Department of Energy
Washington, D.C. 20545
Docket No. 50-537
HQ:S:82:148

DEC 20 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:

ADDITIONAL INFORMATION RESULTING FROM DECEMBER 15, 1982, MEETING ON PLANT AUXILIARY SYSTEMS, PRELIMINARY SAFETY ANALYSIS REPORT (PSAR) CHAPTER 9

In accordance with agreements reached between our respective staffs at the December 15, 1982, auxiliary systems meeting, enclosed are responses to questions (Enclosure 1) and associated amended PSAR pages (Enclosure 2) for the following PSAR Sections of Chapter 9, "Plant Auxiliary Systems."

Section 9.1	See responses in Enclosure 1, amended PSAR Section 9.1 pages (Enclosure 2).
Section 9.2	See responses in Enclosure 1, amended PSAR Section 15.7.3.7 (Enclosure 2).
Section 9.4	See responses in Enclosure 1, amended PSAR Section 9.4 (Enclosure 2).
Section 9.5	See responses in Enclosure 1.
Section 9.6	See responses in Enclosure 1, amended PSAR pg. 9.6-1, 4, and 23.
Section 9.7	See amended PSAR Q/R's CS410.18 and 19 (Enclosure 2).
Section 9.9	See amended PSAR Section 9.9 (Enclosure 2).
Section 9.10	See amended PSAR pg. 9.10-1 (Enclosure 2).
Section 9.15	See amended PSAR pg. 9.15-2 (Enclosure 2)
Section 9.16	See responses in Enclosure 1, amended PSAR pg. 4.2-256 (Enclosure 2).

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The amended PSAR pages of Enclosure 2 will be incorporated into Amendment 75 of the PSAR scheduled in January. Additional information associated with Sections 9.3 and 9.13 will be submitted under separate cover later in December.

Questions regarding these responses may be addressed to D. Robinson (FTS 626-6098) or D. Hornstra (FTS 626-6110) 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

2 Enclosures

cc: Service List
Standard Distribution
Licensing Distribution

Enclosure 1

PSAR Section 9.1

RESPONSES TO NRC COMMENTS

1. Comment: Equipment with active cooling (i.e., EVST, EVTM, FHC, and fuel transfer port cooling insert) should include diesel power provisions or otherwise satisfy clad temperature limits for loss of offsite (normal) power as an anticipated event; the PSAR is unclear with respect to applicability of such a requirement.

   Response: Fuel Clad Failure and subsequent fission product release will result in site boundary doses well below established limits as discussed in PSAR Chapter 15.5. Cooling loops supplied with backup electrical power by diesel generator are provided for EVST sodium and FHC argon cooling. The forced convection cooling system for the EVTM is supplied with normal electrical power but is backed by a natural convection cooling system which can maintain the cladding temperature within its limits. The FHC cooling grapple blowers are supplied with normal electrical power. In the event of an extended loss of power while handling a bare fuel assembly in the FHC, the cladding might be heated to the point of failure. Fission products released would be contained in the FHC because the diesel power-supplied argon circulation system would maintain the FHC pressure negative relative to surrounding areas. Diesel power (from one diesel) is provided to the FHC cooling systems to minimize exposure to operators on a loss of offsite power. However, the FHC boundary is not considered safety-related and credit is taken only for the safety-related RSB confinement to limit the release. The reactor, EVST, and FHC fuel transfer ports have cooling capability provided by blowers supplied with normal electrical power. In the event of a loss of this power and immobilization of a fuel assembly-containing core component pot (the EVTM grapple drive is also supplied with normal electrical power), the peak cladding temperature remains below the clad temperature limit for anticipated events. In any case, emergency power is not required because in case of power failure, the manual drive capability of the EVTM can be used without electrical power to raise or lower a core component pot to a location in which it is passively cooled.

   The enclosed markup of the PSAR revises Section 9.1 to clarify the type of electrical power supplied for each situation in which cooling is needed and the consequences of loss of normal power. The revision consists of a new Table 9.1-2A to list the peak fuel assembly cladding temperature for loss-of-power cases and text in the description of each applicable facility to describe the power supplied and to reference the new table.

Equivalent data for other components can be found as follows:

EVST, paragraph 9.1.3.1.3
EVTM, paragraph 9.1.4.3.3
Fuel Transfer Port, paragraph 9.1.4.7.3

2. Comment: A 2-h station blackout should be assessed (assuming 20-kW fuel and/or blanket in FTP; a bare 15-kW assembly in FHC) with the ensuing radiological evaluation including seal degradation, due to either high temperature or loss of pressure, as appropriate.

Response: A two-hour station blackout while handling a bare fuel assembly during normal fuel handling cell (FHC) operations (assembly decay power less than or equal to 6 kWt) could result in release of fission products to the environment. The potential radiation doses at the site boundary and low population zone distances resulting from such a release would be less than the established limits. Doses calculated for long-term station blackouts are listed in Table 1. The site boundary doses are integrated for a 2-hour period following loss of cooling and thus are applicable for a postulated 2-hour station blackout. The low-population-zone doses are integrated for a 30-day period, and thus envelope those of a 2-hour station blackout. It is noted that station blackout is not a design event for the fuel handling system.

The loss of all station power (both off-site and on-site) could result in release of fission products from a fuel assembly being handled in the FHC. It would disable all FHC equipment, including the cooling grapple (which provides forced cooling of a bare fuel assembly during handling), and the FHC in-cell crane (which transports the fuel assembly between locations where long-term passive cooling is provided). If a fuel assembly were being handled at the time of such a power loss, the assembly would be immobilized in the FHC atmosphere. Cooling would be provided only be natural convection of the FHC argon atmosphere. The heat removal rate by this mechanism is lower than the heat generation rate in the fuel and the fuel assembly temperature would begin to rise. It has been calcubated that the peak cladding temperature would rise to 1500 F in 33 min. At this temperature it is assumed that the cladding would fail, releasing fission products into the FHC atmosphere.

A conservative analysis of the postulated event assumed that fission products from the high-temperature fuel assembly would be released directly to the environment, except for plateout on building surfaces as described in the next paragraph. The two systems which normally operate to eliminate or reduce release to the environment of fission products from the FHC would be disabled by a station blackout. The first of these systems is the FHC argon circulation system (ACS), which removes heat from the FHC argon atmosphere to maintain the cell pressure negative relative to the pressure in surrounding cells. The redundant ACS loops are supplied with power from a standby diesel generator in the event of loss of off-site power. The postulated two-hour station blackout includes loss of this backup diesel power. The ACS would no longer operate to remove fuel assembly decay heat, and the temperature of the FHC atmosphere would rise. The FHC pressure would become positive relative to the pressure of surrounding cells. The FHC liner would remain intact; however, there would be some leakage from the FHC to the atmospheres of adjacent cells. The conservative assumption is made that the FHC liner would provide no holdup of fission products.

The second system which normally operates to minimize radiation releases from the FHC is the reactor service building (RSB) ventilation sytem, which provides RSB confinement in the event of a radiation release. In a station blackout the ventilation fans would be inoperative and the RSB pressure would no longer be maintained negative relative to atmospheric pressure. Fission products released from the FHC into the RSB interior are conservatively assumed to be released directly to the atmosphere.

The building structure is assumed to provide no holdup of fission products released during a station blackout from a fuel assembly in the FHC. All noble gas fission products would thus be released directly to the environment. There would, however, be plateout of volatile fission products on the relatively cold surfaces of the FHC and the RSB interior. It is assumed that 50% of volatile fission products released from a fuel assembly would be plated out before release to the environment. This factor is consistent with the guideline value for iodine releases from LWR design basis accidents used in NRC Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors." The 50% factor is conservative in that it does not consider formation of CsI in the oxygen-depleted atmosphere of the FHC. This reaction would lead to a higher rate of removal for cesium and iodine particulates penetrating the FHC liner. The release of particulate forms of these isotopes would be expected to be reduced to less than 10% of the total amount released from a fuel assembly instead of the 50% assumed.

The analyses were carried out using the SIPOCO aerosol generation code and the procedure in NRC Regulatory Guide 1.25 to determine the integrated radiation doses at offsite locations. The integrated doses to the whole body and to designated body organs are listed in Table 1.

An evaluation of a 2-hour station blackout during handling of a 15kW fuel assembly in the fuel handling cell has not been analyzed. The expected occurrence of handling a 15kW (i.e., fuel assembly greater than 6kW) fuel assembly in the FHC is 3 times in the life of the plant. The combination of this with a 2-hour station blackout is considered to be sufficiently improbable that evaluation of that situation is not required.

TABLE 1

INTEGRATED OFF-SITE RADIATION DOSE
STATION BLACKOUT WITH BARE 6kWt FUEL ASSEMBLY IN FHC

	Integrated Radiation Dose (Rem)			
Location	Whole Body	Thyroid	Lung	Bone
Site Boundary (0-2 hr dose)	0.27	0.46	3.2 (Total) 0.0 (Pu)	0.55 (Total) 0.0 (Pu)
Low Population Zone Boundary (0-30 day dose)	1.5	1.0	17 (Total) 0.004 (Pu)	4.0 (Total) 0.6 (Pu)
Site Boundary Construction Permit Review Limit	20	150	7.5 (Pu)	15 (Pu)

3. Comment: For inflatable or double purged seals, the project should demonstrate that loss of offsite power and loss of purge or inflating gas does not exceed anticipated event guidelines.

Response: There are three types of elastomer seals used on fuel handling machines and facilities: static, dynamic, and inflatable. All of these seals are provided in redundant pairs and have essentially zero leakage (i.e., leakage is almost entirely due to permeation through the seal material). The dynamic and inflatable seals have slightly larger leakage than the static seals on a comparable basis. All three types of seals have a buffer space between seal pairs. The buffer space for static seals is used primarily for periodic leak testing. The effectiveness of these seals does not depend upn the presence of a buffer gas. Dynamic and inflatable seals are provided continuously with a buffer pressure between the double seals. The purpose of this buffer pressure is for leak detection and is not required to prevent seal leakage although it would mitigate an inner seal leak. The inflatable seals are the only ones which depend upon a continuous source of electrical power and inflation gas to perform their function. In case of loss of offsite power, the seal inflation system valves fail open, providing the seals with a continuous source of inflation gas from the normal supply system. In the case of the EVTM, which moves from one location to another this gas supply is from two separate gas bottles, two separate piping lines to separate seals and is independent of the loss of plant gas supply. The argon bottles are standard high pressure bottles that meet DOT requirements and the piping from the bottles to the EVTM inflatable seals is ANSI B31.1. The inflatable seals on the EVTM are on the closure valve. * Insert A

4. Comment: Acceptance criteria and results for offnormal events involving blanket or control assemblies are required (to date, only fuel clad has been addressed).

   Response: The acceptance criteria for blanket assemblies, control assemblies, and radial shield assemblies are less stringent than those for fuel assemblies. To be conservative, the limits for fuel assemblies have been used for all other types of core assemblies.

5. Comment: The project should commit to perform (back up) neutron monitoring to a technical specification during fuel loading (to include number of required operable detectors, calibration frequency, etc.).

   Response: A technical specification (PSAR Section 16.3.10.3.3) is defined by the reactor system for monitoring neutron flux level during refueling. (Calibration PSAR Section 16.6.3)

6. Comment: The Project should address the IVTM grapple impact on assembly flow.

   Response: Calculations have been performed to establish the percentage of the fuel assembly outlet flow area which could be blocked without causing more than a negligible increase in fuel assembly exit temperature under refueling conditions. These calculations resulted in interface requirements placed on the IVTM grapple and holddown sleeve to limit their cross-sectional area in the region of the core assembly outlet. The IVTM grapple is required to provide at least 1.2 in.^{2} of flow area through the outlet of the fuel assembly; the actual grapple provides 1.6[illegible] in.^{2}. The IVTM holddown sleeve, which rests on the six surrounding core assemblies, may not block any more than 50% of any assembly's flow area. Because the holddown sleeve is a simple cylinder, it provides substantially greater flow area than 50%.

   The enclosed markup of the PSAR revises Sections 9.1.4.4.2 and 9.1.4.4.3 to describe the IVTM grapple input on coolant flow through core assemblies.

Insert A

Similarly, the AHM also has two sets of inflatable seals on the closure valve, supplied by separate gas bottles and gas systems. (Note: any failure of the AHM inflatable seal system is enveloped by the accident described in PSAR 15.5.2.4)

The floor valves, located at the reactor, EVST and FHC during operation of refueling equipment, receive electrical power and gas to inflate seals from the EVTM or AHM as appropriate. Prior to motion of the respective machine from the floor valves, the inflation gas is locked into the seals by the respective control valves in the floor valve. A single failure to one inflation system, i.e. failure of the control valve, will only disable one of two redundant seals.

7. Comment: The Project should specify design temperature and pressure of EVST, EVTM, FHC, and their seals; consistency with normal and off-normal accidents should be demonstrated.

Response: The enclosed PSAR markup includes a new Table 9.1-2B to list limiting and maximum calculated pressures and temperatures for the EVST, EVTM, and FHC.

8. Comment: Instrumentation to verify adequate cooling of EVTM and fuel transfer port cooling inserts should be provided.

Response: Instrumentation is provided to verify adequate cooling of the EVTM and fuel transfer ports. Thermocouples are located along the length of the EVTM cold wall and at the cooling air inlets and outlets; the cold wall thermocouples and the air outlet thermocouple will verify adequacy of cooling. Thermocouples are also attached at two places (near the outlet and near the seals) on the reactor fuel transfer port adapter cooling inserts. The thermocouples will indicate the need for cooling and will verify the adequacy of cooling if the adapter blower is in operation. The EVST adapter contains a thermocouple on the inner wall to serve the same function as the reactor fuel transfer port adapter thermocouples. The FHC spent fuel transfer port does not contain instrumentation. The decay power of core assemblies transferred is sufficiently lower than the other ports, that overheating will not occur.

The enclosed markup of the PSAR revises Sections 9.1.4.3.2 and 9.1.4.7.2 to include the above information.

9. Comment: The Project should identify and justify deviations from ANS 57.1 and 57.2 (these standards are required by the SRP for 9.1.2 and 9.1.4).

Response: The ANS Standards 57.1 and 57.2 have been reviewed for applicability to the refueling system. Those requirements which were judged to be applicable have been incorporated into the design of system equipment. Clarification ~~was requested~~ is provided as follows:

a. Pressure Boundary to Radioactive Environ (ANS 57.1, Paragraph 6.1). The referenced paragraph refers to refueling equipment which is part of the primary reactor containment. There is no CRBRP fuel handling equipment in that category. The equipment hatch, which is open for refueling and is the applicable equipment, is part of the containment vessel (referenced PSAR Section 3.8.2.1).

b. Required Interlocks (ANS 57.1, Table 6.2.1). The refueling equipment does not require any safety interlocks as identified in PSAR Section 7.7.1.9. The design of the refueling equipment is consistent with ANS 57.1, Table 6.2.1.

c. Bridge Travel Annunciated (ANS 57.1, Paragraph 6.2.1.4). The EVTM design includes motion alarm horns.

d. Loss of Electric Power Results in Safe Configuration (ANS 57.1, Paragraph 6.2.1.5). The information is added to the PSAR by the enclosed markup of Sections 9.1.4.3.2 (EVTM), 9.1.4.4.2 (IVTM), and 9.1.4.5.2 (AHM).

e. Capability for Emergency Power Disconnect (ANS 57.1, Paragraph 6.2.1.6). The information is added to the PSAR by the enclosed markup of Sections 9.1.4.3.2 (EVTM), 9.1.4.4.2 (IVTM), and 9.1.4.5.2 (AHM).

f. Manual Motor Capability (ANS 57.1, Paragraph 6.2.3.9). The information is added to the PSAR by the enclosed markup of Sections 9.1.3.2.2 (FHC) and 9.1.4.3.2 (EVTM). The AHM has no manual motion capability. It does not handle fuel assemblies or control components, so the referenced requirement is not applicable.

g. Release-Proof Grapple (ANS 57.1, Paragraph 6.2.3.17). The enclosed markup of PSAR Section 9.1.3.2.2 states that the grapple fingers are prevented from operating when supporting a core assembly.

h. Adequate Cooling (ANS 57.1, Paragraph 6.2.4.1.13). The referenced requirement refers to an LWR fuel assembly transfer tube, for which the CRBRP counterpart is the EVTM (including transfers to and from it). Cooling of fuel assemblies in the EVTM is described in PSAR Section 9.1.4.3.2.

i. Position Indication (ANS 57.1, Paragraph 6.2.4.1.14). Position indication is displayed on equipment control panels. The AHM will be positioned visually by movement of the RCB crane.

j. System Needed for Extremely Unlikely Accident Accommodation Shall Do So Assuming a Single Failure (ANS 57.2, Paragraph 4.2.4.5). The EVTM is designed to accommodate a single failure. There is no other equipment in the refueling system which is required to limit the release of radioactivity.

k. 1E Power for Any System Keeping Radioactive Gas from the Environment or for Decay Heat Removal (ANS 57.2, Paragraphs 4.3, 4.3.1(1), and 4.3.1(2)). Response in comment #1 and #3.

l. No Drains (ANS 57.2, Paragraph 5.1.1.1). The design provisions to avoid draining sodium to lower the level below the fuel level are described in Section 9.1.3.1.3, which is included in the enclosed markup.

m. Seismic I Makeup System (ANS 57.2, Paragraph 5.1.1.2). The EVST system is provided with design features which prevent excessive loss of sodium through drain lines (see previous item) and to maintain a minimum safe sodium level even if the EVST itself should rupture. (PSAR Section 9.1.3.1.2 is revised in the enclosed markup to include the EVST guard tank function in the latter case.) Therefore, a makeup system is not needed to accomplish the purpose of the referenced requirement.

n. Provision for Inspection of Fixed Absorbers (ANS 57.2, Paragraph 5.1.12.3). The fixed absorbers in the EVST can be removed using a special tool. This provides the capability for inspection.

o. Rad Monitor on FHM (ANS 57.2, Paragraph 5.4.1). EVTM permanent shielding is provided. As stated in PSAR Section 12.1, personnel are excluded from proximity of transfer port during raising and lowering of fuel assemblies. No permanent monitoring provided for refueling equipment. Radiation monitoring will be conducted by HPs. Section 7.3 of the PSAR discusses radiation monitoring for Containment Isolation. Section 12 of the PSAR, Figures 12.1-1 through 12.1-14 identify the location of area and mobile monitor for the Reactor Service Building and Reactor Containment Building.

10. Comment: Confirmatory monitoring should be provided for the EVST during startup (for example, using a temporary neutron detector) since the calculated 2-sigma upper-bound value of 0.947 for k-effective is close to the established limit of 0.95.

Response: The value of k-effective quoted in the PSAR is for an upper-limit loading of the EVST, assuming the entire 650 storage positions of the EVST are loaded with new fuel of the highest enrichment, which is the worst case. This case also assumes that LWR recycle fuel of the highest plutonium and fissile content is being used. Initial loadings of the EVST will be approximately 167 fuel assemblies. Thus, there will be substantial margin relative to criticality. A capability is available, however, to add a temporary neutron detection system for confirmatory monitoring during EVST loading. The availability of monitoring capability is added to PSAR Section 9.1.2.1.2 in the enclosed markup.

9.1-11 Comment: Can a single failure cause the EVTM to move away from a floor valve without the fuel assembly fully raised and the appropriate floor valves and closure valves closed.

