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Department of Energy
Clinch River Breeder Reactor
Plant Project Office
P.O. Box U
Oak Ridge, Tennessee 37830

Docket No. 50-537 File: 05.10

June 29, 1979

Mr. Domenic B. Dassallo
Acting Director
Division of Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Dassallo:

AMENDMENT NO. 50 TO THE PRELIMINARY SAFETY ANALYSIS REPORT FOR CLINCH RIVER BREEDER REACTOR PLANT

The application for a Construction Permit and Class 104(b) Operating License for the Clinch River Breeder Reactor Plant, docketed April 10, 1975, in NRC Docket No. 50-537, is hereby amended by the submission of Amendment No. 50 to the Preliminary Safety Analysis Report pursuant to 50.34(a) of 10 CFR Part 50. This Amendment No. 50 includes: an update to Section 9.9.2, "Emergency Plant Service Water System"; an update to Chapter 13, "Conduct of Operations"; and other updates and revisions, as well as a response to NRC's request for additional information contained in a letter dated August 17, 1976.

A Certificate of Service, confirming service of Amendment No. 50 to the PSAR upon designated local public officials and representatives of the EPA, will be filed with your office after service has been made. Three signed originals of this letter and 97 copies of this amendment, each with a copy of the submittal letter, are hereby submitted.

Sincerely,

Raymond L. Copeland

Raymond L. Copeland
Acting Assistant Director
for Public Safety

PS:79:161

Enclosure

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Amendment 50

Clinch River Breeder Reactor Plant

Preliminary Safety Analysis Report

(Docket No. 50-537)

This fiftieth amendment to the Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report includes updates to sections describing the Emergency Plant Service Water System and Conduct of Operations, other updates and revisions as well as responses to NRC's request for additional information. Vertical margin lines on the left hand side of the page are used to identify new design information while lines on the right hand side of the page identify question/response information.

A page replacement guide appears following the list of responses to NRC questions.

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Ques. No.

011.23

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PAGE REPLACEMENT GUIDE FOR

AMENDMENT 50

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

Transmitted herein is Amendment 50 to the Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report, Docket No. 50-537. Amendment 50 consists of new and replacement pages for the PSAR text and question/response supplement pages.

The following attached sheets list Amendment 50 pages and instructions for their incorporation into the Preliminary Safety Analysis Report.

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Amendment 50

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Amendment 50

Question/Response Supplement

The Question/Response Supplement contains an Amendment 50 tab to be inserted following page Q-i (Amendment 49, April 1979). Page Q-i (Amendment 50, June 1979) is to follow the Amendment 50 tab.

The Questions/Response Supplement page listed below should be inserted in the proper numerical order following the correct section tabs. The parentheses beside the question indicates the number of pages contained in the Question/Response.

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3.8 DESIGN OF CATEGORY 1 STRUCTURES

3.8.1 Concrete Containment (Not Applicable)

3.8.2 Steel Containment System

3.8.2.1 Description of the Containment

The Containment Vessel is a low leakage, free-standing, all
welded steel vessel anchored to the base mat with a steel lined concrete
bottom in the form of a vertical right cylinder having an inside diameter
of 186 feet and with side walls extending approximately 169 feet from
the flat bottom liner at the base to the spring line of the ellipsoidal-
45 spherical dome. The cylindrical shell is embedded in concrete up to the
47 elevation of the operating floor. On the inside of the Containment
Vessel, there is the continuous reinforced concrete wall comprising the
peripheral boundary of the internal concrete structure. Butting against
the outside face of the steel shell from elevation 733 feet up to the
45 elevation of the underside of the operating floor, there is another rein-
forced concrete wall of sufficient thickness designed to prevent buckling
45 of the steel shell. Neither of the two concrete walls are considered
part of the containment vessel. Alumina-silica insulation is attached 33
to the inside surface of the Containment Vessel from elevation 816 feet
to elevation 823 feet. The insulation is 3 inches thick and has a value
of 0.0267 Btu/hr - ft-^{o}F. Its purpose is to limit the shell temperature
48 at elevation 816 feet during Design Basis Accidents to less than 130^{o}F.

The vessel includes: its shell, a ¼" bottom liner plate, one
45 access airlock, one emergency egress airlock, vacuum relief system, one
equipment hatch, penetrations, inspection ladders, miscellaneous appur-
tenances and attachments. The configuration of the Containment Building
is shown in figures in Section 1.2. The design lifetime of the contain-
39 ment vessel shall be 30 years.

3.8.2.2 Applicable Codes, Standards and Specifications

3.8.2.2.1 Codes

The Containment Vessel will be designed, material procured,
fabricated, installed and tested in accordance with the requirements of
the ASME B&PV Code, Section III, Division 1, 1974 Edition with Addenda
through Winter 1974 and Code cases 1713, 1714, 1809, 1682 and 1785 and
50 43 ASME-III, Division 2, 1975 Edition, Subsection cc, for the steel lined
concrete containment bottom. The design shall also meet the requirements
of the Class MC Section of RDT Standard E15-2T, "Requirements for Nuclear
Components".

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The quality assurance procedures will be in accordance with RDT Standard F2-2 as well as meeting the requirements of the ASME Code, 1
45 Section III, Divisions 1 and 2.

All structural steel non-pressure parts such as ladders, walkways, handrail, etc. will be designed in accordance with the American Institute of Steel Construction (AISC), "Specification for the Design, Fabrication and Erection of Structural Steel Buildings (AISC, February 12, 1969).

3.8.2.2.2 Design Specification Summary and Design Criteria

The Containment Vessel, including all access openings and penetrations will be designed such that the leakage of radioactive materials from the Containment under conditions of temperature and pressure resulting from the extremely unlikely faults could not cause undue risk to the health and safety of the public and will not result in potential offsite exposures in excess of guideline values of 10CFR100.

Tolerances

The Containment Vessel as constructed shall not exceed the tolerance requirements of NE-4000 of ASME-III for fabrication or erection. The dimensional control procedures shall meet the requirements of RDT STD F3-15T.

The out-of-plumb tolerances shall not exceed 1/500. The out-of-roundness tolerance shall not exceed ½ of one percent of the nominal inside diameter.

3.8.2.2.3 Applicable NRC Regulations and Regulatory Guides

15 NRC Regulatory Guides

The applicable regulatory guides are listed below.

1.10: Mechanical (Caldwell) Splices in Reinforcing Bars of Category I Concrete Structures (Revision 1, January 2, 1973).

45 1.11: Instrument Lines Penetrating Primary Reactor Containment (March 10, 1971)

1.12: Instrumentation for Earthquakes(Revision 1, April, 1974)

1.15: Testing of Reinforcing Bars for Category 1, Concrete Structures (Revision1, December 28, 1972)

1.19: Nondestructive Examination of Primary Containment Liner Welds (Revision 1, August 11, 1972)

50 1.29: Seismic Design Classification (Revision 2, August 1976)

1.55: Concrete Placement in Category 1, Structures (June 1973)

45 1.57: Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (June, 1973)

1.60: Design Response Spectra for Seismic Design of Nuclear Power Plants (Revision 1,December, 1973)

45 1.61: Damping Values for Seismic Design of Nuclear Power Plants (Oct. 1973)

1.63: Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants (Oct. 1973)

50 1.85: Materials Code Case Acceptability - ASME Section III, Division 1, 1976

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45 1.92: Combining Modal Responses and Spatial Components in Seismic Response Analysis (Revision 1, Feb., 1976)

Of the above, Regulatory Guide 1.63 is applicable after the following changes:

1. Deleting "water-cooled" wherever it appears.

2. Replacing "Appendix B to 10 CFR Part 50" wherever it appears with "RDT Standard F2-2".

3. Replacing "General Design Criterion 50 of Appendix A to 10 CFR Part 50" wherever it appears with "CRBRP GDC 50".

4. Replacing "loss of coolant accident" with "containment design basis accident".

5. Substituting "(Summer 1972 Addenda)" following".....ASME Boiler and Pressure Vessel Code" with ", 1974 Edition".

Construction

No special construction techniques are anticipated for this containment vessel.

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☑ ILLEGIBLE: HARD COPY AT:

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Bases - The prime objective of this requirement is to assure the functional capability of all core components is preserved. It is possible to postulate a number of potential failure mechanisms associated with a general condition of duct to duct contact in the active core region. One such failure mechanism is duct brittle fracture under seismic loading. Another would be excessive deformation which could occur due to buildup of large interduct contact loads. By not allowing duct contact to become a general condition, the functional capability of all core components is preserved. The consequences of local duct to duct contact will be evaluated.

i) Requirement - The design life for the core formers is 30 years for the 75% plant capacity factor.

Bases - This requirement assures that the core formers will not need to be replaced during the design life of the plant. The primary incentive for this requirement is economic and operational benefits.

j) Requirement - Design of the core restraint system shall be consistent with annual refueling.

Bases - This requirement provides consistency with established plant operating guidelines.

k) Requirement - The dynamic radial deflection of the inner profile of the Upper Core Former (UCF) shall be less than $\pm$ 0.100 inch, relative to the UCF centerline during an SSE or OBE event.

Bases - Proper alignment of the control assembly handling socket is necessary to meet protection system (SCRAM) insertion rate requirements.

The insertion rates required during a seismic event are slightly lower than those required during non-seismic events (see Figures 4.2-93 and 4.2-94).

44| For non-seismic events, control rod system alignment requirements are defined by Figures 4.2-95A and B. The contribution from the core restraint system and core former clearances and tolerances is limited to less than 0.530 for the control assembly handling socket to reactor vessel centerline.

However, for seismic events, a similar alignment limit was considered not to be applicable, since dynamic analysis is being performed to adequately account for deflections of each component interfacing with the control rod system. Qualitatively, the objective for the core former is to limit seismic deflections to as small as practical, within the various design restrictions. The 0.100 inch dynamic deflection limit is thus established as a reasonable design objective for such a large (150 inch O.D.) ring. It should be pointed out

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that the UCF time vs. deflection history in relation to that for the control rod system and other interfacing components is also an important parameter in the scram analysis, and is not easily factored into a simple deflection requirement.

1) Requirement - The Lower Core Former (LCF) radial deflection relative to the LCF centerline during an SSE or OBE shall be limited to less than ±.100 inches.

Bases - The Lower Core Former, interfacing with the above core load pads on the reactor assemblies, performs a geometry control function. To limit the seismic loads on the assemblies and to limit the magnitude of and reactivity insertion during a seismic event, the LCF deflection must be limited. If the distance across the core former were significantly reduced due to an elliptical deflection of the ring, the reactor assemblies would be compacted, causing a reactivity insertion and higher loads on the load pads. By limiting the LCF deflection to less that 0.100, an increase in load pads is avoided since adequate clearances exist between load pads and between the outermost load pad and the Lower Core Former. This is shown in Figures 4.2-84 and 4.2-92. Preliminary core restraint models indicate that cumulative clearance is slightly more than 0.150 inches between the first row blanket assembly load pad and the LCF segment for on-power operation. Therefore, if the LCF deflects up to 0.100 inches, the compaction would primarily affect the shield and outer blanket assemblies and would not be expected to result in a significantly greater seismic reactivity insertion.

During low power operation, the gap existing in the outer core region is less than for on-power operation due to the outward assembly bow. This could result in a slightly larger seismic reactivity insertion; however, the allowable reactivity insertion is also larger for low power operation. Further evaluations are planned to better determine the effect of core former deflection on seismic reactivity insertions. These evaluations will lead to an improved definition of LCF deflection requirements.

There will be a seismic induced core compaction in an orthogonal direction which will occur even with zero LCF deflection. This is discussed in Section 15.2.1.3. The intent of the LCF deflection limit is to prevent an additional source of core compaction during a seismic compaction event. 14

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$$D = \sum \frac{n}{N_d} + \sum \frac{t}{T_D}$$

To meet criterion: $D_e \leq D$

(7) A cycle limiting criterion is required to verify the applicability of the modified rule. The effective number of allowable design cycles is:

$$n_e = n\left(\frac{D}{D_e}\right)$$

Where n is the total number of significant strain cycles between hold periods. Low amplitude high cycle strain fluctuations (such as normal power fluctuations) need not be considered in n if they are elastic excursions that result in negligible fatigue damage.

For the modified rule to be applicable, n_e shall not exceed 3000 for type 316 stainless steel nor 6000 for type 304 stainless steel.

Modification of Creep Damage Rules

In cases where a local stress concentration exists, the creep-fatigue damage evaluation may be modified as described herein.

(1) The material is austenitic stainless steel Type 304 or 316 solution treated.

(2) The structure does not require a Code Stamp under existing Code rules.

(3) Simplified or rigorous inelastic analysis is used.

(4) Stress rupture test data of the same type of stress concentration with similar geometric proportions tested at prototypic temperatures are used as a basis for modification of the Code Strength. The test temperature may be higher than the service temperature in order to more closely simulate the actual component lifetime and the stress level.

(5) The notched stress rupture data shall be from specimens which are comparably or more severely loaded than the component, i.e., membrane loading of a notched specimen should be more severe than a gradient loading.

49 (6) The stress rupture test data include data up to 1/60 of the component lifetime at prototypic temperatures or the equivalent when a short-time high temperature combination is used to simulate the desired long-time service environment.

(7) Subject to the above limitations, the creep damage may be calculated in accordance with F9-4T and Code Case 1592 as modified. The modification is to use a peak stress to rupture design curve based upon the stress to rupture design curve in Code Case 1592 adjusted for the influence of a non-linear stress state caused by the presence of a geometric stress concentration as with the following:

Step 1 - Determine the smooth specimen stress rupture strength curve by tests of the same material at the temperature of interest.

Step 2 - Determine the stress rupture strength curve with the presence of the geometric stress concentrations under the same conditions in (1) with specimens of the same heat of material with the same histories. Analytically determine the peak stress relative to the net stress thus defining the stress rupture strength in terms of "peak stress" vs. time to rupture.

Step 3 - Ratio Code Case 1592 stress to rupture design curve by the ratio of Step 1 divided by Step 2. This must be done for at least 3 points in time with a separation in time of at least two orders of magnitude. In cases where the strength ratio varies with lifetime, the lesser of the value extrapolated to the component lifetime or the experimental value for the longest duration tests shall be used.

(8) The total creep-fatigue damage is determined by adding to the creep damage and fatigue damage calculated in accordance with T-1411, -1412, -1413, and -1414 of Code Case 1592.

(9) The allowable creep-fatigue damage (D) is determined from the lesser of the values from Figure T-1420-2 of Code Case 1592 (See Figure 4.2- and an average of test values from creep-fatigue interaction tests of notched specimens.

(10) The greater of the damage using the modified rule and the damage using the stress unaltered by the stress concentrations and the Code Case 1592 stress to rupture design curve shall be used. 49

4.2.2.3.3 Additional Material Properties

4.2.2.3.3.1 Inconel 718 Fatigue Properties

The Alloy 718 design fatigue curve, Figure 4.2-48, proposed for inclusion in the NSM Handbook as interim data, shall be used until superceded. The effects of the fabrication processes and service environment on the structural integrity of the UIS shall be considered. The effect 41

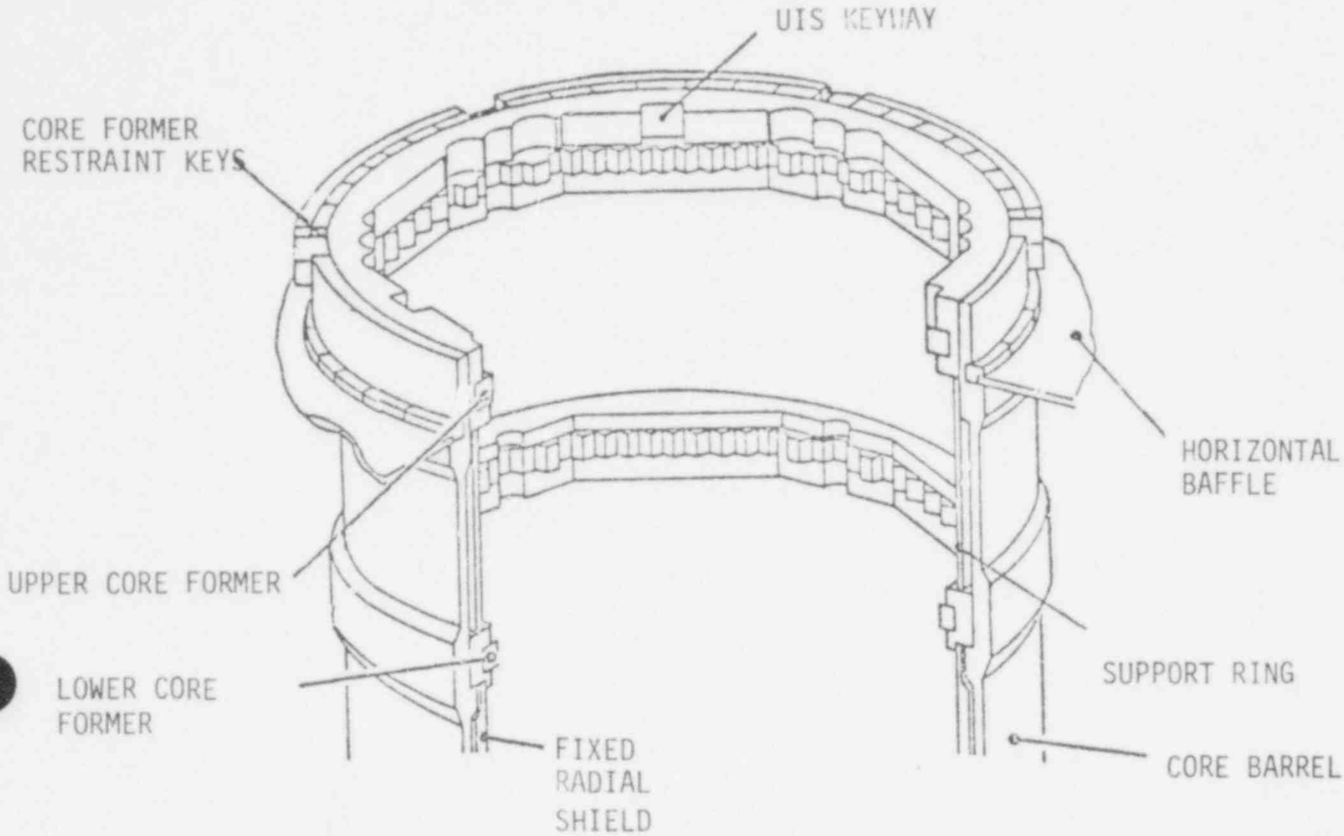

Figure 4.2-46. Core Former Structure

traps are air cooled, and process the sodium at a flow of 60 gpm. The single trap is sufficient to limit oxygen and hydrogen to a maximum of 2 and 0.2 ppm respectively, and to limit the tritium concentration in the sodium to 0.062 μCi/gm (corresponds to a transport of tritium to the steam generator water of 0.016 Ci/day). A more detailed description of the Intermediate
50 Sodium Processing System is given in Section 9.3.2.4.

5.4.3 Design Evaluation

5.4.3.1 Analytical Methods and Data

The design of the IHTS is based upon technology attained during the development, design, construction and operation of sodium systems of similar type IHTS hydraulic and fluid volume changes are calculated and based upon sodium thermodynamic data from "Standard FFTF Values for Physical and Thermophysical Properties of Sodium". (Ref. 2)

The IHTS sodium pressure losses are calculated using Darcy's formula, the general equation for pressure drop of fluids. The factors for equivalent lengths of fittings and pipe friction are based upon "Tube Turns," Bulletin TT725 (Ref. 3) and "Friction Factors for Pipe Flow," by L. F. Moody (Ref. 4), respectively.

5.4.3.1.1 Structural Evaluation Plan (SEP)

The procedure for developing SEP's for the IHTS is the same as described for the PHTS in Section 5.3.3.1.1.

5.4.3.1.2 Stress Analysis Verification

The IHTS stress analysis verification procedures and methods are the same as those described for the PHTS in Section 5.3.3.1.2.

5.4.3.1.3 Compliance with Code Requirements

The design of the IHTS will be in compliance with code requirements as identified for the PHTS in Section 5.3.3.1.3.

AFP Motor Drives

These motor drives will be synchronous speed squirrel cage induction motors of 980 horsepower. These motors will be selected from a
17 vendor's standard line and no special requirements are anticipated.

AFP Turbine Drive

This component will be obtained from an experienced vendor and will be sized to produce 1960 horsepower. The turbine will be constructed with
17 sufficient quality assurance coverage to assure its reliability during service.

The auxiliary feedpump turbine is not kept hot for quick start operation. The drive turbine concept selected for the Auxiliary Feed Pump is based on the capability of this turbine to withstand severe service conditions. This is accomplished by constructing the turbine wheel from a single forging with buckets milled into the forging. The start-up procedure is similar to that for the RCIC turbine in a BWR in that it will occur without
pre-warming. 25

Pump Integrity

The auxiliary feed pumps will be designed to the requirements of ASME B&PV Code, Section III, Class 3. In addition, the pumps and their supports will be designed to Seismic Category 1 requirements. Allowable stress limits are specified in Table 3.9-3 and pressure limits are speci-
17 fied in Table 3.9-4.

5.6.1.2.3.3 Protected Water Storage Tank (PWST)

The PWST holds the protected water to be supplied to the steam drums in the event of loss of normal feedwater or normal heat sink. The size is determined by detailed analysis of the heat removal conditions during the first several hours after shutdown and by anticipated component leakage rates. The tank will be constructed to the requirements for an ASME Section III/Class 2
17 vessel and it will operate at low temperature (<200°F) and low pressure (<15 psig).

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5.6.1.2.3.4 SGAHRS Piping and Support

The SGAHRS piping is described below and is shown in Figure 5.1-5. The SGAHRS piping will be designed in accordance with the ASME Code Section III as specified in Section 5.6.1.1.2. The material
49 specifications are discussed in Section 5.6.1.1.4.

The SGAHRS piping runs can be categorized as follows:

a. PWST Fill Line

This 3 inch low pressure, low temperature, Class 3 carbon steel line runs from the 10 inch alternate water supply line through the motor-driven, normally closed PWST fill valve to the PWST inlet.

b. Protected Water Storage Tank (PWST) to Auxiliary Feedwater Pump (AFP) Inlet

There are three low pressure, low temperature, uninsulated carbon steel lines from the PWST to the three auxiliary feedwater pump inlets. Two of the lines, each of which leads to a half size, motor-driven pump are 6 inches in diameter and the third line to the full size turbine-driven pump is 8 inches. All three lines contain a manually operated, locked open valve and an electrical operated, normally open isolation valve. These lines are Class 2 from the PWST to the electrically-operated isolation valve
49 and then Class 3 to the pump inlet.

c. Alternate Supply Line to AFWP Inlet

The alternate supply line provides the capability for the AFW pumps to take suction from the condensate storage tank. A 10 inch carbon steel line runs from the condensate tank junction to the first branch line. An 8 inch branch line
49 passes through an electrically-operated, normally closed isolation valve and tees into the 8 inch turbine pump inlet piping. Two 6-inch branch lines each pass through electrically-
49 operated, normally closed isolation valves and then tee into the 6 inch motor-driven pump inlet piping. The total run of piping is Class 3.

d. Auxiliary Feedwater Pump Discharge to Discharge Header (Inclusive)

The 6 inch carbon steel turbine pump discharge line leads to a 6 inch discharge header. This header in turn has three discharge points, one to each steam drum feedwater supply loop a 6 inch carbon steel line from each motor driven pump feeds into a 6 inch header which also has three discharge points, one
43 17 to each drum.

5.6.2 Direct Heat Removal Service (DHRS)

5.6.2.1 Design Bases

5.6.2.1.1 Performance Objectives

41 The Reliability Program, discussed in Appendix C, will provide verification that SGAHRS removes residual heat following a reactor shutdown with a high level of reliability. Hence it is judged that only steam and feedwater trains backed by SGAHRS are necessary to safely and reliably remove residual heat following shutdown from three loop, full power operation. To enhance the reliability of decay heat removal, the DHRS provides a fourth redundant heat removal path and heat sink. The impact on overall shutdown heat removal reliability by inclusion of the
41 DHRS is being determined by the Reliability Program described in Appendix C. The DHRS provides this supplementary capability by satisfying the following objectives:

a. Function to remove reactor decay heat following reactor shutdown from three loop rated power operation, assuming loss of all heat transfer through the IHX's at the time of reactor trip. Operation of three primary pump pony motors and maximum reactor decay power is assumed. Under these conditions, the DHRS is designed to provide sufficient cooling to ensure primary coolant boundary integrity and prevent loss of in-place coolable geometry of the core.

   To meet this objective, DHRS components will be sized such that, under these conditions, the average bulk primary sodium temperature will be limited to approximately 1140°F. Capability will be provided to permit remote manual initiation of DHRS from the Control Room. The overflow and makeup circuit and the spent fuel cooling system will be able to be cross connected in a manner which permits both EVST NaK cooling trains to be used
26 when DHRS is removing full capacity heat load.

b. Accommodate the thermal transients resulting from normal, upset, emergency and faulted plant events in which continued performance of its function is not impaired.

c. Accommodate floods (Section 3.4), tornadoes (Section 3.3), missiles (Section 3.5), and earthquakes (Section 3.7), in which continued performance of its function is not impaired.

d. Function in a manner which will not significantly reduce the reliability and availability of the EVST heat removal chain. This objective requires the EVST NaK cooling circuits to be designed to remove concurrently the heat generated by the spent fuel and the reactor decay heat.

5.6.2.1.2 Applicable Code Criteria and Cases

The components of the DHRS shall be designed, fabricated, erected, constructed, tested and inspected to the standards of Section III of the ASME Code, 1974 edition through the summer 1974 Addenda, Class 1 or 2 as
26 listed in Table 5.6-12.

Applicable code cases will be used to supplement the design analysis required by the ASME Code.

5.6.2.1.3 Surveillance Requirements

The need for surveillance of the DHRS piping and components will
26 be determined as the system design progresses and as the need to monitor austenitic stainless steel is determined by ongoing programs. If a requirement is identified, a surveillance program will be designed in accordance with the philosophy of 10 CFR 50, Appendix H.

5.6.2.1.4 Material Considerations

High Temperature Design Criteria

High temperature components in the DHRS will be analyzed in
26 accordance with the requirements specified in ASME Boiler and Pressure Vessel Code, Section III, as supplemented by the applicable code cases and RDT standards.

Material Specifications

Stainless steel materials which satisfy the requirements of the ASME Code will be specified for use in the DHRS system, as noted in Table
26 5.6-13.

49 5.6.2.1.5 Leak Detection Requirements

The DHRS will be monitored for sodium and NaK leaks and leak
26 indication will be provided in the Control Room by the leak detection system described in Section 7.5.5.

5.6.2.1.6 Instrumentation Requirements

DHRS is remote manually activated and controlled from the Control Room. Instrumentation required to monitored the condition of the DHRS consists of thermocouples on the EVST sodium outlet lines (3 loops) and level indicators in the EVST and the Reactor Vessel (RV). These instruments confirm that the sodium levels in the RV remain above the loop outlet nozzles
50 46 and that temperatures remain below design limits. Other DHRS diagnostic instrumentation is not essential for DHRS operation as the pumps and air blast heat exchanger are being operated at maximum design rates. When the reactor decay heat load has dropped sufficiently, the cooling capacity of the system may manually be reduced by lowering flow-rates or fan speed, or by shutting down one one of the EVST cooling trains.

26

TABLE 5.6-5

SGAHRS VALVE CLASSIFICATION

	VALVE	ACTIVE/INACTIVE	NORMAL POSITION	OPERATING MODE
	Alternate AFW Supply	Active	Open	Isolation
44 26	PWST Fill	Inactive	Closed	Isolation
	PWST Drain	Inactive	Closed	Isolation
	PWST Level Indicator	Inactive	Open	Isolation
	AFW Pump Inlet (Manual)	Inactive	Open	Isolation
50	AFW Pump Inlet (Electrical)	Active	Open	Isolation
	Alternate AFW Pump Inlet	Active	Closed	Isolation
26	Pump Recirculation	Active	Closed	Isolation
	Pump Discharge C/V	Active		Check
	Pump Discharge Isolation (Manual)	Inactive	Open	Isolation
	AFW Supply Isolation (Manual)	Inactive	Open	Isolation
	AFW Supply Control	Active	Open	Flow Control
50	AFW Supply Isolation (Electrical)	Active	Closed	Isolation
	AFW Supply Isolation (Manual)	Inactive	Open	Isolation
	AFW Supply C/V	Active		Check
50	AFW Supply C/V	Active		Check
	Steam Drum Vent Control	Active	Closed	Vent, Pressure Control
26	Superheater Vent Control	Active	Closed	Vent, Pressure Control
	PACC Steam Supply	Inactive	Open	Isolation
	PACC Steam Supply Bypass	Inactive	Open	Isolation
	PACC Condensate Return	Inactive	Open	Isolation
	PACC Noncondensible Vent	Active	Closed	Vent
	PACC Noncondensible Vent Isolation	Inactive	Open	Isolation
50	Drive Turbine Steam Supply Isolation (Electrical)	Active	Closed	Isolation
26	Drive Turbine Steam Supply C/V	Active		Check
	Drive Turbine Steam Supply Isolation (Manual)	Inactive	Open	Isolation

TABLE 5.6-5 (Cont'd)

VALVE	ACTIVE/INACTIVE	NORMAL POSITION	OPERATING MODE
Drive Turbine Steam Supply Pressure Control	Active	Closed	Pressure Control
Pressure Instrument (Pump Inlet)	Inactive	Open	Isolation
Pressure Instrument (Pump Discharge)	Inactive	Open	Isolation
Pressure Instrument (Turbine Inlet)	Inactive	Open	Isolation

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Table 6.2-5 Lines Penetrating Containment (continued)

Penetration	PSAR Section	Number of Valves Required	Type of Valve Required	Line Size	Actuation Signal	Loss of Actuation Power Position	Position Accident Conditions	Type of Valve Actuation Required	Valve Position Status at Normal Plant Operations	Primary Actuation	Secondary Actuation	Closure Time Seconds	Valve Configuration (See Fig. 6.2-10)
Failed Fuel Monitoring Supply	9.5	2	Globe	2"	CIS	Closed	Closed	Automatic	Open	Automatic	Remote Manual	<10	F
Failed Fuel Monitoring Return	9.5	2	Globe	2"	CIS	Closed	Closed	Automatic	Open	Automatic	Remote Manual	<10	F
N_2 Supply Line	9.5	2	Globe	2"	Back Pressure	***	***	Automatic	Open	Automatic Back Pressure	Remote Manual	<10	D
Emergency Chilled Water Supply	9.7	1	Ball	3"	**	Closed	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Emergency Chilled Water Return	9.7	1	Ball	3"	**	Closed	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Floor Drain Sump Discharge	9.15	2	Gate*	6"	CIS	Closed	Closed	Automatic	Open	Automatic	Remote Manual	<4	TBD

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Table 6.2-5 Lines Penetrating Containment (continued)

Penetration	PSAR Section	Number of Valves Required	Type of Valve Required	Line Size	Actuation Signal	Loss of Actuation Power Position	Position Accident Conditions	Type of Valve Actuation Required	Valve Position Status at Normal Plant Operations	Primary Actuation	Secondary Actuation	Closure Time Seconds	Valve Configuration (See Fig. 6.2-10)
Normal Chilled Water Supply	9.7.	1	Ball	8"	**	Closed	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Normal Chilled Water Return	9.7	1	Ball	8"	**	Closed	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Normal Chilled Water Return	9.7	1	Ball	6"	**	Closed	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Normal Chilled Water Supply	9.7	1	Ball	6"	**	Closed	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Normal Chilled Water Return	9.7	1	Ball	6"	**	Closed	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Normal Chilled Water Supply	9.7	1	Ball	6"	**	Closed	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Normal Chilled Water Supply	9.7	1	Ball	6"	**	Closed	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Normal Chilled Water Return	9.7	1	Ball	6"	**	Closed	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Containment Ventilation Air Return	9.6	2	Butterfly	24"	CIS	Closed	Closed	Automatic	Open	Automatic	Remote	< 4	A

47

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2. A Heating, Ventilation and Air Conitioning (HVAC) System to keep the main control room slightly pressurized at all times and at temperatures, humidities, and air purity levels adequate for conducting safe, efficient plant control operations.

3. A low leakage enclosure for the main Control Room and its adjoining rooms to provide the capability for keeping a positive air pressure level with n the enclosure.

4. Two alternate and widely separated air intakes and redundant filter units, to limit the amount of contamination entering the Control Room.

5. Airborne hazard monitors that detect unsafe concent ations of smoke, toxic chemicals and unsafe radiation levels, annunciate the presence of the hazard and automatically 22
49 transfer the HVAC system to its accident moc of operation.

6. Appropriate fire suppression equipment.

7. Office and living accommodations appropriate for long term occupancy.

8. An amply stocked inventory of emergency equipment and supplies.

9. Installation of two door vestibules to prevent unfiltered air entering the main Control Room.

49 The habitability system is also required to permit continuous control room occupancy under accident conditions. These provisions enable the operators to remain on duty without relief for as long as required. It is, therefore, not necessary to plan on routine shift changes by the control room personnel. However, if conditions necessitate a change in personnel, this can be accomplished without undue radiation exposure.

The calculated radiation exposure for either ingress or egress
49 at 24 hours after the accident is less than 1 mrem. This estimate is based on direct exposure from fission product gases, halogens, volatile solids, fission products, and activated sodium evenly distributed throughout the reactor containment building free volume. These sources consti-
49 tute direct shine dose only. 25

The ingress or egress exposure resulting from the radioactivity (annulus leakage previously stated) is approximately 3.50 mrem external exposure (whole body dose). The internal exposure to the lungs, thyroid,
49 and bone is less than 1.5 mrem, 9 mrem, and 28 mrem, respectively.

The operator ingress or egress is to be made by motor vehicle to a point adjacent to the control building portal. The car is assumed to travel at an average speed of 15 mph in both directions. The operator walks between the control building and the motor vehicle at an average rate of 4 mph, and personnel will be equipped with supplied air breathing 25
apparel.

49

6.3.1.2 System Design

The Control Room Habitability System is designed to provide a safe, comfortable and appropriately equipped location for personnel controlling plant operations during normal times and during accidents. Features incorporated into this Habitability System to assure these aspects are described below.

A concrete enclosure and special sealed doors are important features in this habitability system. The details of this shielding and 22
the shield wall thicknesses are described in Section 12.1.2.4. Bases used in the design and analyses are also presented in Section 12.1.2.4. Factors considered in these design analyses include thermal margin beyond design basis requirements and the associated activity releases
49 and gamma shine from the containment/confinement. In addition, other features, such as two widely separated air intakes are provided to mitigate the consequences of low probability accidents beyond the design 22
basis.

Another important feature of the Control Room Habitability System is the HVAC system. The HVAC system for the Control Room contains two 100% capacity air conditioning units and two 100% capacity exhaust fans, with one unit each normally operating and the other unit on standby; and two 100% capacity emergency air supply filter units. Complete details of the system are presented in Section 9.6.1. A P&ID of the system is also provided and shown in Figure 9.6-1. This HVAC system contains several aspects that are significant to the Control Room Habitability System. One of these aspects is that this system has full capacity redundancy to assure the capability for controlling the environment after any single component or subsystem failure. A second aspect is the capability to maintain the Control Room at a slightly greater pressure than any of its surroundings at all times. A third aspect of interest to the habitability system is the air cleanup units that purify both make-up and recirculated air flows during postulated accidents. Each air cleanup unit in the HVAC system contains two banks of HEPA filters and a bank of carbon adsorbers. The filter capability is discussed in Section 9.6. A fourth aspect is the capability for selecting emergency pressurizing air during accidents from widely separated intakes, one located at the SE Corner of the Control Building roof at approximately elevation 880' and the other at the NE corner of the Steam Generator
49 Building Auxiliary Bay at approximately elevation 858'. This along with instrumentation provided, allows selection of the cleaner air source during such periods.

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Table 7.1-6

Safety Related Electrical Instrumentation
And Control Equipment

Margin	Equipment	Measured Parameter	Purpose	Location	Temp.	Press.	Humidity	Radiation	Chemical	Seismic	Vibration
39	EVST and										
50	Thermocouples	EVST Outlet Na Temperature	Safe Shutdown PAM	RSB 331, 357, 360	X	NA	X	X	X	X	TBD
47	Thermocouples	OHX Na Outlet Temperature	Safe Shutdown PAM	RCB 107B	X	NA	X	X	X	X	TBD
47											
39	Primary Sodium Makeup Pumps	None	DHRS	RCB 103, 104	X	X	NA	X	X	X	TBD
	EVST Sodium Pumps	None	EVST Cooling	RSB 357A, 360	X	NA	X	X	X	X	TBD
	EVST NaK Pumps and Fld. Panels	None	EVST and Reactor cooling	RSB 352A, 353A	X	NA	X	X	X	X	TBD
	EVST ABHX and Field Panels	None	EVST and Reactor cooling	RSB 352A, 353A	X	NA	X	X	X	X	TBD
39	Makeup Pump Field Panels	None	Reactor DHRS Service	RCB 105A	X	X	NA	X	X	X	TBD
	EVST Sodium Pump Field Panels	None	EVST Cooling	RSB 352A, 353A	X	NA	X	X	X	X	TBD
	Control Room Panel	None	DHRS & EVST cooling control	CB 431	X	NA	X	NA	X	X	TBD
46	Valve Operators	None	Various	RCB 107B	X	X	NA	X	X	X	TBD

TABLE 7.1-6

Safety Related Electrical Instrumentation
And Control Equipment

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	Equipment	Measured Parameter	Purpose	Location	Temp.	Press.	Humidity	Radiation	Chemical	Seismic	Vibration
	Na Valve Operators	None	Various	RSB 357A, 360	X	NA	X	X	X	X	TBD
23	Na K Valve Operators	None	Various	RSB 352A, 353A	X	NA	X	X	X	X	TBD
46	Local Control Panels	None	DHRS & EVST Cooling Control	RSB 311 RCB 105V	X	NA	X	X	X	X	TBD

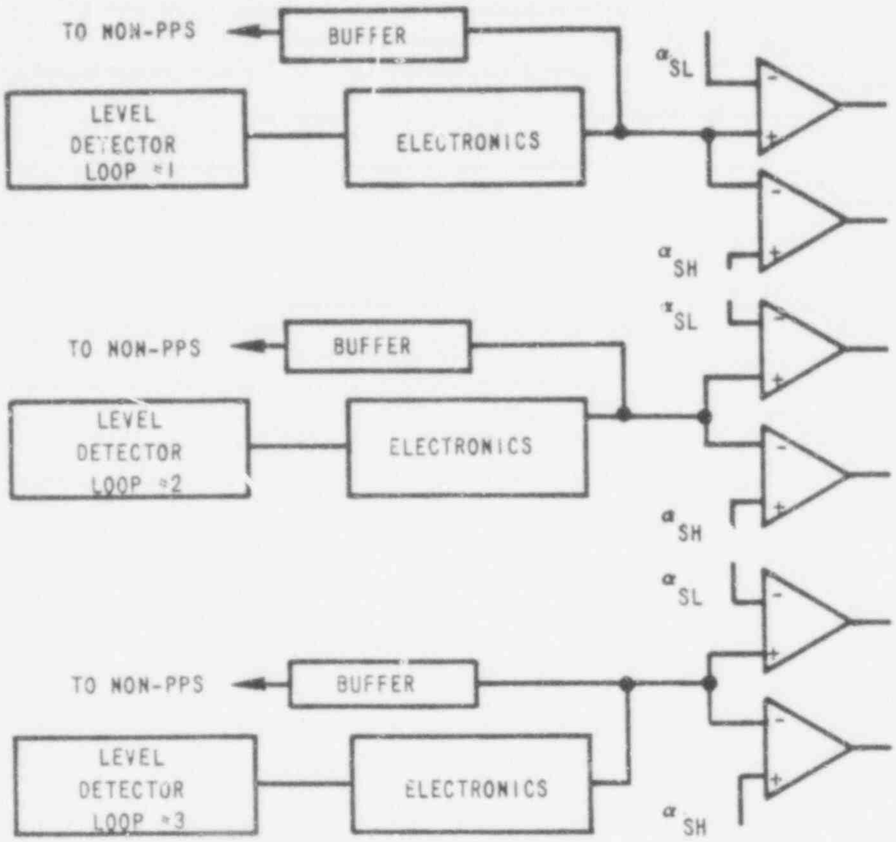

Figure 7.2-9 Functional Block Diagrams of the Steam Drum Level Subsystem. One Channel of Three is Shown.

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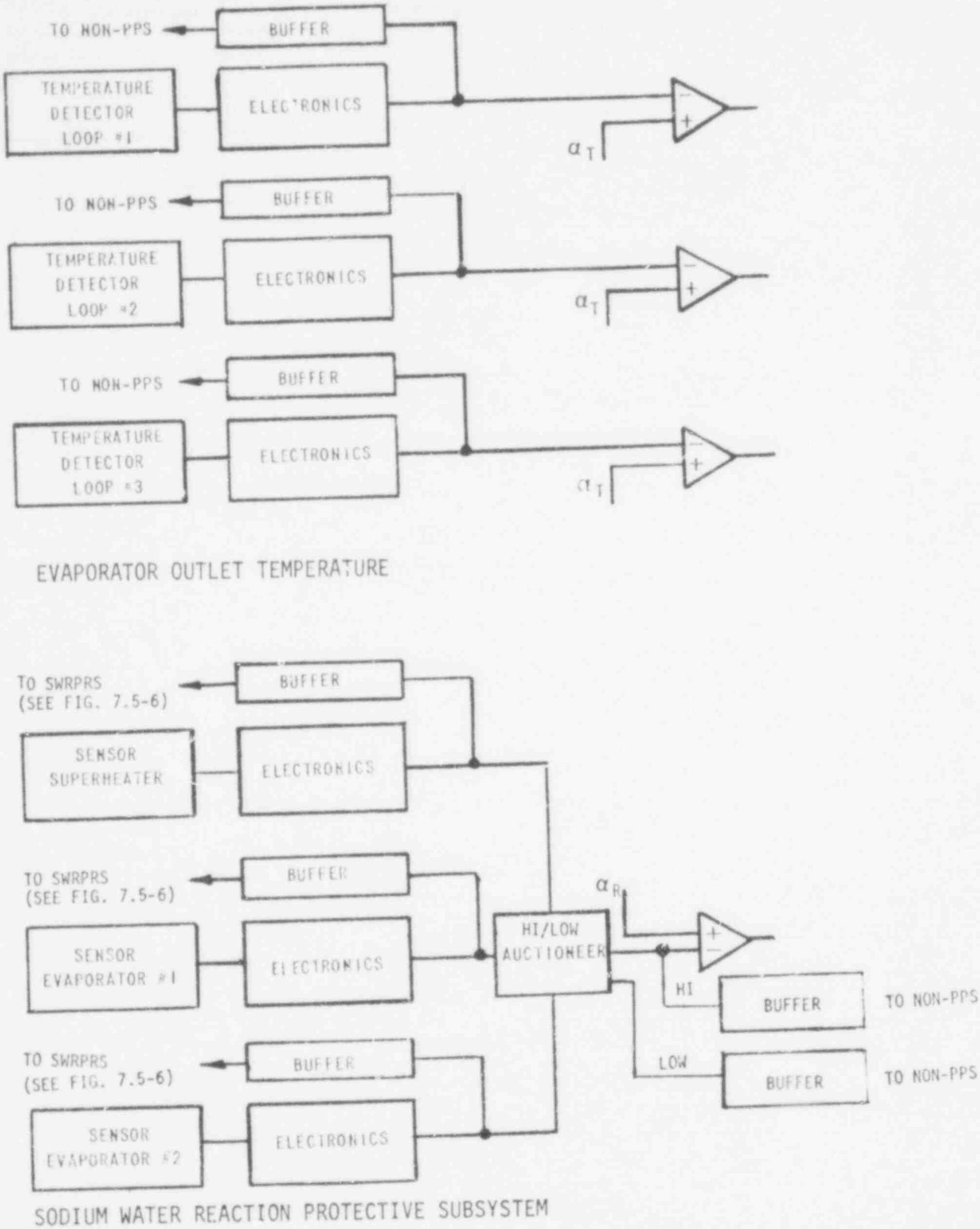

Figure 7.2-10 Functional Block Diagrams of the Evaporator Outlet Sodium Temperature and Sodium Water Reaction Protective Subsystems, One Channel of Three is Shown

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provide the required time response. The thermowell is also swaged at the tip. The thermocouples are spring loaded against the bottom of the well. Although failures of the wells are not expected, as confirmed by tests and analysis, the head of the thermowell, including the cable penetration, is sealed to provide a secondary boundary for the sodium. Tests have shown that this system will provide a time response less than 5 seconds. Flexible mica, polyimide and fiberglass insulated thermocouple extension wires in conduit are used to bring the signals out of the Heat Transport System Cell. The signals are then routed to the containment mezzanine into reference junctions and signal conditioning equipment. The conditioned signals are transmitted to the control room for the Reactor Shutdown System logic. The Reactor Shutdown System provides buffered signals to the PCS and DH & DS.

Primary and Intermediate Hot and Cold Leg Temperature

The primary and intermediate hot and cold leg temperatures are
measured to determine and record operating conditions and to calorimetrically
49 calibrate the permanent magnet flowmeters. The measurement is made by two
duplex element resistance temperature detectors (RTDs) per loop, installed in thermowells. Although failures of the wells are not expected, as confirmed by tests and analysis, the head of the thermowell, including the cable penetration, is sealed to provide a secondary boundary for the sodium. The signals from the RTDs are routed to signal conditioning equipment which converts the resistance variation to a standard signal level for transmission to the DH & DS.

Primary and Intermediate Pump Discharge Pressure

The primary and intermediate pump discharge pressure measurements monitor pump performance and the primary loop/intermediate loop differential pressure. The measurements are made by pressure elements installed in the elevated section of the drain line from the discharge piping of the sodium pump. NaK filled capillaries from the pressure elements are connected to pressure transducers which develop electrical signals proportional to the pressure. These pressure transducers provide a secondary boundary if the bellows in the pressure elements should fail. The conditioned signal is supplied to the DH & DS.

Intermediate IHX Outlet Pressure

43 The intermediate IHX outlet pressure measurement is used to
monitor the loop and IHX operational performance history. The measure-
43 ments are made by pressure elements installed in the intermediate loop
piping between the IHX and the superheater. NaK filled capillaries from the pressure elements are connected to pressure tranducers which develop electrical signals proportional to the pressure. The pressure transducers provide a secondary boundary if the bellows in the pressure elements should fail. The conditioned signal is supplied to the DH and DS.

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IHX Differential Pressure

The primary sodium pump discharge pressure and the IHX Intermediate Loop outlet pressure detectors are used to provide a differential measurement of the IHX Primary/Intermediate pressure difference. The differential pressure measurement is alarmed to alert the operator for corrective action to assure intermediate to primary differential pressure
50 49 is maintained above the minimum required.

Intermediate Pump Inlet Pressure

The intermediate pump inlet pressure measurements provide a signal to monitor pump performance. Used with the pump outlet pressure, the differential pressure across the pump is obtained. In the primary loop, the reactor pressure is used for this surveillance. The measurements are made by pressure elements installed on the piping between the evaporators and the pump inlet. NaK filled capillaries from the pressure elements are connected to pressure transducers which develop electrical signals proportional to the pressure. The pressure transducers provide a secondary boundary if the bellows in the pressure elements should fail. The conditioned signal is supplied to the DH & DS.

Intermediate Expansion Tank Level

Two separate level measurement channels are provided; both channels are used for indication in the control room and DH & DS and for alarm. Alarm channels provide a broad range measurement that covers possible high and low levels during plant operation as well as the IHTS
49 fill level. The DH & DS uses measurements for intermediate loop sodium inventory (see also Section 7.5.5). The level probes are designed to be replaceable.

