 2 of the PSAR to determine the 2 hour site boundary and 30 day LPZ doses.

In view of the assumptions of total release of aerosols, leak tightness is not a design consideration in this analyses.

See PSAR Section 6.2.6 for a discussion of the RSB Confinement Filtration and Ventilation System.