9.1-11 Motion of the EVTM requires that (1) an interlock be satisfied to energize the drive motors and (2) that the operator unlock the seismic locks based upon electro-mechanical indication. The interlock and the imput to permit operator disengaging seismic locks are independent. Hence no single failure will cause the EVTM to be moved until the F/A is fully raised and EVTM is properly sealed.

PSAR Section 9.2

9.2-2 Comment: The appropriate seismic classification of the Nuclear Island General Purpose Maintenance System should be specified.

Response: The seismic category of the decontamination facility, the primary sodium removal and decontamination system, the handling containers and the remote viewing equipment are established consistent with the impact on plant safety related equipment and with their impact of their failure on public safety.

9.2-4 Clarification is required on the analysis of sodium-water reactions in the Nuclear Island General Purpose Maintenance System.

Response: Revised PSAR section 15.7.3.7 is attached.

9.2-5 Comment: Caustic-induced stress corrosion cracking of the process system and the components to be cleaned should be evaluated.

Response: For the stainless steel materials used, the stress cracking regime due to NaOH has a lower temperature boundary of approxiamtely 240°F as shown in attached Figure 2 of article by R.E. Swandby "Corrosion Charts: Guides to Material Selection", Chemical Engineering 69:22, 186-201 (November 12, 1962). Expected process temperatures are below 200°F. Hence, stress corrosion cracking is not considered to be a problem for either the process system or the components cleaned.

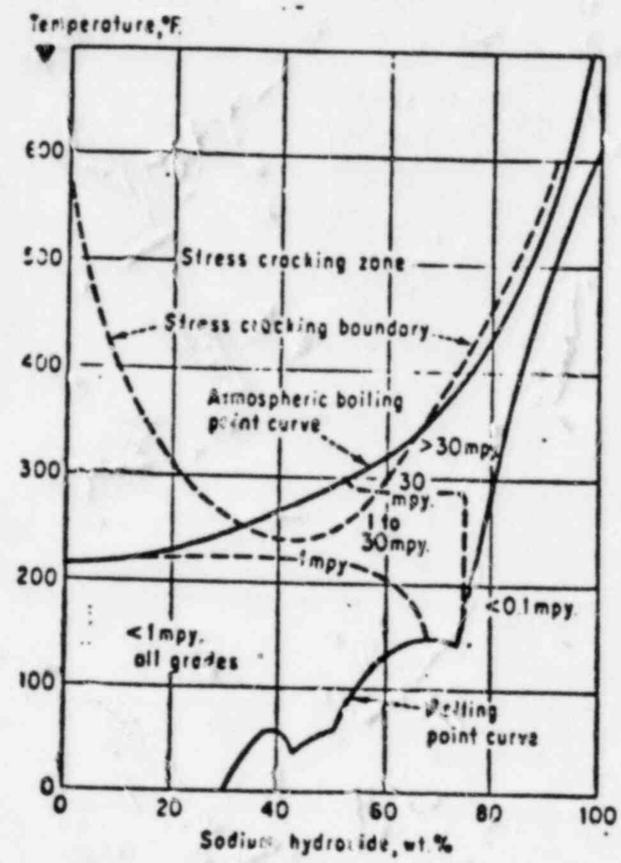

Figure 2. Corrosion of Austenitic Stainless Steels in Sodium Hydroxide

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*R. E. Swandby, "Corrosion Charts: Guides to Material Selection," Chemical Engineering, 69:22, 186-201 (November 12, 1962)

PSAR Section 9.4

Comments: Additional information is required on the trace heating of the following safety related equipment:

Intermediate Heat Transport System Drain Lines
Sodium Water Reaction Product Relief System
  (Piping, Valves and Tanks)
Primary Cover Gas Equalization Line
Overflow Heat Exchanger and associated Sodium and
  NaK piping
Ex-Vessel Sodium Loops
Active safety-related valves
Closure head

Response: Revised PSAR Section 9.4 and 9.5.2-4C is attached (Enclosure 2).

Technical Specifications will be provided in the FSAR for the operation, as appropriate, of trace heated safety related equipment to consider thermocouple and heater malfunctions.

PSAR SECTION 9.5

Question 9.5-1

The portions of the argon system which are extensions of the primary system boundary, the EVST system boundary or the inert gas supplies to any sodium containing component or system should be designed to seismic Category I requirements up to and including the first isolation valve.

Response

Seismic Category I design is provided at least up to (and including) the first isolation valve in the argon supplies to and, vents from, all liquid metal to gas interfaces.

Question 9.5-2

A backup argon supply (such as gas bottles) should be provided for the reactor and EVST to maintain the inert atmosphere on these systems in the event of a loss of normal argon. The capacity of the backup supply should be based upon a reasonable estimate of the duration of a loss of normal argon supply.

Response

Due to the low pressure of the cover gas over the reactor and EVST, and the heavy nature of argon, a continuous supply of argon is not required for purposes of maintaining an inert atmosphere for either.

In both cases, the flowing supply of argon is provided primarily for purging of gaseous impurities. For the reactor, the normal supply is from recycled argon with on-line back-up supply of fresh argon supplied from liquid argon storage via the RCB gaseous argon header. For the EVST, an immediate on-line back-up is not required. The RSB header (normally supplied from its own liquid argon storage) can manually be supplied for back-up purposes through a valved cross-connect to the SGB header. Argon is provided as a buffer gas for seals of both the reactor vessel head and the EVST closure head to detect leaks in the seals. Loss of this argon will not, in itself, cause any cover gas leaks. Argon is also provided as a purge gas in the reactor vessel closure head ~~risers~~ and control rod nozzles to decrease the background radiation in the HAA for ALARA purposes. This purge gas is [illegible]

Question 9.5-3

The design code to be applied in the design of the piping needs to be specified. Also the material for the nitrogen distribution system piping needs to be specified.

Response

Safety classes and safety class changes are shown on PSAR

Figures 9.5-1 through 9.5-10. Design Codes correspond directly (e.g., "SC-2" design code is ASME-III Class 2; "SC-None" design code is ANSI-B31.1). The material for the nitrogen distribution system piping is substantially carbon steel. Stainless steel is specified, however, for those portions of the system containing liquid nitrogen, and up through the first valves on the outlets of the nitrogen vaporizers.

Question 9.5-4

The nitrogen sampling and analysis units should be designed to be seismic II and should be capable of periodic testing and calibration. In addition, the radiation sensor, control and valves which divert the exhaust from the ex-containment inerted cells to CAPS upon detection of high radioactivity should be Seismic I.

Response

The nitrogen sampling and analysis units perform no nuclear safety functions and thus are specified as high quality commercial equipment. Features to facilitate periodic testing and calibration are provided in the design and include connections to introduce known gas and vapor test mixtures.

The potential for direct radiation release thru H&V venting of the ex-containment inerted cells (without diversion to CAPS) is well within 10CFR20 limits. Automatic and manual diversion features to CAPS are provided on the basis of maintaining releases As Low As Reasonably Achievable. Seismic I requirements are not warranted. Radiation detection, in addition to that provided by the nitrogen sampling and analysis unit, is also provided at least twice along any of the potential exit paths through CAPS or the H&V systems. The potential for release is discussed in PSAR Section 11.3.3.3.

Ultimate discharge through the RSB HVAC system will be past a Seismic I radiation sensor as described in PSAR section 9.6.

PSAR section 9.6

RSB RAPS & CAPS HVAC CHANGES (BECOMES CAPS HVAC)

1. RAPS, cells removed from RSB and placed in RCB.

2. Redundent RAPS & CAPS exhaust fans removed, along with associated independent exhaust structure.

3. CAPS cell provided with exhaust (hense negative presure) via RSB Filtered Exhaust System, connected upstream of first class 3 isolation damper.

4. Three unit coolers for CAPS cells rather than one.

5. Rad Monitors indicate high radiation from groups of cell. High radiation will initiate closure of quick closing isolation dampers.

Question

9.6.5 Diesel Generator Rooms HVAC System

1. Describe the design for ventilation of the rooms and associated equipment for the third emergency diesel generator.

2. Provide justification for a fail closed mode for the diesel generator rooms emergency supply fans outside air intake dampers and discharge dampers. Verify that there is no common mode failure source that could result in loss of air flow to the diesel generator rooms when all dampers fail to open. A fail open mode appears to be the proper fail safe position for the above dampers.

Response

1. The ventilation system for each of the three diesel generator cells is typical as follows:

   During normal plant operation (i.e. diesel is not operating) heating and ventilation is provided by an H&V unit to remove the heat load. outside air

   During diesel operation, cooling is provided by two 50% capacity fans which circulate the room air through cooling coils served by the emergency plant service water.

2. The DGB HVAC System has been revised to a recirculating system cooled by emergency plant service water. Outside air is no longer the cooling medium for the HVAC system serving the diesel generators. Accordingly, the outside air intake dampers and discharge dampers have been deleted. which previously supplied that outside cooling air during diesel operation

   It is to be noted that the HVAC system serving Division 1 & 2 switchgears remains as described in <u>Sections</u> 9.6.1.1.2.

   The two diesel generator cells (511 and 512) above the 816' elevation have been removed and relocated to the new diesel generator buildings. The cells below el. 816' remain(with the exception of fuel oil transfer pump cells 526, 527) and are redesignated as being in the electrical equipment building (EEB).

   The switchgear for Division 3 1E power will be located in the SGB and the cooling system serving the area will use outside air and will be powered for the 3rd Division 1E in the event of loss of normal power. Intake and exhaust air ducts will be Tornado and missile protected.

PSAR Section 9.16

Section 9.16, Question 1

The design temperatures and pressures of the subsystems should be made the same as those of the cells which they serve in order to ensure that a sodium or a NaK leak in a cell will not rupture the gas cooling system, even assuming that an isolation valve fails to close. Added assurance of cooling system integrity will preclude opening a path for combustion product release of air in-leakage to the liquid metal.

Response

The design pressure of the subsystems is at least equal to the maximum cell design pressure due to Na or NaK leak.

The piping and system components are located outside the cells cooled, and thus are not directly exposed to the cell environment unless an isolation valve fails to close. The piping design temperatures calculated, used are based upon consider the maximum cell temperature due to Na/NaK leak, the piping and component location is related to cell, whether natural or forced circulation is present, thermal inertia of the system, and thermal conductance of the piping system. In all cases the design temperature will be equal to or greater than the maximum expected temperature.

Section 9.16, Question 2

If control rod cooling from subsystem CR is required to ensure a safety function, then that subsystem should be safety class 3, and a minimum of seismic II (also ASME code III, class 3).

Response

Primary Control Rod Drive Mechanisms are cooled by nitrogen gas, supplied by Subsystem CR of the Recirculating Gas cooling System.

The effect of a failure in any part of these systems to supply this cooling gas has been investigated by a series of tests at W-ARD. The results of these tests were presented to NRC (R. Stark, D. Moran) in a meeting on 10/14/82 and officially transmitted to NRC by DOE letter HQ:S:82:107, J. R. Longenecker to P. S. Check. A summary of these tests and results is presented below.

The PCRDM Loss of Stator Coolant Flow tests were conducted with prototypic hardware in 1000°F sodium flowing at the design flow rate of 45,000 lbs/hr. The PCRCM stator temperature is normally measured by redundant thermocouples located in the outlet of the stator coolant flow. For these tests, additional thermocouples were located in the stator winding to measure the maximum stator winding temperature as a function of coolant flow.

Normal stator coolant flow is 157 scfm N_2 at 95 psig. For these tests the coolant flow was reduced in a series of steps until and coolant flow was zero. At each flow rate, the stator temperature was measured as a function of time until the temperature reached an asymptotic value.

During the stator heatup, the PCRDM was placed in a hold condition with the PCA withdrawn 36 inches. When the stator temperature reached its asymptotic value, the PCA was driven in eleven inches and then withdrawn to 36 inches a total of five times to demonstrate that the mechanism would operate properly with the stator at an elevated temperature. The mechanism was then scrammed and the unlatch time and scram insertion time recorded.

During this test, the stator was held at the design value of 175 volts D.C. As the stator winding temperature increased, the resistenace also increased and the current decreased. Thus power to the stator decreased and eventually an asymptotic condition was reached. This process was aided by radiant heat transfer from the surface of the stator.

At the worst case condition, i.e. complete loss of coolant flow, the maximum stator winding temperature reached 658°F in approximately 260 minutes. In this condition of the PCRDM would run, hold and scram properly. However, once the mechanism had scrammed, the stator winding resistance was too high to latch immediately. There was no loss in scram insertion speed, and at higher temperatures the time to unlatch became shorter.

The only negative effect of high stator temperature on the operation of the Primary Control Rod System was the erratic behavior of the absolute rod position indication (ARPI). This occurs in a narrow band from 12.5 inches to 15 inches when the coolant outlet temperature reached 380°F, approximately 170 minutes after complete loss of coolant flow. Continued operation of a PCRDM without coolant flow would cause the affected zone in the ARPI to increase and potantially cause permanent damage to the ARPI. However, plant procedures call for a consideration to remove power from a PCRDM, which has reached 300°F, and subsequently shutdown the plant. This procedure is intended to prevent permanent damage to the ARPI.

As a result of the above described tests, it is concluded that the Primary Control Rod System can perform its required safety function to shutdown the plant with no loss in performance in the event that there is a complete loss of coolant gas to the Primary Control Rod Drive Mechanisms. Consequently there is no need to upgrade the safety class of Subsystem CR of the Recirculating Gas Coolant System, which supplies cooling for the PCRDMs.

PSAR pg. 4.2-256 (attached) documents that cooling gas is not required to maintain the safety function of the PCRDM.

Section 9.16, Question 3

If the ability of the primary and intermediate heat transport system loops to effect decay heat removal could be adversely impacted by a loss of cell cooling and subsequent, consequent impacts on freeze vent or freeze seal integrity, then the associated subsystems should be updated to safety class 3, seismic catetory I, and should have redundant cooling loops with the capability of being powered by IE power.

Response

Loss of cell cooling has been analyzed for its affect on the freeze vent or freeze seal integrity.

As a result, once the primary heat transport system has been filled with sodium, the cover gas lines to primary heat transport system freeze vents are capped external to the cell to preclude ingestion of cover gas in the event of a freeze vent melt.

Section 9.16, Question 4

A description of the instrumentation providing signals to the isolatin valves, causing them to close, is required, the description should include location, seismic category, safety class and power supply. The staffs position is that the safety class should be the same as that of the cooling system it serves and that all instrumentation should be seismic I and powered by IE power.

Response

PSAR Section 7.6.6 provides the detail description and figures of the instrumentalton and controls provided in the Recirculating Gas Cooling System. This section contains all the information required above. It should be pointed out that the Project has recently changed the failure position of the isolation valves to fail-as-is. Chapter 9.16 and 7.6.6 of the PSAR will be revised in a future ammendment to reflect this.

Section 9.16, Question 5

The power supply and the failure mode of all isolatio valves should be specified.

Response

PSAR Section 7.6.6, provides the information regarding power supply and failure mode of the isolation valves. It should be noted that the Project has recently changed the failure position of the recirculation gas cooling system isolation valve to fail-as-is. This will be incorporated into the PSAR in a future ammendment.

Section 9.16, Question 16

The rationale for automatic closure of isolation valves in RGCS cooling systems (e.g. MA, ME, EA, EB, and FC) where cooling may be required to support decay heat removal should be provided. Alternately, remote manual operation may afford operator flexibility to better ensure safety.

Response

The Project has recently changed the failure position of the recirculation gas cooling system isolation valves to fail-as-is. The valves will close automatically only upon receipt of a leak in the cooling coil or a high return gas temperature. Automatic operation is preferred over remote manual operation in order to provide an additional barrier in the unlikely event either a sodium or water leak to assure no sodium/water reactions can occur.

Enclosure 2

Each system consists of a control assembly, a control rod driveline and a control rod drive mechanism. The control assembly, located in the core, consists of a movable absorber pin bundle called the control rod and an outer duct assembly. The control rod driveline provides a linkage between the control rod and the control rod drive mechanism, located above the reactor closure head, which positions the control rod at appropriate axial core positions.

The control rod systems operate on the principal of varying the neutron absorption in the core by movement of the control rods in and out of the core. The primary system provides a means for starting up the reactor, regulating the power level of the reactor and compensating the reactivity loss due to fuel burnup as well as functioning as the primary shutdown system. The secondary system provides reactor shutdown in the extremely unlikely event of failure of the primary system to shutdown the reactor.

4.2.3.2.1 Primary Control Rod System

The 9 core locations comprising the Primary Control Rod System (PCRS) are shown in Figure 4.3-1. The three major components of primary control rod system are shown schematically in Figure 4.2-100. The PCRS design is essentially the same as the FFTF CRS design in order to maximize the use of the FFTF design, analysis, and testing experience. Figures 4.2-101 through 4.2-104 show the principal design features of
51 the primary control rod system.

4.2.3.2.1.1 Primary Control Rod Drive Mechanisms

Principal features of the Control Rod Drive Mechanism (PCRDM)
51 are shown in Figures 4.2-101, 102 and 103.

The PCRDM is an electro-mechanical actuating device which utilizes a collapsible rotor roller nut drive, and is actuated by signals from the reactor control system. These signals cause the stator to be energized and magnetically actuate the rotor assembly arms, causing the roller nuts to engage the threaded portion of the leadscrew. Rotation of the electrical field of the stator causes rotation of the roller nuts with respect to the leadscrew which is rotationally restrained. This rotation raises or lowers the leadscrew whereas, stopping the rotation causes the rotor assembly to hold the leadscrew at any desired position. De-energizing the stator causes the roller nut to disengage the leadscrew, causing the leadscrew, driveline and the control rod absorber to drop into the core at a rapid rate of insertion (scram). Two independent control rod position indicating systems are incorporated in each control rod drive mechanism.

Heat from the stator is removed by the recirculating gas cooling system (see PSAR section 9.16.2.2). Loss of stator cooling will not inhibit the ability of PCRDM to perform its safety function.

5.2.1.5 Reactor Vessel Preheat

The Reactor Vessel Preheat System will control the dry heat-up and cool down of the Guard Vessel, Reactor Vessel and Internals between ambient (70°F) and 400°F and if required will provide make-up heat for that lost to the Reactor Cavity during prolonged shutdowns.

The heat will be provided by tubular electrical heaters mounted between the Guard Vessel and insulation. These heaters will be arranged circumferentially around the Guard Vessel and will be grouped and controlled in zones of uniform heat output. Temperature sensing devices will monitor the Guard Vessel temperature in each of these zones and provide the necessary feedback for power level adjustments in the heaters.

The heaters will be mounted to the same framework which supports the Guard Vessel insulation. Attachment clips will offset the heaters from the Guard Vessel surface. Convective barriers, reflective sheaths and Guard Vessel insulation will be used to optimize heat input to the Guard Vessel and minimize losses to the Reactor Cavity.

Preliminary preheat, startup, shutdown analyses have been performed on the Reactor Vessel and Guard Vessel to determine the temperature differences which will result in opening and/or closure of the annular gap between the two vessels. By necessity the preheat analysis is very preliminary since no firm preheat procedure has yet been developed. Figures 5.2-4 through 5.2-6 show the temperature differences between the Reactor Vessel and Guard Vessel in the inlet and outlet plenum regions for the three transients in question. As shown the largest positive temperature difference between the Reactor Vessel and the Guard Vessel occurs in the outlet plenum region during startup (335°F) while the largest negative temperature difference occurs in the outlet plenum region during shutdown (-214°F). The nominal radial gap between the reactor vessel and guard vessel is 8 inches at assembly and at the end of preheat. This gap decreases to approximately 7.6 inches minimum during start-up and increases to approximately 8.3 inches maximum during shutdown. During preheat the gap also increases but to a lesser value than during shutdown due to the smaller maximum temperature difference.