Evaporator Sodium Outlet Temperature

Three thermocouple channels are provided to measure the sodium temperature at the outlet of the evaporators in each loop. The thermocouples are placed just after the pipes from each evaporator join to form two single lines. These three signals are conditioned separately and provided to the Reactor Shutdown System logic. The Reactor Shutdown System in turn provides buffered signals to the DH & DS.

7.5.2.1.2 Sodium Pumps

Sodium Level

Sodium level is measured in each pump tank. The signal provides indication and alarm. The alarm is used to notify the operator of abnormal operation and allow initiation of action to prevent pump damage. The signal is also provided to the DH & DS where it can be used in calculation of sodium inventory.

7.5.4.1 Design Description

The following subsystems make up the FFM system.

1. Cover Gas Monitoring

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

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

34

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

2. Reactor Delayed Neutron Monitoring

34

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

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

3. Failed Fuel Location

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

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

7.5.4.1.1 Cover Gas Monitoring Subsystem

The Cover Gas Monitoring Subsystem receives sample gas
50 from the Inert Gas Receiving and Processing System. The sampling
control system is located in a shielded cell in the Reactor Service
Building. The cover gas passes through a sodium vapor trap and into
34 a charcoal delay bed.

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The purpose of the delay bed is to increase the concentration of gas (primarily
xenon and krypton) isotopes of interest, and allow the argon cover gas to
pass through without delay, thus enhancing signal-to-signal background ratio.
A fixed-channel analyzer, located in the reactor service building, analyzes
the cover gas sample for radioactive xenon and krypton fission products. If
a leak in a fuel pin occurs, an increase in activity will occur. This indication
of a leak will be annunciated by the Plant Annunciator. A multi-channel
34 analyzer, including a mini-computer, analyses the signal from the detector
to display the entire gamma spectrum, thus providing burnup and other information
characterizing the failed fuel pins to supplement the failure location subsystem
through correlation with core and blanket history.

7.5.4.1.2 Reactor Delayed Neutron Monitoring Subsystem

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

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

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

Signals from the three BF_3 detectors are routed to individual inputs
of a preamplifier. Each input of the preamplifier provides individual
electronic discrimination against gamma-caused counts from each detector. After
discrimination, the neutron-caused counts from the three BF_3 detectors, 1 per
coolant loop, are combined in a summing circuit, and routed to a local control
34 panel located in the Reactor Containment Building. The local control panel
provides controls for remote adjustment of the preamplifier discriminators,
a low voltage power supply, a detector-bias high-voltage supply, a comparator
for low detector bias voltage alarm and a front panel function switch to select
TEST, CALIBRATE or OPERATE modes. These signals are routed to a panel in the
main control room displaying the output on three five decade logarithmic count
34 rate meters and a three pen five decade logarithmic strip chart recorder.

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7.5.5.3.1 Design Description

General

Steam or wate.· leakage into sodium increases the hydrogen and oxygen concentrate in the sodium stream. The leak detection is based on measurement of the hydrogen and oxygen concentration in the sodium.
47 Oxygen concentration measurements in the sodium are complementary to the hydrogen detectors, thus providing a diverse method to ensure early detection.

To provide a sensitive leak detection capability, the background concentrations of oxygen and hydrogen are maintained as low as practicable. The hydrogen in the sodium is removed by cold trap precipitation of sodium hydride. Oxygen in the sodium is removed in the same way by sodium oxide precipitation. The background concentration of hydrogen in the system is
50 normally maintained below 200 ppb through cold trap operation. Oxygen
33 31 13 concentration is maintained at about 2 ppm.

Hydrogen Detectors

The measurement of hydrogen in the sodium is performed by allowing the hydrogen to diffuse through a thin walled nickel membrane, and detecting the hydrogen with an ion gauge and an ion pump. The monitor may operate in a static mode using the ion gauge to monitor steady state hydrogen concentration, since hydrogen content is directly related to the pressure measured in the chamber. The monitor may also operate in a dynamic mode, using the ion pump to constantly pump the chamber since then the hydrogen concentration is directly related to the ion-pump current.

47 34

Oxygen Detectors

Oxygen electro chemical cells are used to continuously monitor in-sodium concentration and consist of a reference oxygen electrode separated from the sodium by a solid electrolyte. The electrical potential drop between the reference electrode and the sodium measures the in-sodium oxygen concentration.

Detectors Location

The three steam generation loops utilize identical oxygen-hydrogen (O-H) detector modules at the following locations

(a) On one evaporator outlet piping downstream of mixing tee. This leak detection module under normal operation samples sodium that is a combination of sodium exiting evaporator A and evaporator B. However, by the use of valves in the sample line, the leak detection module may be utilized to sample sodium exiting either evaporator A or evaporator B.

(b) Superheater vent line. This leak detection module under all conditions samples only sodium exiting the superheater vent line.

(c) On one evaporator vent line downstream of evaporator A/B vent line junction. This leak detection module under normal operation samples sodium that is a combination of sodium exiting the evaporator A vent line and evaporator B vent line. However, by the use of valves in the sample lines, the leak detection module may be utilized to sample sodium exiting either evaporator A vent line or evaporator B vent line.

(d) On one superheater outlet piping. This leak detection module under normal operation samples sodium that is a combination of sodium exiting both the superheater outlets. However, by the use of valves in the sample lines, the leak detection module may be utilized to sample the sodium exiting each sodium outlet independently.

47 Figure 5.1-4 shows the instrumentation location.

Indication in Control Room

Measurements from the hydrogen and oxygen detectors are monitored by the Data Handling and Display System. Each channel is limit checked and its trend is limit checked. Low level, intermediate level and high level alarms and channel failure alarms are also provided to the Plant
47 Annunciator System.

System Operation

47 The leak detector detects two leak signal categories:

1. a strong signal in a single pass,

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47 2. detection of a gradual concentration increase or decrease through several passes through the sodium.

Figure 7.5-4 illustrates typical first pass hydrogen concentration change as a function of water leak rate. As illustrated, a change in hydrogen concentration of a few ppb would be indicated at the detector for leak rates in the range of 10^{-4} lb/sec. Approximately one minute is required for the hydrogen to reach the detector and signal a leak. Detection capability can be extended to smaller leak sizes through the use of a rate of rise detection system. Several passes of sodium through the system would be required to allow the hydrogen concentration to build up. The sensitivity of this system will allow detection of leaks in the range of 10^{-5} lb/sec. Similarly, Figure 7.5-4A indicates the first pass oxygen
48 47 concentration as a function of water leak rate. Figure 7.5-5 illustrates the hydrogen concentration change with time for various sizes of leaks.

7.5.5.3.2 Design Analysis

A Steam Generator Leak Detection System is provided to comply with CRBRP General Design Criteria 4 which calls for provision of leak detection in the Steam Generators. In order to show how the criterion will be satisfied, a review of leak damage studies is presented with the
13 resulting instrumentation requirements.

Leak Damage Studies

Experimental studies have been conducted in the United States and in Europe over the past ten years which have given a broad base background to the understanding of the behavior of leakage damage effect. Most of the experimental data taken with 2-1/4 Cr- 1 Mo material have been obtained by injecting water or steam through hole type geometrics at selected target configurations (jet leaks). In general, results of these studies have indicated that both adjacent tube wastage and self wastage
45 13 are possible damage mechanisms, as described in Reference 1, 3 and 4.

Adjacent tube wastage will occur with the proper leak size and orientation. For very specific conditions and geometries, some experiments have been performed where adjacent tube wastage occurred very rapidly in a localized area. Relating these specific conditions to CRBRP, adjacent tube wastage could occur which would result in tube failure in a very short period of time, less than one minute. However, it is not likely that leaks would be optimized as to leak geometry, location and orientation, as those utilized for the experiments. In the event that this did occur, the steam generator rupture discs provide necessary protection.

13

The second class of damage is self-wastage around the leak site. Some experiments have noted that some very small leaks have experienced a sudden enlargement after a period of relatively steady operation as reported in Reference 1. The effect of this type of characteristic has been studied by the GE/ANL Steam Generator Systems Development Program and
46 13 is reported in References 3 and 4.

Design Requirements

The design requirements for the Steam Generator Leak Detectors
13 have been selected as described below.

SGS Leak Detection Requirements

47

	Hydrogen Detectors	Oxygen Detectors	
50 Sensitivity	6 ppb	24 ppb	23
Range	0.04-2 ppm	0.1-10 ppm	
Response Time	≤30 sec.	≤30 sec.	

47

Instrument Sensitivity

- The wastage rate studies for jet leaks show that leaks below 10^{-4} lb/sec persist without major damage for more than one loop transit time. The loop transit time can be calculated
47 from a 13.49×10^6 lbs/hr flow rate and 4×10^5 lbs sodium inventory in the IHTS loop; the hydrogen generated from the quantity of H_2O leaked in one transit time divided by the total sodium inventory yields an increase of 6.3 ppb in the
50 47 concentration of hydrogen, thus a 6 ppb sensitivity for the hydrogen detectors.

- A resolution of 3 ppb change in the hydrogen background con-
50 centration ranging from 60 - 200 ppb (i.e., a change of 3-4%) under steady-state SG operation is a design goal for the leak detector.

- The oxygen detector is as sensitive as the hydrogen detector. Taking into account an oxygen background concentration of
47 33 1ppm (with 2ppm maximum), the sensitivity is 24 ppb.

13

Instrument Range

- Detection capability of leaks up to 10^{-1} lb/sec.

47

13

Instrument Availability

- Sodium loop leak detection capability provides continuous monitoring and indication of the impurity level whenever sodium and water/steam co-exist in the steam generator modules.

13

13 Operation Requirements

- In order to effect an orderly plant shutdown which minimize plant unavailability, the following operator actions are required.

	Alarm	Leak Size (lb/sec)	Operator Action
	Low	$<2 \times 10^{-5}$	Confirm leak Monitor leak data
	Intermediate	2×10^{-5} to 5×10^{-3}	Confirm leak Initiate orderly loop Shutdown
47	High	$>5 \times 10^{-3}$	Confirm leak Initiate rapid module blowdown

For leakages greater than about 0.1 lb/sec of water, the pressure buildup in the system will occur rapidly, causing the Sodium-Water Reaction Pressure Relief System to be activated (See Section 7.5.6).

7.5.6 Sodium-Water Reaction Pressure Relief System (SWRPRS) Instrumentation and Controls

7.5.6.1 Design Description

7.5.6.1.1 Function

The Sodium-Water Reaction Pressure Relief System Instrumentation and Control equipment detects the inception of a large or intermediate sodium-to-water leak in any of the steam generator modules (see Section 5.5.2.6). The following automatic actions in the affected loop are initiated by a large sodium-to-water leak which bursts the steam generator
43 module main rupture discs:

a. Initiation of reactor and main sodium pump trip.

b. Trip of recirculating water pumps.

c. Isolation of steam generator modules water side.

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7.5.7 Containment Hydrogen Monitoring

The objective of Containment Hydrogen Monitoring is to provide indication in the Control Room of the hydrogen concentration in the upper levels of containment.

7.5.7.1 Design Description

The hydrogen instrumentation consists of three self-contained electro-mechanical cells located near the top of the RCB with signal lines running outside of containment to preamplifiers. From there, the amplified signals go to the display and alarm panel in the Reactor Control Room so that continuous readout will be provided to the plant operator.

7.5.8 Containment Vessel Temperature Monitoring

The objective of Containment Vessel Temperature Monitoring is to provide indication in the Control Room of the containment vessel temperature.

7.5.8.1 Design Description

The temperature instrumentation consists of thermocouples provided for the outside of steel containment vessel. Signals will be provided to the display and alarm panel in the Reactor Control Room so that continuous readout will be provided to the plant operator.

7.5.9 Containment Pressure Monitoring

The objective of the Containment Pressure Monitoring System is to provide indication in the Control Room of the pressure inside the containment above the oeprating floor.

7.5.9.1 Design Description

The pressure instrumentation consists of a pressure detector inside the containment vessel. Signals will be provided to the display
25 and alarm panel in the Control Room so that continuous readout will be
provided to the plant operator.

7.5.10 Post Accident Monitoring

Table 7.5-4 provides a listing of those parameters which are
monitored to assure the plant is maintained in a safe shutdown
status. Equipment to condition, display, and record the instrument
50 signals is provided in the Control Room. The instruments which
serve the Post Accident Monitoring function are included in those discussed
49 in Sections 7.4.1, 7.5.2, 7.5.3, 7.5.8, 7.5.9, and 7.6.3. The functions
of these instruments corresponding to the parameter of Table 7.5-4
50 are described in the following paragraphs.

The reactor sodium level is monitored to enable the operator to determine whether manual action is necessary to mitigate conditions which may cause low Reactor Vessel sodium levels. The operator may also use the reactor sodium level to determine conditions in the primary sodium systems such as the volume of primary coolant leakage.

The IHX (Intermediate Heat Exchanger) inlet and outlets temperatures are monitored to verify the heat transfer from PHTS to the IHTS. These monitors allow the operator to take manual actions necessary to assure decay heat removal from the reactor is maintained within design limits.

The DHRS cold leg temperature is monitored to assess the systems' decay heat removal performance so the operator may take manual actions necessary to achieve or maintain core temperature at a safe level.

Reactor Containment Building pressure and temperature monitors are provided to follow pressure and temperature changes due to an accident, and provide confidence that the accident consequences are within the capability of the Containment Vessel. Also, these monitors can be used to detect conditions beyond the design basis as discussed in Reference 10 of Section 1.6.

The PWST (Protected Water Storage Tank) water level is monitored so the operator is assured of the adequacy of that water supply to remove reactor decay heat for an extended period of time. In addition, the level monitor provides indication of the long term need to refill the PWST or draw auxiliary feedwater from an alternate source.

The Auxiliary Feedwater Flow is monitored to inform the operator of normal, abnormal or inadequate flow. He can take manual actions to provide adequate flow and thus maintain adequate decay heat removal through the steam generator system.

Steam drum level and pressure are monitored to identify an accident and to allow the operator to take manual actions to initiate and control systems required to achieve and maintain decay heat removal via the Steam Generator System.

EVST sodium hot leg temperature is monitored to enable the operator to make manual actions during normal, transient, and accident conditions which are necessary to prevent or mitigate the exceeding of
50 Ex-Vessel Storage Tank and stored fuel assembly design limits.

References to Section 7.5

1. Ford, J. A., "A Recent Evaluation of Foreign and Domestic Wastage Data from Sodium Water Reaction Investigation", APDA CTS-73-05, January, 1973.

2. Morejon, J.A., "Sodium-to-Gas Leak Detection Mockup Tests", N707-TR-520-C04, September 17, 1975. (Atomics International)

3. Greene, D. A., J. A. Gudahl and J. C. Hunsicker, "Experimental Investigation of Steam Generator Materials by Sodium-Water Reactions, Volume 1, GEAP-14094, January 1976.

4. Gudahl, J. A. and P. M. Magee, "Microleak Wastage Test Results",
46 GEFR-00352, March 1978.

TABLE 7.5-1

INSTRUMENTATION SYSTEM FUNCTIONS AND SUMMARY

	System	Measured Parameters	Instrument	Measurement Location	Purpose
	Flux Monitoring	Source Range*	BF_3	Thimbles on periphery of guard vessel	Determines: 1. Flux status at shutdown, startup and power levels 2. Signals to PPS logic (except source range) 3. Signals for reactor and plant control (power range only) 4. Signals for display, annunciation and recording
		Wide Range*	Fission Chambers	Thimbles on periphery of guard vessel	
		Power Range*	B-10, Compensated Ion Chamber	Thimbles on periphery of guard vessel	
50	Heat Transport Primary/ Intermediate Loops	Reactor Inlet Pressure*	Pressure Element	Cold leg primary loop	PPS and display PHTS performances
50		Primary and Intermediate Flow*	PM Flowmeter	Cold leg of primary and intermediate loops (hot leg in intermediate loop 2)	PPS, Plant Control and Display, PHTS performance
		IHX Primary Outlet Temperature*	Thermocouple	Cold leg piping nearest to IHX primary outlet	Plant Control System (PCS), PPS, and Display
50		Primary and Intermediate Hot and Cold Leg Temperature	Resistance Temperature (RTD)	Primary and Intermediate hot and cold leg	Surveillance, display and use to calorimetrically calibrate PM flowmeters
50		Primary and Intermediate Pump Disharge Pressure	Pressure Elements	Drainline from discharge piping of the loops's sodium pump	Surveillance, display and monitor differential pressure between primary & intermediate loops PHTS performance
50		Intermediate IHX Outlet Pressure	Pressure Elements	Intermediate between IHX & Superheater	Surveillance, display & monitor differential pressure between intermediate loops

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TABLE 7.5-1 (Continued)

System	Measured Parameters	Instrument	Measurement Location	Purpose
Heat Transport Primary/ Intermediate Loops (cont'd)	Intermediate Pump Inlet Pressure	Pressure Elements	Pipes between evaporator and pump inlet	Display pump performance
	Intermediate Expansion Tank Sodium Level	Level Probe	Intermediate expansion tank	Display-intermediate loop sodium inventory and alarm
	Evaporator Sodium Outlet Temperature*	Thermocouple	Downstream where the two evaporator outlets join into the header	PPS and display
Sodium Pumps	Sodium Level	Level Probe	Pump Tank	Display-used for sodium inventory and pump protection (alarm)
	Primary/Intermediate Pump Speed*	Tachometer	Main Shaft of each pump	PPS, display, pump speed control, performance
	Po[illegible] Running	Speed Switch	Pumps	Display, performance
	[illegible] Instrument[illegible]	Various	Pumps	Display, pump performance
Steam Generator	Sod[illegible]	Venturi	Superheater sodium outlet (1 loop)	Display & superheater & evaporator performance
	Sodium Temperature	Thermocouple	Superheater evaporator outlet (3 loops)	Display & steam generator performance evaluation
	Sodium Pressure	Pressure Element	1 loop-superheater inlet, outlet (both legs) and evaporator outlet (one leg)	Display & steam generator performance evaluation
	Feedwater Flow*	Venturi	Inlet line to steam drum (feedwater)	PPS, display & steam generator performance evaluation
	Superheat Steam Flow*	Venturi	Outlet of each superheater (steam)	PPS, display & steam generator performance evaluation
	Steam Drum Drain Flow	Orifice	Steam Drum Drain line for each steamdrum	Performance evaluation
	Evaporator Inlet Flow	Venturi	Inlet to one evaporator (1 loop)	Performance evaluation

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TABLE 7.5-4

POST ACCIDENT MONITORING

	Parameter	Sensor Location
	1) Reactor Sodium Level	RCB
	2) IHX Inlet Temperature	RCB
	3) IHX Outlet Temperature	RCB
	4) DHRS Cold Leg Temperature	RCB
	5) RCB Pressure	RCB
	6) RCB Temperature	RCB
	7) PWST Level	SGB
	8) Auxiliary Feedwater Flow	SGB
	9) Steam Drum Level	SGB
	10) Steam Drum Pressure	SGB
50	11) Deleted	
49	12) EVST Outlet Piping Temperature	RSB

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Rotating Plug drive system/IVTM grapple position
Rotating plug drive system/IVTM hold down sleeve
Rotating plug drive system/EVTM position
44 EVST drive motors/EVTM grapple position

7.6.2.2 Design Analysis

Postulated Reactor Refueling System (RRS) accidents with potentially severe consequences were analyzed in detail to determine requirements for safety interlocks. The techniques employed included safety assurance diagrams, fault trees, mechanical and thermal analyses, and radiological release calculations. None of the analysis results showed off-site doses exceeding those presented in Section 15.5 or 15.7. The off-site doses in Section 15.5 and 15.7 resulting from postulated RRS accidents are all well below the 10 CFR 100 guideline exposures without taking credit for interlocks. It was therefore concluded that the RRS does not require safety interlocks. 44

7.6.3 Direct Heat Removal Service (DHRS) Instrumentation and Control System

7.6.3.1 Design Description

7.6.3.1.1 Function

The DHRS (fluid system and mechanical components as described in Section 5.6, and electrical components as described below) provides a supplementary means of removing long term decay heat for the remote case in which none of the steam generator decay heat removal paths are available.

The DHRS Instrumentation and Control System is provided to permit the monitoring of system conditions and to provide alarm indication of off-normal conditions. These are the same instrumentation and controls that are provided for EVST cooling (Section 9.1.3.1.5) and the reactor primary sodium overflow circuits (Section 9.3.2.5) with the addition of a few temperature monitoring instruments located on the NaK lines connecting the overflow heat exchanger with the EVST NaK cooling loops (see Figures 9.3-2 and 9.3-3).

7.6.3.1.2 Design Criteria

Design criteria that are applicable to DHRS electrical equipment are as follows:

A. No single failure of an instrument, interconnecting cable or panel shall prevent a key process variable from being monitored.

B. DHRS valves shall be manually operated and DHRS electrical equipment shall be manually controlled (see 5.6.2) from a panel in the Control Room to provide 1/2 hour start up capability.

C. Physical and electrical separation of redundant portions of DHRS
26 (EVS cooling system, primary makeup pumps, instrumentation, and

controls) shall be provided.

D. Electrical power supplied to DHRS electrical equipment shall be independent of off-site power.

46 E. DHRS control instrumentation and DHRS electrical equipment shall function during and after an SSE.

F. Capability for periodic calibration and testing of DHRS electrical equipment shall be provided.

7.6.3.1.3 Equipment Design

As shown on Figure 5.1-7, the DHRS is part of the primary sodium processing, and the EVS Sodium Processing System. Description of the functioning of these systems for reactor decay heat removal is provided in Sections 9.1.3 and 9.3.2. The P&I diagrams are given in Figures 9.3-2 and 9.3-3.

DHRS electrical equipment meets the design criteria listed in Section 7.6.3.1.2 above in the following manner:

A. Control Systems

The following DHRS control functions are provided from separate, redundant control panels (local and main control room):

(1) Remote manual control of voltage to all NaK and sodium pumps.

(2) Remote manual control of ABHX dampers and fan speed.

(3) Remote manual override of pump and ABHX interlock circuits.

(4) Remote manual control of all valves required to provide DHRS.

B. Monitoring Instrumentation

Some instrumentation required to monitor the functional performance of the decay heat removal process loops is redundant from the sensor out to and including the readout panel, so that a single failure of an instrument, interconnecting cable or panel does not prevent the process loop from being monitored. In those cases where a redundant sensor is not provided, separate indicators on separate panels are provided. Where redundant sensors are not provided, loss of the sensor does not prevent the acquisition of equivalent diagnostic information from other sensors on the process loop.

The following EVST cooling and DHRS process variables are monitored with completely redundant instrumentation (sensors, cabling, and panels):

50 * (1) EVST outlet sodium temperatures

46
26 * Required for post accident monitoring.

(2) Overflow vessel sodium level

*(3) Reactor overflow sodium inlet temperature

(4) EVST sodium pump control signal (each pump)

(5) EVST NaK pump control signal (each pump)

50 46 (6) Primary makeup pump control signal (each pump)

The following EVST cooling and DHRS process variables are monitored using a single sensor and redundant cabling and panels.

(1) EVST sodium flowrate (each loop)

(2) EVST NaK flowrate (each loop)

(3) Primary overflow makup sodium flowrate (each pump outlet)

46 (4) EVST airblast heat exchanger fan speed setting (each loop)

(5) EVST NaK expansion tank level (each loop)

(6) EVST sodium inlet temperature (each loop)

(7) EVST sodium and NaK remotely operated valve position indicators (each loop)

(8) EVST airblast heat exchanger damper position indicator (each loop)

(9) DHRS heat exchanger bypass valve position indicator

46 (10) DHRS NaK expansion tank level

C. Annunciation and Data Handling

The following EVST cooling and DHRS process variables are connected to the plant annunciator system to alert the plant operator of off-normal conditions:

(1) Low sodium and NaK pump gas cooling flow rate (each pump)

46 (2) High EVST sodium, EVST NaK and primary makeup pump stator temperatures (each loop)

26

(3) Low EVST sodium, EVST NaK and primary makeup flowrate (each pump loop)

(4) High and low EVST sodium inlet temperature (each loop)

(5) High and low EVST NaK expansion tank level (each loop)

(6) Hgih and low EVST sodium level

(7) High and low EVST sodium temperature

(8) Low sodium valve temperatures

(9) High and low DHRS expansion tank level

46 Key process variables that are connected to annunicators are also connected to the plant data handling and display system.

D. <u>Other Features</u>

Remotely operated valves in EVST cooling and DHRS circuits in-
46 corporate either "fail safe" or "fail in place" features and are provided with direct manual (reach rods on sodium valves) override capability in event of I&C or gas supply failures.

Type 1E power is supplied to the equipment and instrumentation required to provide the safety related functions of EVST cooling and DHRS as shown in Figure 7.6-13. This assures independence of off-site power.

Functional testing of all portion of DHRS that are not used during the course of normal operations will be tested on an annual basis during reactor refueling.

Equipment required to provide power to EVST and DHRS pumps, airblast heat exchangers and the monitoring instrumentation in the control panels shall be designed and tested to Seismic Category I requirements.

7.6.3.1.4 <u>Initiating Circuits</u>

Reactor decay heat removal through DHRS is initiated from the Control Room panel as described in Sections 9.1.3 and 9.3.2.

7.6.3.1.5 <u>Bypass and Interlocks</u>

When the DHRS is activated, automatic control of the EVST airblast heat exchangers is bypassed so that control of the complete system is manual. All valves in this circuit are also operated on a direct or
26 remote manual basis. The flow in the primary sodium overflow makeup

It is not anticpated that the FHC requires alpha-decontamination after handling each failed fuel assembly since the cell is an alpha-tight facility and can operate with contamination.

At regular intervals, swipe samples will be taken at various locations in the FHC and particle fallout samples (dishes) will be analyzed to monitor the amount of alpha-contamination. The samples will be compared to established contamination levels and will provide warning of any contamination buildup.

49 One method of minimizing the radioactive spread of alpha contamination through the EVTM and SFSC to other facilities will be by continuously filtering the FHC argon atmosphere. Alpha emitting particles suspended in the argon atmosphere will be removed through a HEPA filter bank installed upstream of the FHC argon gas circulation blowers. If excessive contamination buildup on the FHC liner is indicated, cleaning measures will be initiated.

It is, therefore, concluded that:

1) Maintenance of the FHC equipment is consistent with the current practice of maintaining equipment in hot cells.

2) Alpha-decontamination of the FHC atmosphere will be carried out on a continuous basis. Removal of alpha-contaminated deposits or debris will occur after a predetermined contamination threshold has been exceeded. 25

This section will cover the safety aspects of the FHC containment,
44 and the spent fuel transfer station. Spent fuel decay heat removal by the spent fuel transfer station and by the gas cooling grapple are discussed in Section 9.1.3.

9.1.2.2.1 Design Bases

Adequate shielding is provided in the FHC containment structure for radiation protection outside the cell, to meet the requirements and radiation
49 zone criteria of Section 12.1.

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
44 criteria of 10CFR100.

Criticality of fuel assemblies temporarily stored in the spent fuel transfer station is prevented by physical separation and by limiting their
44 number.
49

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49 44 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 (5fpm), 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. 20

44 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 will be 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.

9.1.2.2.2 Design Description

The top of the FHC is located at the operating floor of the RSB,
44 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,
49 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 20
floor are shielded by conventional concrete to protect the neighboring
49 vaults and the spent fuel shipping cask handling corridor against radiaation, 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
44 spent fuel transfer station.

The liner seams on the cell interior walls, roof and floor, and welded penetrations through the FHC walls, roof, and floor are alpha-tight
44 welded and inspected. Fuel transfer ports, the maintenance and service station port, window seals, and slave manipulator penetrations each have double elastomeric seals buffered with pressurized argon gas. Sealed cover glasses are provided on the interior side of the window penetrations.

44 The spent fuel transfer station within the FHC is shown in Figure 9.1-8. A maximum of 3 spent fuel assemblies in CCP's can be stored in

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where contamination can be contained and where modification to shop tools can be made.

35 The welding cart provided by this group is used primarily in the Reactor Containment Building. The welding cart is sized such that it can be moved to the job within the Reactor Containment Building. After being lowered through equipment hatches, or transported in the RCB elevators,
35 49 it can be moved around on various floors of the building.

9.2.1.2.2 Handling Equipment

Most of the handling equipment provided is used within the Reactor Containment Building to lift and transport components. Some handling equipment is also provided for use in the Steam Generator Building.

All the principal overhead handling systems for servicing safety-related equipment are designated as Seismic Category I Cranes.

All the structural, automatic and manual mechanical and electrical components of the Category I cranes will be designed so that no single failure or malfunction will result in dropping or losing control of the heaviest loads to be handled.

Design, fabrication, installation, inspection, testing and operation of the Category I cranes will be defined in the crane specifications to ensure capability of the crane to retain the maximum design load during a Safe Shutdown Earthquake, although the crane may not be operable after the seismic event.

For those cranes which may be used for construction prior to start of plant operation, a separate performance specification will be prepared and at the end of the construction period, the crane handling system will be modified to conform to the performance requirements of permanent plant service. After construction use, the crane will be thoroughly inspected using nondestructive examinations and will be
performance tested. 25

49 Lifting fixtures are provided for the purpose of handling PHTS components and large maintenance equipment. These fixtures interface with the component through the use of adapters. Special lifting fixtures, for the purpose of installing or removing other components, are the responsibility of the component system, and are not provided by the Nuclear Island General Purpose Maintenance Equipment System.

Patterned bag containers, for handling sodium-service and/or radioactive service components, are provided. Patters are provided for these bags, and each bag has an envelope interface with the component to be handled.

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Transport dollies are provided to move components between the Reactor Containment Building and the Reactor Service Building. A small dolly is provided to transport components which will pass through the service airlock. A large component transporter is provided to utilize the Refueling System gantry rails, and is positioned on the gantry rails when components too large for the airlock must be moved between the two buildings. The large transporter is removed from the gantry rails and stored elsewhere when not in use.

The large component floor valve is provided to maintain containment of the inert atmosphere and radioactive gases during the performance
43 of general purpose maintenance in the PHTS primary pump, EVST cold trap and the PHTS cold trap access ports.

Interface with the building floor at each of the access ports is provided by means of an adapter. A separate adapter will be provided
43 for the PHTS primary pump and a common adapter for both traps. 1

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9.2.2.2 Design Description

The Sodium Removal and Decontamination (SR&D) System is composed of three subsystems located as shown on Figure 9.2-1. These are:

49 1) Primary Sodium and Decontamination (PSR&D) System - for
removing sodium from fuel handling. primary reactor vessel
internals, and reactor enclosure components, and decontam-
43 35 inating the Primary System pump of fission and corrosion
product plate-out.

49 2) Intermediate Sodium Removal (ISR) System - for removing
sodium from the modular steam generator and Intermediate
System pump.

49 3) Small Component Autoclave (SCA) - for removing sodium and NaK
from Auxiliary Liquid Metal and Fuel Handling System com-
ponents.

The Primary Sodium Removal and Decontamination System, shown in Figure 9.2.2, is located in an 83 ft. deep cell that is below the operating floor in the northwest part of the Reactor Containment Building (RCB). This system is remotely monitored and controlled from a control panel on the RCB operating floor.

The Intermediate Sodium Removal System, shown in Figure 9.2-4, is
located in the maintenance bay of the Steam Generator Building. Most of
this system is manually controlled. The critical process parameters are
49 monitored and controlled at the panel board located near the process equip-
ment. This facility will be designed and built after reactor startup.

The small component autoclave is a horizontally positioned, 4-ft. diameter by 5-ft long vessel with cover, located just outside of the decontamination facility in the Reactor Service Building.

The cleaning vessels are designed to provide the process con-
ditions required to remove sodium from sodium-coated components by the
moist nitrogen process. These conditions include inerting, preheating,
35 reaction with water vapor diluted with nitrogen, water rinsing, and drying.
A decontamination phase is provided in the Primary Sodium Removal and
Decontamination System. 14

In the PSR&D System, the moist nitrogen gas and all liquid can
50 be recirculated to enhance process control. A vacuum of ~1 in. Hg
(absolute) and/or hot gas at 180° F is used for drying. The waste water
rinse liquids are sent to the Radioactive Liquid Waste System, and the
waste gases are sent to the Heating and Ventilation System. The reprocessed
waste rinse water (purified by the Radioactive Waste System) is reused in
49 the cleaning operation.

In the ISR System, the contaminated water (tritium only) is transferred to the waste water system for disposal, and the supply
49 water is obtained from the treated water system.

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Additional information on controlling and disposing of hydrogen follows. Hydrogen concentration in the Primary Sodium Removal and Decontamination (PSR&D) System is controlled by:

Regulating water vapor concentration in the water vapor nitrogen (WVN) mixture introduced to the cleaning vessel.

Regulating nitrogen purge rate.

In the normal sodium removal process, (1) the vessel is purged with N_2 until the O_2 level is below 1%, and (2) saturated steam at 15 psig is metered into the nitrogen purging stream. The steam and N_2 gas are mixed in a mixing tee and then introduced into the cleaning vessel where it reacts with the Na on the component. Water vapor concentration is controlled to be between 5-15% by means of remotely operated flow control valves. The lower limit is provided so that Na will effectively react with H_2O vapor, the upper limit of 15% is provided to preclude condensation of water vapor in the WVN mixture at 150°F, which is the component temperature. The normal operating condition maintains the hydrogen, which is a reaction product of Na and H_2O in the vessel, to below 1 v/o. WHen the H_2 concentration increases indicating a faster Na-water reaction, the steam supply introduced to the mixing tee is throttled down and the nitrogen purge rate is increased. Both of the flowrates are controlled remotely from the control panel. The H_2 concentration is monitored by a H_2 meter installed in the vent line for the effluent gases. Should the H_2 concentration exceed 4 v/o, the meter sensor will set off a high H_2 alarm and at the same time activate a control interlock which automatically shuts off the steam supply flow control valve while the nitrogen purge continues. The steam flow may be resumed when the excess hydrogen is reduced to below 1 v/o by continuing nitrogen purging.

In addition to the nitrogen purge feed and bleeding of the gas effluent, the gas mixture in the cleaning vessel is circulated by a blower through the cooler at the rate of 2,000 cfm during the WVN Na removal cycle. Although the primary purpose of the circulation is to remove the heat of reaction, it also promotes uniformity of the mixture of gases in the cleaning vessel. The pressure in the vessel is kept at 5 psig during operation by means of a back pressure valve. The positive operating pressure is provided to prevent air infiltration into the PSR&D system.

Since the maximum hydrogen concentration in the vessel is 4 v/o, any leak of the LCCV gas mixture will [illegible] diluted by air to below the hydrogen flammability limit in air which is [illegible] v/o.

The Primary Sodium Remo[illegible]l and Decontamination System is designated as nonnuclear safety in accordance with PSAR Section 3.2.2.4; namely, (1) failure of system will not result in exposures at the site boundary or beyond in excess of 0.5 rem whole body or its equivalent, and (2) failure of the system will not damage any in-service safety class components or the plant shutdown capability. 25

The equipment of the Primary Sodium Storage and Processing System is mounted in cells that have an inert atmosphere, but are accessible after system or component shutdown for inspection after de-inerting cells and radioactivity decay. The equipment is mounted or supported so that inspection of vessels, pumps and piping can be accomplished.

9.3.2.5 Instrumentation Requirements

Instrumentation and controls (I&C) are provided for operation, performance evaluation and diagnosis of the Primary Na Storage and Processing System. These functions are required for off-normal as well as for the full range of normal operation. Details of the I&C for the subsystem are shown in the piping and instrumentation diagram, Figure 9.3-2.
26 DHRS instrumentation is discussed in Section 5.6.2.1.6. The following I&C is required to ensure safe operation of and to prevent extensive damage to the Primary Na Storage and Processing System.

Temperatures at the inlet and outlet of all heat source and sink components, in conjunction with loop flow measurements are provided for all systems to monitor their status. Critical temperatures and flows are alarmed to alert the operator to off-normal operations. All EM pumps
46 are provided with winding temperature measurements and winding coolant low flow indication. These measurements are alarmed for off-normal conditions and interlocked to automatically shutdown the pump to prevent damaging it.

Storage tanks are provided with level measurements, which are alarmed for abnormal low and/or high level. This information, in conjunction with leak detection data, is required to diagnose external liquid metal leaks. The operator is alerted to NaK to sodium leakage by NaK expansion tank high-low pressure alarms. Differential pressure sensors and
50 46 flow meters are provided to alert the operator to possible plugging of the cold traps or insufficient cold trap flow. All the bellows seal valves are provided with leak detectors (Section 7.5.5.1). All valves are provided with position indicators. The stem portion of the sodium valve is monitored and alarmed for low temperature to ensure free operation and
46 protect the valve sodium seal from damage. To provide for continued operation and prevent possible system damage resulting from control system failures, hand controllers are provided for all controllers. The
50 hand controller allows the operator to manually operate the system while the defect is repaired.

Redundant temperature sensors are provided for each primary cold trap. High temperature conditions in either cold trap are indicated and alarmed in the Control Room to ensure that the cold trap is isolated prior to plant cooldown to refueling temperature. Thus plugging from high impurity content in the PHTS is precluded. 36

9.3.3 EVS Sodium Processing

9.3.3.1 Design Basis

See Section 9.1.3.1.1.

9.3.3.2 Design Description

See Section 9.1.3.1.2.

The equipment is mounted in cells that have an inert atmosphere, but are accessible after system or component shutdown for inspection after de-inerting. The equipment is mounted or supported so that inspection of vessels, pumps and piping for possible deterioration of NaK containment integrity can be accomplished.

9.3.4.5 Instrumentation Requirements

Instrumentation and controls (I&C) are provided for operation, performance evaluation and diagnosis of the NaK cold Trap Cooling System. These functions are required for off-normal as well as for the full range of normal operation. Details of the I&C for the subsystems are shown in the piping and instrumentation diagram, Figure 9.3-4. The following I&C is required to ensure safe operation of, and to prevent extensive damage to the NaK Cold Trap Cooling System.

Temperatures at the inlet and outlet of all heat source and sink components, in conjunction with loop flow measurements are provided for all systems to monitor their status. Critical temperatures and flows are alarmed to alert the operator to off-normal operations. The EM pump is
46 provided with winding temperature measurements and winding coolant low flow indication. These measurements are alarmed for off-normal conditions, and interlocked to automatically shutdown the pump to prevent damaging it.

46 The storage tank is provided with a level measurement which is alarmed for abnormal low and/or high level. This information, in conjunction with leak detection data, is required to diagnose external liquid
15 metal leaks. The operator is alerted to NaK to sodium or Dowtherm J
46 to Nak leakage by NaK storage tank high-low pressure alarms, in conjunction with the level measurement mentioned previously (see also Section 9.1.3). All the bellows seals valves are provided with leak detectors. All valves are provided with position indicators. To provide for continued operation and prevent possible system damage resulting from control system failures, hand controllers are provided for all controllers. The hand controller allows the operator to manually operate the system while the defect is repaired.

9.3.5 Intermediate Na Processing System

9.3.5.1 Design Basis

The system provides the capability to limit the oxygen and hydrogen concentration of IHTS sodium to 2.0 ppm and 0.2 ppm, respectively.
50 The system, working in conjunction with the primary cold traps, limits the tritium content of IHTS sodium to levels consistent with plant radiological release criteria.

The system also provides the capability to (1) fill the IHTS loops from the sodium dump tanks (these tanks are part of the Steam Generator System), (2) purify sodium in the dump tanks, independently of the IHTS loops, and (3) permit transfer of sodium from one dump tank to another.

9.3.5.2 Desig[illegible]cription

The Intermediate Sodium Processing System provides purification of the sodium in each of the three IHTS loops. The System does not provide for storage of the IHTS sodium. This capability is provided by the sodium dump tanks, which are part of the Steam Generator System. The Intermediate Sodium Processing System does provide the capability of transferring sodium into the loops from the dump tanks. The same piping network allows the filling of each dump tank with fresh sodium from tank cars or drums at the sodium receiving station. Sodium removal from the tanks into tank cars can be accomplished through the same fill lines.

The system includes the following components:

a. Intermediate Sodium Cold Trap Pumps

b. Intermediate Sodium Cold Traps

c. Interconnecting Piping and Valves

Refer to Figure 9.3-5 for the P&ID and Figures 1.2-8 and 1.2-22 for layout and arrangement.

Each of the three IHTS loops is provided with a separate purification system consisting of a pump, two cold traps, and the necessary valves and piping. A single trap per IHTS loop is sufficient to remove anticipated oxygen and hydrogen inleakage and to limit these impurities to a
maximum of 2 and 0.2 ppm, respectively. In addition, the intermediate
50 cold traps maintain the tritium level in the intermediate sodium at 0.012
μCiT/gm Na by effectively trapping about 98% of the tritium which enters
the system by diffusion through the IHX. Most of the remaining tritium,
46 50 0.016 Ci/day, diffuses through the steam generators and enters the water
system. Cold trap flow is ~60 gpm at normal system operating temperatures. The Intermediate Sodium Processing System is also connected to the sodium dump tanks such that the sodium in the dump tank may be processed by the system.

The intermediate sodium cold trap pumps are used to pump the
sodium from the dump tanks into the loops with a small, ~22 psig, cover
46 gas pressure being maintained on the dump tanks. Sodium can also be
transferred from one dump tank to another by gas pressure.

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Amend. 50
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The use rate of argon by these services is variable and is dependent on operator options. Under start-up conditions, the flow will be maximum, and a minimum supply capability of 95,000 scfd of argon is to be provided.

Argon is to be used for all services involving sodium-wetted components, such as fuel handling, sampling, and maintenance services. This gas also is ultimately exhausted through CAPS to the atmosphere.

Argon is also to be supplied for purging and inerting IHTS components and for sodium-water reaction control purposes.

9.5.1.2 Design Description

The argon distribution subsystem is composed of liquid argon Dewars with vaporizers, gaseous argon bottles, piping, valves, vapor traps, filters, vessels, relief systems, freeze vents, and oil traps as necessary to distribute the argon to meet the requirements described in Section 9.5.1.1.

9.5.1.2.1 Recycle Argon Distribution

Argon from the primary recycle cover gas storage vessels in the RCB is reduced in pressure to supply cover gas to the reactor vessel, primary sodium overflow vessel, and primary pumps cover gas spaces, which are all interconnected by a pressure equalization line. This cover gas system is maintained at a pressure of 6 in. w.g. by a feed and bleed control system.

There is a continuous transfer of argon cover gas from the reactor and the primary pumps via the equalization line to the primary sodium overflow vessel and then through a 5-scfm vapor trap that removes sodium vapor. This vapor trap consists of a vapor condenser and two parallel aerosol filters (one redundant). The gas flows back to RAPS for processing before recycling. A 1-scfm sample of cover gas is taken from the equalization line and is passed through a 1-scfm sodium vapor
50 trap to the Failed Fuel Monitoring System. This gas and the cover-gas bleed from the primary pumps are also returned to RAPS.

9.5.1.2.2 Fresh Argon Supply at RSB

Argon for services in the Reactor Service Building (RSB), the Reactor Containment Building (RCB), and the Intermediate Bay (IB) is stored as liquid in two Dewars, located on the RSB pad. These Dewars have a capacity of 1500 gal. each and are equipped with fill and vent lines. Normally, only one of the Dewars is in operation. When it is nearly empty, a low-liquid-level instrumentation signal operates automatic controls that shutoff that Dewar and open the other Dewar to the supply header. A control override allows drawing on both Dewars simul-
48 taneously.

Two ambient-air vaporizers on each Dewar can evaporate the liquid argon at a nominal maximum gas flow rate of 2000 scfh each, at 175 psig. With both Dewars on-line, therefore, approximately 8000 scfh of argon gas at 175 psig can be delivered.

The argon from the Dewars passes through a filter and is then divided into three main headers that supply argon to the RCB, RSB, and other ex-containment components.

9.5.1.2.3 Fresh Argon: RCB Distribution

The RCB header enters the building with isolation valves on each side of the penetration. This header supplies argon to the primary sodium storage vessel, with a feed and bleed system at a normal pressure of 1 psig, and to the recycle argon storage vessels.

The RCB header also supplies argon to the primary sodium plugging temperature indicator, the primary sodium sampling package, the floor/wall service stations, the reactor head inflatable seals, and the IVTM storage facility.

The RCB header also supplies argon to the primary sodium line freeze vents, which are furnished argon during startup, maintenance, and sodium drain and fill at a nominal pressure of 5 psig; the pressure can be increased, if needed, to 50 psig. This header also supplies cover gas argon for the NaK system and the make-up pump drain vessel.

9.5.1.2.4 Fresh Argon: RSB Ex-Containment Distribution

The RSB ex-containment header supplies make-up argon to the ex-containment primary sodium storage vessels in the Intermediate Bay. The normal pressure in the storage vessels is 1 psig, but this can be increased to 50 psig during tank drain. These vessels can be vented either through a vapor trap and a pressure control valve to the Cell Atmosphere Processing System (CAPS) or to a vacuum station and then to the CAPS.

9.5.1.2.5 Fresh Argon: RSB Distribution

The RSB header supplies argon at the required pressures to the gas chromatograph, the fission gas monitor module, and the gas sampling trap. A branch line provides argon purge to the RAPS cold box.

The RSB header supplies argon through regulators to the Auxiliary Liquid Metal System EVS Na and NaK components and to the Impurity Monitoring and Analysis System EVS sodium sampling package. The sodium lines have freeze vents that are furnished with argon during startup, maintenance, and sodium drain and fill operations at a nominal pressure of 5 psig. This pressure can be increased to 50 psig.

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9.5.2.2.1 Nitrogen Supply at RSB

The RSB and RCB nitrogen supply is stored as liquid nitrogen in two Dewars, each with 6000 gal. capacity, on the RSB pad. An ambient air vaporizer on each Dewar can evaporate the liquid nitrogen at a nominal flow rate of 15,000 scfh. Normal nitrogen usage is supplied from one Dewar, with a level sensor automatically switching tanks upon depletion of a pre-set level. A control override, however, allows the option of simultaneously supplying nitrogen from both tanks so that doubling the flow rate to meet peak demands is possible.

9.5.2.2.2 Nitrogen: RCB Distribution

The header feeding the RCB contains one isolation valve on each side of the containment penetration, providing automatic shutoff capability on either side in the event of nitrogen pressure loss. The header inside containment branches off into (1) a low pressure header feeding all of the normally inerted cells and pipeways within containment, (2) a high pressure line for actuation of valves in cells that are normally inerted, (3) a line to the CRDM assembly recirculation cooling system, and (4) a line to provide sparging gas to the sodium component cleaning operation.

Cells and pipeways containing sodium components in the RCB are normally inerted with nitrogen atmosphere, as is the CRDM cooling system. Each inerted cell or group of cells has inlet and outlet control valves that maintain preset cell pressures, in addition to having automatic cell purging for maintaining required oxygen or water-vapor levels. Purge flow is automatically activated by a cell atmosphere sampling and analysis unit that periodically monitors the O_2 and H_2O levels in each cell atmosphere. Radioactivity is also monitored but does not activate purging.

The inerting system for RCB and RSB cells (except FHC) is designed for normally controlling the oxygen concentration within the cells to a maximum of 2 vol %. The design base for the cell gas inerting system is a net inward leakage of air of 1% of the cell volume per day. When the cell is inerted to 2% oxygen, which amounts to 10% of the oxygen content of the inleaking air, the water vapor content in the cell will also be 10% of that in the in-leaking air. The Heating and Ventilating System normally controls the humidity of the air in the building to 40% R.H. at 75°F (water partial pressure, 0.12 atm). Because the cells are to be steel-lined, dehydration of the concrete will not contribute directly to the water content of the cell gas, so that the normal partial pressure of water vapor is 0.0012 atm, or 1200 vppm water vapor.

48 During initial warm-up and prior to sodium loading, should the water vapor content of the cell atmosphere (which can be air) exceed the normal maximum value, this water will be removed first by cell purging with air, and then, as the Recirculating Gas Cooling System (RGCS) goes into operation, by condensation on the cooling coils. At steady-state, this unit will limit the water vapor content of the cell atmosphere to

about 10,000 vppm. Reduction from this value to the 1000 vppm limit will be done by purging with nitrogen.

