PAGE REPLACEMENT GUIDE FOR

AMENDMENT 59

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

Transmitted herein is Amendment 59 to Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report, Docket 50-537. Amendment 59 consists of new and replacement pages for the PSAR text.

Vertical lines on the right hand side of the page are used to identify question response information and lines on the left hand side are used to identify new or changed design information.

The following attached sheets list Amendment 59 pages and instructions for their incorporation into the Preliminary Safety Analysis Report.

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AMENDMENT 59

QUESTION/RESPONSE SUPPLEMENT

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Q323.26-1	Q323.26-1
Q323.35-1	Q323.35-1
Q323.37-1	Q323.37-1

the secondary shutdown system is arranged using a general coincidence logic. These logics are described in Section 7.2.1. Primary and secondary systems are electrically and mechanically isolated. Sufficient redundancy is included within each system to assure that single random failures will not degrade protection by either system.

1.2.7 Auxiliary Systems

The Auxiliary Liquid Metal System provides the facilities for receipt, storage and purification of all liquid metal used in the CRBRP. It also provides the capability for controlling reactor sodium level variations, accommodates primary sodium volumetric changes, provides cooling for the core components stored in the Ex-Vessel Storage Tank (EVST), and by means of the Direct Heat Removal Service (DHRS) gives a means of long term reactor decay heat removal that is independent of the
26 intermediate heat transport system and steam generator system loops.

The Compressed Gas System processes ambient air to provide compressed dry air for pneumatic instruments, maintenance systems, unloading devices, tooling, and miscellaneous cleaning and inspection services. This system provides for sodium removal systems and as required for plant usage.

The Recirculating Gas Cooling System provides cooling service to cells and equipment located in the Reactor Containment Building and the Reactor Service Building.

15 The Chilled Water Systems provide heat removal capability from certain equipment and areas in the Reactor Containment Building and the Reactor Service Building.

The Inert Gas Receiving and Processing System (IGRPS) provides inert gases as required by other systems of the CRBRP, including cover gas, cell inerting atmosphere, valve actuation gas in inerted cells, cooling
59 gas, gas for certain seals, for component
cleaning and other services, and vacuum for out-gassing and gas-collection purposes. In addition, the IGRP System provides for the control of reactor cover gas radioactivity and for the processing of gases to be released from the system to remove their contained radioactivity.

The Impurity Monitoring and Analysis System provides for the sampling, monitoring, and analysis of the sodium, NaK, and argon cover gas systems in the plant, and acceptance sampling and analysis of incoming
56 sodium, NaK, argon, and nitrogen.

The Treated Water System includes the domestic (potable) water system, the closed cooling water system, water (makeup) treatment system and the cooling water makeup system.

The River Water Service System handles and treats river water for the plant. The system includes the river water pumps and piping, intake filtration equipment and the plant service water system.

The Heat Rejection System provides the heat sink using the main
cooling tower for waste heat loads from the turbine condensers, and from the
various plant auxiliary and service systems such as sodium pump oil coolers,
air conditions, air compressors, pump coolers and the turbine oil coolers.
41 44 The Emergency Plant Service Water System emergency cooling tower structure pro-
vides the heat sink for the safety related components listed in Table 9.9-3. De-
33 tails of the auxiliary system are given in Chapter 9.

1.2.8 Refueling System

The reactor core is designed to be refueled annually. Under
equilibrium conditions, all fuel and inner blanket assemblies are replaced
as a batch every two years, with a planned mid-term interchange of 6 inner
41 blanket assemblies for 6 fresh fuel assemblies designed to add sufficient
20 excess reactivity to the system to complete the (550 fpd) burnup. The
radial blanket assemblies in the first and second rows are replaced as a
51 59 44 batch at 4 and 5 year intervals, respectively.

The In-Vessel Handling Subsystem (IVHS) provides for the transfer of
44 core assemblies in the reactor vessel, between their normal positions in the
41 reactor core and storage positions outside the core accessible by the Ex-
vessel Transfer Machine. The major equipment comprising the IVHS are the In-
Vessel Transfer Machine (IVTM), Auxiliary Handling Machine (AHM), AHM Floor
Valves (FV), IVTM Port Adaptors, and associated maintenance and storage facili-
ties and equipment. The IVTM is installed in the small rotating plug in the
reactor head after reactor shutdown. The machine raises or lowers core
44 assemblies by means of a grapple. Translation to a new position is by rota-
41 tion of the reactor head rotatable plugs. The AHM is used to install and re-
44 move the control rod drivelines, port plugs, and in-vessel section of the IVTM
in the reactor. The port adaptors and floor valves provide a means for closure
of the reactor and storage ports during the time the transfer of refueling equipment
59 in preparation for refueling operations.

The Ex-Vessel Handling Subsystem (EVHS) provides for the transfer
of core assemblies between the reactor, the Ex-Vessel Storage Tank (EVST),
41 44 and the Fuel Handling Cell (FHC) located in the Reactor Service Building (RSB). The
system consists of the Ex-Vessel Transfer Machine (EVTM) mounted on a Gantry-
44 Trolley (G-T), EVTM Floor Valves (FV), Core Component Pots (CCP), port plugs
and adaptors, and associated maintenance and storage equipment and facilities.

The Ex-Vessel Storage Subsystem (EVSS) consists of the Ex-Vessel
59 Storage Tank (EVST). The EVST is a sodium-filled tank used to store and cool
44 spent fuel prior to shipment offsite, and to preheat new core assemblies. The
41 capacity of the EVST is about 650 assemblies.

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Amend. 59
Dec. 1980

The Conditioning and Service Subsystem (CSS) and Receiving and
Shipping Subsystem (RSS) consist primarily of the facilities necessary to un-
load, inspect and prepare the new core assemblies prior to loading in the
44 EVST; and to handle, inspect and load the spent core assemblies in shipping
41 casks for shipment off-site. The facilities include a Fuel Handling Cell
44 (FHC) which is a shielded inerted hot cell. The major equipment consists of
20 cask handling and transporting machinery.
59

Amend. 59
Dec. 1980

41 1.5.1.3.5 Fallback Position

57 In the event that operating with failed blanket assemblies cannot be shown to be satisfactory from a public safety viewpoint, the reactor may be required to shutdown when the blanket material is exposed to the sodium.

1.5.1.4 Sodium-Water Reaction Pressure Relief Test

41 1.5.1.4.1 Purpose

The principal concern associated with the large water to sodium leak in steam generators is potential system damage, principally to the IHX by propagation of transient pressure waves through the Intermediate Heat Transport System (IHTS). The objective of the Sodium Water Reaction Pressure Relief Subsystem (SWRPRS) is to relieve pressures from the IHTS and thereby protect the primary coolant boundary from damage in the region of the primary sodium to intermediate sodium heat transfer interface.

The approach to design of the CRBRP SWRPRS is to assume a realistic yet
59 conservative design basis water to sodium leak and to use a validated calculational method to predict pressure loads on the IHX. It is a design requirement that the IHX be able to withstand the sodium-water reaction pressures without compromising the primary coolant boundary.

A survey of available existing analytical methods was completed to select the best method for improvements consistent with CRBRP requirements. The TRANSWRAP computer program (Ref. 5) was selected for use in the CRBRP analysis. An improved version of this code was used to establish loads on the IHX for the reference design IHTS piping and component arrangement and the reference design SWRPRS. A design basis leak was assumed to consist of an Equivalent Double-Ended Guillotine (EDEG)* failure of one steam generator tube followed by the equivalent of two additional EDEG tube failures. The two additional failures occur as follows:

one EDEG failure occurs one second after the initial EDEG failure.

one additional EDEG failure occurs two seconds after the initial EDEG failure

This sequence is superimposed on a system which has been pressurized by an undetected moderate-sized leak to just below the rupture disk burst pressure. The
59 three tube EDEG failure is not intended to represent a realistic event, but rather it provides a basis for calculating conservatively large pressure loads for the design of IHX and the pressure relief system. Results of analyses using this basis are reported in Section5.5.3.6.

To increase confidence in assuring integrity of the primary coolant boundary even during a large sodium-water reaction, the development program will provide technical information which is not available for inclusion in the PSAR. The safety related objectives of the development program are:

a) to validate the computer program used to predict pressures in the IHX during a postulated sodium water reaction, and

b) to confirm that effects of the design basis leak assumed for determining pressures in the IHX are conservative.

41 1.5.1.4.2 Program

As part of the Steam Generator Development Program, AI
has constructed the Large Leak Test Rig (LLTR). The test programs
41 included pulling apart a notched tubular specimen in the sodium filled test
article to simulate a DEG failure. A steam/water mixture was forced through
44 41 the burst tube into the sodium. For most tests, surrounding tubes contained
stagnant, pressurized steam/water mixtures. In general, the development
44 effort provided technical information regarding the design of pressure
relief systems to handle unexpected water-to-sodium leaks.

Measured values of pressure at various locations in the test rig are
being compared with calculated pressures obtained using the modified TRANSWRAP
computer program to analyze the test rig and test article. It now appears that
the computer code predicts values of pressure that are either in agreement with
measurements or are conservatively large relative to measured pressures for the
test rig and test article. Thus, it appears that the analysis of CRBRP for
41 sodium-water reaction pressures using this code are being conservatively
44 accomplished. This conclusion is still under review and evaluation and there-
fore subject to adjustment as the remaining test data are examined.

Examination of the test article following intentional bursting of
44 a single tube gives some indication of the nature and extent of damage
propagation to other tubes. It is expected that the tests will demonstrate
47 that the calculated loadings from sodium-water reactions are conservative.

41 1.5.1.4.3 Schedule

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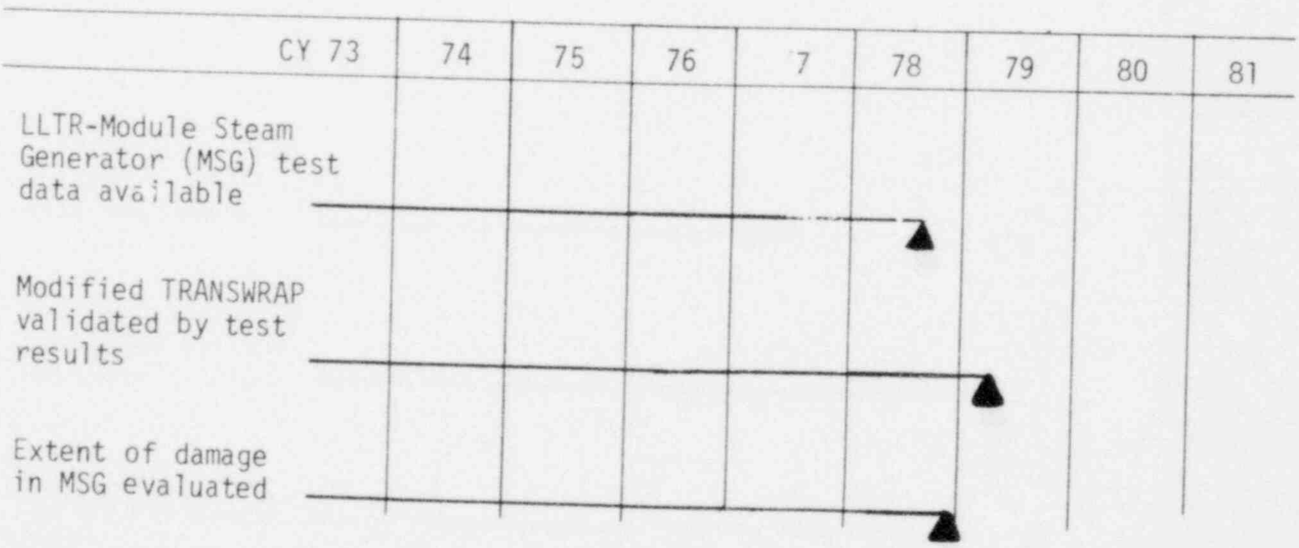

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Amend. 47
Nov. 1978

1.5.2.4 Core Restraint System Tests

1.5.2.4.1 Purpose

In order to provide both axial and radial support for the reactor assemblies, the core restraint system will: (1) maintain fuel, control, blanket and removable shield assemblies in predictable and reproducible positions; (2) assure no damage to the assemblies during refueling, and (3) prevent excessive operating loads from occurring. The key safety related feature is the ability to limit reactivity insertions arising from lateral
53 assembly motion to an acceptable level through control of residual load plane gaps throughout the core.

Variables affecting the core restraint system are the radiation induced
59 53 materials effects of swelling and creep, load pad dimensional and friction characteristics. Irradiation swelling and creep in conjunction with transverse temperature gradients in the core assembly ducts cause bowing deformations of the core assemblies during life. The restraint designs of fast reactors now in operation are among those concepts developed before the recognition of irradiation induced swelling and creep effects in core components. Prototype Fast Reactor and FFTF are among the transitional designs based on recognition of the problem and these systems accommodate design uncertainties due to limited characterization of materials effects. The CRBRP design, effectively the third generation, draws on the design and development experience of the previous two generations. To reduce sensitivity to mechanical
53 effects uncertainties, the following development program is in place.

1.5.2.4.2 Program

41 This program has been undertaken by Westinghouse at its ARD facility.
59 The performance of the CRBRP core restraint systems is being simulated in a full scale mechanical test facility. The facility provides the capability of
53 testing a full core array of simulated reactor assemblies. This three dimensional test effort will provide qualitative output for analytical studies, as well as quantitative core restraint system test data.

To provide data to the CRBRP in a timely manner tests consisting of a full core mock-up including fuel, blanket and two rows of removable shielding will be carried out. This will be accomplished through the use of simulated fuel assemblies designed to provide nominal duct bending stiffness,
53 load pad compliance, inlet nozzle clearance and contact surface frictional characteristics. This effort will consist of core compaction tests with bowed core assemblies and prototypical load pad friction during which duct bowing patterns based on analytical predictions for chosen material conditions will be simulated. Inter-assembly loading patterns and geometries will be determined by using multiple "instrumented" load measurement ducts. The compaction characteristics of the core will be determined for core restraint
53 system evaluations.

Amend. 59
Dec. 1980

1.5.2.4.3 Schedule

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CY	74	75	76	77	78	79	80	81
Full Core Tests								

59 41 53

1.5.2.4.4 Criteria of Success

The test effort is to provide core compaction characterization data and provide a basis for determination of residual load plane gaps within the core under prototypic loading conditions. This will be used to evaluate load
42 41 plane gap related reactivity insertion in the core. The criterion of success
53 is the limitation of reactivity insertions arising from assembly lateral motions to an acceptable level.

1.5.2.4.5 Fallback Position

If the predicted reactivity insertion based on these experimental data is not acceptable, an adjustment of gaps in the load planes will be
53 made.

1.5.2.5 Critical Experiments for Reactivity Coefficients, Control Rod Worth and Fuel Assembly Movement

1.5.2.5.1 Purpose

Critical experiments are required to provide information pertaining to the following parameters and components for the safety analysis of the CRBR: the primary and secondary control systems, the reactivity feedback coefficients and the mechanical motion of core and blanket assemblies.

The following information is required for the determination of the adequacy of the two control systems to meet the safety requirements:

a. The bias factors and associated uncertainties for the control rod worth predictions, of the tritium concentration in the liquid will be conducted to assure compliance with Federal and State Regulations. The rate of liquid radwaste discharge will be a function of the activity and the flow rate of the discharge from the mechanical draft cooling tower.

The design bases for the liquid radwaste system is discussed in Section 11.2 and release points are indicated in Section 11.2.6. Accidental spillage or release of liquid radwaste in the plant area outside of the plant buildings is extremely unlikely. A rupture of the blowdown line could release dilute liquid radwaste to the ground prior to direct discharge to the river. Any other liquid radioactive waste spilled inadvertently on the ground in the vicinity of the plant will either enter the ground or flow into the local drainage courses. The ability of the ground to disperse and dilute normal or accidental plant releases is discussed in Section 2.4.13.

The yearly average anticipated activity in the Clinch River due to operation is discussed in Section 11.2. The dilution factors used to evaluate the liquid radwaste effluents are described in 11.2.7.

The water velocity at the CRBRP site is determined primarily by the Melton Hill Dam with Fort Loudoun and Watts Bar Dams having a much smaller effect. Should the need arise for any regulation of Melton Hill Dam which would result in an extended period of no flow or reverse flow at the site, the operations would be coordinated to meet flow requirements at the CRBRP site.

Users of surface water in the area of the CRBRP are indicated in Section 2.4.1.2. The design of the discharge structure and operating procedures assure that adequate mixing of plant effluent and river water flow will occur prior to use. As discussed in Section 2.4.13.2, there are no users of ground water in the immediate area of the Site who could be affected by liquid releases from the CRBRP.

A description of the final heat sink design bases for both normal and accident operating modes is found in 16.3.5.10.

Recirculation of normal liquid releases to the plant makeup pumps during periods of normal river flow is unlikely due to the relative location of the river water pumphouse and the discharge structure as shown in Figure 2.4-1. Induced temperature changes are unlikely since the Clinch River is shallow and unstratified in the vicinity of the site. Preliminary investigation indicates that there will be little or no increase in turbidity from sediment disturbance. No hydraulic short circulating of cooling ponds will occur since no cooling ponds are employed for either normal operation or operation following a casualty event.

Extensive monitoring programs will be designed to monitor any effects of plant effluents on the environment. These programs are discussed in Section 11.6.

2.4.13 Groundwater

2.4.13.1 Description and Onsite Use

2.4.13.1.1 Regional Groundwater Hydrology

The CRBRP site lies within the southeastern section of the Valley and Ridge Physiographic Province of eastern Tennessee. The province is a long, narrow belt of faulted and folded predominantly calcareous Paleozoic rocks which extends for 500 miles from Virginia to Alabama. In eastern Tennessee its average width is about 40 miles. This province lies between the Blue Ridge Province on the east and the Appalachian Plateau Province on the west and is characterized by a succession of northeast trending ridges of various widths. The ridges are capped by the less soluble cherty limestones, dolomites, and shales, and the valleys are developed in the more soluble limestones, dolomites and shales. Ancient thrusting and folding have resulted in nearly all of the beds dipping to the southeast. The groundwater hydrology of the Valley and Ridge Province in eastern Tennessee has been described in Reference 33. More general discussions of the hydrology of carbonate rock terraces may be found in references 33, 34, 36, and 43.

The most important aquifers in the Valley and Ridge Province in eastern Tennessee are the carbonate rocks which underlie the majority of the province. The openings in which ground-water may be found in these carbonate rocks are of two types: primary openings, which were formed at the time of the formation of the rock, and secondary openings, which have a later origin. The primary openings in the carbonate rocks in the Valley and Ridge province in eastern Tennessee are generally small and of little hydrologic importance. The size and distribution of the secondary openings largely control the hydrologic properties of the carbonate rocks. Secondary openings, which are largely the entrance of chemically reactive water which by weathering processes tends to increase the size in these two figures coincides with the area of deepest weathering and lies about one-third of the way upslope from the topographic low towards the crest of the ridge to the northwest.

Water levels were initially measured during the period of the site
59 investigation at 33 of the observation wells shown in Figure 2.4-67 on a
55 regular basis from December 19, 1973 through April 1, 1974. The water level
in some of the wells changed as much as 20 feet during the 15 period of ob-
servation. The rainfall recorded at Melton Hill Dam (Reference 45), about
six miles from the site, is shown in Figure 2.4-71. Watts Bar headwater
levels during the same period are listed in Figure 2.4-72.

Water levels measured subsequent to the site investigation indi-
cate fluctuations which comprise an annual cycle, with the maximum water
levels occurring during the months of January and February, decreasing to
low values recorded during the months of October and November. Water
level fluctuations due to rainfall conditions are shown for fifteen
selected wells in Figure 2.4-70a through 2.4-70o covering the periods of
measurement both during the site investigation work and subsequent to it.
56 55 59 The groundwater monitoring program was suspended in July 1978.

The rapid response of the water levels to precipitation is indicative of rapid recharge, which occurs largely in areas of exposed rock and small sinkholes along the northwest and southeast ridges which bound the Plant Island. The large fluctuations in the groundwater table on the topographic highs and the quick response to precipitation are likely due to the proximity of these areas to recharge areas.

Eleven piezometers were installed in nests of twos and threes near borings 6, 7, 12, and 40 to supplement the information obtained from the observation wells described above. A typical piezometer installation is shown in Figure 2.4-73 and the locations of the piezometers are shown in Figure 2.4-74.

The water levels at the piezometers were also recorded at regular
monthly intervals after the completion of the site investigation work until
47 the suspension of the groundwater monitoring program. They were measured
on a regular basis during the investigation from the beginning of February
55 12 1974 to April 1, 1974. Groundwater fluctuations in selected piezometers
are plotted in Figures 2.4-75a through 2.4-75f covering the periods of measure-
59 ment both during the site investigation work and subsequent to it. The piezo-
metric head decreased with depth in the piezometers located on topographic
highs, indicating downward flow and thus confirming that these areas are recharge
areas. The piezometric head increase with depth in those piezometers located
in the groundwater trough around boring 27. The upward piezometric gradient
indicates that this is an area of upward flow and, thus, is a discharge area.

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2.4.13.2.3 Movement of Groundwater

In general, movement of groundwater occurs in a direction normal to the groundwater contours. At the site, movement is generally from topographically high areas to topographic lows; however, this pattern is modulated by the extent of weathering of the bedrock aquifers. Ultimately, the Clinch River acts as a sink to which all groundwater at the site migrates. Reference 33 lists instances in carbonate rock terraines in which weathering in topographically high areas is so deep that interchanges between adjacent valleys separated by these topographic highs may occur. Such situations are conducive to important reversals of groundwater flow. No evidence of such deep weathering action has been encountered at the site. Sound rock was encountered in the core of the ridges at elevations higher than the adjacent valley floors. Thus, at the site, the major ridges may be regarded as approximate locations of groundwater divides. Reversals in direction of flow which may occur because of the rather large fluctuations of the groundwater table will be local in extent and will not represent a diversion of groundwater from one major groundwater basin to another.

The Clinch River itself may act as a source of recharge during those periods when the river is subject to a rapid increase in stage. During such periods, water will flow from the river into the aquifer. This reverse flow will occur until a new condition of dynamic equilibrium within the groundwater system is established.

2.4.13.2.4 Effects of Plant Construction and Operation on Groundwater System

The groundwater environment at the site will be substantially changed by the construction of the Nuclear Island. The foundation of the Nuclear Island Structures is to be placed generally at elevation 715.
55 Excavation for the Nuclear Island foundation will be concurrent with dewatering. Due to the proximity of the

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Clinch River and the contiguous area of deeply weathered, more permeable rock between the Nuclear Island and the river, and since the excavation will extend generally to a depth of 14 feet below the normal low stage of the river, considerable dewatering of the site to accommodate the proposed construction will be required.

The effect of the above described lowering of the water table will probably be to lower the water table around the Nuclear Island to an elevation below the normal water-surface elevation in the Clinch River, and thus flow of water from the river toward the plant site may be induced.

2.4.13.2.5 Groundwater Quality

Eighteen groundwater samples for water-quality analyses were obtained from five observation wells and eleven piezometers. The results of these analyses are compared to regional values (Ref. 33) in Table 2.4-20. In general, the chemical and physical quality of the groundwater at the site is not significantly different from regional chemical and physical quality. Groundwater at the site is chemically suitable for human consumption but may require softening for industrial uses.

2.4.13.3 Accident Effects

2.4.13.3.1 Direction of Movement of Groundwater and Contaminants

At the site, under existing conditions, the movement of groundwater is from groundwater ridges to adjacent groundwater lows and thence to the Clinch River, which serves as a groundwater sink to the site area. Thus, the Clinch River acts as a barrier to the movement of groundwater from the site to the wells and springs in present use within a 2-mile radius of the site, all of which are located to the south of the Clinch River. Movement of groundwater in a northwest-southeast direction is restricted by the ridges and groundwater highs associated with these ridges running across the site in a northeasterly direction. Thus the movement of groundwater from the site is to the Clinch River. While extreme groundwater conditions or the construction of the plant may cause local redirection of flow, these redirections of flow will not represent a diversion of groundwater movement away from the Clinch River, nor toward any existing groundwater user.

2.4.13.3.2 Dispersion, Dilution and Adsorptive Potential of the Groundwater Environment

2.4.13.3.2.1 Ion-Exchange Potential of Soil and Contaminant Movement Rates:

Standard methods of chemical analysis were used to determine the cation-exchange capacity of the soil at the site. Four samples were selected from borings around the site and tested to determine their ion-exchange capacity and distribution coefficient. The results of these tests are presented in Table 2.4-21.

The ion exchange capacity of soil affects the rate at which a radioactive groundwater contaminant moves through the soil. The rate of movement of the contaminant depends on the composition of the waste, the composition of the soil, and the rate of movement of groundwater. The radioactive contaminant will move less rapidly than the groundwater because it will be adsorbed, to some degree, by soil particles. A relationship has been developed (Ref. 43) which provides an estimate of the effect of ion adsorption on the travel time of radioactive contaminant. This relationship may be expressed as

27

$$t_c = \left\{ 1 + B \frac{(1-P)}{P} K_d \right\} t_w$$

where

t_c = time of travel for contaminant
B = bulk density (g/ml)
P = porosity
K_d = distribution coefficient (ml/g)
t_w = time of travel for groundwater

The distribution coefficient provides a measure of the exchange characteristics of the soil. From Table 2.4-21 the minimum value of K_d for the soils tested is about 38 ml/g. Using this value for K_d, a conservative travel time for strontium is estimated to be approximately

$$t_c = \left\{ 1 + 1.925 \frac{(1-.3)}{0.3} 38 \right\} t_w$$

$$t_c = (1+171)\, t_w$$

i.e, the conservative travel time for strontium is about 172 times longer than the travel time for water. If the largest measured value of the distribution coefficient were used, the travel time would be increased by a factor of 437 instead of 172. Laboratory tests on remolded soils from the site indicated that reasonable values of unit weight and porosity of site soils were in the ranges of 120 lb/ft^3 (1.925 g/ml) and 0.3, respectively. These values were used in the above calculations. Strontium was used in these calculations because it frequently represents a critical potential nuclear contaminant.

The range of travel times for strontium is summarized in the following table:

RANGE OF TRAVEL TIME FOR STRONTIUM

K_d (See Table 2.4-21)	TRAVEL TIME OF CONTAMINANT	REMARKS
38	172 t_w	Minimum K_d, Minimum Travel Time
53	238 t_w	Median K_d, Median Travel Time
97	437 t_w	Maximum K_d, Maximum Travel Time

The above conclusions are valid provided that the contaminant is released in and transported by groundwater flow through soils such as those encountered on the site.

Shorter travel times would be expected for transport by groundwater flow through rock joints. In terms of the relation

$$t_c = \left\{1 + B \frac{(1-P)}{P} K_d\right\} t_w,$$

the porosity, P, along rock joints may be larger than the porosity in soil. Thus, the term $\frac{(1-P)}{P}$ would be smaller along rock joints. Also, the distribution coefficient, K_d, would be smaller in rock because the ion exchange capacity is smaller in rock than in soil. Thus, the bracketed term $\left\{1 + B \frac{(1-P)}{P} K_d\right\}$ would be smaller for flow in rock than in soil, and so t_c would be smaller for flow in rock than in soil. In extreme cases, the time of travel of the contaminant may approach the time of travel of groundwater in the joints.

Water-level data (Figures 2.4-68 and 2.4-69) indicate a hydraulic gradient of about 0.007 feet/foot from the groundwater low west of the plant to the Clinch River. Porosities of the subsurface rocks were determined as a part of the Birdwell continuous velocity-logging program (see Section 2.5.4.4.2). These measurements indicated porosities ranging from about 5 to 33 percent. Since higher values of porosity are generally associated with higher values of permeability, an estimate of groundwater travel time from the Nuclear Island area to the Clinch River was made using a value of permeability equal to 2000 feet/year (highest measured value was 1510) and a value of porosity equal to 0.3. This computation indicated a travel time of about 28 years for groundwater flow from the Nuclear Island to the Clinch River.

2.4.13.3.2.2 Dispersion and Dilution Potential of the Groundwater Environment:

The dispersion potential of the groundwater environment is dependent upon the characteristics of the aquifer through which the groundwater and contaminant move. Where movement is occurring in fine-grained soils with appreciable fractions of clay-size particles, such as are encountered on the site, the effects of adsorption are likely to completely overshadow the effects of dispersion in modulating the concentration of a groundwater-borne contaminant (reference 31). When transport is occurring in complicated calcareous aquifers such as are found on the site, the effects of dispersion are likely to be relatively minor due to the tendency of groundwater flow to occur along well-defined joints and cracks in the calcareous rocks. Thus, while it is difficult to quantify the effects of dispersion of possible groundwater contaminant concentrations within the groundwater system at the site, it may conservatively be assumed that whatever dispersion occurs will not provide a significant lowering of contaminant concentrations at possible groundwater discharge points.

The dilution potential of the groundwater system is related to the volume of groundwater in storage within the aquifer and to the opportunity for mixing of the induced contaminant with the groundwater in storage. Volumes of stored groundwater are relatively large in those areas in which the groundwater occurs in saturated residual clays on the site, and opportunities for mixing of the contaminant by dispersion are available. However, in these clay soils the effects of dilution are likely to be overshadowed by the effects of cation adsorption of the nuclear contaminants as described in a previous section. Volumes of water in storage within the carbonate aquifers at the site are unknown. Regardless of the volume of water in storage within the carbonate aquifers at the site, the opportunities for mixing of contaminant concentrations with the groundwater in storage may be limited due to the tendency of groundwater movement in these rocks to be limited to flow along joints and cracks in the rocks as previously discussed. Thus, it may be conservatively concluded that contaminant dilution within the groundwater system will not provide a significant lowering of contaminant concentrations at possible groundwater discharge points.

2.4.13.3.3 Effects of Accidental Spills

Accidental spills of water-borne nuclear contaminants are not seen as posing a danger to present or future groundwater users. The site topography and the characteristics of the groundwater aquifer at the site are such as to preclude the migration of groundwater from the site toward any present or future off-site groundwater user by subsurface flow alone. The mobility of possible nuclear contaminants through soils encountered at the site is limited by the cation exchange characteristics of these soils; however, movement of contaminants which reach the underlying carbonate rocks is likely to proceed at velocities which approach the velocity of the transporting groundwater.

The ultimate destination of contaminants induced into the groundwater system is the Clinch River.

2.4.13.4 Monitoring and Safeguard Requirements

The potential for transmission of groundwater contaminants to present or future off-site groundwater users is small. For this reason the monitoring of groundwater for possible contamination may be restricted to periodic sampling and analysis of groundwater in the area of the Plant Island, near other potential sources of contamination, and near any on-site groundwater withdrawal points which may be required.

A groundwater radiation monitoring program is in process and consists of the following: A well, located in a main area of groundwater movement between the plant and the river, was installed and readings commenced in October, 1975. This well is equipped with an automatic pumping sampler that will composite once-daily samples. The composited samples will be analyzed monthly for gross alpha, gross beta, and tritium activities. These analyses will provide information on existing radiation levels and their variations and will continue throughout the plant life.

In addition, shortly before the plant goes into operation, another monitor
12 well will be constructed in the immediate vicinity of the reactor. It will also be equipped with an automatic pumping sampler. Weekly composite samples

from each auto-sampled well will be analyzed monthly for gross alpna, gross beta and tritium activities. If any significant increase in gross alpha or gross beta activity is detected, an analysis will be made to identify the specific radionuclide.

In addition to the radiation monitoring program, a groundwater monitoring program has been developed consisting of the following elements:

a) Pre-Construction

Prior to the completion of the site foundation investigation program in May, 1974, a total of 48 permanent groundwater stations were installed. These stations include 37 observation wells and 11 piezometers. Water level readings were obtained at these stations during the investigation, and were read at monthly intervals thereafter until the groundwater monitoring program was suspended in July 1978. The
47 groundwater monitoring program will be resumed for a period of several months prior to start of construction. The locations of the wells and piezometers are shown on Figure 2.4-67 and 2.4-74, respectively.

b) Construction

During construction, until final grade is establishea, a system
55 33 of observation wells and piezometers strategically located in the areas of the Nuclear Island and Category I Emergency Cooling
43 Tower will be maintained at the site and monitored weekly. This
55 system will incorporate approximately 16 existing observation wells and 11 piezometers. These will be augmented by the installation of approximately 27 additional piezometers to be located around the periphery of the Nuclear Island and
55 Emergency Cooling Tower excavations.

55 Dewatering of the Nuclear Island will be accomplished by installing
59 horizontal gravity drains in the excavated rockfaces and pumping from
33 sumps located at intervals around the perimeter of the excavation in selected berms, including the berm installed at top of rock. Pumping will also be conducted from sumps located at the base of the excavation.
55 The planned monitoring system will adequately define the groundwater
12 regime and permit construction to proceed in a safe manner. Rock treatment will be provided where required to reduce potential inflow from previous aquifers occurring in the weathered strata encountered in the excavation.

2.4.13.5 Design Bases for Subsurface Hydrostatic Loadings

As is shown on Figures 2.4-68 and 2.4-69, the present groundwater
55 59 level is above the proposed foundation level of the Nuclear Island common mat (elevation 715) and the Emergency Cooling Tower structure (elevation 765).
43 Dewatering of these areas during construction is presently planned to be
33 accomplished concurrent with the excavation to foundation levels by pumping
55 from sumps located in selected berms and in the bottom of the excavation and by
59 installing horizontal gravity drains in the excavated rockfaces. Grouting of some local areas around the excavation may be initiated during construction if it appears that such grouting would be of benefit to the dewatering program.

55 Backfill around the Nuclear Island and the Emergency Cooling Tower will be a free draining granular material. No permanent dewatering system will be installed around these structures, however, groundwater monitoring wells will be installed in the granular backfill for the purpose of maintaining a record of water levels around the Category I structures. Post-construction water levels around the Category I structures are expected to be approximately the same as the present levels. The Category I structures will be designed to resist lateral and vertical loads resulting from a static groundwater level of 809 MSL. This elevation corresponds to the maximum water level at the site resulting from the concurrent occurrence of 1/2 the PMF,
55 1/2 SSE and postulated Norris Dam failure. A groundwater elevation of 780', which conforms with the precipitation induced normal maximum water level, has
59 been used in the dynamic analysis of the Category I structures.

2.4.14 Technical Specification and Emergency Operation Requirement

All safety related facilities are designed and constructed to withstand the effects of flooding to an elevation of 809 feet, which is the maximum
53 computed flood level as discussed in Section 2.4.4. Emergency protective measures required to minimize the impact of the maximum flood on safe shutdown are not required.

The maximum flood level stems from the concurrent occurrence of the most pessimistic time sequence of 1/2 the PMF and 1/2 SSE and postulated Norris Dam failure. The worst case analyses are summarized in Table 2.4-8.

2.4 References

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2.5 GEOLOGY AND SEISMOLOGY

2.5.1 Basic Geologic and Seismic Information

The CRBRP Site is a peninsula on the north bank of Watts Bar Lake between Clinch River miles 14.5 and 18.6 in Roane County, Tennessee. The Site is located in the southwest corner of the U. S. Department of Energy's Oak Ridge Reservation.

Two subsurface investigations have been performed at the Site by the Tennessee Valley Authority (TVA), and a third has been performed by Law Engineering Testing Company. Four core holes were drilled by the TVA in February and March, 1972. The second TVA investigation consisting of 20 core holes was conducted from October 23, 1972, through December 29, 1972.

During the period of May 1973 to February 15, 1974, Law Engineering Testing Company prepared a Site Report (SR) of the Geology and Seismology of the Clinch River site. The activities attendant to this Site Report included literature studies, surface mapping, remote sensing studies and 42 additional core borings. The borings were made for several investigations which consisted of: the completion of the grid pattern initiated by TVA, an investigation of the Copper Creek fault, a sinkhole development investigation and borings for specific geologic and engineering purposes.

Following the submittal of the Site Report and continuing through April 1974, 40 foundation borings were made at Category I structure locations. In-situ testing, laboratory testing and detailed engineering studies were made of the foundation materials.

An additional 23 borings were completed between September 1976 and
55 June 1977 on the east side of the Nuclear Island, 16 of which were foundation borings for the Steam Generator Maintenance Bay, a Category I structure.

59 55

The investigations of the CRBRP Site have shown that the site is underlain at shallow depths by sedimentary rocks of Ordovician age. The rock types basically consist of siltstone and limestone with rock units occurring as wide, northeast trending bands. The rocks were deformed during the Paleozoic (more than 230 million years ago) and are now tilted to the southeast at an angle of about 30°. Category
55 I structures will generally bear entirely within a siltstone band (Chickamauga, Unit A, Upper Siltstone), which was determined to be the most favorable founding material based on its engineering properties and its

resistance to weathering. The Steam Generator Maintenance Bay will be
55 founded on competent rock of the Unit B limestone formation. The Fuel Oil Storage Tank will be supported on compacted Class "A" structural backfill placed directly on the Unit "A" Upper Siltstone.

59 Between May 1975 and March 1977, additional borings were completed for Balance of Plant structures and also in the Knox Dolomite to check on a potential source of aggregate (References 174, 175, 176, 177 and 178). Information
41 contained in these references does not change any of the conclusions reached regarding foundation conditions for the Category I structures.

2.5.1.1 Regional Geology

The following discussion of regional geology is based on a comprehensive review of available data including published and unpublished reports and maps, and interviews with recognized authorities. The study region includes the area within 200 miles of the site, with major emphasis placed on the Valley and Ridge Physiographic Province.

59

2.5.1.1.1 Geologic History

The CRBRP site is located in the Appalachian Highland Physiographic Division of the eastern U. S. The proximity of the site to the physiographic provinces within the Appalachian Highlands is shown on Figure 2.5-1. As the name Appalachian Highlands implies, this area is presently characterized by rugged terrain which varies from rolling hills to mountains.

Early in the Paleozoic Era (600 million years ago) the location of the present Appalachian Highlands Physiographic Division was a geosyncline occupied by a mediterranean sea in which up to 40,000 feet of sediments accumulated. The sea was never that deep, but the sea floor subsided gradually under the weight of the sediments while the adjacent highlands rose. Some of the crustal movements that repeatedly occurred during the Paleozoic are recorded as unconformities which represent interruptions of the sedimentation process. The present Blue Ridge Physiographic Province coincides with the previous boundary between the deeper southern seas and the shallower, northwestern portion of the geosyncline. In earliest Paleozoic time, the Piedmont Province was part of the geosyncline but eventually became a source area.

Between middle and late Paleozoic time, several episodes of diastrophism occurred. The sediments southeast of the present Blue Ridge Province underwent a high degree of metamorphism, were injected with magmas and were severely deformed. Deformation of sediments north of the Blue Ridge was restricted to folding and major thrust faulting with no extensive metamorphism occurring. Toward the end of the Paleozoic, the whole region was uplifted and has not since been submerged except for localized basins in the Piedmont during the Triassic Period (180 million years ago).

During the Mesozoic (63 to 230 million years ago) and Cenozoic Eras (present to 63 million years), erosion of the present Appalachian Mountains supplied the sediments that comprise the Coastal Plain Formations.

2.5.1.1.2 Regional Tectonics

During the Paleozoic Era, tectonic forces directed toward the northwest deformed the rocks of the Appalachian Geosyncline. Deformation was greatest in the southeastern portion (Piedmont and Blue Ridge) where the rocks were metamorphosed and injected by magmas. This portion was thrust to the northwest along boundary faults such as the Great Smokey Fault (see Figure 2.5-2). Rocks for several miles to the northwest of this boundary (Valley and Ridge) were shingled into a series of thrust blocks. The thrusting abruptly diminished toward the northwest where a broad syncline was developed (Appalachian Plateaus) and large lateral movements occurred along bedding.

It is generally accepted that these thrust blocks do not extend into the
basement, but are bounded below by a lateral sole fault in some relatively deform-
59 able formation above the crystalline basement (thin-skinned structural concept).
The consistent repetition of the Rome Formation as the basal formation of the thrust blocks substantiates that the thrust blocks do not extend to the basement. In 1973, G. D. Swingle (Reference 97) prepared a cross section within the Central Tennessee section of the Valley and Ridge which shows the basement to be at a depth of 13,000 feet and a sole fault to be at a depth of 9,000 feet. This recent publication confirms the previous concept of a sole fault above the basement rocks.

Figures 2.5-1A, 2.5-1B, and 2.5-1C are geologic cross-sections
showing the thin-skinned structure concept in the Valley and Ridge
38 province of Tennessee. These cross-sections are similar in concept and
show the relationship of surface structures in this region to the regional geology, including "basement" geology. Together these cross-sections include the Cumberland Plateau, Valley and Ridge, and Blue Ridge provinces. The thin-skinned structural concept is discussed in Supplement 2.

Information concerning the extent and nature of the Rome formation "sole thrust" in the CRBRP site area includes the Joy Test Well (location shown on Illustration 6 of Supplement 2) and seismic reflection profiles made by Geophysical Services, Inc. (Illustration 1 of Supplement 2). Inquiries to the Tennessee Division of Geology confirmed that these are
59 the only available deep geologic data for the site area, and that no deep
oil or gas test wells are recorded for Roane County, Tennessee.

There were several episodes of tectonic forces during the Paleozoic Era; however, one episode caused the major deformations of the rock strata in the vicinity of the CRBRP site (Reference 81). This is referred to as the Allegheny Episode which occurred during either the Pennsylvanian or Permian Period, or at least 230 million years ago.

Northwest of the geosyncline, the strata were gently folded to form the Cincinnati Arch. This feature was formed during the Paleozoic Era by downwarping along its margins. Several deep-seated faults developed in this area during the Paleozoic, and constitute the Kentucky River and Rough Creek Fault Zones.

The Kentucky River fault zone (see Figures 2.5-2 and
2.5-2A) trends east-west from eastern Kentucky westward across the
Cincinnati arch. This fault zone dies out on the western flank of the
38 Cincinnati arch. The fault zone has a total length of about 150 miles
and a width of about 25 miles. The closest fault to the CRBRP site
38 within this zone is approximately 90 miles to the north.

Faults in the Kentucky River fault zone are mostly steep,
en echelon normal faults, and bound small grabens. These faults also
38 show some strike-slip movements. The maximum displacement along the Kentucky
River fault zone is approximately 600 feet (Reference 145). Underlying
basement rocks are faulted, with movement having begun early in the
Paleozoic (Reference 81, Section 2.5). Latest movements along this fault
are post-Pennsylvanian (310 MYBP) (Reference 165).

38 The Rough Creek fault zone begins west of the Cincinnati arch and
extends across western Kentucky into southern Illinois. This fault zone
has an east-west trend similar to that of the Kentucky River fault zone.
Near Shawneetown, Illinois, the Rough Creek fault zone curves southwest-
ward around Hicks dome and merges with the New Madrid faulted zone (Ref-
erence 145). This fault zone has a length of about 120 miles and a width
of approximately 25 miles. The nearest point of this fault zone to the
site is about 120 miles to the northwest.

The Rough Creek fault zone includes horsts, grabens, and en
echelon normal faults. These faults also show some strike-slip movement
(Reference 81, Section 2.5). The Rough Creek fault zone has displace-
ments up to 3000 feet (Reference 147). East of Shawneetown, Illinois,
38 movement along this fault zone was probably pre-Cretaceous (References
153 and 166). However, west of Shawneetown the portion of the Rough
Creek fault that curves southwestward has been postulated as being active
since Pliocene (5 to 2 MYBP) and maybe into the Recent (Reference 147).

Both of these fault zones are outside the Southern Appalachian
Tectonic Province and are 90 to 120 miles from the CRBRP site. Neither
38 27 of the faults affects the geologic or seismic design at the CRBRP site.

2.5.1.1.3 Physiographic, Lithologic, Stratigraphic and Structural Settings

Areas of similar lithology, stratigraphy, structure and geomorphic history are associated with physiographic provinces. The physiographic provinces within 200 miles of the CRBRP site include the Interior Low Plateaus, Appalachian Plateaus, Valley and Ridge, Blue Ridge, and Piedmont, as shown on Figure 2.5-1. The lithologic and stratigraphic relationships are shown on the Regional Geologic Map, Figure 2.5-3, which may be regarded as a bedrock map.

2.5.1.1.3.1 Valley and Ridge Physiographic Province:

The CRBRP Site is located in the southeast section of the Valley and Ridge Physiographic Province. This section of the Province is about

25 to 50 miles wide and is over 500 miles long, extending northeastward from Alabama to Virginia. It is characterized by narrow, elongate ridges and intervening valleys which trend in a northeasterly direction, reflecting the regional orientation of the inclined strata of sedimentary rock.

The CRBRP Site is located in the southern section of the province. In this area, ridges are fairly evenly crested and are developed in areas underlain by resistant sandstone, siltstone, and the more siliceous limestone and dolomite. Valleys are typically broader than the ridges and have been formed in areas underlain by easily erodible shale and more soluble limestone formations.

Overall drainage in the Valley and Ridge Province follows the northeast-to-southwest trending valleys. Major streams flow across this trend for short distances due to entrenchment of ancient stream courses which have gradually eroded downward to their present levels. Remnants of river terraces (often over 100 feet above the present flood plain) represent ancient flood plains which were severely eroded as the streams eroded downward to their present levels.

In the vicinity of the CRBRP Site, the topography conforms to the regional northeasterly trend. Surface elevations range from about 800 feet in the valleys to 1,000 feet on the ridge crests. The Clinch River is entrenched across several ridges but generally follows a southwesterly direction (see Figure 2.5-1).

The Rome Formation and the Conasauga, Knox and Chickamauga Groups comprise the majority of the bedrock in the Valley and Ridge Province in Tennessee and they outcrop as repeated bands trending in a northeasterly direction. The Rome Formation is middle-Cambrian in age and its maximum exposed thickness is about 1200 feet. It is composed mainly of red, green and yellow shale, siltstone, and sandstone, with minor amounts of gray dolomite. The Conasauga Group is late-Cambrian in age and is about 2,000 feet thick. It is composed mainly of alternating gray shale and limestone in the southeastern portion of the province. The amount of limestone decreases toward the northwestern boarder of the province where the Conasauga is nearly all shale. The Knox Group is of late-Cambrian and early Ordovician age and is 2,500 to 3,000 feet thick. These rocks are predominantly chert-bearing dolomites and lesser amounts of limestone. The Chickamauga Group is of middle-Ordovician age and ranges in thickness from about 8,000 feet in the southeastern portion of the Valley and Ridge Province to 2,000 feet in the northwest portion of this province. It is composed of alternating layers of gray and maroon limestone, calcareous siltstone and shale. The remaining small portion of the Valley and Ridge Province in Tennessee is underlain by alternating sandstone, shale, and limestone formations of late-Ordovician to Pennsylvanian age, and lesser amounts of early-Cambrian dolomite.

The geologic structure of the Valley and Ridge Province is characterized by elongate folds and thrust faults that strike in a northeasterly direction. In the southern section of the province the faults, and in most places the bedding, dip southeast. In addition to the faults, localized bedding distortions and minor offsets are commonly exposed in outcrops, roadcuts, and excavations.

There is no geologic evidence that indicates that any of the thrust faults can be considered to be active faults still undergoing movement. Geologic evidence indicates that the final episode of movement occurred during the Pennsylvanian or Permian Periods, or at least 230 million years ago (Reference 81). The CRBRP Site is situated between two inactive thrust faults - the Copper Creek and Whiteoak Mountain Faults which are discussed in detail in Sections 2.5.2 and 2.5.3.

2.5.1.1.3.2 Southeast of the Valley and Ridge Physiographic Province

The Valley and Ridge Province is bordered on the southeast by the Blue Ridge and Piedmont Physiographic Provinces. The Blue Ridge Province is about 15 to 70 miles wide and 600 miles long, extending from Pennsylvania to Georgia. The province encompasses the most rugged terrain and the highest elevations in the eastern United States. Surface elevations are generally 1,500 to 5,000 feet with the highest elevation being 6,684 feet. This province is characterized by closely spaced ridges trending in a northeasterly direction.

Rock units included in this province consist of slate, phyllite, schist, gneiss, granite, pegmatite and quartzite. These units are pre-Cambrian and lower Paleozoic metamorphic rocks, generally of amphibolite grade, with some intrusions of predominantly Paleozoic age. The schist and gneiss are considered the oldest rocks in the region. Recent radiometric dating places the peak of Paleozoic metamorphism at a minimum of 430 million years (Reference 4). Pegmatite is younger, with recorded age determinations of 380 million years (Reference 4). The Blue Ridge Physiographic Province is highly deformed with the metamorphic grade increasing from west to east. The northwestern boundary of this province generally coincides with major faults, along which metamorphic rocks have been thrust to the northwest over younger unmetamorphosed sedimentary rocks of the Valley and Ridge.

The Piedmont Province is about 40 to 125 miles wide and about 1,000 miles long, extending from Alabama to New York. In the southern section, surface elevations are about 1,000 feet near the Blue Ridge Physiographic Province and decrease to about 500 feet near the Fall Line. Elevations gently decrease northward and are about 100 to 500 feet near the northern terminus of the province.

In the Piedmont, saprolitic soils have developed over the underlying rock and erosion has produced a smooth, rolling landscape. Metamorphosed volcanic and sedimentary rocks and igneous rocks underlie this province. These rocks have been faulted and severely distorted.

Radiometric dates indicate that regional metamorphism occurred in this province in the time period of 300 to 520 millions of years ago (Ref. 4). The youngest intrusions are diabase dikes and sills formed during the Triassic Period. Some of these dikes exhibit small fault offsets. A recently obtained radiometric date of mylonite occurring in one of these offsets indicates this faulting to have occurred approximately 150 million years ago, thus indicating the most recent fault movement as Jurassic (see Table 2.5-1).

2.5.1.1.3.3 Northwest of the Valley and Ridge Physiographic Province

The Valley and Ridge Province is bounded to the northwest by the Appalachian Plateaus Physiographic Province. This province is about 20 to 200 miles wide and 1,000 miles long, extending from Alabama to New York. Its southeastern margin is an abrupt topographic [illegible] known as the Allegheny Front, which in Tennessee is known as the Cumberland Escarpment. Surface elevations range from about 1,000 to 3,000 feet. The topography is gently sloping to undulating with localized mountainous areas. Most of this province is underlain by alternating sandstone and shale of Pennsylvanian age. Rock strata are gently folded into a broad syncline, with a few elongate anticlinal folds superimposed on this overall structure. Paleozoic bedding plane faults underlie much of this province.

The Appalachian Plateaus are bordered on the northwest by the Interior Low Plateaus Physiographic Province. This province is about 300 miles long and 300 miles wide with a general surface elevation of about 800 feet. The eastern and southern boundaries of the Interior Low Plateaus are formed by the Highland Rim, the western boundary by the Coastal Plain Province and the northern boundary is formed by the southern limit of glacial deposits. Two large, shallow topographic basins occur within this area and are known as the Nashville Basin and the Lexington Plain. This province is underlain by sedimentary rock, mostly limestone of Ordovician and Mississippian age, with lesser amounts of sandstone and shale of Pennsylvanian age. The rock strata are gently inclined over the Cincinnati Arch, which includes the Nashville Dome and the Jessamine Dome.

2.5.1.1.4 Areas of Surface Subsidence and Uplift

2.5.1.1.4.1 Solution Activity in Carbonate Terranes

Slightly acidic groundwater produces solutioning in carbonate rocks. The extent of solutioning is dependent upon their mineralogic composition. In those areas where the rocks are limestones and dolomites, solutioning is most severe. The degree of solutioning decreases as the rocks grade toward more siliceous and clayey sediments. In those sediments which do not contain carbonate materials, solutioning is negligible. Solutioning in Valley and Ridge carbonates generally advances along structural features such as joints and bedding (Refs. 32, 41, 66). Advanced stages of solutioning produce nearly planar zones which diminish in size with depth. Steeply inclined solution joints are often soil filled and are commonly referred to as slots.

Since solutioning is concentrated along structural features, areas where closely spaced jointing occurs generally display advanced development of solution voids (Ref. 84). The structural control of solutioning is illustrated by the northeasterly trend of small caves, corresponding to the regional structural trend (Ref. 66).

Solutioning of carbonate rock is expressed at the ground surface by surface depressions and drop-outs commonly referred to as sinks. Sinks generally result from the ravelling of soil overburden into the underlying caves and slots. The shapes of the sinks are governed by the extent to which ravelling has progressed and the overburden thickness. Sinks vary in shape from nearly circular to elongate and in horizontal extent from a few feet to several hundred feet. In cross-section they vary from saucer-like to steep-sided.

As discussed in Section 2.5.1.2.4.4, the Unit A Limestone and Unit B Limestone outcrop areas were inspected for evidence of solution activity in the vicinity of Category I structures. Five small diameter holes were observed in the Unit A Limestone outcrop area at locations shown on Illustration 7 of Supplement 2. The character of the small holes is shown in Figure 2.5-1D, which includes photographs of two representative locations. At the surface these small holes were 1 to 2 feet in diameter and generally circular in shape. The depth of the holes ranged
38 from 1 to 2 feet. The sides of the holes were steep and irregular. These five small holes were manually probed and in most instances loose
37 materials were penetrated to depths from 3 to 10 feet.

Reconnaissance in the vicinity of the site indicated the presence of a number of small caves. Eight small caves in the site vicinity are shown on Figure 2.5-3A. Six of the caves are more than 2 miles distant from the Nuclear Island. The remaining two caves are located on the northwest facing slope of Chestnut Ridge, about 1.2 miles from the Nuclear Island. Other caves were encountered near the Chestnut Ridge caves, but were too small to permit entrance and are not considered significant. The reconnaissance, which included interviews with local residents, did not disclose any caves in addition to those described above. All of these caves occur in the Knox Group outcrop band.

The locations of northeasterly trending small caves discussed above, which illustrate structural (joint and bedding) control of solutioning, are
too remote from the site to appear on Figure 2.5-3A. 23

Steep-sided sinks in some cases have been mistakenly identified as the result of the collapse of caves or caverns within the rock (Ref. 41). In those cases investigated, these steep sided sinks have occurred in areas of thick overburden soil and were the result of ravelling of the overburden into underlying caves and slots.

2.5.1.1.4.2 Injection Well

A discussion of several facilities for injecting radioactive wastes into subsurface strata at the Oak Ridge National Laboratory about 4 miles east of the CRBRP site is presented in Supplement 2 to Chapter 2 of this document. It is concluded that these injections have no effect
59 on the CRBRP site.

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2.5.1.1.4.3 Mineral Extraction

Coal and petroleum are presently being extracted from beneath the Cumberland Plateau several miles to the west of the CRBRP site, but no known commercial quantities of either are known to exist in the Valley and Ridge Province of Tennessee (Ref. 28). There are no existing or abandoned coal mines, nor oil or gas wells, within five miles of the site. In the Valley and Ridge Province of Tennessee, the only active underground mining is north and northeast of Knoxville, where zinc ore is being extracted. The mineral sphalerite is being mined from deposits in Knox dolomites. There is no evidence of existing or abandoned underground mines within five miles of the CRBRP site. There are many surface quarries in the Valley and Ridge, most of which are located in limestone areas. An abandoned limestone quarry, the closest quarry to the site, is located about two miles to the northwest.

2.5.1.2. Site Geology

The following discussion of site geology is based on interpretations of geologic maps, topographic maps, photographs, remote sensing data, mineralogy studies, and subsurface data acquired from
55 beneath the site. A total of 129 borings and 6360 linear feet of refraction traverses have been completed at the site. Information was also obtained from the regional geology studies described in Section 2.5.1. The location of the borings is indicated in Figure 2.5-4.

2.5.1.2.1 Geologic History

The Clinch River site lies near the western border of the Appalachian geosyncline which was active during most of the Paleozoic Era. During the early portion of this Era, in Cambrian time, sands and clays were deposited in shallow muddy waters and these consolidated to form the sandstones and shales of the Rome Formation. The geosyncline gradually depressed and the sea became deeper and broader. At the beginning of the Conasauga deposition the sea received a small amount of sand and much clay. The sediment load gradually changed until the end of the Conasuaga when only limy sediments were deposited. Throughout the succeeding Knox deposition, the sea was deep and still as indicated by the great thickness of limestones and dolomites that were deposited. At the close of the Knox deposition, the site area uplifted slightly and was exposed to erosion. During the middle Ordovician the land subsided again and was covered by a shallow and oscillating sea in which a great thickness of calcareous shale and limestone was deposited. It was at this time that the interbedded limestone, shaly limestone, calcareous shales, and calcareous siltstones forming the foundation at the Clinch River site were laid down. Silurian through Pennsylvanian deposition included sand, clay and limy sediments, which consolidated to form sandstone, shale and limestone. At the end of Paleozoic time, during the Allegheny orogenic episode, the rocks at the site were tilted to the southeast (Copper Creek Fault) and northwest (Whiteoak Mountain Fault). Since Paleozoic time, weathering and erosion have been the dominant geologic processes at the site with sediment accumulation being restricted to terrace and floodplain deposits of the Clinch River.

2.5.1.2.2 Physiography

The topography in the vicinity of the CRBRP Site is characterized by subparallel ridges with intervening valleys. In the Site area the major ridges are Chestnut Ridge to the northwest and Dug-Hood Ridge to the southeast. Elevations of the ridge crests range between 900 and 1200 feet. The valley separating these ridges is regionally referred to as Raccoon Valley and in the site vicinity locally referred to as Poplar Springs Valley and Bethel Valley. The valley consists of rolling hills which range between elevations 750 to 850 feet. The ridges and valleys in the site vicinity correspond with the northeast regional trend described in section 2.5.1.1.3.1.

Figure 2.5-3A is a topographic map encompassing an area around the CRBRP site of approximately 100 square miles. This map shows nearby towns, governmental boundaries, and the northeasterly trending topographic expression of underlying geologic formations. 23

The CRBRP site is a peninsula formed by a U-shaped bend of the Clinch River as shown in Figure 2.5-5. The majority of the site is located on the southeastern flank of Chestnut Ridge. Within the site boundaries Chestnut Ridge consists of two subordinate ridges which crest at about elevation 900 feet. These sub-ridges are separated by a minor ravine which trends northeasterly across the peninsula. A topographic saddle in the bottom of this valley rises to about elevation 800 feet. The valley slopes from this saddle both in a northeasterly and southwesterly direction down to the level of the Clinch River (normal pool 735 to 741 feet). The sides of the valley slope upward at an inclination of about 13 degrees. The major directional control to the topography in the vicinity of the Category I structures is the northeasterly strike of the underlying rock strata. As discussed in Section 2.5.1.2.4, the rock bedding is inclined down to the southeast causing the various rock layers to occur as northeast-trending bands.

The Clinch River floodplain borders the western side and the southern tip of the peninsula as shown in Figure 2.5-5. The floodplain is flat to gently sloping and extends up to about elevation 752 feet. Its maximum width is about 500 feet. The floodplain is essentially absent from the eastern border of the site due to the presence of steep bluffs.

There are no perennial streams at the site, and flow along valleys and gulleys occurs only after periods of heavy rainfall. Surface drainage from the Chestnut Ridge section of the site occurs along northeast-trending ravines and reflects the control of the underlying rock strata. Drainage in the Poplar Springs Valley portion of the site generally occurs in a more random pattern.

Drainage also occurs vertically in the site area in areas
38| which are underlain by limestones and dolomites. Sinkholes were noted in the portion of the site where the Knox outcrops and in the southeast portion where the Chickamauga Units B and Undifferentiated outcrop. The sinkholes in these areas occur as: 1) small diameter vertical holes (generally less than 1 foot), 2) asymmetrical depressions which are generally less than 100 feet across and 10 to 15 feet deep and 3) gentle sided depressions which are generally several hundred feet across and 10 to 15 feet deep. In the Unit A portion of the Chickamauga Group sinkholes are absent. Only a few
27| small diameter holes as described in Section 2.5.1.1.4.1, were noted in the outcrop area of the Unit A Limestone.

2.5.1.2.3. Stratigraphy and Lithology

Rock at the site is comprised of two major geologic units, the Knox Group and Chickamauga Group. Scattered rock outcrops occur in the central portions of the site, however, the rock is generally covered by a veneer of residual soil, except in the southern portion of the peninsula and near the river where ancient terrace and recent alluvial soils have been deposited. Stratigraphic relationships be-
38| tween the geologic units are presented in Table 2.5-4, Figure 2.5-5 |7
illustrates the areal distribution of these formations and Figures 2.5-7 through 2.5-11 illustrate their occurrence in profile.

2.5.1.2.3.1. Knox Group

The Knox Group is predominantly dolomite with lesser amounts of limestone, and contains thick, chert-bearing sequences. The Knox overlies the Conasauga Group and is in turn overlain by the Chickamauga Group. The boundary between the Knox and Chickamauga is a disconformity formed by erosion of the Knox and subsequent deposition of the Chickamauga during the Ordovician Period.

Nineteen site borings penetrated the upper portion of the Knox Group and encountered predominantly light to medium gray, crystalline dolomite (see Table 2.5-5). Pink and green shale laminations also occur in the core. Pockets of chert-bearing calcareous siltstone similar to the basal portion of the Chickamauga were encountered at the top of the Knox and represent sediments deposited in irregularities on the pre-Chickamauga erosion surface.

2.5.1.2.3.2 Chickamauga Group

The Chickamauga Group consists of alternating layers and laminations of maroon and gray siltstone, limestone, and shale with thin layers and nodules of chert in the lower portion. The Chickamauga Group outcrops on the southeastern half of the CRBRP site and is overlain by the Rome Formation to the southeast of the Clinch River. The boundary between these two formations is the Copper Creek Fault which
38 truncates the upper portion of the Chickamauga. The total stratigraphic thickness of the Chickamauga at the site is approximately 1700 feet.

The lower portion of the Chickamauga is subdivided into Units
38 A and B as illustrated in Table 2.5-6. Eleven key stratigraphic horizons within the Lower Chickamauga were previously determined by TVA and have been used as aids in determining the continuity and orientation of the Chickamauga. These keys are described in Table 2.5-7, and appear on the Graphic Logs and Geologic Profiles.

2.5.1.2.3.2.1. Unit A

Unit A is the basal portion of the Chickamauga and is subdivided into the Unit A Lower Siltstone, Unit A Limestone and Unit A
59 Upper Siltstone. The Unit A Lower Siltstone comprises alternating layers and laminations of maroon and gray calcareous siltstone, gray limestone,
55 and maroon, gray, and black chert layers and nodules. The constituent percentages of the alternating layers of the Chickamagua units were determined by measurement of core samples and interpretation of geophysical
38 logs. Siltstone constitutes about 45 percent of the Unit A Lower Siltstone. Mineralogy studies indicate the composition of

the siltstone layers to be about 30 to 50 percent carbonate minerals (calcite plus dolomite), about 30 to 50 percent minute particles of
38 disseminated silica (quartz plus chert) with the remainder being clay minerals. Iron oxide, up to about 2 percent, is also present and apparently imparts the maroon color to the siltstone. Limestone constitutes about 50 percent of the Lower Siltstone with the remaining 5 percent being chert. Mineralogy studies for the limestone layers indicate about 80 to 98 percent carbonate material and 2 to 10 percent clay minerals; the remainder of the samples consist of finely disseminated quartz. The thickness of the Lower Siltstone is variable because its lower boundary is the undulating disconformity at the top of the Knox. The stratigraphic thickness as determined from borings ranges from 151 to 207 feet, averaging about 175 feet.