Variations in the axial gap between the bottom of the reactor vessel and the inner surface of the guard vessel are noted between the states shown in the table. Thus the largest axial gap is 11.0 inches at the dry cold condition and the smallest gap is 6.2 inches at the end of the heating phase of preheat.

→ Insert Section 5.2.1.6

5.2.2 Design Parameters

Overall schematic views of the reactor vessel, closure head assembly, inlet and outlet piping, and guard vessel are shown in Figures 5.2-1, 1A and 1B. The top view is given in Figure 5.2-2.

Insert

5.2.1.6 Closure Head Heating

The Closure Head Heating temperature control system consists of a single master temperature set point device which is used as the set point reference by the individual heater zone controllers.

The individual heater zone controllers control the temperature of the closure head and reactor vessel support ring based on the temperature reference indicated and the individual zone temperatures as indicated by embedded temperature sensors located within the individual zones. Each zone has multiple temperature sensors. Failure of a single temperature sensor will be detected and alarmed.

The heaters are placed within individual zones to account for heat sinks, i.e., risers, nozzles and other head mounted equipment.

All elements of the head heating and control system are classified as Seismic Category II.

Sufficient redundancy has been incorporated into the design to assure that failures of individual components shall not cause degradation of system performance to the point where the thermal requirements are not met.
A technical specification will be provided to address operations with failed heaters and temperature sensors

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a floor valve is mated to the EVST, and extends from the striker plate top to the cooling insert. The cooling sleeve, shield collar, and floor valve adaptor reduce the transient dose rate from a spent fuel assembly being transferred into the EVST from the EVTM to less than 200 mrem/hr at the surface. The port penetrations through the closure head are stepped to limit radiation streaming through the gaps. In order to allow sufficient time for inspection and maintenance of the main bearings and seals, shielding is provided to attenuate the direct and scattered radiation levels to less than 125 mrem/qtr.

The EVST internals, storage vessel, and guard tank thicknesses are based on structural considerations, but also attenuate radiation in the radial and downward directions. However, the bulk of shielding to reduce radiation levels in adjacent vaults is provided by the concrete vault walls which are discussed in Sections 3.8 and 12.1.

The fuel transfer port plugs in the EVST head have double, static elastomer seals. Large diameter metallic seals are between the storage vessel and the closure head. The operating floor striker plate has a seal at its mating surface with the side wall vault lining. The turntable driveshafts have double, dynamic elastomer seals. All seals in the EVST are double with capability for convenient leak testing by pressurizing the buffer space between seals in a pair. The effectiveness of the seals does not depend on the presence of a buffer gas, although it would mitigate an inner seal leak.

The EVST is designed with sensors and interlocks to prevent any unscheduled movement of the turntable while the EVTM is mounted on the EVST. The interlock allows the turntable to rotate only when the EVTM grapple is in the full up position. The EVST is designed to prevent excessive relative motion between the head and turntable during an SSE.

Temperature instrumentation and sodium level sensing probes will monitor cooling capability. High EVST sodium outlet temperature, and high or low sodium levels will sound an alarm. Other monitors will be provided in the EVST cooling system (see Section 9.1.3). Sodium leak detectors will monitor the space between the storage vessel and the guard tank. An argon cover gas activity monitor will be provided. An area monitor of the gamma scintillation type will measure the gamma radiation on the RSB operating floor above the EVST. The EVST design also includes the capability to add a temporary neutron detection system for confirmatory monitoring during EVST loading.

9.1.2.1.3 Safety Evaluation

The minimum center-to-center separation distance between storage tubes and the 9 storage positions permanently filled with B_4C will keep the storage array subcritical even if the EVST were completely loaded with new fuel assemblies of the highest reactivity. The B_4C neutron absorbers are designed such that they cannot be removed inadvertently, i.e. cannot be removed with the normal refueling equipment. Based on the calculations reported below the K_{eff} of this array, either with sodium or void of sodium, will be less than 0.95, as required.

The system provides the capability to maintain the oxygen content of the sodium in the EVST at, or below, 5 ppm. The cold trap used for this service is separate from those used for reactor and primary loop sodium purification.

The system, working in conjunction with the Primary Sodium Storage and Processing System described in Section 9.3-2, provides a means of removing reactor decay heat in the event of loss of normal heat removal paths. These two systems, operating together, provide the Direct Heat Removal Service (DHRS). The DHRS is sized to limit the average bulk primary sodium temperature to approximately 1140°F when the DHRS is initiated one-half hour after reactor shutdown. Under this condition, all primary pump pony motors are assumed operational. When the DHRS is initiated twenty-four hours after shutdown, the average bulk primary sodium temperature is maintained below 900°F, assuming operation of a single primary pump pony motor. Total heat rejection capability of the EVS Sodium Processing System is based on removal of the required reactor decay heat in addition to the heat generated by spent fuel within the EVST. The maximum simultaneous EVST and reactor decay heat load is approximately 11-1/2 MW, with DHRS initiated one-half hour after reactor shutdown.

9.1.3.1.2 Design Description

The EVST design and operating decay heat loads and sodium coolant outlet temperatures are given in Table 9.1-1.

The major assemblies of the EVST important to decay heat removal, other than the cooling system itself, are the storage vessel, the guard tank and the internals. The internals, specifically the turntable, separate and support the spent fuel assemblies (contained in sodium-filled CCPs) permitting them to be satisfactorily cooled. The structural design of the turntable has already been discussed in 9.1.2.1.

The storage vessel has been classified as Safety Class 2 and is designed, fabricated and inspected in conformance with the appropriate codes and standards (see Section 3.2) to provide a leak-proof containment for the sodium coolant. The sodium level is maintained at a high enough elevation so that normal fluctuations due to changes in temperature or number of stored components do not uncover the top of the CCPs in which the spent fuel is stored. During off-normal conditions, such as a leak or rupture in either the vessel or the cooling system, the vessel sodium outside the CCPs cannot fall below the minimum safe level. This level is defined as that below which fuel cladding temperatures would exceed the limits specified in Table 9.1-2 for the fuel assembly stored at the highest possible location within the storage vessel. The sodium nozzles in the vessel are located in the upper elevations of the vessel wall (see Figure 9.1-6). The EVST sodium inlet lines contain antisyphon devices which prevent a cooling system leak from lowering the vessel sodium below the minimum safe level. The EVST guard tank is sized to contain sodium leaked by the storage vessel and maintain the sodium level above the minimum safe level.

The EVS Processing System includes two independent forced convection cooling circuits, designated circuit Nos. 1 and 2, each of which can remove the required EVST heat loads.

During normal operation, one forced convection circuit is used for EVST cooling and the other is on standby. Each of the circuits is composed of two loops, one a sodium loop and the other a NaK loop. The sodium loop circulates sodium from the EVST through a sodium-to-NaK heat exchanger and back to the EVST. The NaK loop circulates NaK through the exchanger where it picks up EVST heat, to a forced-draft airblast heat exchanger, for dissipation of heat to the atmosphere, and back to the sodium-to-NaK heat exchanger. The system also includes a cold trap to provide purification of the EVST sodium.

The two forced convection cooling circuits are supplied with Class 1E electrical power. Standby electrical power is provided for both circuits in the event of loss of normal power (see Section 8.3.1.1.1). Standby power is supplied to the two circuits by different diesel generators.

In addition, the EVS Processing System includes a third independent natural convection backup cooling circuit designated No. 3 which can also remove the required EVST heat loads. In the extremely unlikely event of loss of both normal cooling circuits, the backup natural convection cooling circuit is used to remove the required EVST heat loads. Sodium circulates from the EVST through a backup sodium-to-NaK heat exchanger and back to the EVST. The NaK loop circulates NaK through the exchanger where it picks up EVST heat, to a natural-draft heat exchanger, for dissipation of heat to the atmosphere, and back to the sodium-to-NaK heat exchanger.

The EVST sodium outlet downcomers within the EVST terminate at different elevations above the stored fuel. Loop #2 (forced circulation) has two outlets; the highest outlet used for normal operation, and a second outlet at a lower elevation such that any sodium leakage from Loop #1 (forced circulation) will not uncover the Loop #2 outlet. Loop #1 has one outlet nozzle located at an elevation between the Loop #2 nozzles. The lower Loop #2 nozzle would be used only in off-normal conditions when both Loop #1 and the higher Loop #2 flow paths will not function. The third (backup) cooling circuit (Loop #3) has one outlet located below all Loop #1 and Loop #2 nozzles such that the Loop #3 outlet will not be uncovered by a leak in either Loop #1 or Loop #2. A leak in the Loop #3 piping will not uncover any of the loop outlets because it is entirely elevated above the minimum safe level in the EVST.

The entire EVS processing system includes the following components:

EVST Sodium Pumps (2)

EVST Sodium Coolers (2)

EVST Backup Sodium Cooler (1)

EVST NaK Pumps (2)

expansion tank is isolated and the EVST NaK pump is increased to 400 gpm each. The cover gas space in the two EVST NaK expansion tanks is cross-connected to equalize tank NaK levels.

9.1.3.1.3 Safety Evaluation

The EVST cooling capability can be provided by either of two identical, forced convection cooling circuits, each of which can remove 1800 kW while maintaining a maximum EVST sodium outlet temperature of .

In the extremely unlikely event that the normal circuits are unavailable, heat will be removed through a third independent (backup) natural convection cooling circuit. At 1800 kW this backup cooling circuit will maintain sodium temperatures within the EVST below 775°F.

The critical temperature in a fuel assembly, from the standpoint of safety, is the peak fuel cladding temperature. The normal and emergency limits are given in Table 9.1-2.

The peak fuel cladding temperatures shown in Table 9.1-2a, are within the limits. Hence, no damage to the stored fuel assemblies will occur.

The codes and standards to which the EVST vessel and the surrounding guard tank are designed and fabricated assure that leakage of sodium will be a very low probability event. At the minimum level, adequate cooling is maintained with no temperature increases from those shown in Table 9.1-1.

Each of the three sodium cooling loops is designed against the possibility of common-mode failure. Two pump suction lines are provided within the EVST for normal sodium circuit No. 2. The open end elevation of each is different, one high, one low. Each of the two lines is separately valved externally to the EVST. After the initial fill of the loop, the isolation valve in the low suction line is locked closed and remains closed (except for periodic testing) throughout the plant life. This low suction line is used only in the event of a major loop or vessel rupture. One pump suction line is provided within the EVST for normal cooling circuit No. 1. The open end elevation of this line is between those for circuit No. 2. This line is valved externally to the EVST, and is called a "high" pump suction line. During normal system operation, one of the normal cooling loops is operated using the "high" pump suction line. The suction line(s) in the standby normal loops are closed. In the event of a major failure (rupture) of the operating normal sodium cooling loop, the isolation valve in the pump suction line is closed by operator action from the control room, signalled by concurrent alarms, indicating low level in the EVST and a sodium leak within the cooling loop cell. If the isolation valve should not be closed the EVST sodium level could only be siphoned to the (high) pump suction outlet within the tank. Siphoning from the return line is prevented by an antisiphon vent in this line within the EVST. If a failure of normal cooling loop occurs, as described previously, the standby normal cooling circuit can be immediately activated by valving in its lower pump suction and increasing pump flow to the design rate of 400 gpm.

In the extremely unlikely event that the second normal loop cannot be activated after the first loop has experienced a failure, the third (backup) circuit will be brought into operation. One suction line is provided within the EVST for the backup cooling circuit. The open end elevation of this suction line is below the lower suction line of normal cooling circuit No. 2. Flow back to the EVST is through the fill/drain line. Siphoning from this return line is prevented because the entire backup loop is elevated above the sodium level in the EVST. The drain line for the Na heat exchanger in the circuit has a removable pipe spool at an elevation above the EVST sodium level to prevent siphoning through this circuit.

Failure of any component, in any of the sodium or NaK loops, can cause loss of only the circuit in which it is located. The normal standby or backup cooling circuit can then be put into operation within minutes to provide essentially continuous cooling of the EVST sodium. The potential radiological consequences of an extremely unlikely release of EVST sodium to an inerted cell is described in Section 15.

All components of the normal sodium and NaK loops which require electrical power are on the Class IE power system, to ensure continuous EVST cooling and reactor decay heat removal. In the event of complete loss of external power to the plant, power to both of the normal cooling circuits is provided by the plant diesels. Immediate activation of the diesel-powered supply is not necessary for the EVST sodium pumps since the sodium volume within the EVST provides a heat sink to minimize sodium temperature rise during loss of circulation. Sodium circulation can be lost for approximately 2 hours before the maximum sodium temperature in the upper portion of the EVST reaches 600°F. Activation of the emergency power supply to the NaK pumps and airblast fans is required within 1/2 hour, however, to ensure the availability of DHRS for reactor decay heat removal.

The only "active" component in the backup loop is the damper on the natural draft heat exchanger. It is operated manually and, therefore, does not require connection to the emergency power system.

Isolation of all of the cooling circuits (sodium plus the associated NaK loop) in separately shielded, inerted cells precludes both radioactive sodium fire and the possibility of any failure in one loop imparing the operability of the other.

is used for transfer and will normally be empty or contain an empty CCP. Normally, the decay power of fuel assemblies handled in FHC will be limited to ≤6 kW each. However, under unusual conditions, it may be desired to examine a short-cooled assembly, i.e., with a decay power greater than 6 kW, but less than 15 kW. Under this condition, no more than a single fuel assembly shall be permitted in the FHC. The transfer station will be designed to cool a single fuel assembly of up to 15 kW decay heat without exceeding the normal cladding temperature limit.

The gas cooling grapple will have sufficient cooling capacity to maintain the cladding temperature of a fuel assembly below the normal cladding temperature limit with a decay heat load of 15 kW.

9.1.3.2.2 <u>Design Description</u>

The spent fuel transfer station shown in Figure 9.1-8 consists of a lazy susan assembly containing three transfer locations for core component pots (CCPs), a bearing and drive system for the lazy susan, a structural support frame and bracketry, heaters, insulation, and seismic restraints for the lazy susan. The transfer station is designed to ASME III/Sub NF3 and Seismic Category 1 requirements.

The spacing between the storage locations is determined such that adequate natural convection cooling is provided.

The lower portion of each storage location is a tapered cylindrical socket to support the CCP while providing a catch basin for sodium drippage. The cylindrical socket houses heaters on its outside which prevent sodium freezing when storing core assemblies with little or no decay heat.

The decay heat will be removed by natural convection to the FHC argon atmosphere, which in turn is cooled by the redundant argon circulation system. Under the worst case conditions the cladding temperature will not exceed 1100°F.

Cooling of the FHC argon atmosphere is provided by the Argon Circulation System, which has two loops, each consisting of a fan, gas heat exchanger and a piped distribution system. The heat exchanger removes heat from the argon gas and rejects it to the recirculating Dowtherm J System which rejects it to the Chilled Water System (Normal or Emergency as applicable) which in turn rejects the heat to the ambient air through the Emergency Cooling Tower in the emergency mode and through the Normal Cooling Tower in the Normal mode. The argon circulation system and supporting heat removal systems operate during normal plant operation, accident conditions, and periods of normal electrical power failure. The Argon Circulation System and Recirculating Dowtherm J System are Non-Class 1E systems supplied with standby electrical power by the same diesel generator (see Section 8.3.1.1.1). The chilled water system loops are Class 1E systems supplied with standby electrical power by diesel generators. One generator serves the argon circulation system loops, also (see Section 8.3.1.1.1). The low-pressure argon system, including the shell of the cooler, is designed to ANSI B31.1, and Section VIII of the ASME Boiler

circulation

and Pressure Vessel Code, Seismic Category I, and is located within a hardened structure.

The crane handled gas cooling grapple, shown schematically in Figure 9.1-9, is mainly used to transfer bare fuel assemblies from the spent fuel transfer station to the spent fuel shipping cask. Design of the grapple finger actuation mechanism prevents actuation of the fingers to release a core assembly while the fingers are supporting the weight of the assembly. The crane hook includes a latch to prevent inadvertent disengagement of a cooling grapple from the hook. In the event of a loss of electric power, the crane will stop at its position at the time of the power failure. Design of the crane includes the capability for manual operation. Access to the crane for manual operation is through ports in the wall and roof closure.

Two redundant argon gas-cooling blowers are mounted on the upper end of the gas-cooling grapple. These blowers draw argon gas from the surrounding cell environment and blow it through the grapple and fuel assembly, discharging it back into the cell through the nozzles at the bottom of the fuel assembly. The argon gas flow rate will be large enough to maintain the cladding temperature of a fuel assembly below the normal cladding temperature limit for decay heat loads up to 15 kW. The blowers are supplied with normal electrical power.

9.1.3.2.3 Safety Evaluation

A CCP containing a fuel assembly is cooled sufficiently by natural convection of the adjacent FHC atmosphere to maintain the peak fuel cladding temperature below the limits given in Table 9.1-2. The peak temperatures, given in Table 9.1-2A for normal operations in which the FHC atmosphere temperature is maintained by the argon circulation system and for the unlikely event of loss of cooling of the FHC atmosphere, are within the limits.

The argon cooling gas flow rate through the spent fuel assemblies while being handled by the gas cooling grapple is sufficient to maintain the maximum steady-state cladding temperature of a 15 kW fuel assembly below 600°F. In the event of loss of argon cooling gas, sufficient time exists for the assembly to be transferred back to a Na-filled CCP in the spent fuel transfer station within the FHC before the fuel cladding reaches 1500°F.

Adequate cooling of a spent fuel assembly suspended from the cooling grapple is maintained by the following means:

1) The grapple blowers are redundant to protect against loss of cooling capability by failure of one blower.

2) Each blower will be tested before beginning FHC spent fuel shipping operations to ensure its operability.

Evaluation of the loss of power for cooling systems for fuel assemblies in the FHC shows that the consequences are acceptable. In the event of loss of normal offsite power, operation of the cooling blowers would stop and the temperature of a suspended fuel assembly would rise. The loss of power would also prevent movement of the FHC in-cell crane to return the assembly to a sodium-filled CCP. The extent of the temperature rise would depend on the decay power of the assembly and the duration of the power loss. During normal operations with the maximum powered 6-kWt fuel assembly, there would be about 33 minutes before the peak cladding temperature would reach 1500°F. If normal power were not restored before the temperature limit was reached, it is assumed that fission products would be released into the FHC. They would normally be retained within the FHC, however, because the argon circulation system and supporting heat removal systems (supplied with electrical power from an onsite diesel generator), would continue operating and maintain the FHC atmosphere at a negative pressure relative to the surrounding cells. In the unlikely event of failure of the diesel generator supplying the argon circulation system, the FHC pressure would become positive relative to surrounding areas and fission products would leak to the building. No credit is taken in accident analysis for the seals of the FHC. This event is enveloped by the event discussed in PSAR Section 15.5.2.3.

the

After removal from the reactor core, the sodium wetted core special assemblies are first brought to the EVST, and are later transferred into the FHC. 162 of these core special assemblies simulate fuel assemblies in the core and have full-flow filters. Some of these assemblies are partially disassembled in the FHC and made ready for sodium removal performed in the large component cleaning vessel. All other core special assemblies are only inspected.

Whenever core special assemblies are handled by refueling equipment, they are accounted for using the same inventory control system as "real" core assemblies. Before entering and after leaving the reactor core lattice they are electromechanically identified by the IVTM using identification notches (see 9.1.4.4.2). In the FHC, core assemblies are identified and differentiated both visually and electromechanically. The core special assemblies leave the FHC in a polyfilm wrapped transfer rack. The outer surface of this polyfilm wrap is checked for radioactivity immediately after sealing and leaving the FHC port. The core special assemblies are transferred from the FHC to the Large Component Cleaning Vessel (LCCV) located in the RCB for sodium removal. Cleaned core special assemblies are packaged in polyethylene bags, loaded into holding transfer racks, and transferred to a storage area.