Nitrogen for service maintenance operations is available at service stations located within the RCB.

9.5.2.2.3 Nitrogen: RCB Auxiliary Supply

An auxiliary supply of nitrogen gas is stored in high-pressure standard cylinders located within a cell in the tornado-hardened RCB. This nitrogen is used to ensure the uninterrupted operability of certain essential valves in the event of pressure loss in the nitrogen supply header. A control valve automatically restores pressure in the valve actuation circuit when an abnormal decrease in operating pressure is sensed. A check valve then isolates the valve circuit from the main supply line in order to preclude auxiliary supply blowdown to the remainder of the failed supply circuit.

9.5.2.2.4 Nitrogen: RSB Distribution

The 150 psig RSB header branches off into several lower pressure headers that service the needs of other systems as well as those of the RAPS and CAPS subsystems within the RSB.

RSB cells and pipeways containing sodium components are inerted with nitrogen during normal operation. The cell pressures are maintained by a feed and bleed arrangement and a purge function controls impurity levels. (See Section 9.5.2.2.2.)

The RAPS and CAPS cold boxes are inerted with nitrogen at a continuous low flow rate during operation. These flows are vented directly to the respective cells so that the cell atmospheres become nitrogen-rich. The RAPS cell pressure is maintained by a back-pressure regulator that bleeds the cell atmosphere to CAPS. The CAPS cold box cell atmosphere is vented to the Heating, Ventilating and Air Conditioning
50 System.

The nitrogen requirement to the cold boxes serves two purposes: to inert the cold boxes so that water condensation within the cryogenically-cooled structure is prevented and to provide gas for valve operation. The cold boxes would not be effected adversely by high purge flows nor would there be an impact on the CAPS decontamination process. The only consequence of such flows would be increased nitrogen utilization.

Nitrogen for service maintenance operations is available at service stations located within the RSB.

Nitrogen gas is provided as a cover gas for the Dowtherm tanks used in the chilled water system.

9.5.2.2.5 Nitrogen: RSB Auxiliary Supply

48 An auxiliary supply of nitrogen gas, stored in high-pressure standard cylinders located within a cell in the tornado-hardened RSB,

TABLE 9.6-4 (cont'd).

EQUIPMENT TITLE	LOCATION BLDG.	LOCATION ELEV.	SAFETY CLASS	SEISMIC CLASS	PRIMARY PARAMETER (CFM)
RCB Inerted Cells Booster Fan	RCB	752'-0"			200
RCB Inerted Cells Booster Fan	RCB	733'-0"			200
50 RCB Portable Filter Fan	RCB	-			1,000
RCB Supply Isolation Valve	RCB	816'-0"	SC-2	I	14,000
RCB Exhaust Isolation Valve	RCB	816'-0"	SC-2	I	14,000
RCB Supply Isolation Valve	ANNULUS	816'-0"	SC-2	I	14,000
RCB Exhaust Isolation Valve	ANNULUS	816'-0"	SC-2	I	14,000
RCB Annulus Air Cooling Fan	RSB	816'-0"	SC-3	I	133,000
RCB Annulus Air Cooling Fan	RSB	816'-0"	SC-3	I	133,000
RCB Annulus Air Cooling Fan	RSB	828'-9"	SC-3	I	133,000
RCB Annulus Air Cooling Fan	RSB	828'-9"	SC-3	I	133,000
RCB Annulus Air Cooling Fan	RSB	841'-6"	SC-3	I	133,000
RCB Annulus Air Cooling Fan	RSB	841'-6"	SC-3	I	133,000
RSB Cleanup Filter Unit Fan	RSB	755'-0"	SC-3	I	18,000
RSB Cleanup Filter Unit Fan	RSB	816'-0"	SC-3	I	18,000
RSB Cleanup Filter Unit	RSB	755'-0"	SC-3	I	18,000
RSB Cleanup Filter Unit	RSB	816'-0"	SC-3	I	18,000
RSB Annulus Pressure Maintenance Fan 49	RSB	846'-0"	SC-3	I	3,000

TABLE 9.6-4 (cont'd).

EQUIPMENT TITLE	LOCATION BLDG.	LOCATION ELEV.	SAFETY CLASS	SEISMIC CLASS	PRIMARY PARAMETER (CFM)
RCB Annulus Pressure Maintenance Fan	RSB	864'-6"	SC-3	I	3,000
RCB Annulus Filter Unit	RSB	846'-0"	SC-3	I	14,000
RCB Annulus Filter Unit	RSB	864'-6"	SC-3	I	14,000
RCB Annulus Filter Fan	RSB	846'-0"	SC-3	I	11,000
RCB Annulus Filter Fan	RSB	864'-6"	SC-3	I	11,000
RCB Cleanup Scrubber Fan	RSB	785'-0"	SC-3	I	18,000
RCB Cleanup Scrubber Fan	RSB	816'-0"	SC-3	I	18,000
RCB Cont. Cleanup Scrubber	RSB	785'-6"	SC-3	I	18,000
RCB Cont. Cleanup Scrubber	RSB	785'-6"	SC-3	I	18,000
RCB Cont. Cleanup Air Washer	RSB	785'-6"	SC-3	I	18,000
RCB Cont. Cleanup Air Washer	RSB	785'-6"	SC-3	I	18,000
RCB Purge Supply Isolation Valve	RCB	842'-0"	SC-2	I	18,000
RCB Purge Supply Isolation Valve	RCB	816'-0"	SC-2	I	18,000
RCB Purge Supply Isolation Valve	ANNULUS	842'-0"	SC-2	I	18,000
RCB Purge Supply Isolation Valve	ANNULUS	816'-0"	SC-2	I	18,000
RCB Containment Cleanup Exhaust Isolation Valve	RCB	TBD	SC-2	I	18,000
RCB Containment Cleanup Exhaust Isolation Valve	RCB	TBD	SC-2	I	18,000

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9.8 IMPURITY MONITORING AND ANALYSIS SYSTEM

The Impurity Monitoring and Analysis System consists of the following subsystems:

1) PHTS Sodium Characterization

2) Primary Cover Gas Sampling and Monitoring

3) EVST Sodium Characterization

4) IHTS Sodium Characterization

5) EVST, FHC, and IHTS Cover Gas Sampling

45 6) Analytical Services Laboratory

All components and piping in contact with sodium will be constructed of Type 304 austenitic stainless steel. This includes sampling inserts in the multi-purpose sampler in the sodium sampling package. Argon cover gas sampling and monitoring piping and components will be constructed of Type 304 stainless steel. 15

9.8.1 Design Basis

9.8.1.1 Sodium Sampling and Monitoring

45 The sodium characterization (sampling and monitoring) subsystems are to be provided a continuous flow of sodium from the systems being monitored. The subsystems will be designed for automatic, on-line impurity monitoring of this sodium, and for sample collection of aliquots of sodium (or equilibration devices or particulate samples) for laboratory analysis. These sample streams are monitored, and samples analyzed, to verify that the circulating sodium meets the allowable oxygen and hydrogen levels. Analytical results 45 from these subsystems provide information for the operation of the sodium purification units of the Auxiliary Liquid Metal System.

9.8.1.2 Cover Gas Sampling and Monitoring

50 Primary cover gas samples are provided by the Failed Fuel Monitoring System. These samples will be processed in the analytical services laboratory to determine impurities. Sample connections are to be provided for sampling EVST, FHC, and IHTS cover gas with sample bottles whose contents are processed in the analytical services laboratory to determine impurities. Analytical results of these samples will be used by the Inert Gas Receiving and Processing System to establish and maintain the various argon gas purity requirements within specified limits.

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45 ### 9.8.1.3 Analytical Services Laboratory

Space is to be provided in the hot laboratory in the plant services building for equipment to perform out-of-loop analyses of sodium (and equalibration devices and particulate samples) and cover gas samples.

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9.8.2 Design Description

9.8.2.1 PHTS Sodium Characterization Subsystem

The PHTS Sodium Characterization Subsystem, shown in Figure 9.8-1 is provided a continuous sample of the primary sodium from the overflow
45 vessel or the storage vessel tanks. The piping is designed to accept sodium from the discharge side of the make-up pumps.

45 The flow rate through the subsystem sampling loop is controlled at 10 gpm, and returned either to the primary sodium overflow vessel or to the primary sodium storage vessel tanks, whichever is the source of the sample.

A portion of the primary sodium sample stream can be diverted through
45 the primary plugging temperature indicator (PPTI) by the proper operation of valves. A plugging temperature indicator (PTI) is a device to determine the saturation temperature of impurities dissolved in the sodium. The results,
45 however, are not specific for a given impurity, but do give an indication of impurity levels which are consistent with the saturation temperature.

In parallel with the Primary PTI is the primary sodium sampling
45 package (PSSP), which provides samples for several types of laboratory analyses. A portion of the primary sodium sample stream can be diverted through the PSSP by the proper operation of valves. A continuous flow of sodium at a controlled temperature is then provided to either or both of the two sam-
45 pling devices [multi-purpose samplers (MPS)] in the primary SSP. By the use of inserts, the MPS can:

1) Provide a sodium sample for laboratory analyses of sodium impurities

2) Be used to expose foils and/or wires of selected materials which will equilibrate with oxygen, hydrogen, or carbon dissolved in the sodium and which can be analyzed in the laboratory

3) Filter known quantities of sodium, to provide measure of particulate impurities in the sodium.

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45|36| Operation of the Primary SSP is manual, and requires remote operation. Master-slave manipulators are provided. Typical laboratory information obtained from these samples are:

1) Total coolant impurity levels

2) Oxygen, hydrogen, and carbon activity

3) Tritium level

4) Fission product levels

5) Corrosion product levels

6) Particulate impurity levels

45|36|27| 7) Other impurities as considered necessary

9.8.2.2 Primary Cover Gas Sampling and Monitoring Subsystem

Primary cover gas samples in shielded, gas sample bottles are provided by the Fuel Failure Monitoring System. These samples will be processed in the gas chromatograph in the analytical services laboratory to determine impurities (helium, hydrogen, oxygen, nitrogen, methane, and carbon monoxide). Samples are also taken for determination of impurities not detectable by the gas chromatograph such as tritium.

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9.8.2.3 EVST Sodium Characterization Subsystem

The EVST Sodium Characterization subsystem is shown in Figure 9.8-2.
45 As EVST sodium is circulated by either EVST sodium pump in the Auxiliary
Liquid Metal System, a portion of the sodium is diverted to the EVST
45 sodium characterization subsystem components.

The ex-vessel plugging temperature indicator (EVPTI) is identical
to the primary PTI (Section 9.8.2.1) in design and function. In parallel
45 with the EVPTI is the ex-vessel sodium sampling package (EVSSP), identical
in function to the primary SSP; only one multi-purpose sampler is provided.
Since EVST sodium will become contaminated with radioactive sodium and/or
fission products, provision is made for remote operation of the EVPTI
and EVSSP. Master-slave manipulators are provided for the remote manual
45 operational requirements of the EVSSP.

9.8.2.4 IHTS Sodium Characterization Subsystem

45 As shown in Figure 9.8-3, the three loops in the Intermediate Heat
Transport System (IHTS) are sampled and monitored by a three intermediate
sodium characterization package (ISCP), each containing a plugging tempe-
rature indicator module and a multi-purpose sampler. The inlet to each
characterization package is connected to the outlet of the cold trap pump
in each IHTS loop and the return line is connected to the return line of
the IHTS loop. Sampling and monitoring can be performed only while the
cold trap pump is functioning. Since the sodium in the three IHTS loops
49 is not radioactive, this subsystem is not designed for remote operations.

9.8.2.5 EVST, FHC, and IHTS Cover Gas Sampling

45 Provisions are made in the EVST and FHC cover gas systems and in
the IHTS pump-expansion tank pressure equalization lines, to obtain cover
gas grab samples, using evacuated gas sample bottles whose contents will
be analyzed in the laboratory. Since fission gases can be present in the
45 EVST and FHC cover gas samples, these gas bottles will be shielded for
operator protection. Shielded sample bottles will not be required for
IHTS cover gas samples.

45 ### 9.8.2.6 Analytical Services Laboratory

Space and equipment will be provided in the h[illegible] laboratory in
the Plant Service Building for out of loop analyses of sodium and cover
45 gas samplers. Due to the radioactive nature of many of these samples, the

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equipment will be designed for shielded and/or remote operation. Sample preparation equipment will include components such as glove goxes, sodium distillation unit and gas transfer systems. Analytical equipment will include components such as gas analyzers and a gas chromatograph. Controlled waste
45 disposal will be provided for radioactive sample residues.

9.8.3 Design Evaluation

The Impurity Monitoring and Analysis System components are designed to accepted industrial and nuclear standards to insure structural integrity and operational reliability. The components, applicable design code and class, plus their seismic category are listed in Table 9.8-1.

45 9.8.3.1 PHTS Sodium Characterization Subsystem

If the Primary PTI is being repaired or has malfunctioned, monitoring of the sodium impurity level can be effected by using the Primary SSP, and performing impurity analyses of sodium samples. This subsystem performs no control functions, but analytical results are used to indicate out-of-range
45 conditions, and are also used by the Auxiliary Liquid Metal System to determine primary sodium cold trap operational requirements.

This subsystem is not required to function during an emergency, nor is it required for the safe shutdown of the reactor. Isolation valves are provided to separate failed components (Primary PTI or Primary SSP) from the sodium sampling loop piping. Sodium leak detectors are provided on bellows seal valves. The cells containing this equipment are provided with sodium
36 45 aerosol leak detectors to signal the release of sodium to the inerted cells.

The components and piping in this subsystem are contained in lined, inerted cells in the Reactor Containment Building, which can be vented to CAPS of the Inert Gas Receiving and Processing System to prevent the escape of radioactive gases. Sufficient shielding is provided to prevent radiation overexposure of operating personnel under normal and anticipated fauited conditions.

9.8.3.2 Primary Cover Gas Sampling and Monitoring Subsystem

The Failed Fuel Monitoring System gas sampling connections are used to obtain a bulk sample of cover gas in a shielded gas sample bottle which is analyzed in the analytical services laboratory. Sufficient shielding is provided to prevent radiation overexposure of operating personnel under
50 normal and anticipated faulted conditions.

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Since there is little delay time and no dilution of the cover gas sample, a representative sample is obtained. This subsystem performs no control functions, but analyzed results are used to indicate out-of-range conditions, and are also used by the Inert Gas Receiving and Processing System to adjust clean argon purge. This subsystem is not required to function during an emergency, nor is it required for the safe shutdown of the
50 reactor.

9.8.3.3 EVST Sodium Characterization Subsystem

This subsystem is not required for the operation of the EVST during an emergency. Isolation valves are provided to separate failed components (Ex-Vessel PTI or Ex-Vessel SSP) from the sodium sampling loop piping. Sodium leak detectors are provided on bellows seal valves. The cells containing this equipment are supplied with sodium aerosol leak detectors to signal the release of sodium to the inerted cell.

If the Ex-Vessel PTI is being repaired or has malfunctioned, monitoring of the sodium impurity level can be accomplished by using the Ex-Vessel SSP, and performing impurity analyses of sodium samples. This subsystem performs no control functions, but analytical results are used to indicate out-of-range conditions, and are also used by the Auxiliary Liquid Metal System to determine EVS sodium cold trap operational requirements.

The components and piping in this subsystem are contained in lined, inerted cells in the Reactor Service Building, which can be vented to CAPS of the Inert Gas Receiving and Processing System to prevent the escape of radioactive gases. Sufficient shielding is provided to prevent radiation overexposure of operating personnel under normal and anticipated faulted conditions.

45

47

9.8.3.4 IHTS Sodium Characterization Subsystem

Continuous availability of this system is not required for the operation of the IHTS or the leak detection instrumentation. The sodium in this component is not radioactive, so that hands-on manual operation is planned.

This subsystem is not required for the operation of the IHTS during an emergency. Isolation valves are provided in case of piping or component failure. The intermediate sodium characterization packages serving IHTS loops 1, 2, and 3 are installed in cells equipped with catch pans with
47 49 Class 1E leak detection capability.

If the PTI is being repaired or has malfunctioned, monitoring of the sodium impurity level can be effected by using the MPS, and performing impurity analyses of sodium samples. Sodium sampling and analysis is also required to provide calibration services for steam generator leak detection instrumentation (Section 7.5.5.). This subsystem performs no control functions, by analytical results are used to indicate out-of-range conditions, and are also used by the Auxiliary Liquid Metal System to determine intermediate sodium cold trap operational requirements.

9.8.3.5 EVST, FHC, and IHTS Cover Gas Sampling Subsystem

45 The shielded gas sample bottles for EVST and FHC cover gas sampling are interchangeable with those used to collect primary cover gas. The shielding will be sufficient to protect personnel during sampling, and transfer of the sample to the analytical laboratory. The IHTS cover gas sampling bottles will not be shielded, as this gas is not radioactive.

9.8.3.6 Analytical Services Laboratory

The sodium and cover gas analytical equipment and supplies will be
45 appropriate for analyses planned. Radioactive sodium wastes will be transferred to the Radioactive Waste System for controlled disposal. Radioactive gas wastes will be disposed of under controlled conditions.

45 9.8.4 Tests and Inspection

9.8.4.1 Sodium Sampling and Monitoring

45 The PHTS and EVST sodium characterization components are located in inerted cells, so visual inspections will be performed while the system is down, such as during fuel loading, for the primary sodium components or while the impurity monitoring components are isolated from the rest of the system. Sodium leak detectors will be employed. The IHTS components are located in air atmosphere cells and present no restrictions to direct inspection methods.

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9.8.5 Instrumentation Requirements

Instrumentation for the Impurity Monitoring and Analysis System provides measurement and control of process variables as required for system operation, performance evaluation, and annunciation of off-normal conditions. Components will be shut down upon deviation of process variables beyond the normal operating limits.

9.8.5.1 Sodium Sampling and Monitoring

Sodium sampling loop heating and temperature controls are provided by Piping and Equipment Electrical Heating Control System. Sodium temperatures within the PTI's and SSP's are measured by thermocouples which also provide input signals to temperature controllers for regulation of process temperatures. Thermocouples are also provided on the magnet of each permanent magnet (PM) flowmeter to allow temperature correction of the flowmeter
49 output. Separate control panels are provided for each PTI, SSP, and ISCP. Specific local annunciators are provided on each control panel. The annunciator signals from each control panel are grouped and transmitted to the main control room to indicate abnormal conditions.

All sodium valves are provided with leak detectors, heaters and thermocouples, and limit switches.

PHTS sodium sampling inlet flow is measured by a PM flowmeter and recorded to assist the operator in flowrate adjustments and to monitor system operation. Two remotely-actuated valves are provided in both the supply and return lines of the sampling loop. The control switches for these valves are interlocked with a high/low flow alarm signal from the inlet flowmeter such that the valves will automatically close upon abnormal flow. The valves are designed to fail closed upon loss of electrical or pneumatic power.

A remotely-actuated valve is provided in both the supply and return lines of the EVST sodium sampling loop. The control switches for these valves are interlocked with high/low flow alarm signals derived from PTI and SSP inlet PM flowmeters such that the valves will automatically close upon abnormal flow conditions. The valves are designed to fail closed on loss of electrical or pneumatic power.

9.8.5.1.1 Plugging Temperature Indicators

Flow control valves are installed on the PTI inlet and bypass lines. Panel mounted, hand-indicating controllers provide remote control of these valves based on PM flowmeter signals.

Remote manual and automatic control of sodium temperatures are provided in the PTI for determination of the plugging (saturation) temperature. Automatic cycling control is accomplished by using high- and low-flow signals for control of the cooling gas blower. At the high-flow setpoint,
45 the blower is energized to cool the sodium until the flow decreases to the

low setpoint, where the blower is deenergized. With the blower off, the sodium temperature and flow increase and the cycle is repeated. Sodium temperature at the PTI orifice and PTI outlet flow are recorded on the same recorder to allow fast direct comparison and interpretation of the data. Manual PTI control is accomplished by manually switching the blower on and off as the flow variations are observed on the recorder. Alarms are provided to indicate that the high- or low-flow and high orifice temperature setpoints have been exceeded.

9.8.5.1.2 Sodium Sampling Packages

Sodium flow rates through the MPS and SSP bypass are controlled manually. PM flowmeter signals are recorded and also indicated at the manipulator station.

MPS furnace temperatures are controlled in three zones: (1) one three-mode controller (proportional plus rate plus reset) is used to control all heaters in the upper furnace, which is the main heat zone for the sampler cup, (2) two two-mode controllers (proportional plus reset) are used to control the two axial zones in the lower furnace. The remaining heaters on the MPS assembly on the argon-vacuum line, and inlet and outlet sodium lines and valves, are controlled by automatic on-off controllers. A high/low temperature alarm is provided for the upper furnace.

TABLE 9.8-1

IMPURITY MONITORING AND ANALYSIS SYSTEM

STRUCTURAL DESIGN CRITERIA

System	Component	Design Code		Seismic Category
		Code	Class	
Primary Na Characterization	Piping to/from Auxiliary Liquid Metal System and isolation valves	ASME Section III	1	I
	Sampling loop piping and valves	ASME Section III	3	I
	Primary plugging temperature indicator	ASME Section III	3	I
	Primary sodium sampling package	ASME Section III	3	I
	Electric Hoist	Coml	-	III
	Master Slave Manipulator	Coml	-	II
	Radiation Shielding	Coml	-	I

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TABLE 9.8-1 (cont'd.)

IMPURITY MONITORING AND ANALYSIS SYSTEM

STRUCTURAL DESIGN CRITERIA

	System	Component	Design Code: Code	Design Code: Class	Seismic Category
50 45		Electric Hoist	Coml.	-	III
	EVST Na Characterization	Piping to/from Auxiliary Liquid Metal Sys.	ASME Section III	3	I
47 45		Sampling loop piping and valves	ASMF Sec.III		
		Ex-vessel plugging temperature indicator	ASME Section III	3	I
49 48 36		Ex-vessel sodium sampling package	ASME Section III	3	I
		Electric hoist	Coml.	-	III
		Master-slave manipulator	Coml.	-	II
		Radiation shielding	Coml.	-	II
36	EVST Ar Sampling	Ar sampling piping	ASME Section III	3	I

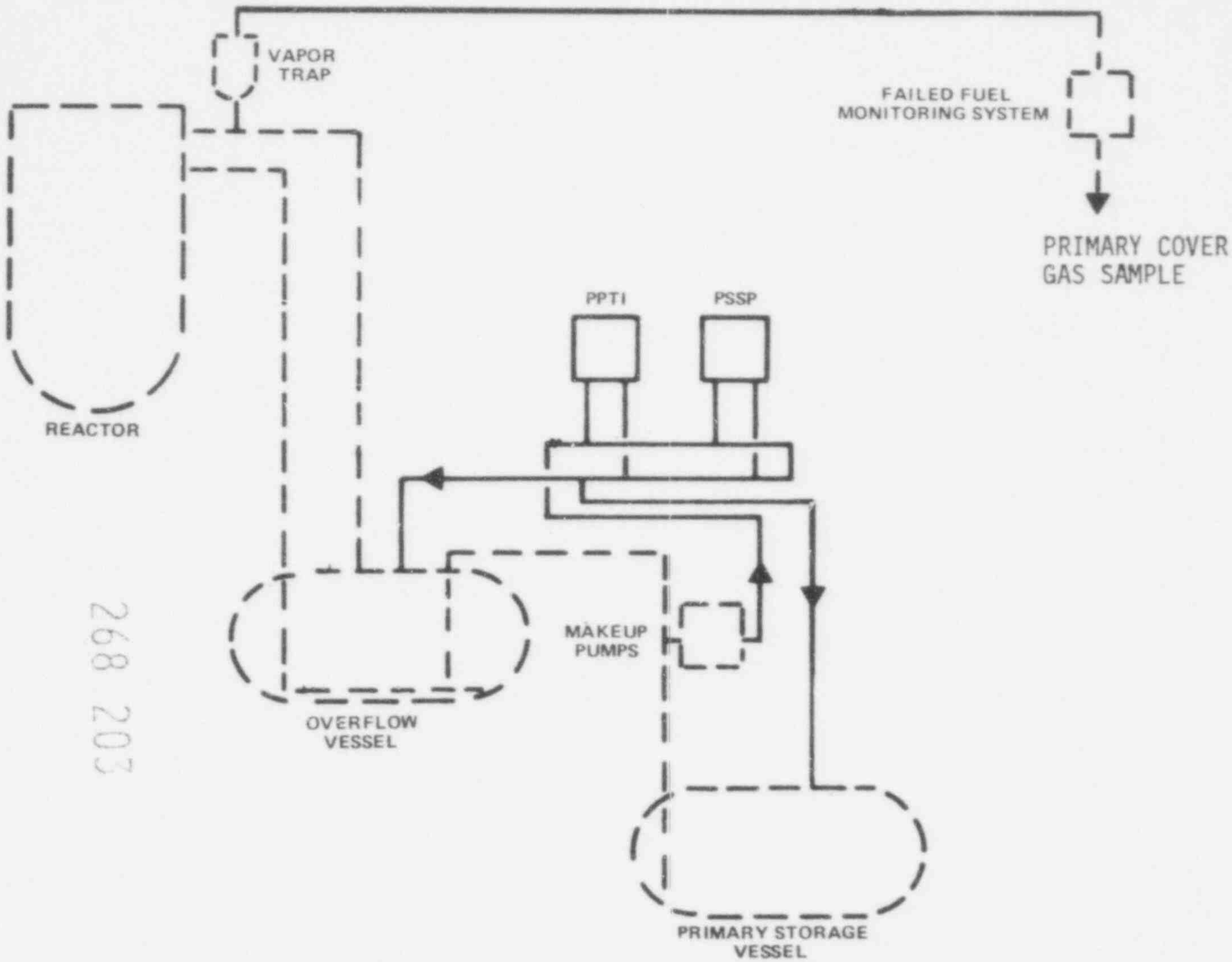

Figure 9.8-1. PHTS Sodium Characterization and Primary Cover Gas Sampling and Monitoring Subsystems

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6669-2

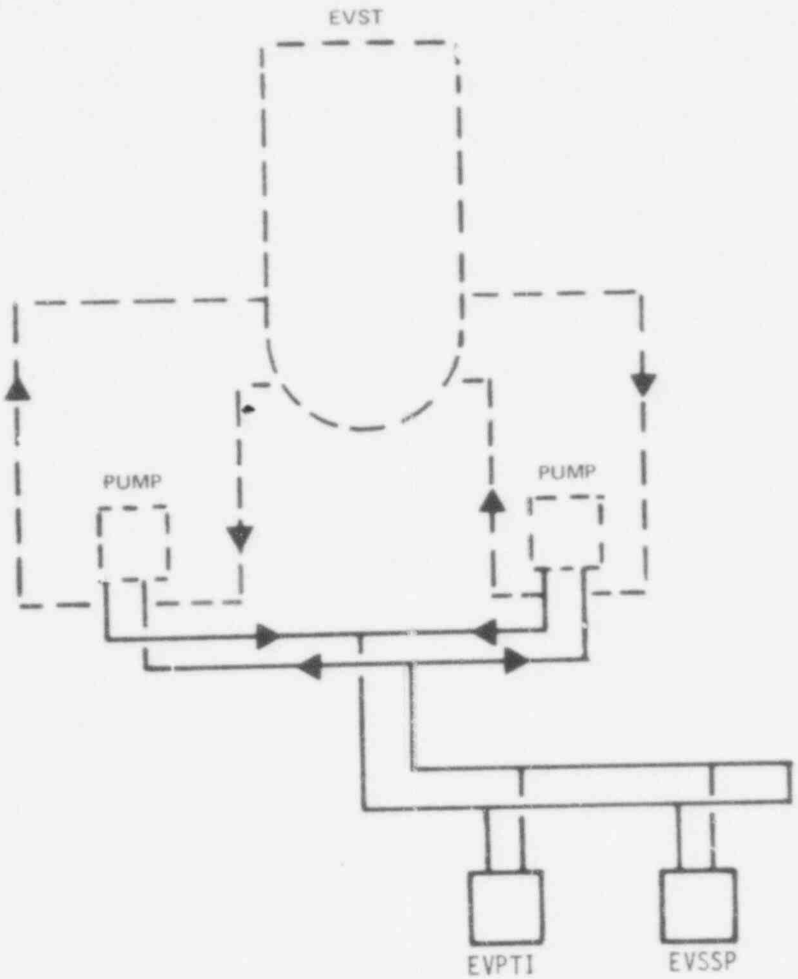

Figure 9.8-2 EVST Sodium Characterization Subsystem

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9.9 SERVIC WATER SYSTEMS

9.9.1 Normal Plant Service Water System

9.9.1.1 Design Basis

The Normal Plant Service Water System is a non-safety related system designed to provide cooling water for the Normal Chilled Water System chiller condensers, the Secondary Service Closed Cooling Water
33 System and other equipment listed in Table 9.9-1 during normal plant operation and planned outages. The system will be designed according to the ASME Section VIII/ANSI B31.1 requirements.

9.9.1.2 System Description

The Normal Plant Service Water System is shown in Figure 9.9-3. The system consists of two (approximately 26,600 GPM) 100 percent
43 15 capacity electric motor driven deep well turbine pumps and the required piping, valves and instrumentation. The Normal Plant Service Water is pumped from the basin of the Circulating Water System cooling tower to the equipment to be cooled, and is returned to the cooling tower
43 return header. The pumps are located in the Circulating Water Pumphouse. Normally, one pump is operating with the second pump in an
33 auto-standby mode.

The components served by the Normal Plant Service Water System are listed in Table 9.9-1. Design data for the major system components are listed in Table 9.9-2.

15

9.9.1.3 Safety Evaluation

15

The Normal Plant Service Water System is a nonseismic, nonsafety class system. Cooling water for the Emergency Chilled Water System chiller condensers, and standby Diesel Generators, for the safe shutdown and the maintenance of the safe shutdown condition, is provided
15 by the Emergency Plant Service Water System as described in Section 9.9.2.

33

9.9.1.4 Tests and Inspections

15

The Normal Plant Service Water pumps are tested at the manufacturer's facility and retested in the system prior to continuous plant operation. The operation of the pumps will be rotated to equalize wear.

9.9.1.5 Instrumentation Application

15

Indication of the Normal Plant Service Water header pressure is provided in the Control Room. Normal Plant Service Water discharge

header pressure 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.

15 33

9.9.2 Emergency Plant Service Water System

9.9.2.1 Design Basis

The Emergency Plant Service Water System is designed to provide
43 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
43 33 15 chiller condensers and the Standby Diesel Generators with cooling water. 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
43 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 onsite power supplies.
15 33 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. The Emergency
50 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. Each cooling loop includes one circulating pump, one make-up pump, one emergency cooling tower and associated piping, valves, instrumentation and controls. Figure 9.9-4 shows the various equipments and represents the system
50 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 com-
43 33 ponents 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
50 Fans will automatically start and provide cooling water at 90°F maximum

to the Emergency Chilled Water Chiller Condensers and the Standby Diesel Generators. The EPSW pumps take suction from the Emergency Cooling Tower operating basins which are located directly below the pumphouses and adjacent to the common storage basin. During system operation the EPSW makeup water 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 basins 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 chilled water system and electric power system loads, 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 a flow totalizer 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
50 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. 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. Should the Control Room become uninhabitable for any reason, such as smoke or fire, redundant controls have been provided that will allow full operation of the
50 system from a control panel in the Diesel Generator Building.

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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.

15 9.9.2.4 Tests and Inspections

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
50 on the supply and return headers in the pumphouses and the DGB permit inservice inspection of the buried piping by hydrostatic testing.

9.9.2.5 Instrumentation Application

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

- pump discharge pressure (local/remote)
- 50 supply temperature (local/remote)
- 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)
- 50 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 Secondary Service Closed Cooling Water System

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

15 9.9.3.1 Design Basis

The Secondary Service Closed Cooling Water (SSCCW) System is desig. to provide adequate cooling water supply for the power generation equipment during normal plant operation.

The SSCCW System is designed in accordance with ANSI B31.1 and is not safety related.

15 9.9.3.2 System Description

The SSCCW System is shown in Figure 9.9-5 and consists of a single closed loop with two 100 percent capacity centrifugal pumps in parallel. The system utilizes two 100 percent capacity SSCCW heat exchangers are cooled by the Normal Plant Service Water System. The cooling water for the SSCCW discharges into a common discharge header where the SSCCW pumps take suction. The SSCCW System provides cooling
15 water to the equipment listed in Table 9.9-5.

A surge tank, located above the SSCCW pump suction, accommodates system volume changes, and maintains static head on the pumps in the SSCCW System. Makeup water to the SSCCW System is supplied by a connection from the condensate pumps to the SSCCW pumps common suction header. Tank level is maintaned automatically by means of level transmitters and controllers mounted locally. A signal from these transmitters opens the level control valve on the condensate line to maintain the surge tank at the desired level. The surge tank is readily accessible during operation for manual level adjustment if desired.

A bypass line is provided around the SSCCW heat exchangers and includes a temperature controller installed at the discharge manifold downstream of the heat exchangers to regulate the bypass flow thereby providing a tempering effect to maintain a cnstant 95 degree F cooling water.

15 9.9.3.3 Safety Evaluation

The Secondary Service Closed Cooling Water (SSCCW) System is not a safety related system and is not required during an emergency shutdown of the plant.

15 9.9.3.4 Tests and Inspections

Pumps for the SSCCW System are tested prior to installation and again prior to plant operation. System subsections normally closed to flow are tested periodically to ensure their operability and integrity of the system.

15 9.9.3.5 Instrumentation Applications

The common discharge header of the SSCCW pumps is monitored for low pressure and alarmed in the Control Room. Pressure indicators are provided at each pump discharge and each heat exchanger outlet. Tempera-
43 ture indication is located on each suction line and on the common discharge manifold of the SSCCW pumps.

9.9.4 Emergency Cooling Towers and Emergency Cooling Tower Basin

9.9.4.1 Design Basis

The Emergency Cooling Towers (Figure 9.9-4) operate as part of the Emergency Plant Service Water System (Section 9.9.2) to provide cooling water for the Emergency Chilled Water System chiller-condensers, and for the Standby Diesel Generators. Uninterrupted cooling water supply is required for the above equipment. The failure of the Normal Plant Service Water System requires the operation of the Emergency Cooling Towers for the safe shutdown and the maintenance of the safe shutdown condition of the plant. The Emergency Cooling Towers do not operate under normal plant conditions except for routine testing.

The Emergency Cooling Towers and Emergency Cooling Tower Storage Basin are designed according to the applicable requirements of Regulatory Guide 1.27. The integral piping associated with the cooling towers is
33 designed according to ASME Section III, Class 3 requirements. The capacity of the Emergency Cooling Tower Basin is sufficient to permit the uninterrupted operations of the Emergency Plant Service Water System for that period of time (minimum of 30 days) needed to evaluate the situation, to take corrective action to mitigate the consequences of an accident, and to take any necessary measures to permit water replenishment.

50 The storage capacity of the Emergency Cooling Tower Storage Basin is based on the historical regional measurements, combining the worst recorded 30 day average period of maximum difference between dry bulb temperature and dew point temperature (ΔT) and the highest wind speeds recorded during the same 30 day period, such that the combination of ΔT and wind speed occurring simultaneously results in the maximum amount of evaporation and drift loss of water from the cooling tower.

The Emergency Cooling Towers are designed not to exceed the maximum permissible cooling water supply temperature, using the worst one day and worst 30 day periods of regional meteorological records when the heat transfer to the atmosphere is minimized and maximum cooling water supply temperature is induced. The worst one day period of the
43 record is assumed the first day of the worst 30 day period.

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50 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 of
components.

50 Electrical power for the Emergency Cooling Tower fans, pumps,
and control equipment is provided from the Class 1E AC power supply.
50 One fan is provided with electrical power from System Class 1E Division 1
and the other from System Class 1EB Division 2.

15 43 9.9.4.2 <u>Design Description</u>

The Emergency Cooling Tower Structure consists two 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, 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
60,000 gallons of water storage for the seismic 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 x 10^7 BTU/HR at the maximum Emergency Plant
50 Service Water Flow of approximately 3600 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 and applied biocide additives provide compensation for the increased tube fouling, resulting from the change in
the water chemistry. 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 from the nearby potable water systems.

The top elevation of the Emergency Cooling Tower Basin is
818 ft. which is 9 ft. above the probable maximum flood level. The
basin maximum water level is at 810 ft. elevation. 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
50 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
50 43 in a Seismic Category I, tornado missile protected structure. The water

intake and 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
50 has a design flow rate of 3600 GPM.

The Emergency Cooling Towers are of counter flow, induced mechanical 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 flow air to provide evaporation for the required heat removal. The air counter 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
50 concrete storage basin. The top of the cooling towers is approximately 44 ft. above the maximum water level.

50 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

The Emergency Cooling Tower structure consists of two 100
50 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
50 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
43 procedures discussed in Section 3.7.

268 212

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 for the worst day of record.

43

b. Water Usage Maximizing Conditions

The conditions for maximizing water usage for 30 days may be summarized as follows:

(1) Wet bulb and dry bulb temperatures for the worst 24 hours on record are assumed for the first 24 hours after design basis accident. For the following 29 days, the worst month of record is assumed.

(2) 90^{o}F initial basin water temperature.

50 (3) Maximum specified cooling tower drift loss of .01% maximum flow.

43 (4) All pumps and fans operating in the active trains.

(5) 60,000 gallons of water for fire protection use is not
50 considered available for cooling.

3. The maximum water usage based on the above assumptions will be calculated by a computer program that model time history of the heat loads and the cooling tower heat removal capability. Normal component leakage and losses due to a postulated pipe rupture will also be taken into account.

U. S. Department of Commerce weather data for Oak Ridge, Tennessee Township and area stations for the years January 1951 through December 1971 will be used in the analysis.

Evaporation rate from the Emergency Cooling Tower is calculated using the heat balance across the Emergency Cooling Tower.

50 9.9.4.4 Test and Inspection

The Emergency Cooling Tower fans will be tested prior to installation of the manufacturer's facilities. After construction of the Emergency Cooling Tower structure is completed, but prior to normal plant operation, the cooling towers will be tested for cooling performance, evaporation and drift rates according to the Standards of the Cooling Tower Institute.

50 The applicable Emergency Cooling Tower components will be
43 tested periodically in conjunction with the Emergency Plant Service
50 Water System according to the requirements of ASME, Section VI.

9.9.4.5 Instrumentation Application

The following instrumentation is provided at the Emergency Cooling Tower structure with signals transmitted to the Control Room:

a. Level transmitters, for storage basin level indication readout.

b. Temperature sensors (for each cooling tower) for discharge cooling water temperature readouts.

c. Temperature sensor (in the storage basin) for discharge cooling water temperature readout.

43 33 d. Air flow switches (at the cooling tower fan discharge) to indicate proper fan operation by status lights.

15 9.9.5 River Water Service

The River Water Service supplies Clinch River water as makeup to
43 the Main Cooling Tower, Emergency Cooling Tower structure and Plant Water Treatment Facility during normal operation. A basic flow diagram of the River Water System is provided in Figure 9.9-6.

15 9.9.5.1 Design Basis

The River Water Service (RWS) is designed to provide adequate river water to replace circulating water lost from the Main Cooling Tower during normal operation due to drift, evaporation and blowdown. The RWS
33 also supplies the Plant Water Treatment Facility to meet all process and potable water demands during normal operation. Design flow rate for the RWS is 9,000 gpm.

The RWS piping is designed and tested in accordance with ANSI B31.1 and is not safety related.

The River Water Intake design incorporates two submerged, perforated pipe intakes which are specifically designed to minimize their impact upon the aquatic life present and eliminate interference with commercial river traffic in the Clinch River.

15 9.9.5.2 System Description

The RWS consists of a pump house located at the shore of the Clinch River, two perforated pipe inlets, two River Water Service pumps
33 designed for 9,000 gpm each and the associated piping and valves necessary to provide river water to meet the plant demands.

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Two backwash lines are provided to allow removal of debris collected on the perforated pipe inlets.

A recirculation line is provided for the river water service pumps to preclude low flow problems associated with the pumps.

15 9.9.5.3 Safety Evaluation

The RWS is not a safety related system and is not required during an emergency shutdown of the plant.

15 9.9.5.4 Tests and Inspections

River Water Service Pumps are tested prior to installation and again prior to plant operation. The RWS is normally in service.

15 9.9.5.5 Instrumentation Application

Flow, pressure, and alarms are provided as required on the RWS. Pump discharge flow will be regulated by level control of the main cooling tower basin.

268 216

TABLE 9.9-1

COMPONENTS SERVED BY NORMAL PLANT SERVICE WATER SYSTEM

COMPONENT	NUMBER OF COMPONENTS	COMPONENT LOCATION BUILDING
NCHW Chillers	6	Steam Generator Building - IB
SSCCW Heat Exchanger	2	Turbine Generator
IALL & LALL Evaporators	2	Radwaste Area
Primary & Intermediate Na Pump Motor Generator Set Fluid Coupler	4	Control Building
Primary & Intermediate Na Pump Motor Generator Set Coolers	4	Control Building
Primary & Intermediate Na Pump Motor Generator Set Fluid Couplers	2	Diesel Generator Building
Primary & Intermediate Na Pump Motor Generator Set Coolers	2	Diesel Generator Building

NOTE:
NCHW - Normal Chilled Water (9.7-')
SSCCW - Secondary Service Closed Cooling Water (Figure 9.9-5)
IALL - Intermediate Activity Level Liquid
LALL - Low Activity Level Liquid

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15 TABLE 9.9-2

NORMAL PLANT SERVICE WATER SYSTEM MAJOR COMPONENTS

	DESCRIPTION	QUANTITY	APPROX. NPSW FLOW FOR EACH COMPONENT
15	Normal Plant Service Water Pump	2	26,600

43

50

15 NOTE: NPSW - Normal Plant Service Water (Figure 9.9-3)

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15

TABLE 9.9-3

COMPONENTS SERVED BY EMERGENCY PLANT SERVICE WATER SYSTEM

Component	Component Location: Bldg.	Cell	Elev.	Component Service Requirements: Flow GPM	*EWT ^{0}F	BTU/HR ($X10^6$)
Standby Diesel Generator A	DGB	511	816'-0"	1500	90^0	13.2
Standby Diesel Generator B	DGB	512	816'-0"	1500	90^0	13.2
Emergency Chilled Water System Chiller A	SGB	216	733'-0"	2100	90^0	10.4
50 Emergency Chilled Water System Chiller B	SGB	217	733'-0"	2100	90^0	10.4

50

*Entering Water Temp.

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33

15 TABLE 9.9-4

EMERGENCY PLANT SERVICE WATER SYSTEM MAJOR COMPONENTS

	Description	Quantity	Design Data For Each Component
	Emergency Plant Service Water System circulating pump	2	3600 GPM 110 ft. total head
43 33	Emergency Cooling Tower	2	3600 GPM
50	Emergency Plant Service Water Make-Up Pumps	2	150 GPM 92 ft. total head

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292M Water Charges for Private Fire Protection, 1974

801 Facilities Handling Radioactive Materials, 1975

802 Fire Protection Practice for Nuclear Reactors, 1974

901 Uniform Coding for Fire Protection, 1976

10. ASME Boiler and Pressure Code, Section III and Section X (For Containment penetrations and isolation valves) 48

9.13.1.2 System Description

a. The Non-Sodium Fire Protection System consists of the following:

   Water Supply System
   Wet Sprinkler System
   Preaction Sprinkler System
   Water Spray System
   Carbon Dioxide Gas Blanketing System
   Halon 1301 Gas Blanketing System
   Standpipe System
   Portable Fire Extinguisher System
   Fire Detection System

   The general description of the above systems is provided in Table 9.13-4. The P&I Diagram of the Water Supply System (Figure 9.13-1) shows the physical location of the plant buildings and the water supply system. The P&ID for the Halon 1301 Gas Blanketing System
50 is shown in Figure 9.13-1a. The fire prevention and protection systems to be provided for all the areas associated with the safety related structures, systems and components are listed in Table 9.13-3.

b. The Control Room fire protection and extinguishing system consists of Halon 1301 portable gas fire extinguishers for manual fire-fighting operations as indicated in Table 9.13-3. The design des- 13 cription of these systems is provided in Table 9.13-4.

   Fire standpipes are provided outside the Control Room for manual 48 water hose spray protection.

   The Computer Room represents a separate fire zone within the Control Room, therefore, the supply and exhaust ducts to the Computer Room are provided with motor operated fire dampers. The automatic or remote manual operation of these dampers provides sealing for the Computer Room. 20

The Computer Room fire protection and extinguishing system consists of the Halon 1301 gas blanketing system with portable fire extinguishers for manual fire-fighting operations as indicated in Table 9.13-3. 48

During a fire, the Computer Room will be completely isolated from the Control Room by the Control Room HVAC System. After the fire is extinguished, the Control Room HVAC system will be placed in a 100% exhaust mode and the Computer Room exhaust duct damper will be opened. The negative pressure established by the exhaust mode will clear out the majority of the combustion and Halon 1301 decomposition products. After some time the supply air damper will be opened to speed up the clean up operation. Since the Computer Room ventilation rate is approximately 8 air changes per hour, within a two hour period, the room atmosphere will be sufficiently cleaned to permit personnel entry without danger to the entering operator.

The Halon 1301 storage tanks are located outside of the Control Room area. To limit the potential areas affected by leakage, the Halon 1301 distribution system is isolated from the storage tanks by redundant isolation valves. At the vicinity of the storage tanks where the probability for small leaks exists, redundant detectors will be provided to alert plant operating personnel of the presence of Halon 1301 in the tank area. Since the Halon 1301 storage tanks are located outside of the Control Room are, the distributing pipeing is isolated from the storage tanks by double isolation valves and the Halon-1301 system will be manually activated by Control Room personnel. For these reasons, an accidental leak of Halon 1301 into the Control Room or its HVAC System is highly unlikely. Therefore, no Halon 1301 detectors have been planned for the Control Room Ventilation System.

The entire Halon 1301 system is located in a tornado missile-proof and flood-protected structure. The Halon 1301 storage system up to and including the storage system insolation valves will be designed to Seismic Category I requirements. The detailed system description including any required redundancy will be provided in the FSAR. 20

c. The design features of the Fire Detection System are provided in Table 9.13-4. The alarm system is designed such that the failure of single fire detection device does not affect the operation of remaining detection devices connected to the same detection zone. The interconnecting circuitry between the detection devices within a zone is continuously supervised, and a break in the circuitry is annunciated both locally and in the Control Room. 13

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TABLE 11.1-9

RADIOISOTOPE INVENTORY IN PRIMARY COLD TRAPS AFTER

15 YEARS OPERATION

	Activated Corrosion Products	Percentage Of Isotope Released in Cold Trap	Activity in Primary Cold Trap (Ci) Shutdown	10 Days After Shutdown
	Co^{60}	10	19.6	19.6
	Co^{58}	10	47.9	43.4
	Mn^{54}	50	239	234
	Ta^{182}	10	3.5	3.3
	Cr^{51}	10	3.9	3.0
50 49	Fe^{59}	10	0.37	0.32
	Fission Products			
	Sr^{89}	10	70	60
	Sr^{90}	10	25.9	25.9
	Y^{91}	10	20.1	17.8
	Zr^{95}	10	37.3	33.5
	Nb^{95}	10	37.3	33.5
	Ru^{103}	10	53.2	44.6
	Ru^{106}	10	38.0	37.2
	Rh^{106}	10	38.0	37.2
	Sb^{125}	90	2710	2680
	Sb^{127}	5	123	20.5
	Te^{127m}	5	133	125
	Te^{127}	5	146	125
	Te^{129m}	10	460	376
	Te^{129}	10	460	376
	I^{131}	5	1675	750
49	Te^{132}	5	1110	132

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TABLE 11.1-9

(continued)

Activated Corrosion Products	Percentage Of Isotope Released in Cold Trap	Activity in Primary Cold Trap (Ci)	
		Shutdown	10 Days After Shutdown
Fission Products			
I^{132}	5	1110	132
Cs^{134}	60	2155	2135
Cs^{136}	5	583	342

Amend. 49
April 1979

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11.2 LIQUID WASTE SYSTEM

11.2.1 Design Objectives

The Liquid Radwaste System is designed to process contaminated liquids from the Clinch River Breeder Reactor Plant prior to reuse or release into the environment. The design objectives are to purify and reuse waste liquids where possible and to minimize the total activity in liquid effluents and the total volume of concentrates that will require drumming. The basic approach is to process liquid radwaste so that virtually all of the radioactive material is contained in solid material, to load all solid radioactive material into containers that meet Department of Transportation regulations, and to transfer the containers to a licensed contractor for processing or disposal. The design objective is to release as little radioactivity as is practicable into the environment under normal operating conditions. The radioactivity released will be as low as reasonably achievable and less than the limits set
49 by 10CFR20.

