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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 FOR SODIUM SPILLS PERTAINING TO PRELIMINARY SAFETY ANALYSIS REPORT (PSAR) CHAPTERS 6 AND 15

Enclosed is additional information for the Clinch River Breeder Reactor Plant regarding sodium spills, specifically sodium spills in cell 102A and spills analyzed in PSAR Chapter 15.6. This will be added to the PSAR in a future amendment.

If you have any questions regarding the enclosure, please contact Mr. W. Pasko (FTS 626-6096) or Mr. D. Florek (FTS 626-6188) of the Project Office Oak Ridge staff.

Sincerely,

Robert L. Woolley for

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

Enclosure

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6.2.1.3 Design Evaluation

A spectrum of postulated in-containment sodium fires has been analyzed. Table 6.2-1 summarizes the results of the analysis for the most limiting fire investigated. This parametric analysis of postulated in-containment sodium fires has shown that the most limiting accident with respect to the containment building temperature and pressure retaining capability is the postulated failure of the primary sodium storage tank during maintenance, assuming the tank is full of sodium and the tank cell de-inerted and open to the upper containment volume.

The primary sodium in-containment storage tank is located below the containment operating floor (Cell 102A) in a normally inerted cell. The floor area beneath the tank is 850 ft^2. The cell walls are concrete, nominally 6 feet thick. The interior surfaces of the cell are protected with steel liners.

In the event that major maintenance requires draining of one of the primary loops, the tank will be used to store the sodium coolant. The maximum volume of sodium stored in the tank will be 35,000 gallons and the sodium temperature will be maintained at approximately 400°F. The cell atmosphere will remain inerted.

In order to identify a scenario that could present a challenge to containment integrity, a hypothetical set of conditions must be postulated to exist where the primary sodium storage tank contains its maximum volume of 35,000 gallons of sodium with the tank cell de-inerted and interfacing with the atmosphere of the RCB. The occurrence of this set of conditions is highly unlikely for the following reasons:

(1) Maintenance activities requiring draining of one or more PHTS loops will occur very few times over the life of the plant. Thus, Cell 102A will contain 35,000 gallons very few times over the life of the plant.

(2) If 35,000 gallons of sodium is present in Cell 102A, it will be at low temperature and low (atmospheric) pressure, and it will be pumped in and out of the cell at a relatively slow rate. These factors minimize the likelihood of any significant transient loads on the sodium containing boundary. The likelihood of any leak is minimized.

(3) De-inerting Cell 102A when it contains more than about 4000 gallons of sodium will be prohibited by administrative procedure.

(4) Opening any door or hatch in Cell 102A when it contains more than about 4000 gallons of sodium will be prohibited by administrative procedure. Violation of this administrative procedure is not likely because of the difficulty of opening the doors and hatches of the cell. In addition, there are only three scheduled entries into Cell 102A during the life

of the plant. These entries are made to perform in-service inspection of the reactor overflow vessel and the primary sodium storage vessel. Thus, operators will not be accustomed to opening the cell; it is unlikely they would do so inadvertantly. The "door" to be used for planned entries is a massive concrete structure that must be moved with a fork-lift truck or equivalent. This door opens to Cell 105 which is below the operating floor. Other cell openings are closed even more permanently requiring extensive effort and use of heavy equipment (e.g., cranes) to open. Further, the number of doors and hatches will be minimized to those necessary for anticipated maintenance and repair activities. All of the above factors assure that it is highly unlikely that the Cell 102A would be in communication with the RCB at all. In the analysis, the door to be used for planned maintenance was assumed to be directly open to containment. Though this assumption was unrealistic, it provided an upper bound on the amount of direct communication of Cell 102A with the upper containment and conservatively enveloped conditions that would occur if there was a large sodium inventory in Cell 102A. Any more direct communication between Cell 102A and containment is incredible.

(5) A non-mechanistic instantaneous failure of the primary sodium storage tank must be hypothesized whereby the total 35,000 gallons of sodium are spilled onto the floor of the tank cell with immediate commencement of sodium pool burning.

For purposes of evaluation, it is assumed that the hypothetical accident occurs near the end of plant life thereby maximizing the primary sodium coolant radiological activity. The radioisotope concentrations in the sodium coolant under these conditions are summarized in Table 15.6.1.4-4. The models and assumptions used in computing the coolant activity levels are discussed in Section 12.1.3. In addition the evaluation assumes that the tank cell environment interfaces directly with RCB environment via a hypothetical passageway equivalent in cross-sectional area to a tank cell door (21 $Ft.^2$).

The rate of sodium combustion with resultant temperature and pressure histories in the containment were computed using the GESOFIRE (Reference 1) computer code. The time behavior of the aerosol generated as a result of sodium combustion was computed with HAA-3 computer code (References 2 & 3). Descriptions of these codes are provided in Appendix A.