The physical difference of identification marks between special and real core assemblies, the positive identification of core assemblies at two locations, and the radiological monitoring of core special assemblies before cleaning them are regarded as sufficient safeguards to insure that no real fuel assemblies are mistaken as special ones, and stored in a storage facility not designed to receive them.

The maximum pressures and temperatures of fuel handling equipment during normal operations and off-normal design basis events are listed in Table 9.1-2B. The values are within the limits in the same table.

9.1.4.2 Deleted

9.1.4.3 Safety Aspects of the Ex-Vessel Transfer Machine (EVTM)

The primary function of the EVTM is to transfer core assemblies between the reactor, EVST, new fuel unloading station, and FHC. The EVTM is designed to handle both new and irradiated core assemblies in sodium-filled CCP's and bare new core assemblies. The EVTM has the following capabilities:

1) Grapple and release core assemblies, CCPs, and port plugs
2) Raise and lower core assemblies, CCPs and port plugs
3) Provide containment of radioactive cover gas
4) Maintain an argon environment
5) Maintain preheat temperature for new core assemblies
6) Provide up to 20 kw cooling for spent fuel assemblies
7) Provide radiation shielding

The EVTM is a shielded, inerted, single-barrel fuel handling machine. The EVTM is mounted on a trolley, which, in turn, is positioned on rails on top of the gantry. The gantry moves on crane rails between the Reactor Containment Building (RCB) and the Reactor Service Building (RSB). The trolley rails are perpendicular to the gantry rails, allowing complete indexing of the EVTM. The EVTM mounted on its gantry is depicted in Figure 9.1-13.

9.1.4.3.1 Design Basis

Adequate shielding for radiation protection is provided in the design of the EVTM to meet the radiation protection requirements of 10CFR20.

The activity released from a damaged or leaking spent fuel assembly while in the EVTM is contained in the EVTM by proper sealing or welding of penetrations and openings. Radioactive leakage and diffusion through seals are well below the limits specified in 10CFR100.

Sufficient cooling capacity is provided in the EVTM to cool spent fuel assemblies with up to 20 kw of decay heat in sodium-filled CCPs and to ensure that fuel cladding temperatures do not exceed the values given in Table 9.1-2.

Mechanical damage to fuel assemblies could potentially be caused by the EVTM due to dropping of a grappled CCP or new fuel assembly or by tipping over of the EVTM. Dropping of new fuel assemblies or of CCPs containing new or spent fuel assemblies is prevented by the design of the grapple special-locking fingers and by suitable interlocks. The EVTM gantry and trolley are designed with anti-lift-off restraints and rail stops to prevent derailing of the gantry and trolley under combined normal operating and SSE loads. Mechanical collision between the EVTM on its gantry and other equipment, especially the control rod drive lines, IVTM and equipment hatch between the RCB and RSB are prevented by a combination of stops, interlocks, and procedures.

If a seismic event occurs while the EVTM is mated to a floor valve at the reactor, or other location, the design limits the transmitted structural loads, such that the reactor head or other mating facility is not damaged. Motion of the EVTM relative to the mated facility is limited to prevent contact with, or damage to a CCP, if a CCP happens to pass through the floor valve or the mating facility and the EVTM at the time of the seismic event. Cover gas release is prevented by maintaining sealing of the EVTM to the mating facility during a seismic event.

9.1.4.3.2 Design Description

Axial and radial shielding is provided in the EVTM to limit the dose rate to less than the criteria given in Sections 12.1.1 and 12.1.2 at the surface of the cask body. Shielding is provided over the entire length of the EVTM and is graduated in thickness, being thinnest at the upper end, where the radiation source from the spend fuel assembly being handled is least. Approximately 11 in. of lead shielding is provided at the lower end of the EVTM.

The pressure boundary of the EVTM is sealed using metallic and elastomer seals. The metallic seals are single seals which serve as backup to two of the pairs of elastomer seals. There are three types of elastomer seals: static, dynamic, and inflatable. All of these seals are provided in redundant pairs and have essentially zero leakage (i.e., leakage is almost entirely due to permeation through the seal material). The dynamic and inflatable seals have slightly larger leakage than the static seals on a comparable basis. All three types of elastomer seals have a buffer space between seal pairs. The buffer space for static seals does not depend on the presence of a buffer gas. Dynamic and inflatable seals are provided continuously with a buffer pressure

is used primarily for periodic leak testing. Effectiveness of the seals between the double seals. The purpose of this buffer pressure is for leak detection and is not required to prevent seal leakage, although it would mitigate an inner seal leak. The inflatable seals are the only ones which depend on a continuous source of electrical power and inflation gas for operation. In case of loss of offsite power, the seal inflation system valves would fail open, providing the seals with a continuous source of inflation gas from the normal supply system. (The valves are closed during normal operation to provide more sensitive seal leak detection.) The gas supply is from two separate gas bottles and is independent of loss of plant gas supply. Because the supply valves fail open, loss of offsite power would not affect seal inflation. The piping and valves from the gas bottles to the inflatable seals are ANSI B31.1. The seal inflation system and controls have been investigated to ensure that there are no common cause failures which would disable both inner and outer seals.

The EVTM is hermetically sealed to a refueling station by lowering the closure valve which mates with a floor valve. The actual sealing at this interface is accomplished by elastomer double seals, which are periodically leak checked.

The space between the closure valve and the floor valve is purged by the plant argon supply and vent system, through hose connections made after the EVTM has been mated to a refueling station.

When the EVTM is mated to a floor valve at the reactor or other location, the large bending moments and shear loads in the combined vertical structure due to a seismic event are relieved by structurally decoupling the EVTM from the floor valve at the joint interface. The joint between the extender and closure valve is designed with two sliding surfaces. One of these can experience limited horizontal motion if horizontal earthquake loads exceed a predetermined value, while retaining its vertical load carrying capability. Similarly, the second surface can experience limited vertical movement during a seismic event but retains horizontal restraint capability. All sliding joint surfaces are sealed against each other to provide cover gas containment under normal and seismic conditions.

Cooling of a CCP within the EVTM is accomplished by heat transfer to a cold wall system consisting of an about 8 in. ID sealed cold wall having an array of axial fins attached to the outside. The cooling concept is illustrated in Figure 9.1-15. Heat from the 3-ft high fueled region of the spent fuel assembly is distributed over the 15-ft length of the CCP by natural convection of the sodium in the CCP. The heat is transferred from the surface of the CCP to the cold wall primarily by thermal radiation and secondarily by conduction across a stagnant argon-filled gap. The cold wall is cooled by forced convection of ambient air circulated past the axial cold wall fins. Forced air convection is provided by a blower with the capacity to circulate sufficient air to maintain fuel cladding temperature to less than the normal limit (see Table 9.1-2). In case of failure of the blower, or complete loss of all power, natural convection of air is initiated by automatic opening of butterfly valves just upstream of the blower. Natural convection air flow is sufficient to maintain fuel cladding temperature to less than the limits for unlikely and extremely unlikely events in Table 9.1-2. The EVTM cold wall is part of the EVTM containment boundary and is designed and fabricated to quality and inspection standards corresponding to Safety Class 3 (see Section 3.2).

Instrumentation is provided to verify adequate cooling of the EVTM. Thermocouples are located along the length of the EVTM cold wall and at the cooling air inlets and outlets. The temperatures measured by these thermocouples will verify adequacy of cooling.

A cooling system for a CCP during normal transfers to or from the EVTM is not necessary. Although there is a region in the transfer path in which the effective cooling is less than needed to prevent fuel assembly heating, the time spent traversing this region is short enough that heating is insignificant. In the event that a CCP became immobilized in this region, however, there would be significant heating of both the fuel assembly and the material surrounding the CCP in the port. Therefore, the capability is provided for removal of heat from an immobilized CCP at each of the spent fuel transfer ports: reactor fuel transfer port (see Section 9.1.4.7), EVST fuel transfer port, and FHC spent fuel transfer port. At each port a blower is attached to a cooling duct in the port to circulate building air as a cooling medium. The blower is normally off but would be turned on if a CCP were immobilized in the port. The blowers are supplied with normal electrical power.

The EVTM grapple drive system instrumentation includes display of the grapple vertical position. The system includes the capability for manual operation with a hand crank to allow raising or lowering a CCP to a region of passive cooling in the event of a power failure. A braking system automatically stops the grapple at its position at the time of power failure. If a CCP is immobilized in a region of reduced effective cooling during a transfer to or from the EVTM, with the transfer port cooling blower inoprable because of the power failure, the CCP could be manually raised or lowered to provide passive cooling.

Electric power to the EVTM can be manually disconnected at either the power interconnecting box mounted on the EVTM gantry or at the substation supplying EVTM power.

The EVTM CCP grapple has an interlocking finger design such that, with the CCP engaged with and supported by the grapple fingers, the fingers cannot be retracted even if the entire weight is supported by the finger actuating chain. Redundant support chains are utilized to insure component safety in the event of a single chain failure. The EVTM is transported and positioned by a trolley, traveling on a gantry. The gantry, in turn, travels on rail tracks (see Figure 9.1-13) secured to the RSB/RCB floor. Trolley and gantry wheel truck structures incorporate anti-lift-off and overturning restraints. Trolley and gantry rails are equipped with rail stops plus shock absorbers as positive travel limitations. Collisions between the EVTM and other equipment adequate assurance for both an OBE and SSE that: (a) the composite component will reseat from much greater than maximum anticipated vertical motion; (b) the clamps will prevent disengagement of the extender from the closure valve; and (c) the lip on the closure valve will limit horizontal motion to one inch. The latter is more than adequate to prevent contact with or damage to a CCP that might be in transit through the plane of the slip joint. The seals between the extender and the closure valve and between the closure valve and the floor valve ensure cover gas containment under normal and seismic conditions.

The EVTM cooling capacity of 20 kW is adequate to provide a substantial margin above the maximum normal heat load expected, which is 15 kW. The active portion of the cooling system, the blower, is capable of providing the specified cooling without exceeding normal temperature limits. In case of failure of the blower or loss of all AC power, completely passive cooling is automatically provided by natural convection. In this case, cladding temperature is maintained to less than the limit for unlikely events. The peak fuel cladding temperatures, shown in Table 9.1-2A, are within the limits given in Table 9.1-2. Therefore, no damage to the fuel assembly will occur.

The temperature of a fuel assembly in a CCP will increase if the CCP becomes immobilized during a transfer to or from the EVTM. The peak fuel cladding temperatures are listed in Table 9.1-2A for immobilization in the EVTM valve stack assemblies (the assemblies below the cask body, see Figure 9.1-14) or the fuel transfer ports for the reactor vessel, EVST, and FHC. In each case, the peak temperature is less than the limits given in Table 9.1-2 for the frequency class of the event and no radioactivity will be released from the fuel assembly.

The design heat removal capability of the EVTM has been experimentally verified in EVTM heat removal tests. These tests were planned early in the CRBRP program; their purpose and outline are described in Section 1.5.2.7 of the PSAR.

The tests have been successfully performed, and the test data have been analyzed. References 7 and 8 of Section 1.6 document test evaluations, test descriptions, and experimental data. Following the review of test data, the EVTM heat transfer computer model was modified to consolidate the model predictions with the experimental data.

The main conclusion from these tests is that the EVTM has heat transfer capability adequate to meet its design conditions for both forced and natural air convection modes.

A summary of the tests and major findings is provided below.

Full-Scale Heat Transfer Tests

Full-scale tests (Reference 7 of Section 1.6) were performed in a HEDL test facility design to simulate the cooling systems of the CLEM for the FFTF and the EVTM for the CRBRP. The fuel assembly was simulated by a full scale, 217-pin, electrically heated "fuel" bundle in a hexagonal duct. The fuel assembly was contained in a sodium-filled core component pot (CCP), surrounded by an inert gas-filled annulus, and cooled by the concentric cold wall. The test facility and test article design assured that accurate extrapolation could be applied to test data for either refueling machine. Major test results showed the following:

9.1.4.4.2 <u>Design Description</u>

The steel shield column of the IVTM has a variable thickness radiation shield (from 2.75 to 1.5 in. above the small rotating plug nozzle) to attenuate gamma radiation from radioactive sodium. The amount of radioactive sodium that will be in the upper part of the IVTM's housing is limited to the surface film remaining on the grapple stem after it is raised from the sodium pool during the transfer operations. The active fuel zone of the core assembly which is under approximately 9 ft of sodium during transfer operations, will also contribute to the external dose rate, but to a lesser degree. There will also be a slight contribution to the dose rate from the radioactive cover gas inside the annulus between the grapple stem and the shield column.

The IVTM is sealed to the SRP by 3 elastomer O-ring seals with a buffer region between them. Between each set of seals, a leak detector port is provided to enable connection of a sensor for monitoring the seals integrity before refueling. This arrangement is typical also for most seal sets that involve dynamic motion, such as the grapple stem and holddown drive shafts. Static seals or dynamic seals with very small displacements, such as those for the identifier pawl shaft, are double, with capability for convenient periodic leak testing by pressurizing the buffer region between the seals. The effectiveness of the seals does not depend upon presence of a buffer gas.

The IVTM drive mechanism has been designed to exert an upward or downward load of 5000 lb maximum. Incorporated into the design is a pneumatic load control system and a load cell system that limits the load exerted on a core assembly to 4300 lb pull and 3000 lb push. The load and the grapple vertical position are displayed on the IVTM control console. In the event of a loss of electrical power, a braking system automatically stops the grapple at its position at the time of the power failure. The grapple drive system includes the capability for manual operation to allow raising or lowering a grappled core assembly. Repositioning would not be needed for cooling the assembly. It would be immersed in sodium and thus passively cooled, whenever handled by the IVTM. Electric power to the IVTM can be manually disconnected at either the IVTM control console or the substation supplying IVTM power.

The IVTM is positioned above core assembly locations by rotation of the reactor rotating plugs. The positions of the plugs are displayed on the IVTM control console to define the IVTM horizontal position.

The load in the pneumatic load control system will be set to provide a normal push or pull force on a core assembly of 1000 lb. The pressure of the load control system may be adjusted to provide higher load capability up to the push or pull load limits.

The design provides for the load to be limited in two ways.

1. The primary limitation is provided by the pneumatic load control system. The pressure in the pneumatic load control cylinders limits the load applied by the electromechanical actuator to the driven core assembly. When the core assembly insertion resistance load exceeds the pressure setting in the system, the core assembly stops moving, but the electromechanical actuator continues driving the pistons for about 0.25 inches. This differential travel trips a set of limit

switches which automatically stop the electromechanical drive. Self-contained hydraulic dashpots prevent sudden actuator movements due to sudden changes of the frictional resistance at the load.

2. Load cells are used as backup to the pneumatic system to shut off the electromechanical drive when the preset load limits are exceeded.

Actuation of the grapple fingers for pickup or release of a core assembly is possible only when all the grapple finger actuation interlocks have been satisfied, the grapple is pushing on the core assembly (i.e., core assembly is in full down position), and the load control limit switches that shut off the electromechanical drive are tripped.

released from the IVTM does not exceed the limits set forth in Sections 12.1.1 and 12.1.2. Leak detectors will be used to continuously monitor the effectiveness of the dynamic seals. Periodic leak checks will be carried out for the static seals. In addition, the RCB will have radioactivity monitors to detect accidental releases and sound alarms.

The drive mechanism and grapple design, together with load controls, limit switches, and interlocks will prevent exertion of excessive forces on core assemblies, unscheduled vertical movement of the grapple or disengagement of the grapple fingers except in the full-down position of the grapple. Limit switches and interlocks (see Section 7.6.2) will also prevent inadvertent rotation of the reactor plugs. The extremely unlikely accident of a fuel assembly dropped from the IVTM is discussed in Section 15.5.2.1.

The combination of position interlock switches and the switches of the load control system in conjunction with the design of the discriminator post and matching receptacle ensures proper seating of core assemblies. The fuel and inner blanket assemblies are divided into 8 groups, each having a different configuration of discrimination post inner and outer diameters fitting into corresponding receptacles. This ensures that a fuel assembly of one group can only fit into its corresponding receptacle, see also Section 4.2.1.2.3.

The IVTM grapple and holddown sleeve, when positioned for pickup of a core assembly, partially cover core assembly outlets, reducing coolant flow through affected assemblies. The calculated minimum flow area to ensure a negligible increase in fuel assembly outlet temperature during refueling conditions is 1.2 $in.^2$. The 1.6-$in.^2$ grappled assembly flow area provided by the actual grapple provides for adequate coolant flow through the assembly. A substantially greater flow area is maintained for assemblies affected by the simple cylinder of the holddown sleeve.

Core assembly identification by the IVTM identification and orientation system prior to core assembly insertion into the core, and the position indicator system of the reactor rotating plug, will ensure core assembly insertion into a correct core location.

If, however, a core assembly is erroneously inserted into a core location belonging to another core assembly group, the core assembly discriminator post will bottom against the top of the receptable, thus resulting in an improper seating. The length of the core assembly discrimination post is sufficient to preclude tripping of the grapple finger actuation interlock switches, thereby preventing core assembly release even though the preset force of the load control system has been exceeded. This condition and the IVTM grapple position indication system signal will warn the operator to initiate a corrective action.

Immediately after core assembly removal from the core, the IVTM identification system identifies the core assembly serial number. In the automatic control mode, this serial number is compared by the control system with the serial number designated by the refueling program that must be in that particular core location, thus ensuring that the correct core assembly is removed from the core. In the manual control mode, the operator compares the calculated serial number to the serial number designated in the refueling plan.

If, however, a core assembly is erroneously removed from the core, the serial number discrepancy will warn the operator to return the core assembly into the core position last serviced and to initiate a corrective action.

provide a second, redundant load path from the handling ball to the polar crane hook. When not in use, the AHM is stored at a parking station located in the northeast quadrant of the building.

The parking station is designed for the SSE seismic loads which are carried into the RCB structure.

In the event of a loss of electric power, a braking system automatically stops the grapple at its position at the time of power failure.

Electric power to the AHM can be manually disconnected at either the AHM console or the substation supplying the floor service station from which the AHM is being supplied.

The vertical position of the AHM grapple is displayed on the AHM control console.

When the AHM is in position at the reactor, only the extender mating flange is resting on the floor valve, which in turn is supported from the small rotating plug (SRP) by an adaptor. If the two components were firmly attached to each other, the resulting combined structure, in effect, would represent a tall, vertical cantilever rising from the SRP, attached at its upper end to the polar crane. The large bending moments and shear loads in this combined structure, resulting from horizontal excitation due to an OBE or SSE, are relieved by structurally decoupling the AHM from the floor valve at the extender/floor valve joint interface. At a predetermined horizontal ground acceleration, complete severance of the AHM from the floor valve ("breakaway" concept) eliminates the cantilever beam effect and significantly reduces all seismic loads.

The joint between lower extender flange and floor valve is designed with shear pins which fail upon reaching a predetermined horizontal load. This enables the AHM to separate from the floor valve during a seismic event. The design incorporates a pneumatic reservoir which initiate raising of the AHM extender following the shearing-off of the shear pins. The actuators can raise the extender by about 3 inches in less time than it takes for the extender to clear the floor valve during the horizontal movement due to an OBE or SSE.