Table 11.2-1 shows the estimated design annual inventory of radioactivity by nuclide which may be discharged after dilution into the Clinch River. The first column lists the isotopes, the second column gives the half-lives. The third column lists the design annual activities
49 released from the monitoring tanks of the Low Activity Level Liquid (LALL)
System. The fourth column lists the annual activities which may be released from the storage tank of the Intermediate Activity Level Liquid (IALL) System. The fifth column lists the sum of the discharged activities per isotope. The two systems are described in detail in the following
49 Section 11.2.2. Activity in the LALL System comes from sodium spillage
washed into the plant drains. Listed activities in the IALL System come from fission products, plutonium, and corrosion products that have plated out or deposited on components washed in the Large Component Cleaning
49 Vessel (LCCV) and the small component autoclave (SCA). Activity levels
of the fission products are based on an assumption of 1.0% failed fuel, 30 year irradiation and 10 day decay time. Corrosion product activity is based on 30 year irradiation and 10 day decay. The activities listed in the column represent a reduction of 10^5 by decontamination procedures of all isotopes entering the liquid radwaste system except for iodine and tritium for which the decontamination factors are 10^4 and 1 respectively. A plutonium limit of 100 ppb in the primary sodium is assumed.

The est[illegible] of the released radioactivity levels shown in
49 Table 11.2-1 [illegible]rvative. The decontamination factors (DF) are
conservative[illegible]stimated from operating experience of Light Water Reactors. Collection [illegible] activity in the LALL System is based conservatively on the assumption of a 850 gallons per day drainage containing 10^{-4} μCi/cc. This activity is due to sodium removed from the reactor for chemical analysis or due to spills and cleanup during normal plant operations.
49 The amount of activity stored and released from the LALL System is
conservatively estimated in assigning a single value to the fraction of

the activity removed in the water washes. Both systems conservatively assume that all of the fluids are processed 10 days after 30 years of irradiation. The assumption of 10 days decay is connected to the assumption of no spare parts availability. Since, in fact, it is planned to have an inventory of spare parts, a more likely decay period is from one to three months. Finally, it should be noted that no planned release
49 of activity from the IALL System is contemplated under normal operating procedures. It may also be noted that the activity associated with the IALL System accounts for the major fraction of the total activity listed in Table 11.2-1. The estimated cleaning process data is provided in Table 11.2-8.

49 The decontaminated water in the IALL System will be recycled for use in washing components in the LCCV. Processed water in the LALL System monitor tanks will not be reused, since to reuse it would mean injecting radioactivity into the laboratories and showers used by plant personnel. In addition, reuse would require control of the water being supplied to many areas which is both complex and costly. Instead, the liquid radwaste, after monitoring to assure compliance with all appropriate
49 Federal and State regulations, will be released into a diluting stream or used as makeup water for the IALL system.

11.2.2 System Description

Figure 11.2-1 shows the schematic for the liquid radwaste system. The system consists of two subsystems. The first is designed to process liquids with intermediate levels of radioactivity that are reused; the second is designed to process liquids with low levels of radioactivity that are diluted by water from personnel showers and other sources and then
49 released into the Cooling Tower Blowdown Stream (CTBS). The decontamination of liquid radwastes in both systems is carried out in the following sequence: The liquids are collected, neutralized if necessary, and then treated in batches to one or more cycles of filtration, evaporation, and demineralization.
49 Normally, the stream in the LALL and IALL Systems are kept completely separate. However, the design provides the option of utilizing the equipment in either system as a backup for the equipment in the other system through a variety of interconnections shown on the schematic. The Radwaste System provides a gross decontamination factor (DF) of 10^5
49 except for iodine and tritium where the decontamination factors are 10^4 and 1 respectively. These DFs are based on operating data compiled in WASH-1258. The concentrated liquid radwaste is collected from the bottom of the evaporator and transferred to a solid radwaste system for solidification and disposal by burial off-site. The contaminated resins in the demineralizer and the contaminated filters are also transferred to the solid radwaste system for packaging and disposal.

49 Table 11.2-2 shows the design annual concentration of activities by isotope flowing into the LALL and IALL systems. The fission
50 product activities are calculated on the basis of 1.0% failed fuel,

TABLE 11.2-1

DESIGN ANNUAL RELEASED ACTIVITY INVENTORY(1)

Isotope(4)	Half-life(5)	Low Level(2) Activity (Ci)	Intermediate(3) Level Activity (Ci)	Total Activity (Ci)
H-3(6)	12.3Y	2.86(-3)*	3.31(-2)	3.60(-2)
Na-22	2.6Y	5.34(-8)	6.24(-7)	6.77(-7)
Na-24	15H	6.65(-9)	7.76(-8)	8.43(-8)
Cr-51	28D	-	5.14(-7)	5.14(-7)
Mn-54	312D	-	1.61(-5)	1.61(-5)
Co-58	71D	-	5.37(-5)	5.37(-5)
Co-60	5.2Y	-	8.37(-6)	8.37(-6)
Fe-59	45D	-	5.92(-8)	5.92(-8)
Sr-89	51D	1.48(-9)	6.58(-7)	6.59(-7)
Sr-90	28.8Y	1.04(-9)	4.73(-7)	4.74(-7)
Y-90	64.1H	1.04(-9)	4.73(-7)	4.74(-7)
Y-91	58D	1.4[illegible](-9)	1.93(-7)	1.94(-7)
Nb-95	35D	2.28(-9)	1.26(-5)	1.26(-5)
Zr-95	64D	2.28(-9)	1.33(-5)	1.33(-5)
Mo-99	67D	-	4.35(-7)	4.35(-7)
Ru-103	40D	3.36(-9)	1.26(-5)	1.26(-5)
Ru-106	1Y	4.23(-9)	2.89(-5)	2.89(-5)
Rh-106	2.2H	4.23(-9)	2.89(-5)	2.89(-5)
Ag-111	7.5D	-	1.27(-8)	1.27(-8)
Sb-125	2.7Y	7.40(-9)	8.63(-8)	9.37(-8)
Te-127m	109D	3.00(-9)	1.35(-6)	1.35(-6)
Te-127	9.35H	3.00(-9)	1.14(-6)	1.14(-6)
Te-129m	34D	1.20(-8)	4.05(-6)	4.06(-6)
Te-129	70M	1.20(-8)	4.05(-6)	4.06(-6)
Te-132	78H	6.41(-9)	2.91(-6)	2.91(-6)
I-131	8.1D	3.23(-6)	7.99(-6)	1.12(-5)
I-132	2.3H	6.09(-7)	7.12(-6)	7.73(-6)
Cs-134	2.1D	3.40(-8)	7.21(-7)	7.55(-7)
Cs-136	13D	1.96(-7)	1.92(-6)	2.11(-6)
Cs-137	30Y	1.30(-6)	1.54(-5)	1.67(-5)
Ba-140	12.8D	5.86(-10)	2.55(-6)	2.55(-6)
La-140	40H	5.86(-10)	2.55(-6)	2.55(-6)
Ce-141	32.5D	1.55(-9)	4.24(-7)	4.26(-7)
Ce-143	33.7D	5.08(-10)	2.28(-7)	2.28(-7)
Pr-143	13.7D	5.08(-10)	2.28(-7)	2.28(-7)
Ce-144	285D	6.90(-10)	3.11(-7)	3.11(-7)
Pr-144	17M	6.90(-10)	3.11(-7)	3.11(-7)
Nd-147	11.1D	2.97(-10)	9.58(-8)	9.60(-8)
Pm-147	2.7D	3.93(-10)	1.74(-7)	1.74(-7)
Eu-155	1.8Y	-	1.80(-8)	1.80(-8)
Ta-182	115D	-	9.97(-7)	9.97(-7)
Pu-238	86Y	2.46(-10)	3.86(-9)	4.11(-9)
Pu-239	2.0(4)Y	6.54(-11)	1.03(-9)	1.09(-9)
Pu-240	6.7(3)Y	8.54(-11)	1.34(-9)	1.42(-9)
49 Pu-241	13Y	7.08(-9)	1.14(-7)	1.21(-7)
Pu-242	3.8(5)Y	1.81(-13)	2.89(-12)	3.07(-12)

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TABLE 11.2-1 (Continued)

Isotope	Half-life(5)	Low Level(2) Activity (Ci)	Intermediate(3) Level Activity (Ci)	Total Activity (Ci)
Np238	2D	2.77(-15)	4.23(-14)	4.51(-14)
Np239	2.4D	1.27(-11)	2.05(-10)	2.17(-10)
Am-241	433Y	2.52(-11)	4.01(-12)	4.26(-10)
Am-242m	152Y	9.95(-13)	1.58(-11)	1.67(-11)
Am-242	16H	9.95(-13)	1.58(-11)	1.67(-11)
Am-243	7.4(3)Y	4.07(-13)	1.85(-11)	1.89(-11)
Cm-242	163D	1.77(-11)	2.84(-10)	3.02(-10)
Cm-243	30Y	2.45(-15)	3.89(-12)	3.89(-12)
Cm-244	18Y	5.12(-12)	8.14(-11)	8.65(-11)

49

(1) 1.0% failed fuel for fission products and 100ppb Pu in the primary sodium. 30 year irradiations and 10 day decay for fission and activated corrosion products. 49

(2) Total discharge of $\sim 10^{-4}$ μCi/cc at 850 gallons per day.

(3) 10% of the annual stored inventory is released.

(4) All released activity has been decontaminated by a factor of 10^5 except for iodine (DF=10^4) and tritium (DF=1).

(5) Y=years, D=days, H=hours, M=minutes.

50 (6) The tritium values do not include Balance of Plant tritium release of 8.45×10^{-3} Ci/day. See Section 11.2.6.2.

49 * 2.86(-3) = 2.86×10^{-3}

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11.3 GASEOUS WASTE SYSTEM

11.3.1 Design Base

The design objective of the Gaseous Waste System is that the levels of radioactive materials in the plant effluents to the environment shall be kept as low as reasonably achievable. Extensive efforts shall be directed toward the development of system designs that will result in minimizing or eliminating planned releases of radioactive material to the environment during normal plant operation. Plant design objectives include conformance with the requirements of 10 CFR 20.

The design base and the expected values of the annual activity release for each gaseous radionuclide are listed in Tables 11.3-11 and
50 11.3-12, respectively. The summed annual dose at the site boundary
from gaseous radioactive effluent for the design base conditions is
approximately 4.5 mrem total skin dose (which includes both α and β), and
50 0.66 mrem whole body gamma dose (see Table 11.3-18). This annual dose
is well below the requirements of 10 CFR 20 for unrestricted areas. Dose rates within accessible restricted areas are below the requirements of 10 CFR 20.

The Radioactive Argon Processing Subsystem (RAPS) shall maintain the primary sodium system cover gas at an acceptable level of radioactivity and shall provide a source of low-radioactivity gas for use in reactor seals. The cover gas is to be contained within the RAPS circuit during its radioactive decontamination and reuse, except for leakages and cover gas samples taken for failed-fuel and impurity monitoring.

The Cell Atmosphere Processing Subsystem (CAPS) shall process plant effluents that contain or might potentially contain radioactivity, in order to reduce the radioactivity to levels that are as low as reasonably achievable during the normal range of routine plant operations. During off-normal operations, the CAPS function is to continue to prevent, to as great an extent as is practicable, the release of radioactivity.

11.3.2 System Description

11.3.2.1 Process Flow

The origin, flow paths, and release points of the gaseous radioactive waste system are shown schematically in Figure 11.3-1. The radioactive gases generated in the reactor (see Sections 11.3.2.2 and 11.3.2.3 for composition, flow rates, and concentrations) consist of tritium and noble gas isotopes. The latter, and some of the tritium, migrate to the reactor and Primary Heat Transport System (PHTS) cover-gas spaces. The Radioactive Argon Processing Subsystem is an essentially closed internal system which continuously processes the cover gas to reduce its activity and then returns the "recycle" argon to the seals and cover-gas spaces. An expanded schematic diagram of the RAPS is
49 shown in Figure 11.3-2.

No significant quantities of iodine nor particulate forms of radioactive isotopes, excluding those daughter products associated with noble gas decay, are expected to be present in the Gaseous Radwaste System. Although some vaporization of nongaseous isotopes from the liquid sodium into the reactor cover gas may occur, all cover gas entering the system is processed through two vapor traps, which are expected to remove essentially all nongaseous isotopes, including trace quantities of sodium iodide. Continuous radiation monitoring of the gases is provided by the process monitoring of RAPS and CAPS and monitoring of the CAPS exhaust.

Radioactive gas leakages into the inerted cells of the Reactor Containment Building (RCB) and Reactor Service Building (RSB) are collected and processed through the CAPS before release to the environment. An expanded schematic diagram of the CAPS is shown in Figure 11.3-3.

50 Most (99.8%) of the tritium generated forms a hydride in the sodium; it is then partially removed from solution in the sodium by cold trapping. A very small portion diffuses into the cells of the Intermediate Bay of the Steam Generator Building. A detailed description of all the identifiable leakage and discharge paths is given in the following paragraph.

In Figure 11.3-1, certain paths have been assigned "numbers" that correspond with the following discussion:

Path 1a. Reactor cover gas is conservatively estimated to diffuse through the reactor head seals at the rate of 0.012 scc/min. This leakage diffuses into the head access area and is discharged to the atmosphere through the RCB heating and ventilating exhaust duct.

Path 1b. The buffered head seals are expected to leak (to the head access area) a maximum of 7 scc/min of recycle argon cover gas.

Path 2. Although the cover gas lines connected to the reactor and other components in the Primary Heat Transport System are not expected to leak, a leakage of 1 scc/min has been assumed, for the purpose of conservatism, in the design basis evaluation. Also, tritium dissolved in the sodium in this system will diffuse through the hot pipe walls into the RCB cells. These two leakages are considered to diffuse into the RCB cell atmospheres, which are collected and processed by CAPS and are discharged to the CAPS heating and ventilation exhaust.

Path 3. Although the RAPS and CAPS piping and components are also not expected to leak, a leakage equivalent to 1 scc/min of RAPS Cold Box influent gas has been assumed, for the purposes of conservatism, in the design basis evaluation. This assumed leakage is considered to diffuse into the RSB cell atmospheres, which are processed by CAPS and are discharged to the CAPS
49 heating and ventilation exhaust.

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Path 4. Ar^{39} and Kr^{85} collected and stored in RAPS (see further below) are bled into CAPS; these also are discharged to the CAPS heating and ventilation exhaust.

Path 5. The Failed Fuel Monitoring System discharges reactor cover gas samples to CAPS. After processing, this gas also is discharged to the CAPS heating and ventilation exhaust.

50 Path 6. Other plant systems, specifically Refueling, Maintenance and Auxiliary Liquid Metal intermittently discharge radioactive or potentially radioactive gases through CAPS to the CAPS heating and ventilation exhaust.

Path 7. Tritium dissolved in the sodium of the PHTS will transfer to the Intermediate Heat Transport System (IHTS) sodium by diffusing through the intermediate heat exchanger (IHX) tube walls. A very small but finite amount will then diffuse through the hot leg piping in the cells of the intermediate bay (IB) and steam generator bay of the Steam Generator Building and will mix with the ventilation streams in that building.

Radioactive gases are thus released to the Heating and Ventilation Systems of the IB, the RCB, and the RSB (Paths 7, 8, and 9 on Figure 11.3-1). The discharge of these streams to the environment is discussed in Section 11.3.2.6.

Balance of Plant (BOP) tritium release (Path 10, Figure 11.3-1) is discussed in Section 11.3.6.2.

A schematic diagram of the process flow in the cover-gas recycle system, which includes the reactor, the overflow vessel, and the PHTS pump cover-gas spaces, the oil traps for the pumps, the Failed Fuel Monitoring System, the recycle argon vessels, and RAPS equipment, is shown in Figure 11.3-2. The recycle system components, distinguished by solid-line blocks, constitute the collection, control, and principal processing portion of the system, although isotope decay occurs in all parts of the system. The function of this system is to continuously draw radioactive gases from the cover-gas spaces, so that noble gas isotopes, both stable and radioactive, are extracted from the cover-gas spaces by condensation in a cryostill, and then to return the purified argon to the cover-gas spaces as a "recycle" argon purge. The activity in the cover gas is thus dependent on the production rate in the reactor, the purge rate, the holdup time, the half-life of each isotope, and the cryostill efficiency.

Argon flows from the recycle vessels nominally at 5.15 scfm: 0.5 scfm to each of the PHTS pumps and 3.65 scfm to the reactor cover gas space. The PHTS pumps gas effluent is divided equally (by design), so that 0.75 scfm (total) passes through the three shaft seal spaces and the three oil traps and enters the RAPS input (vacuum vessel); the other 0.75
49 scfm bleeds to the common pressure-equalization line that joins the reactor,

the reactor overflow vessel, and the PHTS pumps' cover gas spaces. From this pressure-equalization line, 1.0 scfm of the gas passes through a sodium vapor trap and through the Failed Fuel Monitoring System before entering the RAPS input; the remaining 3.4 scfm goes through the overflow vessel cover gas space, then through a sodium vapor trap to RAPS.

RAPS continuously processes a flow of 10.0 scfm which is made up of the 5.15 scfm input and 4.85 scfm of recirculated throughput. The output of RAPS, 5.15 scfm, is delivered to the recycle argon vessels. The RAPS cryogenic distillation column operates with liquid argon as the still bottom. This liquid absorbs the krypton and xenon isotopes and permits their separation by draining, evaporating, and transferring them to the noble gas storage vessel. The transfer to the noble gas storage vessel is to be an annual procedure. During the subsequent year, the transferred gas will be bled at a controlled, low rate from the noble gas storage vessel into CAPS, and through CAPS to the RSB CAPS H&V exhaust. The release process will occur over a period of several months.

The Cell Atmosphere Processing Subsystem process flow circuit is shown in detail in Figure 11.3-3. The individual inputs to CAPS, grouped as shown in the five upper boxes in the diagram, are as follows:

1) Cells and Pipeways - During normal operation, there will be a small but finite diffusion and leakage of radioactive gases through the piping and components. This source of
50 activity will be accumulated in the atmospheres of respective RCB and RSB cells and when the cells are purged to CAPS, the contained radioactivity will be collected and processed in CAPS. It is conservatively assumed for calculational purposes that an average 1 scc/min of reactor cover gas and 1 scc/min of RAPS cold-box process
50 gas will be leaked into the cells; further, they will be exhausted to CAPS without delay, except that Ne23 is assigned a delay of 8 minutes. The cell atmosphere sampling and analysis units will divert to CAPS for processing any activity concentration exceeding a predetermined setpoint value. For other than RCB cells, normal cell-atmosphere nitrogen, when it is not radioactive, passes directly to the heating and ventilation exhaust system, bypassing CAPS.

2) Mass Spectrometer - This equipment, part of the Failed Fuel Monitoring System, periodically samples reactor cover gas and discharges portions of the samples into
49 CAPS.

3) Gas Services Exhausts - Intermittently, CAPS will receive exhausted nitrogen, argon, or air from vessel cover gases, cooling gases, cleaning, bagging, and fuel handling operation, and other services. These are only infrequent, potential carriers of radioactivity; they will not normally contribute a significant amount of radioactivity relative to the first three sources.

4) RAPS Cold Box - These CAPS inputs include RAPS cold box components overpressure relief, purge of RAPS for component maintenance, RAPS cryostill nitrogen coolant effluent and
50 the noble gas bleed from the noble gas storage vessel. The noble gas bleed is normally continuous; the others will be used only in case of a malfunction in the RAPS circuit, and only for short periods of time, as for a repair or correction.
50 With the exception of the noble gas bleed, these sources will not normally contribute a significant amount of radioactivity relative to sources (1), (2), and (3).

The nominal volume input of gases to CAPS is the time-averaged sum of the inputs listed. The CAPS design flow rate is 38 scfm.

A recirculation loop, shown by broken-line in Figure 11.3-3, will return the CAPS output to the vacuum vessel if radioactivity above an acceptable level is detected by the effluent monitoring system.

The tritium-water removal process uses an oxidizer and a freeze-out dryer; it oxidizes tritium, collects the resultant tritiated water, and passes it to the Radioactive Liquid Waste System, where it is incorporated into a solid form for off-site disposal.

CAPS incorporates two cryogenically-cooled charcoal delay beds and a tritium-water removal unit. In the beds, the short-lived gaseous radioactive species are adsorbed and then decay; they are thus removed from the process gas stream.

RAPS and CAPS have different process methods, i.e., the distillation-process removal of noble gases in RAPS rather than the delay beds in CAPS, and the oxidation-process removal of tritium in CAPS. In each subsystem, however, the input is collected in a vacuum vessel, from which it is transferred and stored under pressure in a surge vessel. It is then treated in the respective cold box. The recirculation-loop feature of these cryogenic subsystems increases their effective capacity to reduce activity. In RAPS, it also permits maintaining a
49 steady throughput under conditions of changing input.

11.3.2.2 Gaseous Radioactive Waste Inputs to System

The radioactive waste gases consist of noble gas radionuclides and tritium that are generated by fission and/or neutron activation. The noble gas radionuclides migrate to the reactor cover-gas space, although a time lag occurs in the leakage from ruptured fuel and in the movement to the cover gas. The tritium remains primarily in the sodium, from which it is removed by cold trapping. The tritium concentration in the cover gas will be affected by the sodium temperature, the cover gas temperature and pressure, the cold trapping efficiency, and the concentration of hydrogen in the sodium. The latter factor, in turn, depends on the diffusion rate of hydrogen from the steam-generator tubes into the intermediate sodium and the subsequent diffusion of the hydrogen into the primary sodium system through the IHX tube walls.

Table 11.3-1 lists the radionuclides of concern, their half-lives, decay constants, and design base input rates to the cover-gas space at normal reactor power level (975 MWt). The noble gas input rates to the cover gas are adjusted for decay during their release from failed fuel (modeling described in Section 11.1). The assumed condition of 1% failed fuel is the design base point for the radioactive gas processing systems.

11.3.2.3 Activity Inventories and Concentrations

a. Reactor Cover-Gas Space - The steady-state inventory of a specific radionuclide in the bulk volume of the reactor cover gas can be calculated from the following formula:

$$I = \frac{\dot{I}\Sigma}{\lambda + F/V} \qquad \ldots(1)$$

Where I = inventory (Ci), $\dot{I}$ = input rate (Ci/min), λ = decay constant (min^{-1}) (0.693 ÷ half-life), Σ = processing efficiency factor (typically taken as unity), F = purge rate (3.65 scfm), and V = cover-gas space volume (410 scf). F/V is the "purge factor". The concentration of a radionuclide in the cover-gas space is its inventory divided by the total gas volume adjusted to standard conditions (68°F, 14.7 psia).

Table 11.3-2 lists, for each isotope of concern, the inventory concentration in the reactor cover gas for the design-base condition of 1% failed fuel.

b. RAPS Process Stream - Table 11.3-3 lists the inventories in the principal RAPS vessels, and Table 11.3-4 lists the concentrations
49 of activity at selected points in the RAPS process stream. The

11.3.2.6 Radioactive Gaseous Site Boundary and Restricted Area Concentrations

Radioactive gaseous concentrations at site boundary have been calculated for five Heating and Ventilating System air stream sources; these are compared to 10 CFR 20 unrestricted area MPC limits. The air streams are:

a. Reactor Containment Building Vent - Cover gas diffusion through the reactor head seals and recycle argon gas leakage through the buffered seals mix with the H&V air stream and are exhausted through the main RCB exhaust duct to the exhaust opening located on the top of the Confinement Building. (The release point is #5A on Table 11.3-20 and on Figure 11.3-9.) The flow rate through the exhaust is 14,000 cfm. Associated activity concentrations are shown in Table 11.3-13.

b. Reactor Confinement Building Vent - The Annulus Air Cooling System is provided as a means to mitigate events beyond the design basis. Activity will only be discharged through release point #13, located at the top of the Reactor Confinement Building, in the event of very low probability accidents beyond the design basis. Thermal Margins Beyond the Design Basis are discussed further in Reference 10 of PSAR Section 1.6.

c. CAPS H&V Exhaust - CAPS effluent discharges into the CAPS H&V ducting and is released to the environment through a missile protected exhaust structure located on the roof of the RSB. The air stream flow rate is 3000 cfm, and the release is within the restricted area (release point #6). Associated activity concentrations are shown in Table 11.3-14.

d. Intermediate Bay Vent - The exhaust duct located in the IB receives ventilation exhaust air from the Intermediate Bay area. The flow rate is 64,000 cfm with release within the restricted area (release point #1). The associated activity concentration is shown in Table 11.3-16a.

e. Turbine Generator Building Vent - BOP Tritium discharge will be released to the environment from the TG Building H&V vent (release point #7). The flow rate is indicated in Table 11.3.20. Associa-
49 ted tritium concentration in the release is given in Table 11.3-16.

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The two restricted-area locations that present potential occupational exposure to airborne radionuclides are the IHTS piping cells and the head access area. The estimated leakage of tritium into the IHTS piping cells is 1.6×10^{-4} Ci/day. This is diluted in 1000 cfm of
ventilating air, resulting in an expected concentration of 3.9×10^{-9}
50 $\mu Ci/cm^3$, less than 0.1% of the MPC (occupational) concentration of 5×10^{-6} $\mu Ci/cm^3$.

The normally accessible area with the largest potential atmosperhic radionuclide concentrations is the head access area (HAA). As shown in Table 11.3-13, the concentrations in this region for operation
with 1% failed fuel will be approximately 1.3×10^{-7} $\mu Ci/cm^3$, which
50 results in a sum of the fractional MPC's (occupational) of 0.05. Also
shown on Table 11.3-13 are the expected concentrations for operation
with 0.1% failed fuel, which results in a sum of the fractional MPC
50 (occupational) value of 0.007.

Particulates are not expected to be discharged from the design release points of the CRBRP. However, as discussed in Section 11.4, monitors will be provided, as appropriate, to ensure the capability of detection.

11.3.3 System Design

11.3.3.1 General

The RAPS and CAPS System designs emphasize all-welded construction, wherever practicable, and bellows-sealed valves throughout, so that leak-tightness is enhanced. There will be no field-routed piping in either system.

11.3.3.2 Equipment

RAPS and CAPS flow diagrams are shown in Figures 11.3-4 through 11.3-7. The design parameters of the major equipment components shown in the diagrams are listed in Table 11.3-17; this table summarizes the design codes, the seismic categories, the operating pressures and temperatures, the actual volumes of the components, and their capacities under normal operating conditions. Experience with components is limited. Some testing has been performed as part of the development of FFTF.

These components are all located within the RCB and RSB, which are Seismic Category I structures, and are tornado-protected by the building. Consequences of equipment failures by rupture of leakage are
49 discussed in Section 15.7.

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11.3.3.3 Instrumentation

Process instrumentation is to be installed in RAPS, CAPS, and the inert gas distribution systems in order to effect the control, generally in the automatic mode, of pressures, temperatures, radioactivity concentrations, and flow rates (see Section 9.5.5). The normal pressures and temperatures for the vessels are listed in Table 11.3-17.
50 Radioactivity concentrations and flow rates have been discussed in previous sections of Section 11.3.

Radiation monitoring for the nitrogen-inerted cells in the RCB and RSB is provided by two separate multi-channel sampling and analysis units, typically piped as shown in Figure 11.3-8. The individual cell atmospheres are continuously withdrawn but are sequentially subjected to analysis for detection of radioactivity, water vapor, and oxygen. Detection of oxygen concentration in excess of the high set point will automatically initiate purging of the violated cell with fresh nitrogen to reduce the concentration to the low set point. Initiation for automatic purging to reduce water vapor concentration rather than oxygen is an operator option, selectable by a hand switch. Detection of radioactivity in the cell atmosphere automatically directs the cell effluent to vent to CAPS if the radioactivity is above the set point; otherwise, it is vented to HVAC for direct release. This option is provided for all the inerted cells in the RSB, but not for the inerted cells in the RCB. The effluents from the RCB cells are always vented to CAPS during normal plant operation.

Two radiation monitors in series are provided in a common RSB inerted-cell-vent header before the cell gases are discharged into the HVAC ducting. This provides continuous monitoring of the vented gases, and a high-radiation signal provides automatic closure of a common header-isolation valve located downstream of the radiation monitor. The signal also closes all the HVAC vent valves to the individual cells, to prevent release of radioactive gases.

50 The question of whether or not available RSB radiation-monitoring equipment provides adequate discrimination to guard against excessive releases, is addressed in the following sample calculation:

> If a 100,000 ft^3 cell is at, but not above, the threshold of radioactivity detection (1 E-6 Ci/cm^3) and is then purged within one day to correct its oxygen concentration, the purge flow for other than cells in the RCB will enter the H&V effluent duct. Under these conditions, a nominal (1E+5 ft^3) (2.832 E+4 cm^3/ft^3) (1E-6 Ci/cm^3), or 0.0028 Ci, will be in the cell. In the worst case, all of it (0.0028 Ci/day) could be released in the H&V effluent. This is only 1.1% of the normally
> 49 expected daily plant release rate.

11.3.4 Operating Procedures and Performance Tests

The gas inputs to CAPS (listed in Section 11.3.2.1) are drawn into a vacuum tank by one or more of four 25-scfm compressors, depending on input flow rate, which are instrumented to automatically maintain a 7.7 to 12.7 psia pressure in the tank. The compressors are arranged in parallel and their controls are such that one starts when the vacuum pressure reaches 12.7 psia, and others start in sequence if the pressure
49 is not held below 12.7 psia. All compressors stop when the vacuum

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reaches 7.7 psia in the vacuum tank. If the setpoint vacuum pressure exceeds 13.7 psia, a high alarm is triggered. If the temperature of the effluent from the compressors exceeds the high setpoint, indicating inadequate cooling, can low alarm will be triggered to alert the operator to the abnormal condition. 50

RAPS and CAPS are independently operated, with process control being automatic and with control-room as well as local provisions for overriding automatic controls if conditions so dictate. Both subsystems have control and alarm instrumentation and instant data retrieval available in the control room; this provides the operator with information that ensures proper system operation. The effectiveness of operating procedures has been demonstrated as part of the FFTF development and, to a limited extent, analogous systems used in light-water reactors.

The receiver (surge vessel) of the CAPS compressor(s) normally operates at about 40 psig but can be operated in the range of 35 to 135 psig. The outlet flow from the surge vessel is regulated by a flow control valve. When the surge vessel pressure reaches the nominal 40 psig, the flow valve will permit gas to flow to the processing equipment at a flow rate that increases linearly with surge-vessel pressure differential above 40 psig.

If there is a high gas inflow to the surge vessel and the pressure rises above the nominal setpoint, the outflow from the vessel is increased to accommodate to the increased inflow rate. If the inflow exceeds the maximum processing capability, the surge vessel will act as an accumulator and its pressure can increase to 135 psig, at which pressure the compressors are automatically shut down and a high alarm is triggered. In the event of a compressor diaphragm failure, the failed compressor can be isolated and repairs can be made without shutting down the remainder of CAPS. Similarly, individual radioactive gas filters, upstream of each compressor, can be isolated and replaced when a high pressure-drop alarm is triggered from excessive particulate buildup.

One of two parallel tritium-water removal trains is always on line while the other is being regenerated, with the switchover being automatically controlled by a sequencing timer. In the regeneration cycle, any CO_2 which has been frozen out of the process gas is sublimed off between -20°F and 0°F and is released through the CAPS effluent to H&V. Then, ice formed from tritiated water vapor is melted and, [illegible]n 40°F and 70°F, it is drained to the CAPS rad-water holding vessel. [illegible] this vessel, it is periodically transferred to the Radioactive Waste System by manual actuation of the transfer valve. The waste is then
49 converted to non-compactible solid form for ultimate offsite disposal.

The RAPS design is based on operation with 1% failed fuel. Normal operation and the expected releases are based on operation with 0.1% failed fuel. The estimated radioactivity release rate from the gaseous waste and the H&V systems after a 1-year period under average operating conditions are shown in Table 11.3 12. The release rates based on the design condition of 1.0% failed fuel are presented in Table 11.3-11.

11.3.6 Release Points

11.3.6.1 Nuclear Island

There are a total of eight design release points for the Nuclear Island Buildings. The location, height, discharge flow rate, discharge velocity, discharge air temperature, and size and shape of the discharge orifice for each release point are presented in Table 11.3-20.

Ventilation from the Steam Generator Building Intermediate Bay cells is ducted to a single exhaust point located in the Steam Generator Building Intermediate Bay. (Release Point 1 of Figure 11.3-9).

There is a separate exhaust point for each of the Steam Generator Loop cells. Ventilation from each of the three Steam Generator Loop cells is ducted to its respective exhaust point located in the Steam Generator Building Auxiliary Bay(release Points 2, 3, and 4 of Figure 11.3-9). Levels of radioactivity in these areas will make no significant contribution to offsite dose rates.

There are two design release points provided for the RSB. One design release point exhausts the Radwaste Area. This area involves decontamination of non-volatile isotopes and is not expected to result in the release of activity to the exhaust. However, as described in Section 11.4, monitoring of this release point will be provided. An additional release point for the RSB is provided for the exhaust from the CAPS, which is expected to release activity to the exhaust. Per Section 11.4, a monitor will also be provided for this exhaust. The locations of the RSB CAPS H&V exhaust and Radwaste Area exhaust are Points 5 and 6, respectively, of Figure 11.3-9.

49 Ventilation from Reactor Containment Building H&V system and from the annulus pressure filtration system is ducted to a single exhaust point (release point 5a of Figure 11.3-9) located on the top of the Reactor Confinement Building.

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Two release points (points 20 and 21 in Figure 11.3-9) associated with Thermal Margins Beyond the Design Basis design features receive exhaust from the Annulus Air Cooling System and the Containment Cleanup System (These systems are described in Section 9.6.2). These systems are not required to operate during normal operations or to mitigate the consequences of any accidents in the Design Basis. Activity would only be released from these points in the event of very low probability accidents beyond the design basis, such as a hypothetical core disruptive accident.

The Containment Cleanup System exhausts through a release point (21) near the top of the Reactor Confinement Building. Before being exhausted to the atmosphere, the Containment reaction products pass through one of two filter trains, which consist of an air washer, a sodium scrubber and water separator, a heater, a prefilter, a high efficiency particulate air filter (HEPA), an adsorber bed, and an after-HEPA filter. Particulates, radioiodines, radiogases, and plutonium are monitored continuously in the effluent stream.

11.3.6.2 Balance of Plant

A small fraction of tritium produced in the fuel and control rods passes into the steam-water system by diffusion through stainless steel in the IXH and through chromalloy in the steam generators. Tritium is expected to be in the steam-water system in the form of tritiated water. The condenser air removal system removes non-condensible gases (vapors) from the condensing steam. Tritiated water vapor, present in the off-gas flow, constitutes the only expected gaseous release contribution from the balance of plant.

Mechanical vacuum pumps will remove the vapors together with the non-condensible gases and will discharge them to the exhaust plenum of the Turbine Generator Building (exhaust point 7 on Figure 11.3-9). The vapors will mix with the exhaust air. The resulting gaseous tritium release from the TGB is provided in Table 11.3-16.

BOP tritium contribution is included in the dose calculations presented in Section 11.3-8. Balance of Plant tritium release is based on the following assumptions: (1) Plant Capacity Factor of 0.68, (2) Vacuum Pump Operating Factor of 0.85, (3) Radioactivity Input to Steam-Water System 0.016 Ci/day, and (4) Condenser off-gas removal 7 scfm.
50 The design value release of tritiated water vapor amounts to 6.3 x 10^{-5} Ci/day.

Description, design bases, and evaluation of the BOP design
49 are provided in Section 10.

Average effective grazing area of beef cattle and dairy cows is assumed to be 45 m^2. Assuming that 100% of the tritium ingested is evenly distributed within the meat and the average weight of a steer is 500 kg, the cumulative fraction of tritium transferred per kg of beef is 1.0/500 kg or 0.002/kg. Although some grains produced outside of the immediate area are used to supplement their diet, dairy cows on the farm nearest the CRBRP site are allowed to graze outside all year. It is therefore assumed that 100% of the diet of both cows and cattle comes from the fields. All other variables, such as total intake of beef or milk and elapsed time between butchering and ingestion, are provided in Table 11.3A-6 to the Appendix of Section 11.3.

Maximum total annual whole body internal dose from exposure to gaseous effluents to the hypothetical individual who eats only leafy vegetables grown in the closest home garden to the site, eats only the meat from beef cattle grazing in the closest field to the site, drinks one liter of milk per day taken only from the closest known cow to the site, drinks water from the nearest reservoir, and lives in the closest
50 house to the site, is 0.021 mrem/yr. The estimated annual internal dose
from natural radiation to an individual is 18 mrem/yr (ref. 2). Therefore, the maximum internal dose to an individual from exposure to CRBRP
50 gaseous effluents is approximately 0.12 percent of his internal dose
from naturally occurring radiation. The maximum internal dose is a
50 factor of 4×10^{-5} of the 10 CFR 20 annual dose limits for unrestricted
areas. External and internal doses resulting from exposure to daughter products of gaseous effluents have been included in the dose evaluations.

Analysis of doses due to gaseous fallout on the Clinch River assumes an annual mean depth of a reservoir from which water is taken for drinking supplies downstream of the plant site to be 4.8 m. Approxi-
50 mately 0.5 days are required for processing the Clinch River water into
potable water. Internal whole body doses to persons drinking water is calculated with the assumption that total intake during 24 hours is 2,200 cm^3 and 1100 cm^3 for an 8 hour working day (Ref. 3). Period of exposure is assumed to be 260 days for workers and 365 days for public consumption of water. A conservative dilution factor for gaseous deposition into the river, due to flow of fresh water into a reservoir, is derived as flow out÷flow in. An additional dilution factor (0.16) results from the ratio of the river flow relative to the deposition velocity of the gaseous release. Maximum whole body internal dose to an
50 individual from ingestion of water is 2.0×10^{-4} mrem/yr from the gaseous
fallout. The analysis conservatively assumes, on an annual basis, one month of zero flow condition of the Clinch River at the plant site and
49 11 months of summer average flow (4777CFS).

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Population dose assessments for the area within the fifty mile
radius of the plant site have been performed. The pathways assessed
considered all those previously discussed in this section. Population
dose assessments via the ingestion of milk containing fallout utilized
the impact for milk cow populations from the counties within a 50 mile
radius of the CRBRP as provided in Ref. 4. The same general approach
was applied to the population dose associated with the ingestion of beef
containing fallout, i.e., a survey of the beef cattle (Ref. 4) was
incorporated. The population dose associated with the ingestion of
leafy vegetables utilized an extrapolation of nearby land usage from the
plant site to the 50 mile radius area. Ingestion of water containing
fallout effluent considered nearby reservoirs. Inhalation dose assessments
were also included. The man-rem internal dose values from all the
50 pathways considered sum to a value of 0.017 man-rem per year. The
assessments utilized the model specifics of section 11.3A and 50 mile
49 radius population figures associated with the year 2010.

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3. Recommendations of the International Commission on Radiological Protection, Report of Committee II on Permissible Dose for Internal Radiation, ICRP Publication 2, Pergamon Press, New York, New York, 1959.

49 4. Tennessee Statistical Abstract - 1971 Edited by M. G. Currence, Center for Business and Economic Research, University of Tennessee,
50 Knoxville, Tennessee.

TABLE 11.3-1

RADIONUCLIDE INPUT RATES TO REACTOR COVER GAS*

Isotope	Half-Life	Decay Constant (day^{-1})	Input Rate (Ci/day)
49 Xe^{131m}	11.96 day	0.058	112
Xe^{133m}	2.26 day	0.306	3,760
Xe^{133}	5.27 day	0.131	65,100
Xe^{135m}	15.7 min	63.6	95,600
49 Xe^{135}	9.16 hr	1.81	334,000
Xe^{138}	14.2 min	70.2	170,000
Kr^{83m}	1.86 hr	8.98	16,400
Kr^{85m}	4.4 hr	3.78	30,000
Kr^{85}	10.76 years	1.77E-4	2.05
Kr^{87}	76 min	13.1	52,000
Kr^{88}	2.79 hr	5.96	64,400
Ar^{39}	269 years	7.0E-6	0.129
Ar^{41}	110 min	9.07	314
Ne^{23}	38 sec	1576	1.42E+9
50 49 H^3	12.5 years	1.52E-4	1.95E-7***
		TOTAL:	832,000 **

49 * For the design condition

** Exclusive of Ne^{23}

49 ***The rate at which tritium diffuses into the cover gas to replace losses and maintain equilibrium concentrations at the liquid metal-

50 to-gas interface.

TABLE 11.3-2

GASEOUS RADIONUCLIDE CONCENTRATION IN REACTOR COVER GAS*

	Isotope	Inventory (Ci)	Concentration (μCi/scc)
	Xe^{131m}	8.6	0.74
50	Xe^{133m}	2.8E+2	24.
	Xe^{133}	5.0E+3	4.3E+2
	Xe^{135m}	1.2E+3	1.1E+2
	Xe^{135}	2.3E+4	1.9E+3
	Xe^{138}	2.0E+3	1.8E+2
	Kr^{83m}	7.5E+2	64
	Kr^{85m}	1.8E+3	1.5E+2
	Kr^{85}	0.16	1.4E-2
	Kr^{87}	2.0E+3	1.7E+2
	Kr^{88}	3.4E+3	2.9E+2
	Ar^{39}	9.09**	0.783**
49	Ar^{41}	14.4	1.2
	Ne^{23}	8.9E+5	7.7E+4
49	H^3	1.7E-4	1.5E-5

49 * For the design condition
** After 30 years' operation

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TABLE 11.3-3

ACTIVITY INVENTORIES IN RAPS PROCESS VESSELS

Isotope	RAPS Vacuum Vessel		RAPS Surge Vessel		RAPS Cryostill		Recycle Argon Vessels	
	Design (Ci)	Expected (Ci)	Design (Ci)	Expected (Ci)	Design (Ci)	Expected (Ci)	Design (Ci)	Expected (Ci)
Xe^{131m}	1.2	0.12	28	2.8	1.9E+3	1.9E+2	1.1E-3	1.1E-4
Xe^{133m}	38	3.8	8.2E+2	82	1.1E+4	1.1E+3	3.1E-2	3.1E-3
Xe^{133}	6.9E+2	69	1.6E+4	1.6E+3	4.7E+5	4.7E+4	0.61	6.1E-2
Xe^{135m}	24	2.4	32	3.2	2.0	0.20	6.6E-5	6.6E-6
Xe^{135}	2.5E+3	2.5E+2	4.0E+4	4.0E+3	8.8E+4	8.8E+3	1.1	0.11
Xe^{138}	35	3.5	44	4.4	2.5	0.25	8.2E-5	8.2E-6
Kr^{83m}	50	5.0	3.6E+2	36	1.6E+2	16	3.9E-3	3.9E-4
Kr^{85m}	1.7E+2	17	2.0E+3	2.0E+2	2.1E+3	2.1E+2	3.8E-2	3.8E-3
Kr^{85}	2.2E-2	2.2E-3	0.52	5.2E-2	7.2E+2	72	2.1E-5	2.1E-6
Kr^{87}	1.1E+2	11	6.0E+2	60	1.8E+2	18	5.0E-3	5.0E-4
Kr^{88}	2.7E+2	27	2.5E+3	2.5E+2	1.7E+3	1.7E+2	3.7E-2	3.7E-3
Ar^{39}*	3.5	3.5	81	81	28	28	49	49
Ar^{41}	1.1	1.1	7.9	7.9	1.5	1.5	1.3	1.3
Ne^{23}	17	17	0.97	0.97	3.4E-4	3.4E-4	1.3E-3	1.3E-3
H^3	6.6E-5	6.6E-5	2.4E-3	2.4E-2	3.0E-2	3.0E-3	1.6E-3	1.6E-3
50 49 Total	3.9E+3	4.1E+2	6.2E+4	6.3E+3	5.8E+5	5.8E+4	52	50

* After 30 years' operation

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TABLE 11.3-6

ACTIVITY INVENTORIES IN CAPS PROCESS VESSELS

Isotope	CAPS Vacuum Vessel (Ci)		CAPS Surge Vessel (Ci)	
	Design	Expected	Design	Expected
Xe^{131m}	3.9E-3	3.8E-3	5.8E-2	5.7E-2
Xe^{133m}	5.0E-3	2.3E-3	7.5E-2	3.5E-2
Xe^{133}	0.18	0.13	2.7	1.9
Xe^{135m}	2.8E-2	2.7E-2	0.11	0.11
Xe^{135}	0.23	3.9E-2	3.2	0.55
Xe^{138}	1.1E-3	1.5E-4	4.1E-3	5.5E-4
Kr^{83m}	4.1E-3	7.1E-4	4.5E-2	7.7E-3
Kr^{85m}	1.5E-2	2.6E-3	0.19	3.4E-2
K[illegible] 5m	9.1E-3	4.1E-3	0.14	6.1E-2
Kr^{87}	8.2E-3	1.4E-3	8.0E-2	1.4E-2
Kr^{88}	2.4E-2	4.1E-3	0.28	4.9E-2
Ar^{39}*	1.0E-3	6.7E-4	1.6E-2	1.0E-2
Ar^{41}	6.1E-4	1.6E-4	6.6E-3	1.7E-3
Ne^{23}	9.1E-6	9.1E-6	2.0E-6	2.0E-6
50 H^3	2.9E-5	2.9E-5	4.4E-4	4.4E-4
Total	0.50	0.22	6.9	2.9

49 *After 30 years' operation

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TABLE 11.3-7

ACTIVITY CONCENTRATIONS IN CAPS PROCESS STREAM

Isotope	Influent to Vacuum Vessel (μCi/scc)		Effluent from the Second Charcoal Bed (μCi/scc)	
	Design	Expected	Design	Expected
Xe^{131m}	8.7E-4	8.5E-4	3.9E-10	3.8E-10
Xe^{133m}	1.1E-3	5.2E-4	*	*
Xe^{133}	4.0E-2	2.9E-2	*	*
Xe^{135m}	7.4E-3	7.3E-3	*	*
Xe^{135}	5.1E-2	9.0E-3	*	*
Xe^{138}	3.0E-4	4.0E-5	*	*
Kr^{83m}	9.5E-4	1.6E-4	3.0E-10	5.2E-11
Kr^{85m}	3.4E-3	5.9E-4	5.4E-6	9.4E-7
Kr^{85}	2.1E-3	9.2E-4	1.6E-3	7.3E-4
Kr^{87}	1.9E-3	3.3E-4	*	*
Kr^{88}	5.4E-3	9.4E-4	2.4E-7	4.2E-8
Ar^{39}**	2.4E-4	1.5E-4	1.9E-4	1.2E-4
Ar^{41}	1.4E-4	3.6E-5	4.2E-5	1.1E-5
Ne^{23}	1.1E-5	1.1E-5	*	*
50 H^3	6.8E-6	6.8E-6	5.3E-8	5.3E-8
Total	0.12	5.0E-2	1.9E-3	8.7E-4

*<<E-10

49 **After 30 years' operation

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TABLE 11.3-12

ANNUAL ACTIVITY RELEASE RATES FOR THE EXPECTED SERVICE CONDITION

	Radionuclide	Main RCB H&V Exhaust (Ci/year)	RSB H&V Exhaust (Ci/year)	Intermediate Bay Leakage (Ci/year)	Total Release (Ci/year)
	Xe^{131m}	4.8E-4	2.7E-4	0	7.5E-4
	Xe^{133m}	1.6E-2	2.9E-31	0	1.6E-2
49	Xe^{133}	0.27	1.1E-10	0	0.27
	Xe^{135m}	6.9E-2	+	0	6.9E-2
49	Xe^{135}	1.2	+	0	1.2
	Xe^{138}	0.11	+	0	0.11
	Kr^{83m}	4.0E-2	3.7E-5	0	4.0E-2
	Kr^{85m}	9.9E-2	0.67	0	0.77
	Kr^{85}	8.9E-6	5.2E+2	0	5.2E+2
	Kr^{87}	0.11	8.5E-8	0	0.11
49	Kr^{88}	0.19	3.0E-2	0	0.22
	Ar^{39}	2.9	85	0	88
49	Ar^{41}	8.4E-2	7.7	0	7.8
	Ne^{23}	2.0	+	0	2.0
	H^3++	5.5E-5	7.1E-3	5.8E-2	6.5E-2
50 49	Totals	6.9	6.1E+2	5.8E-2	6.2E+2

\+ E-45

++ BOP Tritium Release (0.9 Ci/yr) from T-G Building Exhaust not included.