The Unit A Limestone segment is light to dark gray limestone with laminations and layers of maroon and gray calcareous siltstone. About 75 percent of this segment is limestone and 15 percent is siltstone. Chert nodules occur within the Unit A Limestone and constitute about 10 percent of this segment. Mineralogy studies show the limestone contains about 5 to 45 percent chert and clay mineral impurities. The thickness of the Unit A Limestone ranges from 67 to 100 feet in the borings, averaging 85 feet.

The Unit A Upper Siltstone consists of maroon and gray calcareous siltstone with laminations and layers of gray argillaceous limestone. The siltstone constitutes about 80 percent of this segment with remaining portion being argillaceous limestone. The argillaceous limestone occurs as thin laminations within predominant siltstone sections and as scattered layers up to a few inches thick. Mineralogy studies show that the siltstone contains about 35 to 50 percent carbonates (calcite and dolomite) and the argillaceous limestone contains about 50 to 70 percent carbonates. The stratigraphic thickness of the Upper Siltstone in the borings ranges between 191 and 216 feet, averaging about 203 feet.

55 The Category I structures, except for the Steam Generator Maintenance Bay and the Fuel Oil Storage tanks, are located directly on the Unit A Upper Siltstone structure.

The Steam Generator Maintenance Bay will be founded on competent Unit B Limestone. The Fuel Oil Storage Tanks will be supported on compacted Class "A" structural backfill placed on the Unit A Upper Siltstone. The detailed engineering characteristics of the foundation materials, both static and dynamic, are presented in Section 2.5.4.

59 2.5.1.2.3.2.2. Unit B

Unit B of the Chickamauga Group is light to dark gray limestone and argillaceous limestone which contains chert nodules. It contains laminations and layers of dark gray calcareous siltstone, and is locally nodular and fossiliferous. Although no borings penetrated the entire Unit B sequence, its stratigraphic thickness is estimated to be 240 feet.

2.5.1.2.3.2.3. Undifferentiated Chickamauga

The sequences of the Chickamauga Group above Unit B have not been differentiated in the same manner as the lower units during this investigation. A general description of the Undifferentiated Chickamauga is alternating maroon and gray limestone, shale and siltstone.

2.5.1.2.3.3. Terrace and Alluvial Deposits

Terrace deposits form a veneer over portions of the site as shown in Figure 2.5-5. This material is high-level ancient alluvium deposited by the Clinch River when stream levels were much higher than at the present. Such deposits are generally regarded by geologists as Pleistocene or Pliocene. The areal extent of the terrace deposits at the CRBRP site is shown on Illustration 7 of Supplement 2. The terrace deposits consist mainly of orange and red silty clay with thin layers of rounded quartz, chert and quartzite gravel.

The largest area of terrace deposits was investigated by a series of twenty-two test pits and seven test borings. Twenty-one test pits and all the test borings penetrated the entire thickness of terrace materials. The terrace deposits at the site exist between elevations 750 and 840.

No materials conducive to absolute age dating techniques were encountered in these test pits or borings. The age of this terrace is probably Tertiary. This age estimate is based on the fine-grained composition and lack of carbonaceous materials.

Archeological excavations at the CRBRP site have yielded datable organic materials in the alluvium that underlies the Clinch River floodplain. The oldest material dated was about 2500 years old (Reference 159).
38 27 The alluvium layer was penetrated by boring 45 and consists of sand and clay extending to a depth of about 32 feet.

2.5.1.2.4 Structure

The attitude of the bedding planes and joints at the CRBRP site were calculated utilizing data obtained from surface mapping and by defining the orientation of key stratigraphic horizons traceable between borings (3 point solution). Interpretations of structural features are based on these data as well as observations of outcrops, rock core, and geophysical bore hole logs.

2.5.1.2.4.1 Bedding Orientation

Outcrop measurements provided the general attitude of the rocks over the site peninsula. In addition, small scale structural distortions were noted during surface mapping. From outcrop data, the rock bedding was measured to strike between N35°E and N 60°E and dip to the southeast between 20° and 52°. The average strike is N 52°E and the dip is 37° to the southeast.

As can be seen from these measurements the attitude of the rocks is quite consistent. However, small scale drag folds are common in these rocks and were also observed at the site. Such features include tight folds or shears which extend over short distances - a few feet to a few inches.

The strike of the bedding, based on stratigraphic keys (Table 2.5-7) is between N48°E and N60° and the dip is to the southeast between 24° and 35°. The average strike from stratigraphic keys is N52°E with a 31° dip to the southeast.

59 The TVA has documented their experience over the past 40 years with major construction projects in the Valley and Ridge in the Bellefonte Nuclear Plant PSAR. They state that no residual stress effects have been observed in the area. Therefore, no specific investigations of unrelieved residual stress in the bedrock have been made.

2.5.1.2.4.2 Jointing

Four high-angle joint sets were observed in the rock outcrops (Figure 2.5-14). One set occurs along bedding (weathered partings) having an average strike of N52°E, and dips to the southeast at 37°. A second joint set has the same strike as the bedding and dips at an average of 58° to the northwest, or about perpendicular to the bedding. Two joint sets were observed to strike nearly perpendicularly across the strike of the bedding direction: N25°W, 80° southwest dip and N65°W, 75° northeast dip. Figure 2.5-15 graphically illustrates the joint systems. In outcrops, the bedding plane joints (partings) were observed to be spaced about one to five feet apart. Most joints observed are hairline cracks with joint surfaces being stained by weathering.

The four joint sets described above were also observed in the core. Because of their orientation, the joint set along bedding (partings) is the one best exhibited in the core, with the set dipping to the northwest being prominent also. The boring data indicate the joints of the N52°E, 58°NW set are spaced about one to six feet apart. A boring inclined in a southwesterly direction indicates the northwest-striking joint sets are spaced one to six feet apart.

2.5.1.2.4.3. Small Folds and Minor Dislocations

During geologic surface mapping for the CRBRP site investigation, a small tight fold and three minor shear dislocations were observed. Minor shear dislocations or offsets are interbed adjustments which formed contemporaneously with the regional thrust faults and represent displacements of traceable beds measured in terms of inches or at most a few feet. Since the original mapping, an additional small tight fold (F_2, Illustration 7 of Supplement 2) was exposed in a bank of a newly constructed road. Illustration 7 shows the locations of the two small tight folds and three minor shear dislocations that were observed in outcrops.

The two small tight asymmetrical folds (F_1 and F_2 shown on Figure 2.5-17c, d) have an amplitude of a few feet with a comparable width. These tight folds are overturned to the southeast and plunge to the southwest. Both tight folds are composed of layers of chert and clay. No "keys" or marker beds are recognizable in these outcrops.

A minor shear dislocation (D_1) shown on Illustration 7 is in the Unit A Upper Siltstone. This dislocation strikes north-south and dips steeply to the east. Left-lateral displacement of the out-
cropping limestone bed in the plane of the ridge slope was approximately
2 feet. The second shear dislocation (D_2) is in the lower portion of
the Unit A Upper Siltstone. This dislocation strikes north 25° west
and dips steeply to the west. Right-lateral displacement of about 2
feet was observed. The third minor shear dislocation (D_3) is in the
38 Undifferentiated Chickamauga. This dislocation strikes north-north-
east; dip could not be determined due to limited exposure of the
27 outcrop. Apparent left-lateral displacement of about 2 feet was
observed.

38 27 Minor undulations in the attitude of the lower Chicka-
mauga units have been interpreted from contouring the elevation of
stratigraphic keys (see Table 2.5-7) observed in the rock core. Immediately
27 northwest of the Nuclear Island, at borings 9, 31, 34, 40, 47, 52 and
38 53, is the nose of a small northeast-plunging anticline (shown on Ill-
ustration 7). The closure, or height, of this fold is about 60 feet at
boring 40, about 25 feet at boring 52 and declines to 0 near boring 9.
It merges with the regional inclination of the strata between borings
47 and 15, and is considered to be a wrinkle imposed on the regional
59 inclination during the late Paleozoic.

In the core, slickensided joint and bedding surfaces occur
sporadically, and an ancient rehealed shear zone can be distinguished
in the lower portion in the Unit A Limestone. This shear zone is
essentially a zone of interbed slippage characterized by a combination
of slickensides, calcite veins, and 1-inch to 1-foot segments that are
either severely warped or brecciated. It was fully penetrated by 37
borings, and ranges from 19 to 46 feet in thickness with an average thickness
59 of 35 feet thick. Locally, such as in the vicinity of boring 10, this
zone cannot be distinguished.

38 The minor structures observed at this site, including the inter-
bed slippage noted in the core, are common to the region and represent
ancient adjustments.

2.5.1.2.4.4. Weathering

Rock at the site weathers to clayey residual soil which locally
55 contains chert gravel. The depth of soil varies from 1 foot to 58 feet
in borings in the vicinity of the Category I structures and is a maximum
of 78 feet deep in the northeastern portion of the site. The Unit B
55 Limestone residuum ranges from 8 to 58 feet deep in the borings and soil
overburden thickness above the Unit A Limestone ranges from 12 to 54 feet.
The Unit A Upper Siltstone residuum ranges from 1 to 29 feet deep in the
borings and the soil overburden thickness above the Unit A Lower Siltstone
38 ranges from 11 to 39 feet. Northwest of the Category I Structures the
Knox dolomite occurs and is covered with overburden ranging from 2 to
78 feet in the borings. The geologic profiles (Figures 2.5-7 through
2.5-11) show the variance in overburden thickness.

Weathering within the rock occurs as solution widening of joints
and bedding features. In areas where the limestones and dolomites outcrop
at the surface, rock weathering along joints is the greatest (0 to 73 feet
below top of rock). In areas where the Upper Siltstone outcrops at the
surface, rock weathering is less (0 to 57 feet below top of rock). The
59 geologic profiles and Nuclear Island subsurface profiles show a zone of
weathered rock whose base is identified as the Top of Continuous Rock.

The Top of Continuous Rock is defined as the rock level below
which there are no significant discontinuities. This level was ultimately
selected for each boring by detailed examination of the core including
evaluation of the percent recovery, rock quality designation (RQD),
weathering, joint and bedding plane parting frequency, water pressure tests
and geophysical logging of the open bore-holes. Major emphasis was
placed on visual inspection of the core. During this inspection, the
criterion followed in establishing the Top of Continuous Rock was to
determine if the discontinuities would have any effect on the mass behavior
of the rock. Several experienced engineers, hydrologists and geologists
examined all the core to establish this level. Subsequently, inspections
and evaluations were made by Professors G. Sowers and A. J. Hendron, (geo
technical consultants to the project) and agreement was reached on the
selected elevation by detailed examination of the core in a number of
representative borings. The foundation level of the Nuclear Island common 25
mat has been established below the Top of Continuous Rock level. Therefore,
59 no cavities or significant discontinuities occur beneath the mat.

Amend. 59
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The topography of the site is governed by the rock types and their resultant residual soil cover. The Unit A Upper Siltstone residuum is a readily erodible soil type which results in a thin soil cover. A small valley has developed through this easily erodible material In this unit, sound and continous rock free of significant weathering and solution occurs within about 50 feet of the ground surface in the outcrop area.

The slope immediately northwest of the previously discussed valley is underlain by the Unit A Limestone. In this area, the clay residuum is more resistant to erosion resulting in the thicker overburden above the limestone. The Unit A Limestone dips beneath the impermeable Upper Siltstone near the bottom of the valley. 25

The ridge along the southeast limits of the Category I structures is underlain by the Unit B Limestone. The overburden
55 thickness in this area ranges from 8 to 58 feet as discussed above. Weathering and solutioning of the underlying limestone has extended to a maximum depth of about 100 feet. Below this depth, joints penetrated by borings do not indicate any significant weathering or solutioning. Results of borings drilled in the area of the Steam Generator Maintenance Bay, which will be founded in the Unit B Limestone, indicate that the proposed foundations for this structure
55 (at Elevation 780 and 760) are supported on competent rock.

The Unit A Lower Siltstone outcrops on the crest and the northern flank of the ridge northwest of the small valley. In this area the depth of overburden extends to a maximum depth of about 39 feet. The rocks which consist of siltstone and limestone have undergone weathering and solutioning to about 85 feet below top of rock. The depth of maximum weathering and solutioning corresponds to about elevations 750 to 830 feet.

In the vicinity of B-40, a small anticlinal fold occurs
59 (described in section 2.5.1.2.4.3). Within the anticlinal area, the rock is apparently more extensively jointed, facilitating weathering to a maximum depth of about 108 feet. The lowest elevation of weathering and solutioning within this anticlinal area is about 704 feet at borings 34 and 56.

The Knox Group outcrops on the northwest flank of the ridge northwest of the small valley. The most prominent feature of the Knox outcrop zone is a band of sinks which are generally aligned along the strike of the Knox. Eight borings, which penetrated through the lower Chickamauga strata into the Knox Group did not encounter any weathering or solutioning at the contact or within the Knox.

The soil-rock profile for inclined strata of limestone and dolomite is typically quite irregular, exhibiting rock pinnacles and intervening gaps. These irregularities are developed by solution-widening along joints and to a lesser extent along bedding. The wide range of soil depths over the Knox Group, Chickamauga Unit A Limestone and Chickamauga Unit B Limestone reflect these irregularities.

Weathering has advanced from the surface along joints and partings
55 in the rock to produce soil seams and cavities. that range from less than one
59 inch to 16 1/2 feet in the borings. Frequency and size of soil seems and
cavities produced by weathering and solutioning diminish with depth. The
occurrence of such discontinuities is dependent on the surface exposure of
the individual limestone units and joint frequency. The term "clay seam"
designates an interval of clay within the rock; the term "cavity" designates
59 an open void within the rock.

Figure 2.5-16 is a contour map of the Top of Continuous Rock.
With the exception of an area west of the Nuclear Island, the Top of 25
Continuous Rock generally follows the same configuration as the surface
topography. The foundation level of the Nuclear Island
59 38 (Elevation 715') is about 20 to 85 feet below the Top of Continuous
18 Rock level. The Top of Continuous Rock rises to about Elevation 825
feet along the crest of the ridge east of the Nuclear Island and
above Elevation 800' along the ridge northwest of the Plant. An
elongate low area in the Top of Continuous Rock occurs immediately west
of Plant Island beneath the flank of the small ridge. Borings indicate
this low area to be centered near the small anticlinal fold described
59 in section 2.5.1.2.4.3. Apparently, jointing is enhanced over this small
fold facilitating deeper ground water circulation and weathering. Even
in this area the borings show that the lowest Top of Continuous Rock
elevation within 75 feet of the Nuclear Island is about 725 feet. This sound
and continuous rock is overlain by 40 to 60 feet of hard rock which contains some weathered seams.

Seismic refraction traverses were made at the site by Weston Geophysical Engineers, Incorporated. A total of 6360 linear feet of refraction profiling were made along 5 traverse locations as shown on Figure 2.5-6. The results of the refraction profiling are included in Reference (133).

Figures 2.5-12 and 2.5-13 are summarized reproductions of the refraction traverse on which boring data has been superimposed. These figures show the correlation of boring information to the refraction data. The velocity zone of 2000-4000 feet per second corresponds with the soil overburden determined by borings. The Top of Continuous Rock determined by the borings always occurs well within the rock zone which is generally distinguished by velocities above 10,000 feet per second.

Intervals of clay up to about 6" thick were indicated as "clay seams" by a dash, and thicker intervals were indicated by a bracket. Clay seams within the rock which were identified on the graphic logs by brackets were encountered in five borings: 25, 31, 56, 62 and 89. Symbols for clay seams and cavities are illustrated in the legend of each individual graphic log. In three of these five borings, "standard penetration tests" were performed within the intervals indicated as clay seams. Samples of the material recovered during the standard penetration testing, and standard penetration resistance values, provided the basis for designating these
38 intervals as clay seams.

A five foot thick interval designated as a clay seam was encountered within the Unit B Limestone in boring 25 (ground surface elevation 854.4) at a depth of 24.5 to 29.5 feet, or from 10.5 to 15.5 feet below the top of rock. A standard penetration test performed within this interval indicated a penetration resistance of 13 blows per foot. The sample recovered by the penetration test had a visual appearance similar to the nearby residual overburden soils and appeared to be derived by in-place weathering of the surrounding rock.

Three soil layers, ranging from 5.2 to 8.2 feet thick, were encountered within the Unit A Limestone in boring 31 (ground surface elevation 802.8). Standard penetration tests were performed within each of these soil intervals to determine consistency and to obtain samples of the materials. Results of the standard penetration tests were:

DEPTH OF CLAY INTERVAL (FT)	STANDARD PENETRATION TEST DEPTH (FT)	STANDARD PENETRATION RESISTANCE
54.0-59.2	57.8-59.2	50 blows per 4" penetration
61.0-68.8	66.0-67.5	7 blows per foot
	67.5-68.0	13 blows per foot
	68.0-68.8	50 blows per 3" penetration
70.9-79.1	74.3-75.7	50 blows per 4" penetration
	77.0-77.8	50 blows per 3" penetration

Examination of the samples recovered during the standard penetration testing indicated that these soil layers were residual materials derived by in-place weathering of the surrounding rock. The standard penetration resistances
59 further indicate that these layers contained thin lenses of rock.

27

Boring 56 (ground surface elevation 804.8) was drilled at an angle of about 19^{o} from the vertical, bearing toward the southwest. Standard penetration testing of the overburden soils was not performed due to the inclination of the boring. Three intervals within the rock which have been designated as clay seams and two cavities (0.4 and 6.7 feet) were encountered within the Unit A Limestone. Standard penetration tests within the three clay seam intervals, ranging from 3.6 to 5.7 feet thick, were not performed because of the inclination of the boring. However, based on observations made during drilling of the boring it was determined that these intervals contained soil or soft rock.

38

Boring 62 (ground surface elevation 844.8) was made in a sinkhole in the
38 Knox formation more than 2400 feet away from the plant site area. A 6 foot thick soil layer was encountered in the boring between the depths of 36.5 and 42.5 feet, or from 5.5 to 11.5 feet below the top of rock. The standard penetration resistance of the soil layer was 2 blows per foot. Visual examination of the recovered sample revealed that some rounded sand-size fragments were included in the material. Therefore, it is possible that this interval might have been a cavity that was filled with soil materials transported by water.

A 4.2 feet thick clay seam was encountered within the Unit A Upper Siltstone in boring 89 (ground surface elevation 818.5) between depths 13.1 to 17.3 feet, or from 0.6 to 4.8 feet below the top of rock. Standard penetration
27 tests were not performed in the interval. However, based on observed drilling pressures and rates it was concluded that the interval was a soft rock or very hard residual soil seam.

38

Cavities within the rock units have not been shown on the geologic
38 profiles through the plant site area (Figures 2.5-7 through 2.5-10) because:

(1) All cavities that were encountered in the borings are above the
38 Top of Continuous Rock;

(2) Most of the cavities were 1 foot or less and would be less than 0.01-inch thick on the profiles;

(3) The clay seams and cavities found above the Top of Continuous Rock represent irregular weathering particular to a boring location, and are not traceable between borings.

38 Figure 2.5-11 shows the cavities encountered in boring 61 and the interval in boring 62 (described above) which may be a zone that was once
27 59 an open void but is now soil filled.

In addition to borings located under all proposed Nuclear Island
structures, geophysical refraction, cross-hole and up-hole surveys, sonic
logging and in situ testing were conducted in this area. A grouting program
38 was completed on a section of the foundation strata located on the
west side of the Nuclear Island (Plant North). With the interbedded
siltstone and limestone beds dipping to the east, the selection of this
zone represented that area where the Unit A limestone is overlain by the
shallowest depth of siltstone and consequently was considered to be most
susceptible to development of solutioning within the bearing influence of
the Nuclear Island structures. In the Unit A limestone outcrop area, the lime-
stone has undergone weathering and solutioning along joints and partings to
a maximum depth of about 100 feet. Two lines of borings were drilled in this
area to investigate solutioning where siltstone cover is minimum. Ten borings
spaced at approximately 50' intervals were completed on a line 50' west of the
western limit of the Nuclear Island. A second line consisting of seven borings
spaced at approximately 70' intervals was completed along the western limit of 25
the Nuclear Island. Inspection of the core from these borings indicated no
evidence of solution activity below the proposed bearing elevation for the
59 structures.

Groundwater elevations generally coincide with the Top of
Continuous Rock with local variations apparently influenced by surface
topography. A detailed discussion on groundwater at the site is presented
in Section 2.4.13. 25

In the vicinity of the Nuclear Island structures, surficial examination
59 indicated a few small diameter depressions and openings occurring in
the Unit B and Unit A Limestone outcrop areas. These features are localized
and less than 2 feet in diameter having the appearance of stump holes. These
features were manually probed and in most instances loose materials were
59 penetrated to depths of 3 to 10 feet. No evidence of deeper solutioning or
continuity of solutioning to other nearby areas was found. Analysis of the
38 boring data also confirmed that voids or cavities noted on the logs generally
appeared to be localized in extent. Apparently, some of these features, 25
particularly those penetrating several fee , are related to ravelling or
erosion of overburden soils into solution widened joints. The refraction
59 profiles indicate several areas of late wave arrivals in the soil over-
burden. These late arrivals could be caused by shallow ravelled zones
similar to the small diameter depressions observed in limestone outcrop
areas.

In addition to the small diameter holes, larger sinkholes obviously associated with solutioning were observed in the Unit B Limestone outcrop area. One such sinkhole was observed near the southeast corner of the Nuclear Island. This sinkhole is approximately circular in surface expression and about 15 feet in diameter. The sinkhole was investigated by four borings (41, 46, 48 and 49). The top of rock in these borings illustrates the irregularity of the limestone rock surface due to solutioning along joints. The borings show that the Unit B Limestone contains soil seams and voids up to about 3 feet in their vertical dimension. The maximum depth of weathering in these borings is about 57 feet (boring 49) and the lowest elevation of solutioning and weathering is 754 feet (boring 41). Unit A Upper Siltstone was encountered in each boring below
the Unit B Limestone. Upon penetrating the Unit A Upper Siltstone, no
59 solutioning, weathering or drilling water losses occurred in the borings.
The borings demonstrate that the surface sinkhole is associated with solutioning along joints that is confined within the Unit B Limestone.

Large sinkholes occur several hundred feet from the Nuclear Island structures in two areas. In the northeast corner of the site, sinkholes occur in a broad swale underlain by Knox dolomite. Elongate depressions ranging up to 175 feet in their longest dimensions occur in the bottom of the swale, and small sinkholes ranging from 3 to 75 feet across are located on the southern slope of the swale. These sinkholes are about 1 to 15 feet deep.

59 Outcrops of dolomite on the southeast side of the larger sinkholes
exposed steeply inclined rock surfaces whose orientation corresponds
38 to the northwest dipping joint set (Section 2.5.1.2.4.2). Nine
borings (B-60 through B-68) made across two of the sinkholes encountered a wide range of soil overburden thickness (2 to 78 feet). These two sinkholes are approximately 100 and 175 feet in width, respectively, and about 15 feet in depth (subsidence) in the Knox Formation. The borings were conducted to evaluate the continuity of solutioning across the depression, the degree of weathering and the rock quality. It
38 was concluded that even in areas where sinkholes had formed to a
significant size, no major increase was noted in the depth of weathering compared with areas where limestone strata had not been exposed to solutioning at the surface. A typical geologic profile of the sub-
surface conditions is shown on Figure 2.5-11. The top of rock in these 25
borings is irregular, illustrating solutioning along the joint set (strike N52°E, northwest dip) which is commonly mapped on nearby outcrops. The maximum depth of weathering in these borings is about 107 feet at boring 61. The zone between the Top of Continuous Rock and the
59 top of rock contains solution voids which have a maximum vertical dimension
of about 4 feet. The rock is nearest the ground surface along the butt slope of the ridges. Near the swale the rock deepens gradually, however, borings within the sink depressions found markedly deeper rock. Progressing northward the rock surface rises slightly. The sudden increase in depth to rock within the sinkholes results from solutioning along the high angle northeast striking (N52°E, 58°NW) joint set. This joint is apparent in the rock walls which border the southern side of the sinkholes, and the sink area is aligned along the strike of this joint. Deep solutioning along steeply dipping joints is commonly found in the Knox Formation.

Over the Knox area the relationship of bedding to joint patterns in borings and outcrops corresponds to the regional trend. Numerous joints with a N 52°E strike and northwest dip have been recorded in the rock core and surface outcrops in this area. The lowest sections of all these borings encountered rock of the Knox Group which contains no significant solutioning or weathering. These observations are further conclusive evidence that solution weathering in the Knox progresses from the surface along joints, with high angle joints being most susceptible to weathering. The major part of the rock mass which occurs between joints has not been affected.

The largest sinkhole within the site limits is located about 1000 feet east of the Nuclear Island in the area underlain by Chickamauga Undifferentiated. This sinkhole is about 850 feet long and up to 150 feet wide having a maximum depth of about 14 feet. Limestone forms a portion of the southeast side of this sinkhole and also exhibits a steep rock surface whose orientation corresponds with the N52°E, northwest dipping joint set.

Locations of sinkholes in the Unit A limestone, Unit B limestone, Knox Formation and Chickamauga Undifferentiated are shown in Illustration
59 7 of Supplement 2.

From the above it is considered that the development of solutioning has been slow and with the added protection of a siltstone cover, it is concluded that there is no possibility of encroachment of solutioning in the Unit A limestone possibly resulting in voids or cavities occurring within the bearing influence of the structures during the lifetime of the plant. 25

2.5.1.2.5 Conclusions

The site is underlain by early Paleozoic siltstones, limestones and dolomites which were subjected to forces which culminated at the close of the Paleozoic Era. This resulted in the formations at the site being tilted. The rocks beneath the Category I foundations are hard, sound, continuous and free of weathering defects.

The ancient forces which provided the attitude of the beds at the site caused some slight distortions and minor dislocations in the rock at the site. These minor distortions and dislocations are common throughout the Valley and Ridge and do not constitute tectonic structures. Distortion and dislocations which have been identified at the site include: 1) a minor anticlinal fold northwest of the Nuclear Island; 2) an ancient healed shear zone within the lower portion of the Unit A Limestone, and 3) two small tight folds and three minor shear dislocations at the locations shown on Illustration 7. These structures
59 are not significant geologically.

Four prominent joint sets have been identified from outcrops and core borings at the site. The orientation of the joints have been considered in the design of excavated construction slopes.

Solutioning has been identified within the outcrop bands of limestone and dolomite formations. Solutioning primarily occurs along steeply inclined joints. Where limestones and dolomites are vertically covered by unweathered siltstones, borings have not found any evidence of solutioning within the limestone or dolomite.

The Top of Continuous Rock has been defined from the borings. The Nuclear Island common mat and Emergency Cooling Tower foundation mat have been established below the Top of Continuous Rock. The Class 'A' structural fill which supports the Fuel Oil Storage Tank overlies
59 continuous rock.

The borings conducted to date have encountered only hard, sound, and continuous siltstones and limestones below the foundation levels of Category I structure locations. Weak or unstable zones have not been encountered. There is also no evidence of significant zones of solutioned, cavernous, or highly weathered areas which could produce subsidence. Solution activity could be a problem during dewatering of the excavation. Extensive solutioning in cut rock faces could permit heavier than normal inflows of water. This would be mitigated by additional pumping and local grouting.

2.5.2 VIBRATORY GROUND MOTION

2.5.2.1 Geologic Conditions of the Site

The lithologic, stratigraphic, and structural conditions of the site and the regions surrounding the site, including its geologic history are described in Section 2.5.1.2.

The Nuclear Island structures and the Emergency Cooling Tower will be founded on rock. The Category I Fuel Oil Storage Tanks will be supported on compacted Class 'A' structural backfill overlying competent siltstone. Consequently, both rock and overburden response to vibratory
59 motions are a consideration in evaluating foundation bearing capability (Ref. 1).

2.5.2.2. Nearby Tectonic Structures

A tectonic structure is a large scale dislocation or distortion within the earth's crust with its extent measured in miles. The tectonic structures in the Valley and Ridge consist
38 of numerous Paleozoic thrust faults and folds (see Figure 2.5-2.) These structures were formed during the Allegheny orogeny at the end of the Paleozoic Era (Ref. 81, 101). The CRBRP site is situated between the traces of two inactive tectonic structures: the Copper Creek and Whiteoak Mountain thrust faults (see Figure 2.5-17). The inactive tectonic structures within the Valley and Ridge do not affect the determination of the Safe Shutdown Earthquake; however, the nearest two tectonic structures to the CRBRP site are discussed in Section 2.5.3 and summarized below.

2.5.2.2.1 Copper Creek Fault

The Copper Creek Fault is mapped approximately 100 miles in length and the CRBRP site is located near its mid-point. The shortest distance from the CRBRP Plant Island to the fault trace is about 3,000 feet south. In the site vicinity the fault strikes north 52 degrees east and dips southeast (away from the site) at an angle of about 25 degrees measured at the ground surface. Nearby borings indicate that the dip angle decreases with depth. In the site area, the Copper Creek Fault has thrust the Rome Formation over younger rocks of the Chickamauga Group for a horizontal distance estimated in miles. The stratigraphic displacement is approximately 7,200 feet (Ref. 54). About 65 miles southwest of the site, the fault becomes a complex zone and merges with the Whiteoak Mountain Fault.

The trace of Copper Creek Fault was identified at several outcrop locations in the vicinity of the site and in boring 43. Additional data on the Copper Creek Fault was obtained from two test wells, the Joy Test Well (Ref. 13) and B29 (Ref. 89), both located on the Oak Ridge Reservation about four miles east of the site.

The best exposure of the Copper Creek Fault near the site is at the I-40 road cut about two miles southwest of the site. The hanging wall is a dark gray dolomite of the Rome Formation, and the foot wall is a gray limestone of the Chickamauga Group. Except for minor undulations, the beds on both sides of the fault are undisturbed. The Rome beds strike north 55 degrees east, dip 35 degrees southeast, and the Chickamauga beds strike north 53 degrees east and dip 29 degrees southeast. The apparent dip of the fault trace is 20 degrees, which implies a 25 degree dip for the fault plane at this location.

2.5.2.2.2. Whiteoak Mountain Fault

The Whiteoak Mountain Fault is composed of a main thrust fault and several subsidiary thrust faults branching from the main thrust. The fault extends southwesterly across the state from a point near Clinton, Tennessee, about four miles northeast of the Oak Ridge Reservation. The Rome Formation (Cambrian) overlies younger rocks of Ordovician Age along the Fault. The nearest trace of the Whiteoak Mountain fault system is 1.7 miles from the site. The main fault and its subsidiaries dip to the southeast, at an inclination estimated to be 45 to 50 degrees, with the angle probably decreasing with depth. Data from outcrops and from a deep core hole drilled about 4 miles east of the CRBRP site (Ref. 13) indicates that the Whiteoak Mountain Fault and its subsidiaries are deeper than 2,000 feet at the site.

2.5.2.2.3 Radiometric Age Determinations

Representative samples of the mylonite and the adjacent hanging wall (Rome Formation) and foot wall (Chickamauga Group) of the Copper Creek Fault were obtained for age determinations. Whole rock potassium-argon dating gives an age of about 760 and 530 millions of years for the hanging wall and foot wall country rocks, respectively (See Table 2.5-1). The age of the mylonite is about 285
59 million years. These dates confirm previous work of experts in the Valley and Ridge (Refs. 64, 69, 81, 97, 101) and indicate that the last active period of Valley and Ridge tectonism was late Paleozoic.

38 The potassium and argon isotopic analyses used in radiometric dating were carried out by Dr. J. M. Wampler, Associate Professor, School of Geophysical Sciences, Georgia Institute of Technology.

Because the analyses for potassium and argon were carried out on different portions of a sample, each rock sample was crushed and pul-
38 verized to a size sufficiently small so that equivalent portions could be obtained by standard sample - splitting methods. Potassium and argon contents are reported in terms of sample weight after drying at 110°C.

Samples for potassium analysis were dissolved in a mixture of hydrofluoric and perchloric acids. After evaporation to small volume, the material was taken up in a standardized diluting solution (4% HNO_3 plus 0.1% NaCl as an ionization suppressant), and further diluted, when necessary, until the potassium content was less than 5 parts per million by weight. The potassium content of the solution was then determined by atomic absorption spectrophotometry. The precision of this method is such that error limits of ± 0.5% of the measured K-content may be assigned to most determinations. Systematic errors may be sig-
27 nificant for samples with very low potassium content (less than 1%).

Intensity estimates provide the basis for the epicentral locations of earthquakes prior to about 1960. Before 1800, much of the region was so sparsely populated that the epicentral locations were identified with the scattered towns, possibly tens of miles from the actual epicenters. Since then, the greater population density, better communications, and more seismograph stations have made it possible to locate areas of greatest intensity within a few miles. Within the past few years strong motion seismographs have been installed at several nuclear power plants being constructed in the general region.

Surface intensities at the site have not been directly observed. The intensities which occurred at the site have been estimated based on the epicentral intensity and distance from the CRBRP site (Ref. 11).

A list of the earthquakes which have produced estimated site intensities of III Modified Mercalli (MM) or greater is presented on Table 2.5-2. In each case the intensity is assessed in terms of the Modified Mercalli scale shown on Table 2.5-2.

The site area has experienced numerous light to moderate earthquakes. The maximum site intensity associated with these earthquakes is VI-VII MM. The site investigation has not produced any physical evidence which can be associated with any earthquakes.

2.5.2.4 Engineering Properties of Materials Underlying the Site

The engineering properties of materials underlying the site are discussed in section 2.5.4.2.

2.5.2.5 Earthquake History

The epicenters of all the reported significant earthquakes in the southeastern United States, with Modified Mercalli (MM) intensities of V or more,have been plotted as shown on Figure 2.5-18 and are listed in tabular form in Table 2.5-3.

It should be noted that these surface intensities are based on the worst effects noted and reflect the response of the poorest surface foundation materials. The plot of earthquake epicenters shows that there have been 11 recorded earthquake epicenters within a 50 mile radius, 19 epicenters within a one hundred mile radius and 44 within a two hundred mile radius of the CRBRP site.

The largest earthquakes ever recorded in the southeastern United States are the New Madrid earthquakes of December 16, 1811, January 23, 1812, and February 7, 1812, and the Charleston, South Carolina earthquake of August 31, 1886. The epicentral intensity at New Madrid, 300 miles west-northwest of the site, is estimated to be XII MM and the epicentral intensity of the Charleston earthquake 315 miles southeast of the site is estimated to be IX MM (Ref. 120).

The observed surface intensity in the vicinity of the site from the New Madrid earthquakes is estimated to be VI-VII MM (Refs. 121 and 126). Topographic changes reportedly resulted from these earthquakes over an area of 30,000 to 50,000 square miles. At least a two million square-mile area was shaken (Ref. 129). Only a very small amount of damage was reported, mainly due to lack of inhabitants. The New Madrid earthquake produced the greatest ground motion at the site of any earthquake in historic time. As previously stated, these earthquakes have been assigned an intensity XII which is described by "total destruction" at the epicenter.

The observed surface intensity in the vicinity of the site from the Charleston earthquake is estimated to be VI MM (Ref. 111). This earthquake is reported to have been felt over an area of two million square miles. For its reported epicentral intensity, the Charleston earthquake was felt over a large area.

A moderately large earthquake within the southeastern region which was felt at the site was the Giles County, Virginia, earthquake of May 31, 1897, with a reported epicentral intensity of VII-VIII MM (Ref. 117). The Giles County earthquake, whose epicenter is about 220 miles northeast, is estimated to have been felt at the site at about intensity V MM (Ref. 112).

As stated in Section 2.5.2.3, the greatest historic ground motion of the site is estimated to have been intensity VI-VII MM and was produced by the New Madrid earthquake which occurred about 300 miles from the site.

2.5.2.6 Correlation of Epicenters with Geologic Structures

The tectonic structures which occur in the CRBRP site area and region have been previously described. The tectonic structures or thrust faults within the Valley and Ridge Province are considered in the literature and by recognized geologic experts as ancient and inactive. Results of the recently completed Law Engineering site
38| investigation substantiate this.

Scattered earthquakes occur in the Valley and Ridge and their "normal" focal depth is 50,000 to 65,000 feet, well within the basement rocks. On February 18, 1964, an earthquake was reported by the U.S. Coast and Geodetic Survey southwest of the site with its epicenter in the Valley and Ridge at a focal depth of 15 kilometers - 49,000 feet (Ref 101). As shown in section 2.5.1.1.2, the tectonic structures in the
46 Valley and Ridge terminate at a sole fault which occurs at a depth of about 9000 feet. Obviously, earthquakes which occur at depths of 40,000 feet below these shallow structures are in no way related to the structures. When plotted in relation to each other, earthquake epicenters and the ancient, inactive faults exposed at the surface within the Valley and Ridge Province are in no way related.

Since epicentral locations can not reasonably be correlated with tectonic structures, earthquakes are identified with the tectonic province in which the site is located. This province has been designated as the Southern Appalachian Tectonic Province by the NRC
38 in their evaluation of the Sequoyah Nuclear Plant. The Province is bounded on the east by the western margin of the Piedmont Province: on the west by the western Limits of the Cumberland Plateau; on the south by the overlap of the Gulf Coastal Plain Province; and on the north by re-entrant in the Valley and Ridge Province near Roanoke, Virginia (Ref. 101)

46

2.5.2.7 Identification of Capable Faults

There is no geologic evidence of surface faulting within the Valley and Ridge or adjacent geologic regions that is even remotely related to earthquakes that have occurred in historic time. This is supported in Bonilla's review of Historic Surface Faulting in the Continental United States and Adjacent Parts of Mexico (Ref. 113)

46

It is concluded that there are no identifiable capable faults that could be expected to produce surface displacement anywhere within the Southern Appalachian Tectonic Province, within 200 miles of the site.

2.5.2.8 Description of Capable Faults

There is no evidence for any capable faulting within 200 miles of the CRBRP site which may be of significance in establishing the Safe Shutdown Earthquake.

2.5.2.9 Maximum Earthquake

The largest historic earthquake which has occurred in the Southern Appalachian Tectonic Province was the May 31, 1897 earthquake in Giles County, Virginia, with a reported epicentral inten-

sity of VII-VIII MM (Ref. 117). A subsequent re-assessment of the
intensity was performed by Law Engineering Testing Company in conjuction
with Burns and Roe, Inc., which confirmed the VII-VIII intensity value
59 of this earthquake (Ref. 137). 14

2.5.2.10 Safe Shutdown Earthquake

The criteria contained in Appendix A to 10CFR100 "Reactor Site
Criteria", on Vibratory Ground Motion was followed to establish the
Safe Shutdown Earthquake. The most severe earthquake occurrence
associated with the tectonic province in which the site is located may
be assumed to occur adjacent to the site. As noted in Para. 2.5.2.9,
this earthquake was reported (Ref. 117, 137) to have an epicentral
intensity of VII-VIII MM. However, in conformance with Nuclear Regulatory
Commission direction, an intensity rating of VIII MM is being utilized
46 in the CRBRP design.

Numerous correlations between intensity and acceleration have
38 been developed. Figure 2.5-19 indicates a number of these relationships
which have been previously considered acceptable to the NRC. These
include Gutenberg-Richter (Ref. 130, 131), Coulter, Waldron and Devine
(Ref. 114) and Neumann (Ref. 138). The recent relationship proposed by
Trifunac and Brady (Ref. 139) is also shown. The Neumann relationship
46 based on period-acceleration graphs has been used to derive the
appropriate seismic coefficient for design. Based on the above consid-
46 erations, a peak horizontal ground acceleration of 0.25g has been
selected. 14

The Design Response Spectra for horizontal and vertical motions
59 for the SSE are shown in Figures 2.5-20 and 2.5-21. These spectra have
been linearly scaled from Figures 1 and 2 of the USNRC Regulatory Guide
38 Series, Number 1.60, Design Response Spectra for Seismic Design of
Nuclear Power Plants, in proportion to the maximum specified horizontal
ground acceleration of 0.25g. The amplification factors and control
points in Tables I and II of the USNRC Regulatory Guide Series, Number
46 38 1.60, were used.

2.5.2.11 Operating Basis Earthquake

The Nuclear Regulatory Commission criteria state the 14
Operating Basis Earthquake (OBE) shall be specified by the Applicant
and shall be defined by response spectra. The OBE loadings will be
applied to all safety related structures, systems and components and
stresses will be evaluated in accordance with applicable requirements.

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46 Even though the maximum horizontal ground acceleration based
on historical data at the site vicinity is less than 1/2 the SSE, an
46 OBE of 0.125g is selected. This value is equivalent to one-half the
maximum ground acceleration of the SSE being used for CRBRP design
The Design Response Spectra for the horizontal and vertical motion for
59 the OBE are shown in Figures 2.5-22 and 2.5-23, which have been linearly
38 scaled from Figures 1 and 2 of the <u>USNRC Regulatory Guide Series, Number
1.60, Design Response Spectra for Seismic Design of Nuclear Power Plants</u>,
in proportion to the maximum specified horizontal ground acceleration of
46 0.125g. The amplification factors and control points in Tables I and II
of the <u>USNRC Regulatory Guide Series, Number 1.60</u>, were used.
38

2.5.3 Surface Faulting

2.5.3.1 Geologic Conditions of the Site

The geologic history of the site and the lithologic, stratigraphic and structural conditions have been described in the previous section 2.5.1.2.

2.5.3.2 Evidence of Fault Offset

As described in detail in Section 2.5.3.8, the site is located between two ancient thrust faults. These are the Copper Creek Fault 3000 feet south of the Nuclear Island, and the Whiteoak Mountain Fault
46 1.7 miles northwest of the Nuclear Island. These faults are well documented in the literature, and studies made during this investigation verify their reported locations and late-Paleozoic period of activity.

2.5.3.3 Identification of Capable Faults

No capable faults have been identified within five miles of the CRBRP site.

2.5.3.4 Earthquakes Associated with Capable Faults

The scattered and infrequent moderate-intensity earthquakes in this area have deep hypocenters. These minor earthquakes are not related in any way to the ancient Paleozoic faults in the Valley and Ridge.

2.5.3.5 Correlation of Epicenters with Capable Faults

Since there is no relation of earthquakes to capable faults for the site area, a correlation of epicenters with capable faults is not applicable.

2.5.3.6 Description of Capable Faults

As stated in Section 2.5.3.3, no capable faults have been identified within five miles of the CRBRP site.

2.5.3.7 Zone Requiring Detailed Faulting Investigation

As previously stated, no capable faults exist within five miles of the CRBRP site.

2.5.3.8 Results of Faulting Investigation

The Copper Creek and White Oak Mountain Faults were identified at several locations in the vicinity of the site and also at locations several miles to the northeast and southwest of the site. The Copper Creek Fault was identified at locations 1 through 5 shown on Figure 2.5-17 and by borings B-43 and Joy Test Well, designated JTW on Figure
59 2.5-17 (Ref. 13). Farther from the site, and to the southwest the fault
38 was observed and radiometrically dated at an Interstate road cut. The White Oak Mountain Fault was observed at location 6 (Figure 2.5-17).

General System

The Copper Creek and White Oak Mountain Faults are related as shown in Figures 2.5-17, 2.5-17A, and 2.5-17B. The Wallen Valley, Hunter Valley, Clinchport, and Copper Creek Faults enter Tennessee from Virginia. These are all thrust faults, along which the Rome Formation overlies younger formations, usually Conasauga, Knox or Chickamauga. At Sneedville the Clinchport merges into the Hunter Valley Fault and at Clinton, the Wallen Valley and Hunter Valley Faults merge to form the White Oak Mountain Fault. The Whiteoak Mountain Fault roughly parallels the Copper Creek Fault to Cleveland, Tennessee where they merge. The White Oak Mountain Fault terminates northwest of Rome, Georgia. The area where Copper Creek and White Oak Mountain Faults merge near Cleveland is very complexly faulted (Refs. 54, 61, 72).

Copper Creek Fault

In contrast to the White Oak Mountain Fault, the Copper Creek Fault is, with a few exceptions, a single fault plane. Throughout most of its length the Rome Formation is thrust over the Chickamauga Group with a stratigraphic displacement of about 7200 feet. Near Cleveland the fault enters the imbricated zone south of the Whiteoak Mountain Fault and loses its individuality (Ref. 82).

In the area of the site, the trace of the Copper Creek Fault appears along the northwestern flank of a northeast trending ridge named in places, Haw Ridge, Dug Ridge and Hood Ridge. The fault has an average strike of N 52° E in the site area and N 55° E in the Oak Ridge area (Ref. 54). The dip of the fault plane at the surface was measured as 25° to 30° to the southeast. Based on the Joy Test Well data where the Copper Creek fault plane was intercepted at a depth of 1370 feet, the dip of the fault plane decreases with depth (Ref. 13). Seismic reflection profile made by Geophysical Services, Inc. show several reflectors whose dip flattens with depth and these reflectors merge into a common, near-horizontal zone. Both of these data sources are consistent with the thin-skinned structural concept
59 presented in Supplement 2.

In the vicinity of Oak Ridge the fault has thrust over the entire stratigraphic sequence of the Conasauga Formation (2000 feet),
38 all of the Knox Formation (2500 feet) about 1800 feet of the Chickamauga Group (Ref. 34) and, by McMaster's estimate (Ref. 54), 900 feet of the Rome Formation, for total stratigraphic displacement of 7200 feet.

At Location 1 shown on Figure 2.5-17, White Oak Creek cuts across Haw Ridge exposing the Rome Formation above the fault trace. The trace itself is covered with colluvium made of soil and cobbles derived from the Rome Formation.

At location 2, State Road 95 crosses Haw Ridge. Southeast
59 of the fault the Rome Formation comprises thinly bedded yellow, brown and maroon sandstones, siltstones and shales and a thick (in excess of 5') layer of dolomite capping the hill. The fault trace is covered with colluvium derived from the Rome Formation. Northwest of the fault, the Chickamauga Group comprises tan and light gray calcareous siltstone and very argillaceous limestone striking N 45° E and dipping 36° southeast.

Location 3 is a roadcut in a dirt road which crosses Dug Ridge and
59 connects Poplar Springs with Bradbury. The Rome Formation at this location consists of maroon and brown shale, sandstone and siltstone with gray massive dolomite. On the southeast flank of the ridge, the Rome is fractured by many small high-angled offsets. The beds have many low amplitude folds. The trace of the Copper Creek Fault occurs approximately halfway down the northwest slope of the ridge and is covered by colluvium. Below the colluvium, the Chickamauga Group outcrops as a
59 partially weathered light gray to cream calcareous siltstone.

At Location 4, Interstate 40 cuts through Dug Ridge exposing the fault on the southwest side of the Interstate. The Rome Formation consists of dark maroon, yellow and tan alternating sandstone (grading to quartzite), shale, and siltstone with gray dolomite layers. Immediately above the fault contact is a gray dolomite of the Rome. At the contact is a thin (3") tan to cream mylonite layer. Immediately below the mylonite is a two foot thick tan to light gray fossiliferous argillaceous limestone layer of the Chickamauga Group. The remaining exposed Chickamauga is tan, gray and maroon argillaceous limestone and calcareous siltstone. The Rome beds strike N 55° E and dip 35° Southeast, and the Chickamauga beds strike N 53° E and dip 29° Southeast. The beds on both sides of the fault are relatively undisturbed. The apparent dip of the fault trace is 20°, which infers a 25° true dip for the fault plane at this location.

At Location 5, Kingston Pike crosses Dug Ridge. On the southeast flank of the ridge, the Rome Formation consists of brown thinly bedded sandstone, green and tan partially weathered shale, brown and tan siltstone and occasional thin beds of green shale seams weathered to clay. The trace of the fault is covered by colluvium on the northwest slope of the ridge. Northwest of the colluvium the Chickamauga Group outcrops as a light gray to cream argillaceous limestone.

Boring B-43 encountered the Copper Creek Fault at an Elevation of 563.6 feet. The Rome Formation above the fault consists of maroon and brown siltstone, shale, and gray to dark gray dolomite. The fault consists of a 10-inch slickensided zone. Below the fault the Chickamauga is a gray crystalline slightly argillaceous limestone.

Boring B-45, drilled at the southern end of the CRBRP site peninsula, encountered the Chickamauga Group directly below Clinch River alluvium. This demonstrates that the Copper Creek Fault does not cross the site peninsula (See Figure 2.5-17).

The Joy Test Well drilled near Lagoon Road penetrated the Copper Creek Fault at a depth of about 1370 feet (Ref. 13). The well began in the Conasauga, fully penetrated the Rome and Chickamauga, and terminated in the Knox Group.

White Oak Mountain Fault

In the Oak Ridge area, the White Oak Mountain Fault is a complexly branching thrust fault along which the lower shales of the Rome Formation have been thrust over middle Cambrian and younger rocks (Ref. 54).

The shale beds of the lower Rome are vertical or dip steeply to the southeast or northwest. The excessive thickness of this formation in parts of the area suggests that a considerable amount of faulting has occurred within the shale (Ref. 42, 70). At Location 6, Figure 2.5-17, a branch of the White Oak Mountain Fault was observed thrusting the Knox Formation over the Chickamauga Group. The Knox hanging wall was folded into a tight syncline and thrust over the Chickamauga.

The total throw of the White Oak Mountain fault system in the vicinity of Clinton, Tennessee is 8200 feet (Ref. 70).

Period of Faulting

The faulting and folding demonstrated within the Valley and Ridge Province is believed to have occurred during the Allegheny Orogenic movement (Permian-Pennsylvania, 230-280 million years). There is some evidence of earlier orogenic periods affecting the Valley and Ridge, but there is no evidence of tectonic activity in this area since the late Paleozoic or about 230 million years ago (Refs. 79, 82, 100, 101).

Commonly, the traces of the White Oak Mountain and Copper Creek faults are covered with residual soil, colluvium or some combination of the two, and nowhere has this material been obviously displaced. It is difficult to arrive at an age for these materials, but TVA geologists believe that they may be of Paleocene or even later Upper Cretaceous Age (Ref. 101). Thus there is no indicated movement along these fault traces during the last 60 million years.

In two recently exposed interstate highway roadcuts, samples of faulted material have been collected and radiometrically dated (see Table 2.5-1). The results of these age determinations (280-290 ± 10 million years) indicate that the last movement of these faults occurred during the late Paleozoic, as virtually all of the literature indicates.

In conclusion, the geologic literature pertairing to Appalachian Valley and Ridge faulting have been examined and found to be in unanimous agreement on a Paleozoic age for these faults. Field work and radiometric age determinations give no reason to suspect any post-Paleozoic activity associated with the Copper Creek or White Oak Mountain Faults.

2.5.3.9 Design Basis for Surface Faulting

As indicated in Section 2.5.3.2, a design basis for surface faulting does not need to be considered at this site.

2.5.4 Stability of Subsurface Materials

The locations of the Category I Nuclear Island and Category I Cooling Towers are shown on Figure 2.5-24. Generalized profiles of typical subsurface materials for these structures are shown on Figures 2.5-25, 2.5-26, 2.5-27 and 2.5-28. The Unit A Upper Siltstone, the Unit A Limestone and the Unit A Lower Siltstone underlie the Category I structures shown in these profiles. The Steam Generator Maintenance Bay, also Category I, will be located on the Unit B Limestone.

Category I structures in the Nuclear Island area include the following buildings: Reactor Containment, Reactor Service, Control, Diesel Generator and Steam Generator. These structures are to be supported primarily by a common mat founded on sound rock in the Unit A Upper Siltstone. One portion of the Steam Generator Maintenance Bay will be founded on a mat on competent rock in the Unit B Limestone and another portion (craneway area) will be founded on piers, also in the Unit B Limestone. The common structural mat will have maximum plan dimensions of about 475 x 360 feet and will be founded at elevation 715. The mat will impose a static bearing pressure of approximately 8.9 KSF.

In selecting the proposed bearing elevation for the structures, a number of factors were taken into consideration and these may be enumerated as follows:

(a) Final grade was established at Elevation 815' based on providing an adequate freeboard allowance for the potential maximum flood elevation with wind waves and runup of 809' resulting from the hypothetical failure of Norris Dam, located approximately 62 miles upstream of the CRBRP site and coincident with 1/2 PMF.

(b) In reviewing the operational and maintenance characteristics of the plant, it was concluded that there was a significant economic advantage to establishing the operating floor at grade. Conforming with this criterion resulted in a required minimum depth of embedment of approximately 100' for all Category I structures located on the common mat (18' thick).

(c) By selecting a bearing elevation of 715 for the common mat, a conservative margin of safety is inherently established below the Top of Continuous Rock to account for minor variations or discontinuities which may be
59 exposed as the excavation is opened up below that level.

The Category I Cooling Tower will also be supported by a single mat founded on sound rock in the Unit A Upper Siltstone. The foundation mat will have dimensions of about 140 feet x 140 feet, will be founded at Elevation 765 and will exert a static bearing pressure of 4.0 KSF. Two Category I Fuel Oil Storage Tanks will be supported
59 directly by Class A structural backfill overlying the Unit A Upper Siltstone

at the locations shown on Figure 10.2-2. The tanks will be anchored to a common reinforced concrete mat measuring 70 feet x 42 feet with its base at Elevation 793 feet.

The Radwaste Area of the RSB and the Turbine Generator Building are both non-Category I structures which are located immediately adjacent to Category I structures as shown in Figure 2.4-3. The upper part of the Radwaste Area, which is a steel framed structure, and the Turbine Generator Building, are designed to ensure that the adjacent Category I structures are not damaged nor their safety functions compromised during an SSE. The Radwaste Area will be supported on a mat measuring 84 feet x 113 feet at Elevation 770, and will exert a static bearing pressure of 3.8 KSF on competent silstone and limestone. The Turbine Generator Building will be supported on a mat measuring approximately 198 feet x 166 feet and having the configuration shown in Figure 2.5-39. The mat will exert an average
59 static bearing pressure of 2.5 KSF on compacted Class 'A' backfill.

Initially, 65 continuous core borings were made along a 200 foot
N-S grid to define general site subsurface conditions. Based on the
55 findings of these borings, the location of the Nuclear Island was set. A
new grid system, rotated 45° from the initial grid system and correspond-
ing to the axes of the plant, was established and borings were made to
55 complete a 140 foot grid spacing at the Nuclear Island area. Additional
borings were then made at the centers of the 140 foot grid blocks to
provide a minimum distance of 100 feet between borings. Also, additional
specific investigation of the Unit A Limestone - Upper Siltstone contact
55 was made for the Nuclear Island area by two lines of borings along the
northwest side of the structures. These two lines of borings were spaced
50 feet apart and consisted of borings with spacings of 50 feet and 70
38 feet along the lines. In addition to borings made for the Category I
Nuclear Island, 5 borings have been made for the Category I Cooling Tower
55 to provide data to assist in selection of the most suitable location and
level for the structure. As a result of classifying the Steam Generator
Maintenance Bay as a Category I structure, an additional site program
consisting of 4 core borings and 12 air-track borings was performed in
that area. Results of the borings indicated that the Unit B Limestone
below Elevation 780 is competent rock, free from voids and solution
activity, and consequently suitable for support of the structure. Locations
59 of borings are shown on Figures 2.5-4 and 2.5-24.

Eleven core borings were surveyed by continuous velocity logging techniques and fifteen additional borings were made to measure seismic compression and shear waves by the cross-hole and up-hole methods. In-situ Goodman Jack tests were performed in four core borings at the site. Locations of these foundation studies are shown on Figure 2.5-29 and the results are discussed in Section 2.5.4.2.

2.5.4.1 Geologic Features

Below the foundation level of the Nuclear Island structures, the borings encountered hard, sound and continuous siltstones and limestones. Weak or unstable zones have not been encountered. There is no evidence of significant zones of solutioned, cavernous or highly weathered areas which could produce subsidence. The Unit A Limestone, which has developed the most weathering and solutioning of the foundation rocks, is at its closest proximity to the foundations and has its thinnest siltstone cover along the northwest side of the structures. However, as discussed in Section 2.5.1.2.4.4, significant weathering and solutioning has not occurred in the Unit A Limestone where it dips beneath and is covered by unweathered Unit A Upper Siltstone. This observation was confirmed by the borings made along the two lines along the northwest side of the Nuclear Island structures.

As discussed in Section 2.5.1.1.4.3 and 2.4.13.2.1 no mining or extensive withdrawal, either of which might allow subsidence, occurs or is expected to occur in the area. In Section 2.5.1.1.4.2, it is concluded that the injection well, located about 4 miles from the site, will not affect the site.

38 | 59 | Loads imposed by the Nuclear Island structures (approximately 8.9 KSF-static) would not be of sufficient magnitude to develop compaction subsidence in the hard, foundation rocks. In most instances, foundation rocks have been subjected to overburden stresses much greater than the anticipated foundation loads.

The TVA has documented their experience over the past 40 years with major construction projects in the Valley and Ridge in the Bellefonte Nuclear Plant PSAR. They state that no residual stress effects have been observed in the area. Therefore, no specific investigations of unrelieved residual stress in the bedrock have been made.

As discussed in Section 2.5.1.2.4.3, small folds and minor dislocations are common in the region and are present at the site. However, the minor structures observed represent ancient adjustments. Results of laboratory and in-situ tests indicate that the rocks which occur within such zones are similar in character and competency to other sound rocks at the site.

2.5.4.2 Properties of Underlying Materials

The investigative program to determine static and dynamic engineering properties of the foundation rocks at the CRBRP site consisted of both laboratory and in-situ testing. Static determination of the foundation rock's engineering properties were based on the results of laboratory unconfined compression and Poisson's ratio tests and in-situ Goodman Jack tests. Geophysically measured engineering properties have been determined by evaluating results of continuous velocity logging, and cross-hole and up-hole seismic velocity measurements. The results of these evaluations are presented in the following subsections.

2.5.4.2.1 Statically Determined Properties

Laboratory tests were performed on rock core samples obtained from core borings made at the site. The laboratory tests were performed on rock core samples of the Unit A Upper Siltstone, the Unit A Limestone and the Unit A Lower Siltstone. The tests included samples from both above and below the Top of Continuous Rock. The laboratory testing program consisted of unconfined compression tests, Poisson's ratio, unit weight and moisture content determinations. Test procedures are described in the Laboratory Data Appendix. Ranges and averages of laboratory test results are listed on Table 2.5-8. These laboratory test results are representative of the intact rock.

Goodman Jack tests to determine the Modulus of Elasticity of the in-situ rocks were performed in borings number 69, 74, 79 and 94 as shown on Figure 2.5-29. The tests were conducted at depths ranging from 15 feet to 150 feet below ground surface and included materials from both above and below the Top of Continuous Rock. The borings for Goodman Jack tests were selected to provide maximum vertical coverage of the Unit A Upper Siltstone and the Unit A Limestone. The tests were performed in several orientations, with plate loading directions corresponding to strike, dip and intermediate orientations. Goodman Jack test procedures are described in the Field Data Appendix. Ranges and averages of test results are listed on Table 2.5-8. Figures 2.5-30 and 2.5-31 show the variation of test results with depth for the Unit A Upper Siltstone and Unit A Limestone, respectively, and indicate that the modulus of the foundation rocks is generally 1×10^6 psi or more. These field test results are considered representative of the intact rock since the stressed zone at the test location is relatively small.

Results of the laboratory and in-situ tests performed on intact rock indicate that the foundation rocks at the CRBRP site are capable of supporting any of the planned static loads. However, geological discontinuities within the zones stressed by applied structural loads are significant factors which determine to a great extent the properties of the rock mass. As the size of the loaded area and the magnitude of applied load increases, the stressed zone becomes larger and more discontinuities are affected. The rock mass properties are influenced by the rock mass quality as well as the properties of the intact rock. The extent to which the modulus of the rock mass is reduced from the modulus of the intact rock, due to the rock mass quality, can be expressed by a reduction factor that is determined by evaluation of the rock quality designation (RQD) as described in Reference (134).

RQD Determinations made at or below foundation level in the borings within or near the Plant Island have been evaluated. Two hundred eighty eight (288) RQD values, representative of nearly 2700 feet of rock cored below foundation level, are shown by Figure 2.5-32. As indicated by this figure, the quality of the rock mass is very good, with only about 10% of the observed RQD determinations being less than 0.9. Based on a "design" RQD value of 0.9, a reduction factor of 0.5 was used to determine the modulus of the rock mass, as shown by Figure 2.5-33. On this basis, the modulus of the rock mass would be about 5 X 10^5 psi. The Shear Modulus of Elasticity can be calculated from the equation:

$$G = \frac{E}{2\,(1 + \nu)}$$

where G = Shear Modulus of Elasticity
E = Young's Modulus of Elasticity
ν = Poisson's Ratio

12 Utilizing a Poisson's Ratio of 0.3, the Shear Modulus is calculated to be 2.0 X 10^5 psi. The properties of the foundation rocks which may be utilized for static analysis are:

Unit Weight	165 pcf
Poisson's Ratio	0.3
Shear Modulus	2 X 10^5 psi
Young's Modulus	5 X 10^5 psi

2.5.4.2.2 Seismically Determined Engineering Properties

The engineering properties of the rocks have been determined by evaluating compression and shear wave velocity measurements. Three test methods were used in order to evaluate the range of velocities. The methods were:

(1) continuous velocity logging
(2) seismic up-hole surveys
(3) seismic cross-hole surveys.

Continuous velocity logging was performed in eleven borings as shown on Figure 2.5-29. The field work was performed by Birdwell Division of Seismograph Services Corporation using their standard 3-D velocity, density and caliper logging tools to make required field measurements. The borings for continuous velocity logging were selected to include the entire thickness of both the Unit A Upper Siltstone and Unit A Limestone. A significant portion of the Unit A Lower Siltstone, as well as minor portions of the Unit B Limestone and Knox Formation, were also logged. Descriptions of the continuous velocity logging procedures are included in the Field Data Appendix. Ranges of velocity measurements obtained are shown on Table 2.5-9. These velocity measurements provide information for 3 foot sections of the rock mass in the vertical direction under in-situ conditions.

12 Seismic up-hole surveys were performed in boring 33-C, as shown on Figure 2.5-29. The work was performed by Weston Geophysical Engineers. The field procedures and results of the velocity measurements are described in Reference (132). These measured velocities provide information for the rock mass in the vertical direction under in-situ conditions.

12 Seismic cross-hole surveys were performed in an array of fifteen borings, as shown on Figure 2.5-29. The work was performed by Weston Geophysical Engineers. The field procedures and results of the velocity measurements are also described in the "Seismic Velocity and Elastic Moduli Measurements" (Ref. 132). These measured values provided information for the rock mass along bedding, under in-situ conditions.