Sodium fire-burning rate with resulting containment pressures, temperatures and aerosol concentration are shown as a function of time in Figures 6.2-1 through 6.2-5. Table 6.2-1 summarizes the important design values of the containment and the significant results of the analyses. Table 6.2-3 provides an itemized listing of the radioactive constituents of the aerosol resulting from sodium burning.

15.6 SODIUM SPILLS - INTRODUCTION

Postulated sodium fires could possibly result in the dispersion of some radioactive material to the atmosphere. Fires involving primary sodium coolant are of most concern since this sodium circulates through the reactor core and accumulates radioactivity due to neutron activation and entrainment of fission products leaking from defective fuel. Postulated fires involving sodium used in the Ex-Vessel Storage Tank (EVST) Cooling System could also result in radiological releases. The EVST sodium is essentially non-radioactive at the beginning of plant life. However, during refueling a small quantity of primary sodium is tranferred to the EVST along with each irradiated assembly, resulting in a slow buildup of radioactivity in the EVST sodium.

Besides the potential radiological impact of postulated sodium fires, these fires can result in pressure/temperature transients. Therefore, for each fire the consequences are evaluated in terms of: 1) the potential individual whole body and organ doses at the site boundary and low population zone and 2) the pressure/temperature transient in the affected cell/building. The possibility of occurrence of any of the fires considered in this section is extremely unlikely. As such, it will be shown: 1) that the potential off-site doses are well within the guideline limits of 10CFR100, and 2) that the pressure/temperature transient does not exceed the design capability of the affected cell/building.

These fires can also result in pressure, temperature and aerosol challenge to equipment contained in the cell where the fire occurs and any connected cells. These challenges are generally mitigated by providing redundant equipment in a cell which is separate and isolated from the cell where the fire is postulated to occur. For those cases where such separation of redundant equipment is not possible, the environments resulting from sodium fires have been explicitly identified as challenges to be considered in the environmental qualification program. This includes both (1) the environment inside the cell or building in which the fire occurs and (2) the environment resulting from ingestion of the combustion products into other buildings after initial release from the plant.

The computer codes utilized in the analysis of sodium spills and fires are SPRAY-3B, GESOFIRE, SOFIRE-II, SPCA, and HAA-3B. These codes are described in Appendix A with identification of supporting references.

Sodium spills at potential locations other than those discussed in this section have been examined. However the results of these spills were considered to be less severe in terms of radiological consequences and cell temperature/pressure transients and for this reason are not presented.

Since cells containing either primary or EVST sodium are normally closed and inerted, the potential for large postulated radioactive sodium fires exists only during maintenance, when these cells are opened and de-inerted, and sufficient oxygen is available to sustain combustion. A spectrum of fires, both in inerted and de-inerted atmospheres, is investigated in this section.

Tne consistent application of conservative assumptions throughout the analyses presented in this section provides confidence that the consequences of the fires are within the predicted results. A number of these assumptions are generic to all the fires evaluated in this section, and are summarized below:

1. The radioactive content of the sodium is based on continuous plant operation for 30 years. The design basis radioisotope concentrations were assumed present in the sodium for the accident analyses. Included in the basis and discussed in PSAR Section 11.1.5 is a design limit of 100 ppb (parts per billion) for plutonium content of the primary coolant.

2. Retention, fallout, plateout, and agglomeration of sodium aerosol in cells of buildings, whose design does not include specific safety features to accomplish that function are not accounted for in the analysis. Neglecting these factors (an assumption that all of the aerosol is available for release to the atmosphere) leads to over-prediction of potential off-site exposure.

3. No credit for non-safety-related fire protection systems is taken.

4. Dispersion of aerosol released to the atmosphere was calculated utilizing the conservative atmosphere dilution factors (X/Q) applicable to discrete time intervals provided in Table 2.3-38 (the 95th Percentile Values). Guidance provided in NRC Regulatory Guide 1.145 was followed in calculating the X/Q values. Detailed descriptions of the atmospheric dilution factors estimates are provided in Section 2.3.4.

5. Fallout of the aerosol during transit downwind was neglected.

6. The cells will be structurally designed to maintain their integrity under the accident temperatures and pressures and the weight of the spilled sodium. For radiological calculations, no credit is taken for cell atmosphere leak tightness.

7. The cell liners, catch pans, and catch pan fire suppression decks are designated as Engineered Safety Features and will have design temperatures equal to or greater than the sodium spill temperature, thus confining the sodium spill.