Also, allowance for 2 weeks per year bypass of the oxidizer unit (amounts to 2.7E-2 curies of tritium exhausted to the RSB H&V exhaust)

49 is not included.

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TABLE 11.3-13

RADIONUCLIDE CONCENTRATIONS IN HEAD ACCESS AREA

Isotope	MPC* (μCi/cc)	Design		Expected	
		Concentration (μCi/cc)	Concentration ÷ MPC	Concentration (μCi/cc)	Concentration ÷ MPC
Xe^{131m}	2E-5	2.7E-11	1.3E-6	2.7E-12	1.3E-7
Xe^{133m}	1E-5	8.6E-10	8.6E-5	8.6E-11	8.6E-6
Xe^{133}	1E-5	1.5E-8	1.5E-3	1.5E-9	1.5E-4
Xe^{135m}	1E-6	3.9E-9	3.9E-3	3.9E-11	3.9E-4
Xe^{135}	4E-6	6.7E-8	2.0E-2	6.7E-9	2.0E-3
Xe^{138}	1E-6	6.4E-9	6.4E-3	6.4E-10	6.4E-4
Kr^{83m}	1E-6	2.3E-9	2.3E-3	2.4E-10	2.3E-4
Kr^{85m}	6E-6	5.3E-9	8.8E-4	5.3E-10	8.8E-5
Kr^{85}	1E-5	5.0E-13	5.0E-8	5.0E-14	5.0E-9
Kr^{87}	1E-6	6.0E-9	6.0E-3	6.0E-10	6.0E-4
Kr^{88}	1E-6	1.0E-8	1.0E-2	1.0E-10	1.0E-3
Ar^{39}	8E-6	1.6E-8	2.0E-3	1.6E-8	2.0E-3
Ar^{41}	2E-6	4.7E-10	2.4E-4	4.7E-10	2.4E-4
Ne^{23}**	1E-6	0	0	0	0
H^3	5E-6	3.1E-13	6.2E-8	3.1E-13	6.2E-8
Totals		1.3E-7	5.3E-2	2.7E-8	7.3E-3

49

*10CFR20, Appendix B, Table I (restricted area), except for Ar39, which has been scaled to Kr85.

** Ne^{23} is not included because a minimum period of 5 minutes will occur for leakage from cover gas to head access area. Thus Ne^{23} will not be present in the head access area. Decay during transit is not considered in evaluating any other isotope.

Head access area flow rate = 12,000 scfm

TABLE 11.3-16

TRITIUM CONCENTRATION AT SITE BOUNDARY
FROM T. G. BUILDING EXHAUST

	MPC* (µCi/cc)	Concentration (µCi/cc)	Concentration ÷ MPC
49	2E-7	4.1E-12	2.1E-5

TABLE 11.3-16a

TRITIUM CONCENTRATION AT SITE BOUNDARY
FROM IB EXHAUST

	MPC* (µCi/cc)	Concentration (µCi/cc)	Concentration ÷ MPC
49	2E-7	1.0E-12	5.0E-6

*10 CFR 20, Appendix B, Table II (unrestricted area).

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TABLE 11.3-17

DESIGN PARAMETERS OF RAPS AND CAPS PROCESS VESSELS

Margin	Items	Number Required	+, ** Design Code	Seismic Category	Design Pressure (psig)	Normal Operating Pressure (psig)	Design Temperature (°F)	Operating Temperature (°F)	Volume (scf)	Capacity at Operating Pressure and Temperature (scf)	Materials of Construction	Margin
	Storage Vessels, Recycle Argon	2	III-2	I	200	35	250	80 to 120	720 (total)	2200	Carbon Steel	
	RAPS Cryogenic Distillation Vessel	1	III-3	I	-14.7, 200	32	-320	-282	3.6 net	58	Stainless Steel	
	RAPS Vacuum Vessel	1	III-3	I	-14.7, 200	-7 to -2	250	120	261	125 to 206	Carbon Steel	
	RAPS Surge Vessel	1	III-3	I	-14.7, 200	103	250	120	500	3600	Carbon Steel	
	RAPS Storage Vessel, Noble Gas	1	III-3	I	-14.7, 200	35	250	70	260	880	Carbon Steel	
6	CAPS Charcoal Bed Vessels	2	III-3	I	-14.7, 200	34	-320	-134	64 (total)	DNA*	Stainless Steel	
	CAPS Vacuum Vessel	1	III-3	I	-14.7, 200	-7 to -2	250	120	260	124 to 204	Carbon Steel	
50 49	CAPS Surge Vessel	1	III-3	I	-14.7, 200	35 to 135	250	70 to 120	698	2360	Carbon Steel	9

*Does not apply because of adsorption variable
**ASME Section III, Class 3 - III-3
+Design Code listed may be higher for reasons other than safety
++Saturation Temperature at normal operating pressure

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TABLE 11.3-18

EXTERNAL DOSES* AT SITE BOUNDARY TO AN INDIVIDUAL VIA GASEOUS EFFLUENTS FROM CRBRP DESIGN RELEASE POINTS

	Total Skin (mrem/yr)	Whole Body Gamma (mrem/yr)
49 Main RCB H&V Exhaust	0.25	0.13
CAPS RSB H&V Exhaust	4.2	5.3E-1
Intermediate Bay Leakage	4.1E-6	0
50 T-G Building Exhaust	1.7E-6	0
49 41 Total (mrem/yr)	4.5	0.66

*Design condition

41

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TABLE 11.3-19

RELEASE POINT DESCRIPTION OF INTERNAL WHOLE BODY DOSES DUE TO CRBRP GASECUS EFLUENTS

	Ingestion of Beef	Inhalation	Ingestion of Vegetables	Ingestion of Milk	Ingestion of Water
A. Individual (mrem/yr)					
Main RCB H&V Exhaust	1.4E-6	1.5E-7	5.4E-7	7.7E-6	9.9E-8
CAPS RSB H&V Exhaust*	8.4E-4	9.5E-5	3.3E-4	4.7E-3	6.1E-5
Intermediate Bay Leakage	1.5E-3	1.6E-4	5.6E-4	8.4E-3	1.0E-4
T-G Building H&V Exhaust	5.8E-4	6.3E-5	2.2E-4	3.3E-3	4.1E-5
Total	2.9E-3	3.2E-4	1.1E-3	1.6E-2	2.0E-4
B. Population. 50-mile radius (man-rem/yr)	2.2E-3	3.7E-4	3.7E-4	1.3E-2	1.1E-3

49 *Includes two weeks oxidizer bypass.

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TABLE 11.3-20

EFFLUENT RELEASE POINTS DESIGN DATA
(Continued)

Release Point		Release Point Elevation	Distance From RCB	Degree From Plant N-S Axis	Discharge Flow Rate (CFM)	Discharge Velocity (Ft/min.)	Discharge Temperature (°F)	Size & Shape of Discharge Orifice
Reactor Containment/ Confinement Building								
27 | 49 | 41 | 5a	RCB H&V and Annulus Filtration System Exhaust	983'-0"	0	0	14,000	1,985	9-97	3' diameter pipe
27 | 20	Cooling Exhaust	890'-0"	0	0	400,000	910	55-430	Annular area 440 $Ft.^2$
36 27 | 21	RSB H&V and Containment Cleanup System Exhaust	983'-0"	0	0	18,000	2,550	55-200	3' diameter pipe

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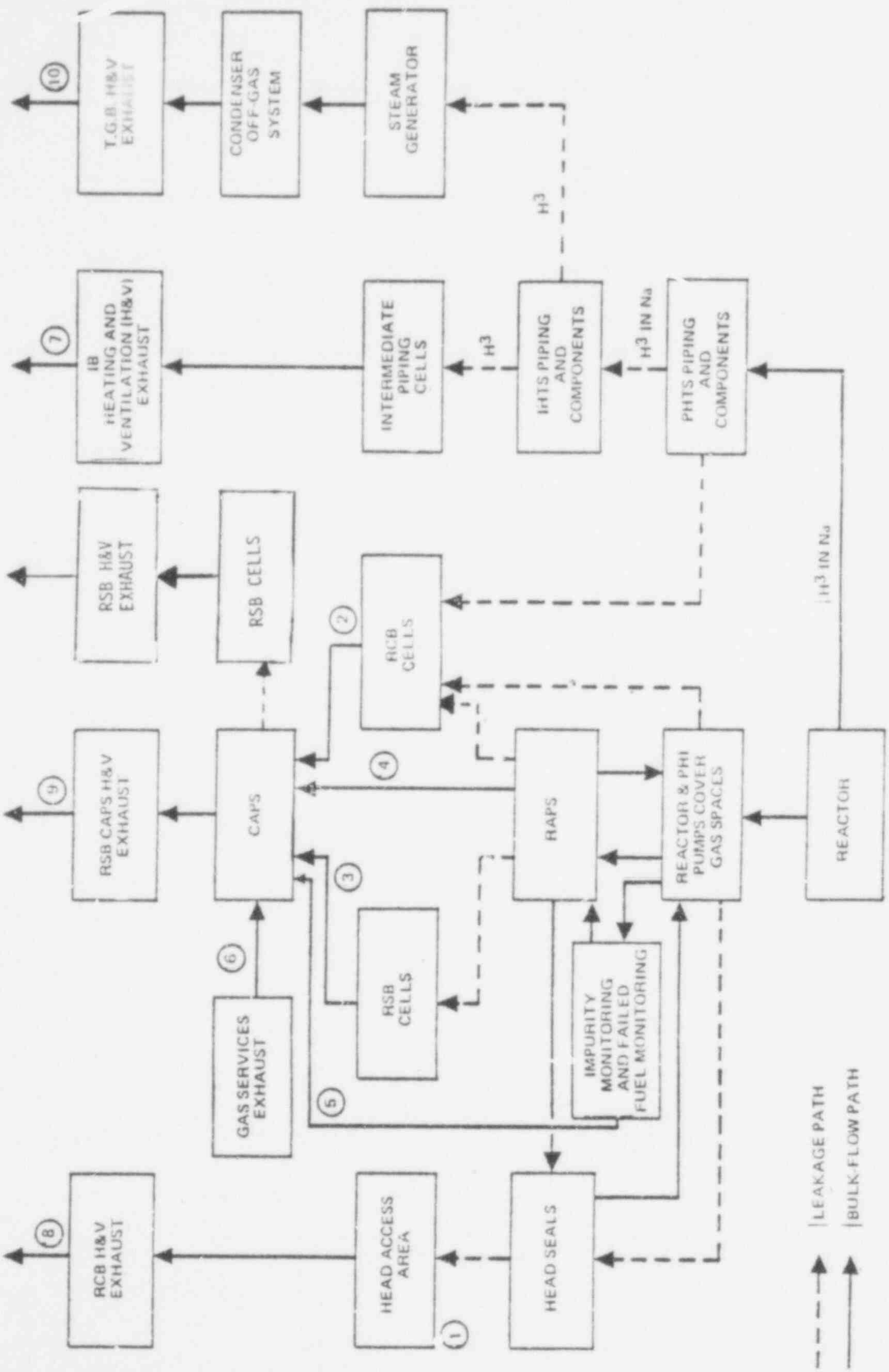

Figure 11.3-1. Radioactive Gas Flow Paths

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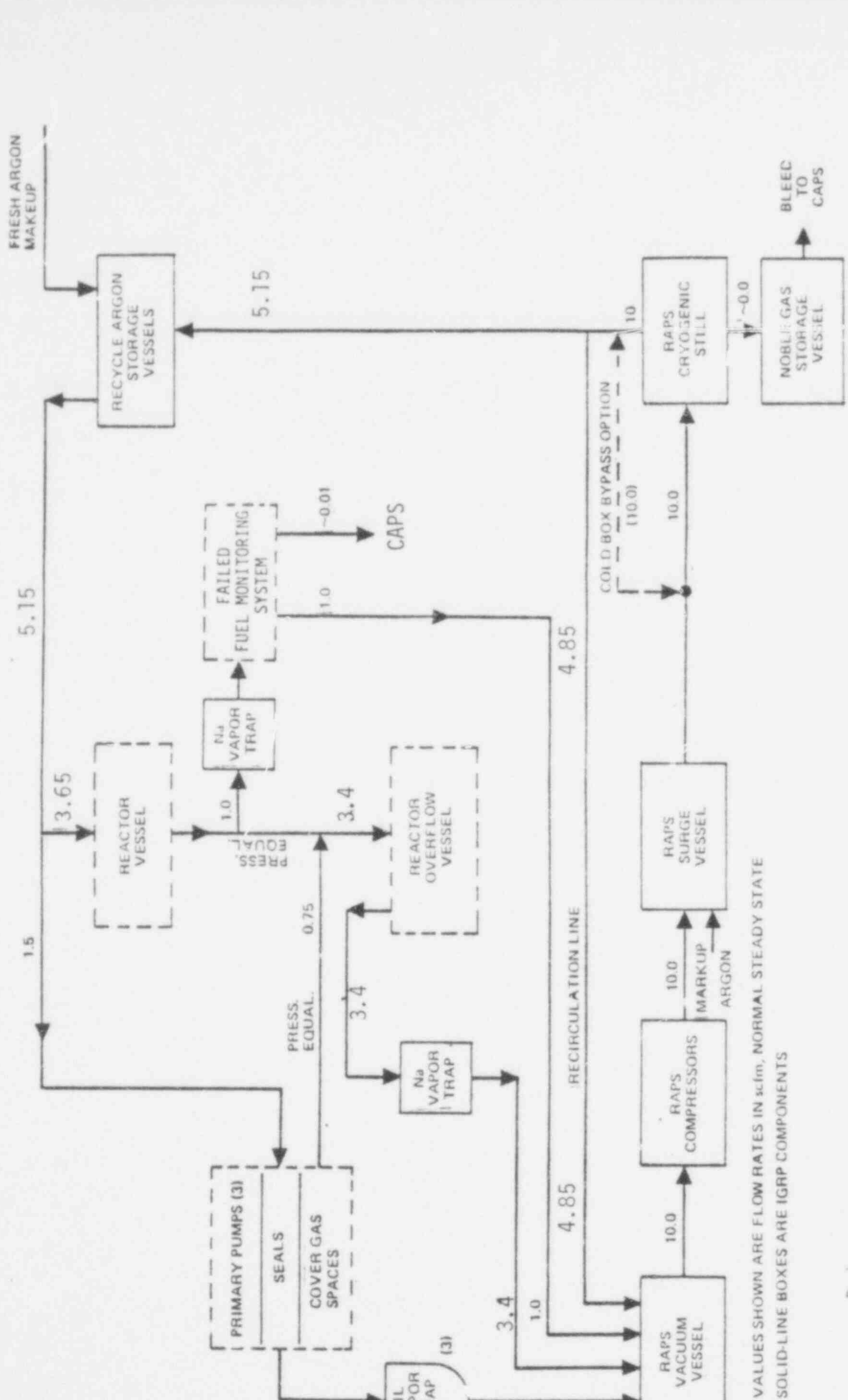

Figure 11.3-2. Schematic Diagram of the Recycle Argon Circuit

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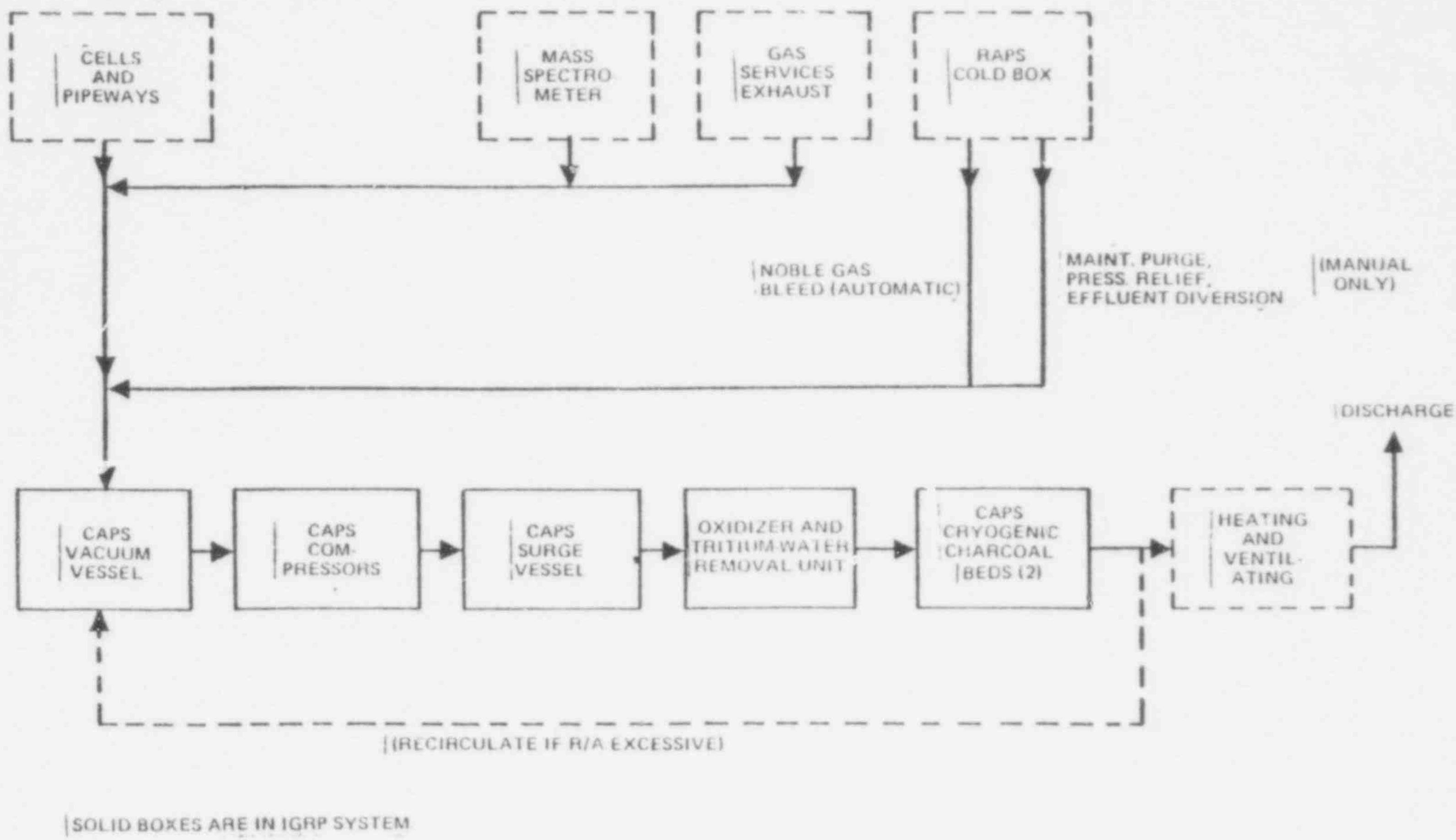

Figure 11.3-3. Schematic Diagram of CAPS Process Flow

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50 Figures 11.3-10 thru 11.3-13 have been deleted.

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three gas monitors. Their output will also be provided to the PPS for
initiation of containment isolation when a preset radiation level is
6 reached by two of the three detectors. A particulate filter will be in-
stalled in line with the gas chamber to prevent buildup in the sample
chamber.

The monitoring system will be designed to comply with IEEE 279-
1971. The overall containment isolation system design and protection
logic is discussed in Section 7.3. Figure 12.2-1 shows a typical block
6 diagram of these channels and Figure 7.3-1 shows the trip logic configur-
ation.

11.4.2.2.3 Building Ventilation Exhaust Monitors

The building exhaust plenums from which potentially radioactive
plant gaseous release may emanate are: one in the Intermediate Bay, three
49 in the Steam generator Building, one in the Plant Service Building, twelve
18 in the Turbine Generator Building, one in the Radwaste Building, Reactor
Service Building exhaust(s), the common RCB H&V and Annulus Pressure
Maintenance and Filtration System exhaust and the Containment Cleanup
System and Annulus Air Cooling System exhaust. Continuous monitoring will
be performed at those exhausts which could conceivably undergo a signifi-
cant increase in detectable levels in radioactivity. The remaining
exhausts will be sampled periodically. 28

18 The exhaust plenum located in the IB receives ventilation exhaust
air from the Intermediate Bay area. A continuous air monitor (CAM) will
be provided to detect particulate, radioiodine and gaseous activity in the
effluent stream. The air sample will be obtained isokinetically from the
exhaust, on a continuous basis. The operation of the three-channel CAM
unit is described in Section 12.2.4.2.1.

The RSB exhaust(s) will be continuously monitored for radioactivity
18 releases.

49 The three SGB exhausts receive ventilation exhaust from the indi-
vidual steam generator cells, and IHS cells in the Intermediate Bay. Each
exhaust will be sampled for tritium activity using silica-gel dessicants;
and analysis of samples will be performed by liquid scintilation techniques.
The exhaust samples will be obtained isokinetically, and flow through the
silica-gel column will be maintained constant by a regulated pump assembly.

49 The exhaust fans in the TGB receive ventilation from the various
Turbine Generator Building operating areas. These exhausts points will
also be sampled for tritium activity as described above.

49 The exhaust in the PSB receives ventilation from the Hot Labora-
6 tory/Counting Room. Samples will be collected isokinetically by a parti-
culate (and iodine, if required) filter and analyzed for isotopic content

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6 in the Counting Room.

The Containment Cleanup System and Annulus Air Cooling System exhausts at the top of the the Reactor Confinement Building. Particulate, radioiodine, gaseous and plutonium activity in the effluent stream will be monitored continuously. The common RCB H&V and Annulus Pressure Maintenance and Filtration System exhaust will be continuously monitored for particulates,
27 radiogases, and radioiodine in the effluent stream.

The reporting of effluent radioactivity released from the CRBRP will be consistent with the guidelines established in Regulatory Guide 1.21. This reporting will be based upon the results of Counting Room analysis of effluent samples obtained at each location listed above.
6

11.4.2.2.4 Condenser Vacuum Pump Exhaust and Deaerator Continuous
49 Vents Tritium Sampler

A continuous gas sample will be withdrawn from the condenser vacuum
6 49 pump air and deaerator exhaust into a silica-gel dessicant column to enable a determination of tritium activity in order to indicate unacceptable tritium diffusion in the steam generators. The sample will be analyzed using scintillation techniques.

11.4.2.2.5 Control Room Inlet Air Monitors

The main and remote control room air intakes will be continuously monitored for gaseous radioactivity to determine which intake should be used during the Control Room isolation condition. Details concerning the sequence of operation during Control Room isolation are given in Section 9.6.1.3.4.B. A three channel (particulate/radioiodine/radiogas) CAM will be installed downstream of the parallel (redundant) HVAC make-up air filter trains to check on the performance of these high-efficiency HEPA filter trains. A detailed description of the operation of each of these three CAM units is given in Section 12.2.4.2.1.
6

11.4.2.2.6 Inerted Cell Atmosphere Monitors

The capability for monitoring the atmosphere of each individual inerted cell for high radioactivity will be accomplished by three methods. One method is the sequential sampling of groups of cells with three on-line gas monitors as described in 3.A.1.3. Each monitor shall have a trip signal determined by the process system to initiate activation of cell purging equipment. In addition, mobile particulate and gas monitors are provided to sample any individual inerted cells atmosphere, as described in 12.2.4.

Finally to provide a sensitive method of sodium leak detection, particulate monitors are provided for continuous monitoring of selected inerted cells within the RCB containing components contacting radioactive sodium. These monitors will alarm for activated sodium present in the cells atmosphere. The individual inerted cells that are continuously monitored for sodium leak detection will be listed in the FSAR.

11.4.2.2.7 RAPS and CAPS Monitoring

Gas monitors will be provided for the Radioactive Argon Processing System (RAPS) and Cell Atmosphere Processing System (CAPS). These monitors will be located at the inlet to these systems for controlling the rate of radioactivity input. Monitors will also be located at the output of these systems to ascertain that the radionuclide activity of the processed gas is within limits for reuse in RAPS or within 10CFR20 limits for those gases exhausted to the H&V system by CAPS.

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Table 11.4-1 (Cont'd.) Process and Effluent Sampling and Monitoring

Description	Bldg.	Elev.	Continuous Monitoring or Sample	Detector Type	μCi/cc Sensitivity	Expected Concentrations or Dose Rates	Quantity Measured	Remarks
SSST Effluent Activity	RCB	981	Continuous					
Particulate				β-Scintillation	10^{-10}(137 Cs	TBD	Gross Conc.	
Radioiodine				γ-Scintillation	10^{-10}(131 I)	TBD	Gross Conc.	
Gaseous				β-Scintillation	10^{-6} (85 Kr)	TBD	Gross Conc.	
TMBDB Effluent Activity	RCB	981	Continuous					
Particulate				β-Scintillation	10^{-10}(137 Cs)	TBD	Gross Conc.	
Radioiodine				γ-Scintillation	10^{-10}(131 I)	TBD	Gross Conc.	
Gaseous				β-Scintillation	10^{-6} (85 Kr)	TBD	Gross Conc.	
TMBDB Effluent Plutonium Activity	RCB	981	Continuous					
Plutonium				α-Scintillation	10^{-12}(239 Pu)	TBD	Gross Conc.	Delayed detection techniques employed

50 |49

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Table 11.4-1 (Cont'd.) Process and Effluent Sampling and Monitoring

Description	Bldg.	Elev.	Continuous Monitoring or Sample	Detector Type	μCi/cc Sensitivity	Expected Concentrations or Dose Rates	Quantity Measured	Remarks
RAPS and CAPS Process Monitoring Gaseous	RSB	765'	Continuous	β Scintillation	10^{-6} Kr-85	See Section 11.3.2.3	Gross Conc.	
Plant Discharge Effluent Liquid Composite	RSB - downstream of low Activity System	-	Sample	-	-	See Section 11.2.5	Isotropic Conc.	Sample analyzed in counting room using proportional and liquid scintillation counters and γ-ray spectroscopy system.
Inerted Cell Atmosphere Monitors	*							
6 Radwaste Disposal	*							

LEGEND

RCB - Reactor Containment Building
RSB - Reactor Service Building
IB - Intermediate Bay
TG - Turbine Generator Building
CRB - Control Room Building

6 * Information will be provided in the FSAR

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TABLE 11.5-1A (CONT'D)

46 14*. Using remote tools from the Maintenance Platform, place and operate automatic welding equipment to cap the cold trap crystalizer inlet and outlet sodium pipes.

46 15. Shut off NaK coolant flow to the cold trap crystallizer and drain the NaK to the lowest drain point on the crystalizer.

46 16. Using remote tools from the Maintenance Platform, cut and cap the NaK coolant lines and complete preparation for removal of the crystalizer.

46 17. Close Floor Valve and remove Maintenance Platform.

46 18. Install Cold Trap Removal Cask to the Floor Valve.

46 19. Open Floor Valve and remove Clod Trap crystalizer into Cold Trap Removal Cask.

46 20. Close Floor Valve and remove cask from Floor Valve containing cold trap crystalizer.

+To be preceded by appropriate decay time.

*This step can possibly be accomplished following Step 14., depending on pipe accessibility.

1

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TABLE 11.5-2

INVENTORY OF DESIGN ANNUAL ACTIVITY OF SHIPPED WASTE

Isotope	Half-Life	Solidified Liq Waste (Ci)	Sodium Contaminated Solids (Ci)
H-3	12.3Y	3.31 (-1)	1.8 (-4)
Na-22	2.6Y	6.24 (-1)	2.9 (-1)
Na-24	15H	7.76 (-2)	6.6 (-2)
Cr-51	28D	6.51 (-1)	1.0 (0)
Mn-54	312D	1.61 (-1)	3.6 (-1)
Co-58	71D	9.19 (0)	4.4 (1)
Co-60	5.2Y	4.24 (0)	6.3 (1)
Fe-59	45D	6.83 (-2)	4.0 (-1)
Sr-89	51D	6.58 (-1)	4.0 (0)
Sr-90	28.8Y	4.73 (-1)	1.7 (0)
Y-90	64.1H	4.73 (-1)	1.7 (0)
Y-91	58D	1.93 (-1)	1.2 (0)
Nb-95	35D	1.53 (1)	2.2 (0)
Zr-95	64D	1.53 (1)	2.2 (0)
Mo-99	67D	4.35 (-1)	-1
Ru-103	40D	1.49 (1)	3.0 (0)
Ru-106	1Y	2.88 (1)	2.5 (0)
Rh-106	2.2H	2.88 (1)	2.5 (0)
Ag-111	7.5D	1.27 (-2)	-
Sb-125	2.7Y	8.63 (-2)	1.8 (2)
Te-127m	109D	1.35 (0)	-
Te-127	9.35H	1.14 (0)	-
Te-129m	34D	4.05 (0)	2.5 (1)
Te-129	70M	4.05 (0)	2.5 (1)
Te-132	78H	6.41 (-4)	8.8 (0)
I-131	8.1D	1.37 (0)	5.0 (1)
I-132	2.3H	7.11 (-1)	8.8 (0)
Cs-134	2.1D	7.21 (-1)	1.4 (2)
Cs-136	13D	2.02 (0)	2.3 (1)
Cs-137	30Y	1.55 (1)	8.5 (3)
Ba-140	12.8D	4.18 (0)	8.0 (-1)
La-140	40H	4.18 (0)	8.0 (-1)
Ce-141	32.5D	4.24 (-1)	2.7 (0)
Ce-143	32.5D	2.28 (-1)	-
Pr-143	13.7D	2.28 (-1)	1.4 (0)
Ce-144	285D	3.11 (-1)	1.9 (0)
Pr-144	17M	3.11 (-1)	1.9 (0)
Nd-147	11.1D	9.57 (-2)	-
Pm-147	2.7D	1.74 (-1)	1.5 (0)
Eu-155	1.8Y	1.80 (-2)	1.64 (-1)
Ta-182	115D	2.35 (-2)	8.7 (-1)
Pu-238	86Y	3.86 (-3)	4.4 (-1)
49 Pu-239	2.0(4)Y	1.03 (-3)	1.1 (-1)

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TABLE 12.1-8

INVENTORY OF RADIOACTIVE PRIMARY SODIUM
IN THE PPTI+ AND PSSP++

	Isotope	Inventory (Ci) PPTI	PSSP
	Na^{24}	103	121
	Na^{22}	1.2×10^{-2}	1.4×10^{-2}
	Cs^{137}	0.29	0.36
	Cs^{136}	6.1×10^{-2}	7.1×10^{-2}
	Cs^{134}	7.8×10^{-3}	9.3×10^{-3}
	Sb^{125}	1.7×10^{-3}	1.9×10^{-3}
	I^{131}	0.17	0.20
	Te^{132}	0.013	0.015
6	I^{132}	0.12	0.13
	Te^{129m}	2.5×10^{-3}	3.0×10^{-3}
	Te^{129}	2.5×10^{-3}	3.0×10^{-3}
	Sr^{89}	3.8×10^{-4}	4.6×10^{-4}
	Sr^{90}	2.4×10^{-4}	2.8×10^{-4}
	Y^{90}	2.4×10^{-4}	2.8×10^{-4}
	Y^{91}	1.1×10^{-4}	1.3×10^{-4}
	Zr^{95}	2.1×10^{-4}	2.4×10^{-4}
	Nb^{95}	2.1×10^{-4}	2.4×10^{-4}
	Ru^{103}	2.9×10^{-4}	3.4×10^{-4}
	Ru^{106}	2.1×10^{-4}	2.4×10^{-4}
	Rh^{106}	2.1×10^{-4}	2.4×10^{-4}
	Ba^{140}	2.3×10^{-4}	2.7×10^{-4}
	La^{140}	2.3×10^{-4}	2.7×10^{-4}
	Ce^{141}	2.7×10^{-4}	3.1×10^{-4}
	Ce^{144}	1.6×10^{-4}	1.8×10^{-4}
	Pr^{144}	1.6×10^{-4}	1.8×10^{-4}
	Pr^{143}	1.9×10^{-4}	2.3×10^{-4}
	Nd^{147}	9.0×10^{-5}	1.1×10^{-4}
	Pm^{147}	9.0×10^{-5}	1.1×10^{-4}
	Co^{60}	8.0×10^{-6}	9.5×10^{-7}
	Co^{58}	1.5×10^{-5}	1.8×10^{-6}
	Mn^{54}	5.9×10^{-5}	7.0×10^{-6}
	Pu^{238}	5.7×10^{-5}	6.8×10^{-5}
6	Pu^{239}	1.5×10^{-5}	1.8×10^{-5}
	Pu^{240}	2.2×10^{-5}	2.3×10^{-5}
	Pu^{241}	1.7×10^{-3}	2.2×10^{-3}
	Pu^{242}	4.1×10^{-8}	5.2×10^{-8}
49	H^{3}	8.2×10^{-3}	9.7×10^{-3}
50	Tc	103.09	121.8

\+ Primary Plugging Temperature Indicator
++ Primary Sodium Sample Package

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50 TABLE 12.1-34A PRIMARY PUMP INTERMEDIATE LEVEL (IALL)
WASTE SYSTEM RADIOISOTOPE INVENTORY

	Isotope	Inventory (Curies) 10 Days After Reactor Shutdown
	Cr^{51}	3.92
	Mn^{54}	119.3
	Fe^{59}	.43
	Co^{58}	63.2
	Co^{60}	31.7
	Sr^{89}	78.8
	Sr^{90}	57.7
	Y^{90}	57.7
	Y^{91}	22.9
	Zr^{95}	43.2
	Nb^{95}	43.2
	Mo^{99}	4.8
	Ru^{103}	57.2
	Ru^{106}	46.3
	Rh^{106}	46.3
	Ag^{111}	1.5
	Te^{127}	157.6
	Te^{127m}	157.6
	Te^{129m}	475.5
	Te^{129}	475.5
	Te^{132}	338.8
49	I^{131}	0.23
	I^{132}	338.8

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TABLE 12.1-34A (Continued)

Isotope	Inventory (Curies) 10 Days After Reactor Shutdown
Cs^{134}	0.085
Cs^{136}	0.42
Cs^{137}	3.2
Ba^{140}	31.2
La^{140}	31.2
Ce^{141}	51.4
Ce^{143}	23.0
Pr^{143}	23.0
Ce^{144}	36.7
Pr^{144}	36.7
Nd^{147}	11.1
Pm^{147}	20.8
Pm^{149}	.61
Eu^{155}	2.7
Eu^{156}	1.02
Ta^{182}	4.13
Pu^{238}	.47
Pu^{239}	.12
Pu^{240}	.16
Pu^{241}	14.1
Pu^{242}	.0003
Np^{238}	.00001
Np^{239}	.07

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50 TABLE 12.1-34 A(Continued)

	Isotope	Inventory (Curies) 10 Days After Reactor Shutdown
	Am^{241}	.13
	Am^{242m}	.005
	Am^{242}	2.3×10^{-8}
	Am^{243}	.002
	Cm^{242}	.095
	Cm^{243}	.001
49	Cm^{244}	.028

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TABLE 12.1-47

SOLID RADIOACTIVE WASTE RADWASTE DRUM
(SPENT RESINS - FILTER SLUDGE)
RADIOISOTOPE INVENTORY

ISOTOPE	INVENTORY (CURIES)	ISOTOPE	INVENTORY (CURIES)
Cr-51	2.31(-1)	Ce-141	2.99(-2)
Mn-54	7.00(0)	Ce-143	1.34(-1)
Fe-59	2.53(-2)	Pr-143	1.34(-1)
Co-58	3.72(0)	Ce-144	2.14(-1)
Co-60	1.86(0)	Pr-144	2.14(-1)
Sr-89	4.58(-2)	Nd-147	6.41(-2)
Sr-90	3.36(-1)	Pm-147	1.21(-1)
Y-90	3.36(-1)	Pm-149	3.55(-3)
Y-91	1.34(-1)	Eu-155	1.57(-2)
Zr-95	2.52(-1)	Eu-156	5.94(-3)
Nb-95	2.52(-1)	Ta-182	2.43(-1)
Mo-99	2.80(-2)	Pu-238	2.74(-3)
Ru-103	3.34(-1)	Pu-239	7.00(-4)
Ru-106	2.70(-1)	Pu-240	9.35(-4)
Rh-106	2.70 (-1)	Pu-241	8.17(-2)
Ag-111	8.76(-3)	Pu-242	1.75(-6)
Te-127	9.17(-1)	Np-238	8.76(-8)
Te-127m	9.17(-1)	Np-239	4.08(-4)
Te-129m	2.77(0)	Am-241	7.53(-4)
Te-129	2.77(0)	Am-242m	2.91(-5)
Te-132	1.98(0)	Am-242	1.34(-10)
I-131	1.34(-3)	Am-243	1.16(-5)
I-132	1.98(0)	Cm-242	1.42(-4)
Cs-134	4.94(-3)	Cm-243	5.84(-6)
Cs-136	2.45(-3)	Cm-244	1.63(-4)
Cs-137	1.86(-2)		
Ba-140	1.82(-1)		
49 La-140	1.82(-1)		

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TOTAL 28.4

Amend. 49
April 1979

TABLE 12.1-48

AREA MONITOR DESCRIPTIONS

	Location: Area	Bldg.	Cell No.	Area Design Dose Rate mR/hr	Instrument Range mR/hr	Basis for Location
49	I&C Cubicle El. 824'-3"	RCB	162	0.2	$0.01-10^3$	1.
49	I&C Cubicle El. 824'-3"	RCB	163	0.2	$0.01-10^3$	1.
50 49	I&C Cubicle El. 824'-3"	RCB	164	0.2	$0.01-10^7$	1.,3.,4.
	Refueling Hatch El. 816'	RCB	-	0.2	$0.01-10^7$	1.,3.,4.
	Airlock El. 816'	RCB	-	0.2	$0.01-10^3$	1.
	Operating El. 794'	RCB	105V	2.0	$0.1-10^4$	1.,2.
	Operating El. 780'	RCB	105U	2.0	$0.1-10^4$	1.,2.
	Operating El. 766'	RCB	105M	2.0	$0.1-10^4$	1.,2.
	Operating El. 766'	RCB	105N	10.	$0.1-10^4$	1.,2.,5.
	Operating El. 766'	RCB	105R	10.	$0.1-10^4$	1.,2.,5.
	Operating El. 766'	RCB	105T	10.	$0.1-10^4$	1.,2.,5.
	Operating El. 766'	RCB	105P	10.	$0.1-10^4$	1.,2.,5.
	Operating El. 766'	RCB	105Q	2.0	$0.1-10^4$	1.,2.
1	Operating El. 733'	RCB	105A	2.0	$0.1-10^4$	1.,2.

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TABLE 12.3-3

ESTIMATED MAN HOURS OF ACCESS TO RADIATION AREAS
DURING NORMAL OPERATION AND OPERATIONAL OCCURRENCES

Operating Area	Man Hours/Quarter
1. Reactor Containment Building	
a. Operating Floor and Adjacent Balconies (Zone 1)	1275
b. Cells Below the Operating Floor in NE, SE, and SW Corners and Cell 152 (Zone 2)	950
c. Pump Wells (Zone 2)	580
d. Head Access Area (Zone 2)	220
e. All remaining Accessible Cells (Zone 3)	365
Total Reactor Containment Building	3390
2. Reactor Service Building	
a. Operating Floor, Balconies, and Refueling Cask Shaft and Corridor (Zone 1)	7560
b. OHRS Cells (Zone 1)	70
c. Access Corridors and Adjacent Sampling and Value Gallery Cells Below the Operating Floor (Zone 2)	2490
d. Fuel Handling Cell (Zone 1)	5720
e. Radwaste Processing Area (Zones 1 & 2)	1315
Total Reactor Service Building	17155
3. Provision for Required Health Physics Coverage	3640
Total man hours in accessible areas of Reactor Containment and Reactor Service Building	24185 man hours/quarter

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TABLE 12.3-4

ESTIMATED MAN HOURS OF ACCESS TO
ACCESSIBLE RADIATION ZONES

	Zone	Man Hours/Quarter
	Zone I	17850
	Zone II	5740
	Zone III	595
49	Total	24185

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TABLE 12.3-5

PERSONNEL PROTECTION MONITORS - AREA MONITORS

LOCATION BLDG.	LOCATION ELEV.	LOCATION CELL	AREA AND/OR PROCESS MONITORED	MONITOR TYPE	DETECTOR TYPE	DETECTOR RANGE (mR/hr)	DETECTOR BACKGROUND (mR/hr)	ALARM SETPT.(s) µCi/cc	MONITOR** OUTPUT	BASIS FOR* LOCATION
RCB	824'	162	I&C Cubicle	Direct Gamma	TBD	$.01 - 10^3$	0.2	0.2, 2.0	A	1.
RCB	824'	163	I&C Cubicle	Direct Gamma	TBD	$.01 - 10^3$	0.2	0.2, 2.0	A	1.
RCB	824'	164	I&C Cubicle	Direct Gamma	TBD	$.01 - 10^7$	0.2	0.2, 2.0	A	1., 4.
RCB	816'	-	Refueling Hatch	Direct Gamma	TBD	$.01 - 10^7$	0.2	0.2, 2.0	A	1., 3., 4.
RCB	816'	-	Airlock	Direct Gamma	TBD	$.01 - 10^3$	0.2	0.2, 2.0	A	1.
RCB	794'	105V	Operating	Direct Gamma	TBD	$0.1 - 10^4$	2.0	2.0, 10	A	1., 2.
RCB	780'	105U	Operating	Direct Gamma	TBD	$0.1 - 10^4$	2.0	2.0, 10	A	1., 2.
RCB	766'	105M	Operating	Direct Gamma	TBD	$0.1 - 10^4$	2.0	2.0, 10	A	1., 2.
RCB	766'	105N	Operating	Direct Gamma	TBD	$0.1 - 10^4$	10.	10, 25	A	1., 2., 5.
RCB	766'	105R	Operating	Direct Gamma	TBD	$0.1 - 10^4$	10.	10, 25	A	1., 2., 5.
RCB	766'	105T	Operating	Direct Gamma	TBD	$0.1 - 10^4$	10.	10, 25	A	1., 2., 5.
RCB	766'	105P	Operating	Direct Gamma	TBD	$0.1 - 10^4$	10.	10, 25	A	1., 2., 5.
RCB	766'	105Q	Operating	Direct Gamma	TBD	$0.1 - 10^4$	2.0	2.0, 10	A	1., 2.
RCB	766'	105A	Operating	Direct Gamma	TBD	$0.1 - 10^4$	2.0	2.0, 10	A	1., 2.
RSB	842'	363	Refuel, Comm Cen	Direct Gamma	TBD	$.01 - 10^3$	0.2	0.2, 2.0	B	1., 3.
RCB	802'	151	Head Access Area	Direct Gamma	TBD	$0.1 - 10^7$	0.2	100, 2.5×10^4	A	1., 2., 4.
RSB	816'	364A	Operating Floor	Direct Gamma	TBD	$.01 - 10^3$	0.2	0.2, 2	B	1., 2., 3.
RSB	816'	364A	Operating Floor	Direct Gamma	TBD	$.01 - 10^3$	0.2	0.2, 10	B	1., 3.
RSB	790'	379	Operating	Direct Gamma	TBD	$0.1 - 10^4$	2.0	2.0, 10	A	1., 2.
RSB	790'	355A	Operating	Direct Gamma	TBD	$0.1 - 10^4$	2.0	0.2, 10	B	1., 2.

TABLE 12.3-5 (CONTINUED)

LOCATION			AREA AND/OR PROCESS MONITORED	MONITOR TYPE	DETECTOR			ALARM SETPT.(s) μCi/cc	MONITOR** OUTPUT	BASIS FOR* LOCATION
BLDG.	ELEV.	CELL			TYPE	RANGE (mP/hr)	BACKGROUND (mR/hr)			
RSB	781'	338	Fuel Handling Cell	Direct Gamma	TBD	TBD	TBD	TBD	B	3.
RSB	779'	339A	Operating	Direct Gamma	TBD	$.01 - 10^3$	0.2	0.2, 2.0	B	1., 2., 3.
RSB	749'	313F	Spent Fuel Cask Corridor	Direct Gamma	TBD	$.01 - 10^3$	0.2	0.2, 2.0	B	3.
RSB	765'	313A	Operating	Direct Gamma	TBD	$0.1 - 10^4$	2.0	2.0, 10.0	B	1., 2., 3.
RSB	733'	313B	Operating	Direct Gamma	TBD	$0.1 - 10^4$	2.0	2.0, 10.0	B	1., 2., 3.
SGB	816'	262	Operating	Direct Gamma	TBD	$0.1 - 10^7$	0.2	100, 2.5×10^4	B	1., 4.
SGB	794'	253	Emer Airlock	Direct Gamma	TBD	$0.1 - 10^7$	0.2		A	1., 4.
CB	816'	431	Control Room	Direct Gamma	TBD	$0.1 - 10^3$	-	0.2, 25	A	1.
PSB	816'	-	Hot Lab	Direct Gamma	TBD	$.01 - 10^3$	-	0.2, 25	A	1.

LEGEND

RCB - Reactor Containment Bldg.
RSB - Reactor Service Bldg.
SGB - Steam Generator Bldg.
CB - Control Bldg.
49 PSB - Plant Service Bldg.

*BASIS FOR LOCATION

1. Provide personnel protection in trafficked area
2. Monitor adjacent high radioactivity arch
3. Monitor refueling operations
4. High level reactor containment radiation monitor
5. Monitor PHTS fan-cooler unit cells

**MONITOR OUTPUT

A. Local and Control Room: Loss of signal indicator light, high level radiation alarm, high-high level radiation alarm, exposure meter (mR/hr), input to PDH & DS (System 91).