Comparisons of the velocity ranges measured by continuous logging techniques (shown on Table 2.5-9) with velocity ranges measured
12 by the up-hole method have been made. These two test methods, that provide information in the vertical direction under in-situ conditions, indicate good agreement, as shown by Figures 2.5-34 and 2.5-35.

The continuous velocity logging techniques resulted in a mass of data which was statistically analyzed. Results of the analyses of data from borings that penetrated a major portion of the geologic unit for the Unit A Upper Siltstone, the Unit A Limestone, and the Unit A Lower Siltstone are tabulated on Tables 2.5-10, 2.5-11, 2.5-12 and 2.5-13 respectively. Table 2.5-14 lists the overall average, standard deviation and coefficient of variation obtained by combining velocity measurements for the same rock unit made in different borings over the site. As indicated by these tables, most velocity measurements across the site do not show significant variance for each rock unit.

An apparent velocity anisotropy for the Unit A Upper Siltstone has been indicated by the Weston cross-hole survey. Comparisons of average velocities determined by the continuous velocity and up-hole methods with those determined by cross-hole methods were made to evaluate this possible phenomenon. As shown by Figures 2.5-36 and 2.5-37, these average up-hole and continuous logging velocities, representative of the in-situ rock in the vertical direction, fall near the middle of the ranges of the cross-hole velocities, which were made to represent the in-situ rock along the bedding. Some anisotropy may be present. However, an evaluation of available data indicates that the apparent velocity anisotropy is influenced by refraction along hard layers which is common in interlayered sedimentary rocks.

The two methods of geophysical measurements - continuous velocity logging (CVL) and seismic refraction - resulted in computed values of Poisson's ratio which are slightly different as shown by the following comparison:

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GEOLOGIC UNIT	CVL	POISSON'S RATIO RANGE UP-HOLE SEISMIC	CROSS-HOLE SEISMIC
Unit A Upper Siltstone	.39-.23	.43-.23	.42-.32
Unit A Limestone	.36-.22	.40-.19	.40-.32

The majority of Poisson's ratios determined by CVL techniques are on the order of 0.28 to 0.32. Poisson's ratios determined by seismic methods averaged from 0.33 to 0.39. These values are reasonably close to the values obtained by other techniques. The Poisson's ratio values determined by the continuous velocity logging method best agree with the values obtained by laboratory techniques and with values appropriate to hard sedimentary rocks.

In summary, the three methods of in-situ measurements of the foundation rocks seismic velocities result in similar values. The moduli of the foundation rocks may be calculated from the measured seismic velocities.

Shear wave velocities were measured by each of the in-situ dynamic test methods. Shear wave velocities in the siltstone were found to range from 3900 to 8800 fps. The minimum velocity, which was determined by up-hole techniques, was located in the drainage swale which presently occupies portions of the Nuclear Island area. This determination is indicative of the upper weathered portion of the Unit A Upper Siltstone and is not representative of the sound foundation rocks which will support the Nuclear Island structures. The minimum shear wave velocity determined by up-hole seismic methods which is representative of the foundation rocks is 5580 fps. The range of shear wave velocities which is representative of the sound siltstone is from 5580-6820. A conservative value for the shear wave velocity within the foundation rocks of 6200 fps was selected for determination of the Shear Modulus of Elasticity. With a bulk density of 165 pcf and a shear wave velocity of 6200 fps, the
59 Shear Modulus of Elasticity is calculated to be 1.37 X 10^6 psi. Young's
12 Modulus of Elasticity can be calculated from the equation:

$$E_{seis} = 2G_{seis}(1 + \nu)$$

where

E_{seis} = Young's Modulus of Elasticity

G_{seis} = Shear Modulus of Elasticity

ν = Poisson's Ratio

38 Utilizing a Poisson's Ratio of 0.3, Young's Modulus is calculated to be
12 3.55 X 10^6.

In evaluating in-situ moduli calculated from seismic velocities the significance of geologic discontinuities and scale effects must be considered. Since the seismic waves can travel along preferred paths and directions the measured wave arrivals are likely to be representative of the more intact rock which, in the case of cross-hole measurements, would tend to mask out softer zones or geological discontinuities. The dynamic moduli calculated from seismic velocities are higher than the in-situ moduli appropriate for real structures because the seismic pulse is of very short duration and, more significantly, it is a very low stress-level pulse. Whereas, earthquake loads will last for several seconds and produce significantly higher stresses within the rock mass than the microseisms associated with the geophysical methods. Therefore, to account for the geological discontinuities and the low stress-level measurements, a reduction of moduli calculated from seismic wave arrivals is warranted.

Deere, Hendron, Patton and Cording (Reference 134) have compared moduli calculated from dynamic test methods with moduli determined by large scale in-situ tests of rock masses. They indicate that the in-situ modulus is usually less than that calculated by seismic methods. The relationship between the modulus of the rock mass and the modulus determined by seismic velocities can be expressed by a reduction factor that is determined by evaluation of the rock quality designation (RQD),
59 as shown by the solid line on Figure 2.5-38.

The scatter of data presented in Figure 2.5-38 is relatively large and a reduction factor ranging from about 0.15 to 1.0 is possible. Based on a "design" RQD of 0.9 as previously determined, a reduction factor of 0.4 was conservatively selected as representative of the foundation rocks. Using this reduction factor and the moduli previously calculated, properties of the foundation rocks are:

Unit Weight	165 PCF
Poisson's Ratio	0.3
Shear Modulus	5.45×10^5 psi
Young's Modulus	1.42×10^6 psi

For comparative purposes an alternate approach was selected to determine a representative value for the rock modulus which could be used in the seismic analysis of the structures.

Figure 2.5-38 shows the reduction factor (E_r/E_{seis}) as a function of rock quality (RQD). The reduction factor data in the figure is also applicable as a function of the square of velocity ratios $(V_F/V_L)^2$. Dr. A. Hendron has indicated that most of the data contained in this figure is based on velocity data. Therefore, the extent to which the modulus obtained by seismic methods should be reduced to account for geologic discontinuities, stress levels, etc. can also be determined by comparison of measured velocities.

An estimate of the laboratory velocities can be made based on the continuous velocity logging (CVL) measurements. The CVL measurements, which are representative of one foot segments of the foundation rocks, would be expected to provide velocities representative of intact rock (where no dis-
12 continuities are present within a one foot segment) that would be similar

to laboratory determined values. The CVL measurements would be expected to
provide unrealistically low velocities where an unusually high precentage
55 of discontinuities are present within a one foot segment. Below the primary
38 bearing elevation for Nuclear Island structures (El. 715) the number of discon-
tinuities is minimal and consequently the maximum velocities obtained by CVL
techniques are considered representative of the probable upper limit of the
average V_L and the average CVL velocities are considered as somewhat less
than the average of V_L. Therefore, the average V_L is considered to be between
the ranges shown in Table 2.5-15.

Comparison of the estimated range of V_L with field seismic velocity values (V_F) which were measured in the same direction (i.e. up-hole velocities at Boring 33-C) can be used as a basis for determining the extent to which E_{seis} should be reduced. Table 2.5-15 lists the range and average field seismic velocities reported for up-hole measurement in Boring 33-C at the CRBRP site. Table 2.5-16 lists the ratios obtained by comparing the possible combinations of V_F and V_L. As shown in Table 2.5-16, the Unit A Upper Siltstone is computed to have a representative velocity ratio of 0.8. This ratio can be used in conjunction with Figure 2.5-38 as a basis for reducing moduli calculated from field seismic velocities.

Results of in-situ rock mass moduli calculated based on transient
(dynamic) loadings generated by large nuclear blasts are considered to be
representative of the range of moduli appropriate for the seismic analysis
of structures. Dynamic test results from three sites have been included
on Figure 2.5-38. These values have been obtained from rock masses
loaded by large explosions. Reduction factors have been weighted towards the
three data points resulting from these dynamic measurements as indicated by the
59 dashed line. The Upper Siltstone velocity ratio of 0.8 results in a modu-
lus reduction factor of 0.44. Utilizing an average shear wave velocity (V_S)
for the stratum of 6200 fps and a Poisson's Ratio of 0.3, the Shear Modulus
"G" is calculated to be 6.0×10^5 psi and the corresponding Young's Modulus
"E" is 1.56×10^6 psi.

2.5.4.2.3 Recommended Design Properties

The dynamic Modulus of Elasticity for the Upper Siltstone stratum has been computed by two approaches resulting in an average value of "E" approximately equal to 1.5×10^6 psi. The corresponding Shear Modulus "G" is 5.8×10^5 psi. These values represent the best estimate of the modulus of the in-situ rock below the foundation level considering in-situ tests and seismic techniques. To account for minor variations and inconsistencies in the data it is considered appropriate to complete the seismic analysis of the structures utilizing a range of $\pm$ 25% in the average computed modulus.

59 The following values are recommended for the Upper Siltstone for seismic
design:

Shear Modulus (G) = 5.8×10^5 psi $\pm$ 25%

Young's Modulus (E) = 1.5×10^6 psi $\pm$ 25%

12

The following values are recommended for the Unit B Limestone for seismic design:

59 Shear Modulus (G) = 1.2×10^6 psi $\pm$ 25%

Shear Modulus (E) = 3.0×10^6 psi $\pm$ 25%

12 2.5.4.3 Plot Plan

55 The locations of all borings that have been made at the site are shown on Figure 2.5-4, with respect to the site grid coordinates. The locations of borings that have been made near Category 1 structures are again shown on Figure 2.5-24. The locations of borings made or used for special foundation studies are shown on Figure 2.5-29. The locations of the Nuclear Island and Cooling Tower structures are superimposed on these figures. Figures 2.5-25, 2.5-26, 2.5-27 and 2.5-28 are cross
59 sections through of the subsurface materials.

2.5.4.4 Soil and Rock Characteristics

Table 2.5-8 summarizes representative static engineering properties of the foundation rocks at the site. Dynamic properties are discussed in Section 2.5.4.2.2. Logs that graphically describe the foundation rocks are included in the Field Data Appendix.

2.5.4.5 Excavations and Backfill

2.5.4.5.1 Nuclear Island

2.5.4.5.1.1 General

Site preparation will consist of clearing, stripping, excavating, dewatering and backfilling operations.

2.5.4.5.1.2 Clearing and Stripping

Trees and brush will be cleared, and grass, roots and other deleterious materials will be stripped from all areas being excavated or filled. Usable timber will be removed from the site; all unusable material will be wasted at specified on-site locations in conformance with state and local code requirements.

2.5.4.5.1.3 Excavation

Excavation in soil and rock will be required to achieve the final planned foundation grade for the plant structures. The proposed base level of excavation will be 712.5 to permit pouring of a common
55 foundation mat with a bottom elevation of 715 for most Category I structures included in the Nuclear Island, resulting in a top of mat elevation of 733. The maximum depth of the excavation will be approximately 100 feet below the final grade elevation 815. The minimum depth of excavation will be approximately 50 feet, corresponding with the
38 topographic low elevation at the bottom of the valley trending in a northeasterly direction across the site. The eastside of the Nuclear Island excavation will be designed to accommodate the Category I Steam Generator Maintenance Bay, which will be founded on competent rock of the Unit B Limestone formation on a mat at elevation 760 and on piers extending
55 to spread footings bearing at elevation 780.

Portions of the excavation extending through the residual overburden will be cut on slopes no steeper than 2 horizontal to 1 vertical.

59 Generally, excavations in partially weathered and sound rock will be essentially vertical on north, south and east faces (Plant North as reference). Rockbolts will be installed in the exposed rock faces as excavation proceeds to ensure stability of the slopes with an adequate factor of safety. Along the west side, it is planned to excavate both the weathered and sound rock above Elevation 734 along the bedding plane (approximately 2:1 horizontal to vertical) and install rock bolts between Elevation 734 and Elevation 755. Also, on the west side, between Elevation 734 and the base of the excavation at Elevation 712.5, it is planned to excavate the rock vertically and install rock bolts.
38 However, a 50' wide test section will first be excavated vertically within this elevation range to permit evaluation of susceptibility of slope to undergo a wedge type failure. Depending on the results of the evacuation, the excavation within this elevation range will proceed either vertically or along the bedding plane. Dewatering of the
59 excavation is discussed in Section 2.5.4.5.1.4.

Due to the anticipated variable hardness characteristics of the upper siltstone stratum, it is planned to conduct a test blasting program at the time of construction under supervision of a qualified engineer. This program will determine the optimum blasting arrangement and procedure which will result in minimum damage to the foundation rock and permit removal generally in 15 foot depth increments (maximum). The
38 perimeter walls of each excavation level will be blasted using pre-split blasting procedures. Berms will be provided at a predetermined vertical interval. It is expected that ripping may only be feasible for the highly weathered section. 25

Consideration will be given to removing the final 18 inches of rock by controlled means, e.g., air hammers, however, it is probable
38 that if the final excavation lift is limited to 7 feet and careful control is exercised in blasting, the foundation grade will not be unduly disturbed or cracked from the blasting effect. This assumption will be checked in the field prior to deciding on the method for removal of the final layer of rock above foundation grade. 25

38 Unbolted side slopes in siltstone and the base of the excavation will be protected from deterioration and weathering caused by frost, ponding of water and construction activity by a layer of gunite prior to construction of the mat foundation.

An extensive inspection verification program will be established and implemented during construction, and will consist essentially of the following:

a. A qualified and experienced geologist will be on site immediately prior to the start of excavation and will monitor progress of the work until the base of the excavation has been prepared for the initial mat pour. He will report directly to the engineering and design organization and will be charged with the responsibility in the field of reviewing and commenting on the adequacy of the construction procedures proposed by the excavating contractor for ripping, blasting and removal of rock, inspecting exposed rock strata including side slopes and base of excavation and preparing a detailed geological map of the area.

b. A progress report will be submitted to the engineering and design organization on a weekly basis including photographs and detailed mapping of any significant geological features.

c. A consulting geotechnical review group consisting of specialists in rock mechanics and geology will inspect the excavation and report to the engineering and design organization on their findings at regular intervals, not exceeding one month.

d. If a geological discontinuity is noted, the engineering and design organization will be notified immediately and an inspection will be made by qualified personnel including members of the review board if considered necessary.

e. The site geologist will be assisted in his inspections if required by readily available air track drills which may be utilized for investigation purposes including geophysical logging of the holes and analysis of the data, and use of back-hoe or bulldozer with ripping capability. Zones of potentially high permeability may be checked by drilling from accessible berms constructed around the perimeter of
12 the excavation.

f. The representivity of Boring 55, selected as the central boring in a test grouting program on the west side of the Nuclear Island, will be checked after excavation to confirm the homogeneity and satisfactory bearing capability of the foundation strata. This will be done by completing a series of airtrack holes supplemented by geophysical logging and additional core borings as required, extending through the Unit A Limestone. It is the consensus of opinion among the geotechnical consultants engaged in this project that the satisfactory bearing characteristics of the foundation strata
59 will be confirmed. All borings will be gravity grouted on completion of the program.

g. Formal approval of the prepared base of the excavation will be re-
12 quired by the review board prior to proceeding with the pouring of the mat.

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h. An instrumentation program will be implemented which will permit monitoring fluctuations in the ground water table, stability of slopes during construction and possible heave or rebound in the foundation stratum. Data will be analyzed and reported to the engineering and design organization at weekly intervals.

i. The Nuclear Regulatory Commission (NRC) will be notified
12 if a geological discontinuity is noted.

Emphasis will be placed on exploratory work along the western boundary of the plant after excavation and no additional checking is considered necessary over the remaining area of the Nuclear Plant Island foundation.

59 If localized weathering zones are observed by the field site geologist at the proposed excavation level for the base of the foundation, additional excavation will be conducted to expose sound siltstone. All depressions at the base excavation level will be filled with mass concrete to reach the desired final bearing elevation. Since the Nuclear Island common
38 | 55 mat has a maximum width of approximately 360' and is located on a 400' wide band of siltstone, the establishment of a common mat foundation concept results in a bearing elevation being selected for the Nuclear Island which is below the maximum recorded depth of weathering for the siltstone stratum. 25

2.5.4.5.1.4 Dewatering

The groundwater level at the time the boring investigation was being conducted at the site was approximately at the Top of Continuous Rock. It will be necessary to dewater the excavation for the plant area. Dewatering will be accomplished by installing ditches in berms located in the overburden soil and at the top of rock which will intercept rain, runoff and seepage. The water will be directed to sumps located at an optimum spacing along the ditch and subsequently pumped to impoundment ponds located beyond the immediate influence of the excavation and/or directed to the Clinch River, depending on percentage of solids (soil particles) in the effluent. Inclined drains (slotted PVC pipe) will be installed in the rock faces as excavation proceeds. Drainage from the rock faces will be directed to sumps excavated
59 below the base excavation elevation of 712.5. A clean sand and gravel drainage layer, ranging from 15" to approximately 21" in thickness will be installed at the base of the excavation to dissipate any pore pressures which may develop below the foundation during construction. The base of the drainage layer will be sloped to a trench located around the periphery of the excavation, permitting the water to be directed to sumps for
38 removal from the excavation.

Material to be used for the drainage layer beneath the mat
will be placed in two layers. Each layer will be compacted to a minimum
85% relative density as determined by ASTM D 2049. An extensive monitoring
system of wells and piezometers will be located around the excavation
to detect and record fluctuations in the water table before, during
and after the excavation has been completed, as outlined in Section
38 2.4.13.4.

The exposed side slopes on the north and south ends of the
excavation (Plant North) will be primarily in the Unit A Upper Siltstone
stratum which is relatively impervious (results of water pressure tests
are given in Appendix 2A and Table 2.4-17). It is concluded that similar
59 conditions will be encountered in the exposed siltstone in the west
side slope which is planned to be excavated along the bedding plane (approx-
38 imately 2:1 horizontal to vertical) above elevation 734. Where the thickness
of siltstone immediately overlying the Unit A limestone is thin or where no
55 siltstone cover is present, it is anticipated that some seepage will occur due
to the probable occurrence of joints which may provide a seepage path from
38 55 the adjacent weathered Unit A limestone. Slotted PVC pipe drains will be
provided as described above. The major quantity of seepage is expected
where the weathered limestone is directly exposed on the east slope and
partially on the north and south ends.

If significant seepage is encountered at a specific elevation
or location during the course of the excavation which may inhibit the
ripping or blasting operations, it is planned to treat these areas by
38 grouting from berms or level areas at higher elevations. Results
of the site investigation program have indicated that the continuity
of the solution activity encountered in the borings located in the Unit A
and Unit B limestone is not extensive and consequently, the grouting
treatment, if required, is expected to be localized and should effectively
reduce flow sufficiently to permit excavation to continue and permit
control by sumping and pumping.

It is estimated that the normal flow into the excavation will be in the order of 500-1000 gallons/min based on an assumed conservative selection of the coefficient of permeability "k" for the rock matrix of 1×10^{-3} cms/sec. An upper limit coincident with the heavy rain and
run-off is not expected to exceed 3000 gpm. 25

Adequate pumping capability will be provided to ensure that con-
struction at the base of the excavation is conducted in a reasonably dry
condition at all times. The groundwater will be permitted to rise only under
controlled conditions, utilizing pumping techniques which will establish
a phreatic surface outside and below the excavation acceptable to the
55 engineer. The dewatering system will be designed such that a suitable

factor of safety against uplift exists at all stages of construction,
and that it is capable of lowering the groundwater table below bottom
59 of lean concrete and Class 'A' fill behind Nuclear Island structural walls.
Dewatering operations will be compatible with the sequence of placement
of concrete and Class 'A' fill behind Nuclear Island structural walls.
The contractor will be required to provide standby equipment, available
for immediate operation, as may be required to maintain the continuous
operation of the dewatering system in the event that all or any part
of the system becomes inadequate or fails. No dewatering will be
permitted to start in any area until contractor's plans have been
approved by the engineer. Dewatering operations will not be discon-
55 tinued in part or in full without approval of the engineer.

2.5.4.5.1.5 Structural Fill and Backfill

55 Most Nuclear Island structures will be located on a common mat
bearing directly on highly competent siltstone of satisfactory bearing capability.
The Steam Generator Maintenance Bay will be supported on competent rock of
59 the Unit B Limestone. It will be necessary to backfill around the structures
within the immediate horizontal influence of the mat with lean concrete
fill. The lean concrete fill will also extend up to the top of the
(nominally) vertical cuts as shown on Figure 2.5-39. Class A structural
fill will be placed in thin, near horizontal lifts, above the lean concrete
fill and compacted around the structures, extending to the design grade
(Elevation 815). The material will be compacted to minimum densities of
95% Modified Proctor and/or 85% Relative Density. The Category I Fuel
Oil Storage Tanks will be supported on Class A fill, compacted to a
density of at least 95% Modified Proctor and/or 85% Relative Density,
after removal of all weathered siltstone from within the bearing influence
59 of the tanks. Related non-Category I structures located in the immediate
vicinity of the Nuclear Island, which will include the Radwaste Area of
55 the RSB, Plant Service Building, Turbine Generator Building, and Maintenance Shop/Warehouse , will be supported on compacted Class A structural fill
or rock. Residual overburden and highly weathered rock will be excavated prior
38 to placement and compaction of supporting fill.

Class A structural fill will consist of a free draining, well-graded, granular material. Investigation of on-site residual soil indicates that this material is not suitable as Class A fill due to its variable engineering properties, including percentage fines, moisture content and degree of compressibility related to support of adjacent non-Category I structures. It is planned to obtain Class A fill by crushing limestone of the Knox formation obtained from an on-site quarry developed beyond planned excavation limits. A comprehensive field exploration and laboratory program will be performed to evaluate such design considerations as engineering properties of quarried rock, quantity of material available, impact on solutioning potential of the Knox formation, slope stability and environmental impact. Additional material is also available from off-site sources. Appendix 2-D contains a report which presents the results of investigating a number of off-site limestone quarries to determine the suitability and availability of material for Class A fill. It was concluded that, based on comprehensive laboratory testing, suitable material is available in desired quantity. A test fill program will be conducted on potential Class A fill material to determine the
59 compaction characteristics required to attain minimum specified densities.

The quality control of the backfilling operations will be under the direction of an experienced soils engineer. He will have the authority to alter the compaction procedure as required to ensure that the desired uniformity and quality as specified in the design are achieved. 27

Field control test methods will be checked to establish suitability and testing frequency for compacted fill. Various testing devices such as the Washington densometer and/or sand cone apparatus
55 will be checked for consistency of test results to obtain the desired field control during the placement and compaction sequence.
38

As indicated in NRC question 323.38, dynamic testing would not be required if a commitment is made to compact to at least 95% Modified Proctor
59 and/or 85% Relative Density. As noted above, this commitment is being made. 38

Assumptions have been made with respect to the dynamic characteristics of the proposed Class A structural backfill. Data has been obtained from in situ and laboratory tests on compacted granular fill used at other nuclear power plant sites. Average values for the Shear Modulus, Shear Strain and related Damping Coefficient are indicated below:

Shear Modulus	Shear Strain	Damping
G (psi)	γ (in/in)	%
22,600	5×10^{-6}	5.6
20,850	1×10^{-5}	5.6
20,000	2×10^{-5}	5.6
19,650	5×10^{-5}	5.6
18,000	1×10^{-4}	5.6
15,000	2×10^{-4}	5.7
11,200	5×10^{-4}	6.7
8,000	1×10^{-3}	8.0
5,000	4×10^{-3}	14.8
3,500	1×10^{-2}	17.2

An evaluation of the liquefaction potential of granular fill and/or crushed rock compacted to densities of at least 95% Modified Proctor and/or 85% Relative Density is outlined below. An evaluation of dynamic lateral loads is made in Section 2.5.4.5.1.6.

It is apparent from the laboratory studies conducted on representative samples of crushed rock obtained from off-site quarries
59 (Appendix 2-D) which is a potential source of the Class A fill, that the 25
95% Modified Proctor values as determined by ASTM D-1557-70 (Method D)

generally resulted in densities slightly greater than the 85% value
obtained by Relative Density tests conducted in conformance with ASTM
D-2049. In reviewing the extensive compilation of data which has
46 been published to date on the question of liquefaction, and in recent
discussions with Professor H. B. Seed of the University of California,
who has been a major influence in arranging for the conducting of the
various research programs on this subject, and other geotechnical con-
sultants, the consensus of opinion is that samples of well-graded
granular material similar to Figure #1 of Appendix 2.D will not liquefy
or be susceptible to significant strain amplitude when compacted to the
46 above densities and subjected to an equivalent acceleration of 0.25g
and vibratory motion in the range of 10-100 cycles.

38 Although most of the research conducted to date has been on
sands, recent work by Wong, Seed and Chan (Reference 142) on gravelly
soils with a gradation similar to that proposed for the CRBRP site,
indicated that coarse-grained material generally tended to have a much
lower susceptibilty to earthquake induced liquefaction as compared to
38 fine to medium sands. This was considered to be due primarily to their
capacity to dissipate pore pressure changes during the period of earthquake
shaking.

It has been generally agreed that the number of cycles required
38 to liquefy sands compacted to approximately 85% Relative Density is
considerably in excess of any postulated seismic event. It is concluded,
therefore, that the proposed Class A structural backfill consisting of
a well graded crushed rock, when compacted to at least 95% Modified Proctor
59 density and/or 85% Relative Density, will not be susceptible to lique-
faction and a dynamic testing program to demonstrate this conclusion is
not required.

38 25

2.5.4.5.1.6 Substructures and Subsurface Walls

All Category I substructures will be designed to resist full
46 hydrostatic groundwater pressure at all levels below the maximum flood
elevation 809. All mat foundations established below elevation 809 will
be designed to resist hydrostatic uplift pressure. Sufficient concrete
will be provided to ensure that an adequate factor of safety has been
12 incorporated in the design for this condition.

Subsurface walls will be designed to resist lateral earth pressures and hydrostatic water pressure under static and dynamic loading conditions.
46 Dynamic soil and water pressures have been based on the approach outlined by Seed and Whitman (Ref. 138). In static analysis, an at rest lateral earth pressure coefficient equal to 0.7 has
59 been used for compacted well-graded granular fill (crushed rock). The 35
parameters used in the dynamic analysis are summarized below:

a. Fill

Add increment of pressure to static force equivalent to:

ΔP_f = $\frac{1}{2} \gamma H^2 \times 0.75 K_h$ where

H = Height of wall (force applied at 0.5H - pressure diagram is one-half sine curve.)

γ = Unit weight of soil (γ_{moist} above Elev. 780; $\gamma_{submerged}$ below Elev. 780)

46 K_h = 0.25g for SSE and 0.125g for OBE.

b. Water

Add increment of pressure to static force equivalent to:

ΔP_w = $7/8 \times 0.70 K_h \times \gamma_w (hy)^{\frac{1}{2}}$ where

γ_w = Unit weight of water

h = Depth of water

y = Depth from Elev. 780 to plane where dynamic water pressure is determined.

46

Anchors will not be required to resist hydrostatic uplift. Adequate factors of safety for foundation stability will be provided in accordance with the requirements outlined in Section 3.8.5.3.4.

12

2.5.4.5.2 Emergency Cooling Tower

2.5.4.5.2.1 General

Site preparation in the vicinity of the Emergency Cooling Tower located approximately 400 feet north of the Nuclear Island in the Unit A Upper Siltstone will consist of stripping, excavating, dewatering and backfilling operations. Clearing and stripping shall be conducted in conformance with Section 2.5.4.5.1.2 The dewatering control required is expected to be minimal in the relatively impervious residual soil overburden and Upper Siltstone stratum. Drainage ditches will be provided along berms established on the excavated slope faces to intercept rain and run-off. Relief wells located at base level and inclined PVC drains in rock faces will be installed if required.

2.5.4.5.2.2 Excavation

The "Top of Continuous Rock" profile gradually increases toward the north, and it has been possible to locate the Emergency Cooling Tower on the band of siltstone at a higher bearing elevation than the Nuclear Island common mat, viz., EL. 765'. Engineering characteristics of the siltstone below this elevation at this location are consistent within the bearing influence of the structure. Excavation in soil and rock will be required to establish the proposed foundation bearing elevation 765'. Maximum depth of excavation will be approximately 50 feet below the final grade elevation 815'. The minimum depth of excavation will be approximately 35 feet, occurring along the eastern edge of the tower area corresponding to the topographic low elevation 800'.

The excavation made through the cohesive residual overburden soil will be cut on 2:1 slopes (horizontal to vertical), or flatter. In a fashion similar to the Nuclear Island area, excavation will then proceed downward through weathered and sound rock; however, only rock of the Unit A Upper Siltstone formation will be encountered. Cut slopes will be determined and controlled by orientation of joint patterns and bedding planes. Adequate factors of safety against instability will be incorporated. Proposed excavation slopes are illustrated on Figure 2.5-40.

Blasting will be required to remove sound rock and probably a significant portion of the weathered rock. Consequently, a carefully planned blasting program will be implemented to minimize rock overbreak and damage to the foundation area. All blasting will be monitored and controlled by a qualified engineer.

The inspection verification program outlined for the Nuclear Island will also be implemented for the Emergency Cooling Tower foundation. Due to increased thickness of siltstone cover overlying the Unit A Limestone beneath the foundation mat as compared with the Nuclear Island, no additional
59 exploratory borings will be required after excavation.

2.5.4.5.2.3 Structural Fill and Backfill

The Category I Emergency Cooling Tower structure will be supported on a mat foundation placed directly on sound, unweathered siltstone (Unit A, Upper Siltstone). Lean concrete fill will be placed between the edge of the mat and the excavated rock over the full height of the mat. Class A structural backfill will then be placed and compacted around the structure from the level of the top of the lean concrete fill to the design grade elevation 815' as shown in Figure 2.5-40.

Class A structural fill, as described in Section 2.5.4.5.1.5, will be placed in thin, near horizontal lifts and each lift will be compacted to at least 95% of the maximum dry density as determined by the ASTM Test Designation 1557-70 Method of Compaction and/or 85% relative density as determined by ASTM Test Designation 2049, whichever is greater. A qualified engineer will be present during the backfill operations to ensure that the required compaction is being attained by performing in-situ density tests in a prescribed manner and frequency.

2.5.4.5.4 Diesel Fuel Oil Storage Tanks

Two Category I Diesel Fuel Oil Storage Tanks, each measuring approximately 12 feet in diameter and 60 feet in length, will be supported directly by Class A structural backfill overlying the Unit A Upper Siltstone. The tanks will be anchored to a common reinforced concrete mat measuring 70 feet x 42 feet with its base at Elevation 793 feet. The anchorages and the mat will be designed for earth, seismic, tank and hydrostatic uplift loads.

The Class A fill supporting the tanks will be compacted to at least 95% of the maximum dry density as determined by ASTM Test Designation 1557-70 Method of Compaction and/or 85% relative density as determined by ASTM Test Designation 2049, whichever is greater. Compaction to the high densities required by this criteria ensures that liquefaction will not occur in the
59 fill during ground motions generated by the SSE.

2.5.4.6 Groundwater Conditions

Groundwater levels at the site are related to depths of weathering and are influenced by surface topography. Groundwater at the site occurs generally under water table conditions. A history of groundwater fluctuations at the site is presented in Section 2.4.13.2.2.

55 38 18 The planned Nuclear Island common mat foundation elevation of 715 is below site groundwater levels and 20 feet below the normal low stage of the Clinch River. Therefore, to facilitate conventional construction of foundations for the Nuclear Island structures, dewatering will be required. In Section 2.4.13.2.4, it is indicated that the volume or quantity of water to be removed may be large. After construction of the plant is completed, groundwater levels may rise to their normal levels since permanent dewatering operations are not planned.

2.5.4.7 Response of Soil and Rock to Dynamic Loading

Refer to Section 2.5.4.2.2 and 3.7.

2.5.4.8 Liquefaction Potential

Category I structures will primarily be supported directly on the competent Unit A Upper Siltstone stratum. Fuel Oil Storage Tanks will be embedded in Class 'A' granular backfill overlying the Upper Siltstone The structural backfill will be placed and compacted to a specified density which will ensure that the potential for liquefaction will not exist. One portion of the Steam Generator Maintenance Bay will be founded on a mat directly on competent rock of the Unit B Limestone formation. The craneway portion of that structure will be supported by piers extending to competent Unit B Limestone. Class 'A' structural backfill shall be placed around the piers from base level up to grade and shall be compacted
59 to a specified high density which will preclude liquefaction.

2.5.4.9 Earthquake Design Basis

Refer to Section 3.7.

2.5.4.10 Static Analysis

Category I Nuclear Island structures located on the common mat will be founded at Elevation 715 on sound rock in the Unit A Upper Siltstone formation. The static bearing pressure to be produced by the mat is approximately 8.9 KSF.

38

The Emergency Cooling Tower is a Category I structure and will also bear on sound rock in the Unit A Upper Siltstone, at Elevation 765,
59 with a static bearing pressure of approximately 4 KSF.

Surface displacements or settlements under the action of a point load on a semi-infinite elastic solid may be determined by the equation (Ref. 136):

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$$W_o = \frac{P(1-\nu^2)}{\pi E r}$$

where W_o represents the normal displacement of the surface at radius r from a concentrated normal load P, ν is Poisson's ratio and E is Young's Modulus.

Average displacements for a loaded area can be expressed in the form:

$$\overline{W_o} = \frac{\overline{m} P\ (1-\nu^2)}{E\sqrt{A}}$$

where $\overline{W_o}$ is the average displacement of the loaded surface, $\overline{m}$ represents a displacement coefficient dependent on the shape of the loaded surface and the distribution of the load, A is the area of the loaded surface and P represents the total normal load.

55 The average surface displacement of the Nuclear Island common mat foundation can be estimated from the above equation. Assuming that the entire Nuclear Island area of 131,600 ft^2 is uniformly loaded with a static loading intensity of 8.9 KSF, ν = 0.3, E = 5 X 10^5 psi and $\overline{m}$ = 1 the average surface displacement is estimated to be approximately 0.5". Since the
38 55 structural loading of 8.9 KSF is generally less than the effective weight of overburden, settlement under the Nuclear Island will be limited chiefly to the recompression of rebound experienced during excavation.

The average surface displacement of the Emergency Cooling Tower can be estimated in a manner similar to that used for the Nuclear Island. Assuming the entire Emergency Cooling Tower area of 20,500 ft.^{2} is uniformly loaded with a static loading intensity of 4 KSF, ν = 0.3, E = 5 x 10^5 psi and $\overline{m}$ = 1, the average surface displacement is estimated to be less than 0.1 inch. Here again, the structural loading of 4 KSF is generally less than the effective weight of overburden, and settlement under the Emergency Cooling Tower will be limited chiefly to recompression
38 59 of rebound experienced during excavation.

2.5.4.11. Criteria and Design Methods

An estimate of the minimum bearing capacity of the siltstone bearing strata has been made using the following assumptions:

a. Foundation rock is siltstone.

b. Unconfined compressive strength (q_u) of siltstone is a minimum of 3,000 psi.

c. C = 1/2 q_u = 1500 psi and $\emptyset$ = 0.

d. Mat foundation has no surcharge.

With these very conservative assumptions, the bearing capacity for the foundation has been calculated, using the relationship outlined in reference (136) to be about 800 KSF. This calculated bearing capacity exceeds
38 all anticipated static loads by a factor greater than 130.

2.5.4.12. Techniques to Improve Subsurface Conditions

Based on inspection of the core from 115 drill holes, including borings 41, 54 and 55, which were specifically evaluated by NRC, and 41 borings in the immediate vicinity of the Nuclear Island and Emergency Cooling Tower, it is concluded that the Unit A Upper Siltstone is the optimum bearing stratum for the Nuclear Island structures located on a common mat, the Emergency Cooling Tower and the Fuel Oil Storage Tanks. It is also concluded that the engineering characteristics of the foundation stratum are consistent below the proposed base of the common mat (Elevation 715). The results of a detailed geophysical investigation including refraction, cross-hole and up-hole methods, in-situ and laboratory test data and examination of the walls of exploratory holes by geophysical logging devices have supported these conclusions. The significant factors considered in the evaluation of the approximately 400 foot wide band of siltstone and the bordering Unit A and B limestone included depth of weathering, susceptibility to development of solution activity and extent of recorded variations in the engineering characteristics. The low permeability, infrequnet joint spacing, inherent resistance to solutioning and adequate available strength ensure the satisfactory bearing capability of the silstone stratum at the foundation grade elevation of 715 for the Nuclear Island and 765 for the Emergency Cooling Tower.

The interface of the underlying Unit A Limestone and upper siltstone stratum was also checked in some detail with two lines of core borings spaced 50' apart with a 50' to 70' spacing between borings. It
38 has been concluded that the limestone stratum within the immediate static influence of the Nuclear Island structures is competent below Elevation 715. Inclided boring 56 had indicated soil layers extending to Elevation 704, however, the Nuclear Island structures are located
59 approximately 100 feet to the east of this zone.

To confirm the homogeneity and satisfactory bearing quality of the foundation strata, a test grouting program centered around boring 55 has been conducted on the west side of the Plant Island extending through the silstone and Unit A limestone. Water pressure test results, negligible grout take and geophysical logging indicated that foundation conditions applicable to that zone below the Top of Continuous Rock were reasonably consistent for all borings completed in this area. It is concluded that foundation treatment of the siltstone and underlying Unit A limestone will not be required and the results of the test grouting program are essentially
8 confirmatory in scope. Photographs were taken of the core from many of the borings shortly after removal from core barrel and these are available for inspection if additional information is required. Results of this
59 investigation are provided in Appendix 2-C.

Subsequent to the completion of the site investigation for the Nuclear Island, the width of the common mat supporting the Category I structures increased slightly due to a series of design changes. Consequently, a minor portion (10' - 20') of the mat may be supported on Unit A Limestone. It is apparent from the results of the test grouting program that this would not result in a change in the foundation conditions for the Nuclear Island. Additional modifications will be made herein regarding precise mat outline and thickness after the design has been finalized, which is dependent on final seismic
41 analysis.

Some dental work is anticipated, which will consist primarily of removing loose materials including any pinnacles of rock which may be evident after excavation,and filling local depressions with mass concrete. The rock surface will be air cleaned and relief points or vertical cracks will be treated with a slush grout application to prevent entry of rain and run-off. 25

When the proposed base excavation elevation is reached, the area will be carefully cleaned with high pressure air hoses and inspected to determire if additional material should be removed because of deterioration by weathering or shattering by blasting. Shortly after the required base elevation has been exposed and approved for foundation support, the siltstone surface will be coated with a layer of gunite approximately 1" thick to prevent deterioration from weathering. The base shall then be additionally
59 protected by a concrete leveling mat of approximately 3" minimum thickness.

Localized solution zones may be encountered in the side slopes of the excavation. Treatment by grouting will be provided if required to inhibit seepage which may exceed drainage ditch, PVC pipe drain,
59 or base drainage larger capacity.

If solutioning is exposed in the construction slopes on the east side in the Unit B limestone and the finished weathered rock excavation line is adjacent to the Class A structural backfill, dental treatment will be provided by plugging exposed cavities with mass concrete and/or slush grout to provide a contiguous rock excavation line and ensure that the required density of the overlying Class A backfill will be established and maintained during the lifetime of the plant. 25

As indicated in Paragraph 2.5.4.5, very careful control will be exercised in the excavating procedures to ensure that the foundation grade is subjected to minimal disturbance from blasting and construc-
59 tion activity in the removal of rock. To minimize rock damage, a carefully designed and controlled blasting program will be developed before start of construction and will be supervised by a qualified engineer/geologist during construction. The program will contain the elements outlined in the following paragraphs.

The blasting contractor will provide the services of at least one (1) person, thoroughly qualified in the use of explosives, for designing each blast and directing the execution of the blast. The identification and qualifications of such person or persons will be submitted in writing to the engineer, who will approve or disapprove their employment on the project. The contractor will also be required to notify the engineer at least 48 hours before blasting operations are to commence.

All blasts must be designed by the contractor and approved by the engineer before drilling and loading is begun. Each design will include the following information:

1. Number, location, diameter and depth of drill holes shown on a blast plan drawn to scale.

2. Type and grade of explosive, size of cartridge and weight of explosive in each hole.

3. Total amount of explosives in the blast and maximum pounds of explosive per delay interval. 27

4. Delay arrangement scheme showing delay interval proposed for each hole. Type and brand of millisecond delays will also be shown on the blast plan.

The contractor will not deviate from the approved blast design
without written permission from the engineer. The blast plan will be
55 included as part of a permanent blast report compiled by the engineer
showing the contractor's comments on blast results and any recommended
changes to be incorporated in subsequent blasting. The contractor will
be required to comply with any alterations in the blast design ordered by
the engineer.

Where rock requires drilling and blasting, explosives
will be used in controlled amounts. Test blasts will be required
59 These blasts will be monitored by the engineer and used for de-
termining the characteristics of the rock in transmitting vibrations.
Energy ratios less than or equal to 3.0 and/or particle velocities of
less than or equal to 3.3 inches per second will be permitted. A program
to monitor energy ratio and/or particle velocity will be developed as
part of the blast design procedure. This program will be reviewed and
approved by a qualified engineer/geologist. It will include the use of
equipment comparable to the Sprengnether VS-1100 or VS-1200 to measure
particle velocity.

A qualified engineer/geologist will be present at the site
during all excavation operations, including blasting. The engineer/
geologist will determine the stratigraphic and structural relationships
of the materials encountered and evaluate the blasting effects on rock
remote from the blasted area. Corrective action will be taken if required
by overexcavating and backfilling with lean concrete. Records will be
maintained which will include accurate descriptions, photographs and
surveys of any rock zone which has been damaged during the blasting 27
operation.

59

2.5.5 Slope Stability

There are no slopes identified which could affect the nuclear power plant. Therefore, response to requirements of this section and its subsections is not needed.

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Dec. 1980

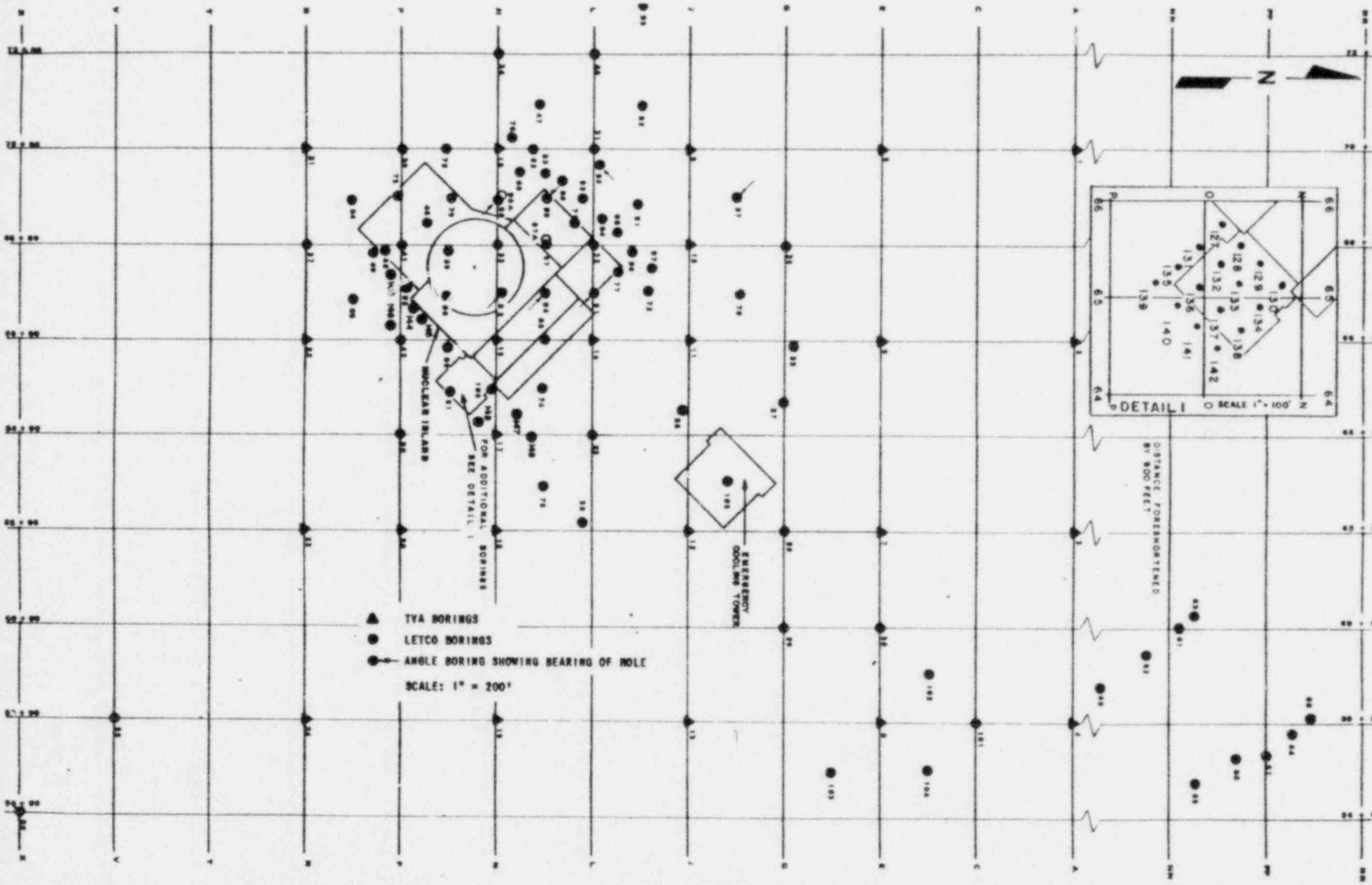


FIGURE 2.5-4 Boring Locations

BCT 405

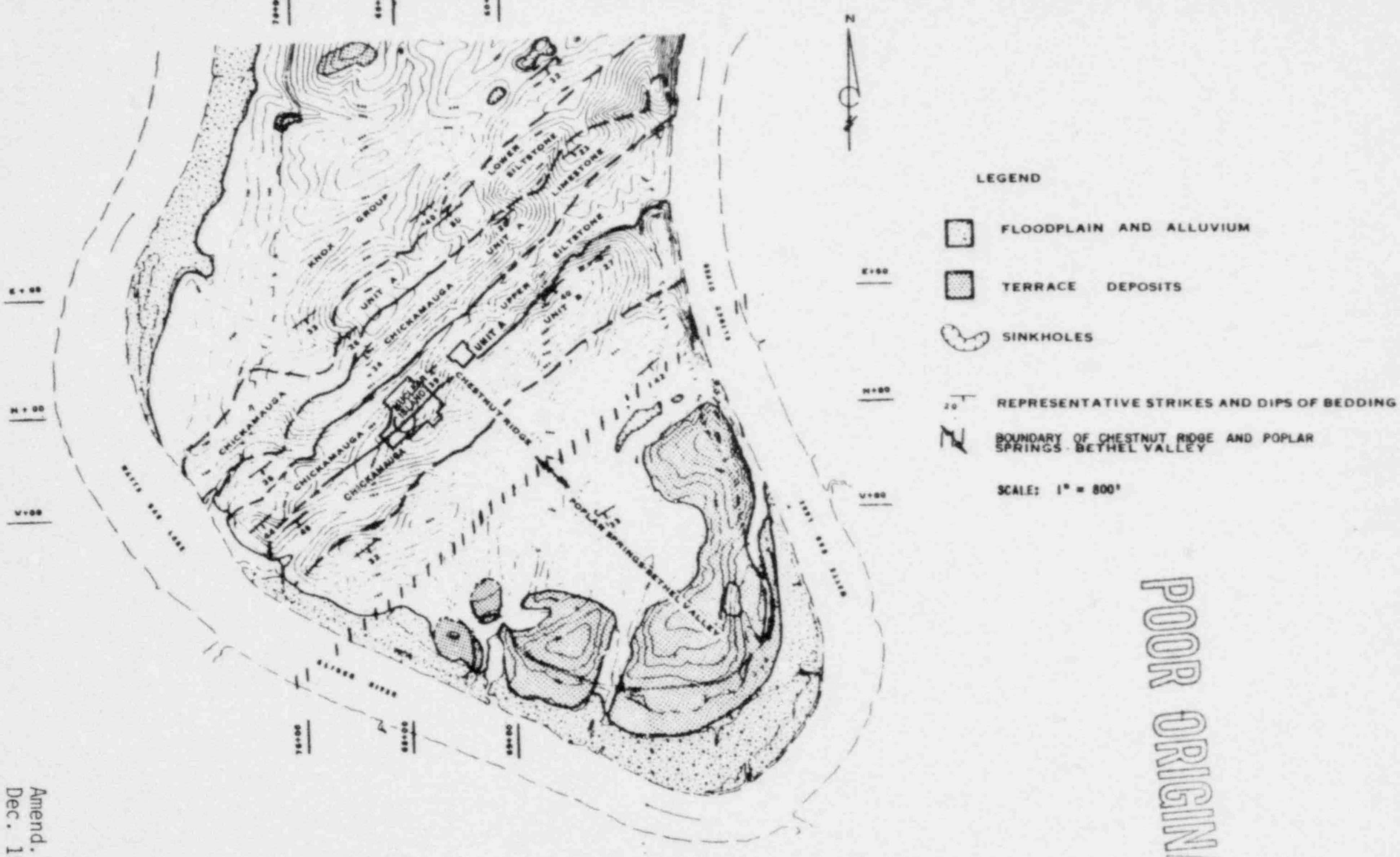


FIGURE 2.5-5 Site Geologic and Physiographic Map

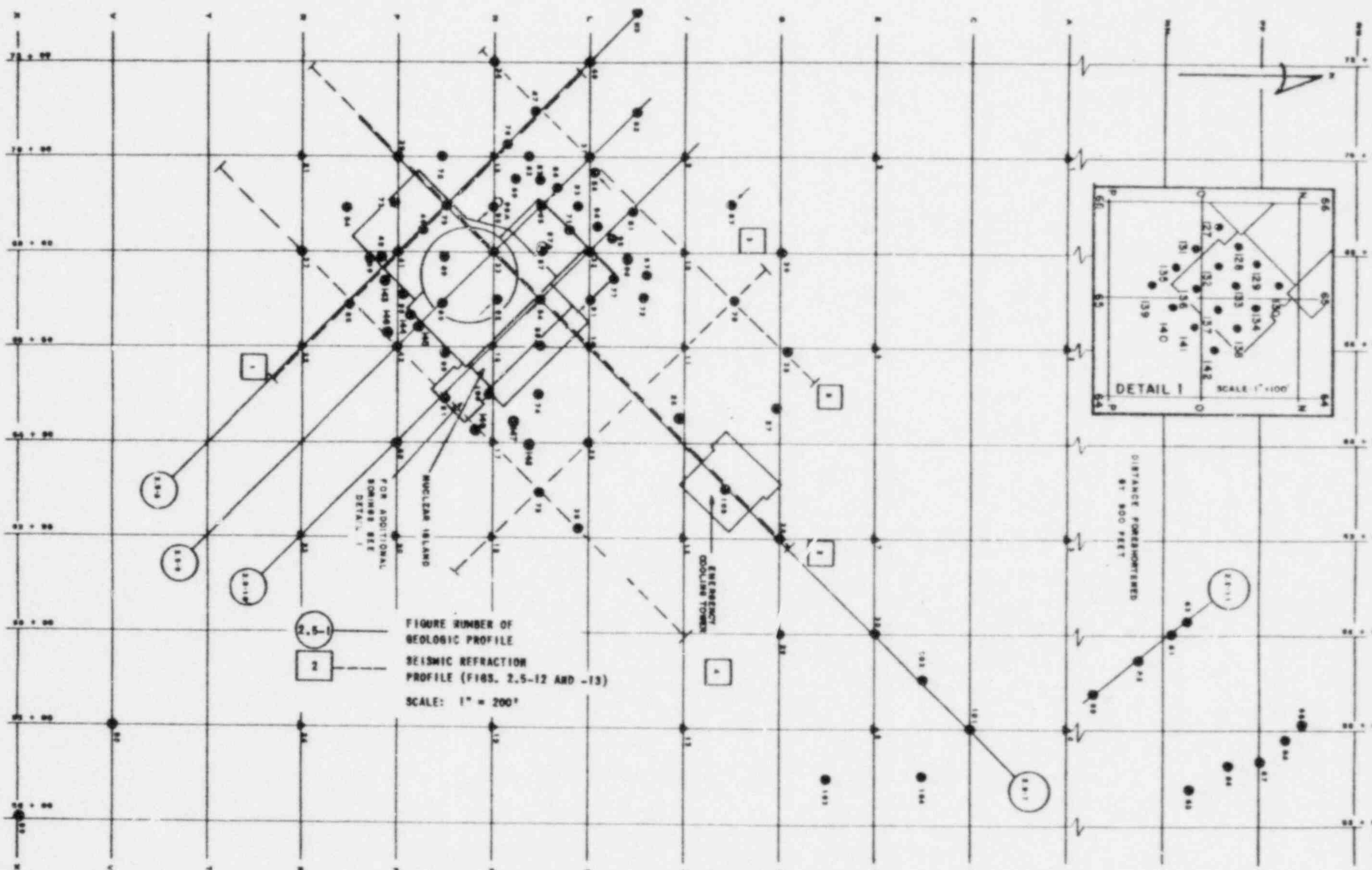
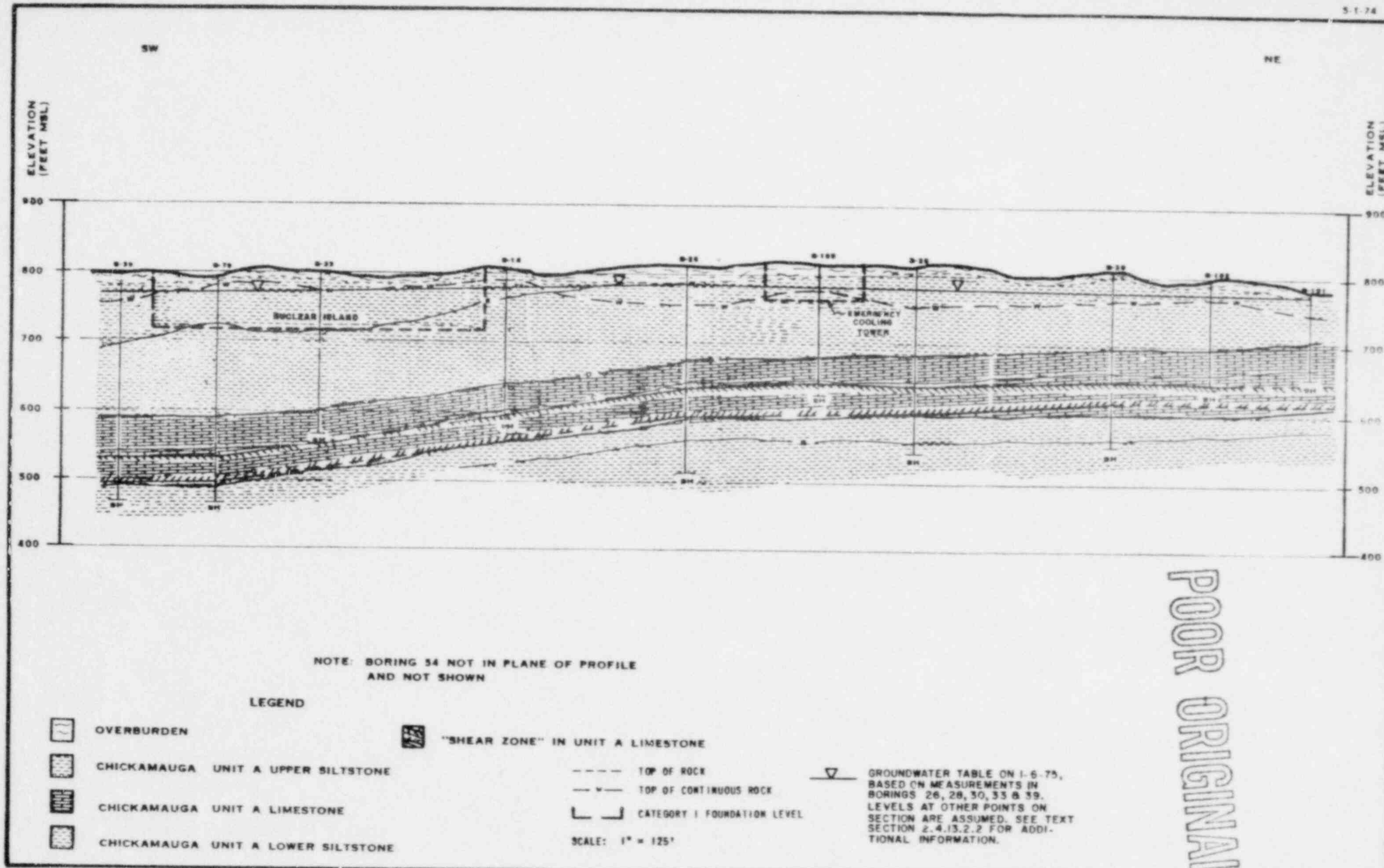