8. Both inerted and air-filled cells will be designed to accommodate liquid metal spills resulting from a leak in a sodium or NaK pipe/component in the cell producing the worst case spill/temperature condition. The leak is based on a Moderate Energy Fluid System break (1/4 x pipe diameter x pipe thickness) as defined in branch technical positionn MEB3-1 with the sodium or NaK system operating at its maximum normal operating temperature and pressure.

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

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

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

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

TABLE 15.6-1

SODIUM SPILL EVENTS

Section No.	Events	Sodium Spill Gallons	Sodium Spill Temp (F)	Atmosphere	Location* Bldg.	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 Design Press 10 psig ~~Liner Design Temp 1015°F~~	Overflow Tank Cell	0.19	0.8 psig 138°F**
15.6.1.2	Failure of ex-vessel sodium cooling system during operation	15,250 ~~7,500~~	600 ~~475~~	Inerted	RSB Design Press 12 psig ~~Liner Design Temp 1015°F~~	Ex-Vessel Sodium Tank Cell	0.48	3.8 psig 254°F***
15.6.1.3	Failure of ex-containment primary sodium storage tank	50,000 ~~45,000~~	450	Inerted	SGB/IB Design Press 4 psig ~~Catch Pan Design Temp 1000°F~~	Storage Tank Cell	2.13	3.5 psig 260°F***
15.6.1.4	Primary Heat Transport System piping leak	35,100 (PHTS Cell)	1015	Inerted	RCB Design Press 30 psig ~~Liner Design Temp 1015°F~~	PHTS Cell	$<10^{-4}$	14.4 psig 580°F***
		29,200 (Reactor Cavity)	750	Inerted	RCB Design Press 35 psig ~~Liner Design Temp 1015°F~~	Reactor Cavity		10.3 psig 650°F***
15.6.1.5	Intermediate Heat Transport System piping leak	39,000	800°F	Normal Air	SGB/IB Design Press 3 psig ~~Catch Pan Design Temp 1000°F~~	IB	3	0.4 psig 630°F***

*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

15.6.1 Extremely Unlikely Events

15.6.1.1 Primary Sodium In-Containment Storage Tank Failure During Maintenance

15.6.1.1.1 Identification of Causes and Accident Description

A detailed description of this postulated event is provided in Section 6.2. Section 6.2 includes a complete discussion of the analysis methods and the calculated consequences for this event.

15.6.1.2 Failure of the Ex-Vessel Storage Tank Sodium Cooling System During Operation

15.6.1.2.1 Identification of Causes and Accident Description

There are three Ex-Vessel Storage Tank (EVST) sodium cooling circuits, two forced convection circuits normally used (alternately) to cool sodium circulated to and from the EVST, and one backup natural convection circuit used in the event the normal circuits are unavailable. Each normal circuit is located below grade in the Reactor Service Building (RSB); the backup loop is located above grade. Each cooling circuit is located in separate cells. The pump suction line for each circuit exits from the EVST at an elevation above the normal sodium level in the tank. There are internal downcomers within the EVST which extend down below the sodium level. A remotely operated isolation valve in the pump suction line for the normal cooling circuits is located slightly above the tank outlet elevation.

During operation, all the sodium cooling circuit cells are closed and inerted. The interior surfaces of the cells are protected with a steel liner, 3/8-in. thick. The cell walls are nominally 4-ft. thick concrete. The free volume of the cell is approximately 14,950 ft^3 and the cell floor area is 680 ft^2. The postulated accident is a leak in ~~the~~ a loop pump ~~suction~~ return line in the operating normal cooling circuit in cell 337. In the event of this postulated accident, the other normal or backup cooling circuit would be brought on line to permit continued EVST cooling. The rupture is assumed to occur ~~at the low point of~~ the pump ~~suction~~ line, resulting in the siphoning of sodium down to the bottom of this ~~high point suction line~~ within the EVST. This postulated ~~rupture results in the~~ maximum possible quantity of sodium discharged ~~from the system during operation.~~ Approximately 7500 gal (57,000 lb) of 475°F sodium would be spilled into the cell.

Replace with Insert A

The maximum spill postulated would require a simultaneous major piping failure plus failure of the remotely operated isolation valve (which is located in a separate environment from the spill). As such, the accident is extremely unlikely and is not expected to occur over the life of the plant.