B. Local, Control Room and Refueling Communication Center: (same as above).

TABLE 12.3-6

PERSONNEL PROTECTION MONITORING - CONTINUOUS AIR MONITORS

LOCATION			AREA AND/OR PROCESS MONITORED	MONITOR TYPE	DETECTOR			ALARM SETPT.(s) μCi/cc	MONITOR OUTPUT (note 2)	REMARKS
BLDG.	ELEV.	CELL			TYPE	SENSITIVITY (μCi/cm)	BACKGROUND (mR/hr)			
RCB	816'	-	Operating Floor	Particulate	β SCINT	10^{-10} Cs-137	0.2	TBD	A	Mobile Monitor - Provides continuous monitoring of containment atmosphere
				Radioiodine	α SCINT	10^{-10} I -131	0.2	TBD		
				Gaseous	β SCINT	10^{-6} Kr-85	0.2	TBD		
RCB	816'	-	Operating Floor	Particulate	β SCINT	10^{-10} Cs-137	0.2	TBD	A	Mobile Monitor - can be used to monitor individual inerted cells
				Gaseous	β SCINT	10^{-6} Kr-85	0.2	TBD		
RCB	816'	-	Operating Floor	Alpha	α SCINT	10^{-12} Pu-239	0.2	TBD	A	Mobile Monitor - provides continuous monitoring of containment atmosphere
RCB	766'	105M	Operating Floor	Particulate	β SCINT	10^{-10} Cs-137	2.0	TBD	A	Mobile Monitor - can be used to monitor individual inerted cells
				Gaseous	β SCINT	10^{-6} Kr-85	2.0			
RSB	816'	364A	Operating Floor	Particulate	β SCINT	10^{-10} Cs-137	0.2	TBD	A	Mobile Monitor - can be used to monitor individual inerted cells
				Radioiodine	α SCINT	10^{-10} I-131	0.2	TBD		
				Gaseous	β SCINT	10^{-6} Kr-85	0.2	TBD		
RSB	816'	364A	Operating Floor	Alpha	α SCINT	10^{-12} Pu-239	0.2	TBD	A	Mobile Monitor - provided continuous monitoring of RSB Atmosphere
RSB	790'	355A	Operating Floor	Particulate	β SCINT	10^{-10} Cs-137	0.2	TBD	A	Mobile Monitor - can be used for post-accident of containment atmosphere
				Gaseous	β SCINT	10^{-6} Kr-85	0.2	TBD		
IB-SGB	816'	262	Operating Floor	Particulate	β SCINT	10^{-10} Cs-137	0.2	TBD	A	Mobile Monitor - can be used for post-accident monitoring of containment atmosphere
				Radioiodine	β SCINT	10^{-10} I-131	0.2	TBD		
				Gaseous	β SCINT	10^{-6} Kr-85	0.2	TBD		

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12.3-16

TABLE 12.3-6 (CONTINUED)

LOCATION			AREA AND/OR PROCESS MONITORED	MONITOR TYPE	DETECTOR			ALARM SETPT.(s) μCi/cc	MONITOR OUTPUT (note 2)	REMARKS
BLDG.	ELEV.	CELL			TYPE	SENSITIVITY (μCi/cm)	BACKGROUND (mR/hr)			
CB	816'	-	Control Room	Particulate	β SCINT	10^{-10} Cs-137	-	TBD	A	Mobile Monitor - Provides continuous monitoring of control room intake air
				Radioiodine	α SCINT	10^{-10} I -131	-	TBD		
				Gaseous	β SCINT	10^{-6} Kr-85	-	TBD		

LEGEND

RCB - Reactor Containment Bldg.

RSB - Reactor Service Building

IB-SGB - Intermediate Bay-Steam Generator Building

CB - Control Building

SGB - Steam Generator Building

SCINT- Scintillation

NOTE: Monitor Output

A. Monitor housing includes: loss-of-signal, high radiation, high-high radiation indicator lights, sample flow reading, countrate meter (per detection channel), multipoint strip-chart recorder, audible alarms.

B. Radiation Monitoring Panel (Control Room): loss-of-signal, high radiation, high radiation indicator lights, countrate meter (per detection channel). Alarm annunciation (loss-of-signal, high radiation, high-high radiation) provided by Plant Control (System 90). Permanent recording by PDH & DS (System 91).

C. Same provisions as B.

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CHAPTER 13.0 CONDUCT OF OPERATIONS

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CHAPTER 13.0 CONDUCT OF OPERATIONS

13.1 ORGANIZATIONAL STRUCTURE OF THE APPLICANT

Contract AT (49-18)-12 has been established to design, construct
and operate a Liquid Metal Fast Breeder Reactor demonstration plant. The
50 parties to the contract are the Department of Energy (DOE), the Tennessee 25
Valley Authority (TVA), the Commonwealth Edition Company (CE), and the
Project Management Corporation (PMC). The organizational structure
50 of the applicant (DOE, PMC, and TVA) is covered in Section 1.4. 25

TVA, as part of its lead role responsibility described in Section 1.4, will be responsible for the safe operation of the CRBRP.

13.1.1 Project Organization

13.1.1.1 Functions, Responsibilities, and Authorities of Project Participants

The functions, responsibilities, and authorities of Project participants are described in Sections 1.4 and 1.4.2. The qualification requirements of Project participants are described in Section 1.4.4. 25

13.1.1.2 Applicants' In-House Organization

This material is covered in Section 1.4.2.

13.1.1.3 Interrelationships with Contractors and Suppliers

This material is covered in Section 1.4.3.

13.1.1.4 Department of Energy (DOE) Participation

50 The participation of DOE in the CRBRP Project is described in
Section 1.4. In addition, DOE participates in R&D in support of the CRBRP
Project through its LMFBR base technology programs described in Section 1.5. 25

13.1.1.5 Technical Support for Operations

TVA's Office of Power will be responsible for carrying out the
operator role for the Clinch River Breeder Reactor Plant. Within the
50 TVA Office of Power, the Division of Power Production(P PROD) will be
responsible for the operation and maintenance of the CRBRP. The TVA
technical staff supporting the operation of the CRBRP will consist of
50 P PROD's home office staff in Chattanooga and also support from other
divisions and offices within TVA (Section 13.1.1.5.1). In addition,
50 technical support will be supplied by the technical staffs of PMC, DOE, 25
WARD, and Burns and Roe (Section 13.1.1.5.2).

13.1.1.5.1 TVA's Technical Staff

13.1.1.5.1.1 Division of Power Production

The Division of Power Production is responsible for the operation and maintenance of all TVA power plants and will have this responsibility
50 for the CRBRP. Included within P PROD for technical support are the Nuclear Generation Branch, the Plant Engineering Branch, and the Power Plant Maintenance Branch. Table 13.1-1 provides a summary of the number of personnel in each of these branches as well as their educational background and technical experience.

The Nuclear Generation Branch will direct the operation and maintenance of the CRBRP; review and analyze operating and engineering data, regular and special reports, test results, and other information pertaining to the operation and maintenance of the CRBRP; and review and coordinate operating maintenance, and surveillance procedures to ensure that the CRBRP is operated to provide maximum safety along with achieving project operating and demonstration goals.

The Nuclear Generation Branch is responsible for providing input to the Office Power for operational aspects of the plant design.

During initial operation of the Plant, the Branch coordinates activities through the plant manager including preoperational, startup,
50 and acceptance tests with PMC, DOE, Nuclear Regulatory Commission (NRC), reactor manufacturer, and the equipment suppliers. The Branch will coordinate the activities of the CRBRP with other branches and divisions within TVA in such areas as onsite fuel management and waste disposal.

50 The staff of this branch meets the "staff specialist" definition of ANSI N18.1-1971, and consists of the following:

Chief, Nuclear Generation Branch

Assistant Chief, Nuclear Generation Branch

Supervisor, Reactor Engineering Staff

Supervisor, Nuclear Operations Staff

50 Supervisor, Preoperational Staff

The Nuclear Generation Branch will analyze reports of abnormal equipment operation or faults to determine the initiating cause so that remedial measures can be taken. The Branch also assists in training the plant staff and evaluating training programs.

The Chief, Nuclear Generation Branch, meets both the definition and qualification of "Engineer in Charge" as set forth in "Standards for Selection and Training of Personnel for Nuclear Power Plants", ANSI N18.1-1971.

The Plant Engineering Branch provides a variety of engineering services for all TVA generating plants. For the nuclear plants, it provides technical assistance in the area of nuclear engineering, chemical engineering, instrument engineering, and testing.

The staff of the Plant Engineering Branch meets the "staff specialist" definition of ANSI N18.1-1971. These staff specialists are:

Chief, Plant Engineering Branch

Staff Environmental Engineer

Staff Mechanical Engineer, Equipment and Structures

Staff Mechanical Engineer, Testing

Staff Chemical Engineer

Staff Mechanical Engineer, Thermal Cycles

50 Staff Instrument Engineer

The Power Plant Maintenance Branch provides services from the central office staff on electrical and mechanical maintenance problems and major shop services from its central service shops.

The staff of this branch meets the "staff specialist" definition of ANSI N18.1-1971. These staff specialists are:

Chief, Power Plant Maintenance Branch

Staff Nuclear Engineer

Staff Electrical Maintenance Engineer

Staff Turbine Maintenance Engineer

Staff Boiler and Reactor Maintenance Engineer

Staff Technical Support and Planning Engineer

50 Superintendent, Power Service Shops

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Personnel within the organization of the Plant Engineering Branch and the Power Plant Maintenance Branch provide technical support to nuclear facilities when called upon to do so.

13.1.1.5.1.2 Other TVA Organizations

Other organizations within TVA which supplement the CRBRP and Division of Power Production Staff are as follows:

Division of Power System Operations (PSO)

Division of Transmission Planning and Engineering (TP&E)

Division of Power Resource Planning (PRP)

Division of Engineering Design (EN DES)

Division of Construction (CONST)

Division of Chemical Development (CHEM D)

Division of Medical Services (MED SV)

Division of Property and Services (P & SVS)

50 Division of Environmental Planning (ENV PL)

Office of Power Management Services Staff

Office of Power Quality Assurance and Audit Staff

A description of the duties of these organizations is given in Sections 1.4.2.4.1 and 1.4.2.4.2.

13.1.1.5.2 Project Technical Support

Project technical support for the operation of the CRBRP will
50 be provided to the Division of Power Production in TVA by PMC, DOE, WARD, and Burns and Roe. Areas of support will be in accordance with the responsibilities described in Section 1.4.

13.1.2 Operating Organization

13.1.2.1 Plant Organization

The plant organizational chart is shown in Figure 13.1-1. The principal groups that function directly under the supervision of the Plant Manager and Assistant Plant Manager are the Plant Operations Section, the

Plant Engineering Section, and the Plant Maintenance Section. Staff services are provided by an administrative staff, a Quality Assurance staff, and the Health Physics Unit of the Radiclogical Hygiene Branch. The latter is under the administrative supervision of the Division of Environmental Planning. The CRBRP organization follows the pattern developed through experience and used at all TVA fossil and nuclear generating plants.

Plant employees are selected primarily from existing TVA conventional and nuclear plant staffs and DPP's central office. Personnel qualifications shall meet the criteria set forth in the ANSI N18.1-1971.

13.1.2.1.1 Plant Operations Section

The Plant Operations Section is responsible for all plant operations. It provides operating personnel for the preoperational
50 testing, fuel loading, startup testing, startup, and plant operation. It is responsible for coordinating and scheduling the training program for all operations personnel. It provides the nucleus of emergency teams such as the plant rescue and fire-fighting organizations.

The Plant Operations Section is under the direction of the Operations Supervisor who holds a valid NRC Senior Reactor Operator (SRO) license. He is assisted by an inline Assistant Operations
50 Supervisor who also holds a valid NRC SRO license.

Within the Plant Operations Section are five shift crews. The minimum shift crew will consist of the Shift Engineer who holds an NRC SRO license, one Assistant Shift Engineer who holds an NRC SRO
50 license, one Unit Operator who holds an NRC Reactor Operator (RO) license, and three Assistant Unit Operators. One Health Physics Technician will also be assigned to each shift. Additional operators are assigned as necessary. Plant management and technical support will be present or on call at all times.

13.1.2.1.2 Plant Engineering Section

The Plant Engineering Section is under the direction of the Engineering Supervisor. He is assisted by a complement of engineers. The Plant Engineering Section is responsible for providing technical direction and staff assistance in the areas of nuclear, mechanical, instrumentation, and chemical engineering. Responsibilities of this section include plant and equipment performance tests, inplant fuel management, waste management, chemistry control, and instrumentation maintenance.

The Plant Engineering Section carries out a comprehensive program of plant tests, studies, and investigations for the purpose of monitoring the reactor, engineered safeguards, and plant operating conditions to assure compliance with the operating license and technical specifications and to improve the efficiency of the plant. This includes the coordination of the surveillance test program with other plant sections.

13.1.2.1.3 Plant Maintenance Section

The Plant Maintenance Section is under the direction of the Maintenance Supervisor. He is assisted by two inline Assistant Maintenance Supervisors.

The Plant Maintenance Section is responsible for mechanical and electrical maintenance work and inspections in the plant. This includes scheduling and conducting periodic inspections and tests on the systems assigned to this section associated with the reactor
50 and engineered safeguards, as required by the technical specifications and operating license. This section develops and carries out a preventive maintenance program that assures that the repair and replacement of parts are consistent with the intent of applicable codes and basic requirements of the original equipment. A record file is maintained by the section on all equipment, inservice tests, inspections, and maintenance reports.

13.1.2.1.4 Health Physics Unit

The Health Physics Unit of the Radiological Hygiene Branch is responsible for all health physics activities at the plant. It develops and applies radiation standards and procedures; reviews proposed methods of plant operation; participates in development of plant documents; and assists in the plant training program, providing specialized training in radiation protection. It conducts comprehensive onsite environmental radiation monitoring before, during, and after plant startup and provides radiological health coverage for all operations including maintenance, fuel handling, waste disposal, and decontamination. It is responsible for personnel and inplant radiation monitoring and maintains continuing records of personnel exposures, plant radiation, and contamination levels.

This unit is under the administrative supervision of the Chief, Radiological Hygiene Branch in the Division of Environmental
50 Planning, and under the functional supervision of the Plant Manager.

13.1.2.1.5 Administrative Group

The administrative staff, under the supervision of the Supervisor, Administrative Services, performs management service functions and clerical services for the plant.

50 13.1.2.1.6 Division of Power System Operations (PSO) Engineering Unit

50 The PSO Engineering Unit, of the Division of Power System Operations, is responsible for the maintenance and testing of the relaying associated with the transmission system. They are also responsible for maintenance of all external communications systems at the plant (with the exception of the Bell Systems Equipment). They are responsible for maintenance of portions of the onsite distribution and bus protection relaying.

This unit is under the administrative supervision of the Chattanooga Area Superintendent in the Division of Power System Operations and under the functional supervision of the Plant Manager.

13.1.2.1.7 Nuclear Plant Quality Assurance Staff

The nuclear plant quality assurance staff is under the direction of the Quality Assurance Staff Supervisor.

The nuclear plant quality assurance staff is responsible for developing, planning, initiating, and directing a comprehensive nuclear plant quality assurance/quality control program in the plant. Responsibilities include informing and advising other plant sections of the applicability, requirements, and implementation of the quality assurance program.

The nuclear plant quality assurance staff is responsible for coordinating, scheduling, and verifying surveillance monitoring and
50 inspections of safety-related structures, systems, and components.

13.1.2.2 Personnel Functions, Responsibilities, and Authorities

During normal plant operations, the plant manager is responsible for all plant activities. In the event of absences, incapacitation of personnel, or other emergencies, the following persons will be responsible in the order listed for all plant activities:

Plant Manager

Assistant Plant Manager

Plant Operations Supervisor

Plant Engineering Supervisor

Shift Engineer

13.1.2.2.1 Plant Manager

The Plant Manager has direct responsibility for all plant activities. He is responsible for safeguarding the general public and plant employees from hazards associated with the operation of the CRBRP through implementation of the TVA hazard control standards and requirements, applicable DOE and NRC rules and regulations, and plant procedures,
50 and for adherence to all requirements of the operating license and technical specifications. He receives direction and supervision from the Chief, Nuclear Generation Branch, and staff assistance from the Division of Power Production Central Office.

13.1.2.2.2 Assistant Plant Manager

The Assistant Plant Manager assists the Plant Manager in planning, coordinating, and directing the plant activities. In the absence of the Plant Manager, he is responsible for management of the plant activities.

13.1.2.2.3 Plant Operations Supervisor

The Plant Operations Supervisor is responsible for the safe and efficient operation of the plant in accordance with the operating license, technical specifications, and approved procedures and TVA hazard control standards and requirements. He is responsible for the preparation and maintenance of up-to-date operating procedures and the preparation of operating records. He is also responsible for operator training programs and operating personnel schedules and is charged with the responsibility of keeping the Plant Manager fully informed in all matters of operating significance.

13.1.2.2.4 Assistant Plant Operations Supervisor

The Assistant Plant Operations Supervisor assists the Plant Operations Supervisor in reviewing, coordinating, and planning the activities of the plant Operations Section. In the absence of the Plant Operations Supervisor, he assumes the responsibilities of that position.

13.1.2.2.5 Plant Engineering Supervisor

The Plant Engineering Supervisor serves as supervisor of the Plant Engineering Section and as a staff engineer in providing engineering advice and assistance to the Plant Manager. He is responsible for initiating, planning, and coordinating the technical training programs. His experience and training must provide him with a good understanding of nuclear reactor technology, hazards, safeguards, and licensing requirements and a knowledge of the control systems used in a nuclear plant. He is responsible for analysis of the performance of the reactor and turbine cycle and associated equipment during the test, startup, and operation of the plant.

13.1.2.2.6 Plant Maintenance Supervisor

The Plant Maintenance Supervisor is responsible for all mechanical and electrical maintenance work and inspections in the plant. He is responsible for maintaining safe working conditions for his employees and for their adherence to safe working practices. He is assisted in his work by two Assistant Supervisors with experience in mechanical and electrical maintenance. He is also assisted by foremen of the various crafts within the organization and engineers who will be assigned to the plant as the workload demands. The Plant Maintenance Supervisor must have a thorough knowledge of the operation and maintenance of all plant mechanical and electrical equipment.

13.1.2.2.7 Assistant Plant Maintenance Supervisors

The two Assistant Plant Maintenance Supervisors--one a mechanical specialist, the other an electrical specialist--assist the Supervisor in planning, coordinating, and directing the maintenance work and inspection in the plant.

13.1.2.2.8 Health Physicist

The Health Physicist is the onsite supervisor of the Health Physics Unit of the Radiological Hygiene Branch and is responsible for direction of an adequate program of health physics surveillance for all plant operations involving potential radiation hazards. He keeps the Plant Manager informed, at all times, of radiological conditions related to personnel exposure and potential contamination of site and environs. His duties include training and supervising health physics technicians; planning and scheduling monitoring and surveillance services; maintaining current data files on radiation and contamination levels; personnel

exposures, and work restrictions; and ensuring that operations are carried out within the provisions of developed radiological hygiene and procedures. He provides monitoring assistance and technical advice to plant operations and provides assistance to the medical staff in emergencies where radiation and contamination hazards are involved.

13.1.2.2.9 Supervisor, Nuclear Plant Quality Assurance Staff

The Nuclear Plant Quality Assurance Staff Supervisor serves as supervisor of the nuclear plant quality assurance staff and as a staff advisor to the Plant Manager. He is responsible for advising the Plant Manager of unresolved quality assurance problems and trends significant to plant operation and safety. He is responsible for review and approval for plant procedures and instructions. He also advises the Plant Manager of failures of plant equipment to meet technical specification requirements and other nonconforming aspects of operations. He is responsible for the inplant quality assurance/quality control training programs.

13.1.2.2.10 Safety Engineer

The Plant Safety Engineer provides consultation to plant management on all fire safety matters; coordinates and evaluates testing, maintenance, and repair of all fire-related equipment and systems; conducts periodic safety and fire inspections to identify deficiencies and recommends corrective actions; conducts fire training and evaluates fire drills; provides on-the-scene advice to fire brigade leaders during fire emergencies as applicable. He reviews pre-fire plan and emergency planning documents and coordinates fire safety matters as
50 required with Safety Engineering Services at the Central Office.

13.1.2.3 Shift Crew Composition

Normal Operations

The Shift Engineer on duty is in direct charge of the plant including startup, operation, and shutdown of the reactor and turbogenerators. He may institute immediate action in any given situation to eliminate difficulties or remove equipment from service to preclude violation of the operating license or technical specifications or to avert possible injury or undue radiation exposure of personnel.

The Assistant Shift Engineer is under the immediate supervision of the Shift Engineer. He follows established procedures in doing his work. However, if a particular situation is not covered by a procedure, he may seek advice from the Shift Engineer; or, if the situation is critical, he may use his own judgment to prevent damage

to equipment, injury to personnel, or undue radiation exposure of personnel. He performs operations in the electrical switchyard, diesel generator building, and other areas inside and outside the main powerhouse structure.

The Unit Operator is under the immediate supervision of the Assistant Shift Engineer and the general supervision of the Shift Engineer. He follows established procedures in operating the plant.

The Assistant Unit Operator is under the immediate supervision of the Unit Operator and the general supervision of the Assistant Shift Engineer. He follows established operating instructions in doing his work and does not deviate from those instructions except as directed. He performs assigned routine inspections and manipulative operations without close supervision. He assists in the operation and performs work requirements within defined areas such as the Control Building, Reactor Containment Building, Reactor Service Building, Turbine Generator Building, Diesel Generator Building, Intermediate Building, Steam Generator Building, and Intake Structure.

When on shift, the Radiochemical Analyst is under the functional supervision of the Shift Engineer. These duties consist of periodic sampling of the various systems, such as feedwater and main steam, water makeup, waste condensate, and periodic monitoring of the primary and secondary sodium coolant.

When on shift, the Health Physics Technician is under the functional supervision of the Shift Engineer. He performs routine radiation surveys, personnel monitoring activities, and other assigned duties. He keeps the Shift Engineer informed of radiation hazards and performs special surveys as requested.

During the five-year demonstration test period, a Technical Engineer will be assigned to shift who is under the supervision of a Shift Engineer. However, he receives his technical guidance from the Supervisor of the Plant Engineering Section. He has the responsibility of providing technical assistance to the Shift Engineer in performing the demonstration and test portions of the program.

13.1.3 Qualification Requirements for Nuclear Plant Personnel

All personnel at the CRBRP will be required to obtain and maintain qualification standards equal to or better than those specified in ANSI N18.1-1971. The personnel selection and training program that assures fulfillment of these qualification requirements also satisfies
50| NRC Regulatory Guide 1.8. Specific minimum qualifications for all those personnel discussed in Section 13.1.2 are given below.

13.1.3.1 Minimum Qualification Requirements

Plant Manager

At the time of initial core loading or appointment to the active
position in the licensed plant, the Plant Manager shall have 10 years
of responsible power plant experience of which a minimum of three years
shall be nuclear experience. A maximum of four years of the remaining
seven years of experience may be fulfilled by academic training on a
one-for-one time basis. This academic training shall be in an
engineering or scientific field generally associated with the production
of power. The Plant Manager shall have acquired the experience and
training normally required for examination by NRC for a SRO license
50 whether or not the examination is taken.

If the Assistant Plant Manager meets the nuclear plant
50 experience and NRC examination requirements established for the Plant
Manager, the requirements of the Manager may be reduced so that only
one of his 10 years of experience need be nuclear plant experience,
50 and he need not be eligible for NRC examination.

The Plant Manager or the Assistant Plant Manager should have
a recognized baccalaureate or higher degree in an engineering or
scientific field generally associated with power production.

Assistant Plant Manager

At the time of initial core loading or appointment to the
active position in the licensed plant, the Assistant Plant Manager
shall have a minimum of eight years of responsible power plant experience
of which a minimum of three years shall be nuclear plant experience.
A maximum of four years of the remaining five years of the power plant
experience may be fulfilled by satisfactorily completing academic or
related technical training on a on-for-one time basis. A degree in
science or engineering is desirable. He or the Plant Manager shall
50 be capable of fulfilling the requirements of an NRC SRO license whether
or not the examination is taken. If the Plant Manager has the required
three years of nuclear experience, the requirements of the Assistant
Plant Manager may be reduced so that only one of his eight years
of experience needs to be nuclear plant experience.

Plant Operations Supervisor

At the time of initial core loading or appointment to the
active position in the licensed plant, the Plant Operations Supervisor
shall hold an NRC SRO license and shall have a minimum of 8 years of
50 responsible power plant experience, of which a minimum of 3 years shall
be nuclear plant experience. A maximum of 2 years of the remaining

5 years of power plant experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one time basis. The required nuclear experience for this position may be reduced to one year if the Assistant Plant Operations Supervisor has the required nuclear plant experience.

Plant Engineering Supervisor

At the time of initial core loading or appointment to the active position in the licensed plant, the Plant Engineering supervisor shall have a minimum of 8 years of responsible power plant experience or applicable industrial experience of which 2 years shall be nuclear plant experience. He should have an engineering or science degree.

Plant Maintenance Supervisor

At the time of initial core loading or appointment to the active position in the licensed plant, the Plant Maintenance Supervisor shall have a minimum of 7 years of responsible power plant experience or applicable industrial experience, including at least one year of nuclear plant experience. A maximum of 2 years of the remaining 6 years of power plant or industrial experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one time basis. He further should have familiarity with nondestructive testing and maintenance of sodium containing components, craft knowledge, and an understanding of electrical, pressure vessel, and piping codes.

Supervisor, Nuclear Plant Quality Assurance Staff

At the time of initial core loading or appointment to the active position in the licensed plant, the Supervisor of the Nuclear Plant Quality Assurance Staff shall have 7 years of responsible power plant experience or applicable Quality Assurance experience of which a minimum 2 years shall be nuclear power plant experience. He shall be a graduate with a degree in engineering. A maximum of 2 years of the remaining 5 years of power plant or quality assurance experience may be fulfilled by satisfactory completion of academic or related training on a one-for-one time basis. If the Staff Supervisor has not had the quality assurance experience, he shall receive training from the Office of Power Quality Assurance and Audit Staff relative to basic quality assurance theory and practice. This training shall include an orientation to the Office Power Quality Assurance Program as defined by the Office of Power Quality Assurance Manual.

The Safety Engineer

The Plant Safety Engineer shall have a sound understanding and thorough technical knowledge of safety and fire protection practices, procedures, standards, and other codes relating to electrical utility operations. He shall: be able to read and understand engineering drawings; possess an analytical ability for problem solving and data analysis; be able to communicate well both orally and in writing; be able to write investigative reports and prepare written procedures; have the ability to secure the cooperation of management, employees, and groups in the implementation of safety programs; and be able to conduct safety presentations for supervisors and employees. He shall have experience in safety engineering work at this level or have 3 years experience in safety and/or fire protection engineering. It is desirable that the incumbent be a graduate of an accredited college or university with a degree in industrial, mechanical, electrical, or
50 safety engineering or fire protection engineering.

Health Physicist

50 The plant Health Physicist shall meet the qualifications as specified in NRC Regulatory Guide 1.8.

Plant Operations Section Employees

At the time of initial core loading or appointment to the active position in the licensed plant, the Assistant Plant Operations Supervisor shall have a minimum of 6 years of responsible power plant experience, of which a minimum of one year shall be nuclear plant experience. A maximum of 3 years of the remaining 5 years of power plant experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one time basis.

At the time of initial core loading or appointment to the active position in the licensed plant, the Shift Engineers shall have fulfilled the requirements of TVA's Training Plan for Operators and have a high school diploma or equivalent and six years of responsible power plant experience, of which a minimum of one year shall be
50 nuclear plant experience, and he shall hold an NRC SRO license.

At the time of initial core loading or appointment to the active position in the licensed plant, the Assistant Shift Engineers shall have fulfilled the requirements of TVA's Training Plan for Operators and have a minimum of a high school diploma or equivalent and five years of responsible power plant experience, of which a minimum of one year shall be nuclear plant experience, and he shall
50 hold an NRC SRO license.

At the time of initial core loading or appointment to the active position in the licensed plant, the Unit Operators shall have fulfilled the requirements of TVA's <u>Training Plan for Operators</u> a high school diploma or equivalent and two years of power plant experience, of which a minimum of one year shall be nuclear plant experience. The latter, for operators with no previous nuclear experience, will consist of a basic nuclear course, plant technology course, plant systems and operators course, and control board experience. He shall have an NRC
50 RO license.

At the time of initial core loading or appointment to the active position in the licensed plant, the Assistant Unit Operators working within the plant shall have a minimum of a high school diploma or equivalent, completed requirements of TVA's <u>Training Plan for Operators</u>, completed a basic nuclear course and plant systems course, and had several months of onsite plant familiarization. This position
50 does not require an NRC RO license.

<u>Plant Engineering Section Employees</u>

At the time of initial core loading or appointment to the active position in the licensed plant, the Instrument Engineer shall have a bachelor's degree in science or engineering and minimum of one year's experience in the field of instrumentation. Six months of this experience shall be in nuclear instrumentation and control.

At the time of initial core loading or appointment to the active position in the licensed plant, the Chemical Engineer shall have a bachelor's degree in science or engineering and a minimum of one year's experience in radiochemistry.

At the time of initial core loading or appointment to the active position in the licensed plant, the Reactor Engineer shall have a minimum of a bachelor's degree in engineering or the physical sciences and two years of experience in such areas as reactor physics, core measurements, core heat transfer, and core physics testing program.

The Radiochemical Analysts and other engineering aides shall be high school graduates with a minimum of two years' experience in their respective fields.

Each Instrument Mechanic shall have a minimum of three years' experience in his craft and shall be a skilled journeyman.

<u>Plant Maintenance Section Employees</u>

At the time of initial core loading or appointment to the active position in the licensed plant, the Assistant Maintenance Supervisor shall have a minimum of five years of responsible power plant

experience or applicable industrial experience, including at least one year of nuclear plant experience. The position requires familiarity with nondestructive testing, craft knowledge, and an understanding of electrical, pressure vessel, and piping codes.

Each TVA craftsman shall be a skilled journeyman. These experienced journeymen will predominantly be transferees from other TVA generating plants and installations. Craftsmen shall have a minimum of three years in one or more crafts. The primary source of new journeymen is the TVA apprenticeship program. This program, jointly administered by a TVA labor-mangement council, normally requires in excess of four years for completion. The program requires assignments designed so that he will develop skills equal to the recognized journeyman standard. Related classroom and correspondence lesson assignments provide the technical information needed in the actual work being done on the job.

Preoperational testing will be carried out for all safety-related fire protection systems in accordance with the requirements of nationally recognized fire codes and standards. The test director will be an experienced system-oriented test engineer assisted by one or more engineers from the reactor manufacturer, the architect-engineer, and/or the constructor.

The engineer responsible for developing and assisting in implementing the fire protection program shall have a B.S. degree in engineering and shall have 3 years or more of experience working on mechanical engineering projects. He must have acceptance testing experience in fixed fire protection systems and have a general knowledge of nationally recognized fire codes and standards.

13.1.3.2 Qualifications of Plant Personnel

The positions listed in 13.1.3.1 have not yet been filled. These positions will be filled as indicated in Figure 13.2-1.

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TABLE 13.1-1

TECHNICAL SUPPORT SUMMARY

GROUP AND NUMBER OF PERSONNEL	MANYEARS OF EXPERIENCE			
	TOTAL UTILITY EXPERIENCE		NON-UTILITY EXPERIENCE	
	NUCLEAR POWER FIELD	OTHER UTILITY EXPERIENCE	NUCLEAR POWER FIELD	OTHER ENGINEERING FIELDS
Reactor Engineering Staff Nuclear Generation Branch - 23	91.25	6.0	41	7
Preoperational Test Staff Nuclear Generation Branch - 40	180.25	130.25	43.75	95
Nuclear Operations Staff Nuclear Generation Branch - 3	24	25	0	0
Chemical Section Plant Engineering Branch - 12	63.25	82.50	37.50	44.00
Mechanical Section Plant Engineering Branch - 12	31.50	161.5	5	53
Structural Section Plant Engineering Branch - 11	40.75	148.25	6.50	25.00
Instrument and Controls Section Plant Engineering Branch - 61	106.85	121.5	42.5	177.5

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TABLE 13.1-1 continued

GROUP AND NUMBER OF PERSONNEL	MANYEARS OF EXPERIENCE			
	TOTAL UTILITY EXPERIENCE		NON-UTILITY EXPERIENCE	
	NUCLEAR POWER FIELD	OTHER UTILITY EXPERIENCE	NUCLEAR POWER FIELD	OTHER ENGINEERING FIELDS
Special Projects Section Plant Engineering Branch - 11	0	61	0	49
Test Section Plant Engineering Branch - 22	29	76.25	1	69.5
Economy & Statistical Section Plant Engineering Branch - 8	0	13	0	0
TOTALS FOR 211 TECHNICAL SUPPORT PERSONNEL	566.85	825.25	177.25	514.0

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TABLE 13.1-1 continued

GROUP AND NUMBER OF PERSONNEL	DEGREES HELD		
	BACCALAUREATE LEVEL	MASTERS LEVEL	DOCTORATE LEVEL
Structural Section Plant Engineering Branch- 11	Mechanical Engineering - 3 Civil Engineering - 2 Engineering - 2 Industrial Engineering - 1 Engineering Physics - 1 Engineering Mechanics - 1		
Instrument and Controls Sections Plant Engineering Branch- 61	Electrical Engineering - 41 Nuclear Engineering - 2 Math - 3 Business Management - 2 Mechanical Engineering - 2 Computer Science - 1 Technical Training - 3 High School - 5	Electrical Engineering - 1 Business Administration-1	
Special Projects Section Plant Engineering Branch- 11	Mechanical Engineering - 6 Chemical Engineering - 2 Biology - 1 Industrial Technology - 1 Accounting - 1	Mechanical Engineering - 3 Chemical Engineering - 1 Safety Engineering - 1	
Test Section Plant Engineering Branch- 22	Mechanical Engineering - 12 Math - 1 Aeronautical Engineering - 1 Pre Law - 1	Mechanical Engineering - 1	
Economy and Statistical Section Plant Engineering Branch- 8	Mechanical Engineering - 3 Math - 2	Nuclear Science - 1 Math - 1	Statistics - 1
211 TOTAL TECHNICAL SUPPORT PERSONNEL			

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TABLE 13.1-1 continued

GROUP AND NUMBER OF PERSONNEL	DEGREES HELD		
	BACCALAUREATE LEVEL	MASTERS LEVEL	DOCTORATE LEVEL
Reactor Engineering Staff Nuclear Generation Branch- 23	Nuclear Engineering - 14 Electrical Engineering - 2 Engineering Physics - 2 Chemical Engineering - 1 Chemistry - 1 Engineering Science - 1 Physics - 1 Physics and Math - 1 Mechanical Engineering- 1	Nuclear Engineering - 7 Nuclear Science and Engineering - 1	
Preoperational Test Staff Nuclear Generation Branch- 40	Electrical Engineering-16 Mechanical Engineering-11 Nuclear Engineering - 6 Marine Engineering - 1 Engineering Physics - 1 Physics - 1	Electrical Engineering-2 Nuclear Engineering - 1	
Nuclear Operations Staff Nuclear Generation Branch- 3			
Chemical Section Plant Engineering Branch- 12	Chemistry - 5 Chemical Engineering - 4	Chemistry - 2 Physics - 1	Chemistry - 2 Nuclear Chemistry - 1 Environmental Hygiene - 1
Mechanical Section Plant Engineering Branch- 12	Mechanical Engineering-9 Engineering Physics - 1 Marine Engineering - 1		

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13.2 TRAINING PROGRAM

13.2.1 Program Description

The basic objectives of the training program are:

a. To assure that all plant personnel are properly trained and qualified to perform their assigned tasks in a safe and efficient manner.

b. To assure that the CRBRP is operated in accordance with
50 NRC regulatory requirements and guidelines.

c. To assure that all training is formally documented.

50 d. To meet or exceed NRC licensing requirements.

In achieving these objectives, individual training needs are established by comparing job requirements with individual experience.

The training program, as it is initially constructed, is approved by the Chief, Nuclear Generating Branch, after being approved by the Plant Manager. This ensures that the content and the intent of the training program provide the necessary training for personnel associated with reactor operations. The program is designed to train personnel both with and without previous nuclear experience.

The effectiveness of the training program is evaluated by the
50 performance of employees on TVA and NRC examinations in carrying out
their assigned duties. In addition, periodic audits of the training program are performed by designees within the Office of Power, but outside the Division of Power Production.

13.2.1.1 Program Content

At the time of manning the CRBRP, TVA should have highly trained nuclear plant operating personnel at the Browns Ferry, Sequoyah, and Watts Bar Nuclear Plants. These plants will be the primary source of personnel for the CRBRP.

Those positions at the CRBRP which require an NRC licensed
50 SRO shall be filled with personnel who have or are eligible to sit
for an NRC SRO license on a commercial size light-water reactor. Other positions shall be filled with personnel within the TVA organization as available and selected from competent applicants from outside. All CRBRP personnel will be given comprehensive training to produce personnel who have that combination of education, experience, and skills commensurate with their level of responsibility which provides reasonable assurance that decisions and actions during all normal and off-normal conditions will be such that the plant is operated in a safe and efficient manner.

The TVA student operator training program and replacement training at operating TVA nuclear plants shall ensure no loss of operator efficiency at those plants because of transfer of personnel to the CRBRP. Individual training needs shall be established by carefully examining the individual's experience and previous training and comparing these with the job requirements. The formal program to be
50 provided for candidates seeking NRC RO license or NRC SRO license as well as the length of each aspect of the program is discussed in the following paragraph and depicted in Figure 13.2-1. The training program shall consist of the following phases:

Basic Nuclear Courses

Plant Technology and Specialist Training

Reactor Operations (LWR and FBR)

Actual Fast Reactor Training

Onsite Work-Study Program

Nuclear System Special Training to include Sodium Technology

A final training plan based on the needs of the individual staff members will be prepared after personnel selection and shall be included in the Final Safety Analysis Report.

13.2.1.2 Coordination with Preoperational Tests and Fuel Loading

Figure 13.2-1 presents a proposed training schedule for the CRBRP which satisfies the requirements of ANSI N18.1-1971. It is planned that the following personnel shall be licensed in accordance with the requirements of 10CFR55 before initial fuel loading: Operations Supervisor, at least five Shift Engineers, and at least five Assistant Shift Engineers. The Plant Manager or the Assistant Plant Manager shall obtain the training required for an SRO license. It is planned
50 to obtain RO licenses for at least five Unit Operators during startup testing of the plant. The various phases of training available are outlined below, along with descriptions of personnel participation in each phase. 11

13.2.1.3 Practical Reactor Operation

Practical training at TVA's Browns Ferry, Sequoyah, and Watts Bar and at an operating sodium-cooled fast reactor is anticipated in order to provide the experience required for applicants for cold licenses.
50 The persons who will initially obtain SRO licenses shall participate in this training. Training requirements shall be individually determined and training will be supplied to fill these needs.

The program involving the actual participation in the operation of a sodium-cooled fast reactor to gain experience in the areas of liquid metal systems and the characteristics and performance of fast reactors shall be integrated into this phase of the total training program for licensable personnel and management personnel.

13.2.1.4 Reactor Simulation Training

A simulator for the CRBRP is not planned, therefore, simulator
50 training for candidates seeking NRC RO licenses or NRC SRO licenses is not included as a part of the nuclear training program.

13.2.1.5 Previous Nuclear Training

Figure 13.2-1 presents the tentative training schedule showing the relation of the training to the plant schedule for construction, testing, and operation. The actual training schedule will be dependent upon the background and experience of the individuals chosen for positions requiring a cold license. Hence, the schedule is tentative and subject to change before submission as a part of the Final Safety Analysis Report.

13.2.1.6 Other Scheduled Training

Basic Nuclear Courses

The nuclear courses for operators will consist of basic atomic and nuclear physics; nuclear reactor principles, including neutron and reactor physics, reactor kinetics, reactor control, reactor instrumentation, and reactor materials, with special reference to fast reactors; reactor core thermal-hydraulic characteristics, such as hot channel factor, sodium boiling and voiding, linear heat rate; and radiation protection and radiation safety. In addition, the course will include work on sodium technology. Other personnel whose duties require basic nuclear training, such as the Chemical Technicians and Instrument Mechanics, will receive more abbreviated instruction in nuclear fundamentals as part of their on-site specialist training.

The prerequisite qualifications for participation in the basic nuclear courses and succeeding phases of the operator training program are:

a. High school education or equivalent.

b. Knowledge of mathematics through high school algebra.

c. Ability to use a slide rule.

d. Demonstration during previous experience of a desire to learn and produce quality work.

e. Demonstration by past performance of maturity, good judgment, and high moral character.

f. Satisfactory completion of medical examination.

g. Satisfactory performance of the work in his present classification.

Plant Technology and Specialist Training

A design lecture series covering the function, design and operation of nuclear systems and components shall be conducted at the plant
50 site. The persons requiring NRC SRO licenses shall participate in this plant technology training. In addition, the plant management and engineers shall also participate.

Various specialist training, consisting of work-study assignments, shall be conducted for plant engineers, technicians, and maintenance personnel. This training shall be specifically tailored to the individual's needs. It is planned to use the Browns Ferry, Sequoyah, and Watts Bar facilities for as much of this training as feasible. Specialist training in LMFBR technology is planned for the Supervisors, the Nuclear Engineer, and certain central office staff engineers. Specialist training is planned to varying degrees in instrumentation and controls including process computer programming and maintenance for the Instrument Engineer, several Instrument Mechanics, Plant Engineering Supervisor, and certain Central Office staff engineers. Training assignments at TVA nuclear plants of varying lengths are planned for plant staff personnel such as the Instrument Engineer, Reactor Engineer, Mechanical Engineer, and Chemical Engineer. After completion of this specialist training, the appropriate personnel shall organize and conduct necessary specialist training onsite for the Chemical Technicians, Maintenance personnel, Instrument Mechanics, and others.

Onsite Work-Study Program

This phase, which begins before fuel loading, integrates personnel into their plant assignments. It is conducted under the direction of TVA supervisory personnel. During this period, plant personnel participate in the preparation of procedures and manuals, preoperational testing, preoperational checkout of the operating procedures, initial fuel loading, and initial startup program. The
50 applicants for NRC RO and NRC SRO licenses will participate in further training and examination preparation related to obtaining
50 the required NRC license. All plant personnel will participate in a plant indoctrination and radiation-protection course.

Fire Brigade Training

Although Fire Brigade Training is not a prerequisite to sitting for an NRC Operator License exam, it is included as a portion of the operator, assistant shift engineer, shift engineer, assistant operations supervisor, and operations supervisor training.

The objective of this training is to ensure that plant fire brigade members, leaders, and other plant employees performing fire-related functions receive comprehensive first aid firefighting instructions and application that will be instrumental in the prevention, control, and suppression of plant fires. This training will be updated or revised
50 as necessary to ensure that current, acceptable practices are included.

13.2.1.7 Training Programs for Non-Licensed Personnel

TVA, on a continuing basis, plans and administers training programs for the professional and managerial development of its employees. Relationships are maintained with both local and state educational institutions as well as with the vendors of various items of equipment. Advantage is taken of appropriate seminars, specialized courses, and training activities offered by these groups to keep employees abreast of new developments in power production and safety.

13.2.1.8 General Employee Training

Personnel with specific duties and responsibilities in the plant shall receive instruction in the performance of these duties and responsibilities. All persons having unescorted access to the plant areas shall have completed either (1) intensive nuclear training, which will include radiation protection techniques and the site emergency plan, or (2) a brief plant indoctrination and radiation protection course which will include discussion of plant organization and layout, controlled zones, radiation and contamination hazards, exposure limits and controls, elementary health physics, and pertinent sections of the site emergency plan.

When persons who have not completed either (1) or (2) above enter the plant areas, they will be escorted by an employee who has received training in radiation protection and plant emergency procedures.

A. permanent plant personnel shall receive training periodically in the plant's fire protection policies. Temporary personnel who have responsibilities in fire protection will also be included.
50

Training and indoctrination relating to quality assurance will be provided to all employees as applicable in conformance to RDT-F2-2, Section 7.3.2.

Periodic retraining of plant personnel regarding radiation hazards with emphasis on individual actions, will be conducted at monthly meetings attended by all available plant personnel, and at short weekly meetings held within the various plant groups. The emergency plan will be discussed at least once annually in these meetings.

13.2.1.9 Responsible Individual

The individual responsible for conducting and administration of the nuclear power plant training program is the Assistant Plant Manager. The plant Quality Assurance Staff Supervisor shall be responsible for developing and directing the Nuclear Plant Quality Assurance Program which complies with RDT-F2-2.

13.2.2 Retraining Program

This information will be included in the FSAR.

13.2.3 Replacement Training

This information will be included in the FSAR.

13.2.4 Records

13.2.4.1 TVA

Official records of employee qualifications experience, training, and retraining of each member of the plant organization are maintained in the official TVA Personal History Record (PHR) by the Division of Personnel. The PHR provides in standardized arrangement, the information officially recognized in recording and supporting employee status. The PHR is maintained in current and accurate status and is controlled as to availability. The material admitted to this record is restricted to items for which authenticity has been confirmed through established procedures; e.g., official TVA forms, signed statements from the employee, management representatives, etc.

13.2.4.2 Plant

50 Records supporting requests for NRC SRO and NRC RO licenses are maintained in the plant master file. These records include training courses attended, retraining classes, number of reactor startups, and other information necessary to ensure that training requirements have been met. Some of these records are duplicated in the PHR.

A training file for each member of the plant organization is maintained in the plant master file. Information regarding participation in training and retraining activities and records of employee participation in training activities leading to promotion to a higher level of competence will be maintained in this training file.

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13.3 EMERGENCY PLANNING

13.3.1 General

TVA's emergency plans contain the precautionary planning, delegation of authority and responsibility, and plans of action to protect the public, plant employees, and equipment in case of unusual incidents. As specified in 10CFR50, Appendix E, these plans are for use at the local level for the control of general emergencies such as fire, personnel injury, tornadoes and high winds, and incidents that could result in the release of a significant amount of radioactivity.

The TVA Radiological Emergency Plan (REP), for the CRBRP will contain the overall TVA REP, the Nuclear Emergency Medical Assistance Plan, and the CRBRP Annex. The CRBRP Annex will contain four documents. They are the (a) Division of Power Production REP, (b) Site REP, (c) Enrivons Emergency Plan (EEP), and (d) State of Tennessee REP.

These documents are briefly described below. The actual TVA REP for the CRBRP will be submitted as a separate document along with the Final Safety Analysis Report.

a. The TVA REP is designed to handle all radiological emergencies which might occur within TVA. During a nuclear emergency at a plant site, the Central Emergency Control Center (CECC) staff will function to provide assistance as necessary to the site and division emergency organizations and will provide all information requested by outside agencies.

b. The Nuclear Emergency Medical Assistance Plan will outline all arrangements which have been made for medical services which may be required for the CRBRP employees or others affected by the emergency.

c. The CRBRP Annex will contain the four following documents:

50 | 1. The Division of Power Production (P Prod) REP requires automatic staff actions to provide required assistance for the site by alerting support facilities, concluding arrangements with civilian support facilities, and providing any support requested by the plant. The major assistance provided by the division emergency staff will be to the plant itself although the staff will provide personnel services as required by state and local agencies. The division emergency staff will also coordinate the efforts of other divisions within TVA.

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2. The Site REP will deal with control of the emergency within the site boundaries.

3. The Environs Emergency Plan (EEP) will deal with the emergency beyond the site boundary.

4. The State of Tennessee REP will provide the support of state organizations in the event of a nuclear emergency and is principally concerned with the well being of area citizens. This plan will work hand in hand with the CRBRP EEP.

13.3.2 Emergency Organization

The normal shift operating crew provides the nucleus of the plant's emergency organization. The shift crew has an adequate number of personnel with the authority to take required immediate action in any emergency. The plant emergency organization is headed by an Emergency Director. The Shift Engineer is responsible for declaring an emergency and acting as Emergency Director until relieved by the Plant Manager or a designated alternate from the plant staff. After relief, the Shift Engineer remains in charge of detailed inplant operations. The shift organization is supplemented by predesignated individuals from the remainder of the plant staff after notification by telephone or messenger. The plant emergency organization has preassigned duties and responsibilities and is trained to perform all actions that may be necessary to cope with the emergency and to implement the emergency plan. In addition to the plant emergency organization, the unaffected plant staff could provide additional personnel to assist as necessary.

In the event of an emergency involving the possibility of danger to the public or the offsite environment, the plant Emergency Director notifies TVA's operating duty specialist who notifies the Central Emergency Control Center (CECC) Director. The CECC organization consists of TVA management personnel from various TVA divisions and offices and is located in Chattanooga. The CECC has the authority to make arrangements and expend funds as necessary to protect the environment from the adverse effects of an emergency. They coordinate TVA offsite activities and work with various other Governmental emergency groups. The members of the CECC staff are predesignated, aware of their responsibilities, and conduct periodic drills to maintain a high degree of readiness.