FIGURE 2.5-6 Geologic and Seismic Refraction Profiles

BCT 407

POOR ORIGINAL



POOR ORIGINAL

FIGURE 2.5-7 Site Geologic Profile

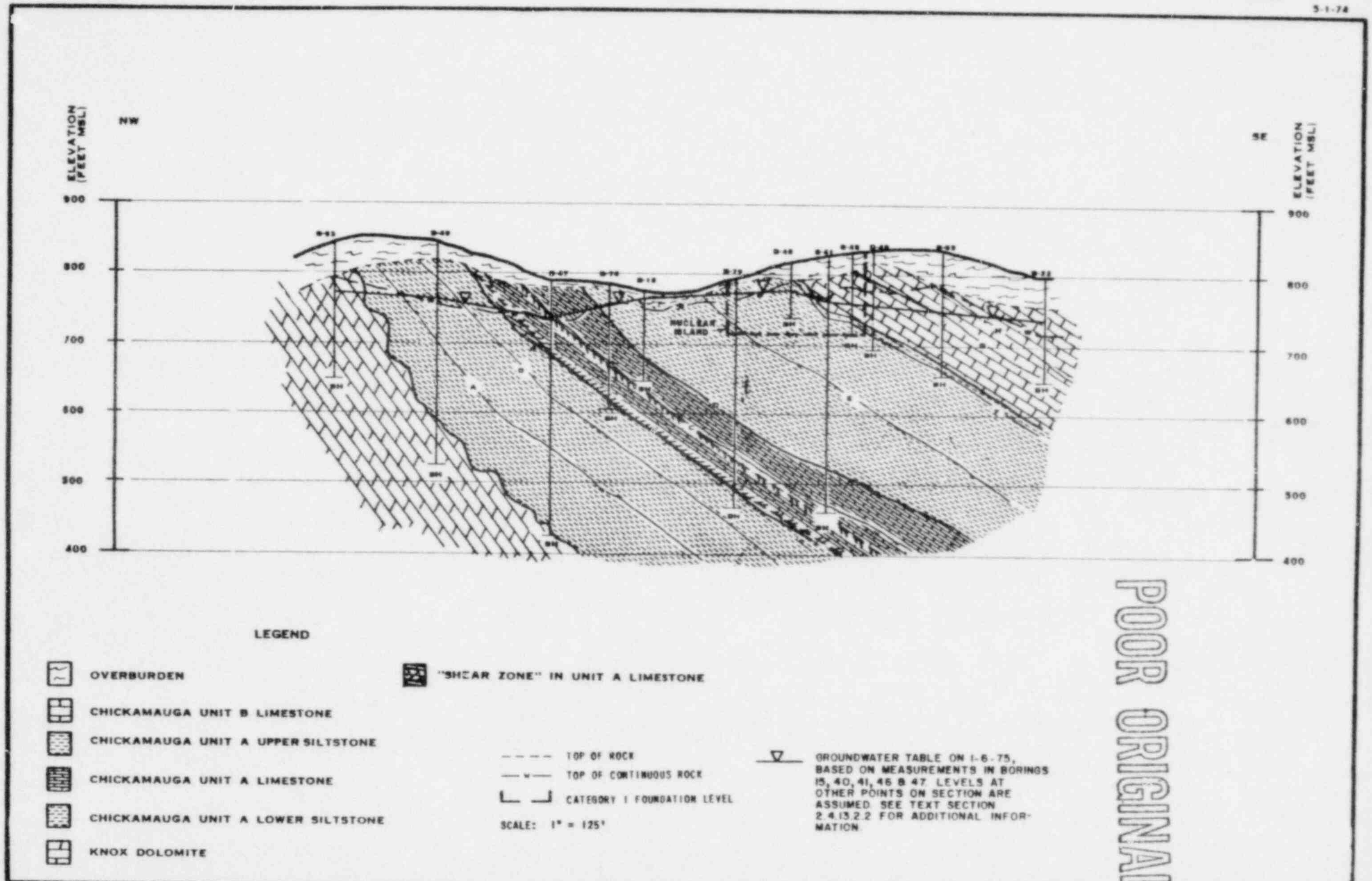


FIGURE 2.5-8 Site Geologic Profile

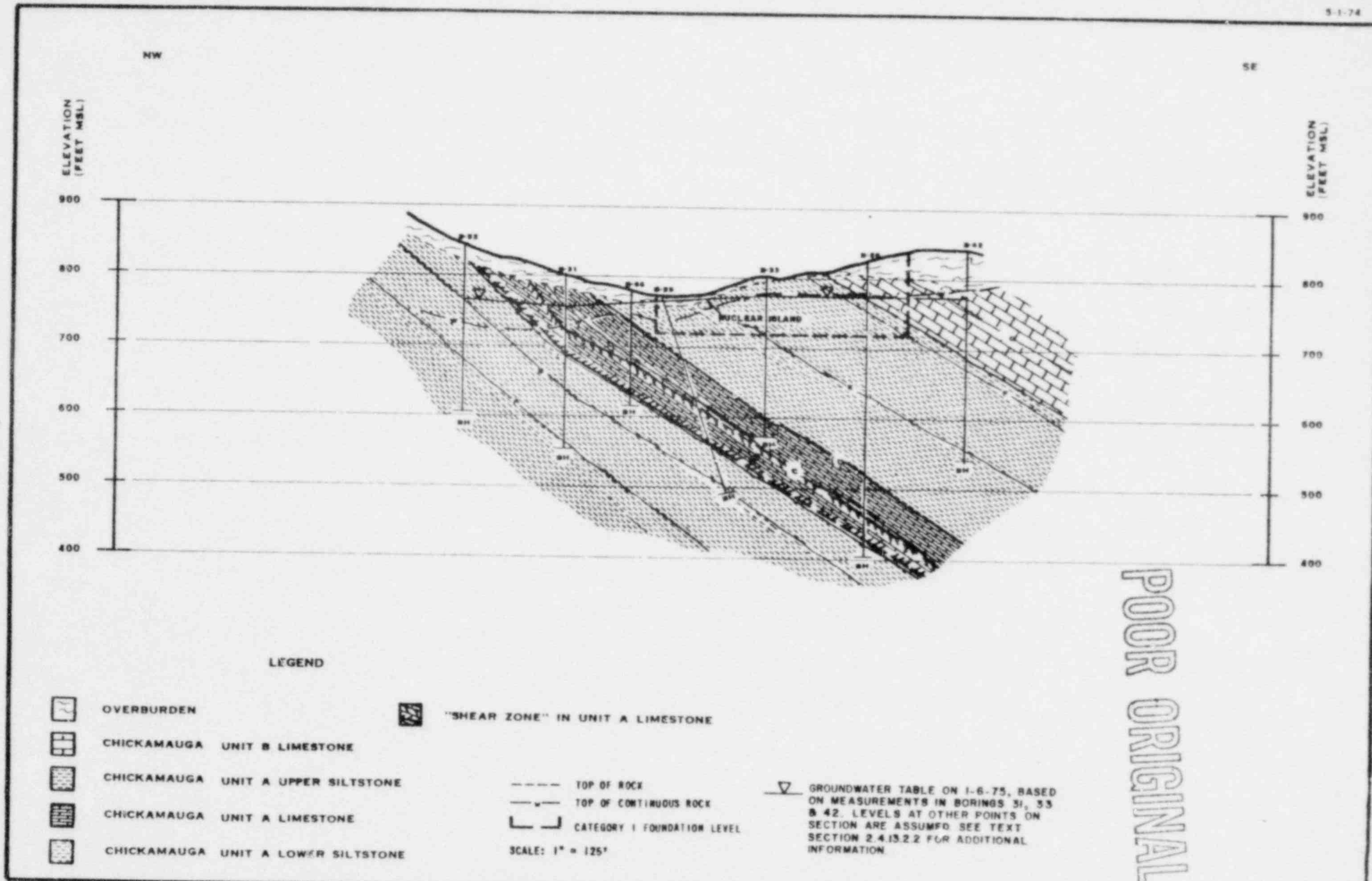


FIGURE 2.5-9 Site Geologic Profile

BCT 403

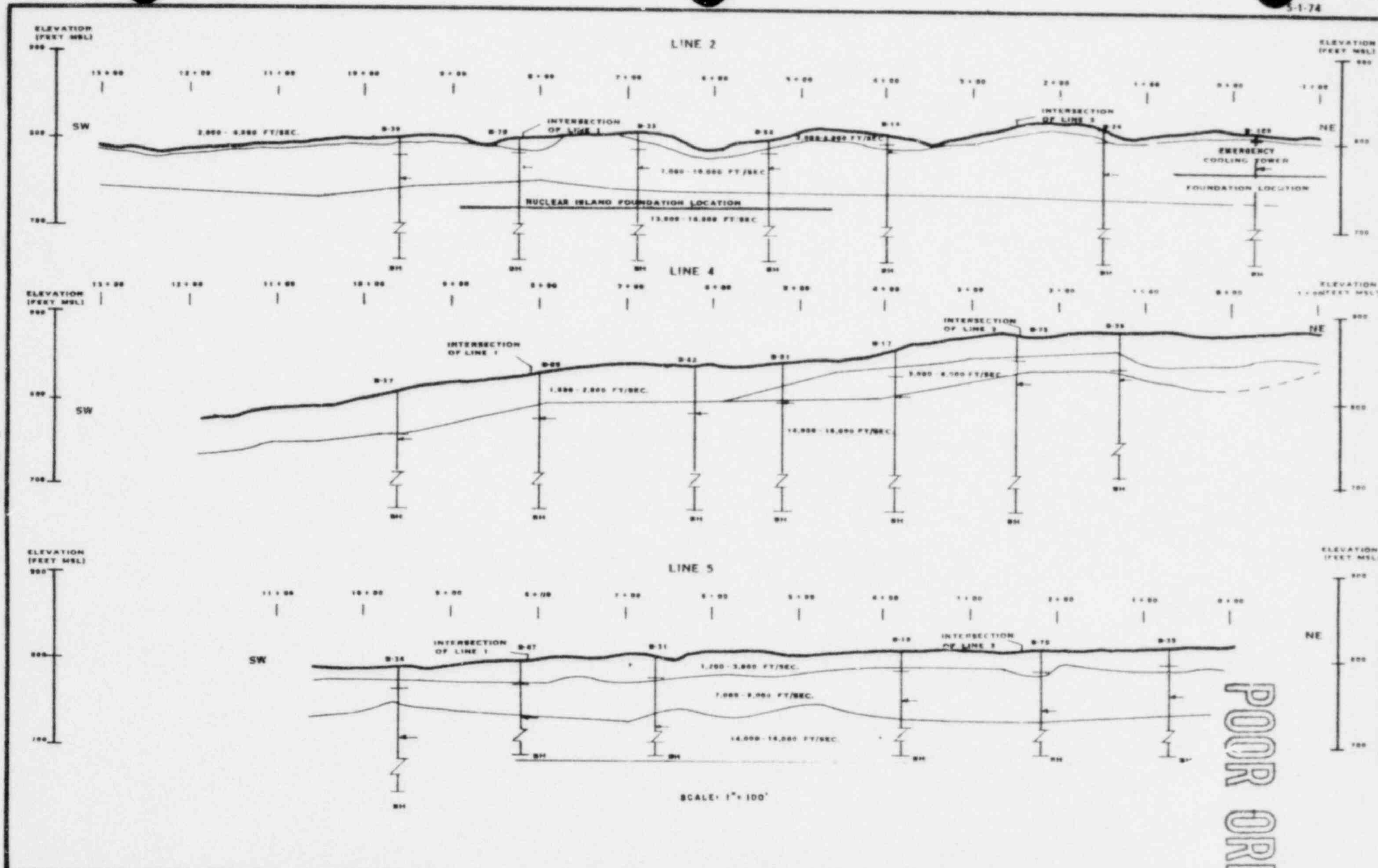


FIGURE 2.5-12 Comparison of Seismic Refraction Profiles and Boring Data

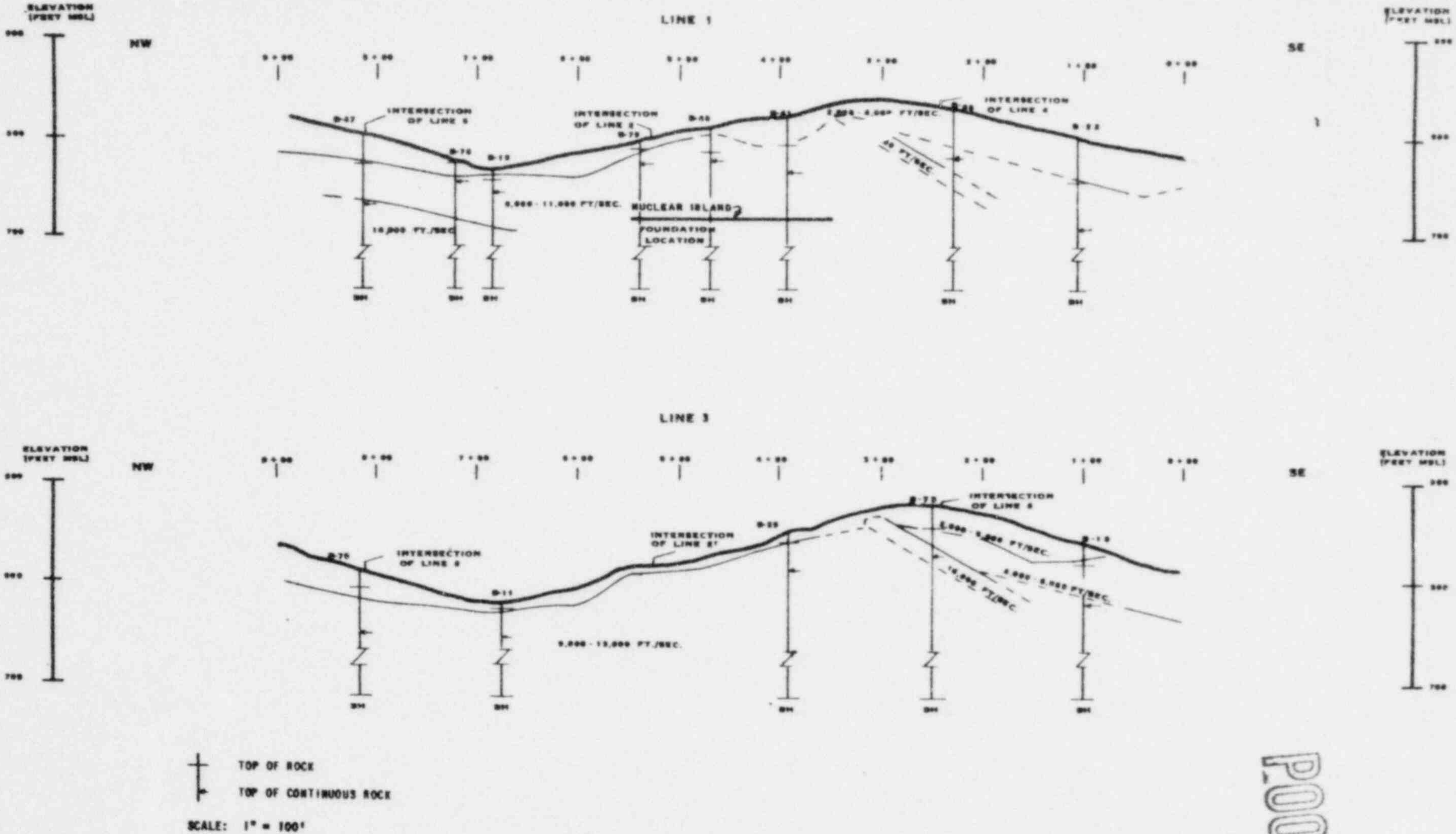
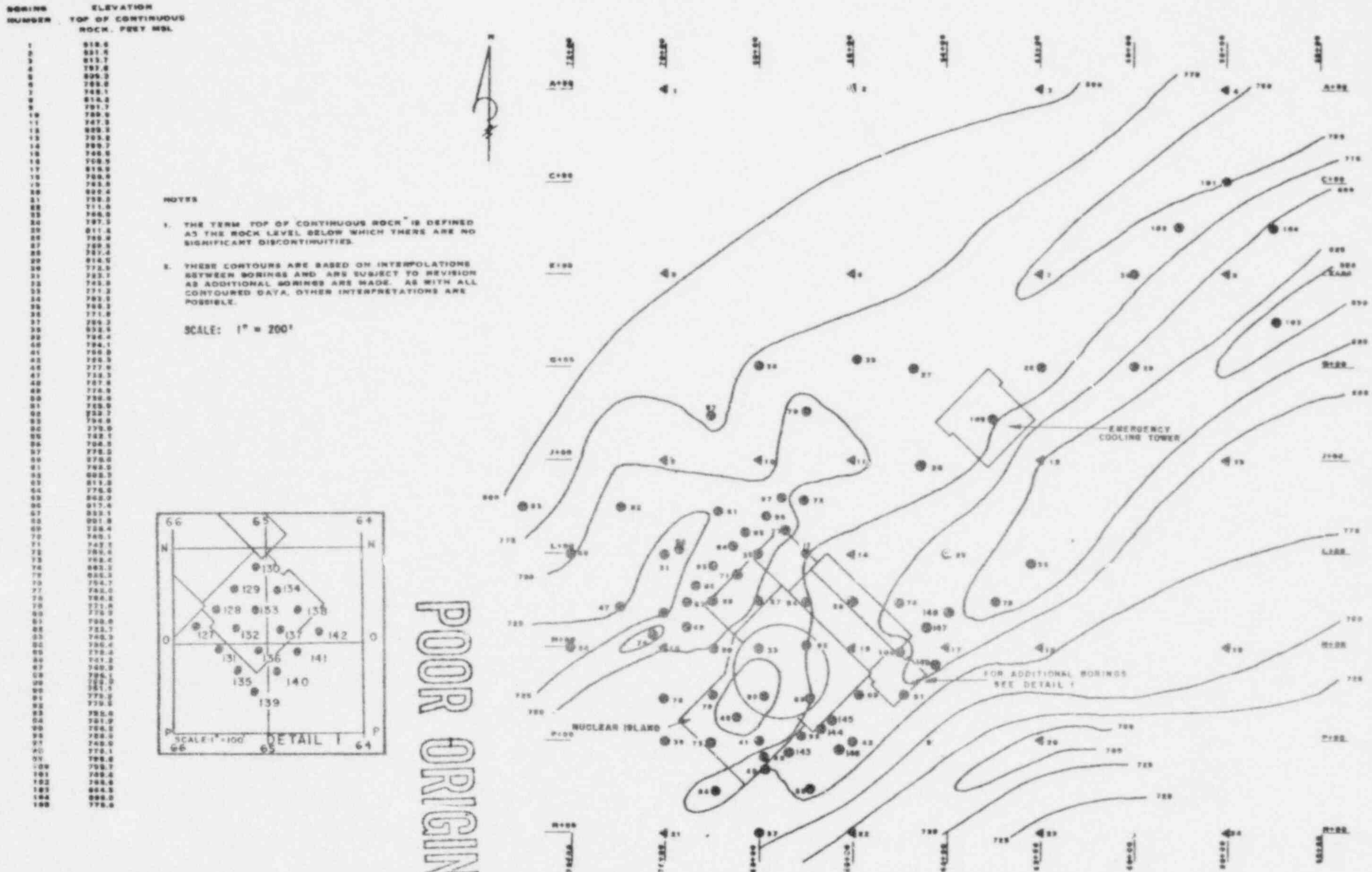


FIGURE 2.5-13 Comparison of Seismic Refraction Profile and Boring Data



POOR ORIGINAL

FIGURE 2.5-16 Top of Continuous Rock

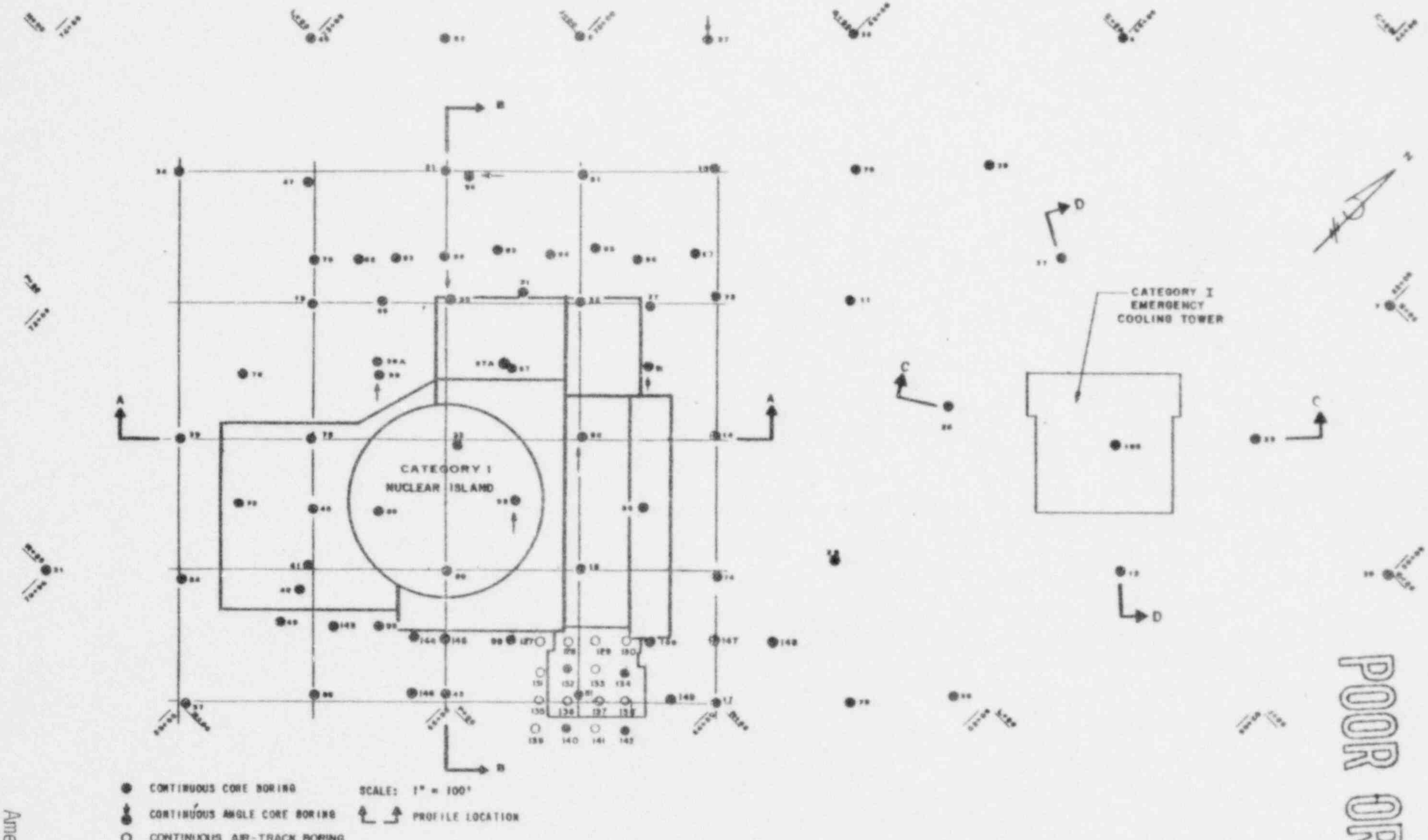


FIGURE 2.5-24 Foundation Boring Plan

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BCT 411

POOR ORIGINAL

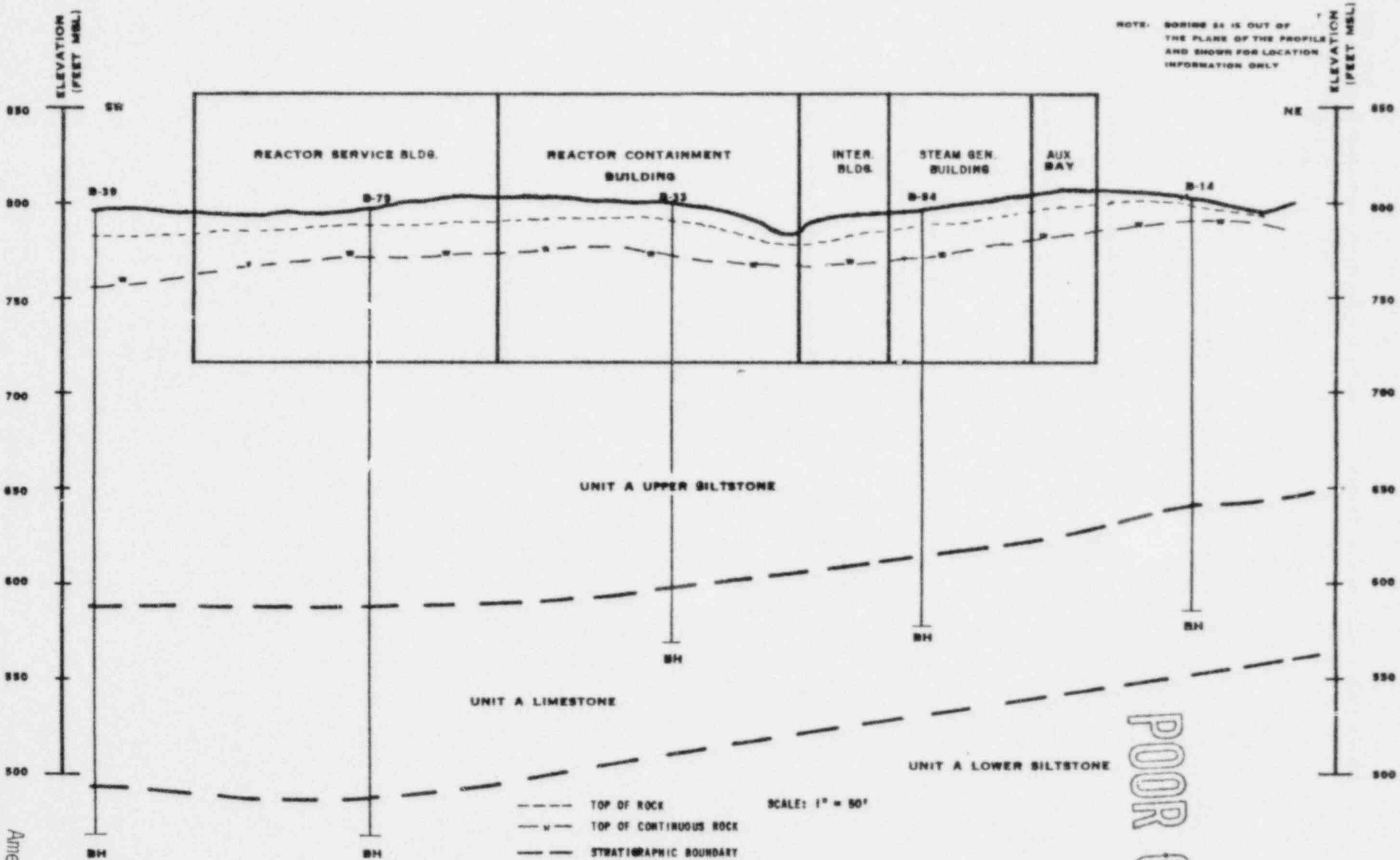

Figure 2.5-25 Nuclear Island Subsurface Profile A-A

6651-27

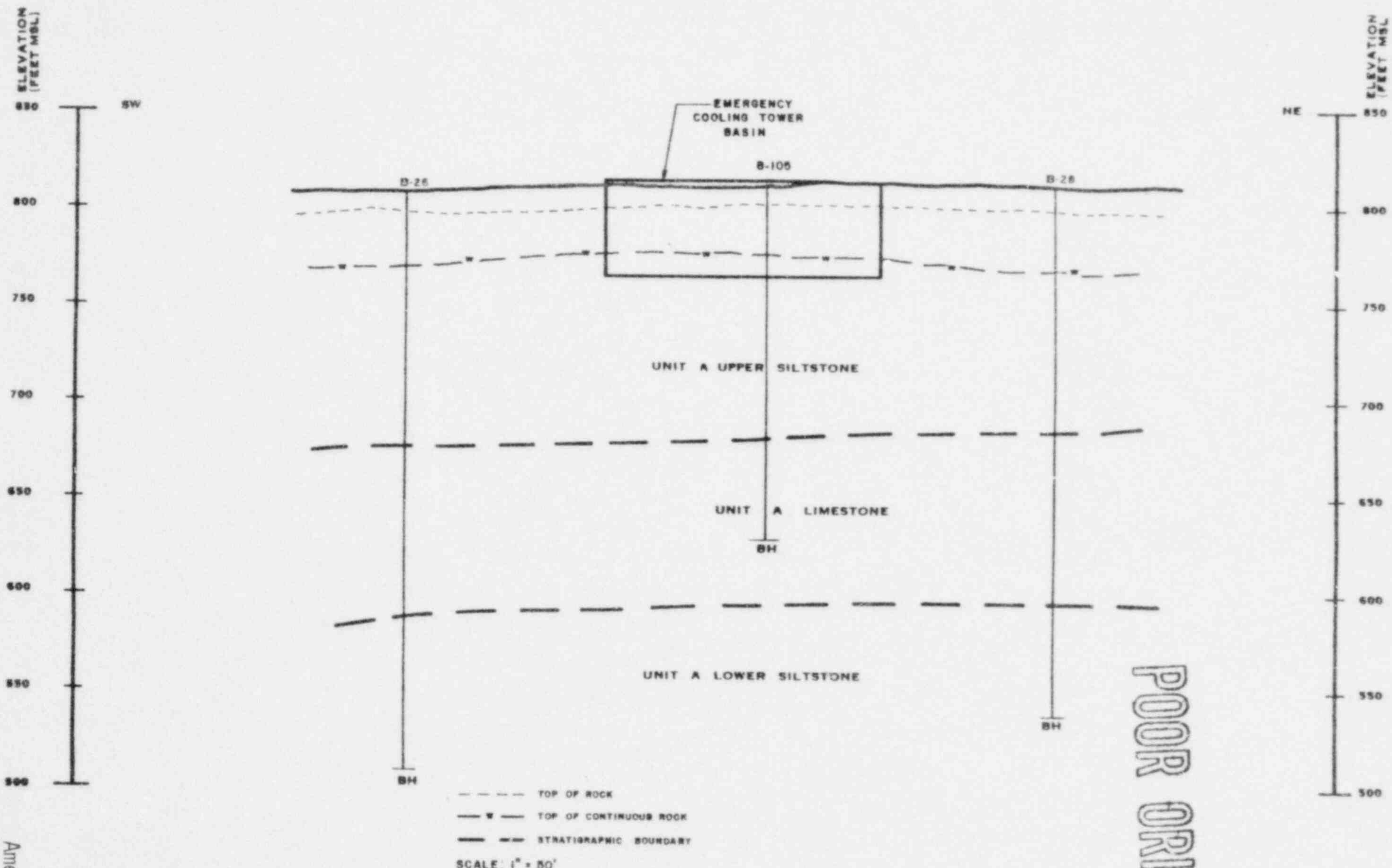

FIGURE 2.5-27 Cooling Tower Subsurface

POOR ORIGINAL

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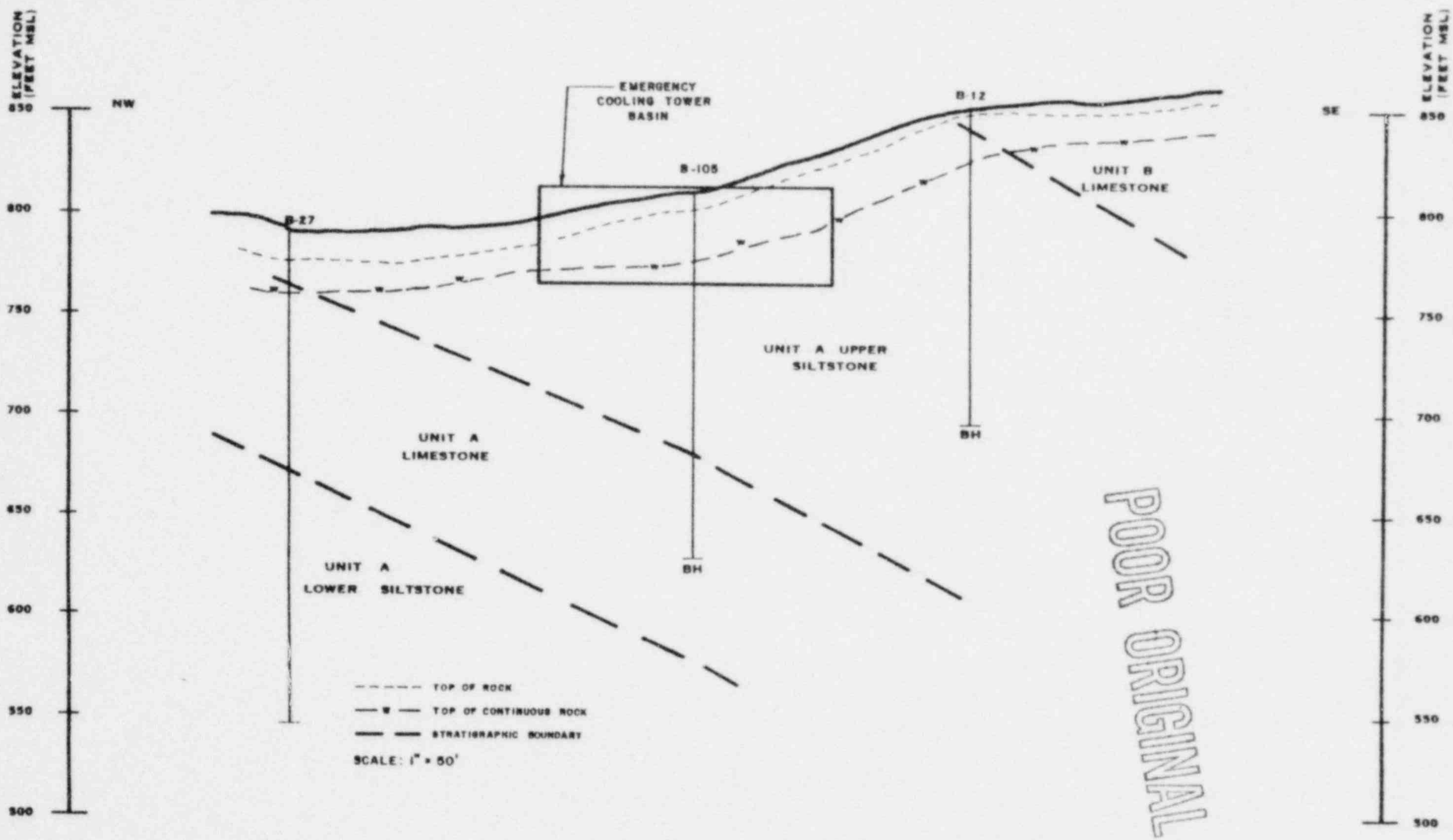

POOR ORIGINAL

FIGURE 2.5-28 Cooling Tower Subsurface

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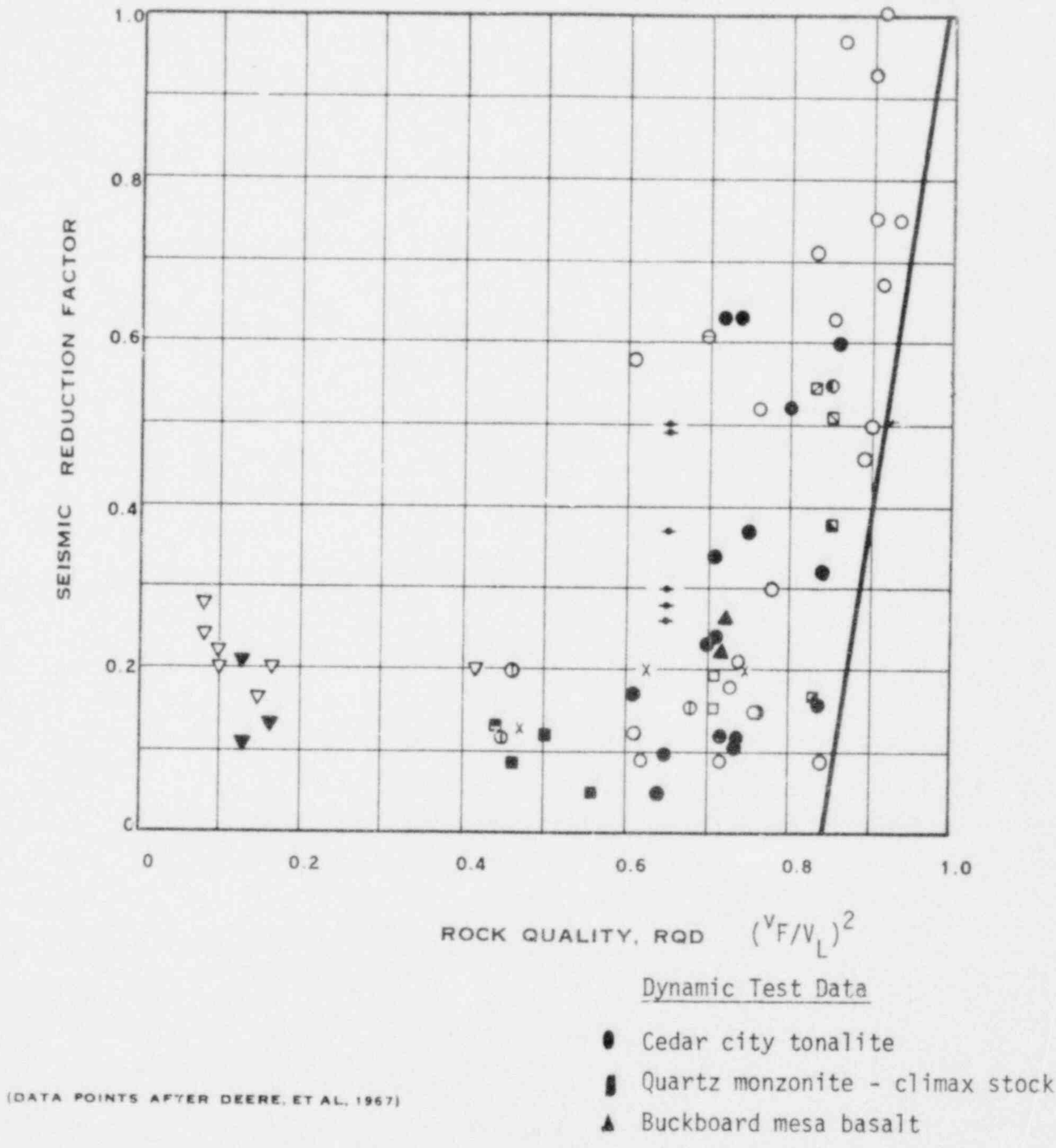
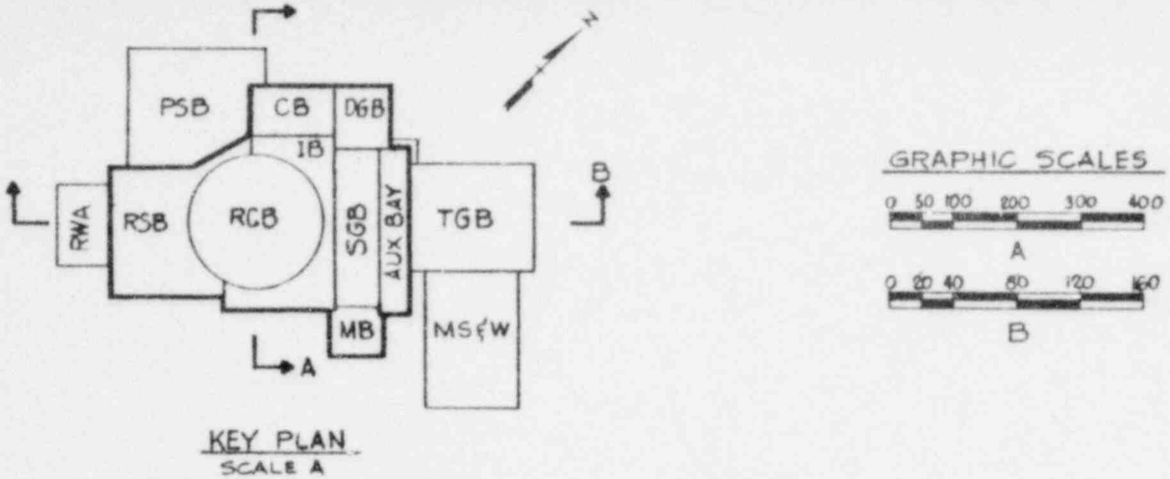
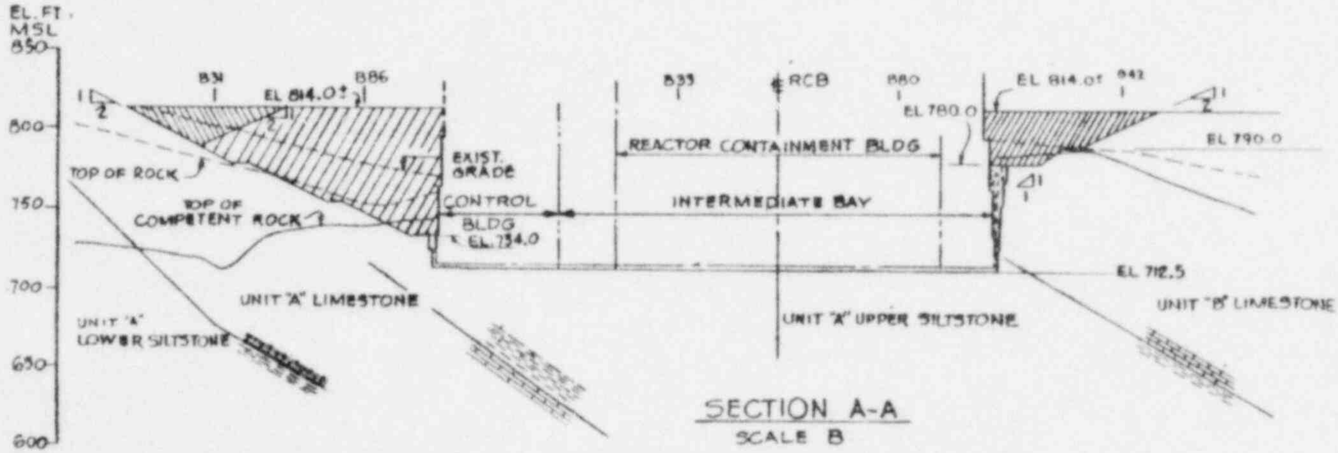
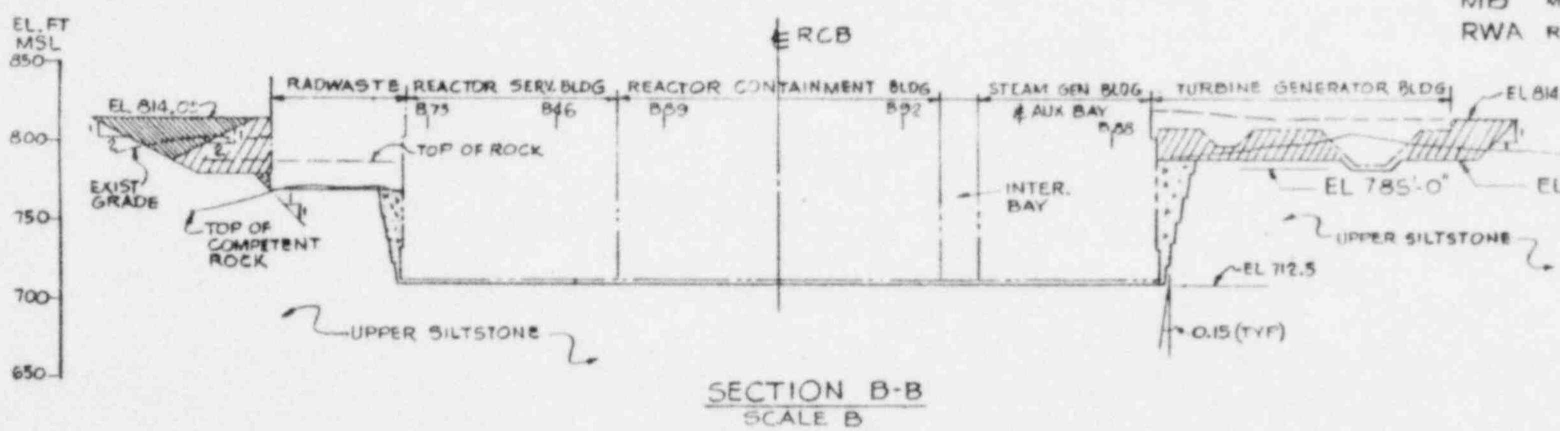


Figure 2.5-38 Seismic Reduction Factor







NOTES

1. CLASS "A" FILL CONSISTS OF WELL GRADED GRANULAR MATERIAL COMPACTED TO AT LEAST 95% OF MODIFIED PROCTOR DENSITY (ASTM TEST DESIGNATION D-1557-70) AND/OR 85% RELATIVE DENSITY, WHICHEVER IS GREATER.
2. CLASS "B" FILL CONSISTS OF WELL GRADED GRANULAR OR COHESIVE MATERIAL COMPACTED TO AT LEAST 90% OF MODIFIED PROCTOR DENSITY (ASTM TEST DESIGNATION D-1557-70) AT OPTIMUM MOISTURE CONTENT
3. HEAVY LINES INDICATE CATEGORY I STRUCTURES

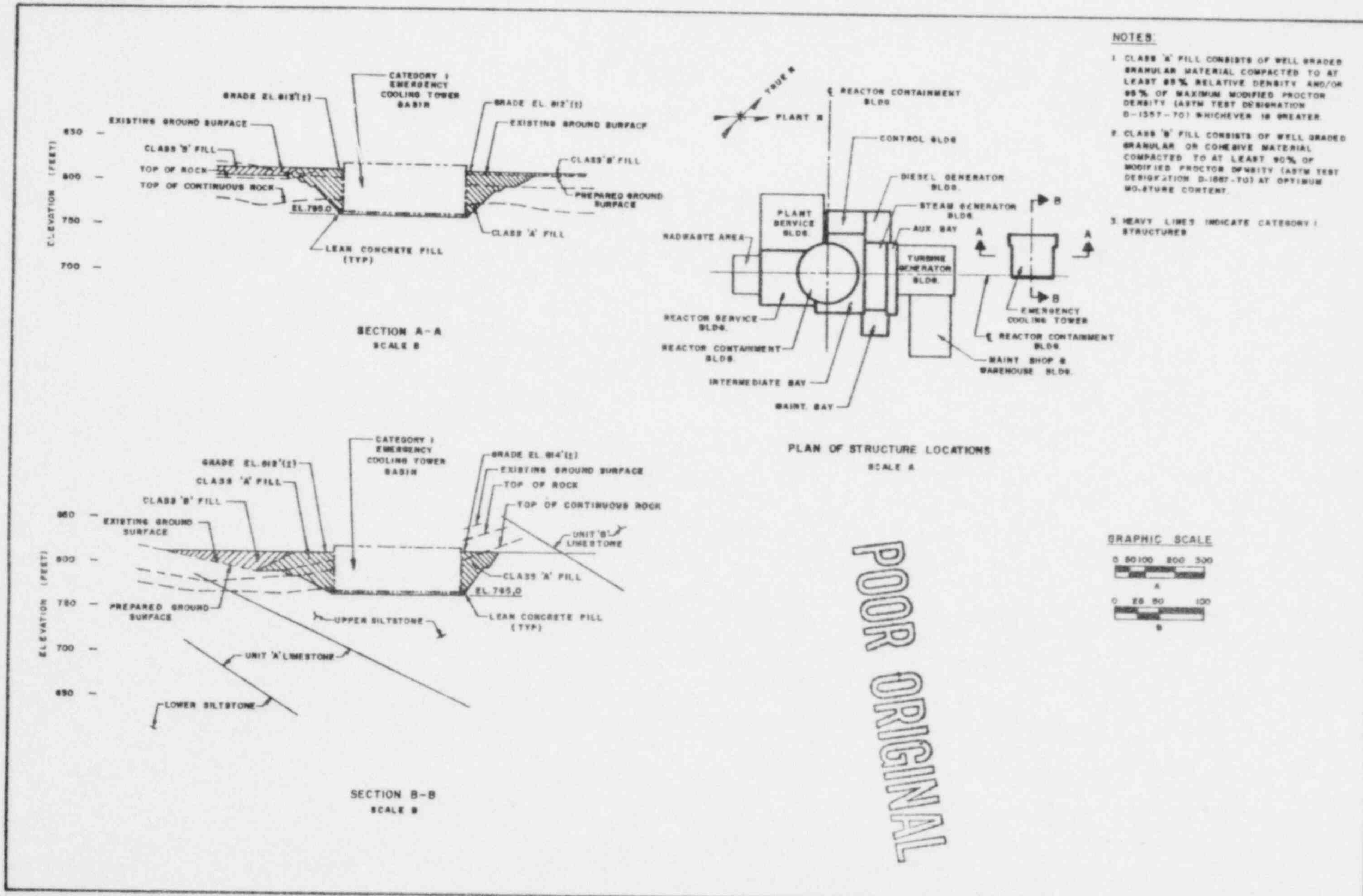
LEGEND

- CLASS "A" FILL
- CLASS "B" FILL
- LEAN CONCRETE FILL
- RCB REACTOR CONTAINMENT BLDG
- SGB STEAM GENERATOR BLDG
- IB INTERMEDIATE BAY
- RSB REACTOR SERVICE BLDG
- CB CONTROL BLDG
- DGB DIESEL GENERATOR BLDG
- TGB TURBINE GENERATOR BLDG
- PSB PLANT SERVICE BLDG
- MS&W MAINTENANCE SHOP & WAREHOUSE
- MB MAINTENANCE BAY
- RWA RADWASTE AREA

POOR ORIGINAL

Amend. 59
Dec. 1980

FIGURE 2.5-37 Nuclear Island Section Excavation & Backfill



BCT 416

FIGURE 2.5-40 Excavation and Backfill Schemes - Category I Cooling Tower

49 TABLE 3.2-3 and 3.2-4

59 HAVE BEEN DELETED

(next page is 3.2-14)

59 FIGURE 3.8-3 and 3.8-4

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(next page is 3.8-52)

1. SCOPE

This appendix established the baseline requirements of the design and analysis of the steel cell liners for the Clinch River Breeder Reactor Plant.

2. APPLICABLE DOCUMENTS

The edition and addenda of the following publications are part of this document and are applicable to the extent specified herein.

2.1. American Society of Mechanical Engineers (ASME)

59 45 2.1.1 Boiler and Pressure Vessel Code, 1974 Edition including Addenda through the Summer 1976

(a) Section II, Material Specifications
(b) Section III, Division I, Nuclear Power Plant Components
(c) Section V, Nondestructive Examination
(d) Section IX, Welding and Brazing Qualifications

59 45 37 2.1.2. Boiler and Pressure Vessel Code, Section III, Division 2, Code for Concrete Reactor Vessels and Containments, 1975 Edition including Addenda through Summer 1976.

59 2.1.3 Boiler and Pressure Vessel Code, Section VIII Division 1, 1974 Edition including Addenda through Summer 1976.

2.2 American Institute of Steel Construction (AISC)

Specifications for the Design, Fabrication and Erection of Structural Steel for Buildings. (1969 including Supplements 1 (11/70), 2 (12/71), and 3, (10/75).)

2.3 Westinghouse Electric Corporation, Advanced Reactor Division (WARD)

45 WARD document No. WARD-D-0037, Seismic Design Criteria for Clingh River Breeder Reactor Plant (Rev 1, 1977), (PSAR Appendix 3.7-A).

3.0 TECHNICAL REQUIREMENTS

3.1 Design Requirements

Cell liners are located in Na and Nak cells in order to maintain the inert atmosphere (Nitrogen) under normal operating conditions, to facilitate decontamination and decrease plant downtime following an accidental sodium spill, and to protect the structural integrity of the cell for the preservation of the capital investment.

The design requirements and the associated criteria used to satisfy each of the requirements for liners, anchors, penetration assemblies, brackets and attachments, and seismic equipment and other structural supports, are described as follows:

3.1.1 Liner Requirements

44 37 .1 The lined cells shall be designed so that the accumulated cell average leak rate shall not exceed 0.36% volume per day in-leakage under a 2.5 inch H_2O negative pressure differential.

Criterion

The leak-tightness of the liner will be preserved by not exceeding the
45 strain limits under Load Combinations A, B and C, per Table 3.8-B-1. The
allowable oxygen concentration and Cell Argon Processing System (CAPS)
44 flow rate shall be based on a 0.36% per day maximum oxygen leakage into
inerted cells under a 2.5 inch H_2O negative pressure differential. 37

37 .2 The liner shall be designed for maximum long term operating conditions of 180°F.

Criterion

45 Strain limits under Load Combinations A and B per Table 3.8-B-1 shall not be exceeded. 37

37 .3 The duty cycle for the liners shall be ten times from 70°F to 140°F, 100 times from 100°F to 140°F and 100 times from 140°F to 180°F for the 30 year plant life.

Criterion

For fatigue evaluation, the methods and limits established by Division 1 of Section III of the ASME B&PV Code will be used. The number of cycles and the temperature ranges are such that based on Section NE-3222.4d of that code, no fatigue analysis is required. 37

37 .4 The liner shall be designed to withstand radiation fluence commensurate with its location within the plant, without radiation degradation which would impair its function.

Criterion

Portions of the Reactor Cavity are the only lined cell areas that are exposed to neutron fluence. The potential for liner brittle fracture will be minimized by controlling the initial Nil-Ductility Temperature (NDT) and controlling the impurity levels of key trace elements in the liner steel with high probabilities of neutron capture. Liner steel degradation due to gamma rays does not present any problems. 37

37 .5 The liner shall be designed for corrosion allowances commensurate with environmental conditions for a 30 year plant design life.

Criterion

A maximum corrosion allowance of 1/6 of the thickness of the liner plate has been assumed in accordance with Section VIII, Division 1, Subsection C, Part UCS-25 of the ASME code.

A maximum of 1/16 inch corrosion allowance will be included in the analysis. This allowance corresponds to the thirty year plant design life. 37

and attachment assemblies for heavy loads shall be anchored directly to the structural concrete walls such that the liner plate and insulating concrete is not loaded in the through-thickness direction. Brackets and attachments for light loads that are not directly anchored in the structural concrete shall, wherever possible, be attached directly to the liner so that they are backed by an anchor or a group of anchors.

3.1.5 Seismic Equipment Support

Seismic Category I equipment will not be directly attached to the liners. Structural supports shall be used for this purpose.

3.1.6 Structural Supports

Structural supports penetrating the liner shall be designed to accommodate all design loads and deformations without loss of structural integrity or leakage requirements specified in paragraph 3.1.1.1. Structural supports shall be such that the liner plate and insulating concrete are not loaded in the through-thickness direction and that the load in the
37 transverse direction is included in the load combinations.

Criterion (for Sections 3.1.4, 3.1.5 and 3.1.6)

Brackets and attachments, seismic equipment supports and structural supports will be designed per AISC "Specification for Design, Fabrication and Erection of Structural Steel for Buildings" for resisting mechanical loads in the construction, test and normal categories. For all other categories, excluding sodium spill conditions, the allowables may be increased by a factor of 1.5. For the load combination 37
including sodium spill conditions the allowables may be increased by
59 1.6.

3.1.7 Concrete Temperatures

The concrete structures shall be designed according to the American Concrete Institute Standard "Building Code Requirements for
54 Reinforced Concrete (ACI 318-71)". As a supplement to the ACI code, paragraph CC 3440 of ASME Section III, Division 2 Code shall be used for concrete allowable temperatures.

3.2 Load Categories

The liner shall be designed for the conditions that are specified to occur during the service life of the liner. These are:

.1 Construction loads
.2 Normal loads
.3 Operating Basis Earthquake Loads
.4 Safe Shutdown Earthquake Loads
.5 Small Liquid Metal Spill Loads
37 .6 Postulated Large Liquid Metal Spill Loads

The design parameters are given in Attachment A of this Appendix.

3.2.1 Construction Loads

Loads resulting from fabrication, construction or preoperational testing are designated as construction loads. Loads to be specified in this category are:

.1 Dead weight during construction
.2 Construction equipment loads
.3 Hydrostatic pressure of wet concrete.

3.2.2 Normal Loads

Normal loads include loads resulting from system startup, power range operation, shutdown and servicing. Loads that are to be specified in this category are:

.1 Dead load, including hydrostatic and permanent equipment load. (D)
.2 Live loads, including any movable equipment loads. (L)
.3 Pressure differential across cell wall. (Po)
.4 Thermal effects due to fluctuations in plant power, loss of cell cooling systems, startup from ambient conditions. (To)
.5 Static reactions and loads from piping and support restraints. (Rc)

3.2.3 Operating Basis Earthquake Load

An Operating Basis Earthquake load is one that could infrequently be encountered during the plant life:

.1 Operating Basis Earthquake (E)

3.2.4 Safe Shutdown Earthquake Load

A Safe Shutdown Earthquake Load is one that is credible but highly improbable:

.1 Safe Shutdown Earthquake (E')

3.2.5 Small Liquid Metal Spill Load

These are loads which were determined to be prudent to provide a capability for accommodation. These loads are:

37 Sodium or NaK leaks less than 25 kg.

The effects to be considered under Small Liquid Metal Spill (SLMS) Loads include:

.1 Differential Pressures on cell walls due to the events within this category. (Pa)
.2 Thermal effects. (Ta)
.3 Pipe reactions from thermal conditions generated by events within this category. (Ra)

45 The above loads are used in Load Combination C.

3.2.6 Postulated Large Liquid Metal Spill Loads

Postulated Large Liquid Metal Spills (PLLMS) are those which include sodium or NaK spills greater than 25 kg.

The effects to be considered under the Postulated Large Liquid Metal Spills include:

.1 Differential pressures on cell walls due to the events within this category. (Pa')
.2 Thermal effects. (Ta')
.3 Pipe reactions from thermal conditions generated by the events within this category. (Ra')

45 The above loads are used in Load Combination D.

3.3 Load Combinations

The following load combinations shall be used for design analysis:

Load Combination A

D + L + To + Po + Ro + E

Load Combination B

D + L + To + Ro + Po + E'

Load Combination C

D + L + Ta + Ra + Pa + E'

Load Combination D

D + L + Ta' + Ra' + Pa' + E'

3.4 Stress and Strain Allowables

37 See Table 3.8-B-1.

TABLE 3.8-B-1

Liner and Anchor Allowables

Use Category	Liner-Stress/Strain Allowable: Membrane	Liner-Stress/Strain Allowable: Combined Membrane plus Bending	Anchors-Force/Displacement Allowable: Mechanical Loads	Anchors-Force/Displacement Allowable: Displacement Limited Loads
Construction	fst=fsc= 2/3 fpy	fst=fsc=2/3 fpy	-	-
45 Load Combinations A and B	ϵsc=0.002	ϵsc=0.004 inch/inch	Lesser of: Fa=0.67Fy	0.25 δu
	ϵst=0.001	ϵst=0.002 inch/inch	Fa=0.33Fu	
Load Combination C	ϵsc=0.005	ϵsc=0.014 inch/inch	Fa=0.9Fy	0.50 δu
59 See Note (2)	ϵst=0.003	ϵst=0.010 inch/inch	Fa=0.5Fu	
Load Combination D	SEE NOTE (1)			

Notes:

(1) For load combinations which include Combination D loads the Von Mises effective strain shall not exceed 0.5 ϵu for membrane
59 and 0.67 ϵu for combined membrane plus bending for both liner plates and anchors.

(2) For local conditions like jet impingement or minor spills the limits given for Load Combination D shall apply.

where:

fst = allowable liner plate tensile stress

fsc = allowable liner plate compressive stress

fpy = specified tensile yield strength of liner steel

ϵsc = allowable liner plate compressive strain

ϵst = allowable liner plate tensile strain

Fa = allowable liner anchor force capacity

Fu = liner anchor ultimate force capacity

Fy = liner anchor yield force capacity

δu = ultimate displacement capacity for liner anchors

ϵu = ultimate strain of liner material under the environmental conditions of interest. ϵu shall be separately evaluated for weld metal and base metal; potential
37 aging and hardening effects shall be considered.

3.5 Design Analysis Procedures

3.5.1 General

.1 An analysis shall be prepared in sufficient detail to show that each of the design allowables of Table 3.8-B-1 is satisfied when the liner, penetrations, brackets and attachments are subjected to the specified loadings.

.2 Experimental results may be used to evaluate the capacity of components to withstand static or cyclic loading.

3.5.2 Liner and Anchors

The liner analysis shall be consider deviations in liner geometry due to the fabrication and erection tolerances which are stated in the construction specification. For loading combinations where the stresses are below yield the liner analysis may be based on plate or beam theory provided all assumptions made in the analysis are conservative.

For those loadings where the stresses in the liner and anchor exceed yield, a finite element elasto-plastic analysis of the liner-anchor system shall be conducted. The material behavior shall be represented using stress-strain properties representative of the actual material at the temperature of the analysis. The finite element technique may also be used for conditions where the stresses are below yield.

3.5.3 Penetrations and Openings

.1 Careful attention shall be given to the analysis of the Steel Cell Liner (SCL) in the vicinity of penetration and openings.

.2 The thermal stresses caused by process piping passing through the wall shall be considered in the design and analysis of the SCL.

.3 Penetration assemblies shall be analyzed using the same techniques and procedures used for metal containments (ASME B&PV Code Section III, Division 1).

3.5.4 Brackets and Attachments

Brackets and attachments connected to the liner shall be designed and analyzed using accepted techniques applicable to beams, columns, and weldments per the AISC, "Specification for Design, Fabrication, and Erection of Structural Steel for Buildings", for resisting mechanical loads in the construction, test and normal categories. For all other
37 categories, for the load combination including sodium spill conditions the
59 allowables may be increased by 1.6 excluding sodium spill conditions, the
allowables may be increase by a factor of 1.5 except for impulse loads and
37 impact effects.

3.6 Design Analysis by Model Testing

.1 Where analytical procedures to predict the liner ultimate strength and behavior in the range approaching failure are not established, structural modeling to establish the liner characteristics and to verify adequate liner performance is required.

.2 The required model shall maintain similitude, including that of materials, to the prototype design and be geometrically similar with respect to the principal dimensions of the prototype.

.3 Failure of the model shall be considered as having occurred upon breach of the liner integrity.

3.7 Design

3.7.1 General

Due to the nature of the loads and other effects together with the types of components, the allowable capacities of the components shall be specified in terms of stress, strain, force or displacement, whichever is applicable. When the allowable capacities are based on ultimate capacities, testing of a prototype may be necessary to verify the ultimate capacity of a particular part.

3.7.2 Liner and Anchors

.1 The calculated strains and stresses for the liner shall not exceed the values given in Table 3.8-B-1. The load combinations are given in Section 3.3.

.2 The liner shall be anchored to the concrete so that the liner strains do not exceed the strain allowable given in Table 3.8-B-1. The anchor size and spacing shall be chosen so that the response of the liner is within the allowable limits for all the loads and load combinations. The anchorage system shall be designed so that it can accommodate the design in-plane (shear) loads or deformations exerted by the liner and loads applied normal to the liner surface.

3.7.3 Penetration Assemblies

.1 Each penetration shall be provided with an anchorage system capable of transferring pressure loads and other mechanical loads such as piping reactions to the concrete. The anchorage design allowables are the same as those given in Table 3.8-B-1 for liner anchors. 37

Amend. 37
March 1977

.2 The design allowables for the penetration nozzles shall be the same as those used for Division 1 of Section III of the ASME B&PV Code.

3.7.4 Brackets and Attachments

The design allowables for brackets and attachments shall be the same as those given in the AISC, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings", for resisting mechanical loads in the construction, test and normal load categories. For all other categories excluding sodium spill conditions, the allowables may
37 be increased by a factor of 1.5. For the load combinations including
59 sodium spill conditions the allowables may be increased by 1.6.

3.7.5 Fatigue

Fatigue is not expected to be a controlling factor in the liner design due to the small number of cycles. The designer shall verify the liner against fatigue using the thermal duty cycle specified in Section 3.1.1.3. The fatigue methods and limits established by Division I of
37 Section III of the ASME B&PV Code shall apply.

3.7.6 Welds

Cell liner seam welds shall be full penetration welds designed in accordance with the requirements of ASME Code, Section III, Division 2, Article CC-3000. The full penetration cell liner seam welds of the Fuel Handling Cell shall be designed in accordance with the requirements of ASME Code, Section III, Division 1, Article ND-3000. 37

3.8 Design Details

37 See PSAR Section 3A.8.

ATTACHMENT A

37 Design Parameters for Cell Liners

I. Structural Concrete

A. Density: 147 lb/ft^3 at ambient temperature (~70^oF)
B. Thermal Conductivity: (See Figure 3.8-B.1
C. Specific Heat: (See Figure 3.8-B.2)

II. Lightweight Concrete (Wall and Ceilings)*

A. Density: 65 lb/ft^3 at ambient (~70^oF)
B. Thermal Conductivity: (See Figure 3.8-B.3 except for the lightweight concrete behind the cell liner walls and ceiling in cells 101C, D and E (not common with the reactor cavity) which have an effective thermal
59 conductivity of 0.24 BTU/hr.-ft-^oF.)
C. Specific Heat: (See Figure 3.8-B.4)
D. Lightweight Concrete Thickness: 4 inches except in Cell 102A which has 8 inches at the Primary Sodium Storage Vessel Area.

III. Lightweight Concrete at Liners (Floors)*

A. Density: 65 120 lb/ft^3 at ambient temperature (≈70^oF)
59 B. Thermal Conductivity: (See Figure 3.8-B.3 except for Cells 101C, D and E which have an effective value of 0.24 BTU/hr-ft-^oF)
59 C. Specific Heat: (See Figure 3.8-B.4)
D. Lightweight Concrete Thickness: 4 inches except in Cell 102A which has 10 inches at floor Elevation 733'-0"

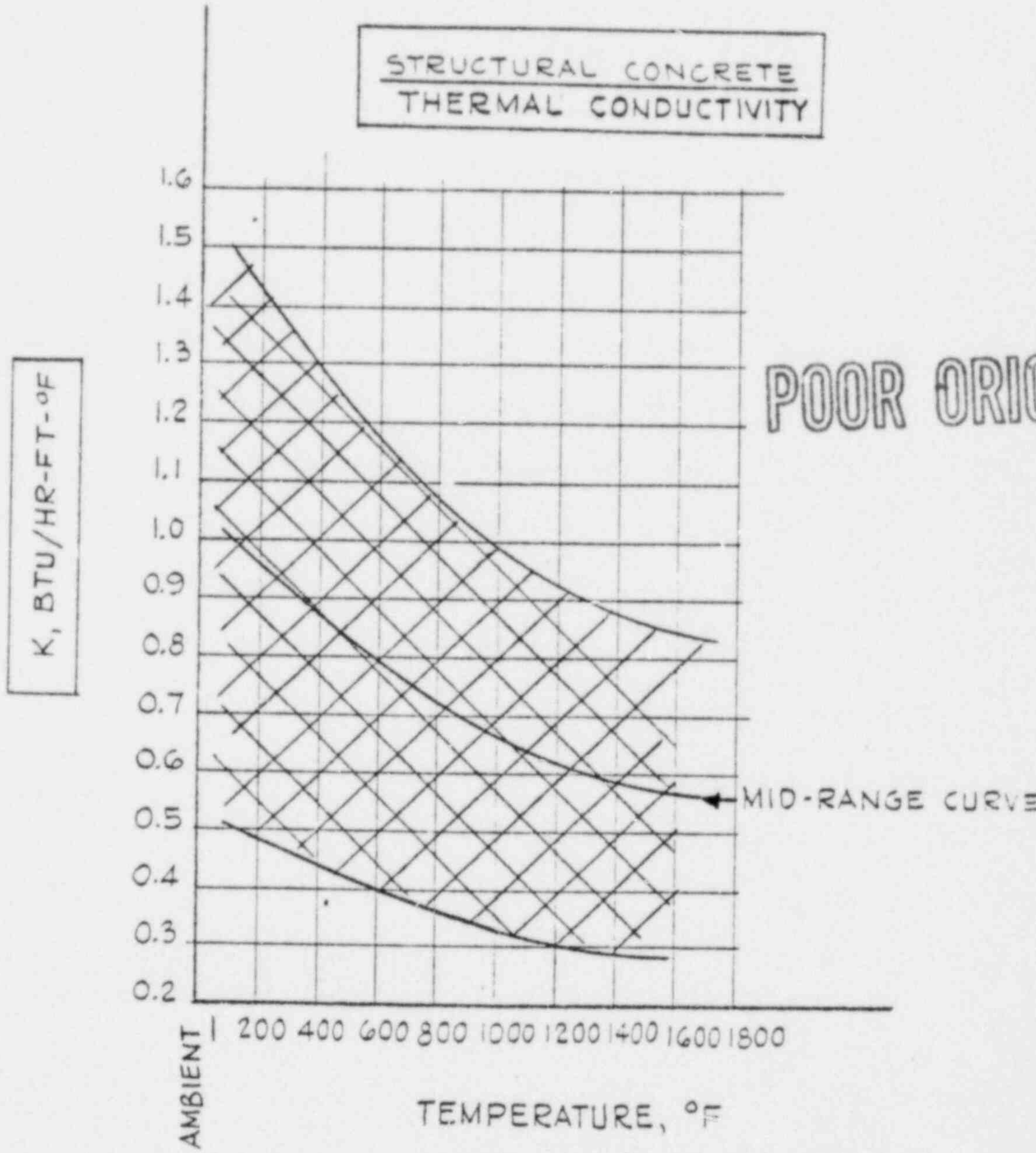
59 IV. Liner Steel: Mild Steel (1% Carbon) for all cells except the Fuel Handling Cell.