The EVST sodium is essentially non-radioactive at the beginning of plant life. However, during refueling a small quantity of primary sodium is transferred to the EVST along with each irradiated assembly, resulting in a slow buildup of the radioactivity in the EVST sodium. For conservatism, it is assumed that the accident occurs at the end of plant life (30 years) and immediately following a refueling operation when the EVST sodium activity has reached its peak. The design basis radioisotope concentrations in the EVST sodium under

INSERT A

The spill volume in cell 337 assumes a leak in the 4-in. EVST return line from sodium cooling Loop 1, with Loop 1 in operation. The spill volume is based on the loss of the loop inventory and pump out of EVST sodium down to the inlet of the suction piping within the EVST. The leak is essentially constant at an MEFS rate of 6 gpm; sodium temperature is assumed to be 600 degrees F.

these conditions are summarized in Table 15.6.1.2-1. Only those isotopes which make a significant contribution to the radiological content of the EVST sodium are included in the Table. The models and assumptions used in computing the radionuclide concentrations are included in Sections 11.1 and 12.1.3.

600°F

The Design Basis spill temperature is ~~(425°F)~~. The potential radiological consequences of this event are controlled by the extent of radioactive sodium aerosol formation. The aerosol formation is controlled by the limited amount of oxygen available in the inerted (2% O_2) EVST cooling equipment cell. Thus, the radiological consequences are rather insensitive to a wide range of initial sodium release (spray or pool) conditions. This is especially true because no credit was taken for retention, plate-out, or settling of the aerosol in either the EVST cooling equipment cell or the Reactor Service Building. It was conservatively assumed that all the aerosol generated during combustion was released directly to the environment.

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TABLE 15.6.1.2-1

DESIGN BASIS RADIOACTIVE CONTENT OF EVST SODIUM 30 YEARS REACTOR OPERATION

Isotope	uCi/gm Sodium	Isotope	uCi/gm Sodium
Na-24	1.47E+1*	I-132	1.50E-1
Na-22	5.80E-1	Sb-125	8.04E-3
Cs-137	7.10E+0	Sr-90	2.87E-3
Cs-136	4.39E-1	Am-241	5.39E-4
Cs-134	7.10E-1	Am-242m	2.60E-5
I-131	8.90E-1	Cm-244	1.22E-4
Pu-238	6.90E-3		
Pu-239	1.86E-3		
Pu-240	2.42E-3		
Pu-241	1.63E-1		
Pu-242	5.18E-6		
H-3	1.40E-2		

*Peak activity during the fuel handling cycle.

15.6.1.2.2 Analysis of Effects and Consequences

The consequences of this postulated event were determined as follows:

a. The sodium reacts with all the available oxygen in the inerted cell (~2% O_2). The burning releases Na_2O as aerosol.

b. The radioisotope concentrations in the aerosol are the same as the initial concentrations in the sodium.

c. Radioactive decay during the accident is neglected.

d. No credit for retention, plate-out, or settling of the Na_2O aerosol in either the EVST cooling equipment cell or the RSB was taken. It was conservatively assumed that all the aerosol generated during combustion was released directly to the atmosphere.

e. Fallout of the aerosol during transit downwind was neglected.

SPRAY-3B

A analysis of the fire in the inerted cell indicates that combustion is completed (O_2 depleted) in less than 2 hours. A total of 45.4 kg of Na_2O, containing 33.8 kg of Na, is released to the atmosphere as a result of the postulated accident. Release during specific time intervals is as follows:

Time (hr)	Mass Na Released (kg)
0-2	15.8
2-8	17.8
>8	0.2

Even though no credit for aerosol retention in the Ex-Vessel Sodium Tank Cell was taken in the analysis, the cell pressure/temperature history was computed for an evaluation of cell integrity. The results of the analysis indicate a peak cell pressure of only 5.8 psig. This peak occurs 42 hours following the postulated spill. The cell gas pressure decreases to less than 4.3 psig after 90 hours. The cell temperature increases from nominally 250°F to 310°F in 42 hours and then decreases gradually to approximately 135°F 90 hours after the postulated spill.

The results of the radiological assessment are provided in Table 15.6.1. 2-2. The radiological assessment was performed utilizing atmospheric dispersion factors (X/Q) in Chapter 2 of the PSAR.

15.6.1.2.3 Conclusions

The calculated transient cell pressures and temperatures are within the design pressure and temperatures. The offsite radiological consequences are small fractions of the 10 CFR 100 guidelines.

TABLE 15.6.1.2-2

POTENTIAL OFF-SITE DOSES FOLLOWING FAILURE OF THE EVST COOLING SYSTEM

Organ	10CFR100	Dose (Rem) SB (0.42 mi) 2-hours	LPZ (2.5 mi) 30-days
Whole Body**	25	2.59 E-2	5.31 E-3
Thyroid	300	2.20 E-2	4.52 E-3
Bone	150+	7.13 E-1	1.46 E-1
Lung	75+	3.51 E-2	7.20 E-3

*2.59 E-2 = 2.59 x 10^{-2}
+Not covered in 10CFR100; used as guideline values.
**Includes both inhalation and external gamma cloud exposure.

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