50 | The P PROD emergency organization is also notified by the Plant Emergency Director and provides additional manpower as required to augment the plant organization. The personnel may come from other
50 | TVA nuclear plants, the P PROD Central Office, or the P PROD Service Shops Section, depending on the nature of the emergency and the
50 | disciplines required. The P PROD emergency organization will also provide technical support groups for emergency planning and recovery operations.

As required, an environs emergency team is dispatched to the vicinity of the emergency to conduct TVA's offsite monitoring activities and to work with monitoring groups from other agencies.

Communication networks are adequate to handle predicted emergencies. These communications facilities include:

a. Public Address Intra-Plant Communication (PA-IC)

This system provides primary communication throughout the plant. It also provides a warning throughout the plant of fire, high radiation, and building evacuating through the use of a multi-tone generator.

b. Private Automatic Exchange (PAX)

This system provides communication paths through tie lines and interfacing circuits of the following:

1) Microwave communication to the TVA-wide direct-dial system

2) The page channel of the PA-IC system

3) The Powerline carrier communication system.

It also incorporates a dial-in executive right of way.

c. Manual Telephone Switchboard at the electrical control desk which is connected through tie-lines and interfacing circuits to the following:

1) Microwave connections to the TVA-wide direct-dial system

2) Powerline carrier communication system

3) The PAX switching equipment which provides full benefits of the PAX system

d. Bell System commercial telephone service - this system is not connected to any of the communications system within the plant and is not a part of the TVA-wide communications system. However, this service is available at key locations in the plant.

e. Maintenance Communications Jacking System (MCJ) - this system consists of sound powered headsets/microphones and jack stations provided to facilitate the testing and calibration in the maintenance discipline.

50

f. Radio equipment including a VHF base station, portable radios (walkie-talkies), vehicle-mounted radios (security officers) and radios located at environmental sampling stations. 50

13.3.3 Coordination with Offsite Groups

TVA has agreements with other Federal agencies through the Interagency Radiation Assistance Plan to assist in the evaluation and
50 control of any radiological emergency. These agencies include DOE's Savannah River and Oak Ridge Operations Offices. The CECC staff will request assistance from these outside agencies as required. The CECC
50 staff is also responsible for notification of NRC's regional office of the Division of Compliance.

Agreements have been made with the Tennessee Departments of Public Health, Civil Defense, Agriculture, Public Safety, and Conservation to provide planning for emergencies at TVA nuclear facilities. This planning includes evacuation arrangements, traffic control, and support from civil defense agencies. The Tennessee Department of Public Health will be notified and will coordinate assistance from other state agencies as required.

TVA maintains liaison with various agencies of county and municipal governments, particularly with respect to the availability of emergency services. The State Department of Public Health informs these agencies of actions to be taken under their respective statutory authority and assists them in developing emergency procedures. TVA may call upon these agencies directly for fire and police protection. TVA will meet with representatives of the various county and municipal governments to discuss their involvement in the Radiological Emergency Plan. TVA will provide training for local fire departments in radiological hygiene practices and recognition of radiological hazards. The attached Table 13.3-1 lists the organizations that will be participating in the CRBRP Emergency Plan.

Arrangements will be made with a local private ambulance service to provide emergency service as required to the plant and affected areas in the event that such service is required. Agreement will also be culminated between TVA and a local hospital to provide emergency treatment of irradiated or contaminated patients as required. TVA will assist in training ambulance attendants and hospital personnel in this type of treatment and will ensure that adequate equipment is made available. Agreement has also been made with the Oak Ridge Associated Universities Radiation Emergency
50 Assistance Center/Training Site for emergency treatment of severely contaminated or irradiated personnel.

13.3.4 Protective Action Levels

Protective action levels are established depending on the nature of the emergency and the value of continuously monitored variables. These protective action levels defined in the REP include those which may affect only a local area of the plant or a small number of employees as well as these which could possibly involve the public in unrestricted areas. The protective action levels are based on control room indications of continuously monitoring variables such that the operator can quickly determine the nature of the emergency. Local emergencies may also be detected by the shift crew during routine plant tours and inspections.

13.3.5 Protective Measures

For each protective action level, definite protective measures or actions are specified in the emergency plan. The Site REP contains such actions which include mustering personnel in preassigned assembly areas, informing personnel of the nature of the emergency, confirming the indication of the emergency, notifying the CECC Director, mustering the plant emergency organization in the Emergency Control Center, conducting radiation surveys, locating missing personnel, evacuating non-essential personnel from the site, accounting for any visitors, and limiting access to the plant areas. Personnel safety is the prime consideration in all protective measures. Protective measures become more detailed and extensive with the increasing protective action levels. The emergency plan contains site drawings designating assembly areas and evacuation routes.

Protective measures, including evacuation, for persons living outside the plant boundary are included in the EEP and State REP and are expected to be required only after evaluation of plant conditions and effluent release rates. However, immediate protective measures will be specified based on previously determined dose rates, population distributions, meteorological conditions, and plant conditions that could cause site boundary conditions requiring action. These protective measures include preplanned evacuation procedures and routes, reassembly points, traffic control, and public announcement.

13.3.6 Review and Updating

The Plant Operations Review Committee will periodically review and update the CRBRP Site REP. All holders of these plans will acknowledge in writing, receipt of all changes.

13.3.7 Medical Support

The emergency plan includes a description of the medical facilities at the plant and the arrangements made with other facilities to provide additional support. The plant medical facilities include a treatment area consisting of an emergency room, treatment room, bedroom, physiotherapy room, and waiting room. A full-time nurse in on duty during the day shift. A complete stock of medical supplies and first-aid equipment is available. One ambulance is maintained at the site. Medical consultation is available from TVA doctors in Chattanooga and other areas. Members of the plant emergency team are trained in first aid.

Arrangements will be made with a local hospital and with attending physicians for the emergency treatment of contaminated, injured, and exposed individuals. The Oak Ridge Associated Universi-
50 | ties Radiation Emergency Assistance Center/Training Site has agreed to provide treatment to severely contaminated or exposed individuals.

Arrangements will be made with a local private ambulance service to provide emergency service as required to the plant and affected areas in the event that more than one ambulance is required.

13.3.8 Drills

Periodic drills will be conducted by the plant staff on the plant emergency plans. Personnel will assemble, be accounted for, and prepare to assume their preassigned duties. Contact will be made with concerned persons outside the plant organization to confirm the adequacy of communications facilities.

On an annual basis, a TVA-wide drill on the Radiological Emergency Plan is conducted. The various members of the emergency staff will assemble, assume their duties, and get in touch with the various outside organizations, identifying the action as a drill.

13.3.9 Training

Each person having unescorted access to the plant will have received either intensive nuclear training which included the emergency plans action or a brief plant indoctrination and health physics course which includes pertinent sections of the emergency plan. Specific training will be conducted for individuals assigned to the plant emergency organization. This will include first-aid training, radiological hygiene training, decontamination training, and training in the emergency procedures.

TVA will assist in providing training in decontamination and treatment of contaminated patients to the staff of the local hospital and the commercial ambulance service.

13.3.10 Recovery and Reentry

The emergency plans provide for the development and implementation of detailed recovery and reentry plans based on evaluation of conditions existing at the time. Recovery and reentry will be a deliberate, thoroughly planned evolution and will be reviewed by the Plant Operations Review Committee depending on the nature of the emergency.

13.3.11 Implementation

Operating instructions, promulgated in the plant operating manual, are used to control plant operations during normal operating conditions. Abnormal operating instructions and emergency operating instructions are used to specify the manipulation of controls of the plant during conditions requiring protective measures to be taken to place the plant in a safe condition. The abnormal and emergency instructions contain assignments of responsibility for the performance of specific tasks not otherwise established by plant practices and instructions.

Plant instrumentation indications requiring implementation of emergency and abnormal operating instructions are specified in these instructions. Protective action levels, also based on plant instrumentation indication, requiring implementation of the Radiological Emergency Plan for protection of personnel and the environment are specified in the emergency plan.

Specific actions required of offsite support groups are delineated in the TVA-wide Radiological Emergency Plan and in a Division of Power Production Emergency Plan.

Instructions for medical treatment and handling of contaminated and exposed individuals are contained in the Plant Emergency
50 Plans manual and a TVA Nuclear Emergency Medical Assistance Plan.

Equipment requirements, inlcuding communications equipment, for implementation of the plant emergency plans are contained in these plans. Storage and calibration requirements are also specified. Alarm signals are described in the respective emergency plans.

Instructions for restoring the emergency situation to normal, from the standpoint of hazard to personnel, plant safety, and the environment, are contained in the emergency plans and the emergency and abnormal operating instructions. Instructions for repair of plant equipment or structures will be prepared after evaluation of the damage or malfunction involved.

13.3.12 Radiological Analysis

A radiological analysis of the facility design features, site layc , and site location with respect to considerations of surroundings in compliance to 10 CFR 50, Appendix E has been conducted. The findings of this analysis are listed in this section.

13.3.12.1 Projected Ground Level Doses

Plots showing projected ground level doses, for both whole body and thyroid, resulting from the most serious design basis accident analysis is depicted in Figures 13.3.1 through 13.3.4. These provide, respectively, the elapsed exposure times to reach specific bone, lung, thyroid, and whole body doses as a function of downwind distance based on exposures resulting from the Site Suitability Source Term (SSST). The use of the SSST is conservative since it envelopes the most serious design basis accident analyzed in the PSAR.

Table 13.3-2 summarizes the data and parameters used in the analysis. The information provided in the Table fully describes the basis of the analysis.

13.3.12.2 Accident Assessment, Warning and Evacuation Times

13.3.12.2.1 Assessment

The time required for the initial accident assessment of the most serious design basis accident may required 15 minutes. This time is an estimate based on the operation of the reactor instrumentation used to follow the course of accidents. Based on TVA's experience, the time required to perform an initial dose projection and notify off-site authorities can be accomplished in 15 minutes.

For most serious design basis accident, the projected two-hour doses at the exclusion area boundary do not reach the protective action guide level for evacuation, i.e., 5 rem whole body dose and 10 rem child's thyroid dose.

13.3.12.2.2 Warning

The notification of persons within the potential evacuation sector (see Figures 13.3-5 and 13.3-6) can be accomplished in less than one hour. This time frame has been established by conversations with DOE-ORO, State Health Department, and local Civil Defense officials who will be involved in evacuation efforts and who will establish the
50 detailed means used to warn all affected resident and transient persons.

Specific discussions regarding emergency planning for the facilities located within the LPZ boundary have been held with DOE-ORO, the Roane and Loudon County, Tennessee Civil Defense Directors, and the Nuclear and Industrial Safety manager of U. S. Nuclear, Inc. Also, with the reduction of the LPZ boundary from the original 5.0 miles (letter, S:L:997, P. S. Van Nort to R. S. Boyd, "Amendment No. 18 to the PSAR for CRBRP", dated April 30, 1976) only U. S. Nuclear, Inc. and approximately 7 of the recreational areas identified in Table 2.1-14 will be located within the LPZ. Consistent with the guidance in the Standard Format and Content of Safety Analysis Reports of Nuclear Power Plants, however, the emergency plans extend outward encompassing a five-mile area.

13.3.12.2.3 Evacuation

Control within the exclusion area, excluding the waterway, will be by the applicant and can be evacuated in approximately 45 minutes. Due to the short distance involved, the applicant (by means of loud speakers used along the patrol road) will be able to warn persons using the waterway of adverse conditions. Control of the waterway will be coordinated with the regional office of the Tennessee Wildlife Resources Agency, through the Tennessee Office of Civil Defense and Emergency Planning. Upon request, the Agency will block the flow of river traffic, evacuate water craft, and evacuate persons along the shoreline. The Agency estimates that within one hour of notification, they can have craft on the river for traffic control at the plant exclusion area.

The estimated time to accomplish notification and evacuation of any sector of the environs (see Figures 13.3-5 and 13.3-6) is approximately two hours. The basis for the environs time frame is derived from discussion with DOE and local civil defense directors who will be responsible for evacuation efforts. The times are estimates and detailed evacuation procedures will be developed for the CRBRP-Radiological Emergency Plan (REP). Buses may be used to evacuate the Edgewood School. The problem has been discussed with the Roane County Defense Director and will be specifically addressed in the CRBRP-REP. Table 13.3-3 shows the combined resident and maximum transient population within the five-mile zone for 1980 (projected approximate level at the time the plant is scheduled to commence operation). Figures 13.3-5 and 13.3-6 outline the proposed evacuation sectors and Table 13.3-4 gives the resident and transient population data for each evacuation sector for 2010 (projected peak level during expected life of the plant). Table 2.1-14 estimates the average peak hour use of recreational areas within 10 miles including Melton Hill Dam (Site Number 12) which are included in the analysis.

50

Figures 13.3-5 and 13.3-6 show the roads available for evacuation of the plant environs out to 10 miles from the site. All road types surface characteristics, and road widths are identified by the legend. Road encumbrances not evident are identified in the CRBRP Map Information block.

Discussions have been held with the U. S. Nuclear, Inc., the only location within the LPZ Boundary where people will normally populate on a pre-arranged schedule, concerning the CRBRP emergency plans. Safeguard measures for the continued protection of SNM at the U. S. Nuclear, Inc., facility and an adequate emergency plan insuring protection of U. S. Nuclear employees is possible either through measures taken to protect persons required to remain at the plant and the evacuation of all non-essential personnel. The CRBRP - Radiological Emergency Plan will contain the specific emergency plans regarding the U. S. Nuclear, Inc., facility.

Regarding the ORGDP and ORNL, both of these facilities are outside the LPZ, but since they are within a five-mile radius of the CRBRP, they will be addressed in the emergency plans. Discussions have been held with DOE concerning these two facilities. Should the need be identified, all non-essential personnel can be evacuated in approximately two hours. However, process requirements and security requirements may prevent the complete evacuation of these DOE facilities. The dose levels to these few remaining personnel would be acceptable and consistent with appropriate guidelines. Special procedures, protection or
50 equipment could be used, as necessary to provide reduction of the dose.

TABLE 13.3-1

PARTICIPANTS IN CRBRP RADIOLOGICAL EMERGENCY PLAN

Tennessee Department of Public Health

Tennessee Department of Civil Defense

Tennessee Department of Agriculture

Tennessee Department of Public Welfare

Tennessee Department of Safety

Tennessee Department of Conservation

Tennessee National Guard

Tennessee Game and Fish Commission

Tennessee Department of Transportation

50 Tennessee Wildlife Resources Agency

City and County Officials of Roane and Anderson Counties

Sheriffs' Departments of Roane and Anderson Counties

Civil Defense Coordinators of Roane and Anderson Counties

Local Police Departments

Local Ambulance Service

Local Fire Department

Oak Ridge Associated Universities Radiation Emergency Assistance Center/Training Site

Department of Energy Savannah River Plant Operations Office (IRAP)

50 Department of Energy Oak Ridge Operations Office (IRAP)

Environmental Protection Agency, Region IV, Atlanta

Eastern Environmental Radiation Facility, Montgomery, Alabama

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TABLE 13.3-2

SUMMARY OF DATA UTILIZED FOR
SOURCE TERM RADIOLOGICAL ANALYSIS

Source Term

100% Noble Gases
25% Halogens (50% release to containments, 1/2 if which [25% total] is airborne and available for release)
1% Solid Fission Products
1% Plutonium

Released instantly to and uniformly distributed in Reactor Containment Building.

Meteorology

Atmospheric dispersion parameters (x/Q's) are the ninety-fifth percentile values (see Section 2.3). Consistent with Regulatory Standard Review Plan, Section 2.3.4, the 0-2 hour exposure intervals were evaluated based on the single-hour 95% x/Q value.

Plume front transit times to downwind positions are based on a wind speed of 1 mile/hour.

Containment Modeling

The following parameters are used to evaluate Source Term releases from containment:

RCB Leakage to Annulus (Direct to Annulus Filter Intake)	0.1% Volume/Day
Annulus Flow Rates	
Filtered Exhaust	3000 CFM
Filtered Recirculation	3500 CFM per 1000 CFM Exhausted
Time Delay from Source Term Release to Initiation of Annulus Filtration	No Delay
Time Delay from Source Term Release to Initiation of Annulus Recirculation	<10 Seconds
Total Bypass Leakage (1% of RCB Leakage)	0.001% Volume/Day
Bypass Leakage Direct to Environment (60% of Total Bypass)	0.0006% Volume/Day
Bypass Leakage to the RSB (40% of Total Bypass)	0.0004% Volume/Day

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TABLE 13.3-2 (Continued)

Sources of Bypass Leakage to the RSB	96.4% Personnel and Equipment Airlock 3.6% All Other Sources
Gamma Shielding	1.5" Steel (RCB) Plus 4' Concrete

Filter Efficiencies

Iodine	95%
Particulate	99%
Noble Gas	-0-

Radiological Parameters

Inhalation dose factors are per Regulatory Guide 1.109 for a standard adult.

Time dependent breathing rates are per Regulatory Guide 1.4.

External gamma whole body exposure is based on a semi-infinite cloud per Regulatory Guide 1.4 for the released material and includes direct exposure from the material within the Reactor Containment Building.

Radioactive decay of nuclides during downwind transit of the plume is conservatively neglected. While conservative, this assumption has minimal impact on the results, since off-site exposures are
50 controlled by relatively long-lived nuclides.

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TABLE 13.3-3

PROJECTED MAXIMUM RESIDENT + TRANSIENT
POPULATION DISTRIBUTION WITHIN
5 MILES OF THE DEMONSTRATION PLANT
FOR CENSUS YEAR 1980

Sector Designation	Radial Interval (miles)				
	0-1	1-2	2-3	3-4	4-5
N	0	95	0	0	0
NNE	0	0	0	0	0
NE	0	0	0	0	0
ENE	5	5	0	0	6520
E	10	5	30	125	1122
ESE	5	5	6015	65	115
SE	0	15	125	95	115
SSE	0	15	20	120	125
S	60	305	20	75	95
SSW	0	35	5	70	65
SW	5	25	30	100	75
WSW	0	30	70	315	345
W	0	55	165	105	150
WNW	0	155	100	30	55
NW	0	20	20	0	45
NNW	0	20	6000	0	75
Sum for Radial Interval	85	785	12600	1100	8902
Accumulative Total up to Radius Indicated	85	870	13470	14570	23472

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Amend. 50
June 1979

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TABLE 13.3-4

PROJECTED MAXIMUM RESIDENT AND TRANSIENT POPULATION* IN EVACUATION SECTORS WITHIN 5 MILES OF CRBRP

Sector+	1980	2010
A	6545	6520
B	7497	9162
C	885	885
D	960	1055
E	1365	1955
F	6220	6295

*Transient population is based on current available information

50 +See Figures 13.3-5 and 13.3-6

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FIGURE 13.3-1

ELAPSED EXPOSURE TIME TO REACH SPECIFIC BONE DOSE VERSUS DOWNWIND DISTANCE
(BASED ON SITE SUITABILITY SOURCE TERM)

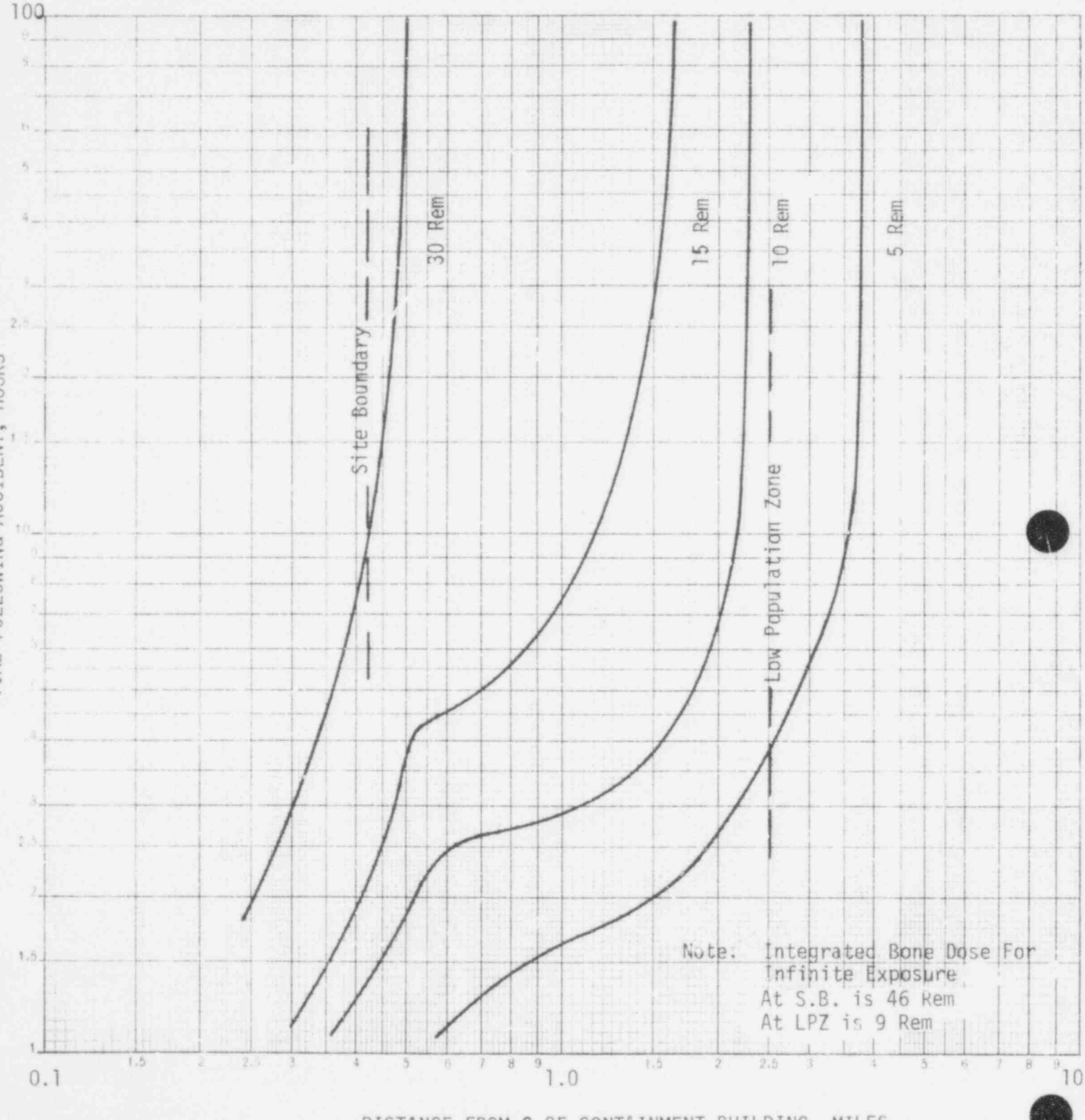

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FIGURE 13.3-2

ELASPED EXPOSURE TIME TO REACH SPECIFIC LUNG DOSE VERSUS DOWNWIND DISTANCE
(BASED ON SITE SUITABILITY SOURCE TERM)

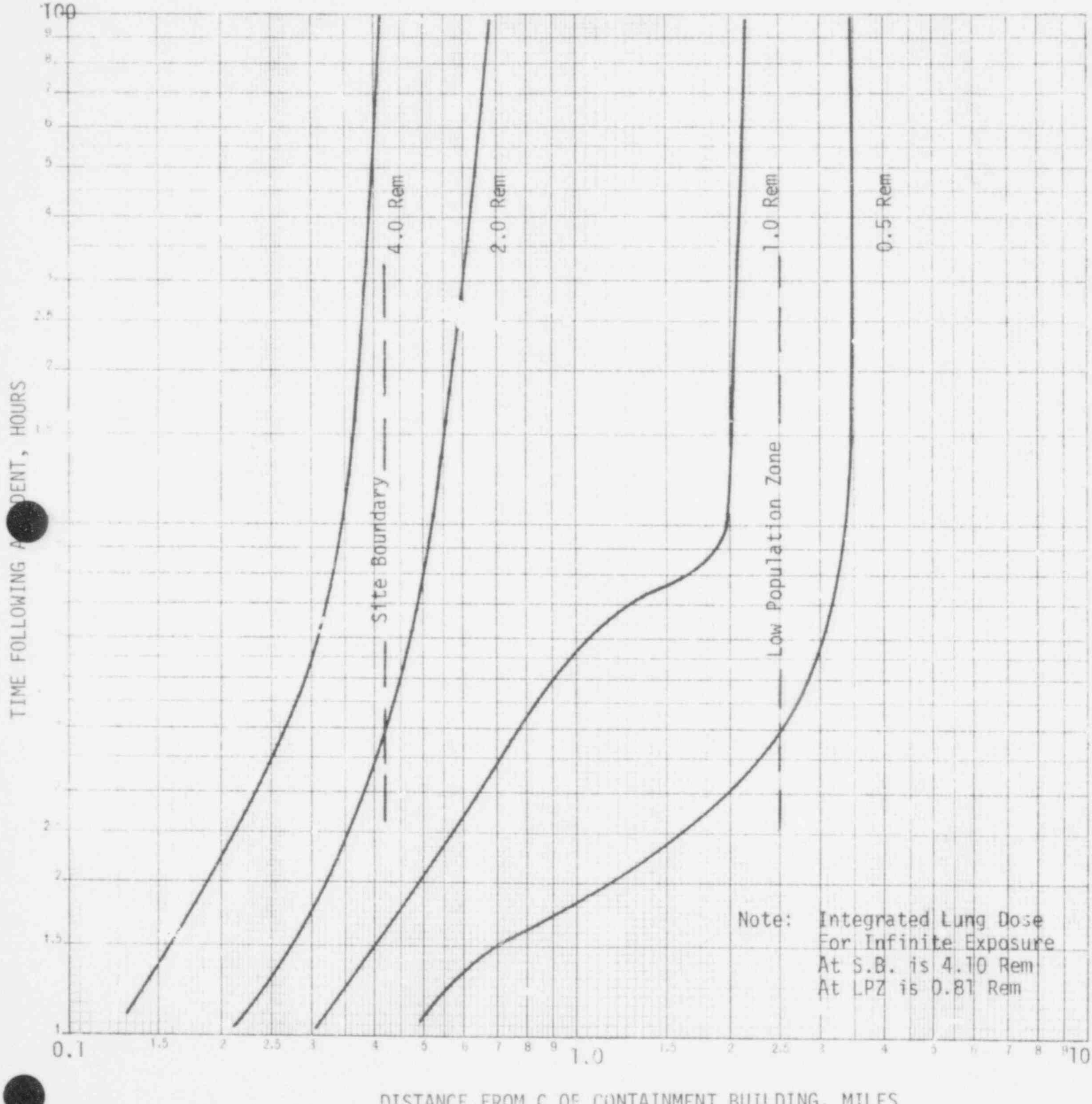

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FIGURE 13.3-3

ELASPED EXPOSURE TIME TO REACH SPECIFIC THYROID DOSE VERSUS DOWNWIND DISTANCE
(BASED ON SITE SUITABILITY SOURCE TERM)

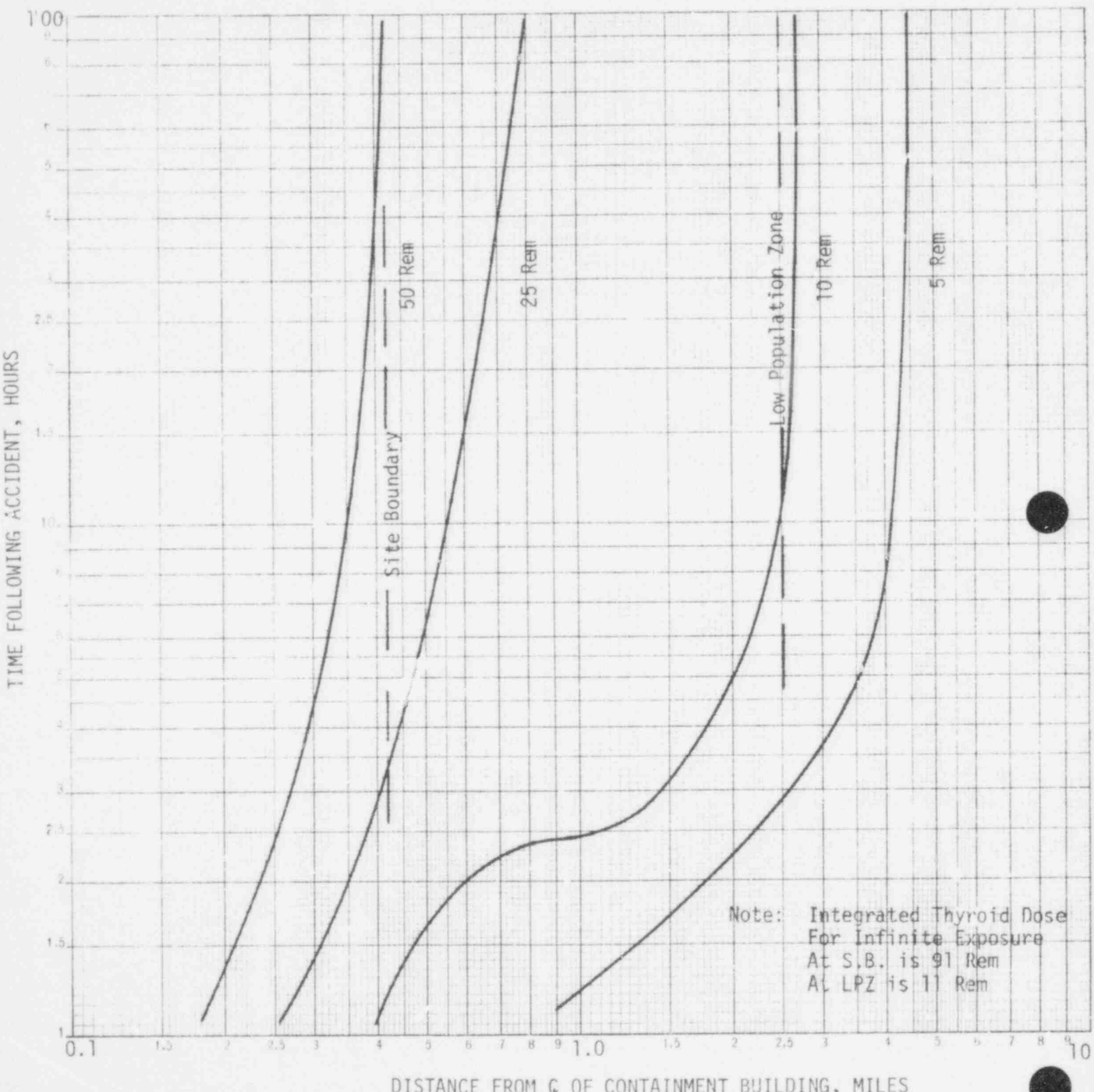

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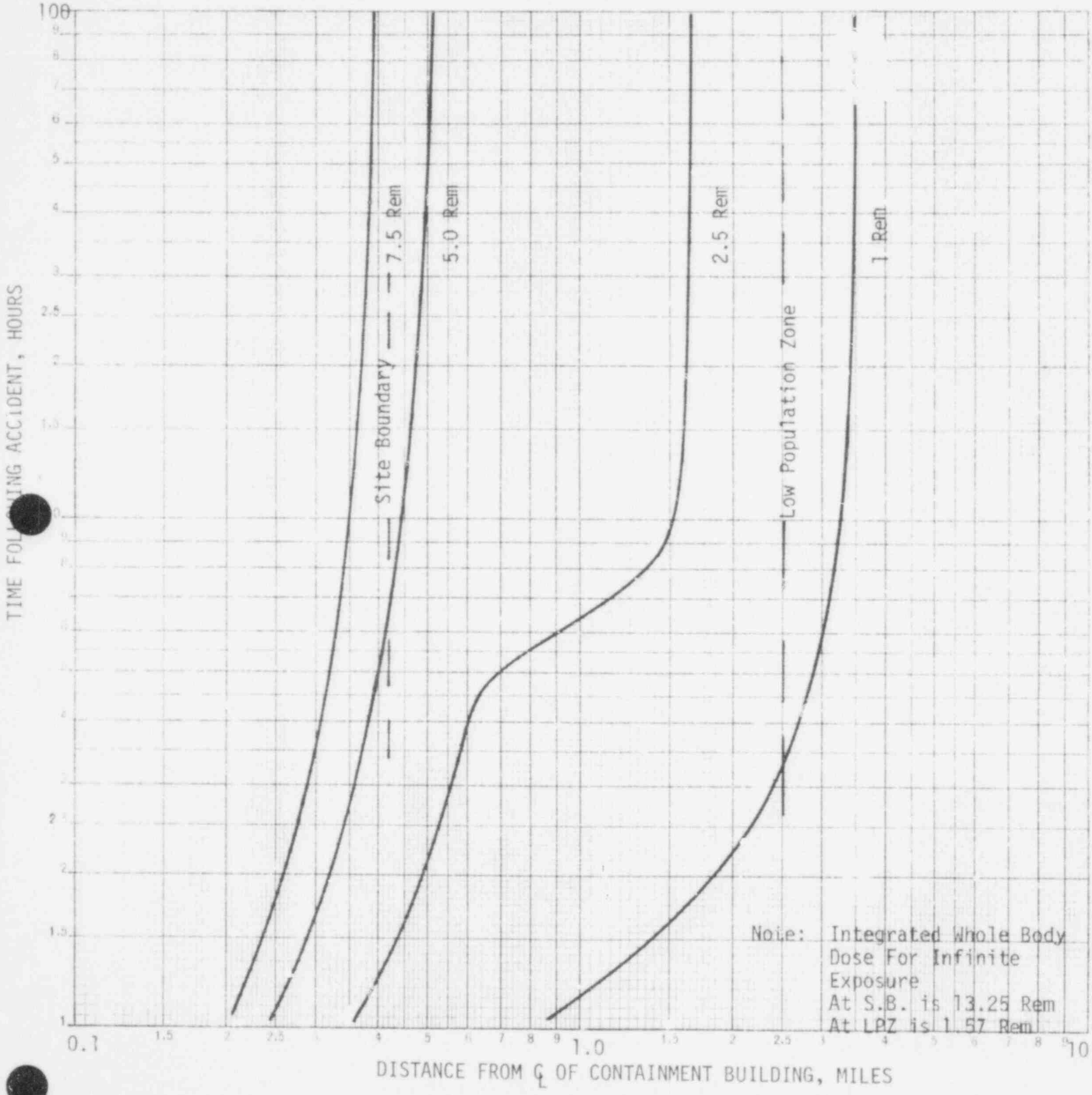
FIGURE 13.3-4

ELAPSED EXPOSURE TIME TO REACH SPECIFIC WHOLE BODY DOSE VERSUS DOWNWIND DISTANCE
(BASED ON SITE SUITABILITY SOURCE TERM)



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instructions will contain information describing the incident or conditions, probable indications, automatic actions which occur, immediate operator action, and any subsequent operator action necessary to correct or control the situation.

The primary responsibility for initiating the corrective action will rest upon the operator who first becomes aware of the situation. He will notify his supervisor of the existing condition and the action he has taken. All operating personnel through training and experience will have learned to recognize and evaluate impending failures or malfunctions and to initiate proper corrective actions.

The emergency instructions will be used to train the operating personnel and make them aware of the accidents or situations that could occur and the proper course of action.

Equipment Clearance Instructions

The clearance instruction is the method used by TVA for protection of workmen, the public, and equipment, whether it be electrical circuits, hydraulic equipment, mechanical equipment or other devices.

No work on such equipment will be performed except under the applicable clearance instruction. The Shift Engineer will be responsible for the tagging of equipment within his area of responsibility. The TVA load dispatcher will issue tagging instructions for the high-voltage switchyard including the transformers. The Health Physics Unit will be responsible for determining the existing radiation hazards. Clearances will be issued only to those persons whose names appear on official clearance lists.

A clearance will be established by the use of colored protective cards placed to indicate the boundary of isolation or special operating limitations.

Protective tags shall not be applied, altered, or removed except under applicable established procedures by authorized employees.

Every person working around equipment that is involved in a clearance will be responsible for recognizing the boundaries established by protective tags and the conditions imposed by the protective tags and must in no way violate the areas and conditions outlined.

13.5.5 Maintenance Instructions

The plant maintenance program will be designed to safely and efficiently provide maintenance and repair to keep the plant in good operating condition. Maintenance work will be initiated through work requests and/or by the preventative maintenance program. Safe working conditions will be assured by the use of TVA's hold order, clearance, and special work permit

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instructions. Complex and critical maintenance operations which require step-by-step performance will be detailed in written instructions. These instructions covering mechanical, electrical, and instrumentation maintenance will provide information to assure proper coordination of operating and maintenance employees as well as step-by-step procedures to be followed by the craftsmen doing the work

13.5.6 Surveillance Instructions

Instructions will be prepared covering the conduct of all surveillance tests and inspections designated in the plant technical specifications. These instructions will specify prerequisites, precautions, references, acceptance criteria, necessary step-by-step actions for conduct of the tests and return to normal, data sheets, and signatures of those conducting and reviewing the tests or inspections.

Detailed test schedules and records will be maintained to assure that all surveillance requirements are conducted in a timely manner and the results are properly documented.

13.5.7 Technical Instructions

Instructions covering routine technical operations will be prepared as required. Examples of these operations are chemical sampling and analysis, chemistry control, and calibration of vital instrumentation.

Fuel accountability instructions delineating the requirements, responsibilities, and methods of nuclear material control from the time new fuel is received until it is shipped from the plant as spent fuel will be utilized. They will provide detailed steps for physical safeguards, invencory, accounting, and for preparing records and reports.

13.5.8 Section Instruction Letters

Each section supervisor will, as the need arises, prepare numbered instruction letters pertaining to administrative routines, responsibilities, and methods to be followed by members of his section.

13.5.9 Site Emergency Plans

These plans are discussed in Section 13.3.

13.5.10 Radiation Control Instructions

Radiation control instructions are written and made available to all plant personnel. These instructions include permissible personnel exposures consistent with 10 CFR Part 20 and other requirements and guidelines to minimize radiation exposures. All plant personnel will be required to follow these procedures.

50

Amend. 50
June 1979

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IMAGE EVALUATION
TEST TARGET (MT-3)

6″

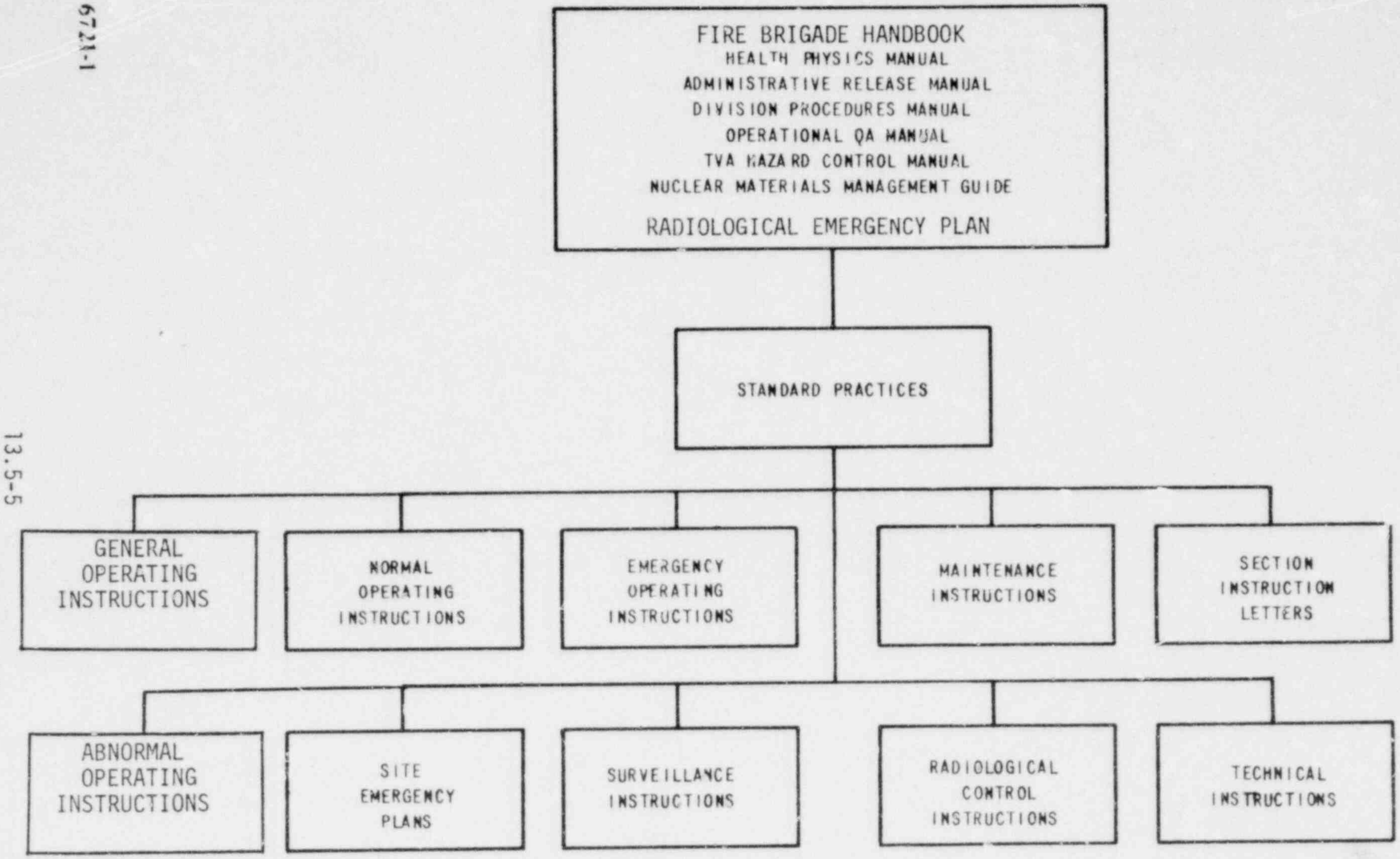

Figure 13.5-1 Plant Procedures

13.7 INDUSTRIAL SECURITY

The requirements of 10 CFR 73.55 and NRC Regulatory Guide 1.17 will be met for the CRBRP. This section discusses in general how the CRBRP will meet these requirements. The CRBRP Physical Security Plan shall provide specifics including the contingency plan and qualification and training plan for security personnel and will be submitted as a separate proprietary document at the FSAR stage. It will provide specific details as required by 10 CFR 50.34(c).

13.7.1 Organization and Personnel

The Division of Property and Services and the Office of Power of the Tennessee Valley Authority shall share the security responsibilities for the CRBRP. The organization chart shown in Figure 13.7-1 delineates this responsibility which is explained in the following Section 13.7.1.1 and 13.7.1.2.

13.7.1.1 Division of Property and Services

The Division of Property and Services (P&SVS), with the assistance of other TVA organizations, develops guides and standards on property protection, reviews protection plans for compliance with these guides and standards, and advises in their development and application.

P&SVS provides police and fire protection service on properties for which it is responsible and furnishes these services to other organizations in accordance with their protection plans. The Public Safety Service (PSS) in the P&SVS Public Safety Service Branch furnishes this service.

The Chief of the Public Safety Service Branch functions as overall TVA Emergency Coordinator in carrying out P&SVS's security responsibilities with other TVA organizations and providing liaison with federal, state, and local agencies on security and emergency preparedness matters. His organization provides supervision for the PSS.

The supervisor of PSS, located in the Public Safety Service Branch office, has several PSS area chiefs, each of which are over several PSS supervisors of units at power generating plants in their area.

The PS Security unit at the CRBRP is under direct supervision of the Public Safety Service Branch, but functions as an onsite armed security force for the Plant Manager according to existing plans and
50 the Manager's requirements.

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The Public Safety Service Branch is responsible for recruiting, training, and assigning security force personnel to the CRBRP unit as required.

13.7.1.2 Office of Power

The Office of Power is responsible for protection of power properties. It develops detailed plans and applies specific measures, with the advice of P&SVS.

The Chief, Management Services Staff of the Power Manager's Office, represents the Power Manager on all security matters. The Power Security Section which reports to the Chief of Management Services Staff coordinates planning and administration of industrial security
50 measures with all concerned.

Each Power division is responsible for security and fire protection and prevention of Power facilities under its control in accordance with general policies and general instructions from the Power Manager's Office. The Division of Power Production is responsible for security of nuclear plants. At the CRBRP, the Plant Manager and in his absence the Senior Plant Supervisor on duty is responsible for security of the plant in accordance with general policies, plans, and instructions received through administrative channels.

13.7.1.3 Employee Selection

As discussed in Section 13.1, TVA will operate the plant, and accordingly, will provide all operating and security personnel who will be regular TVA employees.

TVA appoints, promotes, transfers, and retains employees on the basis of merit and efficiency, as prescribed in the TVA Act and in accordance with other applicable Federal laws and regulations. It is the policy of TVA to promote present employees, whenever possible, who have demonstrated competence, reliability, and stability to vacant positions in preference to hiring persons from outside the organization. This is often accomplished by upgrading employees through internal training programs.

Specific instructions pertaining to personnel matters are contained in Section III of the TVA Administrative Release Manual. These instructions are observed by all plant supervisors, especially as they apply to appointment, transfer, promotion, and retention of employees.

Selection for a position is supportable by records of education, training, and experience, and by records of judgements
which have been made regarding work performance, ability, and condition 1
of health.

In selecting for placement or retention in positions, covered by agreements negotiated between TVA and the employee organizations, the provisions of such agreements are observed.

Because of TVA's conformance to the Veteran's Preference Act, when employing outside candidates for vacant positions, a large number of persons beginning employment have successfully completed tours of duty with the military forces of the USA. The availability for review of the military record of these candidates provides good control in the selection of high-quality candidates.

Each new annual TVA employee is given a physical examination and a national agency check, and written inquiries are routinely made to references such as former employers, schools, and police. Before any employee is allowed unescorted access to a nuclear plant protected area, there must be satisfactory results from his security check and
50 emotional stability check.

PS officer selection procedures include a preemployment interview by the PSS area chief and one or more PSS unit supervisors in addition to the steps previously mentioned. Upon acceptance, the candidate's first six months of employment are probationary. Appointment as a PS officer is dependent upon satisfactory service during this period and satisfactory completion of training and qualification as provided for in the CRBRP Training and Qualification Plan to be
50 submitted with the CRBRP Physical Security Plan.

13.7.1.4 Employee Evaluation

Because of the general policy of promoting present employees rather than appointing candidates from outside TVA, most employees at the CRBRP will be known from their previous employment record with TVA. Although TVA employees are not given routine psychiatric examinations, they shall be given when an employee's on-the-job performance indicates that this is desirable.

Observation of employee service is made as a regular part of day-to-day continuous supervisory function. When performing this function, supervisors shall be alert for any unusual behavioral patterns such as may result from mental stress, alcohol, or other drug abuse.

In addition to this kind of review, the performance of employees in management and salary policy positions are reviewed formally and the results reported in order (1) to further aid in maintaining a high level of employee performance and the maximum util-
ization of employee abilities; (2) to provide recorded evidence of 1

employee performance for use in making judgements concerning transfer, demotion, promotion, and terminations; (3) to assure that employees are adequately and systematically informed of the effectiveness of their service; and (4) to further facilitate the maintenance of a high standard of supervision in TVA. All employees' services are reviewed formally at the time of status changes and at such other times as may be required to achieve the above purposes. A service review shall precede each recommendation for operator licensing or renewal of an operator license.

13.7.1.5 Industrial Security Training

All employees shall receive training in security procedures with emphasis on being alert to the presence of unauthorized persons and evidence of forced entry. This training shall normally be con-
50 ducted by a member of the Plant Security Force under the direction of the Plant Manager.

13.7.2 Plant Design

The physical plant design has been developed so as to accom
50 date the necessary security provisions. TVA, along with DOE and PMC and its architect-engineer, Burns and Roe, will provide a continuing review of the plant design, as well as the detailed security provisions. Burns and Roe, as the architect-engineer for the Project, has been delegated the responsibility for detailing the security provisions. 31
The design criteria used at the CRBRP will assure that the physical security facilities and the plant layout are developed so as to thwart
50 any attempted sabotage. The physical security design will: 31

(1) Control entry to the plant site and portions of the plant;

(2) Deter or discourage penetration by unauthorized persons;

(3) Detect such penetrations in the event they occur; and

(4) Apprehend in a timely manner unauthorized persons or authorized persons acting in a manner constituting a threat of sabotage.