A. Density: 490 lb/ft^3 at approximately 70^oF
B. Thermal Conductivity: 25 Btu/hr-ft-^oF at approximately 800^oF
C. Specific Heat: 0.15 Btu/lb-^oF at approximately 800^oF
D. Steel Plate Thickness: 3/8 inch
E. Air Gap Width Behind Steel Plate: 1/4 inch (walls/ceilings)
59 1/8 inch (Floors)

V. Liner Steel for Fuel Handling Cell: Stainless Steel

A. Density: 501 lb/ft^3 at approximately 70^oF
B. Thermal Conductivity: 13 Btu/hr-ft-F^o at approximately 800^oF
C. Specific Heat: 0.14 Btu/lb-F^o at approximately 800^oF
D. Steel Plate Thickness: 1/4 inch
E. Air Gap Width Behind Steel Plate: 1/4 inch (walls and ceilings)
59 1/8 inch (Floors)

*Specific wall, floor, and ceiling insulation thicknesses have not been
54 established for the Fuel Handling Cell (341).

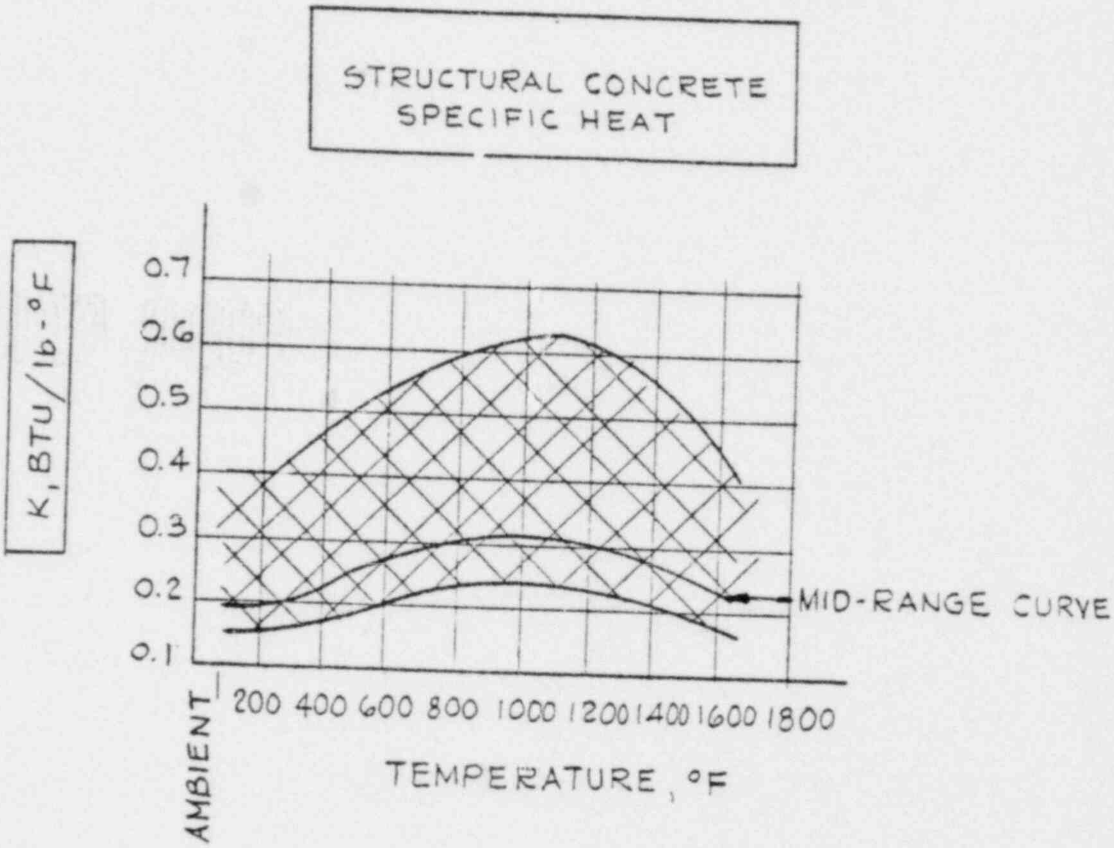


POOR ORIGINAL

NOTE: All CRBRP analyses shall be based upon the Mid-Range Curve values until completion of the DRS (Development Requirement Specification) testing programs.

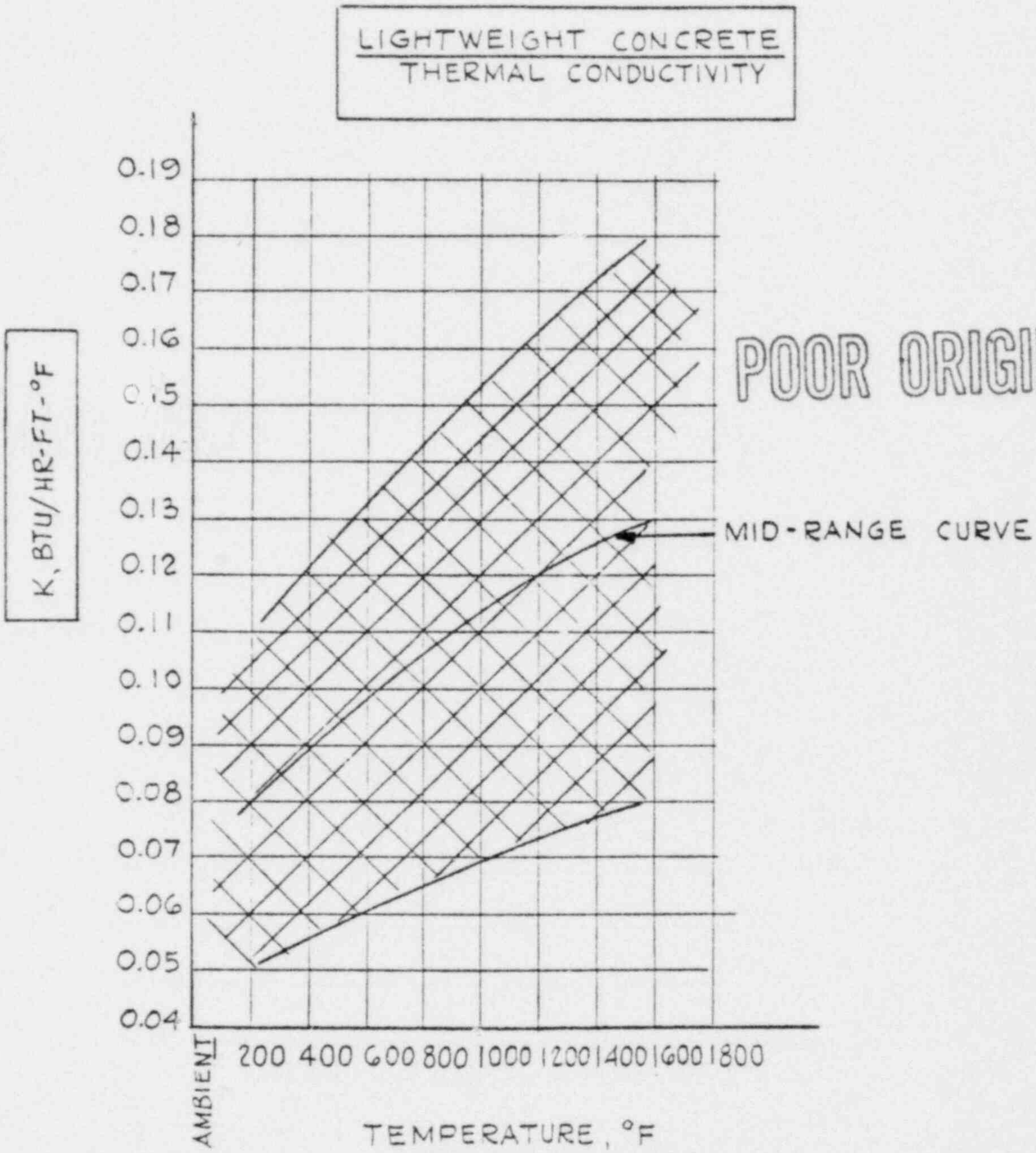
59

Figure 3.8-B.1



NOTE: All CRBRP analyses shall be based upon the Mid-Range Curve values until completion of the DRS (Development Requirement Specification) testing programs.

59 Figure 3.8-B.2



POOR ORIGINAL

NOTE: All CRBRP analyses shall be based upon the Mid-Range Curve values until completion of the DRS (Development Requirement Specification) testing programs.

59

Figure 3.8-B.3

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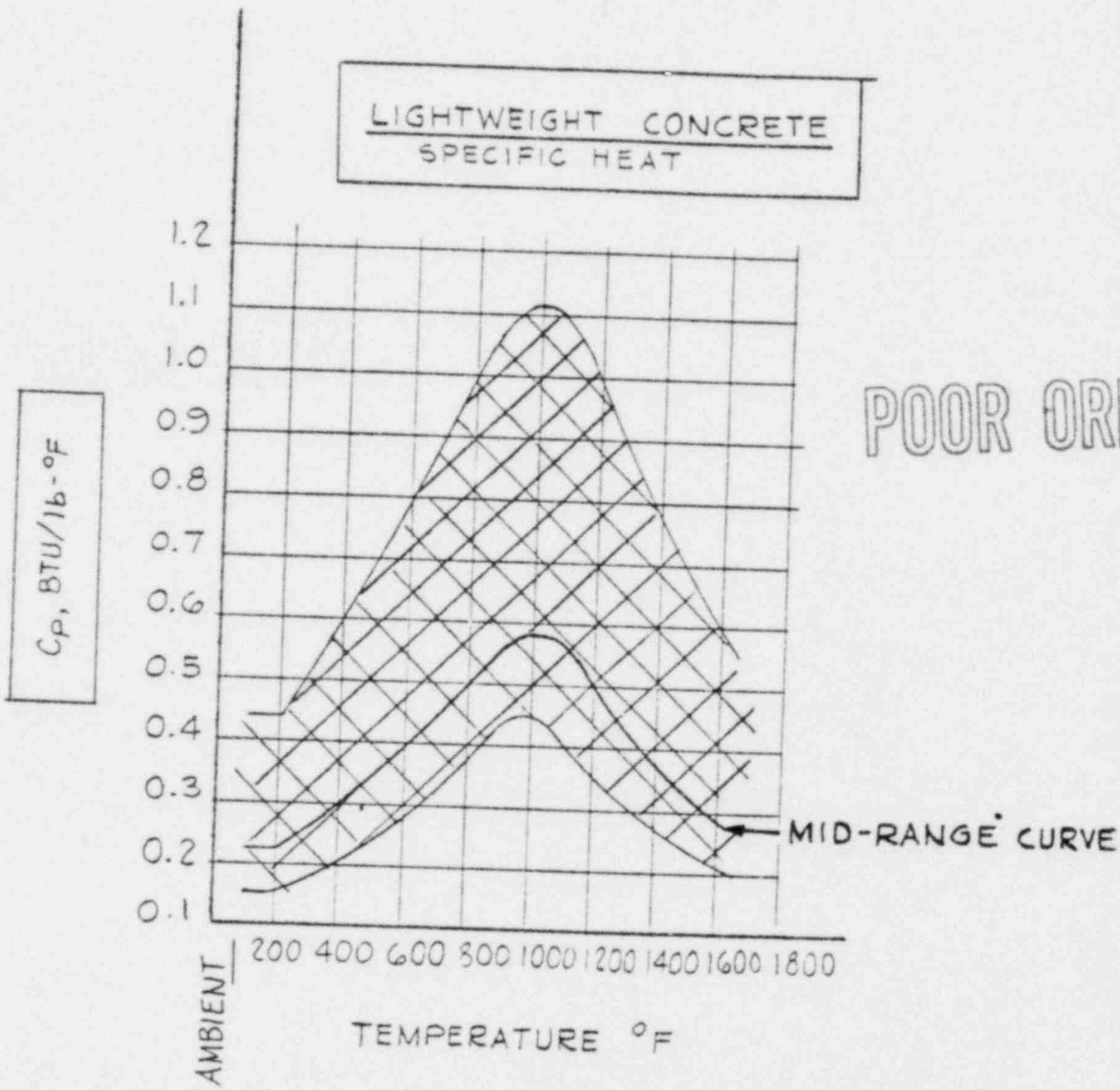

POOR ORIGINAL

NOTE: All CRBRP analyses shall be based upon the Mid-Range Curve values until completion of the DRS (Development Requirement Specification) testing programs.

59

Figure 3.8-B.4

A written welding procedure specification, containing information detailed in Form QW482 of ASME, Section IX, and the qualification test data containing the information detailed in Form QW483 of ASME Section IX shall be submitted to the Purchaser for review and approval.

A certificate of welder or welding operator performance qualification test shall contain the information as detailed in Form QW484 of ASME Section IX, and be available for Purchaser review if desired.

All welding repairs shall be made in accordance with a written welding procedure.

4.3 Studs and Anchors

The welding of the studs or anchors to the liner plates shall conform to the requirements of the ASME Code Division 2, Subarticle CC-4543.5 and ASME Section III, Division 2.

4.4 Storage, Conditioning and Handling of Welding Materials

45 4.4.1 Filler materials shall be stored, conditioned and handled in
accordance with the appendices of ASME Code - Section II, Part C which
are mandatory parts of this specification.

5.0 NONDESTRUCTIVE EXAMINATION REQUIREMENTS

5.1 Fuel Handling Cell

The full penetration cell liner seam welds of fuel handling cell will be examined in accordance with Article ND-5000 of the ASME Code, Section III, Division 1 requirements.

Radiography is a code requirement when the liner welds are accessible,
however, this is not feasible due to the method of construction. The entire
length of all liner seam welds shall be examined by the vacuum-box method
45 using either a bubble solution or a gas detection technique. All welds
shall be examined by magnetic particle or liquid penetrant techniques.

5.2 Cell Liners (except fuel handling cell)

Cell liner seam welds shall be full penetration and will be examined in accordance with Article CC5500 of the ASME Code, Section III, Division 2 requirements.

Radiography is a code requirement when the liner welds are accessible,
however, this is not feasible due to the method of construction. The entire
length of all liner seam welds shall be examined by the vacuum-box method
45 37 using either a bubble solution or a gas detector technique. All welds
shall be examined by magnetic particle or liquid penetrant techniques.

59

TABLE 3.8-B-2

CELL LINER BOUNDARY MATERIALS

	Application	Material	Supplemental Requirements
1.	Liner Plates (floors, walls ceilings)	ASME SA-516 Grade 55	55 ksi $\leq$ U.T.S. $\leq$ 65ksi
2.	Stud Anchors	ASTM A-108 Grade 1020	silicon content 0.15-0.30% U.T.S. $\geq$ 70ksi (7/8" Ø studs only)
3.	Structural Shapes (floor beams, embedments)	ASME SA-36	58ksi $\leq$ U.T.S. $\leq$ 71ksi

59 37 Note (U.T.S. = Ultimate Tensile Strength)

ATTACHMENT C

DESIGN BASIS LEAKS

To accommodate the effects of accidental sodium leaks or spills, the cell liners shall be designed for a design basis sodium spill. Based on the operating experience of existing sodium facilities and previous assessments of sodium spills, the amount and/or leakage rate of an accidental sodium spill into a cell are minor. Leaks which could develop from flaws, fatigue, creep, etc. are expected to be much less than the Upper Limit Design Basis leak as discussed below.

The design basis leak used as the basis for the inerted cell liner design is that leakage which would result from a four inch crack in the largest piping with the highest line pressure.

The following formula is used to determine leak volume:

$$\text{Design Leakage} = R_1 \cdot T_1 + R_2 \cdot T_2$$

where: R_1 = leak rate with pressure through four inch crack.

T_1 = time for detection plus time for operator action.

R_2 = leak rate without pressure.

T_2 = time required to stop the leakage.

The following times were used for operator action:

- Automatic - 0
- Single Operator Action in Control Room - 10 minutes
- Simple Operator Action in Control Room -20 minutes
- Operator Action Outside Control Room -30 minutes
- No Operator Action - Time to Drain System

Sodium spray is considered until such time as the driving pressure is eliminated.

A lesser leak considered is a 100 gram/hour (gm/hr) leak which is assumed to continue for 250 hours before the initiation of operator action. This leak establishes an upper volume limit of the
37 small liquid metal spill loads.

ATTACHMENT D

Justification of Cell Liner Strain Limits Under Large Sodium Spill Accident Conditions

59 D1.0 Strain Limits for Postulated Large Liquid Metal Spill (PLLMS) Loads

Under the PLLMS conditions to assure the integrity of the liner-anchor system design limits based on the effective von Mises strain are specified. They are the following:

Membrane: $\varepsilon_e \leq 0.50\ \varepsilon_\mu$

59 Membrane plus bending: $\varepsilon_e \leq 0.67\ \varepsilon_\mu$

Where

ε_e = Effective von Mises strain

ε_μ = Ultimate (uniform) strain of the material from uniaxial tensile test at the temperature considered (strain at ultimate load).

For carbon steel the ultimate strain at 900°F is 14%. This value
59 was confirmed in Reference 6 of Section 3A.8.

The von Mises or energy of distortion criterion is one of several theories proposed to predict yielding in a multiaxial state of stress based on the stress versus strain properties of the material provided by the uniaxial tensile test.

The von Mises yield criterion assumes that in a multiaxial stress condition yielding occurs when the energy of distortion has the same value as in the uniaxial tensile test when the tensile stress reaches yield.

Based on the equality of the energy of distortion in the multiaxial and uniaxial conditions, an expression can be derived for a representative stress as a function of the multiaxial stresses. This is called the von Mises effective stress. Similarly, an expression can be derived for a representative strain as a function of the multiaxial strains which is called the von Mises equivalent strain. The von Mises flow rule assumes that beyond yield, the effective stress and strain follow the stress-strain relationship obtained from the uniaxial test and based on this assumption the post yielding behavior including strain hardening can be introduced into the analysis.

The results of analyses based on the von Mises yield criterion and flow rule are for steel structures much closer to the results of experiments than those obtained from other criteria. The theory behind the von Mises yield criterion and flow rule is well documented in the literature (References 1 through 4) and has become a familiar and accepted procedure for application in the inelastic analysis of steel structures. Section III, Division 1 of the ASME B&PV Code accepts the yield criterion and associated flow rule based on the energy of distortion method (von Mises) for
plastic analysis of nuclear power plant components (Section F-1321.1.c). 37

D2.0 Background on the Strain Limit Criteria for the Design of Cell Liners for PLLMS

The strain limits for the design of cell liners for
59 PLLMS loads are presented in Section D.3.0. The effective von Mises
strain is not to exceed a constant fraction of the ultimate strain
obtained from uniaxial tension test. This criterion has been formulated
to preclude the possibility of liner rupture under extreme faulted
loads. Justification for the use of the criterion presented is discussed below.

A survey of the relevent literature indicates that no single criterion is available for characterizing the conditions associated with rupture of steels or other metals. The apparent ductility associated with ruptures is seen to depend not only on the properties of a metal, but also on the geometry of the particular structural system, on the nature of the multiaxial state of stress and on the characteristics of the loading. A survey of many earlier tests on yielding and fracture is presented in Reference 5. Some of the effects of combined state of stress are discussed in the above reference and, in particular, the influence of hydrostatic pressure on the apparent ductility is pointed out.

Although much experimental data related to the rupture of steels and other metals are available, it is obvious from the above remarks that not all of these data are relevant to the liner strain criteria in question. The liners are essentially subjected to a biaxial state of stress.

Important experimental data relevant to the liners in question
can be obtained from tests of thin tubes subjected to axial tension and
lateral pressure (References 10, 11 and 12). Many results of thin tube
tests, however, cannot be used because they are obscured by very significant anisotropy or by improper aspect ratios for the specimens. For
the relevent tube tests (References 10, 11 and 12), it is noted that
there is a general trend for the maximum strains and for the effective
von Mises strains to be significantly smaller when the failure occurs
with a rupture parallel to the tube axis. An apparent loss of ductility
is often related to plastic instability or necking (Reference 5, p.
259). Analytical predictions indicate the plastic instability criteria
to be quite different for the circumferential and axial directions of
59 the tubes (See Section D.5.0). It is also noted in Section D.5.0 that the
plastic stability criterion associated with relatively larger axial
stresses is identical to that for membrane stresses in structural systems
such as plates. For these reasons, it is apparent that rupture
data from experiments on tubes which fail with a longitudinal fracture are not directly related to the strain criteria for the
liners in question. However, for the conditions of the liner, circumferential rupture is more relevant. The effects of the biaxiality of
the stress field are noted in Table 3.8-B-3 for the tube tests in which
a circumferential rupture occurred. It is apparent that the effective
von Mises strains at rupture can be represented fairly well by a constant value for all the stress ratios indicated. In Reference 12, the results for a single test case for a tube having a ratio of principal stresses
equal to -1.0 is reported. The effective von Mises strain in this case falls
well in the range of those referred to in Table 3.8-B-3. 37

It is noted in Reference 14 (p. 573) that if equal longitudinal and circumferential stresses are present in a thin tube subjected to axial load and internal pressure, then the strains associated with plastic instability in that case are expected to be no more than one half of the strains expected in a circular, hydraulically loaded diaphragm. This fact suggests that the consideration of the tube rupture data in the development of the cell liner strain criteria is very conservative, since the cell liner behavior resembles that of circular diaphragms more closely than that of thin tubes. Nevertheless, such conservative considerations were used.

It is also shown in References 10 and 11 that for a particular material, the data for all the tube tests can be represented well with a single curve of true octahedral shear stress (or effective von Mises stress) versus natural octahedral shear strain (or effective von Mises strain). Therefore, the effective von Mises strain at rupture should (theoretically) and does (experimentally) correlate well with the energy of distortion at rupture (Reference 11, p. A-18).

Discussions in the above paragraphs lend empirical support to the use of the effective von Mises strain as a parameter in the strain limit criteria for the liners in question. On analytical grounds, there are several reasons for considering the effective von Mises strain in the criteria.

The well-accepted von Mises yield criterion assumes that in a multiaxial state of stress yielding occurs when the energy of distortion has the same value as in the uniaxial tensile test when the tensile stress reaches yield. Beyond yielding, the true stress versus natural strain curves are theoretically expected to be identical for uniaxial and multiaxial stress conditions when plotted in terms of the effective von Mises strains and corresponding effective stresses (Reference 13). This expectation is confirmed well by the experimental data from Reference 11.

As noted earlier, plastic instability (necking) has profound influence on the plastic behavior of a system. Analytically, the criteria for plastic instability can be readily formulated in terms of the effective
59 von Mises strains. This is discussed in Section D.5.0, in which a modified approach of Lankford and Saibel (Reference 14) is referred to.

A consideration of the plastic stability criteria for the A516 Grade 55 steel proposed for the cell liners is particularly significant. For this steel, the ultimate strain (i.e., uniform elongation) and total elongation are considered to be the same (Reference 15). Therefore, it is important to limit the strains for this steel to some values below those associated with the onset of predicted instability. For this purpose, the effective von Mises strain can be analytically shown to be
59 a relevant parameter (see Section D.5.0). 37

The above discussions indicate that the effective von Mises strain can serve, on empirical and analytic grounds as a significant parameter for imposing strain limits for the cell liners. Also, the particular characteristics of the A516 Grade 55 steel referred to above suggest that the ultimate strain from the uniaxial tests (at temperatures and strain rates representative of liner service conditions) may be used as a significant material parameter. It is noted in Section
59 D.5.0, that an analytical expression can be derived for the criterion of plastic stability in which the effective von Mises strain and the ultimate strain from uniaxial tests appear as parameters. This lends further support for using the latter parameter in the strain limit criterion for the particular steel proposed.

With the ultimate strain from uniaxial tests and the effective von Mises strain chosen as parameters for the strain limit criteria, there still remains the question of whether or not an additional parameter accounting for the effects of biaxiality of stress fields should be incorporated. As pointed out earlier, the data in Table 3.8-B-3 can be represented fairly well without any factor to account for the multiaxial stress field.

Davis and Connelly (Reference 6) have proposed a triaxiality factor which may relate to the triaxiality or biaxiality effects on ductility. In the particular case of transversely loaded discs, Riccardella (Reference 7) has demonstrated a good correlation between the effective von Mises rupture strains multiplied by the triaxiality factor and the reduction in area for uniaxial tension tests. However, if the ultimate strain from tensile test instead of the reduction in area were used, the correlation would not be good. In a recent paper (Reference 9), Davis himself has shown that the triaxiality factor is not relevant to all
59 studies. In Section D.4.0, a correlation of the triaxiality factor with the effective von Mises rupture strains is considered for the experimental data from the tube tests already referred to in Table 3.8-B-3. It is concluded that there is no advantage to using the triaxiality factor. Furthermore, it is of interest to note that no analytical justification for the use of the triaxiality factor has been offered in literature.

59 In Section D.5.0, a plastic stability criterion is presented for membrane strains for the case of biaxial tension. It is of interest to note in connection with the above criterion that for the ratio of the principal stresses considered in Table 3.8-B-4 the biaxiality effects vary with the stress ratio in a sense opposite to what an application of the triaxiality factor would yield for rupture. Plastic stability criteria, in whatever form they are expressed, may be more relevent in connection with the A516 Grade 55 steel proposed for the liners than in connection with the steels that have been used in the tube tests. Therefore, at best, there seems to be no advantage to using the triaxiality factor and, at worst, there may be a disadvantage to using this factor. Thus, the strain limit criteria for the design of the cell liners for the extreme faulted loads has been formulated in terms of the effective von Mises strain, the ultimate strain from a uniaxial tension test and a constant factor to provide a safety margin and to account for some uncertainties. 37

59 In the strain limit criteria presented in Section D.3.0, no distinction is made between tensile and compressive biaxial loads. It is usually believed that hydrostatic pressure improves the apparent ductility of a system (References 8 and 9). Although the applicability of this notion to biaxial states of stress is not self-evident, it is also expected to improve the apparent ductility in such a case. However, the same strain limits are used conservatively for biaxial compression and for biaxial tension.

From experimental data on rupture of transversely loaded discs (Reference 7), it is seen that the effective von Mises strains are approximately 31 to 58% higher for ruptures involving combined membrane and flexural strains than they are for membrane strains alone. Therefore, an allowable strain for the former case equal to 4/3 times the allowable strain for the latter case appears reasonable.

The constants appearing in the strain limit criteria (Section
59 D.3.0) were chosen so as to allow a "factor of safety" to account for inconsistencies in the observed experimental data and for some uncertainties in the state of art. In terms of the relevant experimental data considered, this factor is at least 1.33. It is also noted that the strain limit for membrane strains provides a "factor of safety" of approximately 2.0 or more with respect to the strain predicted analytically (see
59 Section D.5.0) for plastic instability.

37

Table 3.8-B-3

Variation of Effective von Mises Rupture Strains with Stress Ratios (Based on Data from Tube Tests)

59 Approx. Ratio of Circumferential to Logitudinal Stress	Relative Effective von Mises Strain at Rupture, $\overline{\varepsilon}_e/\overline{\varepsilon}_0$	Reference
0	1.00	10,11,12
0.250	0.79	10
0.375	0.87	10
0.500	0.75	11
0.500	0.92	10
0.534	0.80	12
0.750	0.90	11
0.762	0.86	11

Note: Tests in References 10, 11 and 12 were made with copper, medium-carbon steel and S.A.E. 1020 steel, respectively.

$\overline{\varepsilon}_e$ is the von Mises strain based on natural strain, and

59 $\overline{\varepsilon}_0$ is the corresponding strain for the same material for the case of pure axial tension. 37

59 D3.0 Strain Limits for Design of Cell Liners for PLLMS Loads

The strain limits specified below for the design of cell liners for the PLLMS loads are expressed in terms of the
59 effective von Mises strain, ε_e and the ultimate strain, ε_μ of the material at the temperature considered. The ultimate strain ε_μ is to be obtained from uniaxial tension tests performed at temperatures and strain rates representative of the service conditions of the liners. The effective von Mises strain is

59 $$\varepsilon_e = \frac{\sqrt{2}}{3}\left[(\varepsilon_1-\varepsilon_2)^2+(\varepsilon_2-\varepsilon_3)^2+(\varepsilon_3-\varepsilon_1)^2\right]^{\frac{1}{2}}$$

59 where ε_1, ε_2, ε_3 are the total principal strains

The following strain limits are to be considered.

59 1) Membrane strains $\varepsilon_e \leq 0.50\ \varepsilon_\mu$

2) Membrane plus bending strains

59 $$\varepsilon_e \leq 0.67\ \varepsilon_\mu$$

It is noted that the limits imposed above are applicable to all multiaxial states of stress for the liners, regardless of whether tensile or compressive components are present. 37

59 D 4.0 Correlation of Triaxiality Factor with von Mises Rupture Strains for Tubes

For biaxial tension, the principal stress

59 $$\sigma_3 = 0$$

Let β be the ratio of the other two principal stresses. That is

59 $$\sigma_2 = \beta\sigma_1$$

Thus, the triaxiality factor

59 $$TF = \frac{2\,(\sigma_1 + \sigma_2 + \sigma_3)}{\left[(\sigma_1 - \sigma_2)^2 + (\sigma_2 - \sigma_3)^2 + (\sigma_3 - \sigma_1)^2\right]^{\frac{1}{2}}}$$

can be written as

$$TF = \frac{(1 + \beta)}{\left[(\beta^2 - \beta + 1)\right]^{\frac{1}{2}}}$$

The effective von Mises strain ε_e can be written in terms of the natural principal strains as

59 $$\bar{\varepsilon}_e = \frac{\sqrt{2}}{3}\left[(\bar{\varepsilon}_1 - \bar{\varepsilon}_2)^2 + (\bar{\varepsilon}_2 - \bar{\varepsilon}_3)^2 + (\bar{\varepsilon}_3 - \bar{\varepsilon}_1)^2\right]$$

In Table 3.8-B-4, a comparison of the inverse of the triaxiality factor with relative von Mises rupture strains for thin tubes is made. The tubes were subjected to axial tension only (when $\beta = 0$) or to axial tension and internal pressure. Only the ruptures with a circumferential fracture are considered here.

For this comparison, it is assumed that

59 $\sigma_3 \simeq 0$ as an approximation. 37

Table 3.8-B-4

Correlation of Triaxiality Factor

with Data on Rupture of Tubes

59 Approximate $\beta = \frac{\sigma_2}{\sigma_1}$	$\frac{1}{TF}$ (Approximate)	Relative von Mises Strain $\bar{\varepsilon}_e/\bar{\varepsilon}_{eo}$	Ref. No.
0	1.00	1.00	10, 11, 12
0.25	0.72	0.79	10
0.375	0.64	0.87	10
0.500	0.58	0.75	11
		0.92	10
0.534	0.56	0.80	12
0.750	0.52	0.90	11
0.762	0.51	0.86	11

Note: $\bar{\varepsilon}_e$ is the von Mises strain based on natural strains, and
59 $\bar{\varepsilon}_{eo}$ is the corresponding strain for $\beta = 0$ for the same material.

Tests in References 10, 11 and 12 were made with copper, medium-carbon steel and S.A.E. 1020 steel, respectively. 37

59 D5.0 Plastic Stability Criteria for Membrane Strains

The plastic stability criteria for flat plates and thin tubes subjected to biaxial tension are presented below. These criteria, which pertain to membrane strains only, are useful in the interpretation of experimental results on rupture, and they may serve as an upper bound guide for developing strain limit criteria for the design of cell liners.

The expressions stated below have been derived by using a modified version of the procedure given by Lankford and Saibel (Reference 14). It has been assumed that in the plastic instability region the elastic strains are negligibly small compared to the plastic strains, and that the material is incompressible in this plastic range. Based on experimental evidence (for example, see References 10 and 11), it is assumed that the octahedral shear stress based on true stresses is related to the octahedral shear strain based on natural strains by the equation

$$\tau_{oct} = k_1 \gamma^n_{oct}$$

where k_1 and n are material constants.

It can be shown that, analytically n is equal to the uniform elongation for a uniaxial tensile test (References 5 and 14). This is confirmed by experimental results (Reference 11).

The expressions stated below can be derived formally by using the above assumptions. This can be done by using either the Prandtl-Reuss flow laws or the associated flow rule with a plastic potential of the same form as the von Mises yield criterion. In either case, the derivation is based on an approach using strains increments.

The plastic stability criteria stated below are presented in terms of the effective von Mises strain, ε_e, the uniform elongation, ε_u, from uniaxial tension test, and a function of β, where

$$\beta = \frac{\sigma_2}{\sigma_1}$$

59 is the ratio of the principal stresses.

It is noted that the plastic stability criterion for a flat plate is identical to that for a thin tube only when the latter is subjected to a relatively large axial stress. 37

Plastic Stability Criterion for a Flat Plate

59 $\varepsilon_e < \varepsilon_\mu \quad \emptyset_1\,(\beta)$

where $\quad \emptyset_1\,(\beta) = \dfrac{2(\beta^2 - \beta + 1)^{\frac{1}{2}}}{(2-\beta)}$

Plastic Stability Criteria for Thin Tubes

$\varepsilon_e < \varepsilon_\mu\, \emptyset_1\,(\beta) \quad$ for $\quad \dfrac{\sigma t}{\sigma a} \leq 0.5$

$\varepsilon_e < \varepsilon_\mu\, \emptyset_2\,(\beta) \quad$ for $\quad 0.5 \leq \dfrac{\sigma t}{\sigma a} \leq 1.0$

$\varepsilon_e < \varepsilon_\mu\, \emptyset_3\,(\beta) \quad$ for $\quad \dfrac{\sigma t}{\sigma a} \geq 1.0$

59 where $\emptyset_1$ (β) has been defined above; where

$\emptyset_2\,(\beta) = \dfrac{2}{3\beta}\,(\beta^2 - \beta + 1)^{\frac{1}{2}}$

$\emptyset_3\,(\beta) = \dfrac{2}{3}\,(\beta^2 - \beta + 1)^{\frac{1}{2}}$

and where σa is the axial stress in the tube and σ_t is the circumferential
59 stress in the tube.

It is noted that

$\emptyset_1(\beta) \geq 1.00$

$\emptyset_2(\beta) \geq 0.667$

$\emptyset_3(\beta) \geq 0.577$

$\emptyset_1$ (β) is valid for any stress ratio in plates and for the range of stress ratios shown above for tubes; $\emptyset_2(\beta)$ and $\emptyset_3(\beta)$ are valid for the ranges of stress ratios shown above for tubes. 37

The following basis will be used in the seismic analysis of typical cable tray supports:

a. All Class IE cable tray supports will be designed to meet the requirement by dynamic analysis using the appropriate seismic response spectra.

b. The support system will be designed to exclude all natural frequencies in a band covering the peak or peaks of response-spectrum curve.

c. Maximum stress will be limited to 90 percent of minimum yield to compensate for effects of higher modes and minor inaccuracies in method of analysis.

The design of typical instrument racks and supports for the instrument tubing is based on the attainment of a fundamental natural frequency of more than 20 Hz so that the floor seismic input will be transmitted through the support without amplification.

The equipment will be analyzed as an assembly that simulates the intended service mounting, thereby accounting for possible amplification of the seismic input by the equipment support.

Seismic documentation submitted by the vendor will be reviewed by the design engineering group to ensure that the support system has been considered.

Where it is necessary to test individual devices (e.g., relays or instruments) separate from the panel on which they are mounted, the acceleration of the panel at the device locations will be checked to ensure a level less than that at which the devices are qualified.

The structures that will be seismically qualified as Seismic Category I are listed in Section 3.2, Table 3.2-1. The Class 1E equipment is listed
59 in Table 3.2-3. 1

3.10.2 Analysis, Testing Procedures and Restraint Measures

The seismic qualification of safety related instrumentation will be
59 performed and documented as specified in Reference 13 of PSAR Section 1.6. 1

Category I electric equipment such as battery racks, instrument racks, and control consoles located in Category I structures will be supported and restrained to resist uplift or overturning resulting from seismic forces. 1

TABLE 3.10-1

LIST OF CLASS IE ELECTRICAL POWER SYSTEM EQUIPMENT

Equipment	Number in Plant	Location	Expected Method of Seismic Qualifications
4.16 KV Auxiliary Power System			
4.16 KV Class IE Switchgear	2	DGB	The switchgear will be seismically qualified by testing*
4.16 KV/480 V Class IE Unit Sub-Station	4	DGB (4)	The unit sub-stations will be seismically qualified by testing*
480 Volt Auxiliary Power			
480 V Motor Control Centers	8	RSB (2) SGB (4) CB (2)	The motor control centers will be seismically qualified by testing*
480 Volt Vital Power System			
Inverter System Components	**1**	**CB**	**The inverters will be seismically qualified by testing.***
120 Volt Vital Power System			
Inverter System Components	3	CB	The inverters will be seismically qualified by testing*
120 V Vital Instrument Power Board	3	CB	The instrument power board will be seismically qualified by testing*
250 Volt DC Diverse Power System (Division 3)			
250 V DC Batteries	**1**	**CB**	**The battery performance under seismic conditions will be predicted by analysis. A similar prototype battery rack will be tested by the manufacturer under simulated seismic condition.**
250 V DC Battery Chargers	**2**	**CB**	**The battery chargers will be seismically qualified by testing.***
125 Volt DC Power System			
125 V DC Batteries	3	CB	The battery performance under seismic conditions will be predicted by analysis. A similar prototype battery rack will be tested by the manufacturer under simulated seismic condition.
125 V DC Battery Chargers	6	CB	The battery chargers will be seismically qualified by testing*

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TABLE 3.10-1 (CONTINUED)

LIST OF CLASS IE ELECTRICAL POWER SYSTEM EQUIPMENT

Equipment	Number in Plant	Location	Expected Method of Seismic Qualifications
125 V DC Battery Distribution Boards	3	CB	The battery distribution boards will be seismically qualified by testing*
Standby AC Power Supply System			
Diesel Generators	2	DGB	The Diesel Generators will be seismically qualified by Dynamic Analysis.
Diesel Control Panels	2	DGB	The control panels will be seismically qualified by testing*
Electrical Penetrations			
(Details to be provided in the FSAR)		RCB	

*Test reports of prototype or previously built equipment proven to be of similar construction and/or dynamic analysis based on previously qualified equipment will be acceptable.

Notes: Motors on Class I driven equipment are included with the driven component listed in Table 3.2-2

SGB - Steam Generator Building
RSB - Reactor Service Building
DGB - Diesel Generator Building
RCB - Reactor Containment Building
CB - Control Building

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TABLE 3A.1-2

ACCEPTANCE RANGES FOR IMPURITIES AND PRESSURES IN CELL ATMOSPHERES DURING NORMAL OPERATION

		Inerted RCB Cells	Inerted RSB Cells	RAPS and CAPS Cold Box Cells	Fuel Handling Cell	
	Oxygen					1
	(Volume %)	0.5 - 2.0*	0.5-2.0*	20	0.005 ± 0.0025	1
	Water Vapor					
47	(Volume ppm)	<8000	<8000	<14,600	50 ± 25	1 23
	Radioactivity					
	(μCi/cc)	<0.001	<0.001	<0.008	0.001	1
	Cell Pressure**					
59	(in H_2O)	-10 to +10	-10 to +10	-10 to +10	-10 to +3	

* Tentative Values

** Control range; the operation specification may require control within the quoted range.

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TABLE 3A.1-3

INNER CELL DESIGNATION LIST

REACTOR CONTAINMENT BUILDING

Sheet 1 of 3

	CELL NO. (1)	TITLE	FLOOR ELEVATION	RADIATION ZONE (2) OPERATION	SHUTDOWN	DESIGN PRESS. PSIG	DESIGN (4) TEMP. (°F)	OPERATING TEMP. (°F)	NORMAL (3) ATMOS.	EQUIPMENT CONTAINED
45	101A	Reactor Cavity **	740'	V	V	35	180	120	N	Reactor Vessel, Guard Vessel
	101C	Flowmeter Cell (PHTS Loop #1)	786'-3"	V	V	35	180	120	N	PHTS Piping Perma-nent Magret Flowmeter
	101D	Flowmeter Cell (PHTS Loop #2)	786'-3"	V	V	35	180	120	N	PHTS Piping Perma-nent Magnet Flowmeter
	101E	Flowmeter Cell (PHTS Loop #3)	786'-3"	V	V	35	180	120	N	PHTS Piping Perma-nent Magnet Flowmeter
	102A	Overflow and Primary Na Storage Tank Cell	733'-0"	V	IV	10	180	120	N	Primary Na Storage Vessel, Reactor Overflow Vessel, Na Drain Tank
	102B	System 81 Reactor Cavity Piping Penetra-tion Area	782'-0"	V	IV	10	180	120	N	System 81 Pri. Na Overflow and Make-up lines from Reactor Vessel
44	103	Primary Na Make-up Pump Cell	733'-0"	V	IV	10	180	120	N	Primary Na EM Makeup Pump 3

3A.4.2.2 Auxiliary Coolant Systems

The decay heat generated in the EVST sodium is removed by a Na-Nak heat exchanger. The Nak, in turn, releases its heat through an air heat exchanger. These heat exchangers, Na-Nak are located in the RSB. A complete, redundant system of heat exchangers is present, as a standby in the event of a failure. The redundant systems and their
39 accompanying lines are made independent of each other by physical barrier separation. The FHC is cooled by the Recirculating Gas Cooling System, which, in turn, gives up its heat in gas-Dowtherm J heat exchangers. Section 9.7 presents more details of this system.

3A.4.2.3 Deleted

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39

3A.4.2.4 Heating and Ventilation

The Heating and Ventilation System provides for air-conditioning and ventilating the plant atmosphere. Section 12.2 presents more details of this system.

3A.4.2.5 Sodium Fire Protection

The Sodium Fire Protection System provides the means of detecting, alarming, containing, and controlling sodium and/or NaK fires.
38 Details of the SFPS are described in Section 9.13.

3A.4.2.6 Recirculating Gas Cooling

The Recirculating Gas Cooling System provides cooling of the inerted cells and is described in detail in Section 9.16.

47

17

3A.4.2.7 Reactor Refueling

The Reactor Refueling System performs all handling operations on
44 core assemblies destined for the reactor, from the new fuel shipping and receiving to the installation of the spent fuel shipping cask into the railroad flat car. Its functions are performed in the RSB and RCB. Section 9.1 presents details of this system.

The three major areas occupied by this sytem within the RSB-hardened portion, and the maximum radioactive sources contained therein, are as follows:

1) Ex-Vessel Storage Tank - The EVST, located below the operating floor, will receive, hold and cool all core assemblies discharged from the reactor vessel prior to shipment offsite.

2) Fuel Handling Cell - The FHC is a subfloor hot cell which can hold up to three core assemblies for inspections, measurements and transfer to spent fuel shipping casks for shipment offsite.

3) Two New Fuel Unloading Stations - Each station will contain one new fuel assembly in a new fuel shipping container.

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3A.4.2.8 Nuclear Island General Purpose Maintenance Equipment

The Nuclear Island General Purpose Maintenance Equipment System provides the capability for maintenance of the Nuclear Steam Supply System (NSSS). These maintenance operations are accomplished within the Reactor Containment Building, the Reactor Service Building, and the Steam Generator Building. The system provides general purpose equipment for removal and replacement of radioactive and/or sodium service components of the Fuel Handling System, Heat Transport System, Auxiliary Systems, and Reactor Systems. Capability is also provided for sodium removal and decontamination. The system also provides the general purpose equipment used for the removal, repair, maintenance, and reinstallation of equipment and components housed within the Nuclear Island.

59

3A.8 Cell Liner Systems

Cell liners are located in Na and NaK cells in order to maintain the inert atmosphere (Nitrogen) under normal operating conditions, to facilitate decontamination and decrease downtime following an accidental sodium spill and to protect the structural integrity of the cell for the preservation of the capital investment.

3A.8.1 Design Bases

See Section 3 of Appendix B to Section 3.8 for a discussion of cell liner requirements.

37 3A.8.2 Design Description

The CRBRP cell liner system utilizes a steel plate liner, including edge embedments, to restrain the thermal inplane expansion - contraction forces, and stud anchors (in walls and ceilings) and structural sections (in floors) to provide out of plant restraints to minimize the effects of bending caused by behind the liner pressure or by the buckling of the plate. Additional features of the cell liner design include an integral insulation panel to, protect the structure from severe thermal loading, and an integral vent system, to relieve the behind the liner steam buildup under
45 sodium spill accident conditions. The cell liner system consists of the following typical wall and ceiling panel elements:

1. Carbon steel liner plate - to provide a leaktight sodium spill boundary

2. Insulating Concrete panel - to protect the structural concrete from the elevated temperatures resulting from the 'Na' spill

3. Nelson Steel Anchors (wall/ceilings) or Embedded Structural Sections (Floor) Welded to Liner Plate - to minimize the out of plan bending of the cell liner plate

4. Continuous Air Gap Between Insulating Concrete and
59 Liner Plate - To vent and relieve the buildup of gas (steam) pressure behind the liner due to the heating of the insulating and structural concrete

45 The typical wall and ceiling panels are prefabricated in large
modular-panel sections. This will minimize the amount of field welding
required. The stud anchors extend deep enough into the concrete structure
45 so that the integrity of the wall/ceiling liner system is maintained and the
59 full strength of the stud anchor is developed. At the corners of the liner
plates are attached to steel sections embedded in the structural concrete in
order to prevent the in plane thermal expansion of the liner plate and
the excessive strains in the anchors that would result.

The typical floor liner is a carbon steel plate welded to rolled
45 steel sections which are embedded in the concrete slab such that approx-
imately half their depth projects above the top of the floor slab. The
floor liner will be supported on embedded steel sections. The space
37 between the floor slab and the liner is filled with a precast insulating
59 45 concrete panel. The function of the floor insulation is to:

a) Provide an insulating barrier between the steel floor liner plate and the structural concrete such that the temperature of the concrete floor slab will not exceed the limits specified in Section 3.1.7 of Appendix 3.8-B.

b) Provide a vent path for the gasses generated beneath the floor liner as a result of a sodium spill.

c) Limit the deformations of the floor liner plates under a positive pressure differential by transmission of the internal cell pressure through the insulation to the structural slab. 37

59 The insulating concrete floor panel is assumed to provide no
lateral support to the embedded steel sections. The size and spacing of
the embedded rolled steel sections are designed such that stresses and
strains in the beam web and the liner plate fall within the limits spe-
cified in Table 3.8-1 of Appendix 3.8-B. 37

A liner vent system will be installed to limit the pressure be-
hind the liner generated by the heatup of structural concrete during a
sodium spill. The liners will be designed to withstand the pressure
45 37 under the maximum liner temperature.

The steam generated below the floor liner by the heat up of the
structural and insulating concretes will be vented through the air gaps
provided as shown in Figure 3A.8-4 and through holes in the webs of the
floor liner embedment beams to collective points along the periphery of the
cell. The steam generated below the reactor cavity floor liner is vented
independently of the cell walls due to the need for compartmentalization of
the cell liner vent system to satisfy TMBDB requirements. Each zone in the
59 reactor cavity is vented independently and is separated by a baffle plate
from the adjacent zone. In areas other than the reactor cavity, the steam
from the floors will be released with the steam from the walls and ceilings
into the liner vent system piping. Effects on stiffness caused by liner
corrosion will be accounted for in the liner plate/anchors analysis.
Equipment supported on the floor liner will be provided with special supports
to transmit the loads directly to the structural slab. During construction
and maintenance the floor liner will be protected from loading as specified
in Section 3.1.1 of Appendix 3.8-B. Diagrams of the cell liner configurations
47 are shown in Figures 3A.8-4, 3A.8-5 and 3A.8-6.
48

The vent path for the cell liner wall and ceiling system is
provided by a 1/4" continuous air gap as shown in Figures 3A.8-4 and
3A.8-5. The air gap is prefabricated with the modular cell liner panel. 37
The air gap between the insulating concrete and the cell liner plate will
48 be inspected before installation. The continuity of the air gap is maintained
during construction and the life of the plant by a) sealing the joints between
adjacent insulating concrete panels during construction to prevent the
48 entrance of the structural concrete; b) the bearing and/or bond strength
of the insulating concrete panel to the cell liner plate at the stud
anchors to prevent the closure of the air gap during construction and
59 installation; c) the preoperational testing of the cell liner vent system.
Local plugging of the air gap is precluded since the air gap
is continuous over the entire surface area of the lined cell. Therefore, 37
there are no effects on the liner or liner anchors due to pressure buildup.

Liners will not ordinarily be exposed to sodium. The structural
48 concrete will be protected by an insulating concrete between the steel
liner and the structural concrete. During accident conditions, some
spalling of this non-structural concrete insulation may occur. However,
this is considered acceptable since liner failure due to spalling of
the insulating concrete is prevented by embedding liner anchors into
the structural concrete.

The inner cells are constructed reinforced concrete with steel
liners designed to maintain the atmospheric leak tightness requirement of
section 3.1.1.1 of Appendix 3.8-B during normal operating conditions. Piping
penetrations entering inerted cells are designed to prevent leakage, and
59 are sealed by any of the following methods, depending upon individual design
requirements:

59 a) packing between pipe and a pipe sleeve which is
welded to the cell liner using full penetration welds.

b) flued head or flexible bellows attachments welded
to pipe and pipe sleeve with sleeve welded
59 37 to cell liner using full penetration welds.

c) pipe embedded in concrete and welded directly to cell liner with full penetration welds.

Penetrations between inerted cells having a common atmosphere may also consist of an open pipe sleeve which is welded to the cell liner
59 at each face with full penetration welds.

3A.8.3 Design Evaluation

The piping integrity investigation analysis of crack growth due to all design duty cycle events indicated negligible crack growth. Based upon this evaluation, it is concluded that no leaks will occur under operation in accordance with the piping design specifications.

The sodium and NaK components and piping in the CRBRP nuclear steam supply system and auxiliary systems are all designed to prevent leakage. The liners in cell which contain sodium or NaK should therefore not be exposed to any conditions more severe than those corresponding to normal plant operation. However, accidental sodium leaks or spills cannot be precluded and therefore
45 must be considered in the design of the liners.

To accommodate the effects of accidental sodium leaks or spills, the cell liners will be designed for a design basis sodium spill. Based on the operating experience of existing sodium facilities and previous assessments of sodium spills, the amount and/or leakage rate of an accidental sodium spill into a cell are minor. Leaks which could develop from flaws, fatigue, creep, etc. are expected to be much less than the Design Basis Leak as discussed in Reference 2 of Section 1.6.

3A.8.3.1 Sodium Spill Evaluation

The evaluation of the consequence of sodium spills is provided in PSAR Section 15.6. The method and criteria for evaluation of the cell liners are discussed in Section 3.8-B.

3A.8.3.2 Brittle Failure Potential of the Liner in Irradiated Areas

The increase in ductile-brittle transition temperature due to neutron damage is estimated to be less than 10oF for the reactor cavity liner. This is based on damage function analysis, which indicates that
37 the damage level for the neutron spectrum in the reactor cavity will be approximately 100 times lower than that for LWR reactor vessels.

For the neutron embrittlement evaluation of the cell liner plate, the methods and limits established by USNRC Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials" will be used. The only area of the plant exposed to neutron fluence is the reactor cavity. By considering the worst case exposure condition for the reactor cavity cell liner where the maximu fluence is 7.8×10^{18} n/cm^2, (E >0.1 M_eV) and 6.1×10^{13} n/cm^2 (E <1.0M_eV), the maximum adjustment of Nil-ductility temperature (NDT) is 10^oF and does not require trace element control. This indicates that the liner steel is not effected by neutron
59 embrittlement nor does gamma radiation result in steel degradation.

3A.8.3.3 Liner Analysis

.1 General

The liner system is described in Section 3A.8.2. The Design Requirements, Load Categories, Load Combinations, Stress and Strain allowables and Design Analysis procedures are given in paragraphs 3.1 through 3.5 of PSAR Appendix 3.8-B. Attachment D to Appendix 3.8-B gives the bases for the strain criteria and strain limits adopted for the Postulated Large Liquid Metal Spill (PLLMS) Loads.

The spacing and size of the Nelson stud anchors in the wall and ceiling panels and of the floor anchors are designed such that the stresses and strains fall within the limits specified in Table 3.8-B-1 of Appendix 3.8-B.

The anchors will resist the shear forces induced when unbalanced forces exist between sections of the liner and axial forces caused by the maximum specified pressure (5 psig) acting on the backside of the liner under the PLLMS loads. Since there is a 1/4 inch gap between the cell liner and the insulating concrete, some axial loads in the anchors will be caused by the cell's internal pressure.

The insulating concrete does not act integrally with the structural concrete and a bond breaker will be provided on the surface separating the two materials to reduce shear transfer. The insulating concrete is not considered a main structural element; its main function is to provide a thermal shield to prevent degradation of the structural concrete under the elevated temperatures of the PLLMS conditions. The adequacy of the insulation thickness has been demonstrated by a preliminary finite element thermal analysis using the computer program ANSYS. The temperatures calculated at the face of the structural concrete did not exceed the limits established in Section 3.1.7 of Appendix 3.8-B. Local hotspots due to heat transfer into the structural concrete through the studs may occur. These effects will be evaluated by both analytical and testing methods.

Spalling or degradation of the insulating concrete under the PLLMS Loads will not cause a failure of the liners or liner anchor system. The anchors will be embedded in the structural concrete to ensure adequate restraint and the design is such that even if no lateral support to the anchors is provided by the insulating concrete, the specified anchor strain limits will not be exceeded.

45 | Liner failure due to behind the liner steam pressure is prevented by the provision of a venting system on the backside of the liner where necessary, to reduce steam pressure generated from heatup of the insulating material and structural concrete during a sodium spill. Specific vent paths behind the liner will be provided where analysis and/or testing indicates
37 they are required. Steam produced would be vented to the non-inerted areas

within the RCB or RSB consistent with the location of the sodium spill.
Preliminary analysis under Postulated Large Liquid Metal Spill (PLLMS)
conditions indicates that this venting scheme will not require the
containment to be purged. A more detailed analysis will be performed to
verify these preliminary indications. Since any steam produced during a
59 sodium spill would be vented to the non-inerted areas, hydrogen evolution
due to sodium/water reactions would occur only following a liner failure.
Failed liner testing is planned and the amount of hydrogen evolved
during these tests will be monitored. Even in the unlikely event of the
liner failure, purging of containment is not expected to be required.
The liner system will be designed to withstand a backside pressure of 5
psig.

Due to the magnitude of the compressive thermal forces caused
by the restraining actions of the concrete structure, buckling of the
liner plates is anticipated. Buckling in itself will not produce failure
since the thermal deformations are self limiting. However, due to the
reduced load carrying capacity of a buckled panel, unbalanced lateral
forces can be induced at the anchor. The liner-anchor system will be
designed such that under the unbalanced lateral forces due to panel buckling,
59 the strains will not exceed the allowable limits. Buckling of panels
will improve the stress-strain conditions at the corner anchors since
the unbalanced lateral forces will be reduced.

The dead and live loads, seismic loads, operating pressure and thermal loads, etc., will affect the cell liners through the interaction of the liner-anchor system with the structural concrete. Since the structural concrete is by far more rigid than the liner, the deformations of the concrete under these loads and the restraint it provides to the liner will determine the stress-strain condition of the liner-anchor system for these loads. For these conditions other than sodium spills, the stress levels in the cell liners are expected to be below the yield strength of the material. The maximum normal operating temperature (peak) will not exceed 180^oF and no significant stresses and strains will be imposed on the liners under these conditions.

The cyclic temperature variation in the cells during the lifetime of the plant (10 cycles from 70^o to 140^oF, 100 cycles from 100^o to 140^oF and 100 cycles from 140^o to 180^oF) are within the ASME Code limitations such that the cyclic fatigue should not be a problem. Based on Section NE-3222. 4d of Section III, Division 1 of the ASME B&PV Code, for the specified temperature ranges and number of cycles, no fatigue analysis is required.

.2 Analysis

45 37 Calculations have been conducted to investigate the adequacy
of the liner-anchor system under the PLLMS Condition. They consist of
elasto-plastic analyses using the computer program ANSYS. The strain values
obtained from the finite element analyses under sodium spill conditions
are compared against the allowable strains at the exposure temperature. The
allowable PLLMS strains are determined using Table 3.8-B-1 and the materials test
data presented in 3A.8.4. Table 3A.8-1 summarize the allowable strains under
59 load combination D (PLLMS spill).

An analysis conducted at ORNL (Ref. 1) considered a wall liner panel, 15 inches square. It was assumed that the corners, where the stud anchors are located, were rigidly supported, which is justified when there are no unbalanced lateral forces acting on the anchors. The analysis considered transient conditions immediately after the sodium spill, an isothermal condition under a steady state of 1000°F and a cooldown to an
37 isothermal condition of 150°F. The maximum calculated strain was 1.7%.

To investigate the cell wall liner-anchor system, a finite element analysis of a typical cell liner panel having a 1/8" bow at the middle of the panel was performed. A typical 75" x 75" portion of the wall liner having a line of symmetry along the edges and 1/8" bow at the center span was considered to determine the effect of an unbalanced lateral force on the stud anchors. By using symmetry, the model was reduced to a one eighth segment (Figures 3A.8-1 and 3A.8-2). The liner studs were modeled at 15" on center. The insulating concrete was assumed to provide full lateral support to the studs. A behind the liner pressure of 5 psi was included.

The cell liner (plate) temperature was raised prototypically from 70°F to 1000°F and the allowed to cool down gradually to 200°F. The cell liner strains and displacements were observed at different stages. At 1000°F the maximum strains in the bowed panel were 2.31% while the maximum strains in the unbowed "typical panel" were 1.5%. The result of these calculations show the maximum effective strain in the liner system well below the allowable
59 limits presented in Table 3A.8-1.

Another analysis included a bi-planar corner (wall to floor). The mathematical model is shown in Figure 3A.8-3. A 15 inch wide strip (equal to the spacing of the stud anchors) was considered. Two models were used: in one the insulating concrete layer was assumed to provide the full lateral support to the stud anchors; in the second it was assumed that the insulating concrete layer provided no lateral support. In both models
59 it was assumed that the insulating concrete under the floor liner provided no lateral support to the floor anchors. A 5 psi pressure was applied on
59 the back face of the liners (to simulate steam pressure buildup) and a uniform temperature of 1000°F in both liner and anchors. The results give a maximum effective strain of 2.3% in the liner plate and 1.7% in the anchors. There is no substantial difference between the results of the two models. Since the strains obtained are much below the allowables specified in
59 Table 3.8A-1, liner integrity is maintained. The effect of the Appendix 3.8B corrosion allowance of 1/16 inch was investigated using the same mathematical model described above. A strain response was obtained after reducing the thickness of the cell liner plates (both wall and floor) by 1/16 inch to account for the effect of corrosion while keeping the remaining parameters the same. The results obtained were of the
45 37 same order of magnitude as was obtained from the same model without corrosion.

This result was anticipated since the major components of the strain
59 induced in the liners are membrane strains. Accordingly it has been
concluded that the effect of a maximum 1/16" corrosion over a thirty
year plant life will not compromise liner integrity.
45

A finite element analysis was performed to determine the response of the cell liner wall/ceiling system in the close proximity of a large diameter penetration. The cell liner panel, penetration sleeve, and collar plate were evaluated under the PLLMS condition to verify the integrity of the cell liner system under sodium spill conditions.

The large diameter pipe penetration sleeve detail used in this analysis utilized a thickened plate collar welded to the penetration sleeve and the cell liner, for resisting the thermal expansion choking forces developed by the fixed cell liner plate at the penetration opening. The collar plate is anchored directly into the structural concrete.

The penetration analyzed consisted of a 24 inch diameter schedule 80 pipe sleeve reinforced with a circular stiffening collar with gusset plates spaced at 45° around the penetration. A cross-section of these elements is presented in Figure 3A.8-7. The penetration assembly was located in a wall/ceiling liner area having the standard cell liner configuration (Figure 3A.8-4).

The analysis performed considered tempratures ranging from normal operating to 850°F. An elastoplastic computer analysis was performed using one eighth symmetry as shown in Figure 3A.8-8. The analysis considered the insulating concrete not available to support the cell liner stud anchor lallerally, but available to support the liner plate in the event of local closure of the air gap at the penetration/liner interface.

At 850°F the maximum membrane strain of 2.02% was calculated at the liner interface with the penetration collar. The combined membrane plus bending strain at the same location was 2.14%. The maximum membrane stud strain adjacent to the penetration was 1.7% with membrane plus bending strain of 2.6%. These strains compare favorably with the allowable values of 7.6% membrane and 10.2% membrane plus bending strain at 850°F as presented in Table 3A.8-1.

A thermal analysis was performed to determine the consequences
of thermal shock if any on the 3/8" wall liner plate resulting from a
design basis sodium spill accident. A finite element model was utilized
which considered the sodium film coefficient, 3/8 inch liner plate, 1/4
inch air gap, 3-7/8 inch insulating concrete and 35 inch structural
concrete. The analysis assumed a one dimensional heat flow. The temperature
distribution through the thickness of the liner plate and the corresponding
59 time were required for the evaluation of thermal shock.

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Using an initial sodium temperature 1000°F at time t=0 hours and a final sodium temperature of 100°F at time t=60 hours, the temperature distribution through the thickness of the plate and the concrete were obtained from the finite element computer solution. It was observed that the max. temperature difference through the 3/8 inch thickness of the liner was 95.5°F at 3.12 seconds and that the difference was gradually decreasing. This resultant temperature difference is insufficient to cause any thermal shock. Moreover, during the heat up and cool down phase of the analysis, the liner was subjected to yielding and high strain as demonstrated in previous analyses. Accordingly, thermal shock is not considered to be a problem in the analysis of the liner. Similar results were also obtained in the case of a thermal shock analysis
45 performed on a floor liner.

The stud anchor yield force, displacement capacity and ultimate force and displacement capacity have been evaluated based on manufacturers test data.

59 The final analysis considered load categories including thermal shock, thermal transients, and the heat-up and cool-down of the liner under the sodium spill accident; hot spots; the effects of variations in the steel properties and thicknesses; the effect of the 1/16 inch corrosion allowance, etc.

The postulated breaks of sodium lines may generate hot sodium
37 sprays on the liner. The effects of the hot spot on the liner including
59 the dynamic effects, if any, of the jet impact will be considered.

