Department of Energy
Washington, D.C. 20545
Docket No. 50-537
HQ:S:83:217

FEB 23 1983

Dr. J. Nelson Grace, Director
CRBR Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Grace:

ADDITIONAL INFORMATION ON THE SECONDARY CONTROL ROD SYSTEM - CLINCH RIVER BREEDER REACTOR PLANT

Enclosed are marked-up Preliminary Safety Analysis Report (PSAR) pages providing additional information on the secondary control rod system capability. This information will be placed into Chapter 4 of the PSAR in a future amendment.

Any questions regarding the information provided can be addressed to S. Frye (FTS 626-6354) or K. Peterman (FTS 626-6186) of the Project Office Oak Ridge staff.

Sincerely,

John R. Longenecker

John R. Longenecker
Acting Director, Office of
Breeder Demonstration Projects
Office of Nuclear Energy

Enclosure

cc: Service List
Standard Distribution
Licensing Distribution

D001

7.2 REACTOR SHUTDOWN SYSTEM

7.2.1 Description

7.2.1.1 Reactor Shutdown System Description

The Reactor Shutdown System (RSS) consists of two independent and diverse systems, the Primary and Secondary Reactor Shutdown Systems, either of which is capable of Reactor and Heat Transport System shutdown. All design ~~anticipated and unlikely~~ events can be terminated without exceeding the specified limits by either system even if the most reactive control rod in the system cannot be inserted. ~~In addition, the Primary RSS acting alone can terminate all extremely unlikely events without exceeding specified limits even if the most reactive control rod in the system cannot be inserted.~~ To assure adequate independence of the shutdown systems, mechanical and electrical isolation of redundant components is provided. Functional or equipment diversity is included in the design of instrumentation and electronic equipment. The Primary RSS uses a local coincidence logic configuration while the Secondary RSS uses a general coincidence. Sufficient redundancy is included in each system to prevent single random failure degradation of either the Primary or Secondary RSS.

As shown in the block diagram of the Reactor Shutdown System, Figure 7.2-1, the Primary RSS is composed of 24 subsystems and the Secondary RSS is composed of 16 subsystems. Figure 7.2-2A is a typical Primary RSS instrument channel logic diagram. Each protective subsystem has 3 redundant sensors to monitor a physical parameter. The output signal from each sensor is amplified and converted for transmission to the trip comparator in the control room. Three physically separate redundant instrument channels are used. When necessary, calculational units derive additional variables from the sensed parameters with the calculational units inserted in front of the comparators as needed. The comparator in each instrument channel determines if that instrument channel signal exceeds a specified limit and outputs 3 redundant signals corresponding to either the reset or trip state. The 3 outputs of each comparator are isolated and recombined with the isolated outputs of the redundant instrument channels as inputs to three redundant logic trains. The recombination of outputs is in a 2 out of 3 local coincidence logic arrangement.

Operating bypasses are necessary to allow RSS functions to be bypassed during main sodium coolant pump startup and ascent to power. ~~loop operation~~. Operating bypasses are accomplished in the instrument channels. For bypasses associated with normal three loop operation, the bypass cannot be instated unless certain permissive conditions exist which assure that adequate protection will be maintained while these protective functions are bypassed. Permissive comparators are used to determine when bypass conditions are satisfied. When permissive conditions are within the allowable range, the operator may manually instate the bypass. If the

TABLE 7.2-2 (Continued)

Fault Events	Primary Reactor Shutdown System	Secondary Reactor Shutdown System
Failure of Steam Dump System	Steam-Feedwater Flow Mismatch	Steam Drum Level
Sodium Water Reaction in Steam Generator(3)	Steam-Feedwater Flow Mismatch	Sodium-Water Reaction
III. Extremely Unlikely		
A. Reactivity Disturbances		
Positive Ramps ≤$2.0/sec		
Startup	Primary Low Flux	Startup Nuclear
25-40% Power	Flux-Delayed Flux or Flux- Pressure	Modified Nuclear Rate or Flux-Total Flow
40-100% Power	Flux- Pressure	Flux-Total Flow
Full Power	High Flux	Flux-Total Flow
SSE	HTS Pump Frequency	HTS Pump Voltage
B. Sodium Flow Disturbances		
PHTS Leak	Manual Trip(4)	Manual Trip
IHTS Leak	IHX Primary Outlet Temperature	Primary to Intermediate Flow Ratio

(1) The maximum anticipated reactivity fault results from a single failure of the control system with a maximum insertion rate of approximately 4.1 cents per second.

(2) The maximum unlikely reactivity faults result from multiple control system failures leading to withdrawal of six rods at normal speed or one rod at the maximum mechanical speed.

(3) The PPS is required to terminate the results of these extremely unlikely events within the umbrella transient specified as emergency for the design of the major components.

(4) No automatic PPS protection is required for the DBE PHTS leak of 8gpm since the required response time is significantly greater than 30 minutes and safety-related information systems are provided to inform the Operator of the presence of a PHTS leak. For additional protection (margin), a reactor sodium level trip subsystem is included in the Primary RSS to provide protection for PHTS leaks beyond the design basis.

TABLE 4.2-35

PRIMARY AND SECONDARY SHUTDOWN SYSTEM DAMAGE SEVERITY LIMITS

	Damage Severity Limit			
	Primary System Only Functioning		Secondary System Only Functioning	
	Without Stuck Rod	With Stuck Rod	Without Stuck Rod	With Stuck Rod
Normal: Operational (2)	Not Applicable	Not Applicable	Not Applicable	Not Applicable
Upset: Anticipated Faults	Operational Incident	Operational Incident	Minor Incident (1)	Minor Incident (1)
Emergency: Unlikely Faults	Minor Incident	Minor Incident	Major Incident	Major Incident
Faulted: Extremely Unlikely Faults	Major Incident	Major Incident	~~Not a Design Basis (3)~~ MAJOR INCIDENT	~~Not a Design Basis (3)~~ MAJOR INCIDENT

(1) Failure of the primary system to scram when required for an anticipated fault is defined as an extremely unlikely event (faulted condition). However, the damage severity limit for the secondary shutdown system is conservatively specified to assure fuel pin integrity even for the concurrent anticipated fault and failure of the primary shutdown system.