9.1.4.5.3 Safety Evaluation

The radial and axial shielding provided by the AHM limits the integrated dose to personnel to less than the maximum allowable dose rate during the installation or removal of the components handled by the AHM. As with the EVTM (see 9.1.4.3) the radiation source in the machine is intermittent and short term.

The AHM has adequate seals to prevent radioactive emissions to the RCB operating floor. Radioactivity released does not exceed the limits as set forth in Sections 12.1.1 and 12.1.2.

The design of the grapple release mechanism, interlocks, and the redundancy of chains will limit the potential for dropping grapple loads. The interlocks are similar to those of the EVTM, and similar to those discussed in Section 7.6.2. The structural support and restraint of the AHM storage facility, and the rigging of the AHM when attached to the polar crane prevent toppling or dropping of the AHM due to an SSE or other loads.

in. diameter opening. The increased thickness of the EVTM floor valve in radial and axial direction provide the additional shielding required for the much higher radiation source which passes through an EVTM floor valve (spent fuel assembly) compared to an AHM floor valve (IVTM port plug).

The stepped upper and lower steel plates of the floor valves, concentric to the valve port, (see Figure 9.1-18) prevent diffusion and radiation streaming through the minimal mating surface gaps. These design features limit the transient dose rate at the surface to less than 200 mrem/hr during transfer of radioactive components, and 5 mrem/hr when closed over the reactor ports.

The floor valve is sealed to the fuel transfer port adaptor by double seals, and bolted to the adaptor flange. The movable circular disk which closes off the port opening in the valves is also sealed by double seals.

The rotor driveshaft is sealed by dynamic seals. All seals are double buffered and of elastomer material. The discussion of EVTM elastomer seals in Section 9.1.4.3.2 is applicable to the floor valve seals also.

The position of the floor valve gate (open or closed) is displayed on the floor valve control panel and, if the EVTM is mated to the floor valve, on the EVTm control console.

9.1.4.6.3 Safety Evaluation

The radial shielding limits the dose rate on the floor valve surface to less than the criteria in Sections 12.1.1 and 12.1.2 during transfer of the highest powered spent fuel assembly (for the EVTM floor valve). The floor valve is considered a piece of equipment whose main function is to permit transfer of radioactive components, both fueled and non-fueled, between a machine and a facility. The radiation source is transient and short term (less than 1 min per transfer) in nature. Hence, it results in a low integrated dose.

Another function of the floor valves is to provide axial shielding to replace that normally provided by the port plugs. The axial shielding limits the dose rate to personnel to 5 mrem/hr when placed over a reactor port and to 0.2 mrem/hr when placed over EVST or FHC ports. Personnel cannot receive a direct axial dose because of the large diameter of the floor valve. In addition, the valve is covered by a mating machine much of the time. In all cases, sufficient axial and radial shielding for the EVTM and AHM floor valves is provided to limit the integrated dose to less than 125 mrem/quarter, and dose rates to the zone criteria of Section 12.1.

The floor valve has adequate seals to prevent excessive radioactive release to the RCB and RSB operating floors. Accidental cover gas release through inadvertent opening of a floor valve in the absence of a mating fuel handling machine (EVTM, AHM) on top of the floor valve is prevented by interlocks. One interlock prevents energizing the valve operating motor unless a mating machine is on top of the floor valve. (Electrical power to the floor valve motor is supplied by connection to the mating machine.) Other interlocks prevent (1) depressurizing the buffer gas zone, and (2) raising the closure valve extender, unless both the closure valve and the floor valve are in their closed positions.

As discussed in Section 15.5.2.4, an unlikely accident releasing radioactive cover gas from the reactor leads to a site boundary dose well below the guideline value of 10CFR20.

9.1.4.7 Safety Aspects of the Reactor Fuel Transfer Port Adaptor and Fuel Transport Port Cooling Inserts

The reactor fuel transfer port adaptor (see Figure 9.1-19) is positioned on top of the reactor fuel transfer port and extends from the reactor head to the bottom of the floor valve which is located at the elevation of the RCB operating floor. It serves as an extension of the reactor cover gas containment and provides shielding when irradiated core assemblies are removed from the reactor. The adaptor also guides cooling air from an air blower to a cooling insert inside and below the adaptor.

The function of the cooling inserts located around the EVST and FHC fuel transfer ports as well as the reactor port (see Figure 9.1-19), is to remove decay heat should an irradiated core assembly in a sodium-filled CCP become immobilized in a fuel transfer port during transfer between the reactor vessel, EVST or FHC and the EVTM.

9.1.4.7.1 Design Basis

The design bases for shielding and radioactive release of the fuel transfer port adaptor are the same as for the EVTM (see 9.1.4.3.1). The reactor, EVST, and FHC fuel transfer port cooling inserts have the capacity to remove decay heat from 20 KW irradiated core assemblies in sodium-filled CCPs to prevent exceeding the 1500°F spent fuel cladding temperature limit specified for unlikely or extremely unlikely events (Table 9.1-2).

9.1.4.7.2 Design Description

The reactor fuel transfer port adaptor extends from the upper surface of the fuel transfer port in the reactor head to the operating floor, see Figure 9.1-19. The upper surface of the reactor fuel transfer port adaptor consists of a flange to which is bolted to the flange of the cooling insert. Shielding is provided by a thick, annular lead cylinder surrounding the adaptor cover gas containment tube over its entire length to limit the dose rate at the shield surface to less than the limits given in Sections 12.1.1 and 12.1.2. The lower part of the adaptor is bolted to the reactor head permanently, while the upper part is installed during refueling only.

The reactor fuel transfer port cooling insert extends from the top flange of the adaptor to the fuel transfer port nozzle. The cooling insert uses a cold wall cooling concept, similar to the EVTM. The CCP containing a spent fuel assembly is cooled by thermal radiation and conduction across the argon gas gap to the cold wall which forms the confinement barrier for the reactor cover gas. Ambient air is blown down the outside annulus of the cooling insert, and discharges into the reactor head access area. Air flow from the blower is adequate to limit the cladding temperature of a 20 KW fuel assembly to less than 1500°F. Thermocouples are attached at two places (near the cooling air outlet and near the seals) on the reactor fuel transfer port adapter cooling insert. The thermocouples indicate the need for cooling and will verify adequacy of cooling if the adapter blower is in operation.

The EVST adapter contains a thermocouple on its inner wall to serve the same function as the reactor fuel transfer port adapter thermocouples. The FHC spent fuel transfer port does not contain instrumentation. The decay powers of core assemblies transferred are lower than for the other ports.

The cooling insert is sealed to the rotating guide tube and the EVTM floor valve to form a continuous reactor cover gas pressure boundary through the adapter during refueling. Sealing to the rotating guide tube is by three static elastomer seals with a continuously monitored buffer region between the center seal and each outboard seal. Sealing to the floor valve is by two sets of double static elastomer seals for which the floor valve provides the sealing surface. The buffer region of each seal pair is continuously monitored to detect a loss of seal integrity.

9.1.4.7.3 Safety Evaluation

The transient dose rate from the highest powered spent fuel assembly is less than the criteria in Sections 12.1.1 and 12.1.2 at the surface of the adaptor body. This significant dose rate exists only during the short time (a few minutes) when a spent fuel assembly travels through the adaptor and floor valve into the EVTM. The closest location where personnel can be exposed to the radiation source is more than 10 ft. from the adaptor surface for normal operation, and more than 2 ft. from infrequent maintenance operations. Both locations are on the RCB operating floor, above the adaptor. Therefore, integrated exposures are low.

Cooling of spent core assemblies in the reactor or EVST fuel transfer port is adequate to maintain the assembly cladding temperature below the 1250°F limit for normal operations and anticipated events. During normal operations, the transit time of the core assembly through the port is short (a few minutes) so that there is no significant heatup. In the event that an assembly becomes immobilized in the port, design provisions to maintain the cladding temperature below 1250°F will be used. If immobilization of an assembly is the result of a mechanical failure of the EVTM grapple drive system, the backup cooling system for the port may be turned on to provide the necessary cooling. If immobilization is the result of a loss of power (which would also disable the backup cooling system), the EVTM grapple drive system may be operated manually to raise or lower the core assembly to a location (EVTM or sodium pool) where adequate passive cooling is provided to maintain the cladding temperature within the 1250°F limit. However, those operator actions are required only to maintain fuel temperatures within normal limits. In the unlikely event that a core assembly becomes immobilized in the port for a longer time by coincident drive system mechanical failure and loss of power or failure of the operator to respond to this condition the cladding temperature would exceed 1250°F but would remain below the 1500°F limit for unlikely and extremely unlikely events.

The seals of the reactor fuel transfer port cooling insert are adequate to prevent excessive radioactive release due to cover gas leakage into the RCB from the reactor fuel transfer port adapter portion of the equipment stackup at the port. Radioactivity released from the cooling insert does not exceed the limits set forth in Sections 12.1.1 and 12.1.2. The pressure in the seal buffer regions is continuously monitored to ensure continued seal integrity. In the unlikely immobilized fuel assembly event seal temperatures increase but remain below the seal limit. One of the sets of cooling insert-to-floor valve seals is in a cooler region than the other set and serves as a backup to the inboard pair.

9.1.4.8 Spent Fuel Shipping Cask

The integrity of the SFSC design will ensure sufficient margins to meet all requirements stipulated in the applicable regulations, especially 10 CFR 71. The shipping cask is discussed in this section only to the extent that conditions to which it is subjected inside the RSB are potentially more severe than those design conditions specified in 10 CFR 71.

Regulation 10 CFR 71, paragraph 71.36, states that the cask design shall withstand a hypothetical accident characterized by a 30-ft drop onto a flat, essentially unyielding, horizontal surface without exceeding a specified reduction in shielding and containment of radioactive material. The LMFBR spent fuel shipping cask will be designed to withstand, with no release of radioactivity, a maximum deceleration of 123 g if dropped 30 ft onto an unyielding surface. The largest height for a potential SFSC drop in the CRBRP is the 72-ft vertical distance of the SFSC handling shaft.

the integrated dose at the same level as the remainder of the reactor head (see Chapter 5.2.1.3).

Activity in the reactor cover gas is contained by plug and cap seals during reactor operation and by adapter and floor valve seals during refueling. Under all conditions, radioactive leakage and diffusion through seals are in conformance with the limits listed in Chapter 5.2.1.3.

Mechanical damage to core assemblies is prevented by control interlocks governing RGT positioning during refueling and the RGT cap locking the RGT in place during reactor operation.

9.1.4.9.2 Design Description

The shield plug is so designed as to limit the total radiation dose rate at the upper end of the RGT to less than 2.0 mr/hr at a distance of 3 feet from the closest accessible surface.

Hermetic sealing is provided by both plug seals and seals in the RGT cap. A means to purge the cap-plug interface volume before removal of the cap is also provided. The pressure boundary consists of metallic seals between the RGT and the reactor fuel transfer port nozzle, elastomer seals for the RGT drive shafts (static during reactor operation, dynamic during RGT movement for refueling), and static elastomer seals in the RGT gear housing which seal to the RGT cap during reactor operation and are not part of the pressure boundary during refueling. The RGT is sealed to the reactor fuel transfer port adapter cooling insert during refueling as described in Section 9.1.4.7.2. All RGT seals are double with a pressurized, continuously monitored buffer region between each pair of seals. The buffer region pressure is for leak detection and is not necessary for seal effectiveness.

Control logic interlocks prevent improper sequences of core assembly-RGT movement whenever the RGT is in use. During reactor operation, the RGT end cap locks the RGT in position and prevents all tube movement. Also, no electrical power is provided to the RGT during reactor operation.

9.1.4.9.3 Safety Evaluation

The RGT, RGT plug, and RGT cap are so designed that refueling and/or operating personnel will never receive a total dose greater than 125 mrem/quarter. (Actual allowed dose and leakage levels are shown in Chapter 5.2.1.3.)

Double seals and a capability of purging the cap-plug interface volume ensure that gaseous radioisotope leakage from all sources to the head area will never cause a dose rate in excess of that given in Section 12.

Control interlocks are designed to prevent mechanical damage to core assemblies contained in core component pots (CCP) and reactor components by preventing the following actions:

1) Inadvertent attempt to insert an assembly in an occupied location.

2) Motion of the RGT with a CCP or grapple extending below the base of the RGT.

3) Any motion of the RGT during reactor operation.

4) Positioning of the RGT over any position except one of the storage/transfer locations.

1. FHC Normal Fuel Handling

In a typical spent fuel handling sequence, a spent fuel assembly in a core component pot is lowered through the fuel transfer port (see Figure 9.1-7) by the EVTM, into the spent fuel transfer station directly below the port. A lazy susan assembly, with three transfer positions supported by a stainless steel gridwork, provides the storage locations. Each position holds one fuel assembly, in a sodium-filled core component pot. Decay heat is removed by natural convection to the FHC atmosphere.

The spent fuel assembly is removed from the core component pot by the in-cell crane, using a gas-cooling grapple, and allowed to drip dry. If for some reason not identified as a part of normal procedures, it is deemed necessary to remove a sodium film from the exterior surfaces, the exterior surfaces will be wiped with alcohol wetted swabs.

Then the spent fuel assembly is lowered into the spent fuel shipping cask located in a shaft below the cell floor. The sequence is repeated for the number of assemblies necessary to fill the shipping cask. The above functions within the FHC are performed remotely by operators in the adjacent operating gallery, and can be observed through the viewing windows.

Normal core assembly handling operations in the FHC are conducted with assemblies having a decay heat of 6 kW or less. Infrequently, it may be necessary to (1) examine a fuel ~~SDD-41 Pg. 1-16.~~1. In this event a single assembly may be handled in the FHC with a decay heat not to exceed 15 kW.

2. Spent Fuel Examination

Spent fuel examination in the FHC is limited to inspecting the exterior surfaces of fuel assemblies to determine their geometrical condition before loading into the spent fuel shipping cask. Spent fuel assemblies will not be disassembled or sectioned in the FHC.

It is planned that only a few selected spent fuel assemblies will be examined, after the plant operation has reached its equilibrium. During the first few refuelings, it is expected that more spent fuel assemblies may be inspected.

The extent of the spent fuel examination covers the following operations, all of which will be performed in the fuel examination fixture:

1) Visual inspection of all exterior surfaces

2) Determination of axial and radial dilation of fuel assembly by measuring its length and distances across flats

3) Measurement of the fuel assembly bow

Radioactive releases and contamination from spent fuel assemblies that are being prepared for shipment in the FHC are contained within the FHC by proper sealing or closure welding of penetrations. Radioactive leakage and diffusion through seals, in the unlikely event of release of the entire fission gas inventory of a fuel assembly, are limited to well below the criteria of 10CFR100.

Criticality of fuel assemblies in the spent fuel transfer station in the FHC is not possible because only three fuel assembly locations are provided.

The spent fuel transfer station design considers all normal loadings in combination with the loads from an SSE in maintaining the necessary physical separation. The FHC roof closure is designed to absorb the load of the heaviest equipment handled by the RSB bridge crane over the FHC: (a) for the main hook, lowered at the maximum crane speed (5 fpm), and (b) for the auxiliary hook, accidentally dropping from the maximum handling height to which it is raised, onto the center of the roof closure without affecting the integrity of the fuel separation lattice. The FHC is located such that heavy equipment not belonging to the fuel handling and storage system is not carried over it by the RSB bridge crane.

The spent fuel transfer station within the FHC is designed so that movement of the lazy susan will not occur while a CCP is being inserted or withdrawn. This design condition prevents mechanical damage to the CCP or its contents.

Monitoring instrumentation is provided for the FHC for conditions that might result in a loss of the capability to remove decay heat, and to detect excessive radiation levels.

9.1.4.10.2 Design Description

The top of the FHC is located at the operating floor of the RSB, as shown in Figure 9.1-7. Sufficient shielding is provided so that the radiation level above the FHC does not exceed the radiation Zone I criteria, see Section 12.1. This shielding is provided by the cell's roof closure assembly, a load-bearing structure which is part of the RSB operating floor. The FHC side wall facing the operating gallery is shielded by high density concrete to protect the operating gallery against radiation dose rates exceeding the radiation Zone I criteria. The other walls and the floor are shielded by conventional concrete to protect the neighboring vaults and the spent fuel shipping cask handling corridor against radiation, see Section 12.1. All windows, and port penetrations through the roof, walls, and floor are stepped to limit radiation streaming in the gaps. The main source of radiation in the FHC is spent fuel assemblies in the spent fuel transfer station.

The pressure boundary of the FHC is sealed using welded seams and elastomer seals. There are three types of elastomer seals used: static, dynamic, and inflatable. All of these seals are provided in redundant pairs and have essentially zero leakage (i.e., leakage is almost entirely due to permeation through the seal material). The dynamic and inflatable seals have slightly larger leakage than the static seals on a comparable basis. All three types of elastomer seals have a buffer space between seal pairs. The buffer space for static seals is used for periodic leak testing. Effectiveness of the seals does not depend on presence of a buffer gas. Dynamic and inflatable seals are provided continuously with a buffer pressure between the double seals. This pressure is monitored continuously for leak detection and is not required to prevent seal leakage, although it would mitigate an inner seal leak. The inflatable seals are the only ones which depend on a continuous source of electrical power and inflation gas for operation. In case of loss of offsite power or gas supply, the seal inflation system valves would fail open, providing the seals with a continuous source of inflation gas pressure. (The valves are closed during normal operation to provide more sensitive seal leak detection.) Since the supply valves fail open, loss of offsite power would not affect seal inflation.

The liner seams on the cell interior walls, roof and floor, and welded penetrations through the FHC walls, roof, and floor are alpha-tight welded and inspected. Fuel transfer ports, the 28-in. maintenance floor valve, window seals, utility penetratons, and slave manipulator penetrations each have double, static elastomeric seals. The 28-in. maintenance port FWDR valve also has double, dynamic elastomeric seals. Sealed cover glasses are provided on the interior side of the window penetrations.

The 28-in. maintenance port floor valve and equipment transfer drawer are sealed by double inflatable seals. The floor valve inflatable seals are active only during periods of valve use. At other times the maintenance port is sealed by a plug with double static seals. The equipment transfer drawer inflatable seals are on the closure doors on each side of the penetration and are pressurized except when a door is open.

The spent fuel transfer station within the FHC is shown in Figure 9.1-8. A maximum of 3 spent fuel assemblies in CCPs can be stored in this interim storage; however, in normal operation, a maximum of 2 fuel assemblies will be stored (1 storage position left empty). The storage positions within the transfer station consist of short, tapered, cylindrical inserts at the bottom, the middle, and the top of a substantial supporting steel grid structure. Each CCP is held in place by the three cylindrical sections. The center-to-center distance of the three storage positions is about 25 in. Each position can hold only one spent fuel assembly in a CCP.

During loading of the spent fuel shipping cask (SFSC), a cask-FHC seal assembly forms a gas-tight extension of the FHC containment to the cask interior. The SFSC is, therefore, connected to the FHC atmosphere and is separated from the air atmosphere of the cask corridor.

The FHC roof closure assembly consists of a large north closure plug (21-ft by 18.5-ft) and a smaller south closure plug (8-ft by 18.5-ft), both joined by a cross beam assembly.

The two closure plugs are composite structures consisting of 34.5-in. thick reinforced concrete, enclosed on four sides by a steel liner, and resting on five 8.5-in. thick steel shield plates. The entire 43-in. thick composite structure provides sufficient radiation shielding and is designed to support normal structural loads as well as the accidental impact loads given in the design bases.