3A.8.4 TESTING AND INSPECTION

45 3A.8.4.1 Development Testing Programs

A series of development testing programs have been developed to support the cell liner design. These programs provide materials data to support the objective of designing the cell liners to accommodate large sodium spills without failure, demonstrate through qualification testing that integrity of the liner is maintained under sodium spill conditions, and provide test materials data on sodium-concrete reactions to assess the consequences of cell liner failure.

Five individual testing programs have been completed or are ongoing in support of the cell liner design. These development programs are:

(a) Comprehensive Testing Program for Concrete at Elevated Temperatures

(b) Sodium-Concrete Reaction Tests

(c) Sodium Spill Design Qualification Tests

(d) Cell Penetration Sealant Tests

(e) Base Material Tests for Liner Steels

The tests included in the development programs listed above are modeled to minimize the difference between small scale tests results and the actual mass concrete response at elevated temperatures. The development programs indicated above are directed toward the goal of designing and testing a cell liner system which will not fail, even
59 under the unlikely event of a large sodium spill.

Comprehensive Testing Program for Concrete at Elevated Temperatures

45 This ongoing experimental program will define the variation with temperature of various physical and thermal properties of prototypic CRBRP limestone aggregate concrete and lightweight insulating concrete. The properties include, but are not limited to, compressive strength, modulus of elasticity, shear strength, bond strength, thermal conductivity, specific heat, and coefficient of thermal expansion. The series of experiments will be carried out at various temperatures including those
37 representative of accident conditions.

The results of this testing program can be directly applied to
37 the analysis of the building structures supporting the cell liners. The
testing program is nearing completion and the results will be included in
59 an ORNL/CRBRP report following completion.

Since the biaxial and triaxial testing of concrete at elevated temperatures will yield a greater compressive strength than uniaxial testing due to the influence of the lateral confining stress, the concrete tests performed on specimens in the uniaxial state of stress will
45 yield a more conservative value of strength. Therefore the consequences
59 of biaxial and triaxial loading can be disregarded.

Sodium-Concrete Reaction Tests

45 The objective of this ongoing program is to determine the rate and extent of penetration due to sodium-concrete reaction. The effect of reaction product accumulation and gas release on the sodium-concrete reaction rates will be determined to allow upgrading of analytical capability. Additionally, intentionally defected liner tests will be performed to assess the response of the liner to a sodium-concrete reaction.

The dimensions of the test articles have been selected to ensure that results representative of the actual mass concrete structure can be obtained.

37 Sodium Spill Design Qualification Tests

A large scale model of a CRBRP cell liner has been performance tested to demonstrate the ability of the cell liner system to maintain liner integrity, mitigate consequences of a large sodium spill, and prevent sodium-concrete reactions. A total of 3500 pounds of liquid sodium at $1100^{\circ}F$ was spilled against a CRBRP cell liner wall forming a 50 inch deep sodium pool above the CRBRP liner floor in the test article. The sodium pool was then heated, using electric heaters, to temperatures ranging between $1460^{\circ}F$ and $1580^{\circ}F$ and maintained until six days after the spill. The $1100^{\circ}F$ sodium spill simulated a Design Basis Accident sodium spill event and the subsequent heat up to approximately $1600^{\circ}F$ simulated the fission decay heat of a sodium pool under TMBDB Accident conditions.

The test data and post test examination revealed no failures or liner defects and minimal deformation of the liner system under the DBA and TMBDB spill conditions. The results of this testing program are
59 included in the HEDL final report (Reference 5).

Cell Penetration Sealant Tests

59 The objective of this completed program was to determine the effects of temperature, sodium and radiation on various candidate sealant materials for cell penetrations. This series of experiments enables
37 selection of the most suitable sealant material for use in the CRBRP.

Following selections of the prime sealant material, prototypic electrical cable penetration assembly performance testing were conducted. The results of this testing program were published in Reference (4).

Base Material Tests for Liner Steels

59 The objective of this completed testing program was to determine
the response of the cell liner plate material (SA-516 Grade 55) and its
associated weldment material to elevated temperatures up to 1700^{o}F.
The base liner steel will be tested for residual tensile strength (in-
cluding stress-strain response), stress-rupture (Creep) and thermal expansion.
59 The weldment material was tested for residual tensile strength (including
stress-strain response) and stress-rupture (Creep). Both longitudinal
59 and transverse welds were investigated. The results of the base liner
45 steel and weldment material tests have been published in Reference 6.

The material properties information at elevated temperatures
which was obtained in this program has been used in the design and analysis
59 of the cell liner system.

References:

1. McAfee, W.J., Sartory, W.K., "Evaluation of the Structural Integrity of LMFBR cell liners - Results of Preliminary Investigations", ORNL-TM-5145, January, 1976.

2. Chapman, R.H., ORNL-TM-4714, "A State of the Art Review of Equipment Cell Liners for LMFBR's", February, 1975.

3. Sartory, W.K., McAffee, W.J., ORNL-TM-5145, "Evaluation of the Structural Integrity of LMFBR Equipment Cell Liner - Results of Preliminary Investigation", February, 1976. 37

4. Humphrey, L.H., Horton, P.H., AI-DOE-13227 "Selection of a Sodium and Radiation Resistant Sealant for LMFBR Equipment Cell Penetrations", January 31, 1978.

5. Wireman, R., Simmons, L., Muhlestein, L., HEDL TME 79-35, "Large Scale Liner Sodium Spill Test (LT-1)", May, 1980.

59 6. Cowgill, M. G., WARD-D-0252, "Base Material Tests for Cell Liner Steels", February, 1980.

45

TABLE 3A.8-1

STRAIN ALLOWABLES FOR

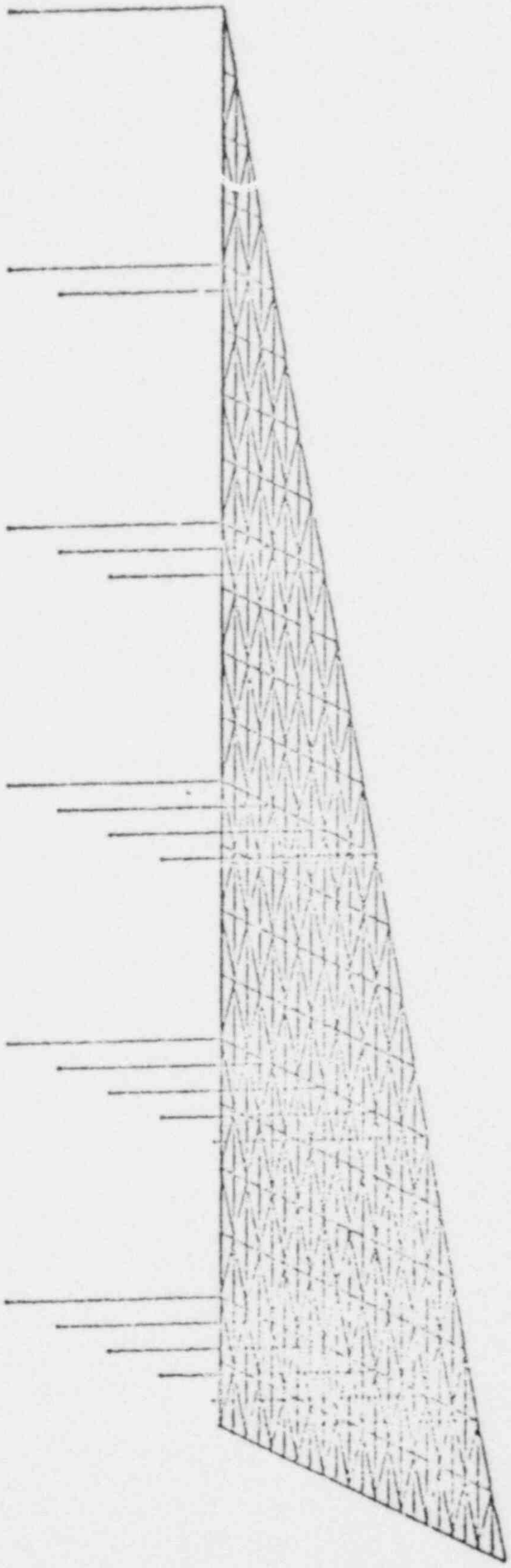
POSTULATED LARGE LIQUID METAL SPILL (PLLMS) CONDITIONS

LOAD COMBINATION D

Temperature ^oF	Membrane Strain $0.50\ \varepsilon_\mu$ in/in	Membrane + Bending Strain $0.67\ \varepsilon_\mu$ in/in
75	0.0955	0.1280
600	0.1185	0.1588
800	0.0815	0.1092
850	0.0759	0.1017
900	0.0703	0.0942
1000	0.0590	0.0791

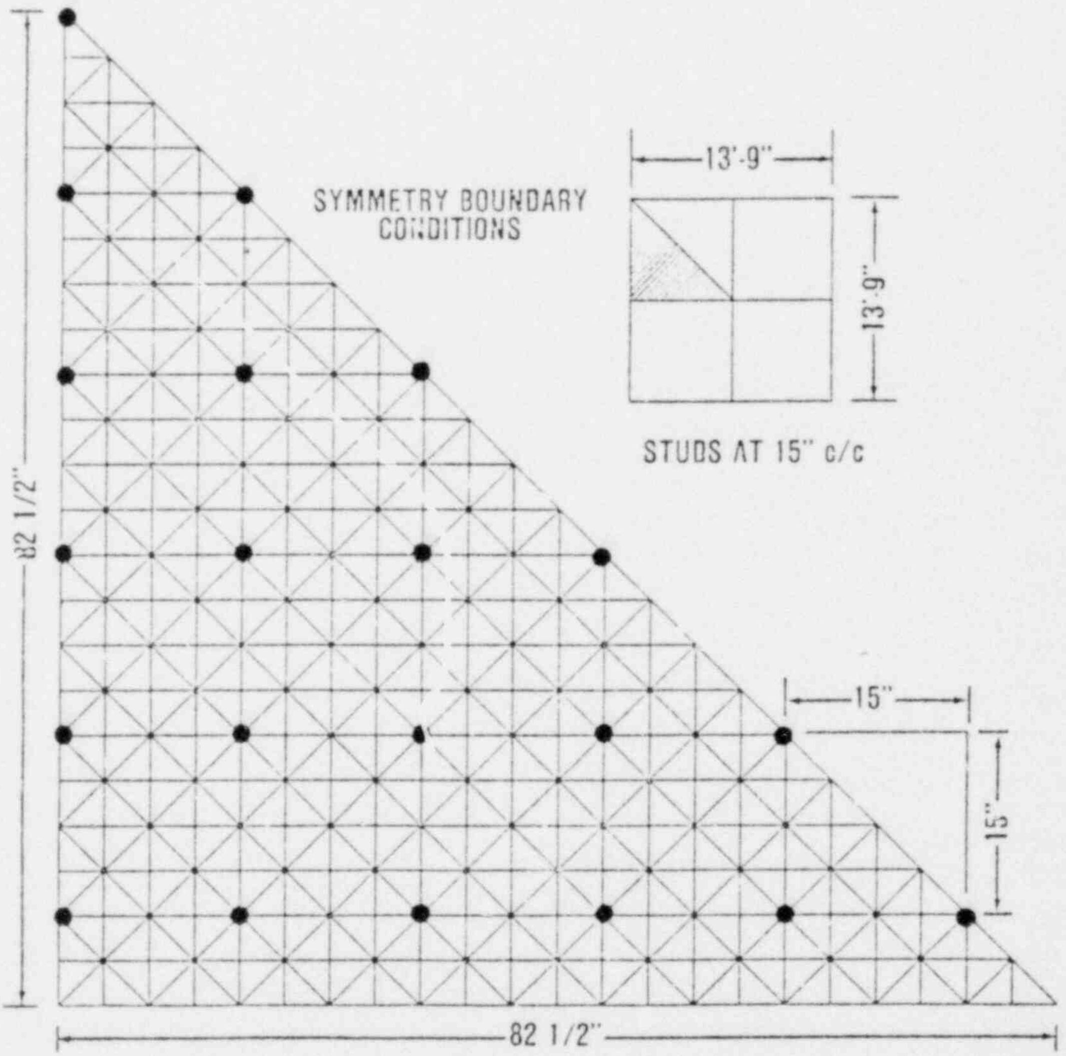
Where:

ε_μ = Uniform elongation or the strain at ultimate load as obtained from the elevated temperature testing of SA 516
59 Grade 55. Liner Steel.



POOR ORIGINAL

Figure 3A.8-1 ANALYSIS OF LINER WITH STUD-TYPE ANCHORS



CRBRP: ANALYSIS OF LINER WITH STUD-TYPE ANCHORS

FIGURE 3A.8-2

Amend. 59
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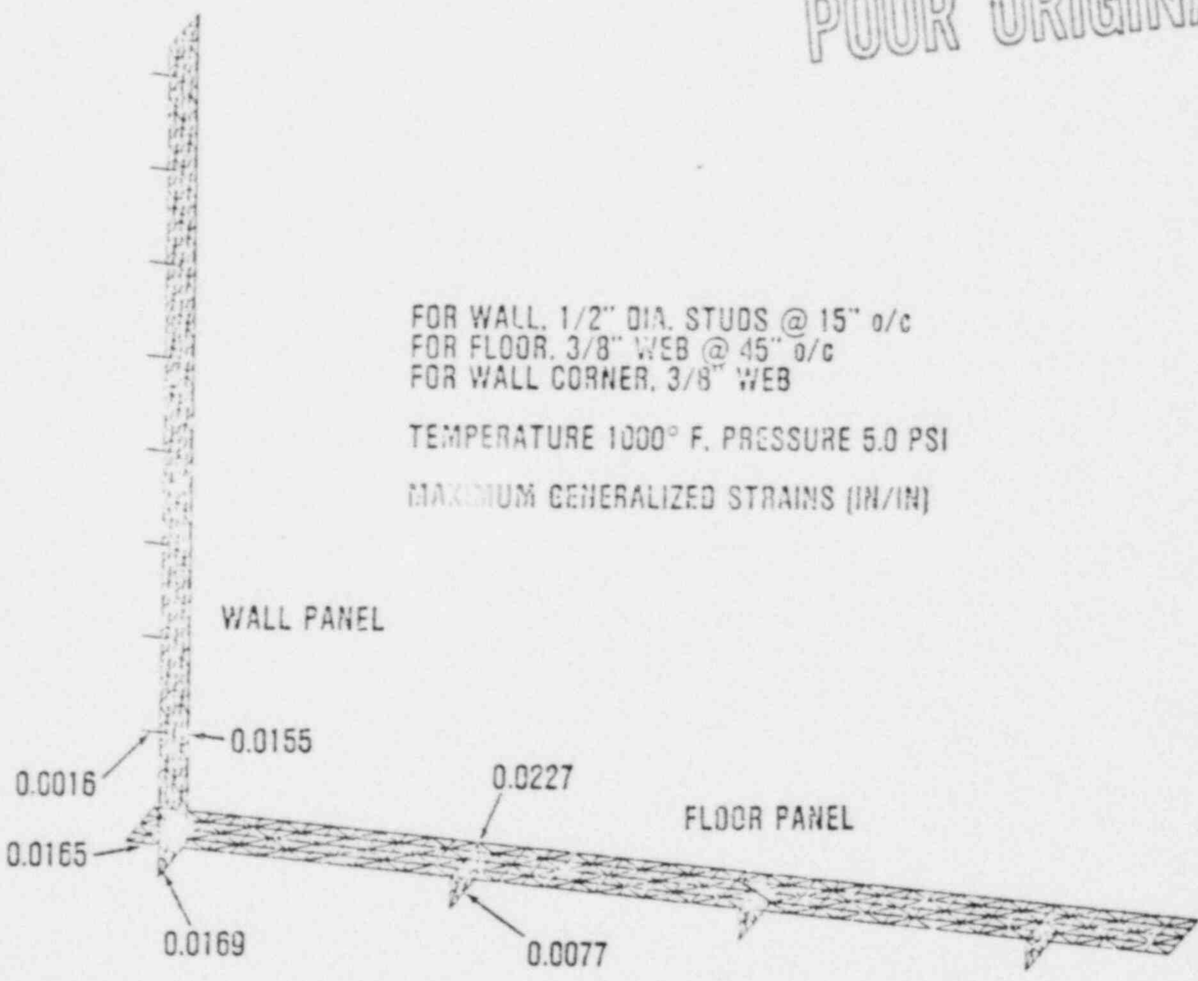
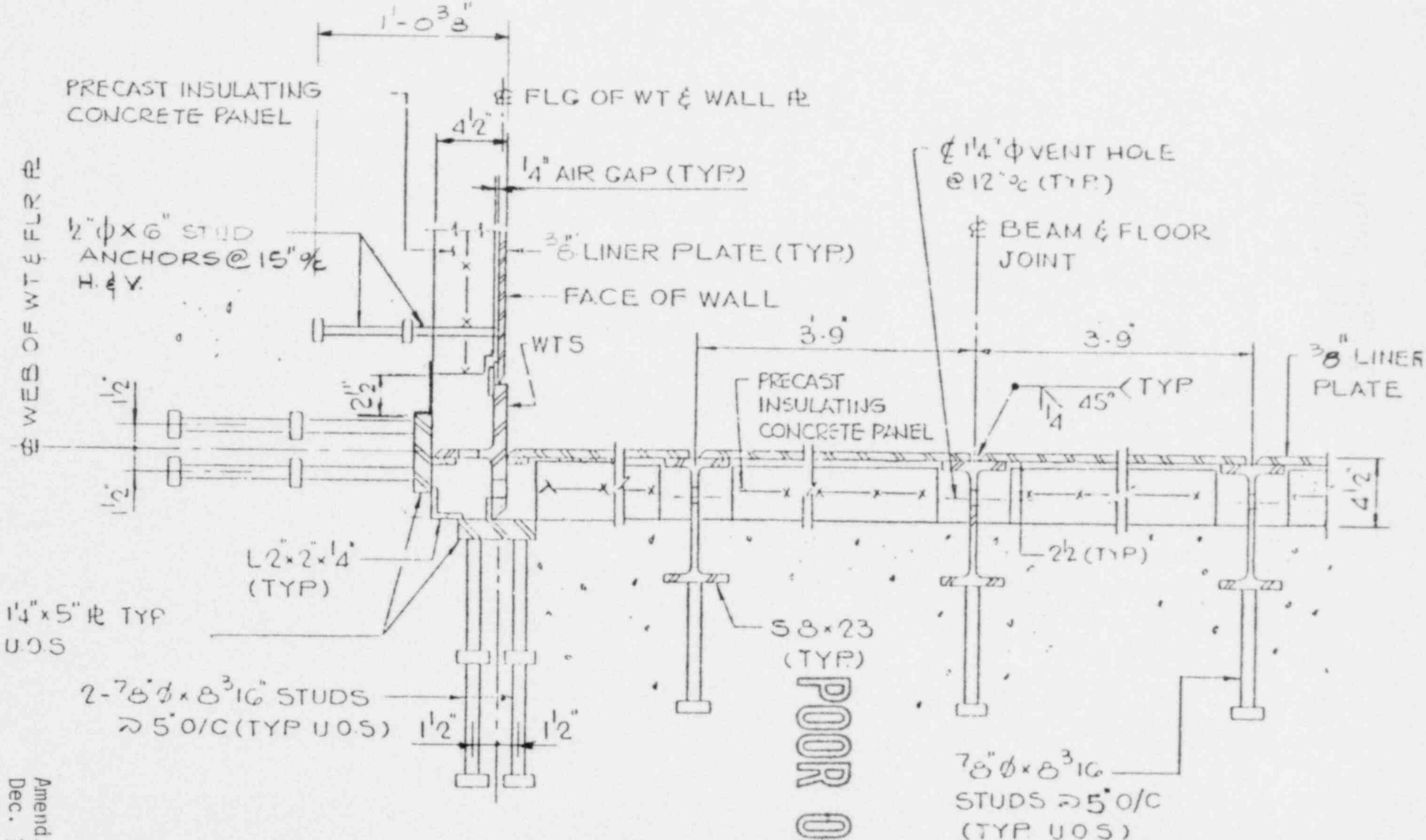


WALL-FLOOR LINER CORNER

FIGURE 3A.8-3

Amend. 59
Dec. 1980

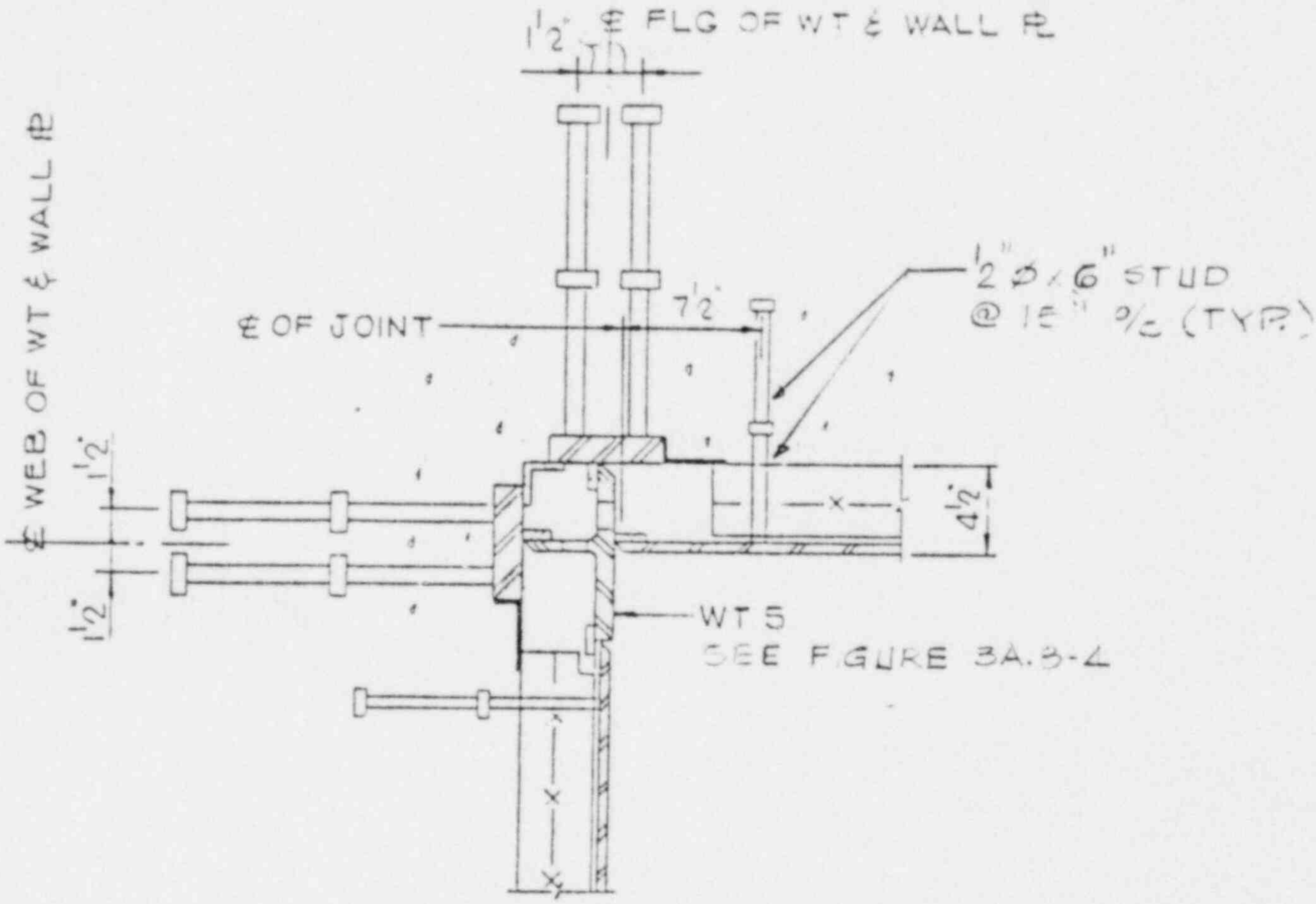


POOR ORIGINAL

FLOOR & WALL DETAIL

FIGURE 3A.8-4

POOR ORIGINAL



TYPICAL INSIDE WALL CORNER DETAIL

FIGURE 3A.8-5

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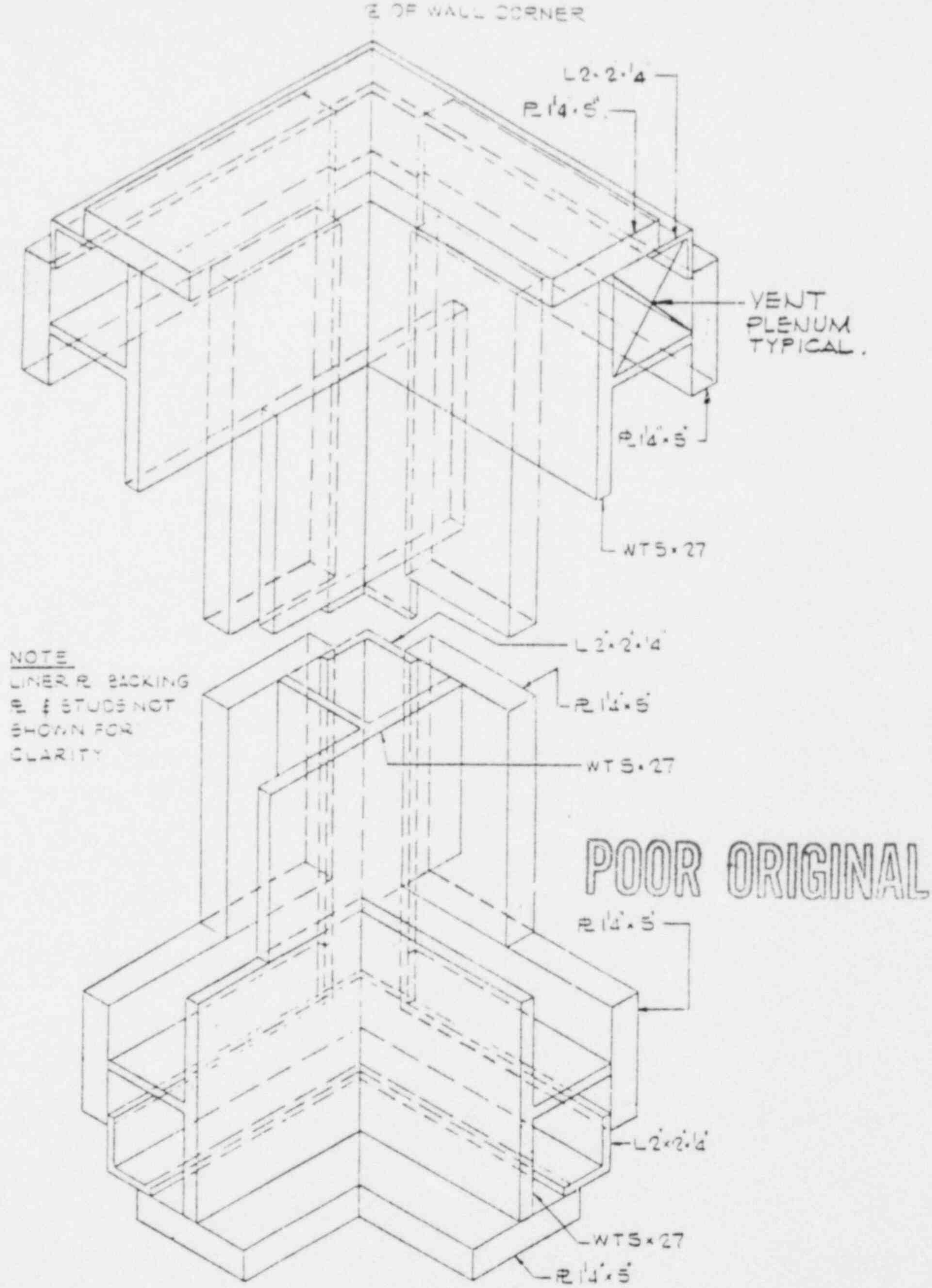
Dec. 1980



POOR ORIGINAL

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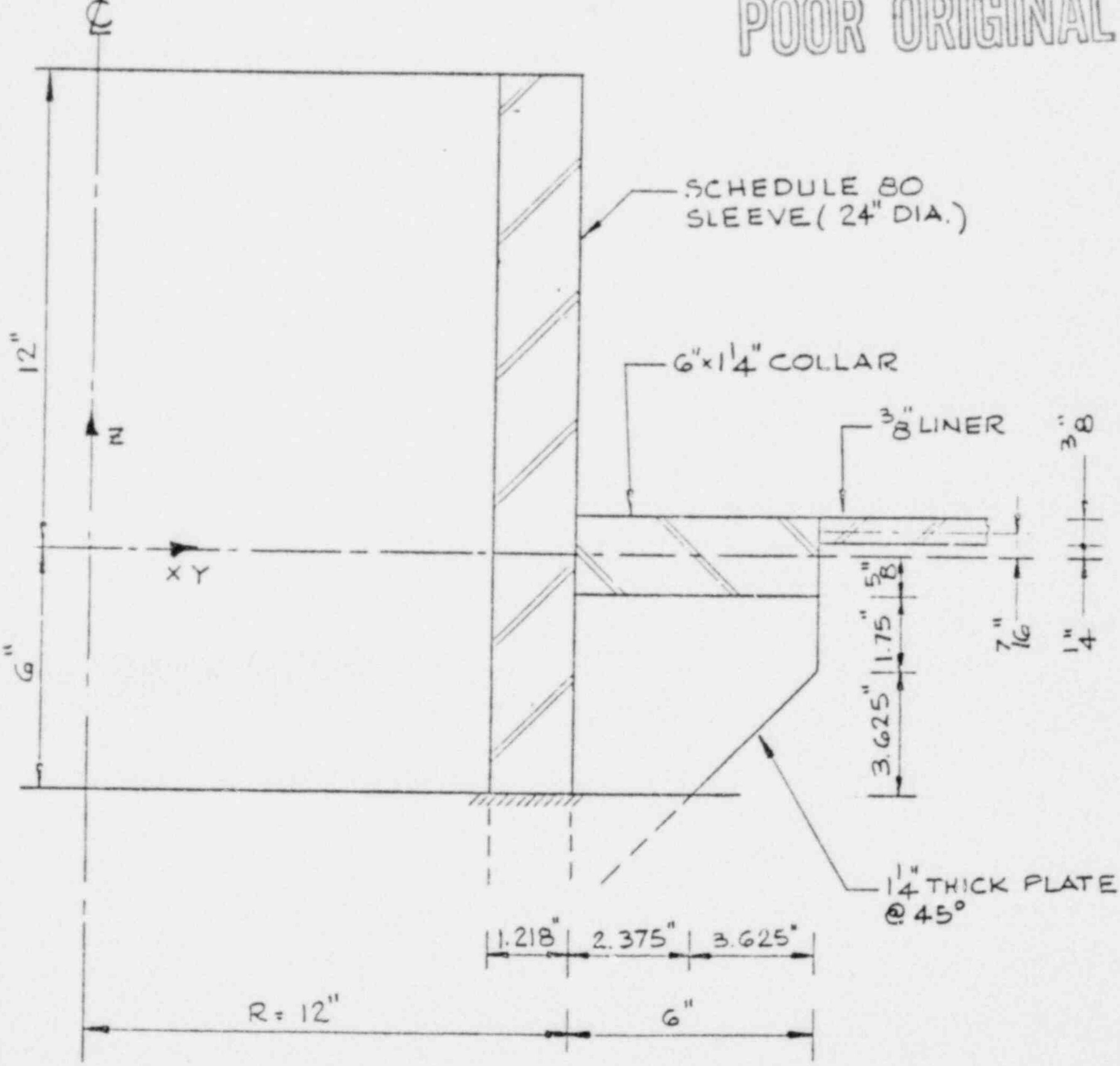
ISOMETRIC OF CORNER ANCHORAGE

FIGURE 3A.8-6



Section Of Wall Liner Penetration

Figure 3A.8-7

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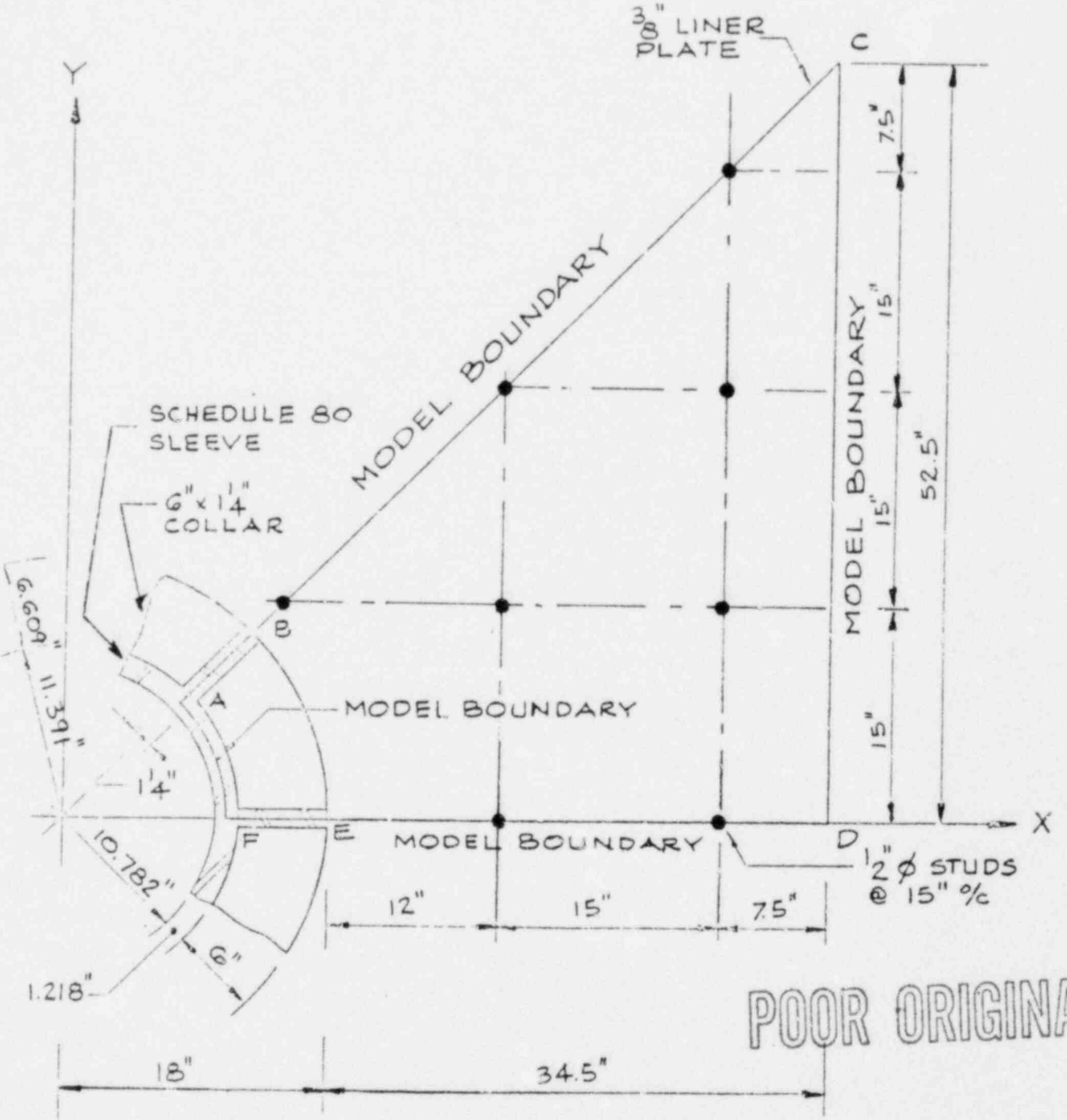
Dec. 1980



Sectional Elevation Of Wall Liner Penetration

Figure 3A.8-8

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LIST OF FIGURES Continued

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LIST OF FIGURES Continued

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51 59

LIST OF FIGURES Continued

LIST OF FIGURES Continued

LIST OF FIGURES Continued

m) Requirement - Key the upper internals structure to the core
51 barrel such that the lateral movement is limited.

Bases - Keying the upper internals structure to the upper core restraint former limits the lateral deflection of the lower end of the upper internals structure during seismic excitation. This deflection control is essential to satisfactory operation of the control rod system.

4.2.2.1.2.8 Core Restraint System

a) Requirement - Provide means for the control of core motion so that it is both predictable and safe.

Bases - Reactivity feedback in an LMFBR core is sensitive to core geo-
59 metric changes. In the case of the CRBRP, uniform radial compaction of
the core inserts reactivity at the rate of approximately 1/2¢ per mil of
inward motion of the outer core radial boundary. The core restraint
59 system is designed such that reactivity insertions from this source
are sufficiently small so as to produce no damage to the core components.
In addition, core design features are employed which enhance predict-
ability of core movement and minimize the occurrence of spurious
59 scrams due to incremental core motion.

b) Requirement - The reactivity feedback attributable to the core restraint system shall be such that Criteria 9 and 10 are satisfied for reactor power range operations.

Bases - Satisfaction of the reactivity related design bases
59 identified in Criteria 9 and 10 in Section 3.1.3. I.
42

c) Requirement - Limit the potential magnitude of core compaction
(and the resulting reactivity insertion) due to postulated
stick-slip motion to less than allowable values which assure that
the fuel and cladding will not be damaged. The allowable step
42 reactivity insertion magnitude is 60¢ at full power and 85¢ at
power levels of one megawatt or less.

Bases - The important parameters for determining a step reactivity insertion limit are fuel temperature, cladding temperature, and sodium temperature.

Analysis presented in Section 15.2.3.3 of this document has shown that a step reactivity insertion of 60¢ at full power will not bring the fuel centerline temperature to the level that would initiate fuel melting, will not result in the clad reaching the emergency event temperature limit of 1600°F, and will not result in sodium boiling in the hot channel. These results were all achieved simultaneous with the assumption of an SSE seismic event which retards insertion of the control rods. An assumed 90¢ reactivity insertion by contrast does produce up to 25% melting of the fuel and peak clad temperatures up to 1700°F. A 60¢ step reactivity insertion limit provides a satisfactory margin against damage to the core components.

59 For a step insertion of 85¢ at power levels of one megawatt or
less the fuel cladding temperature, fuel temperature, and
coolant temperature will not even reach normal full power
temperature levels. Thus no damage would occur and the reactor
59 would be shut down by the PPS system.

d) Requirement - Provision shall be made for release of core restraint interassembly loads such that refueling insertion and withdrawal loads will not exceed 1000 lbs., including the buoyant weight of the assembly being handled.

59 Bases - This requirement is based on the field experience and
recommendations of engineers with reactor refueling experience.
The requirement for refueling loads to be kept at a low level
is based on concern for the possible consequences of applying
high loads to adjacent core assemblies scheduled for continued
59 operation in the core and the reduced ductility of the core
assemblies being removed. Limiting the insertion and withdrawal
loads to 1000 lbs. maximum effectively precludes the potential
for damage to either the core assembly being handled or the
adjacent core assemblies.

e) Requirement - With the core in the refueling configuration, all core component handling sockets must be positioned within ±0.58 inches of true position.

Bases - The IVTM is being designed to locate and engage with
the core components when the IVTM grapple head and the core
component handling socket are misaligned by up to 1.75 inches.
A conservative scoping analysis of the potential positioning
errors in the IVTM grapple due to head tolerances, IVTM
tolerances, and structural deformations gives a maximum IVTM
grapple position error of ±1.1 inches. This leaves an addi-
tional 0.65 inches for core component handling socket posi-
51 tioning errors. The core component handling socket positioning

requirement is set at ± 0.58 inches leaving a 70 mil contingency allowance that will not be exceeded using the most conservative assumptions relative to component tolerance stack up and structural deformation.

f) Requirement - The core restraint system shall position the upper end of the control assembly ducts to within ± 0.53 inches of true position.

Bases - Deviation of the control assembly ducts from their true position results in clearances between the control rod driveline and the interfacing structures being reduced. Reduction of these clearances eventually leads to the generation of interaction loads between the driveline and the surrounding structure. The allowable limit to this clearance reduction is set by the effect that the resulting interaction loads have on control rod scram insertion times. A conservative scram dynamics analysis has shown that the ± 0.53 inch control assembly positioning error is compatible with acceptable control rod system scram performance. A larger control rod duct positioning error may be compatible with acceptable scram performance. Fixing the maximum allowable control rod duct positioning error at ± 0.53 inches represents a conservative design approach which assures that positioning errors in the control rod ducts will not impair in any way the functional capability of the control rod system.

g) Requirement - Positioning control of the fuel assembly duct nozzles shall be such as to minimize lateral hydrodynamic forces imposed on the upper internals structure instrumentation posts.

Bases - Water loop testing of reactor internals dynamic models has shown impingement of the core effluent jets on the upper internal structure immediately above the core to be a potential source of upper internals structure vibration excitation. The potential for this excitation increases when the jets
53 impinge on flat surfaces with hydrodynamic force generating capability. The UIS instrumentation posts and shroud tubes have small transverse dimensions and large vertical dimensions to provide the required bending strength while producing minimum drag. The requirement for core effluent position control assures that the main body of the effluent jet will pass cleanly between the instrumentation posts and shroud tubes, thereby minimizing the potential for hydrodynamic force generation.

h) Requirement - The core restraint system shall be designed to
51 prevent a general condition of duct to duct contact in the active core region.

Bases - The purpose of this requirement is to assure that the functional capability of all core components is preserved. Potential failure mechanisms associated with a general condition of duct to duct contact are: duct brittle fracture under seismic loading, excessive duct deformation with the buildup of large interduct contact loads and adverse pin bundle-duct interaction. By limiting the degree of duct-to-duct contact, the functional capability of all core
59 components is preserved.

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).

For non-seismic events, control rod system alignment requirements
44 are defined by Figures 4.2-95A and B. The contribution from the core re-
straint 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

14

59 4.2.2.3.2.3 <u>Modifications to the High Temperature Design Rules for Austenitic Stainless Steel</u>

<u>Creep-Fatigue Evaluation</u>

Creep-fatigue evaluations will be performed in accordance with the applicable criteria except as modified herein.

The creep-fatigue damage rules of Paragraph T-1400 of Code Case 1592 consider creep damage accumulations resulting from stresses which are clearly compressive to be equally as damaging as creep damage accumulations from tensile stresses. The damaging effects of compressive stresses in a high temperature environment are known to vary considerably from one material to another. Strain controlled fatigue test data of austenitic stainless steels (304 and 316 SS) consistently point to compressive residual stresses having little or no deleterious effect. There is also test evidence that suggests that when subjected to alternate hold periods in both tension and compression that hold in compression has a healing effect on the damage produced by the tensile hold. Based upon these data, the creep-fatigue damage rules are modified as described in subsequent paragraphs.

The effects of the presence of stress concentrations on stress rupture properties are known to vary considerably with the material, geometry of the stress concentration, magnitude of the stress level, the environment, and life. In the case of austenitic stainless steels, test data consistently points to stress concentrations having a less severe effect on stress rupture strength than predicted using the analytical approaches of Code
59 Case 1592 and RDT Standard F9-4 criteria, and in the case of 316 SST, there is a consistent trend to significant notch strengthening for some types of geometries, particularly with a service environment and life at
59 the upper limit of those in the UIS. The rules of RDT Standard F9-4T and Code Case 1592 require comparing the peak stress to the Code strength which is based upon smooth specimen data. They do not recognize that peak stresses may have no adverse effect on stress rupture strength nor do they recognize that non-uniform stress states may alter the strength of the material. Based upon test data, the creep damage rules are modified as described in subsequent paragraphs to allow the use of a peak stress to rupture design curve.

<u>Modifications to Creep-Fatigue Damage Rules</u>

In cases where, in the service life of the component, all three principal stresses are clearly compressive during a hold period, the creep-fatigue evaluation shall be modified as described herein. If prerequisites for the use of the modified rule are not met for a portion of a component's life, the creep-fatigue rules of T-1400 of Code Case 1592 shall be used without modification for that portion of the component's life. The modified rule is described in items (1) to (7), where (1) to (5) are prerequisite conditions, and item (7) is a final applicability criteria to
49 be satisfied.

(1) None of the three principal stresses is tensile during hold period.

(2) The material is austenitic stainless steel type 304 or 316.

(3) Metal temperature does not exceed $1200^{o}F$.

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

(5) Simplified or rigorous inelastic analysis is used.

(6) Subject to the above limitations, creep-fatigue damage may be calculated in accordance with T-1400 of Code Case 1592 as modified by the following steps.

Step 1 - Calculate the fatigue damage in accordance with T-1411, -1412, -1413, and -1414.

Step 2 - Calculate the creep damage in accordance with T-1411 and T-1420.

Step 3 - Multiply the cumulative creep damage by 1/f, taking f = 5, for those hold periods where the stress is compressive, i.e.

$\frac{1}{f} \sum_{k=1}^{q_1} \left(\frac{t}{T_d} \right) k$, where q_1 are the compressive holds.

Step 4 - Calculate the total damage including creep damage from conditions where the principal stresses are not clearly compressive, i.e.,

$$\sum_{j=1}^{p} \left(\frac{n}{N_d} \right) j + \frac{1}{f} \times \sum_{k=1}^{q_1} \left(\frac{t}{T_d} \right) k + \sum_{k=q_2}^{q} \left(\frac{t}{T_d} \right) k = D_e$$

Step 5 - The acceptability of the damage (D_e) is determined in accordance with T-1411. The creep-fatigue damage envelope
53 is shown in Figure 4.2-47A (Figure T-1420-2 of Code Case 1592). If the total damage (D_e) falls within the envelope, the damage level is acceptable.

53 D_e is plotted on Figure 4.2-47A and the allowable damage (D) is the sum of the allowable creep and fatigue damage components
53 at the intersection of the damage envelope and a line extended
49 from the origin through D_e. See Figure 4.2-47A.

The principal reactor internals components involved in normal disassembly (refueling) operations are: the Upper Internals Structure (UIS), the Core Former Structure (CFS) and the Core Restraint System (CRS). In these items, sliding motion occurs in two friction couples; between the UIS keys and the CFS keyways and the second between reactor assembly duct load pads. Neither of these couples is a safety related item in that motion between mating parts of the couples needs to occur only during reactor refueling operations. Any increased friction would be an operational concern only.

1. Friction Couples

Preliminary selection of the mating couple materials has been made for the preliminary designs of the UIS, CFS and reactor assembly duct load pads. The UIS key to CFS keyway couple is
53 Haynes 273 against Alloy 718 at 400 ± 50^{o}F in sodium with approximately one year soak at approximately 1000^{o}F. The couple between the load pads is chromium carbide against
53 chromium carbide at 400 ± 50^{o}F with approximately one year soak at the assembly outlet sodium temperatures. Coefficients of friction to be used in preliminary design are shown in Tables 4.2-31A, and 31B. Test data will be used as the design is preliminary, final selection of mating materials is yet to be made, and the use is not safety related.

2. Antigalling Characteristics

Tables 4.2-31A and 31B show friction testing date of candidate samples.

3. Irradiation Stability

No irradiation test data is available for Haynes 273 against Haynes 273. However, the maximum fluence at the UIS key to CFS keyway location is 1.4 x 10^{19} n/cm^2. The probability for radiation damage at this fluence is very small.

Table 4 lists coefficients of friction measured after irradiation of the chromium carbide test specimens in EBR II to
51 approximately 1 x 10^{22} n/cm^2. Additional chromium carbide specimens are being irradiated in EBR II to higher fluences.

4.2.2.4 Evaluation

4.2.2.4.1 Lower Internals Structure

This section discusses the stress and thermal analyses of the following components:

Core Support Structure
Lower Inlet Module
Bypass Flow Module
Fixed Radial Shielding
Fuel Transfer and Storage Assembly
Horizontal Baffle
Core Former Structure

These components were all analyzed for the critical loading condition and shown to meet the governing structural criteria. Further details are
59 reported in each subsection.

4.2.2.4.1.1 Analysis of Core Support Structure (CSS)

Alternate Low Temperature Design Criteria

Section III of the ASME Code provides stress limits for austenitic steels for temperatures up to $800^{\circ}F$. The design temperature of the lower CSS is $775^{\circ}F$; however, during some thermal transient events the maximum metal temperature does exceed $800^{\circ}F$ for short periods of time.

59

Since the time-dependent failure modes were shown to be insignificant
for the CSS by satisfying the conditions of Test No. 4, Code Case 1592
27 59 and RDT F9-4, the alternate structural limits of the code case were
employed in the CSS evaluation.

Geometry

The core support structure (CSS) concept considered in this analysis
is shown in Figure 4.2-50. The CSS consists of a perforated support plate,
core barrel, and lower inlet module liners. Portions of the support core
and reactor vessel, are included in the analytical model, and all of these
59 components are referred to as the "core support structure" in this analysis.

Thermal Analysis

Two thermal models were developed to calculate transient temperatures in the CSS. A 30 degree sector model (TAP-A computer code) was used to calculate temperatures in the perforated support plate and an axisymmetric model (ANSYS finite element code) was used to determine temperatures in other CSS components. The element geometry of the thermal models is identical with the corresponding stress models shown in Figures 4.2-52 and 4.2-54.

The sector and axisymmetric models were used to analyze the CSS-6N (N-4a), CSS-2U(U-2e), CSS-4U(U-18) and CSS-1E(F-4a) design transients for the CSS. It was shown that these four transients conservatively umbrella all of the plant duty cycle events.

Reactor inlet plenum mixing analyses were performed to determine
the transient sodium boundary temperatures for the CSS. Convective heat
transfer coefficients were calculated for the CSS surfaces exposed to flowing
sodium. Interface conditions with the lower inlet modules (LIMs) were
59 determined with detailed local models.

59 Structural Analysis Models

The ANSYS finite element computer program in conjunction with the "equivalent solid plate" method of analysis for perforated plates (Article A-8000 of Ref. 33) was used to perform the detailed structural analysis of the core support structure. Two types of finite element models were used in this analysis. The axisymmetric model (Figure 4.2-51 was used to cal-
59 culate stresses in parts of the CSS other than in the perforated region.

The perforated plate region of this model defined by R* in Figure 4.2-51 was modeled by using the equivalent solid plate method of analysis for perforated plates. In this method, the perforated plate is replaced by a solid plate which is geometrically similar to the perforated plate but has modified values of elastic constants. The effective elastic constants E* and ν are functions of the ligament efficiency, η. The deflections computed using conventional methods are correct; however, the actual values of the stress intensities in the perforated plate are determined by applying multiplication factors to the nominal stresses computed for the equivalent solid plate. The second model used in the analysis is shown in Figure 4.2-52. This model essentially substructures the perforated region of the core support plate by having boundary conditions applied at the sector periphery correspond to conditions at the equivalent location in the axysymmetric model for the transient time being analyzed. A third model was used in the analysis which is unrelated to the previously described models. Figure 4.2-53 illustrates the lower inlet module liner model used for calculating both primary and secondary stresses. In addition, the key structural evaluation sections
59 are detailed to identify critical regions in the structure.

Structural Analysis

The CSS was analyzed for pressure, dead weight, OBE and SSE seismic, and the thermal transients using the structural models previously described.

The dead weight load used in this analysis is obtained by adding elements to the model shown in Figure 4.2-50 to correctly simulate the masses of the core assemblies, lower inlet modules, bypass flow modules and core former structure. The modified finite element model is shown in Figure 4.2-54.

The CSS axisymmetric model (Figure 4.2-54) was used to evaluate the primary stresses due to the seismic loads using ANSYS. Results of the seismic analysis indicated that OBE and SSE stresses are generally low, with the minimum margin-of-safety being conservatively calculated as 0.77
59 for the faulted event.

Thermal stresses were calculated in the CSS for the thermal
59 transients using the CSS axisymmetric and sector models with the internal temperatures which were calculated in the thermal analysis.

Structural Evaluation

Five sections in the CSS axisymmetric model shown in Figure 4.2-51 and 10 sections of the core support plate sector model shown in Figure 4.2-52 were selected for the structural evaluations. These sections represent the high stress areas in the CSS structure, and their selection was based on a thorough review of the finite element stresses for pressure, dead weight, seismic, and the thermal transients.

The primary plus secondary stress intensity limit of 1.5 Sm was reduced to 1.35 Sm in the perforated region of the core support structure. This reduction was made to account for the actual bending shape factor for the geometry of the ligament. The 1.5 shape factor is only applicable for a rectangular cross section.

The primary, primary plus secondary, and fatigue evaluation results are summarized in Table 4.2-23. These data are limited to the maximum of each stress intensity category for the locations identified in the table. Simplified inelastic analysis techniques were utilized to show that the areas with primary plus secondary stress intensity values exceeding the
59 3 Sm allowable limit are acceptable.

51 4.2.2.4.1.2 Analysis of Lower Inlet Module (LIM)

The LIM was analyzed to the requirements for the criteria listed in Section 4.2.2.3.1.2. Because of the complicated geometry of the LIMs and the varying thermal/hydraulic environment on each of the 61 units, a number of finitie element models were developed for the analysis. Two separate LIM designs are analyzed which envelop the existing configurations. These two basic cases analyzed are for a peripheral module and a central module (See Figure 4.2-50). Cross-sections at four different elevations were analyzed with at least one finite element model generated at each section for both cases.

Primary stresses in the LIM are due to deadweight, differential pressure, seismic events and IVTM loads, while secondary stresses are generated by thermal loading. In particular, the U-18 (Loss of all offsite power) and the F-4a (Saturated steam line rupture) thermal events from Appendix B were analyzed since these were shown to be enveloping down and up umbrella events respectively.

Results of the load controlled portion of the analysis indicate that all applicable requirements are satisfied with the minimum margin-of-safety being 1.09 in the conservatively modeled upper body region of the LIM. Similarly, secondary stresses are relatively low in the component with the greatest fatigue damage occurring in the peripheral module stem and having
59 a magnitude of 0.354.

4.2.2.4.1.3 Analysis of Bypass Flow Modules (BPFM)

59 Geometry

The BPFM geometry and relationship to adjacent components is shown in Figures 4.2-41A & 4.2-41B. Each individual module is composed of an upper forging with holes for the RRS assembly receptacles, side wall plates, and a bottom plate as shown in the section view of Figure 4.2-41B. The geometries
38 of two analytical models are shown in Figures 4.2-63A & 4.2-63B.

Thermal Analysis

The ANSYS finite element program was used to perform the heat transfer analysis of the BPFM for the most adverse thermal transients considered. Two finite element models were developed to calculate the transient temperatures. The outside half of the BPFM was modeled for two-dimensional analysis
51 to determine the critical region in the BPFM due to the most severe transient event. A one-eighth section, three dimensional finite element model was developed for the critical region evaluations.

The U-18 duty cycle was concluded to be the most adverse transient
38 for the BPFM. The most severe temperature gradients occur in the interval

900-1000 seconds, near the termination of the steep down temperature transient. From this data the Intervals Stress Analysts identified the region of the BPFM requiring further detailed analysis.

Figure 4.2-63A shows the three-dimensional thermal model of the BPFM developed cooperatively with the Stress Analysts. In addition to one corner of a BPFM, the model includes the core barrel and stagnant sodium at the side and end walls. Comparison of the two-dimensional analysis temperature contours with those from a similar plate of the three-dimensional analysis shows good agreement between the models.

Structural Analysis

The structural integrity of the BPFM was evaluated to Subsection
NG, Section III, of the ASME Boiler and Pressure Vessel Code. Along with the
59 thermal transients, the seismic loads, differential pressure load, loads due
to core former misalignment and duct bowing, and transient mechanical loads
due to the transfer operations of the Removable Radial Shield (RRS) assemblies
have been investigated.

The functional requirements of the module to module connecting lugs and pins and the BPFM/CSS connecting pins were evaluated for seismic loads using a 3-D finite element model composed of the six interconnected modules. The structural evaluation results for the most critical loads (OBE) versus the Code criteria are given in Table 4.2-29A, showing positive margins in every area.

For the evaluation of the BPFM body, a perforated thick plate model
was used to study the IVTM transient mechanical load. The differential
pressure load was evaluated using the one-eighth 3-D BPFM Solid Model (shown
in Figure 4.2-63B) plus the perforated thick plate model. Thermal transient
stresses were evaluated using the one-eighth BPFM solid model in conjunction
59 with all of the above mentioned mechanical stresses. The BPFM inlet port
region where the bypass flow is fed into the BPFM mixing chamber from the
38 CSS was also analyzed, using an axisymmetric finite element model. This
59 region proved to be the most critical area of the BPFM body.
38

The minimum margin of safety in the structure is 0.23 for the
bearing stress between the BPFM body and the CSS. The minimum margin of
safety in the entire component is 0.64 for the primary stress intensity
category. A simplified elastic-plastic analysis was performed to show that
the structure does shake down to elastic action during its service life. The
maximum calculated creep fatigue damage is 0.27 compared with an allowable
of 1.0. In conclusion, the BPFM has been shown to meet all of its design
59 requirements.

4.2.2.4.1.4 Analysis of the Fixed Radial Shield (FRS)

This section summarizes the stress analysis performed on the Fixed
Radial Shield (FRS) and is based on the requirements of the criteria listed
59 in Section 4.2.2.3.1.3. The load controlled portion of the analysis was

conducted using the 3-D solid finite element model shown in Figure 4.2-75. Since the three segments of the shield are identical, only one sector is modeled. Deadweight, vertical seismic and landing loads on the structure are minimal and do not generate any significant stresses. The only primary load then considered is horizontal seismic. The primary stress tabulation at the critical locations in the FRS is shown in Table 4.2-29D. All primary limits are satisfied for the FRS and its attachment pins. Mode frequency analysis conducted on the FRS indicates a fundamental mode frequency of 67 Hz, precluding any possibility of dynamic amplification.

Strain-controlled stresses in the FRS were also calculated using finite element modeling techniques. The primary source of elevated temperatures in the FRS is due to nuclear heating where the center of the shield remains hot while the surfaces are cooled from being washed by sodium. By analyzing steady-state, U-18 and U-2B transients, the most critical thermal events are analyzed. Effects of neutron irradiation and sodium exposure are shown to either be small or to make the evaluation non-conservative, and were therefore ignored. Results of the secondary stress analysis indicate that creep/fatigue damage effects are small (0.055 out of an 0.9 allowable) and that strain limit requirements are satisfied.

4.2.2.4.1.5 Analysis of the Fuel Transfer and Storage Assembly (FT&SA)

The fuel transfer and storage assembly (FT&SA) was analyzed to the requirements listed in Section 4.2.2.3.1.3. Analyses performed show that the FT&SA satisfies ASME Boiler and Pressure Vessel Code Limits for primary type loadings. For this structure, primary loads include deadweight, OBE & SSE seismic and external pressure. A finite element model using 3-D beam elements is shown in Figure 4.2-76 which is used to determine load-controlled stresses. Results of a mode frequency analysis indicated a fundamental mode frequency of 30 Hz. Seismic stresses calculated in the tubes were shown to be small. Primary stresses in the lower support block region of the FT&SA are also shown to be small as indicated in Table 4.2-29E.

Finite element analyses were also used to generate temperatures and thermal stresses in the FT&SA. The thermal stress analysis of the tubes was conducted elastically for steady-state and the U-4 transient discussed in Appendix B.

The U-4 transient enveloped all thermal events to which the FT&SA is exposed. Results of the thermal stress analysis indicate that the creep/fatigue damages are well within the interaction envelope of code case 1592. In addition, the strain limit requirements are satisfied for the most critical section. Thermal analysis of the lower support block was conducted using a finite element model of a cross-section of the block. Results of this deformation controlled stress analysis indicate relatively low stress levels. Due to the low service temperature of the block no creep damage will occur. The
59 fatigue damage was also low with a value of .149.

4.2.2.4.1.6 Analysis of the Horizontal Baffle (HB)

The Horizontal Baffle (HB) was analyzed to the criteria outlined in Section 4.2.2.3.1.3. Analysis results show that the HB satisfies the ASME Boiler and Pressure Vessel Code Criteria. HB plate stresses were obtained using finite element modeling techniques. Primary loads considered include deadweight, pressure, fuel transfer machine loads and seismic events. It was shown that there will be no dynamic amplification and the rigid plate can be analyzed seismically using maximum support accelerations. Table 4.2-29F indicates results of the load controlled analysis for the HB plate for loadings discussed above. All primary limits for the HB and interfacing components have been satisfied.

Loadings considered for the strain-controlled portion of the analysis are due to the plate preload and thermal transients. The transients examined in the analysis are the U-1, U-4 and E-2 events discussed in Appendix B, and envelop all thermal events. The thermal stress analysis was also conducted using finite element techniques. Structural evaluation of the results of the strain-controlled analysis indicate that all areas of interest demonstrate structural adequacy by elastic analysis except the HB plate. Resulting steady-state stresses and transient strain ranges are small enough so that the creep/fatigue damages are within the limits of Code Case 1592. A thermal striping assessment was made, results of which indicated negligible fatigue damage since the maximum potential is $4^{o}F$. Since the elastic stresses calculated in the HB plate were too high to meet the limits for elastic analysis, a simplified inelastic analysis was undertaken to demonstrate the structural adequacy of the part. Results show that total strain accumulated throughout the life of the component satisfy the allowables of 1%, 2% and 5% on average, linearized and peak respectively. The creep fatigue requirements are also satisfied.

4.2.2.4.1.7 Analysis of the Core Former Structure (CFS)

This section summarizes the results of the Core Former Structure (CFS) stress analysis. The evaluation is made to requirements of criteria used in Section 4.2.2.3.1.2. Primary stresses in both the upper and lower core former rings are determined by generating a 2-D in-plane finite element model. This model consists of beam elements with varying cross-sections to simulate both the irregularly shaped core former ring and the segmented inserts. Loads are applied to the upper ring through the 3 UIS keyways and 6 core barrel lugs, and to the lower ring through 12 equally spaced compression only azimuthal spacers. These loads are due to deadweight, seismic events, core misalignment and core reaction.

Mode frequency analysis was conducted on both the upper and lower geometries. The minimum fundamental mode was 66 Hz, on the upper ring, placing the structure in the zero period acceleration domain.

Results of the load controlled analysis determined that all primary
59 limits were satisfied as required by the applicable criteria.

The deformation controlled stresses and strains were determined by a 2-D axisymmetric model of a cross section of the ring. The model used for both the thermal and thermal stress analyses of the ring is shown in Figure 4.2-77.

Thermal boundary conditions applied to the upper core former model were fluid temperatures applied through a convection coefficient in several different regions of the model as shown in the figure. The structure was analyzed elastically for the U-1b, U-2b, U-18 and E-16 thermal events which may be conservatively used to umbrella all other loadings.

All regions of the CFS were shown to be adequate using elastic analysis methods except the top surface of the upper ring. This area was shwon to be adequate by simplified inelastic methods. The fatigue damage at this location was .414 with a creep damage of .239. This combination of
59 damages falls within the creep fatigue interaction envelope of Code Case 1592.