(2) No action required by Plant Protection System during Normal Operation.

~~(3) Combined probability of two independent failures (extremely unlikely fault and failure of Primary Control Rod System) is exceedingly low and not appropriate as a design basis. However, as an exception, the following concurrent events are being used as a design basis: a) loss of off-site power resulting from a Safe Shutdown Earthquake (with a consequent reactivity insertion of 60%) and b) failure of the Primary Control Rod System. With these concurrent events, the Secondary Shutdown System shall be capable of shutting down the reactor without exceeding Major Incident limits.~~

15.3.3.4 Primary Heat Transport System Pipe Leak

15.3.3.4.1 Identification of Causes and Accident Description

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

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

15.3.3.4.2 Analysis of Effects and Consequences

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

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

The 8 gal/min leak rate is orders of magnitude below the leak rate that could cause a significant core transient. Conservative analysis indicates that for 3-loop operation, a transient maximum loss rate of over 50,000 gal/min would be required for the core sodium temperature to approach the saturation value, and this would require a rupture of more than 1 square foot at the reactor

inlet nozzle. At other postulated primary heat transport system locations, even larger rupture areas would have to be postulated to challenge core cooling. Seperate best-estimate margin analyses have demonstrated that even leaks as large as a double ended rupture can be accommodated without a loss of core coolable geometry. The results of this analysis were confirmed in Reference 16 of Section 1.6 of the PSAR.

15.3.3.4.3 Conclusions

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

9. The only credit for operator action in mitigation of postulated sodium spills is shutdown of the Na overflow system makeup pumps 30 minutes after plant scram for a postulated leak in the Primary Heat Transport System (see Section 15.6.1.4).

10. Analyses of liquid metal burning in inerted cells assumes burning of all oxygen in the cell in which the liquid metal is postulated to leak and burning of all the oxygen contained in cells which are environmentally connected to the cell with the liquid metal leak.

11. The analysis of postulated liquid metal fires in air-filled cells does not include reaction of the liquid metal with postulated water released from concrete. The validity of this approach is presently being verified in conjunction with the large scale sodium fires test program discussed in Section 1.5.2.8 of the PSAR. If the test program does not support the present analysis approach, the appropriate effects of water release from concrete will be included in subsequent analyses.

12. The Moderate Energy Fluid System (MEFS) leak is used in this section to conservatively establish the CRBRP cell structural design performance. For purposes of assessing the functional performance of the heat transport systems, a leak rate corresponding to a 4-inch crack (8 gal/min) was selected based on a detailed evaluation of the PHTS piping integrity (see Section 15.3.3.4).

Table 15.6-1 provides a summary of the initial conditions for each fire considered and the maximum off-site dose as a percentage of the 10CFR100 guideline limits. As the table indicates, a large margin exists between the potential off-site doses and 10CFR100. A discussion of the pressure/temperature transient for each event is provided in the following sections; in no case do the fires result in conditions beyond the design capabiity of the cell/building.

The Project is assessing the impacts of NaK spills in the Reactor Service Building and will provide the results of aerosol released from the Reactor Service Building when the assessments are completed. The aerosols released from the RSB as a result of NaK spill will be controlled so as not to affect safety-related equipment.

TABLE 15.6-1

SODIUM SPILL EVENTS

Section No.	Events	Sodium Spill Gallons	Sodium Spill Temp (F)	Atmosphere	Location* Bldg.	Location* Cell	Max. Off-Site Dose % of 10CFR100	Max. Cell Gas Press/Temp
15.6	Sodium Spills							
15.6.1	Extremely Unlikely							
15.6.1.1	Primary sodium in containment storage tank failure during maintenance	35,000	400	Normal Air	RCB	Overflow Tank Cell	0.19	0.8 psig 138°F**
					Design Press 10 psig			
15.6.1.2	Failure of ex-vessel sodium cooling system during operation	15,250	600	Inerted	RSB	Ex-Vessel Sodium Tank Cell	0.48	3.8 psig 254°F***
					Design Press 12 psig			
15.6.1.3	Failure of ex-containment primary sodium storage tank	50,000	450	Inerted	SGB/IB	Storage Tank Cell	2.13	3.5 psig 260°F***
					Design Press 4 psig			
15.6.1.4	Primary Heat Transport System piping leak****	35,100 (PHTS Cell)	1015	Inerted	RCB	PHTS Cell	$<10^{-4}$	14.4 psig 680°F***
					Design Press 30 psig			
		29,200 (Reactor Cavity)	750	Inerted	RCB	Reactor Cavity		10.3 psig 650°F***
					Design Press 35 psig			
15.6.1.5	Intermediate Heat Transport System piping leak	39,000	800°F	Normal Air	SGB/IB	IB	3	0.4 psig 630°F***
					Design Press 3 psig			

*RCB - Reactor Containment Building
RSB - Reactor Service Building
SGB/IB - Steam Generator Bldg/Intermediate Bay
PHTS - Primary Heat Transport System

** In Containment
*** In Affected Cell
****Although considered to be beyond the PHTS design basis, the MEFS leak is included in the extremely unlikely category for cell structural evaluations only. See Section 15.3.3.4 for systems effects from leaks.