The heaviest load carried over the FHC roof is the EVTM floor valve (9 tons). The lift height of the EVTM floor valve is limited to 2 ft. by administrative controls. All heavy maintenance equipment is transported by the Large Component Transporter (LCT) between the RSB and RCB. Maintenance equipment weighing more than 25 tons and the LCT itself are handled only by the double reeved main hook of the RSB bridge crane. Only maintenance equipment weighing less than 25 tons may be handled by the single reeved auxiliary hook. A load dropped from the auxiliary crane could impact the LCT when it is stationed above the FHC roof. Most or all of the impact energy would be absorbed by the LCT. Seismic restraints prevent equipment loaded on top of the LCT from toppling onto the RSB operating floor or FHC roof during an earthquake. Normal operating procedures require large maintenance loads to be fastened to the LCT seismic restraints before disconnecting them from the RSB bridge crane.

TABLE 9.1-2A

SPENT FUEL ASSEMBLY CLADDING TEMPERATURES
(Sheet 1 of 2)

Location of Fuel Assembly	Frequency Class	Peak Fuel Assembly Cladding Temperature (°F)
CCP in EVST storage location 20 kW Assembly		
Normal operation	Normal	660 ~~[illegible]~~ (1) (510° cool)
Natural convection loop cooling	Unlikely	750 ~~[illegible]~~ (1) (600° cool)
CCP in FHC (15-kWt assembly)		
Argm circulation system operative	Unlikely	1060 (1)
Argon circulation system inoperative	Extremely Unlikely	1265
CCP in FHC (6kW assembly)		
ACS operative	Normal	710 ~~700° (Bottom 710 fuel~~)
ACS inoperative	Unlikely	900°
CCP in EVTM cold wall 20 kW Assembly		
Cold wall blower on	Normal	1240 (1)
Cold wall blower off		
< 30 min.	Anticipated	935
> 30 min.	Unlikely	1350 (1)
CCP immobilized in EVTM stack assembly (assemblies below the cask body assembly, see Figure 9.1-14)		
20 kW Assembly		
< 30 min.	Anticipated	1230
> 30 min.	Unlikely	1415 (1)

TABLE 9.1-2A

SPENT FUEL ASSEMBLY CLADDING TEMPERATURES
(Sheet 2 of 2)

Location of Fuel Assembly	Frequency Class	Peak Fuel Assembly Cladding Temperature (^{o}F)
CCP immobilized in reactor fuel transfer port		
20 kW Assembly		
Blower on		
< 30 min.	Anticipated	950
> 30 min.	Unlikely	1444(1)
Blower off (< 30 min.)(2)	Anticipated	970
Blower off (> 30 min.)	Extremely Unlikely	1482
CCP immobilized in EVST fuel transfer port		
20 kW Assembly		
Blower on		
< 30 min.	Anticipated	<1250
> 30 min.	Unlikely	1389(1)
Blower off (< 30 min.)(2)	Anticipated	<1250
Blower off (> 30 min.)	Extremely unlikely	1482
CCP immobilized in FHC spent fuel transfer port		
15 kW Assembly		
Blower on		
< 30 min.	Anticipated	<1225
> 30 min.	Unlikely	1249(1)
Blower off (< 30 min.)(2)	Anticipated	1225
Blower off (> 30 min.)	Extremely Unlikely	1275 (2 hrs)

(1) Steady-state temperature

(2) For a stopped CCP with the fuel transfer port blower inoperative, the operator is required to take action as described in PSAR Section 2.1.4.3.2 within 30 minutes to raise or lower the CCP.

TABLE 9.1-2B

TEMPERATURE AND PRESSURES FOR FUEL HANDLING EQUIPMENT

	Design Limits		Maximum Calculated Values for Normal Operations and Off-Normal Design Events			
	Pres. (Psig)	Temperature (^{o}F)	Pressure (Psig) Normal	Pressure (Psig) Off Norm.	Temperature (^{o}F) Normal	Temperature (^{o}F) Off Normal
EVST	15	775 (off normal)	10" WG	8	535	625^{o}
EVTM	15	435	10" WG	11.5	400	435^{o}
FHC	+2.5	225	-3" WG	+1.44	110	185^{o}
Inflatable Seals	50	250	30	50	ambient*	250^{o}**
EVST Cooling Sleeve Seals	-	500	-	-	110	500^{o}
EVST Port Plug Seals	-	250	-	-	142	183^{o}
EVTM Static Seals	-	250	-	-	80	250^{o}
RFTP Seals	-	500	-	-	182	242^{o}
FHC Static Seals	-	250	-	-	110	185^{o}

* Closure valve and floor valve transfer of a 20KW assembly could raise the seal temperature to 150^{o}F.

** Closure valve and floor valve steady state (more then 12 hours0 270^{o}F.

9.4 PIPING AND EQUIPMENT HEATING AND CONTROL

9.4.1 Design Basis

The Piping and Equipment Electrical Heating and Control System provides the electrical heaters, electrical heater mounting hardware, heater power controllers, related temperature measuring and controlling instrumentation, and equipment required to heat the following sodium containing systems and components:

Reactor Enclosure *

Reactor Refueling (Storage Tank)

Reactor Heat Transport (Primary and Intermediate) Systems

Steam Generation System (Dump Tanks and Sodium Water Reaction Products Tanks)

Auxiliary Liquid Metal System

Inert Gas Receiving and Processing System

Sodium Impurity Monitoring System

This heat is required to preheat those sodium process systems prior to fill, to prevent sodium freezing when systems heat sources such as reactor decay heat and pumping heat becomes insufficient, and to maintain pre-established temperature differences in

*The trace heating system which services the reactor vessel head, control rod drive lines, and vessel support area is discussed in Section 5.2.1.6.

the system.

To perform the dry heat-up function, the electrical heating system shall be capable of preheating the sodium process systems from ambient temperature to any temperature between ambient and a maximum of approximately 450oF before the system is filled with sodium. The heating requirements for each trace heated component in the above systems will be determined by the particular sodium process system.

The electrical heating system shall also be capable of providing the applicable heatup rate for the particular system or components when filled with sodium and of holding process system temperatures when filled with sodium. Heat provided by this system can be used to melt frozen sodium in piping or components. Freezing of sodium in major systems or components is considered unlikely and is an abnormal event.

The temperatures of all measured points shall be indicated locally and in the Main Control Room. The thermocouple used for monitoring shall be separate from the thermocouple used for control. Thermocouples leads as a matter of good design practice shall use different paths from heater power conductors. The electrical heating, temperature monitoring, and temperature control lines are all non-safety related. As such, the electrical heating, monitoring, and control lines shall be separated from all Class 1E lines per IEEE std. 384-1981.

All piping and equipment in inaccessible areas shall be provided with spare heaters and thermocouples. The spares shall be accessible in a junction box in a man-safe location. All heater circuits shall be provided with ground fault protection. All instruments and controls shall be testable in place.

All components of the Piping and Equipment Electrical Heating and Control System shall remain operational for an OBE (i.e., qualified Seismic Category II). In addition, no component of this system shall, as a result of an SSE, impact in any way the performance of the safety function of the piping and equipment heated by this system. The heaters and thermocouples shall withstand the vibration forces of the components to which they are attached without failure or impact on any safety related function.

The heater physical mounting arrangement and the electrical protection of the heater circuitry shall be designed to preclude damage to the components being heated. The Piping and Equipment Heating and Control System has been designed and applied such that it is not a system which is important to safety. Safety related components which require the application of heat from the Piping and Equipment Electrrical Heating and Control System have also been designated such that any combination of failures, single or gross, of the non-safety related Trace Heating System will not compromise the safety related function of the equipment. (e.g., cover gas equalization lines, overflow heat exchangers, ex-vessel storage tank).

9.4.2 System Description

The electrical heating and control system provides power to the tubular heaters or mineral insulated (Mi) heating cable mounted on the piping and/or components of the systems indicated in Section 9.4.1

The location of thermo couples and heaters on components and piping shall be such that leak detection, loose parts monitoring or operability of inservice equipment is not impaired or prevented.

The heat rates, required by different components are controlled by using thermocouples to monitor piping and component temperatures and to adjust the power supplied to the heaters by means of three mode proportional temperature controllers and solid state relays.

Alarm is provided to the Main Control Room in the event of the following:

1) Setpoint Deviation (Control thermocouple compared to temperature setpoint)
2) High Temperature (Monitor thermocouple compared to a high or low temperature setpoint)
3) Low Temperature
4) Open Thermocouple (Both control and monitor thermocouples)
5) Open Circuit Breakers
6) Load Loss over 10% (Detects a single heater failure)
7) Data Transmission Failure

Tubular heaters apply heat via a spiral wound nickel-chromium alloy resistance wire insulated from its containing metal tubular sheath by tightly packed magnesia (MgO) powder. Several inches on each end of each heater are unheated having a heavy electrical conductor to the electrical termination. In certain cases, i.e., selected piping smaller than eight inches O.D., the heat is applied by mineral insulated heating cable that consists of a metal sheath drawn down over a MgO insulated single heating element.

Separate Chromel-alumel thermocouples are used throughout the systems for the feedback signal to control the operation of the electric heaters and for monitoring the temperature of the metal boundary of the sodium containing piping and equipment. Furthermore, separate signal processing/indication is provided for in the control and monitoring thermocouples. Thermocouple compensation is provided for all thermocouples.

Thermocouples on piping are located at a point on the pipe to enable control of the average temperature of the pipe within specified limits. On equipment, the thermocouples are located in the spaces between heaters for both monitoring and control purposes.

Control of any heater or bank of heaters is by automatic control. This control provides for continuous and automatic adjustment of heat based on an error signal generated from the difference between the temperature setpoint, as set by the plant operator, and the temperature feedback signal from the thermocouple.

The controller compares the temperature control setting (ramp rate in heat-up mode and setpoint in hold mode) as set by the plant operator to the actual temperature of the sodium process metal as measured by a thermocouple and generates an error signal. The error signal is converted into a corresponding "on" to "off" ratio of voltage which is applied to a solid state relay which controls the AC power to the heaters.

The required power is controlled by conducting a fraction of the time over a 17 second period. For example, 50 percent power would be conducting for 11.5 seconds and off 11.5 seconds, 90 percent power would be on for 15.3 seconds, and 10 percent power would be conducting for 1.7 seconds.

Heaters are arranged in a particular control circuit according to the uniformity of heating required by a bank of heaters. This type of heat application is called zoning. A heater zone is an area that can be heated with the same unit input and can be controlled from a single temperature indicating point that is representative of the zone.

The temperature feedback thermocouple is located in a representative position within the pipe run or area within the heated zone. The monitoring thermocouples are located in different areas of the zone from the feedback thermocouple to provide independent checks on the zone temperature.

All heaters are in operation continuously during dry heat-up (system completely empty). Some heaters will be in operation continuously for the occasional fill and drain situations in some piping and components such as cold traps, dump tanks, ex-containment storage tank guard vessel, gas equalization lines, and other components. For all other normal operations (start-up, hot standby, and shutdown) the heaters will be in operation only intermittently to make up for the heat loss through the insulation.

A dedicated, pre-programmed, direct digital control system is provided for the Reactor Containment Building, Steam Generator Building, and Reactor Serivce Building. The system is modular to permit physical distribution of the various functional

From DOE Letter of June 2, 1982

NORMAL CHILLED WATER SYSTEM

Question CS410.18 (9.7.3)

The normal chilled water system is a non-safety related system with some Seismic Category I piping. Verify that the normal chilled water system piping and equipment which is located in cells containing sodium or NaK piping is designed to Seismic Category I criteria.

Response

With the exception of the SGB loop cells, all normal chilled water piping and equipment are located outside the cells containing sodium or NaK piping.

~~For the SGB loop cells the normal chilled water piping is currently designed to Seismic Category III. This seismic classification is currently being evaluated to determine if Seismic Category III piping in the SGB loop cells is adequate. The results of this evaluation will be provided in a future amendment.~~

For the SGB loop cells, the normal chilled water system within the cell Piping and eq[illegible] designed to seismic Category I criteria.

components and to facilitate future expansion and the upgrading.

The panels and Operator Control Centers are located in these three areas of the plant. An additional Opertor Control Center (Master) is located in the Main Control Room. The system arrangement is shown in Figure 9.4-1.

9.4.3 Safety Evaluation

Since the trace heating system is not important to safety, the heating system components are non-class 1E. Therefore, as required by Federal Regulations, the trace heating system design features which make operation highly reliable are excluded from safe shutdown scenarios. In fact, the safe shutdown scenarios must assume gross combinations of heating system failures. This is required without consideration to the design reliability of the non-safety electrical trace heating components. Under these exteme conditions, the apparent failure mechanisms of the trace heating and control system are lack of heat when heat is required, heat when none is required, and current flow through piping and other non-wiring components due to shorts concurrent with multiple failures of the over current protection components. The effects of these potential failures on the safe shutdown of the plant is discussed in this section.

As discussed in PSAR Section 3.~~8~~2.3, the Piping and Equipment Electrical Heating and Control System is not safety-related. The heating system is not essential for the safe shutdown of the reactor, nor will failure of the system result in a release of radioactive material. In those cases where heaters are applied to safety related components, the heaters are not required for the components or the associated systems to perform their safety function.

The loss of vent and drain line trace heating does not compromise the safe shutdown function. In the event of a large sodium/water reaction (SWR), the water side of each evaporator module in the affected loop is dumped and the superheater steam side is vented from the superheater outlet safety relief valves; therefore, no drain of sodium through the

IHTS drain lines is necessary to mitigate the SWR, and these drain lines need not be maintained hot for safe shutdown after this event. If the gas feed and vent lines close, the range of sodium level variation in the PHTS pumps is limited by the sodium level sensing and gas isolation system (qualified for SSE and in-containment sodium fire DBE and provided with Class 1E power) to a safe operating level for shutdown.

In addition, heat transport system temperatures remain above the temperature at which trace heating is required to prevent sodium solidification during the time when the Shutdown Heat Removal System operability is required.

The unwanted additional heating of sodium lines (sensed and controlled normally) due to multiple failures in the trace heating and control system which turn on trace heaters (which should be off) is less than five percent of the long term subsystem heat removal capability per loop. The unwanted heat is less than one percent of the short term subsystem heat removal capability per loop, and it is less than one-half of one percent of the total plant power removal capability. These percentages are sufficiently small in terms of heat transport system capability that the occurrence of this failure mechanism would not compromise the safe shutdown function.

The third potential failure mechanism is a short to a non-wiring component occurring with concurrent failures of the ground heater sheath, the ground fault detectors, and the over current protective devices. This mechanism would not compromise the safe shutdown ~~of the plant.~~ The smallest pipe where a short could occur is greater than ten times the cross-sectional diameter of the electrical wiring. Therefore, for the smallest pipe, the conductivity of the electrical wiring is one-half the conductivity of the pipe, and the conductivity of the pipe is over forty times higher than the heater wire. In either a short or an arcing situation, the pipe would not fail.

Operationally, the failure mechanism requires the failure of the temperature sensing system and one of the following: (1) excess current application, (2) cross-over in mounting of adjacent heaters, and (3) improper setting of protective devices. For design related failures, the failure mechanism can be caused by improper heating wire design, fissures in the magnesium oxide, and bends less than the minimum bend radii. The effect of the failure will not cause failure of the sodium containment.

9.4.4 Design Reliability Evaluation

In order to prevent the effects of heater failure from propagating to the piping or equipment to which heaters are attached, the following operational criteria are used:

(1) For normal operation, the heaters are operated at less than 1/2 rated power. For abnormal operation, each heater control circuit is protected against overcurrent by thermal overload circuit breaker and temperature sensors on the heated component. Ground fault interrupters (GFI) will be used for protection against ground currents.

(2) High and low temperature alarms are provided for all control and monitor thermocouples in all heater control zones.

(3) The cold ends of the heaters are bent 90° and brought out from the component. A spacing is maintained between adjacent heaters to prevent crossover of heaters and significant mutual heating by radiation.

(4) The proper setting of the GFI units will be set at installation.

~~(5) For heaters mounted on stand-offs, separation is maintained between heater sheath and piping or components.~~

5
(~~6~~) To prevent heater failure from design considerations, the heaters are designed to a high quality standard. The use of the standard requires that heaters be radiographed. In addition, the technical, mechanical, electrical, material, fabrication, and quality assurance requirements are specified.

9.4.5 Tests and Inspections

The design of the electrical heating system permits periodic testing to confirm the operation of the ground fault detection system and heater control system. The heater control system will be tested and inspected at installation and at refueling periods as dictated by application. Inspection of the heaters in accessible areas following shutdown will be performed according to the maintenance requirements of each process system. Redundant heaters* wired to accessible terminal blocks will be provided in inaccessible areas as required.

INSERT C

9.4.6 Instrumentation Application

Instrumentation application is discussed in Section 9.4.2.

* Redundant heaters are nonoperating installed spares which can be made to operate in place of failed heaters.

INSERT C

In addition, the trace heating system will be tested prior to startup following an earthquake of intensity greater than or equal to the OBE.