59 4.2.2.4.2 Upper Internals Structure

This section presents the analysis performed in support of the final design of the Upper Internals Structure (UIS) and used to demonstrate the adequacy of this component for the expected service conditions and environment. The adequacy of the design is based primarily upon meeting the criteria of Section III of the ASME Boiler and Pressure Vessel Code including Code Case N-47, and supplemented by RDT standards F9-4T and F9-5T and special project structural design rules. A summary of the components analyzed, material properties, structural design criteria, mechanical loads, thermal environment, methods of analysis, and structural analysis is presented herein.

59 4.2.2.4.2.1 Components Analyzed

The major components of the UIS are identified in Figure 4.2-45. A brief outline of the functions of the UIS is given in Section 4.2.2.2.1.7. A list of the components of the UIS analyzed to demonstrate structural adequacy of the design are:

- o Lower Plate and Ligament
- o Upper Plate
- o Support Columns
- o Shear Webs
- 51 o Core Barrel Key

- Instrumentation Posts
- Upper and Lower Shroud Tubes
- Chimney Assemblies
- Mixing Chamber Thermal Liners
- IVTM Port Plug
- IVTM Port Plug Cover

59 4.2.2.4.2.2 Material Properties

The ASME Code is the prime source for materials properties. For material properties not specified in Section III of the code or applicable code cases, the mechanical properties are based on the Nuclear Systems Materials Handbook TID-26666, RDT Standard F9-4T, RDT Standard F9-5T or the sources given in Section 4.2.2.3.3.

There are no irradiation effects on the mechanical properties of Inconel 718 or 316 stainless steel at the highest fluence levels attained at the end of its 30 year service life. The maximum fluence level is less than:

$$1 \times 10^{21}\ n/cm^2$$

Type 316 stainless steel is a non-age hardenable alloy. Therefore, no significant changes in strength or hardness should result from long term exposures at temperatures up to 1100°F. Inconel 718 is an age hardenable alloy, however, the age hardening process does not result in significant reductions in mechanical properties when subjected to temperatures up to 1100°F for component lifetimes. No allowances have been made for the effects of thermal aging on the properties of either alloy. However, experimental material properties programs to study the behavior of both alloys due to the thermal environment and sodium exposure are discussed in Section 1.5.

The sodium effects of Section 4.2.2.3.3.2.1 are implemented in creep-fatigue damage evaluations of 316 stainless steel by use of the following factors.

51 a. Factor on Stress Strain Curve (see Table 4.2-31)

$$K_N = \frac{\sigma_y (C+N)_e}{\sigma_y (C+N)_o}$$

b. Factors on Creep Rupture Strength

$$F = F_{NI} F_{NS}$$

$$F_{NI} = 1.0 - [(C+N)_o - (C+N)_e] (-.000372 + 9.5 \times 10^{-7} T)$$

$$F_{NS} = 1.0 - (-.2127 + 4.75 \times 10^{-4} T) \quad T \geq 450^oC$$

The following definitions apply to the terms in the above factors:

σ_y (C+N) = Yield Strength (Table 4.2-31)

$(C+N)_o$ = Initial Carbon + Nitrogen Content (Assumed)

= 0.12 wt/%

$(C+N)_e$ = Equilibrium (Carbon + Nitrogen) Surface Content

T = Temperature in ^oC

Alloy 718 is conservatively treated in the same manner.

51 c. Factors on Primary and Secondary Stress Limits

The effect of the average C + N content through the thickness of a section on primary and secondary stress limits was evaluated and shown to be negligible for the thicknesses and temperatures of the upper internals structure for 316 stainless steel.

d. Cyclic Hardening

A factor K_c is applied to alter the stress-strain curve yield surface.

where:

$$K_c = (-0.144 + 3.094 \Delta \varepsilon)^{1/2} \text{ for } \Delta \varepsilon > 0.36\%$$

If the strain range $\Delta \varepsilon$ is $\leq 0.36\%$, then $K_c = 1.0$.

e. High Cycle Design Fatigue Strength

Stainless steel materials subjected to high cycle thermal fluctuations and flow induced vibration phenomena require a fatigue strength evaluation beyond the Code Case 1592 (N-47) curve limit of 10^6 cycles. The Code Case 1592 (N-47) curve is extrapolated beyond 10^6 cycles using a slope on cycles of -0.12 for load controlled situations. In cases where conditions are strain controlled, the special purpose high-cycle fatigue criterion, as described in 4.2.2.3.2.3, is used beyond 10^6 cycles.

59 4.2.2.4.2.3 <u>Structural Design Criteria</u>

The portion of the UIS within the reactor vessel operates at elevated temperatures above 800°F. Under these circumstances the UIS is classified as an elevated temperature structure and is designed and analyzed as an ASME - III Code Class 1 component.

Alternate structural design criteria have been adopted in
59 the cumulative creep-fatigue damage rules of Code Case 1592 (N-47) and RDT Standard F9-4. These criteria assume that in compressive hold, creep rupture damage is 20% as damaging as the damage caused by the same sustained stress in tension. It applies for austenitic stainless steel (Types 304 and 316) at metal temperatures less than 1200°F (649°C). At times in the duty cycle when sustained stresses are
59 51 tensile, damage is computed in accordance with Code Case 1592 (N-47).

59 4.2.2.4.2.4 <u>Mechanical Loads</u>

<u>Design Loads</u>

The design condition loads are the dead weight and pressure. The design temperature of the UIS is 1220°F.

<u>Normal Loads</u>

During normal operation, the UIS carries no mechanical load except its own weight and loads due to actuation of the control rod system. The upper shroud tubes carry the dashpot loads resulting from the primary control rod scram arrest accelerations, and the lower shroud tube is designed to react a 19,000 pound upward load incurred in exercising the control rod breakaway joint.

The UIS is designed to preclude the occurrence of adverse structural and dynamic effects due to flow induced vibration. Where possible, the entire structure and its components are designed such that their natural frequencies do not coincide with any vortex shedding frequencies. Component mechanical stresses caused by flow induced vibration are required to meet the limits of the ASME Boiler
59 Code Section III and Code Case 1592 (N-47) for normal conditions to ascertain structural integrity with regard to fatigue.

During refueling operations, the UIS is raised and lowered so that the rotating plugs may be positioned to provide access to various reactor locations. Misalignment of the UIS keys with respect to the keyways in the CFS will cause loads between the keys and keyways. Both the normal and frictional force resulting from this misalignment are considered in the analysis.

<u>Upset Loads</u>

The upset mechanical loads on the UIS are the seismic input for the operating basis earthquake (OBE). The UIS is designed for OBE in accord with the criteria described in Section 3.7.

<u>Emergency Loads</u>

The UIS is designed to accommodate loads due to loss of primary holddown (hydraulic balance). Loss of hydraulic balance is classified as an emergency event and is assumed to occur five times, but for conservatism it is analyzed as an upset event.

59 | 51

The UIS is designed to withstand the effects of a safe shutdown earthquake (SSE). The SSE is a faulted condition, however, to be conservative, the UIS is designed to satisfy the ASME Code criteria for emergency conditions when in the operating configuration and subjected to the SSE loads.

59 4.2.2.4.2.5 Thermal Environment

Operating conditions for the UIS are specified in a 30 year histogram using the ASME Code categories of normal, upset, emergency, and faulted conditions for the mechanical loads and steady state and transient temperatures. Plant capacity is 75% giving a full power life of 22.5 years.

Normal Loads

The UIS is designed to accommodate thermal striping during
59 normal operation. The UIS surfaces directly exposed to the more severe
thermal striping are the instrumentation posts, control rod shroud
tubes, keys, the internal surfaces of the chimneys, and the UIS mixing
chamber. Sodium exiting from the chimneys will subject the support
59 columns to thermal striping.

The reactor operating temperature, for long term steady
state effects in a cumulative damage analysis, is based on reactor
59 coolant outlet temperature of 1000°F with a 2σ.

During refueling operations, which are normal operating conditions, the UIS will be at the refueling temperature of 400°F.

Normal operating temperature transients such as startup and shutdown are less severe than the upset and emergency events and are enveloped by them.

Upset and Emergency Loads

Steady state temperatures with 2σ uncertainties for a reactor
outlet nozzle temperature of 1015°F are used to begin transient analysis
51 of UIS components.

59| For most areas of the UIS the most severe upset (U) and emergency (E) thermal transients are uncontrolled rod withdrawal from full power, and loss of preferred and alternate preferred power. For the lower shroud tube, the E-16 emergency transient, three loop natural circulation, is also severe. All other UIS transients are grouped with respect to severity under these transients. The fluid temperature changes are less severe farther from the fuel exit as a result of mixing with control assembly flow and blanket assembly flow. These other assemblies also have less severe changes occuring at their exits. The heat transfer analyses of different areas of the UIS account for all these differences.

Faulted Loads

59| Two faulted events are identified in the UIS duty cycle. Only one occurrence of either of these events is considered. Faulted events are not considered in cumulative damage calculations.

59| 4.2.2.4.2.6 Methods of Analysis

Elastic analysis, simplified inelastic and rigorous inelastic analysis have been used to develop the detail design which meets all its structural requirements.

Computer Codes

The following computer codes are utilized in the heat transfer and structural analysis of the upper internals structure:

ANSYS
HOTDAMG
WECAN
TAP-A
TRUMP
VARR-II

Descriptions of these computer codes are given in Appendix A.

59| 4.2.2.4.2.7 Structural Analysis

The detail rigorous analysis can be divided between overall analysis and detail part analysis. The seismic analysis, duty cycle evaluation, and overall thermal stress analysis are overall analyses.
59| Other items discussed are detail part analyses.

Seismic Analysis

The UIS column seismic loads are obtained from the reactor system seismic analysis. The seismic analysis for the remainder of the UIS, was
59 performed with a 180° finite element model of the structure with all of the details essential to dynamic analysis. Modal analysis of the UIS Model for both operating and refueling configurations is performed to obtain natural frequencies and mode shapes.

Seismic Response Spectrum Analysis is used to evaluate primary stresses in the UIS due to seismic excitation. The analysis is performed utilizing the CRBRP Seismic Design Criteria and the
59 dynamic models identified above. Further details are described in Section 3.7. The displacements were checked to verify that impact did not occur at close clearance locations. Stresses due to horizontal support point motion and due to dead weight, are added by absolute value to response spectrum analysis stresses to give final values for comparison to the criteria.

Duty Cycle Evaluation

The UIS is subjected to a large number of Upset and Emergency condition thermal transients. The purpose of the duty cycle evaluation is to reduce the number of events to be applied in the analysis of each area of the structure to only one or two events so as to obtain an equivalent creep and fatigue damage for the entire duty cycle. In a high temperature component time dependent response and environmental effects become governing factors. Residual stresses, hold-time between cycles, elastic-plastic strain, cyclic-hardening, and creep/fatigue interaction must all be properly accounted for. In general, simplified creep-fatigue damage evaluations have been used to perform the duty cycle reduction and determine the umbrella transients.

59 ### Overall Thermal Stress Analysis

The UIS gross model is generated from, 1) shell elements for the lower and upper plates, shear web and skirt, 2) pipe elements for the columns, and 3) rigid beam elements to connect the columns to the plates and shear web. The primary purpose of the elastic lower plate model is to provide boundary forces for the more detailed inelastic lower plate model at various times during the U-18 and U-2b transients (see Appendix B for description of these transients). Since the entire UIS is included in the model, it is possible to obtain boundary forces for other components such as the
51 column center region and the column/top plate joint.

The analysis consists of generating temperatures at various
times in the U-18 and U-2b transients from thermal data obtained
59 from the overall model thermal/structural interface analysis. These
temperatures are applied to the overall structural model. The forces
produced at the boundaries of the individual components by the
thermal loadings are then applied to the more detailed models of
these components.

Lower Plate Analysis

The main purpose of the lower plate inelastic thermal stress
analysis is to, 1) provide nodal forces for the ligament analysis due
59 to the transients, 2) ascertain adequacy of the lower plate to react the
loss of hydraulic holddown load, and 3) determine if the lower plate
satisfies the ratcheting criteria.

59 The boundary forces are obtained from the UIS overall structural
model analysis together with temperatures provided by T/H Analysis.

Lower Plate Ligament Analysis

The critical lower plate ligament is analyzed for the steady state, the critical thermal transient and the applied interaction loads from the lower plate model. A 3-D finite element model is used in the analysis. The results are then compared to the allowables to determine the lifetime of the lower plate.

Upper Plate Analysis

The upper plate overall stresses are evaluated in the analysis
59 of the overall model. A 2-D finite element model of this area is used
and transient solutions obtained for both the governing transients.
In addition, the seismic stresses and steady state thermal stresses
59 from the overall model were superimposed on the detail model.

Column Center Region

The creep-fatigue damage in the column center region due to the critical transient combined with the seismic loads was calculated. Inelastic stress analysis is performed with a through thickness 1-D model applying the critical thermal transient combined with the highest possible seismic load.

Shear Webs

The shear web and its attachment to the lower and upper plates
51 have been considered in the gross model.

Core Barrel Key

The core barrel key is included in the gross model used for seismic analysis. In addition, detailed creep-fatigue analysis of the Haynes 273 hardfacing on the keys is performed using the results from a 3-D finite element model. The thermal stress analysis is based on the critical transient. The creep-fatigue evaluation used a fatigue curve derived from monotonic tensile properties using the "Method of Universal Slopes". Additional structural reliability was generated by a test program which included both low cycle fatigue due to reactor transients and high cycle fatigue due to normal load fluctuations.

Instrumentation Post

The analysis of the instrumentation post considers steady state and U-2b, U-18 and E-16 transient thermal loadings, striping, seismic conditions, loss of hydraulic holddown loads, and loads induced in the instrumentation posts by bowing of the UIS lower plate liners. Appropriate 2-D finite element models are used to represent different portions of the post.

Upper and Lower Shroud Tubes

The upper and lower shroud tubes are both Inconel 718 due to the severe steady state and striping environments. The lower shroud tubes are double wall construction in order to minimize the thermal stresses induced by the severe steady state gradients. These thermal loadings have been defined through the use of small scale thermal hydraulic models. The results of the thermal stress analysis considering both reactor transients and the interference fit between the mating tubes shows an acceptable creep-fatigue damage. These analyses are performed using 2-D finite element models and elastic analysis.

Chimney Assemblies

The Inconel 718 chimney shell was analyzed for steady state and transient conditions. The critical part of the assembly is the spider forging which was analyzed using a 3D finite element model for both steady state and thermal transients.

Mixing Chamber Thermal Liners

The mixing chamber thermal liners are Inconel 718 and are analyzed for the steady state and transient environment using simplified analysis techniques. In general, these parts have low
51 cumulative damage by comparison to other parts.

IVTM Port Plug and IVTM Port Plug Cover

The principal loadings for the IVTM Port Plug and IVTM Port Plug Cover are due to seismic events. Dynamic analysis using finite element models is performed for this condition and the results have been used to design the attachments to the reactor
51 head.

51 4.2.2.4.3 Core Restraint System Evaluation

4.2.2.4.3.1 Summary of Results

The adequacy of core restraint system performance is based on comparison of core restraint system model results with the requirements defined in Section 4.2.2.1.2.8. The core restraint system results include the effects of uncertainties in the model environmental data, material properties and core component dimension. Uncertainties are combined statistically but always in a way to produce a conservative result relative to the defined requirement. The following comparison of core restraint performance to the system requirements utilizes data for the first core of CRBRP.

a) Reactivity

Feedback reactivity attributable to core assembly motions is assumed to occur either predictably, as a result of power-to-flow variations during the reactor startup process or unexpectedly, due to the suddedn movement of the core assemblies within the confines of load plane gaps illustrated in Figure 4.2-84.

As discussed in Section 4.2.2.4.3.3, the reactivity feedback due to predictable assembly motions such as those which occur during reactor startup, may be positive or negative. Performance results show that the positive feedback effect is limited to power-to-flow ratios less than 0.7 and that the rate of positive reactivity insertion is a function of the core assembly duct temperature patterns, assembly motion reactivity coefficients and the lifetime status of the core assemblies. Assembly motion reactivity patterns with uncertainty effects included are provided for use in the evaluation of the reactor control system, plant protection system and reactor stability. These evaluations demonstrate that Criteria 9 and 10 in Section 3.1.3.1 are satisfied.

The analysis of sudden reactivity insertions utilizes conservative assumptions with regard to assembly positioning, interaction and dimensions which although plausible are all extremely unlikely. The event is also assumed to occur at the worst time in assembly life. This very conservative design procedure is used to set the cumulative load plane gap at the ACLP so that a sudden (or step) reactivity insertion at full power no greater than 60¢ is predicted. This design procedure insures that the step reactivity insertion limit at power levels of one megawatt or less is automatically satisfied.

b) Core Component Contact Loads and Distortions

The core restraint system model described in Section 4.2.2.4.3.3 is also used to predict interassembly contact loads, core to peripheral support component loads and overall assembly distortions 59 arising from radiation induced creep and swelling in the core assembly ducts. Uncertainties in dimensions, environment, material properties and the model are all considered to arrive at enveloping contact loads. Contact loads are predicted for both normal and non-seismic off-normal events for both on-power and off-power (refueling) conditions. Non-seismic loads obtained from this model are combined with seismic and other loads as indicated in Section 4.2.1.3.2.3.3. These combined loads are used in the structural evaluation of reactor system components which interface with the core restraint system. Loads calculated at refueling conditions are utilized to demonstrate that the refueling limit for assembly insertion and withdrawal is satisfied. System model results predict core assembly bowing and dilation effects as influenced by the environmental conditions (temperature and flux) within the core. Core assembly distortions are predicted to be sufficiently small so that the interassembly contact limits and the assembly handling envelope limits are satisfied.

c) Top End Misalignments

The evaluation of misalignment of the core assembly handling socket is based on stackup of top load plane gaps from a given assembly to the farthest location on the upper core former ring. Thermal and dimenstional uncertainties and permanent component misalignment effects are also included. This very conservative procedure is used to set the top load plane gaps to insure that assembly top end misalignment limits are satisfied.

d) Duct-To-Duct Contact

Interassembly contact at non-load plane locations due to the combined effects of bowing and duct dilation is also computed. Present evaluations indicate that local non-load plane contact between assemblies will initiate between fuel assemblies during the second cycle of irradiation. The results show, however, that no general duct-to-duct non-load plane contact pattern is established.

4.2.2.4.3.2 Material Properties

The analyses of this section utilize material properties found in Reference 181 for irradiation swelling and creep. Analyses were performed using nominal forms and with uncertainties conservatively applied.

4.2.2.4.3.3 Analysis of Assembly Bowing and Duct Dilation

Core restraint analyses were performed using the NUBOW-3D core restraint system analysis computer code (see Appendix A). The code analyzes a 30° sector of the core restraint system including irradiation and mechanical
59 influences as shown in Figure 4.2-85.

Assembly duct temperature data and neutron flux data used in the
53 core restraint analysis are shown in Section 4.4.3.3.5, "Core Assemblies Duct Temperatures", and Section 4.3.2.9, "Vessel Irradiation". These data were applied over a time period corresponding to two cycles of reactor operation to determine the thermal, irradiation swelling, and creep bowing response of the modeled assemblies. Examples of assembly bowing profiles and interassembly loads are presented in Figures 4.2-88 and 4.2-89 for the following conditions in the operating cycle:

1. At power, start of cycle one.

2. At power, end of second cycle (328 days).

Assembly Bowing and Interassembly Load Patterns

51 Influencing the bowing profiles in the time domain were the effects of irradiation swelling and creep. The effect of irradiation creep is to relax loads caused by on-power thermal bowing, while swelling act to bow the assemblies in the direction of increasing lateral thermal and flux gradient. Figure 4.2-90 illustrates the interassembly load pattern at ACLP due to on-power thermal bowing loads. Irradiation creep effects occur almost immediately, however, a delay or incubation period is required before swelling effects become pronounced. For the row 9 assembly, swelling became significant during the second cycle of operation (compare Figures 4.2-88 and 4.2-89).

Sudden Core Radial Motion

The presence of interassembly gaps at the above core load plane gives rise to the potential for inward radial motion of the core assemblies. A positive reactivity insertion due to assembly motion requires a general radial inward movement of the core. A conservative design procedure employed in CRBRP is to set the ACLP load plane gaps so that the maximum sudden inward motion of the core assemblies within the confines of the ACLP radial gap would result in a step reactivity insertion no greater than 60¢ when the reactor is at full power. The assembly motion necessary to cause a step insertion of this magnitude is improbable. During reactor startup, the establishment of temperature gradients across the assemblies will cause them to bow generally inward tending to close ACLP gaps in the core design in a predictable and controllable manner. Once the ACLP gaps are closed, further inward motion of the core assemblies is not possible. Consequently, the only possible way for a sudden inward core motion to take place is for the core not to compact radially inward at the ACLP as it is brought to power. The only non-compaction effects which have been identified and experienced in core array mechanical interaction testing are the combination of high friction (μ>0.6) and assembly azimuthal rotation.

Consequently, in the analysis of the step insertion event it is assumed that during the insertion of assemblies into the core, the assemblies
59 are rotated uniformly so that the interassembly load plane gap is eliminated.

It is further assumed that the friction coefficient is sufficiently high that thermal bowing forces generated as the reactor is brought to power will not be sufficient to overcome load pad frictional forces and realign the assemblies. Secondly, it is assumed that when full power is achieved, a seismic event occurs which produces forces sufficient to overcome the load pad frictional forces and realign the core assemblies into their nominal orientation. The rotational alignment of the core is assumed to occur suddenly, displacing the core radially inward.

The conservatism of this procedure can be illustrated by examining the assumptions which are employed in the step insertion procedure.

1) Core assembly load pads and the core formers are assumed to be at their nominal dimensions. Fabrication experience indicates that the as-built assembly load pad dimensions in the aggregate will be larger than nominal and that the core former ring smaller than nominal thus the as-built load plane gap is likely to be smaller than the nominal load plane gap. Furthermore, tolerance effects between adjacent assembly faces would result in the smaller of the distribution of interassembly gaps controlling the compaction process. The net effect of load pad tolerances would be to reduce the load plane gap.

2) Load plane surfaces are coated with a low friction hard surface coating of chromium carbide. The mean friction coefficient of this surface coating in the reactor operating range is 0.2 to 0.4. Analytical and experimental evidence on core array mechanical simulations indicates, that for this range of friction coefficient, non-compaction effects are not significant.

3) The rotational alignment of assemblies within the core is expected to be distributed statistically about the nominal orientation as determined by the core assembly and core former load plane as-built surface dimensions. No operational bias has been identified which could preferentially orient the core assemblies so as to close the load plane gap but still permit installation and removal of assemblies into and from the core.

Reactor Assembly Bowing Reactivity

During a change in reactor power-to-flow ratio, temperature gradients change or develop across assembly ducts, causing the assemblies to bow. Lateral motions of the core regions of these assemblies result in a reactivity change. This reactivity change differs from that discussed in the previous paragraph in that it is assumed to occur in a predictable and controllable manner in response to duct temperature changes.

Figure 4.2-92A depicts the row average assembly bowing patterns and corresponding reactivity effects that develop during a power to flow ratio

59

transition. At near-zero power-to-flow ratios, the assemblies tend to bow freely within the constraints of the interassembly and peripheral load plane gaps. The presence of a peripheral gap at the TLP core former ring permits a net outward motion of the core region of the outer core fuel and radial blanket assemblies producing a negative reactivity effect as shown for the 0.2 power-to-flow ratio pattern. When the closure of TLP gaps from the outer blanket rows to TLP core former ring prevent further outward motion, the core regions of the outer core fuel and radial blanket assemblies bow inward as shown for the 0.4 power-to-flow ratio pattern. The net inward motion of the fuel and radial blanket assemblies continues until the ACLP gaps close from the center of the core to the radial blanket rows. During this phase the bowing reactivity contribution is positive. Subsequent increases in power-to-flow ratio result in more complex S-shape bowing patterns and a net outward motion of the active core regions of the high worth fuel and radial blanket assemblies. During this phase, the bowing reactivity contribution is negative as depicted for the power-to-flow ratio equal to 1.0 pattern in Figure 4.2-92A.

43 Figure 4.2-92B shows the bowing reactivity characteristics which are predicted for nominal thermal, nuclear and load pad dimensional data at various times in the fuel assembly lifetime. Differences between the bowing raectivity patterns are attributable to the irradiation effects of creep and swelling in the reactor assembly ducts. Early in fuel assembly life (125 and 250 days in Figure 4.2-92B), the high worth assemblies at the fuel-radial blanket interface are bowed outward at the ACLP at refueling conditions as a result of irradiation creep. On a subsequent reactor startup (power-to-flow transition), these high worth assemblies traverse inwardly to a greater extent than initially straight assemblies. Later in the fuel assembly lifetime (500 days in Figure 4.2-92B), the high worth assemblies at the fuel-radial blanket interface bow inward at the ACLP at refueling conditions as a result of swelling. The startup bowing reactivity life in the fuel assembly life resembles the characteristic behavior of initially straight assemblies.

Figure 4.2-92C shows the bowing reactivity characteristics which are predicted for various assumptions of nuclear and thermal data, load pad mechanical interaction and dimensional uncertainties.

The curves in Figures 4.2-92B and 4.2-92C, when combined with other significant reactivity effects such as the Doppler effect, are used in the reactivity feedback evaluations which are provided in Sections 4.3.2.8, Reactor Stability, 7.7.1.2, Reactor Control System and 15.1.4.5, Reactor Assembly Bowing Reactivity Considerations.

Withdrawal Loads at Refueling

The frictional components of assembly withdrawal loads are obtained
59 from the NUBOW-3D analysis. The effects of uncertainties are combined

statistically and the sodium bouyant assembly weights are added to the friction forces as shown below:

$$F_w = \Sigma\mu_i \, (\Sigma F_j)_i + U + W$$

where: F_w is the withdrawal load μ_i are friction coefficients at load planes, F_j are the duct normal forces at load plane i, U is the uncertainty adder and W is the assembly sodium buoyant weight.

Midcore Duct-to-Duct Contact

Duct-to-duct contact at the core midheight was investigated via NUBOW-3D. It was found that assembly bowing alone was not sufficient to cause midcore contact. However, with duct dilational behavior modeled, midcore contact does occur in the outer core region. Nominal calculations predict the first occurrence just after the end of cycle 2 (~335 vs. 328 days). This is a duct midheight pressure bulge-to-pressure bulge type of contact from internal pressure driven creep dilation, and is not likely to cause high
59 contact loads.

Conclusions and Future Work

Assembly motion reactivity effects are conservatively predicted by current analytical procedures. The results shown indicate that reactivity
42 related core restraint requirements of Section 4.2.2.1.2.8 are satisfied.

Core component contact loads and distortions are predictable using current analytical methods. Additional work is planned to verify dilation induced duct-to-duct contact predicted in NUBOW-3D with more detailed models.

Additional areas where further work is planned include:

1) Detailed analysis of core restraint performance beyond core 1.

2) The simulation of fuel management in the NUBOW-3D model.

3) Verification and improvement if necessary of the duct dilation induced duct-to-duct contact model.

4.2.2.4.4 Removable Radial Shielding (RRS)

The removable radial shielding is in a preliminary phase of design; thus, stress analysis taking into account the effects of environmental conditions has not yet been completed. Analysis will be conducted on the following considerations: thermal stresses and strains, refueling and handling stresses, strain limits for brittle material, and effects of irradiation-
59 induce swelling and creep on the core restraint system.

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tation is that the surrounding structure impacts the driveline and absorber at the guide points, thereby generating impulsive lateral interaction loads which characteristically exist for less than a millisecond. Specific loading effects at the point of impact related to the presence of the coolant and the local dynamic deformation of the material cannot be separated uniquely. However, both are embodied in an "Effective Coefficient of Friction", applicable to impact load condition, which is expected to be substantially less than the design value of a 1.0. A task, described below, was performed to determine this "Effluent Coefficient of Friction".

Coefficient of sliding friction test data from sources external to CRBRP indicate that friction coefficients for material couples occurring in the control rod systems are substantially below 1.5 (Ref. 40). These tests were performed using a pin slider on a flat plate under steady loading. These data, contained in References 79 and 80, are summarized in Table 4.2-36A. This table contains data taken at temperatures ranging from 400°F to 1160°F and represents the maximum observed dynamic friction for each couple. The ETEC data for Inconel 718/718 did not represent the maximum observed value during sliding, but rather the combination of the initial and final values. These data appear to be inconsistent with the other data for Inconel 718/718.

To resolve this inconsistency and to create a large body of applicable data for a variety of material couples appropriate to the control rod systems, a series of tests on the sliding coefficient of friction using a pin slider on a flat plate have been performed. Thse data, presented in Table 4.2-36B, represent the average value of the sliding coefficient of friction for each material couple for a variety of temperatures, pin pressures and over two stroke lengths. For certain couples such as Inconel 718/718 and Inconel 718/316SS the complete tests were repeated and two lines of data appear in Table 4.2-36B. Average values of the sliding coefficient of friction are presented because these data are more appropriate to the conditions that exist during scram than the momentary peak values or the initial and final values presented in Table 4.2-36A. Also, data in Table 4.2-36B has been presented as a function of temperatures and pin pressure to allow review of these functional variables. Since the data in Table 4.2-36B show no significant dependence on temperature, the upper three sigma data have been averaged over temperature for each of the material couples in Table 4.2-36B. The results are shown in Table 4.2-36C for comparison with the prior, less extensive Table 4.2-36A. These data demonstrate a dependence on pin pressure with the lower pin pressures having the higher coefficients of sliding friction. This pressure dependence occurs frequently in pin on plate measurements and may not be a material couple characteristic. There also appears to be a slight correlation with stroke length. In this case, the longer stroke length of .750 inches would be more appropriate
59 for the control rod system during scram.

In summary, the data presented in Table 4.2-36B and 4.2-36C represent the most recent and most appropriate values of sliding friction data for the material couples used in the control rod system. The upper three sigma values of Table 4.2-36B, as selected for the most appropriate analysis conditions, are the recommended values for design analyses.

Tests were performed to determine the effective coefficients of friction under dynamic impacting conditions such as would occur between the PCRS fixed boundaries and the translating elements during a seismic event. In these tests, the effective dynamic coefficient of friction was determined by solution of the equations of motion based on measured impact loads and drop times of test rods subjected to lateral dynamic excitation (Ref. 182). These tests were simplified simulations of the circular PCRS driveline and hexagonal control rod oscillating within their respective constraints. To reduce complexity but still retain the principal dynamic effects on the coefficient of friction, the test articles were a straight circular rod traveling through three bushings, and a hexagonal rod traveling in a hexagonal duct. Bushing clearances were chosen to provide both lateral and rotational impact under lateral dynamic input similar to PCRS driveline response to seismic excitation. The hexagonal rod to duct clearances were typical of PCA design clearances. Material couples were prototypic of the PCRS (i.e., Inconel 718 on
51 59 Inconel 718, circular, and Inconel 718 and 316 SS, hexagonal).

59 The test media were air, water and liquid sodium. Only the
circular rod was tested in sodium to determine the differences, if any,
between the water and sodium media.

The input was a simple sinusoid at a frequency typical of the PCRS
predicted response to seismic excitation. Different frequencies and
59 amplitudes were tested. Input was applied to the test vessel simulating
the PCRS fixed boundaries which are excited in an earthquake. Bushing
59 and hex duct loads were monitored by strain bolts to determine the drop
rod impact magnitudes.

59 The tests have been completed. Table 4.2-36E summarizes the
resultant coefficients of friction. It can be seen that the dynamic
coefficient of friction is significantly less than the 1.0 design value
51 specified in the absence of corroborative test data for a lower value.
59 The effective friction coefficients from these tests are used for seismic
scram insertion speed analyses.

Antigalling Characteristics

Table 4.2-36D summarizes the FFTF Prototype Control Rod System Tests and the completed CRBRP Prototype Control Rod System Tests. Throughout all these tests, there were no failures to scram and no galling was found except for a slight effect in the FFTF tests between the control assembly wear pads and the outer duct which did not impare scram performance. For CRBRP, this wear effect was eliminated by including a rotational joint in the control rod shaft. In all tests except the CRBRP Unlatching Test, the environment simulated a reactor environment of liquid sodium with an argon cover gas. In all cases the sodium pool temperature ranged from 400°F to
51 1100°F. In the CRBRP Unlatching Test the liquid medium was water and a
dummy weight was used to simulate a PCA.

The prototypic FFTF CRDD accumulated a total travel of 23,635 feet and 828 scrams, and the CRD/CA disconnect coupling was operated 50 times. When Phase 2 of the CRBRP tests have been completed, all CRBRP Prototype Units will have exceeded the CRBRP requirements of 17,000 feet of travel and 750 scrams over the 30 year life of the mechanism.

These test results for the PCRS design and associated material
couples show negligible galling. Consequently, galling will not significantly
59 affect PCRS performance.

Irradiation Stability

As stated in an earlier paragraph, the friction coefficient was only slightly affected by sodium exposure and there was no effect due to irradiation. In addition, the control rod drive mechanism and the control rod driveline fluences are too low for the radiation to affect their performance. Core assembly irradiation induced swelling has been conservatively included in the design process to establish the clearances for control assembly components.

Based on the above test data, subsequent analysis and utilization of this data in the CRBRP design, the operability of the reactivity control systems and their disassembly after reactor operation seemed assured. However, to provide the desired assurance, additional testing described in Section 4.2.3.4 and Appendix C, Section C.5.1.2 has been established for the CRBRP Primary Control Rod System. These tests will
51 provide additional data regarding the operability of CRBRP at LMFBR operating temperatures.

4.2.3.1.4 Positioning Requirements

The positioning requirements for the control rod systems are:

1. Both the primary and secondary control rod system shall each provide two independent position indication systems and a means for verification of coupling and disconnect between the driveline and control rod.

2. Each control rod system shall provide capability for measurement of scram insertion times for individual control rods.

3. One of the position indication systems for each control rod system shall have a minimum indication accuracy of ±0.5 inch for the full-in and full-out position of the control rods and ±1.25 inches over the full control rod stroke. These accuracies apply to the positions of the translating assemblies (drivelines)
48 relative to the CRDM housings.

4. One of the primary control rod system position indication systems shall provide an accuracy of ±0.15 inch for the leadscrew relative to the full insertion position.

5. One of the secondary control rod system position indication systems shall provide an accuracy of ±0.5 inch at the full-in,
48 withdrawn operating and refueling positions.

Two independent position indication systems are provided for each system to give positive verification of control rod position and a means to check operation of each system by comparison with the other system. These systems are expected to monitor the positions of the control rod drivelines (leadscrews). Consequently, an additional indicator is provided to verify connection and disconnection operations between the driveline and control rod.

Testing capability for control rod scram performance is planned for all plant conditions between cold shutdown and full power conditions. Measurements of individual control rod scram insertion times are required to ensure this capability and to provide periodic checks for abnormal control rod performance.

Position accuracy of ±0.5 inch at full insertion is provided to verify the fully inserted positions for reactor shutdown and to assure insertion positions for control rod disconnect and subsequent refueling operations. Accuracy in the fully withdrawn position is specified to assure adequate positioning for potential scrams from the parked position. These positions are also used for safety interlocks with the reactor control and refueling systems.

The primary control rod system is used for establishing criticality and subsequent power control operations. While the rod position indication is not fed back directly to the reactor control system, the operator utilizes the position data to evaluate the plant and to interpret reproducibility of reactivity control. The relative position indication accuracy of ±0.1 inch leads to reactivity reproducibility of approximately 1¢ for the highest worth rod in the primary system. In addition, the position indication is utilized for logic interlocks and alarm as described in Section 7.7.1.3.

4.2.3.1.5 Structural Requirements

Control Rod Drive Mechanisms

The primary and secondary control rod drive mechanisms are designed to the following classes of components:

1. ASME Boiler and Pressure Vessel Code, Section III, 1974 edition, Class 1. For the primary control rod system, the mechanism motor tube, motor tube hold-down ring, nozzle extensions and position indicator housing form a part of the pressure retaining boundary. For the secondary control rod system, the extension nozzle, the hold-down ring, the upper SCRDM housing, the upper portion of the lower SCRDM housing, and the connector plate form a portion
59 of the pressure retaining boundary.

2. Seismic Category I. The control rod systems are required to remain functional and shutdown the reactor in the event of an SSE. (See Section 3.2.1 for detailed discussion).

3. Safety Class I. The control rod systems are categorized as Class I because of their control and shutdown functions. (See Section 3.2.2 for detailed discussion).

The primary control rod drive mechanisms shall be designed to the load conditions of Table 4.2-37. For these loading conditions,
51 pressure boundary components shall meet the structural requirements of Section III of the ASME Pressure Vessel Code together with applicable code cases and amendments to the code by RDT Standards. The portion of the Secondary Control Rod System that is coded in accordance with the ASME B&PV code and hence forms a part of the pressure retaining boundary shall be designed to the load conditions of Table 4.2-37. The structural requirements of Section III of the ASME Pressure Vessel Code together with applicable code cases and amendments to the code by RDT Standards shall be met.

The governing stresses in the mechanism are the time independent effects of primary mechanical loads, secondary thermal loads and fatigue. Use of the methods of these codes together with consideration of material effects such as carbon and nitrogen depletion, thermal aging, and environmental correction factors to account for material interaction with sodium leads to conservative structural designs of the mechanisms.

The primary and secondary control rod drive mechanisms shall have a design life of 30 years. This lifetime is consistent with the design lifetime of the reactor. Sufficient shielding shall be provided where appropriate to assure adequate strength to meet the structural criteria over the required lifetime. Interim maintenance will be required in order to achieve this lifetime.

The PCRDM and SCRDM shall remain structurally intact and attached to the reactor vessel, and shall not permit sodium leakage under Structural Margin Beyond the Design Base conditions. (See Reference
56 | 51 10a, Section 1.6). This requirement provides added margin of safety for an event for which no causative mechanism is known.

The PCRDM and SCRDM shall be designed such that no mechanical failure can result in any parts becoming missiles.

Control Rod Driveline

The primary control rod driveline (PCRD) and the secondary control rod driveline (SCRD) shall meet the intent of the structural requirements of Section III, ASME Pressure Vessel Code, together with applicable Code Cases (1592), and amendments by applicable RDT Standards. The stress and stability criteria for evaluation of the design shall be as specified in the above codes for all significant loading conditions including those identified in Table 4.2-37. Material physical property changes due to irradiation, thermal, and sodium environments shall be considered in evaluating the design.

The ASME Code specifies conservative allowable stresses for various loads and combinations of loads. The compressive load limit shall be no greater than 1/3 the buckling stability load of the driveline, and the design stress intensity limit shall be the lower of 1/3 ultimate or 2/3 yield stress. Satisfaction of the criteria assures that conservative margins exist for all conceivable loads including rod ejection for which no causative mechanism is known.

The design lifetimes of the primary and secondary drivelines shall be as shown in Table 4.2-38. The design lifetime requirements are conservative with regard to material considerations, taking into account the irradiation environments of these components (Table 4.2-39).

Thermal transient conditions are defined by the duty cycles described previously.

4.2.3.1.7 Material Selection and Radiation Damage

Material choices for control rod systems were based on prior experience and data from FFTF (Ref. 1, 40, and 41), and on the goal of
51 maintaining adequate mechanical properties in an atmosphere of inert gas, sodium vapor, and liquid sodium. Additional discussion of the interaction of the sodium environment with austenitic stainless steel is presented in Sections 5.3.2.2.2 and 5.3.2.2.3.

Control Rod Drive Mechanisms

The primary mechanisms utilize Type 403 SS in the motor tube and segment arms for its ferro-magnetic properties. In addition, both Types 403 SS and 17-4 PH are utilized in highly stressed areas such as the segment arms, and leadscrews, because of their high strength.
51 The secondary mechanism will utilize many different types of materials. The major load-carrying members are made from high strength materials suitable for the conditions.

Regarding Type 17-4 PH material, only the leadscrew will be fabricated from this metal and exposed to elevated temperatures. The 17-4 PH material being utilized in the leadscrew will be procured and heat-treated per ASME Standard SA-564 as modified by RDT Standard M7-6. The latter standard permits only two heat treat temperatures; 1100°F and 1150°F. The FFTF mechanism leadscrew was purchased and heat treated to the same specifications. In service, the maximum temperature experienced by the lower portion of the leadscrew, which is fabricated from 17-4 PH, will be in the temperature range of from 400 to 450°F. This maximum temperature will be experienced only when the control rods are fully inserted. During reactor operation, with the rods fully or partially retracted, the service temperature of the leadscrew will be less than the 400 to 450°F region. This latter condition should account for the major exposure time for this 17-4 PH material.

In selecting the 17-4 PH material for this application, the early experience (References 140, 141 and 142) with this alloy involving embrittlement was considered, especially when the material is aged at a relatively low temperature such as 950°F and subjected to a service temperature in the 600 to 800°F range. Based on the high aging temperature of 1100°F coupled with a service temperature of 450°F or lower, embrittlement was not considered a problem. It was concluded that below 550°F the embrittlement is minimal. The position is substantiated by the data presented in References 143 and 144. The exposure time required to cause room temperature charpy impact values to fall below 15 ft. lbs is shown in Figure 4.2-97A together with the exposure time to deplete room temperature impact values by one half shown in Figure 4.2-97B. This indicates that at temperatures below 500°F, essentially no effect on impact strength is observed.

Driveline and Bellows

Material choices for the drivelines and bellows were predicated
on maintaining adequate margins of strength in the high temperature
sodium and irradiation environment. Inconel 718 is utilized for the
54 PCRS to take advantage of its high strength and hardness and its favorable
wear and antigalling characteristics. The Primary CRDM bellows consist
of a main bellows, a disconnect actuating rod bellows, and a position
indicator rod bellows. All of the bellows are Inconel 718, and located
above the sodium level in a low-level irradiation environment. The main
bellows is cycled as the control rod system is moved. The outer two
bellows cycle only for the refueling or maintenance modes. Before installation
in the mechanism, all bellows are required to pass a 20 cycle breaking test
with subsequent leak testing. Table 4.2-36D summarizes FFTF and CRBRP
CRDM testing completed. These tests have been completed without a bellows
59 failure. Details of additional FFTF bellows testing is described below.

Assembly Discrimination

51 Two unique noninterchangeable discriminator types are provided for the control assembly discriminator types to preclude insertion of a control assembly into a non-control assembly core location and also preclude insertion of any other type of reactor assembly (fuel, blanket or removable radial shield) into a control assembly core location.
51 In addition, these two unique control assembly discriminator types preclude insertion of each type of control assembly into an incorrect control assembly location. The two types of control assemblies that require discrimination from each other are as follows:

1. Primary control assemblies that are to be located in Row 4 or Row 7 corners, only.

2. Secondary control assemblies that are to be located in Row 7 flats
51 only.

The geometries of the control assembly discriminators are similar to that shown in Figure 4.2-15 for the fuel assemblies. However, the diameters of the discriminator posts on the control assemblies are sufficiently different from the corresponding diameters on the other types of reactor assemblies to preclude incorrect insertion of an assembly during refueling. The discriminator system will stop the insertion of any assembly into an incorrect assembly location 2.35 inches short
51 of its fully inserted position. This resulting 2.35 inch height difference is detected by the In-Vessel Transfer Machine (IVTM) axial position locating system and the IVTM will not release the assembly from the grapple.

4.2.3.2.2 Secondary Control Rod System

The Secondary Control Rod System (SCRS) (shown schematically
51 in Figure 4.2-105) provides six two-position control rods that move from their full-out position to their full-in position to shutdown the reactor. Each control rod is held in its full-out position by a scram latch that engages the control rod's top end. The scram latch is located at the bottom end of a drive shaft extending through the reactor closure. The Secondary Control Rod Drive Mechanism (SCRDM) that positions the drive shaft and latch is mounted above the reactor head. The control rod has gravity and hydraulic pressure forces from the coolant pressure drop across the core which are applied to the control rod in a direction that tends to drive it into the core. The latch holds the control rod in the full-out position against those forces.

The SCRS hardware includes the following major items:

1. The Secondary Control Assembly (SCA) that has a hexagonal configuration similar to fuel assemblies and contains the movable control rod (Figure 4.2-106).

2. The Secondary Control Rod Driveline (SCRD) that extends through
59 the reactor closure and seals the reactor cover gas (Figures
4.2-107a and 4.2-107b).

3. The Secondary Control Rod Drive Mechanism (SCRDM) mounted to the
reactor closure top side and connected to the drive shaft seal
59 assembly (Figures 4.2-108a and 4.2-108b).

The SCRS utilizes hydraulic forces to assist scram action. The control rod moves axially within the control assembly guide tube. During normal reactor operation, the rod is supported above the core by the latch that is actuated by a pneumatic cylinder. Appropriate flow paths and orifices within the assembly allow the reactor coolant to flow from the high pressure plenum to the region above the piston. Sodium below the piston is ducted to the low pressure plenum.

Therefore, a pressure drop in the downward (scram) direction exists across the control rod piston continuously during normal operation. Upon receipt of a scram signal, the control rod is released by depressurization of the latch-actuating cylinder and is forced down by the hydraulic pressure force and gravity into the core region. Coincident with scram initiation, the primary coolant pumps are tripped, that is, turned off; however, the transient pressure decay is relatively slow with respect to the SCRS scram speed, so that sizeable scram assist pressure forces exist throughout the scram stroke. This relatively slow decay of the scram assist pressure due to the primary coolant pump coast down also applies when consideration is given to the postulated
53 undercooling accident.

The latch can be actuated to release the control rod by venting the pressure in the pneumatic cylinder. The weight of the rod and the hydraulic forces cause the drippers to spread and release the rod.

To retrieve the rod from the full-in position, the latch is lowered by the SCRDM. Upon engagement of the latch with the control rod, the pneumatic cylinder is pressurized to secure the rod in the latch. The SCRDM is then actuated to raise the control rod slowly out of the core region to its normal full-out position. In this position the latch is located just below the top of the handling socket as shown in Figure 4.2-105.

For head rotation, the control rods are released and the drivelines below the closure are withdrawn well into the upper internals structure to provide protection for them during head motion.

4.2.3.2.2.1 Secondary Control Rod Mechanism

Extension Nozzle - The extension nozzle that is attached to the
59 reactor head provides the support and mounting feature for the SCRDM and its attached SCRD and control rod. The mechanism holddown ring at the nozzle upper end attaches the SCRDM to the nozzle in a manner similar to that used by the primary drive mechanism.

Sealed Housing - A two part sealed housing (upper drive housing and
38 lower drive housing) forms the outer shell of the SCRDM that will contain the argon pressure that backs up the main shaft bellows. An integral flange at the housing's mid-section mounts and seals the SCRDM to the extension nozzle. The sealed housing upper end is capped off by the connector plate. The fixed end of the main shaft bellows attaches and seals to the lower end of the sealed housing. The sealed housing is pressurized through a connector in the top connector plate. The housing pressure is estimated to be approximately 5 psi above the reactor cover gas pressure. Should a bellows or any other gas seal fail, the leak will be detected, and an alarm sounded, by a system that continuously monitors argon flow in and out of the SCRDM.

38 Mechanism Frame - The main structural framework for the SCRDM is provided by the mechanism frame (see Figures 4.2-107b 108a and 108b). The frame is composed of two rails (beams), and plates, mid-supports, and a number of rail cross tie brackets. These components, bolted and keyed together, form a structurally rigid open frame. The rails provide the guidance for the coil cord assembly and the carriage assembly. The lower end plate contains the leadscrew bearings. The mid-supports contain the needle bearings and provide radial support for the leadscrews at their midpoint. The mechanism frame is coupled to the lower housing where the load is transferred to the extension nozzle. The coupling is made with a flange (formed by brackets at the frame mid-length) clamped to a ledge on the lower
59 drive housing.

Leadscrews - The axial motion for the drive is generated by twin bal-nut leadscrews. Bearing support is provided at three elevations. The lower bearings are in a duplex arrangement that takes the axial load and gives a rigid support for column loading considerations. The mid-support bearings use needle bearings for a thin cross section and provide rigid support. The positioning carriage ball-nuts travel between the lower two bearings. Each leadscrew is rated for 10,000 pounds column loading with rigid ends and 25,000 pounds tensile static load. The leadscrew has 0.2-in. lead per revolution. The upper bearings provide radial support at the gear drive.

Motor and Locking Device - Power to the leadscrews is provided by an
39 electric motor. Two idler gears from the motor pinion gear transfer the torque to the leadscrews. Two methods are provided for preventing the driveline and carraige from coasting downward. During normal operation (when power to the motor is available), the electrical braking capability of the
39 motor is used. During installation and refueling (when power is not available), the motor shaft lock device is used. This device, by mechanical means, prevents motor shaft rotation corresponding to the carriage downward direction. The lock device will maintain its functional state (locked or unlocked) without
59 the application of power.

Latch Mechanism - The latch mechanism is mounted to the upper side of the positioning carriage and provides the function of operating the control rod release latch at the end of the drive shaft. The main components of the latch mechanism are the position indication devices, the pneumatic actuator, and the scram valves. The motion of the concentric shafts inside the drive shaft are indicated by two LVDTs (linear variable differential transformers).

The pneumatic actuator utilizes high pressure argon in order to apply a tensile force to the tension rod to keep the latch locked. The scram valves that vent the pneumatic actuator, which in turn releases the control rod latch, are mounted below the pneumatic actuator. Guides located above the scram valves and on the positioning carriage engage with the split shell guide rails to provide a stable motion path for the positioning carriage and latch mechanism.

The scram valves are redundant and testable. A system of two-out-of-three logic permits any one of the three solenoids to be actuated for test purposes without causing scram. However, actuation of any two of the three solenoids is sufficient to release the control rod. The valves are instrumented to monitor response during testing. This feature permits periodic
39 testing of the scram valves thereby enhancing reliability of the system.

Electrical Lead Coil - The electrical cables needed for the latch mechanism LVDTs and scram valves and the pneumatic tubes needed for the pneumatic cylinder and driveline internal pressurization are routed to the latch mechanism by a helical coil assembly.

Position Indication - Two separate indication systems are provided
59 for indicating carriage position. The two systems consist of (1) a signal

converter driven by the motor input signal which displays a derived carriage position, and (2) a rotary encoder providing continuous absolute position indication (reference position is not lost with power failure) over the full stroke.

The rotary encoder's function is checked by redundant verifier switches actuated as the carriage passes a predetermined position. This initates a comparison of the encoder reading against a calibrated value. This checking function is intended as an advisory indication only.

The carriage position is also indicated by two proximity switches mounted at elevations corresponding to the control rod fully inserted position and the driveline refueling position. An upper limit switch stops the carriage at the position required for refueling. A lower limit switch stops the carriage at a position such that the lower driveline nose cone remains a short distance above the fully inserted control rod. An override permits the driveline to be driven down until the nose cone contacts the top of the
59 control rod.

59

Connector Plate - Hermetically sealed, electrical connector feed-
39 througns are threaded into the top plate and sealed by welding. Connectors
59 from the SCRDM cables can be attached to the feedthroughs on the lower side of the connector plate before it is mounted to its housing. Room is provided at the top of the drive to accommodate the cable length needed for this operation.

4.2.3.2.2.2 Secondary Control Rod Driveline

Latch - The control rod scram release latch is located at the bottom end of the driveline. The coupling head on the control rod is gripped by the latch gripper fingers that are held in the locked position by the latch tension rod. A 1/4-in. downward stroke of the gripper fingers and latch tension rod will release the control rod by permitting the control rod coupling head to slide out of the gripper fingers. The upper end of the latch tension rod is coupled to the pneumatic actuator discussused in Section 4.2.3.2.2.1. The pneumatic actuator has a stroke range of plus or minus one inch, as compared to the 1/4-in. for control rod release, thus allowance is provided for differential
39 thermal expansion over the driveline length. Since the latch tension rod is tensioned by the pneumatic actuator the pneumatic actuator piston can shift either up or down slightly to accommodate drive line expansions and contractions and still maintain a relatively constant force on the latch tension rod.

The angle surfaces contacting the gripper fingers are designed to operate over a coefficient of friction range of 0.2 to 2.0. The latch has been designed to be capable of carrying a 1,000 pound downward load on the latch fingers from the control element. Preliminary estimates indicate an applied load of approximately 450 pounds at reactor full power of which 345 pounds is developed by the hydraulic scram assist feature of the SCRS. The remainder
39 is due to gravity loads.

The presence of the control rod coupling head within the latch fingers is indicated by a mechanical extension of the sensing tube that contacts the top of the coupling head. The sensing tube has a stroke range of minus 0.5-in. and plus 1.0-in. from the latched position shown on Figure 4.2-107a.

Rigid Drive Shaft Section - At the minus 344-in. elevation, where the relative motion between the core assemblies and the upper core support structure is postulated, the drive shaft portion of the driveline has a thick
39 wall to protect the latch tension rod inside. The surrounding core support structure and control assembly handle are expected to deform before the drive shaft crusher, thereby keeping the latch tension rod free to release.

Bellows - The latch tension rod and sensing tube are sealed with metal bellows located at two elevations: the lower end of the upper driveline above the three-shaft coupling and the upper end near the positioning carriage. The volume inside the drive shaft between these two sets of bellows will be pressurized with argon from a source outside the SCRDM. The internal drive shaft pressurization is planned to be a few psi higher than the maximum pressure at the latch end, which is somewhat lower than the core inlet pressure. The drive shaft internal pressure is planned to be set at a fixed value that covers all reactor flow rates (thus, core inlet pressure levels).

Three-Shaft Coupling - The three shaft coupling (shown on Figure 4.2-107a) joins the lower driveline to the upper driveline. This coupling enables disassembly and assembly of the lower driveline without affecting the remainder of the driveline and mechanism.

Coupling of the tension rod is performed by butting the two ends together, engaging the split shell retainer halves in the tension rod grooves, and lowering the lock sleeve to encompass the retainer sleeve. The lower end of the lock sleeve is deformed to retain it over the split shell. The coupling concept used for the tension rod is also used for the sensing tube. The driveshaft connection is made with a coupler tube. The tube is bolted to the lower driveline. Attachment of the coupler tube to the upper driveline utilizes two sets of split rings and a threaded locknut. The locknut in
59 combination with the split rings couples the tube to the upper driveline.

Head Shielding - The head shield plug design is similar to the primary system shielding design.

Attachment to SCRDM - The SCRD attaches to the bottom of the SCRDM positioning carriage by a bolted flange.

4.2.3.2.2.3 Secondary Control Assembly

The control assembly consists of a hexagonal duct that is similar to the fuel ducts, a nosepiece, a shield, a guide tube, a control rod, and an upper handle. The principal dimensions of these components are summarized in Table 4.2-42.

The control rod is composed of 31 absorber pins surrounded by a circular wrapper and includes a piston assembly at the bottom and a damper assembly at the top. Each absorber pin contains a 36-inch column of enriched B_4C pellets, and the minimum total B^{10} loading in the
51 control rod is 4.7 kg. The absorber pin has a gas plenum above and below the pellet column with thin walled spacer tubes in each to maintain position. The upper plenum also contains a holddown spring.

The piston assembly has an inlet plenum region to provide good flow distribution for the coolant as it enters the piston region from the guide tube. There is a small clearance between the piston assembly and the guide tube above and below the inlet region. The lower region requires tight clearances to create high pressure drop for hydraulic scram assist. The upper region has small clearances to minimize leakage flow around the outside of the control rod.

The damper assembly above the absorber pins contain the damper and
dashram, a coupling head to mate with the latch, and an exit plenum for the
flow through the absorber pin bundle. The damper is made up of a series of
plates and belleville-type springs. Two types of plates are alternately
stacked with the springs. All plates have a hole in the middle to accommodate
a shaft. The shaft has a plate at one end and an arresting arm at the other.
After the rod has traveled 90% of the scram stroke, the arresting arm contacts
39 the downstop and starts to compress the stack of springs and plates. Sodium
between the plates is expelled, slowing down the rod. The exterior of the
housing containing the springs and plates is tapered to act as a dashram.
The dashram starts to slow the rod when it enters the guide tube by reducing
the annular flow area between the rod and guide tube.

A downstop is attached to the outer duct such that the boron carbide is fully inserted in the core when the control rod is resting on the downstop and the damper is compressed. The top end of the guide tube is also attached to the downstop, while the lower end of the guide tube contains a shield plug which has a sliding fit with the nosepiece. The guide tube has ports to provide upflow for cooling the control rod and downflow around the piston for hydraulic scram assist. The guide tubes' ports are adjacent to the inlet region of the control rod when the rod is in the withdrawn parked position.

The shield plug at the lower end of the guide tube has a hole
through the middle to permit downflow to the low pressure plenum. The shield
is sized to provide a steel volume fraction of 73% averaged over the 20 inch
shield height and the 4.76 inch assembly pitch. The shield plug also has
grooves in its perimeter to permit the upflow from the nosepiece inlet to
39 traverse the shield region.

The nosepiece contains two concentric flow regions. Coolant from the
reactor high pressure plenum enters the outer flow region of the nosepiece
39 through inlet ports. Sodium flows upward through the shield grooves and then
between the guide tube and outer duct until it reaches the ports in the guide
tube.

Part of the flow goes through and around the control rod to cool the absorber pins and exits the assembly around the latch seal. Most of the coolant at the guide tube ports flows downward around the piston, through the shield and inner flow region of the nosepiece, and into the reactor low pressure plenum.

4.2.3.3 System Evaluation

4.2.3.3.1 Primary System Evaluation

4.2.3.3.1.1 Alignment Analysis

53 This section presents the results of an evaluation of the forces which occur due to lateral misalignment of the CRBRP primary control rod system components. These forces act to retard control rod translation during scram and therefore must be considered in the Scram Analysis (Section 4.2.3.3.1.3).

All the effects contributing to drag forces (product of a lateral load and a coefficient of friction) were considered including lateral bending effects, torsional windup of the driveline, constant operational effects (e.g., out-motion limiter pawl ratchetting along the leadscrew) and PCA duct bowing. The lateral bending loads resulting from system misalignment were determined by a static analysis using the ANSYS finite element structural computer code (APP-A) to model the PCRS. The worst case misalignment of the fixed (non-moving) boundary of the PCRS was determined to be the refueling misalignment, Figure 4.2-95B. This misalignment envelope was further defined by determining the shape of the upper shroud tube, assumed to be initially straight, when forced to interact with a misaligned non-translating assembly (NTA) and constrained to the defined end points according to Figure 4.2-95B. The NTA, consisting of the upper bellows support, torque tube, shield plug, shield plug extension and shield tube, was assumed to be misaligned in the direction opposite to the direction of the installed shroud tube.

A finite element model of the translating assembly (leadscrew, driveline, dashpot cylinder and piston) was prepared and forced to conform to the envelope defined by the misalignments from Figure 4.2-95B design clearances and the analysis of the NTA above (see
53 Figure 109a). The translating assembly itself was assumed to be misaligned, Figure 4.2-109b, as determined by an analysis of tolerances of the component parts. This resulted in a system of static forces required to elastically bend the translating assembly into conformance with the fixed envelope (Figure 4.2-109c). Analyses as described above were performed for each significant withdrawal position
59 (6 total) from fully inserted to fully withdrawn.

Assumptions salient to the lateral misalignment force analysis were as follows:

1. The control rod system is at a uniform temperature which corresponds to the "worst case" refueling conditions for misalignments (see Figure 4.2-95B). Axial thermal expansion of the components
51 has been ignored.

2. Linear interpolation between the maximum misalignment values of
59 the top and bottom of the control assembly is valid.

3. The dashpot cylinder is forced over to one side of the shroud tube and parallel to the slope of the shroud tube.

4. The NTA and TA are initially misaligned as in Figures 4.2-109a & b, and these are the worst possible configuration.

5. The stiffness of the disconnect actuating rod internal to the control rod driveline is negligible when compared to the outer tube.

6. The sum of the absolute values of the reaction forces acting on the T.A. will be greatest (largest retarding forces) for the two-dimensional case with the greatest curvature in the elastic curve
51 of the control rod.

The total lateral loads calculated for each withdrawal position are summarized in Table 4.2-43.

The same analysis was performed for the PCRS sodium test misalignment conditions, which envelop the design basis in order to determine the validity of the worst case assumptions made. These assumptions were a) the position of the dashpot and b) worst case NTA and TA misalignments. Table 4.2-43a summarizes the analysis bases and the resulting total forces for the fully inserted position.

Subsequent system tests at the misalignments defined by Figure 4.2-95a resulted in a measured drag load of approximately 10 pounds. This confirmed that the dashpot moves laterally as designed (i.e. is not fixed as assumed) and that the assumption of combined worst case misalignment of the translating and non translating assemblies is extremely conservative.

Only slight variation in the calculated lateral loads was observed over the control rod stroke except near the full insertion position. The end of stroke drag forces are effective only over the last six inches where the control rod coupling can contact the inside diameter of the scram arrest flange.

The maximum coefficient of sliding friction from the available data (see Section 4.2.3.1.3) was used for each of the material couples in the primary control rod system. These coefficients are greater than the 3σ values given in Table 4.2-36A. Table 4.2-43 summarizes the lateral misalignment drag loads at each withdrawal position.

Torsional effects are minimized by the Rotational Joint at the top of the absorber bundle. Torque taker clearances and normal manufacturing twist of the absorber ducts can lead to twist of the driveline which is torsionally restrained by the Torque Taker at the top and by the hexagonal control rod at the bottom. The Rotational Joint provides the azimuthal degree of freedom to relieve the potential driveline torsion, as well as providing lateral degrees of freedom which minimize moment transmission between the shaft and the control rod. Tests performed on prototype Rotational Joints (Ref. 177) established that the maximum torque transmitted through the joint was 20.5 inch pounds. This was observed after an extended soak period in sodium and represented only a momentary "static" friction peak which immediately reduced to approximately 8 inch pounds torque. Because of the nominal cross corner hex diameter of 4.6 inches, the maximum torsional contribution to lateral load is 9 lbs. acting momentarily at the beginning of a scram.

The out-motion limiter pawl drag must be considered in the scram retarding forces acting on the translating assembly. This load results from the spring loaded pawls ratchetting along the leadscrew as it
51 scrams. The magnitude of this load is approximately 14 lb.

together with any outer duct bow, which must be convex relative to the core centerline at power operation, redistribute the bypass flow to reach a thermal equilibrium condition similar to a straight rod. Therefore, it is concluded that the worst case assembly bowing condition is achieved by assuming no operational bowing of the control rod, while permitting the outer duct to bow as dictated by the adjacent assemblies. Predictions of control rod bowing utilizing the maximum transverse thermal gradients predicted for the rod over a range of radial misalignments and rod withdrawal and using extremely conservative flux data have indicated that the rod bows less than 0.01 inch (deviation from a straight line connecting the end points) in a direction parallel to the outer duct bow. Therefore, the assumption of no control rod operational bow has been analytically confirmed.

Outer duct bowing is primarily dictated by interactions with adjacent assemblies. However, the thermal and flux gradients across the assembly do contribute to a small extent to the equilibrium bowed condition. For bowing predictions a combination of high flux and high temperature was conservatively assumed in the analysis despite the fact that these maxima occur at different rod withdrawal conditions and therefore do not concurrently exist in actual operation. In addition, a low flux case was assumed which maximizes the flux gradient across the adjacent fuel assembly, together with the further conservatism of assuming that the adjacent fuel assembly is initially pre-bowed for one cycle to cover the possibility of non-batch refueling.