In the design and operation of the plant, care is taken to minimize the potential for industrial sabotage by the use of access control measures to prevent unauthorized persons from entering the protected area. Should such persons succeed in entering the protected area, special access control measures will prevent them from entering vital equipment areas and the SNM material access area. 1 31

13.7.2.1 Design Features

The design features and other physical security measures that will protect against or limit the effects of possible sabotage efforts include:

50 a. A security barrier with dual intrusion detection system around the perimeter of the plant, with gates that are kept closed and locked except during times
47 of authorized use.

b. Employee and visitor parking located outside the security barrier.

c. An isolation zone extending from inside the security barrier to outside the barrier in which all activities will be controlled. This zone shall be void of obtrusive structures and plant growth. In addition, a cleared zone will be maintained outside the isolation zone to facilitate observation of persons approaching the isolation zone.

d. A perimeter patrol road extending completely around the plant inside the security barrier.

e. A remote-controlled, outdoor, closed circuit television (CCTV) system to permit observation of the plant perimeter, isolation zone, cleared zone, protected area, and approach roads.

f. An outdoor lighting system to provide illumination to the protected area and isolation zone at a level compatible for both visual and CCTV observation.

g. A minimum number of exterior plant doors leading to vital areas, all of which shall be hardened against 31
penetration and kept locked or otherwise secured when not in use. 11

h. A cardkey electronic access control system to control personnel access to vital areas in conformance with each employee's level of authorization.

i. An alarm system to indicate status of hatches, emergency exits and seldom used equipment or personnel access doors providing access to vital areas not cardreader equipped.

j. An access control building (gatehouse) to control personnel access to the protected area and containing equipment to search personnel for weapons, explosives and special
50 nuclear material. 1

k. A communication system which will allow continuous communications between PSS officers and the central alarm station. Also, redundant communications links will be maintained between the plant and the local law enforcement agency.

l. An electric power system to provide emergency power to the security and lighting loads during periods of "blackout" or loss of normal power.

m. A force of trained, uniformed, and armed PS officers used on a three-shift basis to police the property,
50 provide access control, respond to alarms, evaluate the situations, and neutralize the threats.

n. Fire fighting and other emergency equipment located throughout the plant area to minimize the consequences of fires or explosions.

o. Engineered safeguards and protective systems that are provided to minimize the consequences of fires or explosions or to minimize the effects of postulated major equipment failures, natural disasters, and operator errors which would also serve to minimize the effects of industrial sabotage.

13.7.2.2 Physical Arrangements

The CRBRP site is in a remote location. It is unlikely that major civil disorders would occur at or near the plant area. The plant is located on a peninsula formed by a meander of the Clinch River between river miles 14.5 and 18.6 near the center of a 1364-acre tract owned by and in the custody of the United States Government (see Figure 13.7.2).

13.7.2.3 Owner-Controlled Area

Ultimately, a permanent access road to the plant will lead into the plant. During construction, a temporary construction road will lead into the construction area. The perimeter of the reservation shall be marked prior to the completion of construction with signs to provide reasonable assurance that persons entering the area are aware they are on private property. Adequate roads shall be provided to patrol and control the entire reservation. Employee parking areas shall be located outside the security barrier so that only plant vehicles and trucks making deliveries will need to be admitted. A motor patrol of the reservation area shall be made at least once each evening and night shift. While construction is in progress, the temporary construction road will be the only route of access to the Project. 1

A continuous access control guard post will be maintained on this road for the duration of construction activities. Figure 13.7-2 shows the reservation boundary of the owner-controlled area.

13.7.2.4 Protected Area

When all construction work is completed, there will be an
8-foot high perimeter security barrier enclosing the protected area
excluding the main plant buildings and associated outdoor facilities.
An isolation zone shall be maintained at least 20 feet outside and
50 feet inside the security barrier. A perimeter patrol road will be
located inside this barrier. A sectionalized intrusion detection
system designed to be self-checking and tamper-indicating will be
located along the barrier with sensors located on or between it and
the patrol road. A closed-circuit television (CCTV) system using low-
47 light level cameras with zoom lens and remote pan and tilt controls
as required will be used to provide a means of promptly viewing the
sector or general area involved. Proprietary Figure 13.7-3 indicates
compliance with ANSI N18.18-1973, Section 3-3.

13.7.2.5 Vital Equipment and Vital Areas

All vital equipment and material access areas shall be located
within a vital area or building which, in turn, shall be located within
47 a protected area. Doors and gates to vital areas and to other selected
sensitive areas shall be kept closed and locked at all times when the
areas are not occupied. Proprietary Figure 13.7-4 indicates compliance
with ANSI N18.17-1973, Section 3.4, and other applicable guides and
regulations. The material access area is located within the Reactor
Service Building. No activities other than those which require access
to special nuclear material (SNM) or equipment employed in the process,
use, or storage of SNM will be conducted in the material access area. 31

As construction nears completion and the equipment made operational, the doors and gates to vital areas shall be identified by signs which state that entry through them shall be with the permission of the shift engineer on a need basis. Upon completion of construction, these doors and gates plus others, including some exterior doors and the Power Storeroom shall be controlled by a cardkey access control system.

The cardkey system shall be self-checking and tamper-indicating and an emergency power source provided. The regular power supply and emergency supply will be supervised and the operation of each cardkey controlled door tested no less frequently than once each seven days.

All issues of cardkeys will be authorized by the Plant Manager according to individual needs of employees requiring access to areas controlled by the cardkey system. Each card in the

system will be programmed individually and can be programmed out at any time if lost. The cards will be issued and returned daily to insure that they do not leave the site. Also, since the card will be required in the performance of the employee's duties, this will serve as a continuing availability check of issued cards.

13.7.2.6 Alarm Station

All intrusion alarms will terminate on a graphic security intrusion display panel and annunciate in a continuously manned central alarm station located within the protected area and in at least one other continuously manned station (see Proprietary Figure 13.7-4).

Each sector of the outdoor intrusion detection system and the operation of each cardkey controlled door will be tested no less frequently than once each seven days. Onsite and offsite communication facilities and the CCTV system will be tested at the beginning of each PSS work shift.

This onsite alarm station shall be considered a vital area. All intrusion alarms, emergency exit alarms, and other alarms will be required when purchased to meet the level of performance and reliability specified by Interim Federal Specifications W-A-00450B, GSA-FSS, dated February 6, 1973.

13.7.2.7 Security Barrier

The security barrier shall consist of an 8-feet high No. 9 gauge chain link fence (7-feet fabric and 3 strands of barbed wire on angle brackets) or other equivalent arrangement such as masonry walls. Other fencing located within the protected area may be only 6 feet high without barbed wire. The alignment of the new security barrier will have a minimum number of angles and curves to facilitate effective observation and maximum length sectors of the intrusion detection system.

13.7.3 Security Plan

The Plant Physical Security Plan shall describe security measures used to minimize the potential for industrial sabotage including access control, surveillance of vital equipment, and plans for responding to security threats in more detail than covered in the following paragraphs.

13.7.3.1 Access Control

The CRBRP shall have a perimeter security barrier that encloses all vital areas. The plant shall have two portals for normal access:

(1) a personnel portal and (2) a nearby vehicle gate. General public visitors shall not permitted inside the security barrier. Employees and special visitor's parking areas shall be located outside the security barrier. Vehicle access shall be limited to those required for delivery of material, operations, maintenance, and security of the plant. Persons, packages, and vehicles shall be subject to search upon entering, leaving, and while within the plant area.

There shall be a minimum number of outside accesses to the
47 nuclear island buildings. All of them shall have penetration resistant
doors with frames, hinges, and locks or security devices designed to
prevent forced entry and shall be alarmed or cardkey controlled. These
doors shall be kept locked or secured when not in use. Also, a number
47 of interior doors shall be cardkey controlled to prevent unauthorized
access to certain more important areas.

All persons authorized to enter the protected area unescorted shall have had a satisfactory security check and emotional stability check and shall have completed as a minimum a brief plant indoctrination and radiation protection course which describes plant organization and layout, controlled zones, radiation and contamination hazards, exposure limits and controls, elementary health physics, and pertinent sections of the site emergency plan.

Even those persons who are authorized unescorted access shall have their movement limited by physical barriers, such as locked doors, to prevent them from entering areas containing vital equipment or areas of high radiation levels. Only those who need access to these areas shall be provided means of entering.

When special visitors and other persons who have not completed
50 this training enter the protected area, they shall be escorted by an
employee trained in radiation protection and plant emergency procedures. The escort shall be responsible for the safety and action of the people in his charge until he checks them out of the access portal.

50 13.7.3.2 Control of Personnel by Categories

Employees and visitors authorized unescorted access to the plant protected areas by the Plant Manager will be issued a photo-type identification (ID) badge with tamper-resistant features. These persons will be identified, issued a radiation detection badge and dosimeters, and then be admitted to the protected area by the on-duty PS officer. Other persons not issued ID badges who require escorts may be identified by personal recognization, TVA identification card, or other available identification media. They will be issued a white
numbered visitor's badge to be worn on their outer garment while 1
within the plant protected area.

During construction, temporary fencing and other physical barriers will separate the operating unit and associated equipment from construction activity to prevent uncontrolled access by construction workers into operating unit areas. The only unescorted construction workers who may be inside the confines of the operating unit are those selected to perform maintenance or other work before final acceptance of equipment. Upon being granted unescorted access by the Plant Manager, these persons will be issued a photo ID badge and given a brief security indoctrination course covering the evacuation procedure
50 and relevant sections of the site emergency plan.

Contractor personnel, manufacturers' representatives, and other special visitors who require access to the plant shall be logged in and badged by the PS officer on duty. The officer shall then call the appropriate plant supervisor and arrange for an escort.

50 13.7.3.3 <u>Access Control During Emergencies</u>

Upon hearing of an emergency, the PS officer on duty at the access portal shall lock all doors to ensure controlled entry and exit. Special visitors who are onsite shall be escorted to the access portal. Plant employees shall report the predesignated stations from which they will be dispatched as needed to combat the emergency. All access control procedures will be compatible with the CRBRP Radiological Emergency
50 and Contingency Plans.

13.7.3.4 <u>Surveillance of Vital Equipment and Material Access Areas</u>

Unit operators shall continuously monitor the status of plant systems and equipment by means of annunciators, indicating lights, indicators, and recorders. New equipment or material shall be inspected on delivery. Operating logs and computer printout data shall be periodically examined for changes in equipment performance. Most equipment will be in continuous operation and any change will immediately be detected by the operator. Standby and emergency equipment shall be periodically tested on a routine basis as required by the technical specifications. Assistant unit operators shall inspect equipment and spaces at least once each shift. In addition, the assistant shift engineers and other supervisory personnel knowledgeable in plant 1
conditions shall make frequent unsecheduled inspection tours through the plant. Procedures shall be employed to control access to the vicinity of the material access area. In addition, activities in the vicinity of the material access area will be monitored. The combina- 31
tion of these efforts should provide reasonable assurance that unauthorized physical changes in the status of components of equipment 1
will not be undetected for long periods.

Key operating log sheets and selected recorder tracings shall be reviewed daily except for weekends by the Plant Engineering Section. Abnormal changes observed shall be called to the attention of the Plant Manager and the appropriate supervisors for investigation and corrective action, if required. This operational audit shall serve to assure early detection of physical changes which would have a significant bearing on plant performance.

13.7.3.5 Potential Security Threats

Should an unauthorized person succeed in entering the protected area, the access control measures in use would not allow him access to vital equipment. Operating personnel trained to be alert for unauthorized persons would recognize him as an intruder and arrange for his apprehension by a PS officer.

Plans shall be prepared to cover actions in the event of civil disturbance, emergencies, and bomb threats. Detailed emergency procedures shall be provided plant employees so that they may cope with these and other events in the optimum manner possible.

If there appears to be a real threat of civil disorder or another type of serious security threat to the plant or radiological emergency, all off-duty PS officers shall be recalled, and additional PS officers in the area shall be called in. Local and State law enforcement authorities shall be contacted for assistance. Arrangements
50 with federal, State, and local law enforcement agencies will be addressed in the CRBRP Radiological Emergency Plan. In the plant, precautions shall be taken to protect vital areas from threats of fire or other damage. When appropriate, the Plant Manager will make a written report to NRC.

Bomb threats normally come by telephone. Employees who might receive such a call shall be trained to extract as much information as possible from the caller. Based on this and other information, action would be taken to search for the bomb, evacuate areas, shutdown the reactor, or take any other actions deemed necessary to protect the plant and personnel.

More detailed descriptions of decisions/actions regarding potential security threats shall be included in the CRBRP Contingency
50 Plan.

13.7.3.6 Administrative Procedures

In the event of an incident of suspected sabotage or condition which threatens the security of the plant, the Public Safety Service shall immediately notify the Plant Manager and initiate a thorough 1

investigation. A report shall be prepared which includes as a minimum the cause of the event, extent of damage, if any, and action taken to prevent recurrence of similar event. Copies of the report shall be sent to the Plant Manager; Chief, Nuclear Generation Branch; Power Security Section; Chief, Public Safety Service Branch, P&SVS; and the Division of Law. When appropriate, the Plant Manager shall also report the situation to NRC.

Representatives of the Power Security Section and Public Safety Service Branch, P&SVS, shall make an annual audit of the CRBRP Physical Security Plan for adequacy of content and performance. Based on their audit, they will make recommendations for revising and updating the plan and related plant procedures.

13.7.3.7 Test and Inspections

This information will be supplied in the CRBRP Physical Security Plan. 1

TABLE 15.5.2.4-1

OFF-SITE DOSES FROM COVER GAS RELEASE
DURING REFUELING

		Dose (REM)*	
		Site Boundary (2 hr. -0.42 mi.)	LPZ (30 days-2.5 mi.)
	D_β (Skin)	4.0×10^{-3}	1.1×10^{-3}
49	D_γ (Whole Body)	4.4×10^{-3}	1.2×10^{-3}

49 * Integrated exposure based on puff release.

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15.7.2.3 Generator Breaker Failure to Open at Turbine Trip

15.7.2.3.1 Identification of Causes and Accident Description

In the event of a turbine trip, the generator load break switch is automatically opened by a signal from the turbine trip logic. The turbine trip logic simultaneously causes the generator field breaker to open regardless of whether or not the generator load break switch opens. A generator load break switch failure can occur from electrical or mechanical failure of the tripping mechanism.

15.7.2.3.2 Analysis of Effects and Consequences

If the generator load break switch fails to open after a turbine trip, a Plant Power Supply lockout is initiated. The lockout initiates the disconnection of the Plant Power Supply by tripping the appropriate 161 KV circuit breaker in the Generating Yard. This causes loss of the Preferred AC Power Supply as described in Section 8.2.1.1. Upon loss of power from the Preferred AC Power Supply, the Normal AC Distribution System and the Safety-Related AC Distribution System, are automatically transferred to one of the Reserve Transformers as described in Section 8.3.1.1.4. The reactor can be shut down with no adverse consequence, as described in Section 15.3.1.5, which evaluated the effects of a turbine trip.

15.7.2.3.3 Conclusions

The consequences of a turbine trip with subsequent failure of the generator load break switch to open is negligible, since one offsite power supply is still available to the AC Power Distribution System.

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15.7.2.4 Rupture in RAPS Cryostill

15.7.2.4.1 Identification of Causes and Accident Description

The RAPS cold box contains the cryogenic still in which krypton and xenon are extracted from the reactor cover gas stream. During normal operation, this stream is collected in the surge vessel in the RCB, flows at a controlled rate of 10.0 scfm into the cold box, which is in the RSB, and then through the cryostill. The krypton and xenon isotopes, both stable and radioactive, are condensed in the liquid argon still bottoms. The argon is condensed to liquid, in turn, by coiled tubing through which liquid nitrogen is passed. This coil penetrates the cryostill wall at two locations, sealed by welding. A complex rupture at either of these locations could vent both the still interior and the liquid-nitrogen line to the cold box, and hence to the cell atmosphere. Although such a complex rupture is not expected to occur, it would result in both a significant activity release and a significant increase in cell-atmosphere pressure. This postulated accident determines the RAPS cold box cell leak-tightness requirements. It is presented in order to report the maximum credible resultant doses to unrestricted areas, and the indicated cell leakage specification.

15.7.2.4.2 Analysis of Effects and Consequences

For the purpose of the accident analysis, it is conservatively assumed that the reactor has been operating sufficiently long, with gaseous fission products from 1% failed fuel, for steady-state isotopic composition to exist in the cover gas system. It is assumed, also conservatively, that the cryostill has not been off-loaded to the noble gas storage vessel for one year (maximum period) and therefore contains a maximum inventory of radioactivity. The accident is the rupture of a liquid nitrogen line at the cryostill wall in such a manner as to breach the wall also. This rupture would release liquid nitrogen, liquid argon, and reactor cover gas flowing from the surge vessel into the cold box. The cold box vents to the cell under a slight pressure differential.

The volume of nitrogen released to the cell corresponds to 10 minutes of maximum nitrogen flow; after this time, the nitrogen is automatically valved off by a cell pressure signal. 7300 scf of nitrogen are thus estimated to be released into the cell at the initiation of the incident. Also released at this time is the liquid still bottoms, 1.5 cu ft., which corresonds to 1275 scf of argon.

Redundant radiation monitors, located in the RAPS cold box cell, will
49 sense the presence of radioactivity, sound an alarm, and initiate a signal which will close cell isolation valves. The signal will not close the valve which allows the cell to vent to CAPS since it is a normally open valve. Therefore, the cell will normally continue venting to CAPS after the accident. However, for this analysis, the valve is assumed to be closed in this accident scenario requiring the cell to have a tighter 36
50 leakage specification.

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The most critical shutoff valve in this system is located on the inlet side of the cold box. For purposes of this analysis it is assumed that in addition to the above incident, this valve fails to close. As a result of the alarm, the operator has an assumed 30 min. to take alternative corrective action. One such action is to close this line by resetting the flow control valves, located between the surge vessel and the cold box, to zero flow. During the maximum operator response period of 30 min., radioactive argon will continue to flow at the normal rate of 10.0 scfm from the surge vessel to the cold box and into the cell through the break. This will result in an additional 300 scf of gas being released into the cell.

38

The assumed initial condition, then is that the gases from all three sources (liquid nitrogen, liquid argon and gaseous argon from the surge vessel) come instantly to standard temperature, but elevated pressure. No allowance is taken for radioactive decay during the pressure-rise period. The total amount of gas released into the cell (whose net volume is 8603 acf) is the total of the above or 17480 scf The resultant calculated initial pressure is 15.1 psig. The intital radioactivity inventory is shown on Table 15.7.2.4-1.

The design basis for the leakage specification is that the integrated 2-h site boundary dose (β-γ) be limited to a value below 10% of the 10 CFR 100 value following the rupture incident in the cold box.

15.7.2.4.3 Conclusions

The postulated RAPS cryostill rupture incident requires that the cold box be located in a controlled-leakage enclosure, with a permissible leakage rate such that the site boundary dose is below the CRBRP guideline value of 2.5 rem. The technical specification and testing provisions are discussed in PSAR Section 16.4.8. The analysis of the scenario described in Section 15.7.2.4.2 shows that a cell leakage specification limit of 29% of the cell volume per day at 15.1 psid will prevent the site boundary dose from exceeding 2.5 rem, in the very unlikely event of this accident. With this leakage specification, the calculated 2-h radioactive gas release to the environment is shown in Table 15.7.2.4-2. This cell will require testing, prior to plant startup and demonstration that the
49 leakage is within the specification. Additional testing of the cell will be required, if the cell is accessed. 36

269 018

TABLE 16.2-1

PLANT PROTECTION SYSTEM PROTECTIVE FUNCTIONS

Primary Shutdown System	Secondary Shutdown System
• Flux-Delayed Flux	• Modified Nuclear Rate
• Flux-$\sqrt{\text{Pressure}}$	• Flux-Total Flow
• High Flux	• Startup Nuclear
• Primary to Intermediate Speed Ratio	• Primary to Intermediate Flow Ratio
• Primary Pump Electrics	• Steam Drum Level
• Reactor Vessel Level	• Evaporator Outlet Sodium Temperature
50 • Steam-Feedwater Flow Mismatch	
• IHX Primary Outlet Temperature	• Sodium Water Reaction

269 019

Amend. 50
June 1979

Amendment 50

List of Responses to NRC Questions

Reference: NRC Letter Dated August 17, 1976

NRC Ques. No.
011.23

269 020

Question 001.301 (Chs. 3, 5, 9, 11)

Resolve the numerous inconsistencies in the ASME Code Classes designed for identical systems and components in Chapters 3, 5, 9 and 11 of the PSAR.

Response:

With the exception of the two items listed below, no inconsistencies have been identified in PSAR designations of ASME code classes for the PHTS, IHTS, SGS, SGAHRS, the Nuclear Island Heating, Ventilation, Cooling and Air Conditioning System, Recirculating Gas Cooling System, Chilled Water Systems, or Nuclear Island Treated Water Systems. Two inconsistencies have been resolved as noted below:

a) Table 5.5-7 has been revised to delete the alternative code classification and provide consistency with Table 3.2-5

b) Table 11.2-5 has been revised to indicate "Manufacturers Standards" for pumps per Regulatory Guide 1.26, Rev. 2, June 1975.

As indicated in the response to Question 001.274, the CRBRP Inert Gas Impuring Monitoring, EVST Cooling and Auxiliary Liquid Metal Systems requirements are revised to comply with the NRC position on Safety Classes. 50

269 021

1.0 Introduction

Sodium Leaks from the Primary Heat Transport System (PHTS) piping into the Reactor Cavity and the PHTS Cells and from piping within the Overflow and Primary Sodium Storage Tank Cells have been analyzed. These cells constitute the major cells within the Reactor Containment Building (RCB) both with respect to the size of the cells and the volume of sodium contained in the equipment within the cells. All of these cells operate with an inerted atmosphere (nitrogen) with a maximum of two volume percent oxygen. A spectrum of leaks has been considered for the analysis.

Leaks as small as 100 gm/hr can be detected by the sodium leak detection system in 250 hours and are indicative only of the initiation of a potential breach of the piping. Such leaks are so small, however, that no significant sodium is lost from the affected system and the impact on the cell temperature and pressure and the cell liners is negligible.

The smallest leak rates considered in the analysis presented herein corresponds to the piping Design Basis Leaks (DBL) that have been established for CRBRP. The definition of these leaks is discussed in PSAR Section 3.8-B and represents a conservatively established limit for the leak rates that may be expected to occur in the piping systems within the RCB that contain primary coolant sodium. As shown in Table Q001.581-1 for the three cells considered here, the DBL leak rates are 8 gpm or less.

In order to assess the sensitivity of cell design parameters (pressures temperatures, etc.) to leak rate and quantity, other leak rates were examined. A Moderate Energy Fluid Systems (MEFS) Leak has been defined which is equivalent to the leak size established in the NRC Standard Review Plan - Section 3.6.2. The leak size is defined as a circular opening with area equivalent to a rectangle which has dimensions of one-half the pipe diameter and one-half the pipe thickness. In addition, a still larger pipe break has been considered that results in a leak which approximates the maximum flow through the piping systems in the affected cells. This leak has been termed the Evaluation Basis Leak (EBL). In developing the spectrum of leaks considered, the operating characteristics of the plant were considered. The Design Basis Leak assumes that the leak detection system is functional and that operator actions will be taken to reduce and finally eliminate the source of the leak. The MEFS Leak scenario assumes that leaks in the primary piping which are large enough to reduce the sodium level in the reactor will activate the Plant Protection System (PPS) with resulting pump trip. However, the time of the trip is arbitrarily adjusted to maximize the resulting cell gas temperature and pressure. For the PHTS piping EBL leak it is conservatively assumed that the sodium leak rate is equal to the design sodium flow rate in the piping and that this flow rate is maintained until the maximum available system inventory is discharged through the break. No pump trip is therefore assumed. This is, of course, extremely conservative since the actual plant flow characteristics would tend to reduce the discharge rate

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and the quantity of sodium released. Actually, for a leak of this magnitude, the protection system would detect the failure and shut down the pump within a few seconds. For leaks in the Overflow and Primary Sodium Storage Tank Cells, leak detection systems are assumed operable and operator action is assumed. Other conservative assumptions used in the analyses concerning leak rate and leak volume are noted below. It should be noted that the location of all PHTS leaks in the Reactor Cavity is assumed to be the cold leg. The cold leg leak was selected because cold leg pressure is significantly higher than that of the hot leg ($\sim$100 psig vs $\sim$0 psig). Since the pipe leak geometrics are very close (an area within a factor of 2.5) for the hot and cold legs, leaks in the more highly pressurized cold leg result in significantly higher leak rates and thus higher cell atmosphere temperatures and pressures, even though the cold
50 leg sodium temperature is lower.

Also, the duration of the EBL leaks are slightly different in the RC than in the PHTS Cell. The reason for this difference is that slightly more sodium can be discharged into the RC than the PHTS Cells because of the piping elevations, but the leak rate is assumed to be the same for both.

The EBL leak rate is not as high as might be postulated for a double ended rupture of the PHTS piping. However, as shown in the response to NRC Question 001.700, additional flow, if it were available, would not
50 lead to higher cell pressurization.

In all cases, the elevation of the pipe break was chosen such that the maximum cumulative sodium release was considered. The maximum spray fire consequences were included by assuming that the entire discharge was completely converted into spray. This is, of course, unrealistically conservative for the larger leak sizes.

The effect of various geometric pipe break configurations has been included in the analyses which have been performed. The design basis leak considers a longitudinal crack in the pipe. The MEFS Leak assumes a circular hole in the pipe which has the same flow characteristics as a sharp-edged orifice. The EBL assumes that the break size has the same area as the original pipe. In all cases, it was assumed that the total discharge was converted into spray and the resulting droplets traveled through the entire height of the cell.

The analyses presented herein were performed to determine the transient temperatures and pressures imposed upon a PHTS cell, the Reactor Cavity and Cell 102A (In-Containment Primary Sodium Storage Tank Cell) for a spectrum of pipe leak sizes assuming the cell liners remain intact. To assess the consequences of a failed liner, a parametric study was also performed for the PHTS cell which assumed that portions of the liner system had failed. (The PHTS cell was chosen as a prototypic cell in which to demonstrate that the inerted lined cells in the containment have margin

to accommodate substantial failure of the liners.) A failure mode analysis of the liner system presented in the response to NRC Question 130.89 indicated that in the unlikely event of a liner failure, the extent of the failure would be very limited. Section 4 of this response presents an assessment of the consequences of a liner failure which includes the effects of sodium-concrete interaction.

The analyses were performed using three computer codes, SPRAY, SOFIRE and CACECO. For the conditions in which the liner was assumed to remain intact, SPRAY and SOFIRE were used to evaluate cell transients. For the evaluation assuming liner failure, all three codes were employed with the sodium-concrete-water reactions included in the CACECO analysis. The basic material properties included in the SPRAY and SOFIRE analyses are listed in Table Q001.581-2. The material properties for the CACECO code used in analyses of failed liners are presented in Table Q001.581-5.

Summary results of all the analyses are presented in Section 2. Detailed discussions of the analyses, on a cell-by-cell basis, are presented in Sections 3 and 4.

2.0 Summary

2.1 Summary Results - Cell Liner Design Conditions

Table Q001.581-4 presents the summary results of the analyses for the design conditions where cell liner integrity is maintained. For each cell and leak evaluated, the following peak transient values are itemized: gas pressure, gas temperature, liner temperature, floor structural concrete temperature and the non-wetted wall structural concrete temperature. The concrete temperatures provided in the table represent the temperatures in the first one-half inch of structural concrete behind the floor gravel aggregate and the wall insulating concrete. Note that the peak transient temperatures for the non-wetted wall structural concrete are not specified in absolute terms but rather as a temperature that this concrete will not exceed. This is necessary because it is not feasible to continue the SOFIRE analyses for the time the wall concrete passes through a peak temperature, as is the case for the floor concrete transient. However, by by examining those transients in the insulating concrete which have passed through their peak temperature and are decreasing, it is possible to estimate a peak temperature value that the wall structural concrete will asymptotically approach and thus to specify a temperature it will not exceed over the entire course of the transient. This method was used to specify the bounding temperatures in Table Q001.581-4 for the wall structural concrete.

It should also be noted that the pressures in the reactor cavity quoted for the MEFS and the EBL will be limited by the pressure venting system that is being installed to accommodate the Thermal Margin Beyond the Design
50 Base events. While provisions have been included for a venting system, the rupture disk burst pressure has not been selected. It is expected to

be in the range of 8-10 psig. The primary purpose of this study was to determine the potential pressures that may exist in these cells for a spectrum of postulated leaks. For this reason, use of the Reactor Cavity vent system or consideration of venting the PHTS cells or Cell 102A was not included in the evaluation.

It should be noted that the maximum pressures reached are less than the design pressures for the cells.

Three general observations based on the data summarized in Table Q001.581-4 follow:

- The transients associated with any of the DBLs are minor, both in terms of cell gas pressurization and in terms of structural concrete transients. Specifically, none of the DBLs result in a cell gas pressurization in excess of 1.1 psig. The structural concrete temperatures (floor and wall) do not exceed 120°F for any of the DBLs.

- For each of the cells considered, the MEFS Leaks result in peak gas pressures and temperatures approximately a factor of 2 less than the corresponding EBLs.

- For the PHTS Cell and the Reactor Cavity the structural concrete transients are essentially equivalent for either the MEFS or EBL. However, for Cell 102A, the EBL concrete structural transients are slightly (30°F-80°F) more severe than the MEFS concrete transients.

2.2 Summary Results - Evaluation with Cell Liner Failure

Cell Liners will prevent the interaction of sodium with the concrete cell structures in the event of all sodium spills. However, in response to NRC Question 130.89, a failure mode assessment was performed to determine a scenario that could lead to liner failure. Such a scenario was developed by assuming a series of very pessimistic conditions. The Failure Mode Evaluation results in the determination that the worst failure that can be postulated is a small crack in the liner floor, assumed to occur at or near a weld joint, which extends the entire length of the cell. The crack opening would not exceed approximately 0.25 inches in the PHTS Cell initially and would not appear until the pool of sodium began to cool down. For the conditions assumed in the response to NRC Question 130.89, the crack would not open for over an hour after the spill. During this time delay period, much of the water in the upper portion of the structural concrete will be released and vented through the liner venting system, thus reducing the amount of water available for reaction with the sodium pool following crack opening.

In order to accurately evaluate the effects of a liner failure, a very complex analysis would be required and probably a series of tests conducted to confirm the results. The sodium which leaks through the assumed liner crack would be contained in the aggregate. Since the aggregate is chemically inert to the sodium, it provides two major benefits. First, the aggregate will occupy approximately one-half the volume between the liner and the structural concrete. This will limit the amount of sodium that is available to react with the water released from the concrete. Since the liner failure is very small compared to the cell floor area, there will be little or no opportunity for fresh sodium to replace reacted sodium under the liner. Secondly, the aggregate will tend to hold the reaction products on the surface of the concrete, forming a partial barrier to further

cell atmosphere with sodium discharged after the peak is reached, but neglecting this and assuming only pool-oxygen reaction is conservative with regard to long-term structural transients and insignificant with regard to the cell atmosphere transients since the SPRAY analysis shows these transients to have already peaked.

Figure Q001.581-4 presents the results of the spray-phase transient analysis. As indicated, the peak cell pressure of 11 psig occurs at 3.5 minutes, corresponding to peak cell atmosphere temperature of 570°F. Figures Q001.581-5 through-8 present the longer-term cell transients based on the SOFIRE analysis.

Evaluation Basis Leak (EBL)

The PHTS Cell EBL is a spill of the total spillable volume in a loop (20,000 gallons). The spill rate is conservatively taken to be the normal loop flow rate of 33,500 gpm which yields a spill duration of 0.6 minutes.

The SPRAY analysis for this leak assumed that the entire discharge was in the form of 0.18" droplets at 1015°F and in a manner such that the spray occupied one-third of the cell volume. Figure Q001.581-9 presents the results of the spray-phase transient analysis. The peak cell atmosphere pressure and temperature of 23 psig and 1030°F, occur near the end of the sodium discharge and begin to decrease after the sodium discharge is complete. The entire PHTS Cell oxygen content is depleted by the end of the sodium discharge.

As discussed for the MEFS leak, the longer-term transients were evaluated with SOFIRE, with initial conditions corresponding to the peak spray-phase transient conditions. In this case, however, the oxygen concentration used for SOFIRE was zero, since all the oxygen is consumed during the spray. Figures Q001.581-10 through-13 present the longer-term cell transients based on the SOFIRE analysis.

The Evaluation Basis Leak (EBL) of the IHTS piping within a PHTS cell results in a maximum spill rate of 29900 gpm of 936°F sodium for a period of 30 seconds. When compared to the primary system EBL of 33,500 gpm of 1015° sodium for 35.5 seconds, it is noted that each of the above parameters for the primary system leak envelopes comparable parameters for the intermediate system. The analysis results presented for the primary system EBL therefore conservatively envelopes the intermediate system EBL
50 in the PHTS cell.

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3.2 Reactor Cavity

Design Basis Leak

The Reactor Cavity piping DBL is described in Table Q001.581-1. The total duration of the DBL is 410 minutes and the total sodium injected into the cavity is 390 gallons; 52% of this sodium is injected during the first 40 minutes of the transient after which operator action can be assumed to have tripped the reactor and pumps.

The basic method used to evaluate the transients resulting from this leak is identical to that described for the PHTS DBL. The only principal difference is that the Reactor Cavity DBL originates from a cold-leg piping fault so that the temperature of the injected sodium is taken

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as 750°F, the peak cold-leg temperature.

Figure Q001.581-14 presents the Reactor Cavity atmosphere pressure and temperature transients. The structural transients resulting form the DBL are presented in Figures Q001.581-15 and -16.

Moderate Energy Fluid System Leak

The total duration of the Reactor Cavity MEFS Leak is 150 minutes and the total sodium injected into the cavity is approximately 20,000 gallons; 20% of this sodium is injected in the first 5 minutes after which it can be conservatively assumed that the Plant Protection System has shutdown the plant. The Reactor Cavity MEFS Leak originates from a cold-leg piping fault and the temperature of the sodium injected is specified as 750°F, the peak cold-leg sodium temperature. The same analysis procedure as described for the PHTS Cell MEFS Leak was used to evaluate this leak.

Figure Q001.581-17 presents the results of the spray-phase transient analysis. The peak Reactor Cavity atmosphere pressure and temperature, 9.8 psig and 560°F, occur at approximately 2.5 minutes. The pressure and temperature decline gradually out to 4 minutes, the end of the maximum MEFS Leak rate (880 gpm), and begin to decrease sharply at 5 minutes when the MEFS Leak rate decreases to 120 gpm. Figures Q001.581-18 through-21 present the long-term Reactor Cavity transients based on the SOFIRE analysis.

Evaluation Basis Leak

The Reactor Cavity EBL is a spill of the total spillable sodium in a loop (20,000 gallons). The spill rate is conservatively taken to be the normal loop flow rate of 33,500 gpm, which yields a spill duration of 0.6 minutes. The Reactor Cavity EBL originates from a hot-leg piping failure and the temperature of the sodium injected is taken as 1015°F, the maximum hot-leg sodium temperature. The same analysis procedure as described for the PHTS Cell EBL was used to evaluate this leak.

Figure Q001.581-22 presents the results of the spray-phase transient analysis. The peak Reactor Cavity atmosphere pressure and temperature, 21 psig and 1020°F, occur at approximately 15 seconds. The Reactor Cavity oxygen is also depleted at roughly 15 seconds. For the EBLs, which are essentially identical for both the Reactor Cavity and PHTS Cell, oxygen depletion occurs more rapidly in the Reactor Cavity principally because the Reactor Cavity free volume is only 50% as large as that of the PHTS Cell, so that oxygen available in the Reactor Cavity is roughly one-half that available in the PHTS Cell.

The longer-term Reactor Cavity transients, based on SOFIRE analysis, are provided in Figures Q001.581-23 through-26.

3.3 Overflow and Primary Sodium Storage Tank Cell

Design Basis Leak

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Question 011.23 (11.3.2.1)

In Subsection 11.3.2.1, you discuss the procedure for periodic bottling of Ar-39 and Kr-85 from the RAPS cryogenic still. Discuss the procedure in greater detail; provide bottle storage pressure; discuss procedures and the means for monitoring leakage of gas from the storage bottles; provide the anticipated onsite storage time; describe the shipping container to be used for transport of the storage bottles to a licensed burial site; and discuss the acceptability of bottled radioactive cases at the licensed burial sites.

Justify your conclusion that bottling, shipping and ultimate storage of the long-lived gaseous radioisotopes (Kr-85 and Ar-39) represents a lower risk to public health and safety than releasing these isotopes under controlled and favorable conditions to the environment. Include your consideration of keeping occupational exposures as low as practicable.

Response:

The procedure for disposing of the RAPS cryostill bottoms is discussed in revised PSAR Sections 11.3.2.1 and 11.3.4. The procedure involves controlled gradual release of the noble gases through CAPS during normal operation.

Considerations of keeping occupational exposures as low as reasonably achievable supported the change to the method for disposal of the RAPS cryostill bottoms as described in PSAR Sections 11.3.2.1 and 11.3.4

269 030

Question 222.77 (7.8.1, 7.5-1) (RSP)

Provide a Table similar to Table 7.5-1 listing all instrumentation intended for accident and post-accident monitoring of plant parameters necessary to follow the course of an accident and to achieve a safe shutdown. Branch Technical Position EICSB 23 of Appendix 7A of the Standard Review Plan provides guidance and requirements for the qualification of this type of instrumentation. We will require compliance to these requirements. Moreover, you should provide a discussion outlining and justifying your rationale for the selection of parameters for this group of monitoring instrumentation.

Response:

Section 7.5.10 provides a discussion of instrumentation provided to enable the plant operator to assure that the plant is maintained in a safe shutdown status. Table 7.5-4 lists the parameters monitored to perform this function. Compliance to the qualification requirements of EICSB 23
50 will be carried out as outlined in the response to Question 222.54.

269 031

Question 310.3 (6.4)

Describe the physical location of the outside air intakes for the control room ventilation system. Indicate the locations on plant layout drawings (plan and elevation views).

Response:

The interim response to this question stated that the need for a second Control Room air intake would be evaluated based upon the radiological dose rates, Control Room leakage rate, plant effluent release point locations and site meteorological conditions. To insure Control Room habitability following extremely low probability accidents which are beyond the design basis, two widely separated intakes are provided. One Control Room air intake will be located at the SW corner of the Control Building roof
49 at approximately elevation 880' and the other one will be located at the NE corner of the Steam Generator Building Auxiliary Bay roof at approximately elevation 858'. The selected air intake locations are based on the following:

(1) Control Room Filter Units
   (a) 500 CFM outside air intake through charcoal/HEPA filter train for 1/4 inch W.G. Control Room pressurization.
   (b) 8,000 CFM Control Room air recirculation through same charcoal/HEPA filter train, as (a) above.
   (c) Redundant charcoal/HEPA filter trains with 95% charcoal and 99.97% HEPA filter efficiencies.

(2) Two door vestibules for all Control Room exits/entrances.

(3) 3 CFM unfiltered air infiltration based on Item 2 above.

The following new and revised sections, tables and figures indicate revisions to the design basis of the Habitability System, the addition of redundant toxic chemical and smoke detectors in the Control Room air intake duct, the increase in size of the Control Room filter trains, the deletion of water sprays for the charcoal filter banks, and the conformance to Regulatory Position 4d of Regulatory Guide 1.52:

(a) Revised Section 6.3.1.1

(b) Revised Section 6.3.1.2

(c) Revised Section 6.3.1.3

(d) Revised Section 6.3.1.5

(e) Revised Table 6.3-1

(f) Revised Section 9.6.1.2

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(g) Revised Section 9.6.1.3.1.

(h) Revised Section 9.6.1.3.4.

(i) Revised Table 9.6-1

(j) Revised General Arrangement Drawing 1.2-72. 22 | 50

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Question 331.20 (12.3)

Describe in detail procedural and other controls that would prevent the spread of radioactive contamination from the Reactor Service Building to the cold laboratory and counting room (Figure 1.2-46), and the office areas. Justify placement of the hot laboratory adjacent to the counting room, and the existance of a direct corridor to the office area, apparently not requiring passage through the change area.

Response:

Additional information responding to this question is provided in revised sections 12.1 and 12.3. 33

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269 034

Question 422.1 (13.3)

10 CFR Part 50, Appendix E, requires that the Preliminary Safety Analysis Report contain sufficient information to assure the compatibility of proposed emergency plans with facility design features, site layout, and site location with respect to such considerations as access routes, surrounding population distributions, and land use. To this end, we request that you submit an analysis which includes information and findings which will be needed to assure adequacy of emergency planning with respect to the protective measure of evacuatic of persons from the exclusion area and from any potentially affected sector of the environs, as follows:

(1) Plots showing projected ground-level doses, for both whole body and thyroid, resulting from the most serious design basis accident analyzed in the Safety Analysis Report. These should be based on the same isotopic release rates to the atmosphere and the same (Gaussian) dispersion model as are acceptable for use in Chapter 15 of the PSAR for the purpose of showing conformance to the siting dose criteria of 10 CFR Part 100. Data inputs to the atmospheric dispersion model should be consistent with those used and acceptable for siting dose (10 CFR Part 100) calculations. Plume front transit times, radioactive decay in transit, and dose conversion calculations may be incorporated on a physically realistic basis or a conservatively simplified basis. The bases should be fully described. Note that the plots furnished in response to this item may require revision if the core disruptive accident is, at some later date, considered a design basis accident (DBA) and its consequences are more severe than the most serious DBA analyzed in the current PSAR. Present the data in the following format:

    (a) Use a log-log scale with time (hour) following onset of release as the ordinate, and distance (miles) from the release point as the abscissa.

    (b) Provide curves for whole body doses of 1, 5, and 25 rem, and thyroid doses of 5, 25, 150, and 300 rem. Each curve should represent the elapsed time to reach the specified dose level as a function of distance from the release point.

    (c) Extend each curve from an ordinate of 2.0 hours or from an abscissa equal to the exclusion radius, to an ordinate of 8.0 hours or to an abscissa equal to the LPZ outer boundary, whichever results in the greatest range of coverage.

(2) The expected accident assessment time which includes the time required to identify and characterize the accident, the time required to predict the projected doses resulting from the accident, the time to notify offsite authorities, and the time required to determine the appropriate protective measures from the affected areas. Include sufficient information to support your estimate.

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(3) An estimate of the time required to notify all persons within the potential evacuation area determined in (4) below, and the means assumed for such estimate.

(4) An estimate of the evacuation times, measured from the time of initial warning, to remove persons from the exclusion area, and from <u>each</u> "sector", or increments thereof, of the environs out to a distance determined by the 8 hour terminus of the 5 rem whole body dose curve, the 25 rem thyroid curve, or the outer LPZ boundary, which ever is the greatest. From a practical viewpoint, the "sectors" chosen for analysis may be bounded by certain geographical or manmade features, but should cover an arc of at least 45°. If means for effecting the physical evacuations other than the use of private automobiles are used in estimating any of the foregoing evacuation times, these should be specified. Population data should include both resident and transient persons, including those resulting from the facilities described in Chapter 2 of the PSAR, at levels projected at approximately the time the plant is scheduled to commence operation. Provide a figure showing the combined maximum daily resident and transient population for each 22-1/2° sector in increments of one mile concentric rings from the plant out to a distance as determined above.

(5) Identify any special evacuation problems that arise from the nature of the activities associated with the operation of the Oak Ridge National Laboratory and the Oak Ridge Gaseous Diffusion Plant.

(6) Provide a map or figure showing <u>all</u> roads available for evacuation of the plant environs out to a distance of at least 2 miles beyond that determined in (4) above. Identify the character of each road shown with respect to size, number of lanes, surface characteristics, and other factors which may affect vehicular traffic capacities.

<u>Response</u>:

(1) The information requested in Part 1 of this question is provided in Section 13.3.12.1. 11

(2-6) Parts 2 through 6 of this question were restated in Question 422.2 and are answered in the response to this question. 40 50 35

269 036

Question 422.2 (13.3)

10 CFR Part 50, Appendix E, requires that the Preliminary Safety Analysis Report contain sufficient information to assure the compatibility of proposed emergency plans with facility design features, site layout, and site location with respect to such considerations as access routes, surrounding population distributions, and land use. To complete our evaluation of the feasibility of carrying out appropriate protective measures, an analysis is normally made to determine if there is reasonable assurance that evacuation of persons in the environs of the plant can be accomplished within a time frame such that the potential consequences resulting from the most serious design basis accident analyzed for siting purposes would not have an adverse effect upon the public health and safety. At this time, the above analysis cannot be finalized in the absence of a source term acceptable to the NRC staff. However, the portion of the analysis dealing with the evacuation aspects can be completed. The following is therefore requested:

(1) Provide the expected assessment time for a serious design basis accident having offsite consequences. This should include the time required to identify and characterize the accident, the time required to predict the projected doses resulting from the accident, and the time to notify offsite authorities. Include sufficient information to support your estimate.

(2) Provide an estimate of the time required to warn all resident and transient persons within the LPZ, and the means to be used for such warning. Your response should take into account the following facilities within the LPZ:

(a) Thirteen Recreation Areas as identified in Table 2.1-14, including the stockcar track.

(b) U. S. Nuclear, Inc.

(c) Nuclear Environmental Engineering, Inc.

(d) Nuclear Assurance Company

(e) Oak Ridge Gaseous Diffusion Plant

(f) Edgewood Elementary School

(g) Oak Ridge National Laboratory

(h) Melton Hill Dam

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(3) Provide an evaluation of the evacuation times, measured from the time of initial warning, to remove resident and transient persons from the exclusion area, and from each "sector" of the LPZ. From

a p.actical vie..oin+, the "sectors" chosen for analysis may be bo..nded by certa geographical or manmade features, but should cov.. an arc of a. least 45°. If means for effecting the physical evacuations other than the use of private automobiles are used in estimating any of the foregoing evacuation times, these should be specified, e.g., methods for mass evacuation of the Edgewood Elementary School. Population data should include both resident and transient persons at levels projected at approximately the time the plant is scheduled to commence operation. Note that the peak hour transient population at the Melton Hill Dam was omitted from the PSAR. This number should be provided and included in the analysis.

(4) Provide a map or figure showing all roads available for evacuation of the plant environs out to a distance of 10 miles. Identify the character of each road shown with respect to size, number of lanes, surface characteristics, and other factors which may affect vehicular capacities.

(5) Special problems would be associated with implementing the protective measure of evacuation for some of the facilities identified in (2) above. Discuss your planning for coping with emergencies, including acceptable alternatives to evacuation, in the light of the possibility that

(a) safeguards measures for the continued protection of SNM at Clinch River Consolidated Industrial Park facilities and at ERDA facilities may not be compatible with evacuation of such facilities, and

(b) for other reasons certain facilities may not be left unattended.

Response:

Responses to the five (5) parts of Question 422.2 are provided in the following PSAR Sections. 42

Part (1)	Section 13.3.12.2.1
Part (2)	Section 13.3.12.2.2
Part (3)	Section 13.3.12.2.3
Part (4)	Section 13.3.12.2.3
Part (5)	Section 13.3.12.2.3

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Amend. 50
June 1979

Question 422.3

Figure Q422.2-1 provided in response to item 422.2(4) is not sufficiently legible and detailed to:

(1) identify all roads available for evacuation of the plant environs,

(2) identify road surface characteristics from the legend, and

(3) identify road widths and number of lanes.

Submit the figure and information previously requested in item 422.2(4).

Response:

The subject figure has been revised and re-incorporated as Figures 13.3-5, 6. Full-size maps of this figure have been provided to NRC under separate cover. 50

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