Table 9.4-1

46 SAFETY-RELATED COMPONENTS REQUIRING STAND-OFF ELECTRIC HEATERS

Component

Reactor Guard Vessel

Ex-Vessel Storage Tank Guard Vessel

Primary Heat Transport System

- Main Piping
- Pump Guard Vessels
- Intermediate Heat Exchanger Guard Vessels
- Appendage Piping

Intermediate-Heat Transport System

- Main Piping
- Pumps
- Expansion Tanks
- Appendage Piping

Steam Generator System

- Appendage Piping

Auxiliary Liquid Metal System

- Overflow Heat Exchanger
- Primary Sodium Overflow Vessel
- In-Containment Sodium Storage Vessel
- Primary and EVST Cold Traps
- Overflow Line
- Piping in Reactor and EVST Cavities
- 46 Piping in Primary and EVST Cold Trap Cells

Inert Gas Processing System

- All Safety Class 1, 2, 3 Components, Piping and Equipment

Impurity Monitoring and Analysis System

- All Safety Class 1, 2, 3 Components, Piping and Equipment

Amend. 46
Aug. 1978

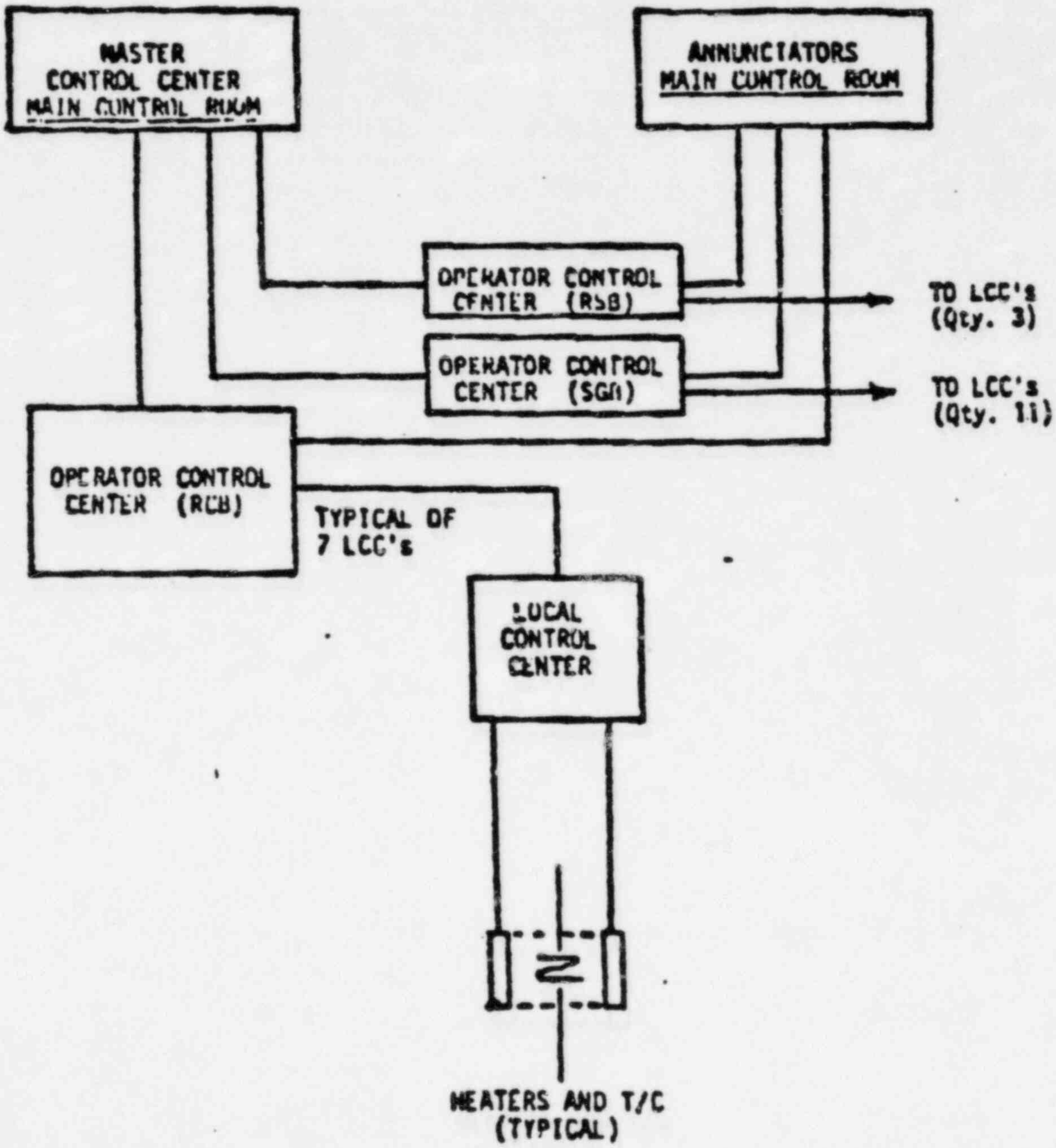

Figure 9.4-1
Power and Control System Arrangement

9.6 HEATING, VENTILATING, AND AIR CONDITIONING SYSTEM

9.6.1 Control Building HVAC System

9.6.1.1 Design Basis

9.6.1.1.1 Control Room HVAC System

The Control Room HVAC System is a safety related system designed to provide filtered and conditioned air to the Control Room as required to permit continuous occupancy of the Control Room and to ensure the operability of all Control Room Equipment and instrumentation under all conditions. The Control Room HVAC System is designed to:

a) Maintain a positive pressure in the Control Room to minimize the infiltration of radioactive or chemical contamination.

b) Prevent the increase of internal airborne radioactivity over the limitations set forth in 10CFR20.

c) Permit continuous occupancy of the Control Room during all operating modes in compliance with the CRBRP General Design Criterion 17 (defined in Section 3.1).

d) Maintain space temperatures within the Control Room at approximately 75^{o}F during all modes of operation.

e) Permit purging of the Control Room following a fire.

f) Comply with the single failure criterion.

g) Operate from the Class IE AC power supply during loss of offsite power.

(h) Comply with the guidance of Regulatory Guide 1.52 and 1.78.

9.6.1.1.2 Control and Diesel Generator Building Emergency HVAC Systems

The Control and Diesel Generator Building Emergency HVAC System is a safety related system designed to provide filtered and conditioned air under all conditions to the Control Building, Battery Cells, Battery Maintenance Area, Upper and Lower Cable Spreading Rooms, Vital AC/DC Rooms and the Diesel Generator Building Class IE Switchgear Rooms. The system provides the required environment to permit personnel access during normal plant operation and to ensure operability of the equipment under all conditions. The HVAC system serving these areas is designed

49 to:

HEPA filters are capable of removing a minimum of 99.97 percent
thermally generated dioctylphthalate particulate of uniform 0.3μ droplet
size at the design flow rate of 8,500 CFM.

The charcoal filter bed is assumed to remove 95 percent of
airborne radioactive elemental iodine and 95 percent of methyl iodine at
relative humidities below 70% at the design flow rate of 8,500 CFM. The
actual tested efficiency of the charcoal bed in removing elemental iodine
is 99.9% and 99.5% in removing methyl iodine.

The Filter Unit Supply Fans are connected to their respective
filter units by a flexible connection. The supply fans are V-belt driven
centrifugal fans provided with automatic inlet vanes. The discharge side
of each fan is connected to the supply ductwork by a flexible connection
followed by an automatic isolation damper. Each supply duct is connected
59 to the corresponding CR air conditioning units.

The 100% redundant return fans are located in their respective
A/C unit cells. Two (2) sound absorbers are located upstream of the return
59 fans and the fans are connected to a common plenum by a gravity damper
followed by a flexible connection and automatic inlet vanes. The discharge
side of each fan is connected by a flexible connection to a discharge
plenum which is connected to three (3) branch ducts.

One duct connects with the Control Building missile protected
exhaust structure and is provided with an opposed blade damper and two (2)
redundant automatic isolation valves connected in series. The second duct
connects with the return air damper of the Control Room air conditioning
unit. The third duct connects with the Control Room filter unit. The
59 return fans are V-belt driven centrifugal fans provided with automatically
adjustable inlet vanes.

The toilet, janitors closet, and the kitchen are continuously
exhausted to the outside of the building by a toilet exhaust fan and a
kitchen exhaust fan during normal operation. The discharge of the kitchen
exhaust fan and the toilet exhaust fan, each with gravity dampers are
joined together into a common exhaust duct and provided with two(2) re-
dundant automatic dampers and is connected to a missile protected exhaust
structure. Insert Upon a containment isolation signal, a high radiation signal from
the redundant radiation monitors or high levels of toxic chemical or
smoke in the main or remote intake ducts, will close the automatic
59 dampers. The toilet and kitchen exhaust fans will be stopped manually.

All cells and corridors served by the Control Room (CR) System
are maintained at a 1/4 inch water gauge positive pressure relative to the
outdoor atmosphere during the normal and accident modes of operation.

Two separate outside air intakes, one main and one remote, are
provided for the Control Room. The main intake is located at the SW corner
of the Control Building roof at approximately elevation 880'. The remote
intake is located at the NE corner of the Steam Generator Building Auxiliary
49 Bay at approximately elevation 858'. Instrumentation is provided to measure

Insert
The two redundant automatic dampers are designed to QC Group C and controlled with IE Controls

pressure sensing device is provided with an alarm set-point to indicate that the differential pressure across the filter is higher than normal. This alarm set-point is selected on the basis that after initiation of the alarm, 72 hours are available to correct the failure without significantly deviating from the system design parameters.

The Containment Isolation Valves and their instrumentation and controls are redundant. The isolation valves are provided with remote position indicators and manual opening devices.

The Below Operating Floor Air Conditioning Unit automatic dampers are provided with remote position indicators and manual operators. The failure of any damper can be detected, identified and corrected within 4 hours. During this time the affected space temperature will be maintained below 120°F.

The Dome Recirculating Fans are not required for the safe shutdown of the reactor and maintenance of the safe shutdown condition.

2. <u>Loss of Normal Chilled Water Supply</u>

The Unit Coolers serving the safety related equipment in the EI&C cubicles are provided with Emergency Chilled Water. During loss of Normal Chilled Water, the Upset Design Temperature shall be maintained in these cells to satisfy the Operational Requirements.

3. <u>Loss of Normal Power</u>

During loss of Normal Power supply, the EI&C cubicle unit coolers are automatically switched to the on-site emergency Class IE AC power supply.

See Insert A

4. <u>Radioactive Contamination Protection of the RCB Areas</u>

The ventilation air quantities for the above and below operating floor areas of the RCB are selected to maintain the radioactive gas concentrations under the 10CFR20 limits.

The source of radioactive concentrations for the below operating floor areas are the probable outleakage of inert gases from the normally inerted cells. Since the inerted gases are continuously purified by the Cell Atmosphere Processing System (CAPS), the initial airborne radioactivity in these cells is low. The cells are designed with steel liners, leak tight penetrations and sealed doors to withstand the pressures resulting from accident conditions. The pressure differential during normal operation is very low, therefore the outleakage is minimal. The ventilation system air quantities for these areas are selected to maintain the acceptable airborne radiation concentration, resulting from the simultaneous design basis pressure differential and outleakage from all inerted cells. The selection provides sufficient conservatism in the
49 design.

Insert A

Atleast one EI &C cubicle shall be capable of operation to perform safe shutdown requirements, and the temperature in all three EI &C cubicles will remain below 120° F, even if any one 1E Division of Emergency Power or one Emergency Chilled Water train is inoperative. Analysis has been performed that shows that these conditions are met.

is annunciated in the Control Room. A logic circuit is available to automatically start the standby pump when the operating pump motor trips or is inadvertently stopped.

9.9.2 Emergency Plant Service Water System

9.9.2.1 Design Basis

The Emergency Plant Service Water System is designed to provide sufficient cooling water to permit the safe shutdown and the maintenance of the safe shutdown condition of the plant in the event of an accident resulting in the loss of the Normal Plant Service Water System or the loss of the plant AC power supply and all offsite AC power supplies. The Emergency Plant Service Water System is not used during normal plant operation. The system provides the Emergency Chilled Water System chiller condensers and the Standby Diesel Generators with cooling water. Additionally, this system provides fire fighting water for the seismically qualified fire pumps of the nonsodium fire protection system. The Emergency Plant Service Water System includes the Emergency Cooling Towers and Emergency Cooling Tower Basin, as described in Section 9.9.4.

The Emergency Plant Service Water System is designed to Seismic Category I requirements as defined in Section 3.2. Pumps, valving and piping required for the safe shutdown of the plant are designed to ASME Section III, Class 3 requirements, as defined in Section 3.9.2. All electric motors serving the system are connected to the Class 1E onsite power supply. In case of loss of plant and offsite power, these motors are switched automatically to the Standby Diesel Generator. The piping and equipment for each redundant loop of the system is physically separated or protected with a barrier to conform to common mode failure criterion. System piping is below ground between the Seismic Category I Emergency Cooling Tower and Diesel Generator Building and Electrical Equipment Building. The Emergency Cooling Tower structure is tornado missile hardened as described in Section 9.9.4.1.

9.9.2.2 System Description

The Emergency Plant Service Water System (EPSW) consists of two 100 percent capacity fully redundant cooling loops and a third loop. Each cooling loop includes one circulating pump, ~~one make-up pump~~, one emergency cooling tower and associated piping, valves, instrumentation and controls. Figure 9.9-2 shows the various equipments and represents the system component configuration and relationship.

The components served by the Emergency Plant Service Water System are listed in Table 9.9-3. Design data on the major system components is listed in Table 9.9-4.

Upon loss of Normal Chilled Water or upon start of the Standby Diesel Generators, the EPSW pumps, ~~EPSW makeup pumps,~~ and Cooling Tower Fans will automatically start and provide cooling water at 90°F maximum to the

Emergency Chiller Condensers in the SGB and the Standby Diesel Generators in the DGB. The EPSW pumps take suction from the Emergency Cooling Tower ~~operating~~ storage basin~~s~~ which ~~are~~ is located ~~adjacent to~~ under the Emergency Cooling Tower. ~~During system operation the EPSW makeup pumps will transfer water from the common storage basin to the redundant operating basins to compensate for evaporative and drift losses from the towers.~~

Cooled water from the Emergency Cooling Tower ~~operating~~ storage basin~~s~~ is pumped via underground supply mains to the emergency loads in the DGB and SGB. After cooling the emergency chillers and the standby diesel generators, warm water is returned, also through underground mains, to the Emergency Cooling Towers. To account for seasonal temperature variations, temperature control valves served by electro-hydraulic operators bypass a portion of the returning water back to the pump suction. A temperature indicator controller automatically adjusts the valves as required to maintain supply temperature above 55°F, the minimum required for chiller operation.

In addition to cooling the Emergency Chilled Water chillers and the standby Diesel Generators, each loop of the EPSW System provides a connection to supply water to the Non-Sodium Fire Protection System. The EPSW pumps and the Emergency Cooling Tower Basin are designed to allow fire protection operation while maintaining the capability for supplying 100 percent cooling to the emergency loads. The fire protection pumps are provided with instrumentation that will automatically terminate operation when a prescribed amount of water has been used (see Section 9.13). This ensures that the guaranteed 30 day supply of water for EPSW system operation will not be compromised. In addition, this system is connected to the EPSW loops in such a manner as to preclude a single failure from compromising the capability of the EPSW system to perform its required function.

9.9.2.3 Safety Evaluation

The EPSW system is a Seismic Category I, safety related system designed to have 100% redundancy in both active and passive components. The system is provided with AC power from the Class 1E power sources. EPSW Loop "A" is supplied from Class 1E Division 1 and Loop "B" is supplied from Class 1E Division 2. Loop C is supplied from Class 1E Division 3. This arrangement assures that 100 percent cooling capability will be available even if one of the Standby Diesel Generators or one of the EPSW loops should fail.

The EPSW system is a fully automatic system, normally controlled from the Main Control Panel in the Control Room. Redundant controls have been provided that will allow full operation of the system from a control panel in the Diesel Generator Building.

Pipe break analysis for this moderate energy fluid system will be provided in the FSAR.

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During the initial phase of recovery from an accident, one Emergency Plant Service Water loop satisfies the cooling of the Standby Diesel Generators and the Emergency Chilled Water Chiller Condensers.

The Emergency Plant Service Water System is capable of accommodating any single component failure without affecting the overall system capability of providing cooling water to achieve a safe shutdown condition. A single failure analysis of the Emergency Plant Service Water
59 System is given in Table 9.9-6.

15 9.9.2.4 <u>Tests and Inspections</u>

The system components will be tested at the manufacturer's facilities, and a complete system test will be accomplished prior to plant operation. The EPSW System does not operate during normal plant operations. However, the system, including all active components will be operated periodically during the year in conjunction with the Standby Diesel Generator testing program as outlined in USNRC Regulatory Guide 1.108. The system can be proven operable at any time by manual initiation. Inservice inspections will be conducted according to ASME Section XI, as described in Section 9.7.2.1.g. In addition, isolation valves and pressure test connections on the supply and return headers in the pumphouses and the DGB permit
50 inservice inspection of the buried piping by hydrostatic testing.

9.9.2.5 <u>Instrumentation Application</u>

Instrumentation will be provided for local and/or remote (Control Room) indication of the following parameters as indicated:

59
- pump discharge pressure (local/remote)
- diesel generator/emergency chilled water chillers supply temperature (local/remote)
50
- storage basin level (local/remote)
- diesel generator and emergency chiller flow rate (remote)
- diesel generator and emergency chiller supply temperature (local)
- diesel generator and emergency chiller return temperature (local/remote)
- diesel generator and emergency chiller supply and return pressure (local)
- operating basin level (local/remote)
- ~~makeup water flow (local/remote - alarm on low)~~

A flow switch, located in the return line from each diesel generator and emergency chiller will detect an abnormal low flow condition
43 33 and energize an annunciator in the Control Room.

15 9.9.3 <u>Secondary Service Closed Cooling Water System</u>

The objective of the Secondary Service Closed Cooling Water (SSCCW) System is to provide cooling to auxiliary equipment located in the turbine building.

Amend. ~~59~~ P
Dec. 1980

The Emergency Cooling Towers, pumphouses, ~~operating basins~~ and storage basin are designed to withstand the most severe natural phenomena (e.g., Safe Shutdown Earthquake, tornado, tornado missiles, wind, Probable Maximum Flood or drought). The design has the necessary redundancy and separation of components.

Electrical power for the Emergency Cooling Tower fans, pumps, and control equipment is provided from the Class 1E AC power supply. ~~One~~ Each loop is provided with electrical power from System Class 1E Division 1, ~~and the other from System Class 1E Division 2.~~ ,2, and 3 respectively.

9.9.4.2 Design Description

The Emergency Cooling Tower Structure consists ~~two~~ three of pumphouses (containing the pumps and piping of the EPSW System, Section 9.9.2) located directly above the ~~operating water~~ storage basin. The cooling towers, and pumphouses ~~and operating basins~~ are ~~100% redundant~~ Seismic Category I, Tornado protected structures. The common storage basin is a Seismic Category I, flood and tornado protected structure. The storage basin has sufficient storage capacity for 30 days of operation, including 30,000 gallons of water storage for the non-sodium Fire Protection System plus adequate allowance for drift and evaporation losses. Each cooling tower is designed to achieve the required heat dissipation rate at any time, approximately 2.36×10^7 BTU/HR at the maximum Emergency Plant Service Water Flow of approximately ~~3600~~ 3970 gpm.

The change in water chemistry due to the absence of blow-down from the cooling towers has minimal effect on operation of the Emergency Plant Service Water System. Proper selection of the Emergency Plant Service Water components, applied biocide additives, and maintainence of proper water chemistry will provide compensation for the increased tube fouling. The maximum makeup water required after 30 days of operation is approximately 100,000 gallons per day. In case the make-up water is not available after 30 days, make-up water can be supplied by either truck, rail or temporary piping from the Clinch River or purchased under agreements with the Department of Energy, Oak Ridge Operations.

The top elevation of the Emergency Cooling Tower Basin is 818 ft. which is 9 ft. above the probable maximum flood level. The entire basin and the cooling tower supports are founded on siltstone. The basin is a below grade reinforced concrete structure. For further details on the basin, refer to Section 3.8.4.1.5.

Each Emergency Cooling Tower consists of a single cell, provided with an induced draft fan system. Each cooling tower is enclosed in a Seismic Category I, tornado missile protected structure. The water intake and

for A&B towers and 7.5×10^6 BTU/hr. at the maximum Emergency Plant Service Water flow of approximately 900 GPM for tower C.

discharge piping are located within the tower or safely below the ground for tornado missile protection. The water intake and discharge piping and the internal distribution piping are Seismic Category I, ASME Section III, Class 3 design. ~~Each Emergency Cooling Tower has a design flow rate of 3600 GPM.~~

The Emergency Cooling Towers are of a counter-flow, wet-type, mechanically induced draft design. The internal distribution piping distributes the intake water evenly over the fill area so that sufficient water area is exposed to the counter air flow to provide evaporation for the required heat removal. The counter air flow is provided by the induced draft fans.

Drift eliminators are located above the internal water distribution piping and below the induced draft fans. The drift eliminators are a zigzag pattern of channels which prevent water carryover through the fan stack.

The Emergency Cooling Towers are supported by the reinforced concrete storage basin. The top of the cooling towers is approximately 44 ft. above the maximum water level of the storage basin.

The Emergency Cooling Tower Basin is filled with potable grade water which is treated for bacteria control. The quality of the stored water is analyzed at regular intervals and the required biocide additive is injected manually in quantities required to control seasonal variations of the bacteria growth.

The Emergency Cooling Towers and Emergency Cooling Tower Basin will be seismically analyzed as described in Section 3.7.

9.9.4.3 Safety Evaluation

a ~~third~~ cooling tower and pumphouse

The Emergency Cooling Tower structure consists of two 100 percent capacity cooling towers, pumphouses, and ~~operating basins~~ and one 100 percent capacity below grade cooling water storage basin. The entire structure is Seismic Category I, tornado, and flood protected. Piping, associated with the Emergency Cooling Tower is designed to ASME Section III, Class 3 requirements. The structure can withstand the most severe natural phenomena expected, and other site related events, such that the Emergency Cooling Tower cooling capability is assured under required conditions. The method of analysis is similar to that used for other Seismic Category I structures. The entire structure is designed to withstand the Safe Shutdown Earthquake. The fill, drift eliminators, motors, mechanical drives, piping, electrical conduit, cables and supports will be seismically analyzed in accordance with the procedures discussed in Section 3.7.

No drains are provided in the cooling tower Basin.

Insert B

Insert (B)

Freezing of the basin water will not affect the operation of the emergency cooling tower and the emergency plant service water system. This is because the suction elevation for the emergency plant service water pumps are located at the bottom of the basin which is 39 ft. from the surface of the colling tower basin water body.

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The Emergency Cooling Towers ~~and operating basins~~ are above the probable maximum flood level. The flood level considerations are discussed in Section 3.4.

The Emergency Cooling Tower pumphouses~~, except for the make-up pump pits which extend down to elevation 771'-0"~~, are also above the
probable maximum flood level. ~~However, the Emergency Cooling Water~~
50 ~~Make Up Pumps are submersible thereby providing system flood protection~~.

The Emergency Cooling Tower structure is designed to withstand
tornado windforces and tornado missiles and the cooling tower internals
50 are protected by the enclosing structure. The tornado and wind loadings
and the Missile Protection are discussed in Sections 3.3 and 3.5
respectively.

All materials used for the Emergency Cooling Tower Structure
43 are designed to be non-flammable in order to negate the possibility of
loss of the cooling function due to fire.

50

In order to evaluate the capability of the Emergency Cooling Towers and Emergency Cooling Tower Basin to act as an ultimate heat sink for the Emergency Plant Service Water System for a minimum period of 30 days, a detailed analysis will be done using the following conservative assumptions:

1. The Emergency Cooling Tower Structure is subjected to the maximum probable heat load. This load corresponds to the heat removal duty of the Emergency Plant Service Water System to control a postulated design basis accident and is listed on Table 9.9-3. During all other modes of operation the Normal Plant Service Water System removes the heat loads.

2. The postulated design basis accident is assumed to occur under conditions that minimize the heat removal rate, and maximize the water usage as follows:

   a. Meteorological Condition for Minimum Heat Removal Rate.

   The meteorological condition for minimizing heat removal rate is the highest wet bulb temperature that may occur at the inlet to the cooling tower. Wet bulb temperature is the only meteorological condition significantly affecting the water temperature produced by mechanical draft cooling towers.

   Each Emergency Cooling Tower is designed to dissipate the
   maximum expected heat load during the first 24 hours after
   a design basis accident assuming average wet bulb temperature
43 for the worst day of record.

15

TABLE 9.9-3

COMPONENTS SERVED BY EMERGENCY PLANT SERVICE WATER SYSTEM

Component	Component Location			Component Service Requirements		
	Bldg.	Cell	Elev.	Flow GPM	*EWT ^{0}F	BTU/HR $(X10^{6})$
Standby Diesel Generator A	DGB	~~511~~ TBD	~~816'-0"~~ STET	~~1500~~ TBD	90^{0}	~~13.2~~ TBD
Standby Diesel Generator B	DGB	~~512~~ TBD	816'-0"	~~1390~~ TBD	90^{0}	~~13.2~~ TBD
Standby Diesel Generator C	DGB	TBD	816'-0"	TBD	90°	TBD
Emergency Chilled Water System Chiller A	SGB	216	733'-0"	2100	90^{0} Max.	10.5
50 Emergency Chilled Water System Chiller B	SGB	217	733'-0"	2100	90^{0} Max.	10.5
Diesel Rm Cooling Unit A	DGB	TBD	847'-0"	TBD	90°	TBD
Diesel Rm Cooling Unit B	DGB	TBD	847'-0"	TBD	90°	TBD
50 Diesel Rm Cooling Unit C	DGB	TBD	847'-0"	TBD	90°	TBD

*Entering Water Temp.

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TABLE 9.9-4

EMERGENCY PLANT SERVICE WATER SYSTEM MAJOR COMPONENTS

	Description	Quantity	Design Data For Each Component
59	Emergency Plant Service Water System circulating pump Loops A&B	2	~~3800 GPM~~ TBD GPM ~~110 ft. total head~~ TBD ft. Total head
3 33	Emergency Cooling Tower A&B	2	~~3600 GPM~~ = TBD GPM
50	~~Emergency Plant Service Water Make-Up Pumps~~	~~2~~	~~150 GPM~~ ~~92 ft. total head~~
	Emergency Plant Service Water System Circulating Pump - Loop C	1	T[illegible] GPM ft. Total Head
43 33	Emergency Cooling Tower C	1	TBD GPM

Amend. 59
Dec. 1980

9.10 COMPRESSED GAS SYSTEM

The Compressed Gas Systems discussed in this section are those which supply instrument air, service air, hydrogen for generator cooling, and carbon dioxide for generator purging. Instrument and Service Air Systems are depicted in Figure 9.10-1. The Compressed Air System Safety Class III Instrument Air Supply is shown in Figure 9.10-2. A final drawing of this system will be provided upon completion of design. The Hydrogen System is shown in Figure 9.10-3, and carbon dioxide system in Figure 9.10-4.

9.10.1 Service Air and Instrument Air Systems

9.10.1.1 Design Basis

The Service Air System is designed to provide clean, oil - free, compressed air which will be used to:

1) Provide air necessary for various maintenance functions.

2) Provide air to required stations for personnel breathing where respiratory protection is required.

3) Provide air for the Instrument Air System.

To fulfill these requirements, the system components and piping are designed, fabricated and inspected in accordance with applicable codes as follows:

- Air receiver tanks, filter bodies, drying chambers, moisture separators, intercoolers/aftercoolers meet ASME Boiler and Pressure Vessel Code, Section VIII, Division 1.

- Piping (except containment penetration piping and isolation valves, instrument air piping and accumulators serving safety related components) meets ANSI B31.1.

- Containment penetration piping and isolation valves meet ASME Section III, Class 2 and Seismic Category I.

- Instrument air piping and accumulators serving safety related components meet ASME Section III, Class 3, Seismic Category I.

- In addition, service and instrument air equipment piping and components meet WARD-D-0037 (Appendix 3.7-A), Seismic Design Criteria for Clinch River Breeder Reactor Plant".

- Instrument Air System meets ANSI MC-11.1

Environmental design requirements will meet those for safety related equipment discussed in Section 3.11.

processing. Non-radioactive drainage is pumped to the equalization ponds of the Wastewater Treatment System. A power failure to the radiation monitor or diversion valves will cause recirculation back to the sump to prevent radioactive drainage from entering the non-radioactive wastewater treatment system. A manually operated, normally-closed valve is located downstream of the diversion valve. The sump will be sampled prior to pumping to the equalization ponds. Treated water and other process water treatment wastes which do not have the potential to be radioactively contaminated, are routed to seperate sumps for transport to the waste water treatment system.

Where there is a potential for oil spills, the drainage is routed to the oil separation system prior to discharge into the waste water disposal system. Oil spills are not allowed to drain in areas that contain radioactively contaminated equipment or fluids. In this case, the oil spill is contaminated with curbs and dikes and removed manually. Oil routed to the oil separation system is collected in a waste oil tank and removed from the site for subsequent disposal.

9.15.3 Safety Evaluation

The plant equipment and floor drainage system is designed so that it is not reasonably possible for any radioactive drainage in these systems to be discharged out of the plant without undergoing the required treatment or processing.

Evaluations of radiological considerations for normal operation and postulated spills and accidents are presented in Sections 11.2.5 and 15.0 respectively.

The plant Equipment and Floor Drainage Systems is not safety related except for the piping and valves required for containment isolation (Section 6.2.4).

EFDS piping within areas containing safety related equipment is supported with Seismic Category I supports.

There are no drains in cells where sodium piping or equipment containing sodium is located, accordingly sodium leaks cannot enter the equipment and floor drainage system.

A water pipe break or fire protection system drainage load cannot enter cells or compartments containing sodium from drain system backflow because these cells do not have any drains. The CRBRP design criteria requires that three passive barriers (or two passive and one active barrier) exist between all sodium and water boundaries. Accordingly, leak detectors located in the drainage system are not required.

Safety related systems containing water have instrumentation to detect leakage.

9.15.4 Tests and Inspections

EFDS pipes embedded in concrete are leak tested prior to the pouring of concrete. All EFDS piping is tested for leaks after installation. All leaking pipes or joints are repaired before the concrete is placed. The piping will be cleaned out to insure that construction debris will not cause a blockage or reduction in the flow. All pumps are tested to ensure that their

ATTACHMENT I

PSAR REVISION: SODIUM WATER REACTION IN LCCV

15.7.3.7 Sodium-Water Reaction in Large Component Cleaning Vessel

15.7.3.7.1 Identification of Causes and Accident Description

A sodium-water reaction accident in the Large Component Cleaning Vessel (LCCV) would be caused by unplanned introduction of liquid water which would react with bulk sodium prior to completion of the WVN phase of the sodium removal process. The consequences of the accident would depend on the amount of sodium on the component in the LCCV and the geometry of the component. This analysis assumes that the component having the largest initial sodium inventory is being cleaned. The frequency of sodium removal from components having enough sodium to make possible a serious sodium-water reaction is very low. This, together with the design features which prevent such a reaction, makes this an extremely unlikely event.

In the normal sodium removal process, all sodium except small amounts isolated in crevices is removed during the WVN cycle by the following reaction:

$$Na + H_2O = NaOH + 1/2\ H_2$$

Monitoring to determine completion of the sodium-water vapor reaction in the WVN cycle is accomplished by measuring hydrogen concentration in the gas leaving the LCCV. The reaction rate is controlled by establishing a water vapor concentration in the WVN entering the LCCV to limit the exhaust gas hydrogen concentration to less than 4%. As sodium is removed, the reaction rate and resulting hydrogen concentration decrease for a fixed inlet water vapor concentration. To maintain the reaction rate, the water vapor concentration is gradually raised to a maximum of 15%. The reaction is considered complete when the hydrogen concentration in the exit gas then falls below 100 ppm.

The rinse cycle in the normal sodium removal process removes the inert reaction products of the WVN cycle. This will not normally involve significant chemical reactions. Presence of the amounts of sodium necessary for a significant chemical reaction could occur only as a result of initiating the rinse cycle without performing the WVN cycle. The design includes an interlock to prevent this error by preventing opening of the water supply valve until 24 hr after opening of the steam supply valve for the WVN cycle. The interlock can be bypassed by use of a key switch whose key is kept under supervisory control. The accidental addition of water while all sodium remains on the component is the worst possible case and is analyzed for the sodium-water reaction accident.

Sodium in the LCCV prior to the WVN cycle would react with water during the accidental rinse cycle by the same reaction as in the WVN cycle. The hydrogen and heat generated would result in high pressure and temperature in the vessel. This would promote the additional reactions listed below; however, the reaction of the above equation would be predominant and was used in the analysis of this event.

$$Na + NaOH = 2NaO + 1/2\ H_2$$

$$2Na + NaOH = NaO + NaH$$

$$NaH + H_2O = NaOH + H_2$$

$$NaO + H_2O = NaOH$$

The sodium of interest for this analysis is in the form of frost deposited on parts which have been in the cover gas space above the reactor sodium pool. Since this form of sodium deposit presents a high-surface area for reaction, it was assumed that the reaction is instantaneous when water reaches sodium.

Many components will use the LCCV for sodium removal; however, all except two components, the intermediate rotating plug (IRP) and the small rotating plug (SRP), contain a quantity of sodium for which complete instantaneous reaction with water would result in an LCCV internal pressure less than the 15 psig design pressure. The design of the SRP is similar to that of the IRP described in the next paragraph. The event for the SRP would be the same as for the IRP, but the amount of sodium involved would be less by a factor of about six. Also, the SRP is expected to be cleaned only once per 30 yr. the same frequency as for the IRP. Therefore, the sodium-water reaction with the IRP is an enveloping event and was the case analyzed.

The IRP consists of a series of horizontal plates supported by four columns supported by the rotatable plug which is part of the reactor vessel closure head. The suppressor plate is the lowest plate. It and the lower 36 in. of its support columns are immersed in the reactor sodium pool during operation and will have a 0.003-in.-thick film of sodium when removed for cleaning. The area of the plate is about 47,000 in.^{2}, giving a sodium content of 4.5 lb. The lower 36 in. of the support columns will contain another 2 lb. of sodium film. The next plate, 48.7 in. above the suppressor plate, is the lowest of the reflector plates. There are 20 reflector plates, each separated by 1/2 in. and having a surface area of about 36,000 in.^{2}. Each has a coating of frost deposits consisting principally of sodium but also containing Na_2O and NaH. The thicknesses of these coatings range from 0.0445 in. for the bottom plate and the upper section of the support columns to 0.0005 in. for the top reflector plate. In this analysis, it is assumed that these are the thicknesses of solid sodium film. The lowest reflector plate contains about 65 lb. of sodium. The next two higher plates contain 51 and 40 lb. The total sodium on the IRP is about 350 lb.

The sodium-water reaction event would begin with the addition of water to the LCCV at a rate of 125 gpm. A flow of nitrogen at 50 cfm would be maintained through the water into the LCCV and out through the vent to maintain a purge of the system. The nitrogen would carry over water droplets which, together with the water vapor above the water surface, would react with the sodium at a rate comparable to that in the WVN cycle. It is assumed, however, that no sodium is removed by this reaction and that it all remains until the water reaches it. When the water reaches the suppressor plate, the 4.5 lb. of sodium on it will react instantaneously. The resulting pressure in the LCCV will be less than the LCCV design pressure of 15 psig. The hydrogen concentration in the LCCV nitrogen will be 2.5%, which is less than the 4% annunciator and interlock setpoint. It is assumed that water addition will continue at 125 gpm. The water level will rise at about 1-1/2 in. per min so that about 30 min will be required to reach the lower reflector plate. During this time, the hydrogen from the suppressor plate reaction and the slow reaction with support column sodium will be purged from the LCCV.

Water and the 65 lb. of sodium on the lower reflector plate will react when the water level has risen to the plate elevation. The pressure and hydrogen concentration in the LCCV gas space will increase. At a pressure of 8 psig, an interlock is activated to close the rinse-water inlet valve. At a pressure of 16.5 psig the LCCV pressure relief valve will open to vent the gas into the Large Component Cleaning Cell. The maximum pressure which would be reached without venting would be 89 psig. This is lower than the burst pressure of all components of the system, so that the hydrogen-nigrogen mixture will be contained except for venting through the pressure relief valve and the normal system vent to the H&V System. The hydrogen concentration in the LCCV gas will be 22%. The increase will be detected within a few seconds by the hydrogen analyzer in the LCCV vent line. When the detected level exceeds 4% an interlock will be activated to close the valve in the water inlet line. This interlock provides backup for the high pressure interlock which closes the same valve.

The hydrogen-nitrogen mixture which is vented through the LCCV pressure relief valve is mixed with the air at atmospheric pressure in the 67,000 ft^3 cell. The pressure resulting from adiabatic expansion of the mixture into the cell is about 2 psig. The hydrogen concentration in the cell after mixing with the air is about 2.5%

15.7.3.7.2 Analysis of Effects and Consequences

The sodium-water reaction described in the above section is an extremely unlikely event because the two components with which it could occur are each cleaned only once in 30 years, and because of the number of failures which must occur to permit the event. The principal failure would be in not completing the WVN cycle before adding water in the rinse cycle. An interlock requires that the WVN cycle must be started by opening the steam valve and must proceed for 24 hrs before the water inlet valve may be opened without using the key switch interlock bypass. Control of the key by supervisory personnel will avoid improper use of the bypass. Once the WVN cycle is begun, failure of a second interlock would be required to terminate it before the hydrogen concentration in the exhaust was less than 100 ppm. This low hydrogen concentration ensures that much of the sodium is reacted even if the inlet water vapor concentration is not raised to the normal 15%.

Analysis of the event hypothesized the instantaneous reaction of the 65 lb of sodium on the lowest reflector plate. The reaction releases hydrogen into the 2,100-ft^3 nitrogen gas space above the water level and releases heat. It is assumed that all heat from the reaction goes to heating the nitrogen and the reaction products (NaOH and H_2). Due to this heating, the gas space above the water would be pressurized to a maximum of 89 psig, which is less than the static rupture pressure of all components in the system. It is assumed for

Cell.

the analysis that all of the gas is released adiabatically into the LCCV. The resulting cell pressure of about 2 psig is less than the cell design pressure of 10 psig. It is also assumed that there is complete mixing of the vented LCCV gas and the LCCV cells air atmosphere. The resulting hydrogen concentration of 2.5% is less than the 4% explosive limit of hydrogen in air.

Since there is no designed vent between Cell 125 and the RCB atmosphere, the aerosol would be confined in the cell. Since the pressure in the cell is only 2 psig, there would be minimal leakage past the cell penetrations seals and by the time this leakage works its way up to the RCB atmosphere and through the system RCB filtration system, the impact on the site boundary would be negligible.

15.7.3.7.3 Conclusion

Based on the analysis described in the preceding sections, it is concluded that the vessel and system design is adequate to protect the plant and the public, and that there are no adverse consequences to the health and safety of the public which would result from this accident. Specifically:

- An uncontrolled sodium-water reaction in the LCCV is an extremely unlikely event.
- The LCCV pressure relief valve is set to vent the gas to the cell at a pressure 10% above the design pressure of the vessel.
- Failure of the relief valve to open will result in a maximum pressure of 89 psig. This is less than the calculated burst pressure for the LCCV and connected process equipment.
- Release to the LCCV cell of all reaction products will pressurize the cell to only 20% of the cell design pressure of 10 psig.
- There will be an imperceptible impact on the site boundary dose.

15.7.3.7.4 Enveloping Other Sodium-Water Reactions

To envelope the site boundary dose of all other sodium-water reactions in any of the cleaning vessels, calculations were made assuming 100% of the radioactivity deposited on the IRP to be released via a hypothetical vent from the LCCC to the RCB HVAC and thus to the environment. Activity content of the assumed release was derived from information in PSAR Table 11.1-7 decayed for 10 days. Such a release would isolate the RCB and the postulated effluent will pass through the filter system before release to the outside environment. A decontamination factor of 20 for iodine and 100 for particulates was assumed in the analysis. The activity is conservatively assumed to be in the form of a "puff."

Table 15.7.3.7-1 provides the resultant doses from this set of conservative assumptions and event. All doses are well within the appropriate requirements and guideline values of 10CFR100.

Question CS410.19 (9.7.3)

The normal and emergency chilled water systems provide cooling for plant HVAC systems. HVAC units serving areas containing sodium or NaK are provided with drains to carry away chilled water leakage to prevent moisture carry-over in the HVAC ducting. Leak detectors are provided in the drains to detect chilled water system coil failure. Activation of the detector results in automatic closure of the chilled water coil isolation valves. Justify the use of non-safety related normal chilled water system piping and valves in HVAC units serving areas containing sodium and NaK.

Response

With the exception of the SGB loop cells, the HVAC units provided with normal chilled water and serving areas containing sodium and NaK are located outside the sodium and NaK cells. These cells do not require safety-related cooling. Accordingly, their associated HVAC units are classified as non-safety-related. For the SGB loop cells. Three barriers between the sodium and water are provided as follows:

a) Chilled water piping walls

b) HVAC equipment walls which serve as spray shields

c) Sodium piping walls

The safety classification of these barriers is currently being evaluated to determine if they provide adequate protection against a sodium/water reaction. The results of this evaluation will be provided in a future amendment.