The three assembly bowing model together with its assumptions introduces conservatism into the analysis. Since assembly interaction effects dominate control assembly duct bowing, the inherent assumption of two face stiffness leads to greater predicted bow. In-service, assemblies adjacent to the four duct faces not in the plane of bowing add stiffness to the assembly and reduce the magnitude of the bow. This conclusion is supported by preliminary data from the Core Restraint Test Facility, and will be further verified as more data becomes available.

Figure 4.2-110 is a predicted outer duct bow for 275 FPD operation in a homogeneous core layout configuration again based on flux
51 data which is conservative by a factor of approximately 1.5. Bowing analyses performed for heterogeneous core layout configurations have resulted in
59 greater end of life clearances between the control rod and the outer duct than obtained using Figure 4.2-110 results. Key elevations are indicated on this figure including the position of a fully inserted rod, the above core load pad (ACLP) and the top load pad. Since the clearance evaluation is a balance of available design clearances and reductions in these clearances from various effects, it is clear that the worst case (greatest reduction of clearance) is with the rod fully inserted. This can be determined by connecting the intersection of the control rod full-in end points with a straight line and gaging the deviation from this straight line to the bowed configuration. Clearly as the rod is withdrawn, the
51 deviation decreases, therefore the clearance increases.

Figure 4.2-110A is a plot of the outer duct bow adjacent to a fully inserted rod - i.e. the maximum deviation from the straight line noted above, superimposed on the clearance envelope of straight inner and outer ducts under the following assumptions:

a) The wear pads are in contact with the outer duct on the same side. (Thus the clearance at 80 inches and 0 inches is zero.)

b) The wear pads are at their maximum tolerance.

c) The inner duct OD is at its maximum tolerance.

d) The outer duct ID is at its minimum tolerance.

Additional assumptions relating to manufacturing are mac as follows:

a) The wear pads are azimuthally misaligned by the maximum tolerance.

b) The inner duct is non-straight by maximum manufacturing tolerance and is assembled so that the bow opposes the predicted outer duct bow.

c) Outer duct manufacturing bow tolerances have been assumed to be cumulative with the predicted operational bow. This is extremely conservative since analyses have shown that the end of life bowed configuration is not significantly altered by assuming an initially bowed configuration due to manufacturing tolerances.

The assumptions on manufacturing tolerances all reduce design clearances and are identified on Figure 4.2-110A. The following conclusions can be drawn from Figure 4.2-110A:

a) Two point (duct-to-du contact would occur under the assumptions of these analyses. T ad from this contact will be small due to the freedom given b rotational joint.

b) Three point contact does r occur. In fact, a large margin to three point contact exists as gaged against the maximum bow in 275 FPD operation.

Therefore, it is concluded that bowing will not detract from the scram performance of the PCA. The large margins and very conservative assumptions
51 lead to a high confidence in this conclusion.

4.2.3.3.1.3 Scram Analysis

This section describes preliminary scram analyses performed for the primary control rod system to demonstrate the expected rates of reactivity insertion during a reactor scram. Considered in this section are available shutdown reactivities, typical rod positions, control rod scram speeds and scram reactivity insertion rates.

Typical Rod Withdrawal Positions

Rod positions at the time of the scram may vary significantly
53 due to: withdrawal over the fuel cycle, potential variations in rod bank positions, uncertainties in rod worths and variations in the fuel cycle length between the first and later cores. The time to insert the first dollar of shutdown reactivity in the reactor scram is typically of greatest importance as this first dollar is sufficient to turn around the power peak or fuel temperature increase for most transients. Table 4.2-44 shows typical rod withdrawal positions at the beginning and end of the first six operating cycles. BOC-5 has been shown to be the worst case for the slowest first dollar insertion and is therefore the basis for the scram insertion analysis.

Control Rod Scram Speed

Control rod insertion analyses are performed by solving the equations of motion considering all the forces acting on the PCRS translating assembly, both scram assisting and scram retarding. Section 4.2.3.3.1.1 presented the analysis of scram retarding forces. Table 4.2-43 gives the total drag force as a function of withdrawal. The drag force is greatest between 6 inches withdrawn and fully inserted; however the loads are still less than the limiting load, which is equal to the driveline and control rod weight (~400 lb), which could cause failure to fully insert the rod.

Typical results of the above distance versus time insertion analysis, using the CRAB computer code, are given in Figure 4.2-112. This figure shows insertion profiles from various withdrawal positions, based on the drag forces given in Table 4.2-43. The CRAB code scram calculation methods have been checked against experimental data obtained in FFTF tests. Results of a typical comparison of calculation and experiment are given in Figure 4.2-113. It can be seen from this figure that the CRAB code predictions show excellent agreement with experiment.

Scram Reactivity Insertion Rates

Scram reactivity insertion rates have been calculated based on
51 the displacement/time profiles given in Figure 4.2-112, the cycle dependent

rod positions of Table 4.2-44, and the minimum rod worths of Table 4.3-32. Results of these calculations are given in Figure 4.2-114. BOC-5 is the worst case with respect to time to insert the first dollar despite the fact that the initial withdrawal height is greater than for BOC-4. The difference in minimum shutdown worth overrides the position effect for BOC-5 vs. BOC-4. All other cases insert reactivity faster due to higher worth or due to farther initial rod withdrawal. End of cycle insertions are significantly faster due to increased shutdown margins and rod withdrawal positions which lead to faster rod speeds due to greater scram assist forces at those positions. It is concluded that the primary control rod system satisfies the speed of response require-
51 ments given in Section 4.2.3.1.3 for the worst case rod positions.

4.2.3.3.1.4 <u>Seismic Scram Analysis</u>*

An analysis was performed to determine the effect of a safe shutdown earthquake (SSE) on the CRBRP Primary Control System's scram capability. Lateral contact forces on the translating assembly were determined for a severe three second segment of the SSE which was then used in evaluation of scram performance under seismic conditions.

The worst time to initiate a scram in this 3 sec. time interval was identified by determining the time required to scram 9 inches. This
59 criterion was used because it represents the required rod travel of the rods to insert approximately one dollar of reactivity. A 1.2 second load time history whose initial point is the worst scram initiation time was then used repetitively until the rods were fully inserted. A dynamic impact coefficient of friction of 0.5 was used since this value is conservative relative to the coefficient of friction averaged
51 over the length of the PCRS (see paragraph 4.2.3.1.3).

The ANSYS computer program was used to perform the seismic analysis, using the semi-linear transient dynamic (time history) option of the program. An overall reactor system model was first used to determine the motions of the important components. The gross motions of the system components were then used as input functions in a decoupled primary control rod system model to determine the response of the leadscrew, driveline and control assembly within the PCRDM, shroud tube and control assembly duct.

The nonlinear primary control rod system model and its use in the seismic impact analysis are discussed in Section 3.7.3.15.3. The results of this analysis used in the scram calculations are the contact forces (vs. time) during the seismic event.

*See footnote to Section 3.7.3.15.

The scram analysis was performed using the CRAB computer code (see Appendix A) incorporating the dashpot model and time variant scram retarding force capability. Calculations were performed for rod positions at the beginning of cycles. The results of the SSE scram insertion predictions are compared with the seismic scram requirements in Figure
51 4.2-119.

It is concluded that the primary control system satisfies the SSE scram insertion requirement of Figure 4.2-93. The reactivity effects of the slightly increased scram time are evaluated in Section 15.2.3.3.

The seismic scram analysis is a conservative evaluation of scram capability under SSE environment in that a conservative calculation of
51 loads and scram initiation time was employed.

4.2.3.3.1.5 Control Assembly Analyses

Absorber Pin

The primary control assembly utilizes enriched B_4C (approximately 92 atom percent ^{10}B in Boron). Data on helium release, thermal conductivity and pellet swelling, required for absorber design, are available
51 in References 44 and 44a.

Currently committed B_4C tests providing EBR-II irradiation data in support of CRBRP control assembly design are given in Table 4.2-46A. This table summarizes each test using the HEDL name for the test. Typical test parameters for pellet temperature, pellet diameter and B-10 captures are given. Also noted is the status of each test including target completion dates for the EBR-II irradiations.

The tests of Table 4.2-46A will extend the irradiation data well above the pellet temperatures and pellet sizes anticipated for the primary control assembly. The BICM-1 test has provided data to $80X10^{20}$ B-10 captures/cc, which is comparable of first core burnups for CRBRP. The BV-2 test for vented pins will provide data on pellet swelling for burnups typical of 275 FPD cycle operation. The tests of Table 4.2-46A cover the
59|51 operating range for the primary control assembly over its required lifetime.

The planned EBR-II B_4C irradiation tests do not include in-reactor transient cycling of absorber rods. Out-of-pile testing of irradiated pellets has been performed under the HEDL development program to determine gas release under transient thermal conditions. Preliminary results indicate that helium release upon temperature increases occurs over a relatively long time (on the order of 15 minutes) characteristic of a Primary Control Assembly thermal transient. Since B_4C temperature increases during transients are small (<100°F) the incremental gas
51 increase from a transient is a small effect. Incremental gas release

during transients based on the thermal transient tests are included in the pin lifetime analyses. Since the absorber pins are designed to preclude pellet to clad interactions or B_4C melting under worst case transient conditions, gas release is the only B_4C variable required to be assessed in transient analyses.

53 Further performance data for the PCA will be obtained from the PCA Irradiation Test (see Section 4.2.3.4.1.1) which will provide integrated lifetime performance data for near prototypic environments and operating
51 parameters.

51 Table 4.2-46 summarizes performance parameters for the absorber pins. The thermal-hydraulic parameters are discussed in Section 4.4. For the current design, the plenum lengths have been established by the maximum available pin length, and the clad stresses at the end of one operating cycle are less than 5,000 psi as shown in Table 4.2-46.

Preliminary strain analyses of the pin have indicated that there is only minimal accumulated strain at the end of the lifetime requirements. Additional analysis utilizing the cumulative damage function approach
51 has been performed which also verifies the lifetime capability of the pins. Use of the CDF for the absorber cladding requires that the duty cycle be separated into various stress state/time segments superimposed on the steady state operating conditions. This introduces conservatism in the analysis since conservative estimates of stress and time form the basis for the analysis. Effects such as sodium interaction with the cladding and pin-duct interactions are included in the lifetime evaluation. B_4C swelling is calculated to assure that no force contact occurs between the pellets and the cladding (see Table 4.2-36) thus reducing the margin for error in the calculations. Figure 4.2-111a shows pellet swelling and associated pellet to clad gap for rod in the Row 7 corner location. Figure 4.2-111b shows axial B-10 burnup profiles for each rod position in the equilibrium cycle.

Based on the results of the preliminary analyses performed, it is concluded that the pellet/clad gap clearance requirements are satisfied for the required 328 FPD lifetime with an initial gap of 0.028 inches (Figure 4.2-111a). The initial gap must be increased to allow for
54 53 51 additional pellet swelling over the goal lifetime of 550 FPD.

Structural Evaluation

A preliminary elastic analysis was performed to evaluate the structural adequacy of the control assembly outer ducts. Design stress
51 limits were derived using the criteria defined in Table 4.2-37B. Both

ductile and brittle failure modes were considered in deriving these criteria. Material data was taken from Ref. 1 for the worst case thermal and irradiated state of the critical duct sections evaluated, that is, the lower duct welds and the ACLP. Plastic analysis, including creep and swelling effects, is not expected to significantly change the result of the elastic analysis.

51 The results of the analysis for fuel ducts are presented in Tables 4.2-7 and 4.2-8. These results are applicable to the control rod duct since the control assembly utilizes the fuel ducts. It can be seen that positive margins exist for all loading conditions and stress categories. The control assembly duct does not attain the same temperatures as the fuel duct, thus the allowable stress and the margins of safety are greater for the control assembly. For the ACLP region of the duct, the increase in material allowable due to the lower control assembly duct temperatures, as shown in Figure 4.4-29, will raise the margins both for primary plus bending and primary plus secondary categories. In addition, the CDF will decrease from the already comfortable value to much less than 1 for both peak and seismic loading conditions.

51 Analyses were performed to evaluate the effects of scram impact on the control assembly and the outer duct. These analyses indicate that scram impact does not pose a problem for the control assembly duct.

The leadscrew-driveline-control rod assembly and the duct were modeled as a dynamic system (Figure 4.2-120). A driveline initial velocity
53 of 25 ips was assumed to provide added analytical margin over the 14 ips design final velocity of the driveline and the FFTF dashpot test results of 9 ips. The results of this preliminary analysis are as follows:

1. There is insignificant rebound (both in magnitude and frequency) of the control assembly following impact on the scram arrest flange. In addition, the outer duct does not appear to be overstressed. Additional effort is in progress to establish acceptability of the
51 end-of-life residual ductility.

2. The "breakaway" link (see Section 4.2.3.1.3) at the base of the control rod shaft is not dynamically overstressed upon scram impact. The dynamic weight of the control rod is less than the force at which the "breakaway" feature is designed to rupture
51 (16,500 pound predicted force, 19,000 pound maximum) to satisfy the stuck rod or stuck coupling requirement referenced above.

3. The impact of the disconnect coupling actuating shaft is slight and does not damage the rod for an initial velocity of 25 ips, and does not occur at the design velocity of 14 ips.

4. There are large margins of safety for the driveline expansion sleeve and the position indicator rod for an impact at 25 ips.

5. The control rod welds (adapter plate-to-duct and pin-to-adapter plate) have adequate margins of safety based on 25 ips impact velocity.

6. Absorber cladding stress is insignificant in scram impact.

7. There is sufficient absorber pellet preload to prevent pellet separation and interpellet impact.

4.2.3.3.1.6 PCRDM and PCRD Structural Analysis

51 The CRBRP control rod system utilizes a modification of the FFTF mechanism to benefit from FFTF design and development experience. A complete structural analysis of the FFTF mechanism is reported in Ref. 41. Review of this analysis and scoping analyses of CRBRP primary control drive mechanisms indicate that loading requirements except the seismic requirements, are very similar to FFTF requirements.

The mechanism is designed to withstand all loads stemming from
SSE, and safely shutdown the plant. Seismic analysis has indicated that
59 51 all seismic loading conditions are within allowable limits.

Preliminary analyses have been performed to determine the
structural adequacy of the control rod driveline, based on conservative
assumptions of minimum material properties and maximum loads on stressed
components. These evaluations considered the maximum load applied to
the driveline by the motor including a stuck rod, latching and a cold
stator. These loads are the peak tensile, compressive and buckling
loads, and envelope all other loads encountered by the driveline. Table
4.2-47 summarizes the applied and allowable loads and stresses for the
PCRDM and PCRD components for the peak loading conditions. Only the
59 maximum loading conditions have been addressed in this assessment of the
PCRDM/PCRD structural analysis. It can be seen in Table 4.2-47 that the
maximum load or stress for these conditions does not exceed the allowable
loads and stresses.

59

4.2.3.3.1.7 Overall System Performance Evaluation

Potential for Functional Failure of Critical Components

51 As stated in Appendix C, a Failure Model and Effects Analysis of the control rod system were performed. In this analysis, each basic

The required total actuation force to be applied to the tension rod is shown in Figure 4.2-126 as a function of friction coefficient for two cases: a) maximum allowable friction coefficient - 2.0, and b) maximum allowable friction coefficient - 3.08. Selected materials combinations will be tested to verify the expected friction coefficient. As reported in Table 4.2-50, the expected friction coefficient ranges from 0.37 to 0.73 for the candidate materials. Inconel 718 on Inconel 718 and chromium carbide on Inconel 718 at 1000°F, and depending on the situation; i.e., static, dynamic, or breakaway.

Avoidance of problems associated with self-welding depends upon proper material selection and upon achieving proper geometry of the surfaces so that contact stresses are limited to sufficiently low values that neither produce severe distortions nor favor self-welding. As noted in Table 4.2-15, Inconel 718 on Inconel 718 shows no self-welding tendency under the operating conditions expected for the latch. Similarly, chromium carbide on Inconel 718 exhibits relatively low breakaway friction coefficients as shown in Table 4.2-50. Breakaway friction coefficients are a measure of self-welding tendencies during long-term holds. Furthermore, the geometry of the contacting surfaces is selected to give relatively
39 low Hertz contact stresses, approximately 70,000 lb/in^2. The surface of the gripper pads in contact with the cam is spherical, and the cam is conical. The inner surface of the gripper, the one in contact with the coupling head, is cylindrical with a circular cross section whereas the coupling head is **spheroidal with a large radius in the vertical plane.** Because of this geometry between the coupling head and the gripper, the point of contact at this interface is not critical; the radius of curvature of the gripper pad is concave and uniform everywhere, and hence, regardless of the location of the contact point, the contact stresses are the same. Consequently, if manufacturing errors result in slight mislocation of the gripper with respect to the cam surface, and the coupling head and gripper contact point differ somewhat from the design point, virtually no change in stress will result. A thorough test program is planned for operation of prototype latches in an environment and under conditions typical of reactor operation. Post-test examination of the latch will divulge wear characteristics and, in conjunction with the friction coefficient measurements made during the test, will provide assurance that the latch is properly designed for the application.

Because self-welding is a diffusion phenomenon, higher temperature and contact stresses are generally considered to favor self-welding due to the fact that these conditions promote diffusion across the contact area. Latches typically sustain high contact stresses and so might be considered candidates for self-welding problems. However, it is important to recognize that relatively high contact stresses are accompanied by relatively large (on a microscopic scale) elastic strains and associated storage of strain energy. This strain is available to rupture any self-weld bonds that might have occurred during intimate contact. If it is assumed that a self-weld bond has occurred on the latch contact areas, the forces acting on the latch as a result of only the gravity forces on the control rod have been calculated to be capable of inducing tensile stresses on the 3

39 order of 1,200,000 psi on the postulated self-weld bond. Clearly such a high stress cannot be sustained by any self-weld bond, and the bond would be ruptured, and the rod would be released. Consequently, it can be concluded that even if self-welding were to occur, the consequences would be limited to slightly greater wear on the latch components due to adhesive wear characteristics. If for some other reason, latch release were not achieved, then the primary control rod system would shut down the reactor. 3

<u>Sensitivity of the Systems to Mechanical Damage</u>

The Secondary Control Rod System is being designed to withstand significant mechanical damage and still permit scram, wherever unknown loads
53 are remotely possible.

39

Analysis has shown that 1.15-in. of relative lateral deflection between the bottom of the head and the top of the control assembly handling socket must occur before significant loads can be developed. At that point, the bottom of the driveline will have rotated one degree from its vertical orientation. The control rod coupling head has been designed to accommodate this rotation without inducing loads into the control rod.

Assuming the case of two opposite control assembly load pads equally loaded at the corner points of curvature, the minimum ultimate capacity of the control assembly duct was calculated as 7100 pounds. An additional 200 pounds resistance is provided by the guide tube within the duct at the above-core load plane. Any other combination of loads on up to all six sides at the load pad region of the control assembly will produce a larger ultimate capacity value. For instance, loading the duct equally on all six faces requires a force of 41,900 pounds per face to reach the ultimate capacity of the duct. Thus, it is important that the tolerances at the load pad region be held as tight as possible to insure that the maximum load carrying capacity of the duct load pads is utilized.

Potential for Excess Reactivity Additions

The planned SCRS operation is to withdraw the control rods to their full-out position before the reactor becomes critical. During reactor operation, the SCRS control rods are always out of the core. When the control rod is in normal parked postion at the top of the core, the top of the damper
59|35 assembly is 0.75" from the bottom of the handle extension. Since the damper assembly cannot fit within the extension, the control rod cannot be withdrawn farther than 0.75 inch. beyond its normal withdrawn position. Movement of the
59 35 control rod from the top of the core to 0.75 inch. beyond the top of the core would be only a small reactivity addition. During shutdown, downflow provides a downward force on the assembly which increases as the flow increases. The operation and design of the SCRS precludes reactivity addition due to unexpected control rod withdrawal.

The design maximum withdrawal speed of the SCRDM is 9 ipm, which is the same as the Primary Control Rod System maximum design withdrawal speed. Reactivity insertion rates associated with this withdrawal speed have been
35 shown to be acceptable (Section 4.2.3.1.3).

Potential for Rod Ejection by Pressure Forces

The Secondary Control Rod System uses a net downward hydraulic pressure force for scram assist. This downward force is achieved by restricting the flow area at the top exit of the assembly (the latch seal), and by having flow communication with the low pressure plenum. The downward flow to the low pressure plenum undergo a large pressure drop when it passes through the narrow clearance between the piston at the bottom of the control rod and the guide tube. With this configuration, the downward hydraulic force on the control rod increases as the flow rate increases, so a flow increase does not present a rod ejection hazard.

The piston that creates the large pressure drop in the downward flow must be between the flow ports in the guide tube and the low pressure plenum to achieve the downward force. However, the control assembly design precludes the piston ever rising above the guide tube ports. There is more than 1.75-in. between the bottom of the port and the top of the piston, so the piston would have to be raised more than that amount before it could even start to affect performance. The clearance between the top of the damper assembly and the
35 59 lower end of the top handle extension is only 0.75". Even if the driveline were withdrawn beyond the normal parked position, the control rod would con-
35 59 tact the top handle extension at 0.75" and prevent any further withdrawal. The piston is thus prevented from being above the ports.

The seal around the driveline (the latch seal) maintains a large flow restriction at the top assembly outlet. This seal is the principal contribution to pressure drop in the upward flow. The seal cannot be withdrawn from the assembly without scramming the control rod because the damper will contact the handle extension. If the driveline is inserted so that the latch seal drops below the handle extension, the driveline still provides adequate flow resistance to prevent an upward hydraulic force on the control rod.

The only time the driveline is completely withdrawn from the assembly is during refueling, when the reactor is shut down and the pumps are operating at low flow. However, the control rods are not lifted even if the pumps are operated at full-flow conditions. This situation can only occur when the rods are fully inserted. With the driveline removed, more flow exits at the top of the assembly, and it passes through the narrow annulus between the dashram and the guide tube. The flow undergoes a large pressure drop at the dashram, and this creates a lifting force on the control rod. However, this lifting force is only 25% of the assembly's weight when the pumps are operating at full-flow conditions.

SCRDM Housing Leak

The details of the two seal boundaries forming and housing buffered seal and leak detection vary between the CRDM and SCRDM.

a) In the SCRDM the first seal boundary consists of three metal
59 bellows and two metal mechanical compression seals. These elements in conjunction with the drive housing structure and driveline structure, seal off the total area inside the drive mechanism nozzle. At the top inside surface of the drive mechanism nozzle, two metal mechanical compression seals are located with the housing pressure ported between the seals to form a
59 buffered seal configuration. Of this seal pair, the lower seal is adjacent to the reactor cover gas and is part of both the first
59 seal boundary and the upper seal boundary, which is described below. The drive housing pressure, which is higher than the reactor cover gas, acts as a buffer against the cover gas so that any leakage is into the reactor. To follow the first seal boundary path through to the drive centerline, the next portion

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59 is at the lower end of the lower drive housing. The drive housing
pressure acts across these seals with respect to the cover gas,
and thus acts as a buffered seal. The lower end of the long
59 stroke welded metal bellows attaches to the lower end of the drive
housing. The top end of the long stroke bellows attaches to the
driveline. The cover gas occupies the inside of the bellows and
the drive housing pressure surrounds the outside. The driveline
configuration extends across the inside of the long stroke bellows
to complete the first seal boundary. The driveline contains two
internal concentric shafts, one for latch actuation and the other
to indicate the control rod coupling head presence within the
latch. These shafts within the driveline are sealed with respect
to each other with two metal bellows at each end of the driveline.
The bellows at each end of the driveline allow it to be internally
pressurized to a higher pressure than the drive mechanism housing.
The second seal boundary, or the primary system boundary, consists
of the three metal compression seals (the upper seal of the nozzle
to housing seal pair, the lower to upper housing seal and the upper
housing to connector plate seal). The electrical and pneumatic
39 59 inputs through the connector plate are sealed with welded seals.

b) The volume enclosed by these two seal boundaries contains the
39 motor, leadscrews, and position indicators. This volume, like the
CRDM, is also purged with clean argon gas that is monitored to
detect leaks. The difference in monitoring is that in the SCRDM
the buffer gas flow rate is measured.

4.2.3.3.2.5 Previous Experience and/or Development Work with Similar Systems and Materials

Various control rod drive systems have received significant development and accumulated extensive experience through the years. EBR-II, Fermi, and FTR are the major U.S. programs that provide control rod drive experience for LMFBR conditions, with FTR being the most applicable.

EBR-II has operated for many years and accumulated reliable drive performance. The main shaft long-stroke bellows of EBR-II are located above the reactor head, as are the SCRS bellows, and the EBR-II experience has demonstrated that sodium vapor migration up the long narrow annulus to the bellows, is minimal and has not caused a problem. The EBR-II control rod is attached to the drive shaft by a gripper mechanism that even though its design is not the same as the SCRS latch and it does not release for scram, it still confirms that latch or gripper mechanisms can function reliably in liquid sodium.

The Fermi reactor utilized a latch at the drive shaft end to release the control rod for scram. After development, the final latch configuration performed well. The reported failure of a latch to scram was the result of a failed bellows exclusive of the latch mechanism. The Fermi latch design differs from the SCRS latch, but its success again adds to the confidence of using latch mechanisms for LMFBR control rod drive systems. The failed bellows prevented the Fermi latch from releasing by permitting liquid sodium to rise in the driveline. The sodium contacted air and reached a cooler elevation where a sodium freeze plug was created inside the drive shaft that prevented the latch release. Subsequent modification to the Fermi drives pressurized the drive shaft with argon and provided leak detection. The SCRS design has incorporated both driveline pressurization and leak detection.

The FFTF roller nut drive is the most recent drive development work, and it is also the CRBRP primary control rod drive. Several of its features will be applicable to the SCRS even though the SCRS is being designed to be diverse from the roller nut drive. The static metallic reactor core gas seals used for FFTF will be applicable for the SCRS. The design and testing experience on FFTF has provided candidate materials for satisfying difficult design requirements for operation in sodium. Inconel 718 was selected for the FFTF drive shaft material and is also planned for some of the SCRS driveline
39 components. The fuel assembly and control assembly load pads require a low coefficient of friction material that will withstand high compressive loads and not exhibit self-welding in high temperature sodium. For the FFTF, chromium carbide was found to be a suitable coating that can be applied to the stainless steel base material. This chromium carbide coating will be evaluated as a
39 candidate for SCRS latch parts. The FFTF main bellows have undergone development that will apply to the SCRS. The SCRS design has maintained the same outside and inside diameters as used for the FFTF bellows, so that the design will be directly applicable. The SCRS stroke is longer but the SCRS bellows do not have to follow the scram motion as do the FFTF bellows.

Boron carbide has been the near universal selection for the absorber material for LMFBR reactors both in this country and abroad. Both FFTF and the SCRS are utilizing this absorber material, so the design data and development experience will be applicable to the SCRS.

4.2.3.4 Testing and Inspection Plan

The testing and inspection plan described herein for the reactivity control system is divided into five areas which verify the design of the systems and the quality of the components installed in the reactor. These five
41 areas are an extensive performance test program, plant tests, surveillance, acceptance tests, and post-irradiation examination.

4.2.3.4.1 Performance Test Program

Extensive testing programs are planned for evaluation of the reliability and design of both reactivity control systems. These tests will include individual component tests and complete prototype systems tests.

4.2.3.4.1.1 Primary Control Rod System

The PCRS testing program consists of the following major testing activities:

A) Component Tests: The following component design test and/or analysis program was established to provide design verification of the PCRS components.

1. Dynamic Seismic Friction Test

This test was performed to evaluate the effective coefficient of friction between a rod and its guide bushings under impact
53 loading conditions. Data obtained are used to provide friction coefficients for seismic scram insertion analyses.

2. Control Assembly Hydraulic (Flow) Test

59 Test results will be used to verify the pressure drops, flow and vibration characteristics of the primary control assembly design under prototypic flow conditions.

3. Control Assembly Pin Compaction Test

Test has provided data to determine inter-pin and pin-to-duct loads for the primary control assembly analyses.

4. Control Assembly Rotational Joint Test

Test has provided performance data on the rotational joint which confirmed the reduction in control assembly wear and reliable operation of the joint.

5. B_4C Data Test

The base technology irradiation test program being conducted by HEDL includes acquisition of data required for design verification
51 of CRBR control assemblies (see Table 4.2-46A).

6. Friction and Wear Tests

The base technology materials test program being conducted at
53 ETEC and ARD provides data for the material couples selected
for fabrication of the primary control rod system.

7. Control Assembly Analytical Methods

Provides an analytical model calibrated with test results for predicting primary control assembly thermal-hydraulics performance, lifetime characteristics and scram dynamics behavior.

B) System Level Tests: A series of Primary Control Rod System Prototype Tests have been performed to verify that the Primary Control Rod System performance is consistent with its design requirements under design basis operating conditions. The Control Rod Drive Mechanism was evaluated in a CRDM Accelerated Unlatching Life Test. This test program verified the unlatch performance characteristics of a prototype primary control rod drive mechanism over twice the design lifetime travel and scrams. The Accelerated Life Test involved testing of a full size prototype primary control rod system in sodium, sodium vapor, and argon gas environments that simulate operations in the Clinch River Breeder Reactor Plant. Phase I testing in this series completed 1/2 of the PCRS lifetime scrams, 1/3 of the leadscrew travel requirement, and about 5 times the PCA travel requirements. Phase II of this series will extend total test scrams and travel beyond CRBRP lifetime
59 requirements.

C) PCA Irradiation Test: A PCA irradiation test is scheduled to be inserted in the FFTF for 600 FPD. The intent of this test is to provide near-prototypic irradiation performance data on the PCA
59 absorber assembly to support the PCA lifetime evaluations. The test assembly will contain 37 pins of enriched B_4C and will function as an integral part of the FFTF Secondary Control Assembly Bank. The parameters of the test assembly have been selected to provide data prototypic of the PCA for burnup, fluence, B_4C and cladding temperatures and cladding strain. Data from this test are expected
59 to be available in 1985.

D) Other Tests: See Appendix C for Reliability Test Program.

4.2.3.4.1.2 Secondary Control Rod System

The SCRS testing program consists of the following major
59 testing activities:

A) Latch Tests: Component development tests of the scram latch configuration for the secondary control rod system verified the design
59 of this component.

B) Damper Tests: Component development tests of the damper configuration for the secondary control rod system verified the design of this
59 51 component.

C) Position Indication Tests: Component development tests of the position indication system for the secondary control rod system will be performed to verify the design concept for this subsystem.

59 D) SCA Static Flow Tests: The static flow test has been performed to verify design calculations for hydraulic forces and flow splits through the secondary control rod.

59 F) Prototype Tests: Four complete prototype secondary control rod systems will be tested in sodium to verify compliance with the design life requirements under prototypic (except radiation) environmental conditions. Experience learned from the first prototype tests is
59 incorporated into the design of the second prototype to be tested.

G) Coil Cord Tests: The latch mechanism that moves with the driveline requires pneumatic and electrical service that is supplied by a coil cord. Component development tests of the coil cord configuration for the secondary control rod system verified the assembly procedures and design of this component.

59 H) Latch Seal Tests: This test provided information necessary to (1) properly size and shape the latch seal flow restriction and (2) to determine the leakage rate through the driveline flow limiters.

I) Nosepiece Flow Tests: The nosepiece flow test provided hydraulic
59 characteristics of an SCA nosepiece assembly.

59 J) Argon Control System Tests: This test has been performed to verify that the Argon Control System (ACS) can control the argon pressure in the three regions of the SCRS and detect leakage to or from each of the three argon pressure volumes.

4.2.3.4.2 Plant Tests

Plant testing is divided into two categories. The first category is Start-up Test where the reactivity control systems components are carefully evaluated to verify conformance with the functional design parameters after initial installation in the reactor. The second category of Plant Testing is selected parameter tests performed during shutdown and refueling to assure there has been no significant degradation of the system since the Start-Up Tests.

Start-Up Tests

Plant start-up testing for the control rod systems will consist primarily of installed performance tests. The specific design parameters to be measured in the PCRS will include scram insertion time, control rod adjustment rates, control rod worth, running and holding current for CRDM stator coils, and stator coil temperature.

Design parameters to be measured in the SCRS will include scram
59 insertion time, control rod withdrawal rate, inferred from motor speed
control rod worth, housing temperatures, scram valve solenoid holding
current, motor torque necessary for rod withdrawal, housing and driveline
59 leak tightness and scram cylinder pressure.

Tests During Shutdown and Refueling

To assure there has been no degradation of the reactivity control
systems, selected design parameters will be periodically measured during
reactor operation. Tests which require access to the mechanism or which
will interfere with safe reactor operation, will be performed during
shutdown and refueling periods. For the PCRS, the following shutdown
tests will be performed: scram insertion time, roller nut drop out
59 current, stator checks, withdrawal speed and position indicator checks.
For the SCRS, the following tests will be performed: scram insertion
time, motor torque and speed latch operation, and position indication
59 operation.

4.2.3.4.3 Surveillance

Surveillance During Operation

Surveillance during operation covers those conditions which can
be monitored or inspected while the plant is in operation through the use
of instrumentation or other means provided in the plant equipment design.
By periodic surveillance, data will be accumulated which will indicate
a drift or change in operating parameters. The more significant PCRS
conditions to be surveyed are: detection of loss of internal PCRDM pressure
including ruptured bellows through CRDM pressure switch; position indicator
59 or drive malfunction by comparison of position indicator readouts; operability
of parked control rods by slight in and out jogging movements; and CRDM
stator/cooling system malfunction by monitoring of stator thermocouples.

The conditions in the SCRS to be surveyed are: temperatures within the SCRDM housing, the electrical holding current for the scram valve solenoid, and the quantity of make-up argon gas required to maintain the SCRDM housing and SCRD pressurization. If a leak in the SCRDM housing or SCRD should occur, it will be noted by an alarm in the control room.

Surveillance During Shutdown and Refueling

This surveillance covers control rod system components which
can only be inspected by removal from the reactor. Inspection requirements
51 vary in this category due to the downtime requirements of the plant.
However, they can be summarized in the following groupings:

o Disassembly surveillance during normal refueling periods

No disassembly surveillance is planned during normal refueling. If, however, the normal tests performed during refueling or operation indicate some degradation in the operation of one or more PCRDMs, a selected removal and inspection of the suspected components may be
59 performed.

o Disassembly surveillance after replacement

On a planned schedule, a mechanism and driveline will be removed and replaced so that it may be partially diassembled and inspected. These extensive inspections will include the following types of examinations: condition of leadscrews, springs, welded joints, wear surfaces, and wiring. During these inspections, expendable items, defective components, and those with limited design life, will be replaced to requalify the mechanism and driveline for reactor service.

4.2.3.4.4 Acceptance Tests

Acceptance tests will be performed both at the fabricator's site and at the CRBRP site after shipment. These acceptance tests follow stringent inspection and quality assurance procedures established for all stages of fabrication beginning with material procurement.

A. Fabricator Acceptance Tests

The fabricator shall perform acceptance tests on plant equipment to establish that the performance of each unit is within acceptable limits. The acceptance limits shall be determined from the performance requirements of the equipment specifications.

For the primary CRDM/CRD, these acceptance tests will include (but not be limited to):

a) Proof pressure test of primary pressure boundary components

b) Stator insulation and winding resistance

c) Scram insertion characteristics from withdrawal heights of 10 inches and 36 inches

d) Unlatch time at standard operating conditions of stator temperature and voltage

e) Minimum latch current

f) Minimum static and dynamic drop out current

g) Position indicator accuracy

51 59 h) Action of outmotion latch pawl

Similar supplier acceptance tests will be established for the SCRDM and SCRD.

For the Primary Control Assemblies, fabricator acceptance tests for pin cladding and B_4C pellets will follow the guidelines of RDT M3 - 28T, May 1972, for cladding and RDT E6 30T, May 1973 for B_4C pellets - and will be fully defined prior to the FSAR. It is currently expected that the fabricator acceptance test requirements will cover the following as a minimum:

B_4C Pellets:

1) Pellet geometry
2) Density
3) B-10 content
4) Stoichiometry
5) Chemical Impurities

Pin Cladding, Pin Assembly, and Final Assembly

1) Geometry and Dimensional Checks
2) Bonding Gas Analysis as appropriate
3) Cladding Metallurgy (grain size, intergranular attack)
4) Cladding Mechanical Properties
5) Weld Integrity
6) Ultrasonic Inspections
7) Cleanliness
8) B-10 Content
9) Weights

Inspections (such as mechanical properties) will be based on sampling plans for lot qualification which will be defined prior to the FSAR.

Critical parameters such as B-10 content will be determined in 100% inspections.

Similar fabricator acceptance tests will apply for the SCA.

B. On-Site Acceptance Tests

In addition to the acceptance tests to be performed by the equipment fabricator prior to shipment, receiving inspection and/or acceptance testing will be performed on each item of plant equipment when it arrives at the site. The purpose of these on-site inspections and tests will be to verify that the performance of the equipment has not
59| 51 deteriorated during shipping to the site, or while in storage.

six poles of the indicating disk in conjunction with the 0.60-in. pitch of the leadscrew provides a resolution of $\pm$0.10-in. and an accuracy of $\pm$0.15-in. for this indication system. However, if a scram or misstepping occurs, the relative position indication system loses its zero-position reference and must be reset at the full-inserted control rod position.

4.2.3.5.1.3 CRDM Pressure Switch

The upper mechanism assembly is in a sealed environment pressurized with argon gas to protect the rotor assembly and leadscrew from the deleterious effects of the sodium vapor in the reactor cover gas. The argon pressure is monitored by a pressure switch located at
51 the top of the CRDM motor tube housing. In the event of a failure of the bellows sealing arrangement for the upper mechanism, the higher pressure argon

gas will leak into reactor cover gas volume, and the resulting decrease in argon pressure will be detected by the pressure switch so that corrective action can be initiated.

4.2.3.5.1.4 CRDM Stator Thermocouples

The CRDM stator coils are equipped with thermocouples to monitor the temperature of the stator windings. In the event of a loss of flow of the stator coolant gas, the resulting increase in stator temperature will be detected by the thermocouple so that corrective action can be initiated.

4.2.3.5.2 Secondary Control Rod System Instrumentation

The instrumentation for the secondary control rod system consists of two independent control rod position indication systems, as well as a scram latch indication system to detect if the driveline is coupled to the control rod, a pressure switch to monitor the integrity of the SCRDM bellows sealing arrangement, thermocouples to monitor the internal temperatures of the SCRDM, and an acoustic monitor (accelerometer) for measurement of scram insertion time for the secondary control rods.

59 4.2.3.5.2.1 Leadscrew Absolute Encoder Position Indication System

The driveline motion is provided through two parallel leadscrews. An encoder geared directly to one of the leadscrews provides the driveline position indication. The driveline position indication is used when coupling the latch
59 to the control rod and raising the control rod to its operational withdrawn position, and for withdrawing the driveline to the fully retracted refueling position.

4.2.3.5.2.2 Proximity Switch Position Indication System

Four proximity indicator switches are used. Two switches are for a limit switch function to indicate the driveline positions which are the near full-down with the control rod in the core, and withdrawn for refueling with the control rod disconnected from the driveline. These two proximity indicator switches will operate indicator lights on the control room operator panel showing the two driveline positions and function in the control logic circuit to automatically stop the drive motor at the stroke end points.

The other two proximity indicator switcnes are used to check the function of the absolute encoder. The two switches are mounted just below the driveline parked position. The encoder function is checked as the carriage actuates these two switches. This initiates a comparison of the encoder reading against
59 calibrated valves.

4.2.3.5.2.3 Scram Latch Indication System

The latch design provides a mechanical feature that contacts the top end of the control rod coupling head over a range of positions on either side of the latched position. The motion of the coupling head is transmitted to the SCRDM by a sensing tube within the driveline. An LVDT (linear variable differential transformer) attached to the upper end of the sensing tube provides an indication of the location of the control rod coupling head with respect to the latch gripper. This indication is used for the coupling operation wherein the coupling head is first sensed as the driveline approaches, and shows when the driveline is in the correct position for actuating the latch gripper. When the coupling is completed, the sensing tube provides continuous indication that the control rod is connected to the driveline and therefore, serves an important safety-related function.

The tension rod that couples the scram latch at the driveline bottom
39 end to the pneumatic actuator within the SCRDM, has its position indicated by a linear variable differential transformer (LVDT) located in the SCRDM. With
39 pneumatic pressure (argon) applied to the pneumatic actuator, the tension rod pulls on the latch gripper to retain the control rod coupling head within the latch. The position of the tension rod provides the indication that the latch gripper is in the state that holds the control rod. When this indication is coupled with the sensing tube indication that shows when the control rod coupling head is within the latch, then the control rod is known to be properly latched to the driveline.

4.2.3.5.2.4 SCRDM Pressure Switches

The SCRS utilizes three different argon pressure levels. The lowest pressure, approximately 5 psig above the reactor cover gas pressure, fills the SCRDM housing and provides a low pressure differential across the main driveline bellows to assure leakage into the reactor if a leak should develop. The
59 mid-pressure, approximately 60 psig, internally pressurizes the driveline to assure a latch bellows leak will not permit liquid sodium to rise inside the driveline if a bellows leak should occur. The highest pressure, approximately 220 psig, is supplied to the pneumatic actuator to keep the scram latch ener-
39 gized. Each of these pressures is static once equilibrium is reached and the pressures are monitored in the control room. The system will include a leak detection capability for each pressure zone to warn of any bellows failure or other possible leak.

4.2.3.5.2.5 SCRDM Thermocouples

The temperature within the SCRDM will be measured by several thermocouples. The thermocouples will be located on the motor housing and at various elevations within the SCRDM housing for an indication of the axial temperature distribution.

4.2.3.5.2.6 Acoustic Monitor

An acoustic monitor and accelerometer, will measure scram times to obtain comparative surveillance data through the drive lifetime. The impact of the control rod meeting the control assembly downstop will produce vibrations that would be sensed. This will be verified by prototype testing. Additionally, the acoustic monitor could be used for diagnostic mechanical signature analysis of such SCRDM mechanical components as bearing, gears, ball-screw/ball-nut, and valve operation.

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TABLE 4.2-20

PLANNED CRBRP BLANKET DESIGN VERIFICATION TESTS

Activity Title	Activity Description	Completion Date	Provides Input To
Assembly Flow and Vibration Test in Water	Full scale assembly test to determine rod and assembly vibrations and pressure losses in the assembly.	12/79	Final Design Verification
Component Pressure Drop Test	Flow test in water various blanket assembly components, measuring the pressure losses as a function of flow rate.	9/80	Final Design
Load Pad Strength and Duct Bending Stiffness Test	Component test to determine the strength of a cold worked duct and load pad for various loading conditions and temperatures.	12/79	Final Design Verification
Rod Cladding Rupture Test	Component test to determine the strength of the cold worked cladding for several combinations of temperature and time and to establish a conceptual stress rupture time curve.	Complete	Final Design Verification
Rod Irradiation Test in EBR-II	Determine the performance characteristics of radial blanket rods in a reactor environment including the effects of a power jump.	Complete	Final Design Verification
Fabricate Rod Bundle Irradiation Test in FFTF	Provide prototypic fabrication experience for blanket assemblies.	Complete	Assembly Fabrication

59 TABLE 4.2.21 IS DELETED.

59 TABLE 4.2.22 IS DELETED.

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TABLE 4.2-23

CORE SUPPORT STRUCTURE STRESS AND FATIGUE DAMAGE SUMMARY FOR NORMAL AND UPSET CONDITIONS

CUT	STRESS CATEGORY	CALCULATED VALUE	ALLOWABLE VALUE	MARGIN-OF-SAFETY*
A thru E	P_m	5734 psi	14,800 psi	1.58
Figure 4.2-51	$P_L + P_b$	8760 psi	21,230 psi	1.42
	$(P_L+P_b+Q)_r^{**}$	94,050 psi	49,185 psi	N/A
	$\Sigma(^n/N)$	.57	.90	(+)
1 thru 10	P_m	2357 psi	14,800 psi	5.28
Figure 4.2-52	P_L+P_b	13,344 psi	19,980 psi	.50
	$(P_L+P_b+Q)_r^{**}$	83,809 psi	49,350 psi	N/A
	$\Sigma(^n/N)$	.80	.90	(+)
1 thru 18	P_m	<10,486 psi	15,000 psi	>.43
Figure 4.2-53	P_L+P_b	10,486 psi	>15,000 psi	>.43
	$(P_L+P_b+Q)_r^{**}$	83,333 psi	49,350 psi	N/A
	$\Sigma(^n/N)$	.41	.90	(+)

$$\text{*MARGIN-OF-SAFETY} = \frac{\text{ALLOWABLE STRESS INTENSITY}}{\text{APPLIED STRESS INTENSITY}} - 1$$

59 ** Since $(P_L+P_b+Q)_r > 3S_m$, simplified elastic-plastic analysis is used.

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59 TABLES 4.2-24 through -28

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PART NAMES	MODEL USED	STRESSES (PSI)				CRITERIA (PSI)			
		MEMBRANE	MEMBRANE + BENDING	BEARING	SHEAR	Sm	1.5 Sm	Sy	0.6 Sm
BPFM	INTEGRATED 3-D SEISMIC MODEL	2,154	2,246	13,565	7.235	15.040	22.560	16.740	9.024
SEISMIC LUG		4,484	13,791	3,948	2,548	15,040	22,560	16,740	9,024
SEISMIC SHEAR PIN		14,155	23,379	8,151	6,919	61,833	K_tS_t= 78,522	130,700	37.100

59

Table 4.2-29a. BPFM Stress Summary (OBE)

59 TABLES 4.2-29b and -29c

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TABLE 4.2-29D

EVALUATION SUMMARY OF THE LOAD CONTROLLED STRESSES (FOR FRS)

Material	Condition	Temp. (°F)	Evaluated Results (PSI)	Criteria (PSI)	Margin*
316 SS	Normal & Upset (OBE)	1000	$P_m = 3202$	$S_{mt} = 154000$	>3
			$P_L + P_b = 5840$	$K_t\ S_t = 22330$	2.82
	Faulted (SSE)	1000	$P_m = 4828$	$1.2\ S_t = 22200$	>3
			$P_L + P_b = 8808$	$1.2\ K_t\ S_t = 26307$	1.99

$$\text{*Margin} = \frac{\text{Allowable}}{\text{Actual}} - 1$$

59

TABLE 4.2-29E

RESULTS OF LOAD-CONTROLLED ANALYSIS: LOWER SUPPORT BLOCK

Material: 304SS

Condition No.	Temp. (°F)	(1,2) Stress Intensity (P_m) (psi)	Allowable (psi)	(3) Allowable Stress Origin	Use Fraction Sum Calculation: Time of Loading t_i (hrs)	Allowable Time t_{ia} (hrs)	t_i/t_{ia}
Design	775	1880	15,100	S_o @ 800°F	-	-	-
A	755	1880	15,100	S_{mt} @ 800°F	262,900	∞	0.0
B/C	755	1880	15,100	S_{mt} @ 800°F	1	∞	0.0
D	1120	1880	22,800	$.6S_r$ @ 1000°F for 1 hr.	-	-	-

$$\sum \frac{t_i}{t_{ia}} = 0$$

Notes:

(1) SSE loads were combined with design condition loadings to conservatively bound all the conditions.

(2) The stresses in this column are actually $P_L + P_B$. To be conservative, they have been treated as P_m, therefore $P_L + P_B$ criteria is automatically satisfied.

(3) Allowables are conservatively taken at higher temperature values.

TABLE 4.2-29F

RESULTS OF LOAD-CONTROLLED ANALYSIS: HORIZONTAL BAFFLE PLATE

Material: 316SS

Condition No.	Temp. (°F)	(1) Stress Intensity (P_m) (psi)	Allowable (psi)	(2) Allowable Stress Origin	Use Fraction Sum Calculation: Time of Loading t_i (hrs)	Allowable Time t_{ia} (hrs)	t_i/t_{ia}
Design	1020	10030	12140	S_o @ 1105°F	-	-	-
A	1000	10030	14800	S_{mt} @ 1105°F	262,900	10^6	0.263
B/C	1105	10030	14800	S_{mt} @ 1105°F	1	50,000	0.0
D	1120	10030	27900	$.6S_r$ @ 1150°F for 1 hr.	-	-	-

$$\sum \frac{t_i}{t_{ia}} = .263$$

Notes:

(1) SSE loads were combined with design condition loadings to conservatively bound all the conditions.

59 (2) Allowables are conservatively taken at higher temperature values.

TABLE 4.2-30

EFFECT OF (C + N) LEVEL ON TENSILE PROPERTIES OF TYPE 304 STAINLESS STEEL

	Mechanical Property	Equation	Standard Deviation
53	Yield Strength, KSI	$89.952 + 181.167\,(C + N) - 0.082\,T\,(C + N) - 4.269\,T^{1/2} + 0.058\,T$ where T is ^{o}R	2.236
53	Ultimate Tensile Strength, KSI	$-1.806 - 227.391\,(C + N) + 1634.644\,T^{-1/2} + 0.626\,T\,(C + N) - 2.607 \times 10^{-4}\,T^{2}\,(C + N)$ where T is ^{o}R	2.645
53	Total Elongation, %	$29.595 - 89.898\,(C + N) + 628.443\,T^{-1/2}$ where T is ^{o}R	2.955
53	Uniform Elongation %	25,748 - 78.[illegible]1 (C + N) + 546.745 $T^{-1/2}$ where T is ^{o}R	2.955

Temperature range validities:

yield strength	75 - 1200°F
ultimate tensile strength	200 - 1100°F
total elongation	400 - 1200°F
uniform elongation	400 - 1050°F

TABLE 4.2-36B

AVERAGE FRICTION COEFFICIENTS
FROM
W-ARD TEST DATA

STROKE = .750 IN.

AVERAGE FRICTION COEFFICIENT		P = 100 psi								P = 1500 psi							
		TEST 1				TEST 2				TEST 1				TEST 2			
		400°F		800°F		800°F		1100°F		400°F		800°F		800°F		1100°F	
Material Couple	Environment	μ	μ+3σ	μ	μ+3σ	μ	μ+3σ	μ	μ+3σ	μ	μ+3σ	μ	μ+3σ	μ	μ+3σ	μ	μ+3σ
316SS/316SS	Liq. Na	.81	.87	.88	.98	.80	.87	.87	.91	.73	.76	.74	.77	.73	.75	.75	.77
Inconel 718/ Inconel 718	Liq. Na	.80 .63	.85 .69	.79 .70	.86 .77	.69 .74	.73 .79	.65 .70	.70 .73	.71 .58	.75 .61	.71 .60	.75 .62	.61 .62	.65 .66	.63 .66	.67 .68
Inconel 718/ 316SS	Liq. Na	.87 .84	.92 .90	.89 .86	.95 .94	.92	1.02	.89	.95	.86 .79	.90 .83	.82 .77	.87 .84	.74	.80	.77	.81
Inconel 718/ Haynes 273	Liq. Na	.70	.76	.67	.73	.79	.85	.78	.81	.66	.70	.69	.73	.73	.78	.70	.72
*Stellite 6/ 17-4 PH	Argon	.24	.29	.27	.32					.17	.21	.13	.14				

STROKE = .188 IN.

Material Couple	Environment	μ	μ+3σ	μ	μ+3σ	μ	μ+3σ	μ	μ+3σ	μ	μ+3σ	μ	μ+3σ	μ	μ+3σ	μ	μ+3σ
316SS/316SS	Liq. Na	.95	1.02	1.07	1.17	.82	.87	.88	.92	.80	.84	.82	.86	.77	.81	.76	.79
Inconel 718/ Inconel 718	Liq. Na	.88 .73	.94 .80	.82 .73	.89 .78	.73 .83	.79 .89	.69 .72	.82 .75	.79 .66	.85 .70	.74 .66	.79 .69	.70 .67	.73 .70	.65 .67	.68 .69
Inconel 718/ 316SS	Liq. Na	.94 .92	1.01 1.00	.94 .91	1.00 .96	.99	1.06	.85	.89	.90 .87	.94 .92	.81 .79	.86 .83	.80	.84	.74	.78
Inconel 718/ Haynes 273	Liq. Na	.77	.83	.90	.96	.76	.80	.70	.73	.78	.82	.75	.81	.75	.79	.68	.70
*Stellite 6/ 17-4 PH	Argon	.21	.26	.17	.21					.18	.21	.13	.15				

*Run at 150°F and 400°F with 17-4 PH vitrolubed.

59

(next page is 4.2-391)

TABLE 4.2-36C

SUMMARY OF AVERAGE
FRICTION COEFFICIENTS

Material Couple	Environment	UPPER THREE SIGMA DATA	
		P=100 psi	P=1500 psi
STROKE = .750 IN.			
316SS/316SS	Liquid Na	.91	.76
Inconel 718/ Inconel 718	Liquid Na	.77	.67
Inconel 718/ 316SS	Liquid Na	.95	.84
Inconel 718/ Haynes 273	Liquid Na	.78	.73
STROKE = .188 IN.			
316SS/316	Liquid Na	1.00	.83
Inconel 718/ Inconel 718	Liquid Na	.83	.73
Inconel 718/ 316SS	Liquid Na	.99	.86
Inconel 718/ Haynes 273	Liquid Na	.83	.78

59

(next page is 4.2-393)

TABLE 4.2-36D

FFTF AND CRBRP PROTOTYPE CONTROL ROD SYSTEMS TESTS

FFTF Prototype Tests

	Phase	Test Description	Temp. Range	Feet of Travel	Scrams	Disconnect Operations
59 53	I	Flow Rates - 0 - 90 GPM Scram Heights - 9, 18, 27, 35, 36" Misalignment - Misaligned	400 to 1100°F	3,420	188	1
59 53	II	Flow Rates - 0 - 100 GPM Scram Heights - 9, 18, 27, 35, 36" Misalignment - Aligned	400 to 1100°F	2,730	157	33
59 53	III	Flow Rates - 0 - 100 GPM Scram Heights - 9, 18, 27, 35, 36" Misalignment - Gross Misalignment	400 to 1100°F	14,400	428	15
59 53	IV	Flow Rates - 0 - 100 Scram Heights - 9, 18, 27, 35, 36" Misalignment - Aligned with Failed Bellows	400 to 1100°F	3,085	55	1

53 100 GPM is 110% of Nominal Coolant Flow 15

TABLE 4.2-36D (Continued)

CRBRP PROTOTYPE TESTS

Phase	Test Description	Temp. Range	Feet of Travel	Scrams	Disconnect Operations
Complete	Accelerated Unlatching Life Test (Prototype Unit 1)	N/A* (H_2O)	35,451	1868	60
I	PCRS Accelerated Life Test (Prototype Unit 2)	400 to 1100°F	5,878	470	42
I	Failed Bellows Test (Prototype Unit 3)	400 to 1100°F	9,146	680	40
I	Real Time Test (Prototype Unit 2)	400 to 1100°F	5,418	368	40

59 *PCRDM and its supporting nozzle were maintained at prototypic temperatures by means of heaters. The drive-line and dummy weight to simulate the control rod weight were in a water environment for this test.

TABLE 4.2 - 36E

MAXIMUM EFFECTIVE COEFFICIENT OF
FRICTION DATA

FOR DYNAMIC IMPACT LOADING

	Material Couple	Configuration	Medium	Max *
	I 718/I 718	Round	Air	.37
	I 718/I 718	Round	Water	.35
53	I 718/316 SS	Hexagonal	Air	.63 **
	I 718/I 718	Round	Sodium	.46
	I 718/304 SS	Round	Air	.45
59	I 718/304 SS	Round	Water	.40

53 51 * Values represent the maximum observed in any test.

** Includes effect of slight duct material yielding resulting from test
59 article configuration Actual coefficient of friction will be less.

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TABLE 4.2-37

CONTROL ROD DRIVE SYSTEM MAXIMUM LOADING CONDITIONS

Condition	Load (lb) Primary	Secondary	Basis
Insertion	1000 (Min.) Max. Under Evaluation	1000 (Min.) Max. Under Evaluation	Primary and secondary system minimum insertion load.
Withdrawal	1500 (Min.) Max. Under Evaluation	Under Evaluation Under Evaluation	Primary and secondary system values dependent on further detailed design and test measurements.
Minimum Breakaway Joint Strength	16000	Under Evaluation	Minimum design strength of breakaway joint (CA to CA shaft) to provide for drive shaft separation with stuck control rod.
Maximum Breakaway Joint Strength	19000	Under Evaluation	Maximum design failure load of breakaway joint to provide for drive shaft separation with stuck control rod.
PI Compression	1000 (Max.)	None	Maximum load to free control rod coupling.
Assembly of Components	Variable	Variable	Torquing preloads required in assembling.
Seismic	OBE SSE	OBE SSE	Seismic design data bases are discussed in Section 3.7.
Thermal	Under evaluation	Under evaluation	Thermal stress for steady state and transient loads.
Rod Expulsion	1800 (Max.)	2400 (Min.)	Conservative design against rod expulsion such that maximum control assembly pressure drop cannot cause significant control rod transients with the driveline connected or disconnected.
Scram	Under evaluation	Under evaluation	Forces from deceleration of translating assembly on scram stop.
Structural Margin Beyond Design Basis	Internal reactor vessel pressure (300 psi).	Internal reactor vessel pressure (300 psi).	Sealing integrity for CRDM housing and nozzle required under SMBDB conditions. Mechanical and hydraulic loads resulting from SMBDB including acceleration vs. time history loading are considered in design basis

TABLE 4.2-41

COMPARISON OF PRIMARY AND SECONDARY CONTROL ROD SYSTEMS

		Primary	Secondary
	Control Assembly (CA)		
	Control Rod	37 Pin Bundle	31 Pin Bundle
	B_4C Enrichments	92%	92%
	Control Rod Guide Geometry	Hexagonal	Cylindrical
59	Control Rod Stroke	36.0" to 37.8"	37.5" Nominal
	Control Rod Driveline (CRD)		
	CA Coupling	Rigid Coupling	Flexible Collet Latch
	CRDM Connection	CRD Leadscrew to CRDM Roller Nuts	SCRD Attached to SCRDM Carriage with Pneumatic Activation to SCRD Latch thru Slender Rod
	CA Disconnect for Refueling	Manual	Automatic
	Control Rod Drive Mechanisms (CRDM)		
	Type of Mechanism	Collapsible Rotor-Roller Nut	Twin Ball Screw with Translating Carriage
59	Overall Mechanism Stroke	37.5 Inches	67.5 Inches
	Cover Gas Seal	Bellows	Bellows
	Scram Function		
	Scram Release	Magnetic, Release CRDM Roller Nuts	Pneumatic, Release SCRD Latch in SCA
51	Scram Assist	Spring in CRDM	Hydraulic in SCA

TABLE 4.2-41 (Continued)

	Primary	Secondary
Scram Function		
Scram Speed Dependence on Flow Rate	Increases with Decreasing Flow Rate	Decreases with Decreasing Flow Rate
36 Scram Assist Length	27 Inches	Full Stroke
Scram Deceleration	Hydraulic Dashpot in CRD	Hydraulic Spring Damper in SCA
Scram Motion Thru Upper Internals	Full Stroke	~0.25 Inch
Operational Functions		
Independent Shutdown Capability	Yes	Yes
Burnup Control	Yes	No
Power and Reactivity Control	Yes	No

TABLE 4.2-43

PRIMARY CONTROL ROD SYSTEM LATERAL MISALIGNMENT FORCES

Rod Position (Inches Withdrawn)		Drag Force (lbf.)
0	Minimum radial clearance (C.R. coupling O.D. to scram arrest flange I.D.)	291
6	Increased radial clearance (C.R. shaft O.D. to scram arrest flange I.D.)	232
10	Just prior to dashpot deceleration	125
15	Typical position of Row 7 Corner control rod at start of cycle	131
30	End of cycle control rod position	101
37	Maximum withdrawal	108

53 | 51 *Based on lateral misalignments from Figure 4.2-95B.

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TABLE 4.2-43a

EFFECT OF WORST CASE MISALIGNMENT ASSUMPTION

59 53

Configuration	System Misalignment According to Fig. 4.2-95B	Non-Translating Assembly Misalignment	Translating Assembly Misalignment	Fixed Dashpot Cup Assumed	Total Lateral Forces Predicted (lbf.)
Analysis of PCRS in CRBRP	Yes	Yes	Yes	Yes	291
Analysis of PCRS Tests Case 1	Yes	Yes	Yes	Yes	300
Case 2	Yes	Yes	Yes	No	102
Case 3	Yes	No	No	No	7.2
Test Results from PCRS Tests	Yes	As-built	As-built	Validity of Assumption Checked by test. Dashpot is <u>not</u> fixed.	10

51

59

TABLE 4.2-44

PCRS CYCLE DEPENDENT WITHDRAWAL POSITIONS
(IN INCHES)

CYCLE	ROW 4	ROW 7			
		Minimum		Expected	
	BOC & EOC	BOC	EOC	BOC	EOC
1	36	19.0	21.4	23.2	25.6
2	36	15.4	25.1	20.0	29.7
3	36	15.3	23.5	20.1	27.9
4	36	13.0	24.7	18.1	29.2
5	36	13.9	21.1	18.9	25.5

51

NOTE: Scram analyses of paragraph 4.2.3.3.1.3 have been based on minimum rod withdrawals approximately 0.4 inches greater than this table. The effects of this difference on scram speeds are negligible compared to the existing margins relative to requirements.

59

Amend. 59
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51 TABLE 4.2-45 DELETED

TABLE 4.2-47

CRDM DESIGN STRUCTURAL ANALYSIS SUMMARY

Component	Allowable Stress or Load	Calculated Stress or Load	Margin
Pressure Boundary Components:			
Motor Tube	60,000 psi	29,540 psi	103%
Position Detector Housing	60,000 psi	29,200 psi	105%
Nozzle Extension	25,000 psi	21,616 psi	15.6%
Non-Pressure Boundary Components:			
Rotor Assembly Bearings:			
Radial Bearing	7,594 lbs	<1,000 lbs	>659%
Synchronizer Bearing	12,476 lbs	<2,000 lbs	>524%
Segment Arm Assembly:			
Segment Arm Springs	70,000 psi	66,442 psi	5.4%
Synchronizer Pin	1.903 kips	.950 kips	100%
Segment Arm Pivot Pin	13,541 lbs	4,081 lbs	232%
Out Motion Limiting System:			
Out Motion Pawl	124,875 psi	56,555 psi	121%
Out Motion Pivot Pin	83,850 psi	45,219 psi	85%
Rotational Stop Latch	24,600 psi	21,201 psi	16%
Scram Assist Spring	70,000 psi	65,070 psi	7.6%
Leadscrew	24,600 psi	18,806 psi	31%
51 59 CRD Disconnect Fingers	53,633 psi	35,100 psi	53%

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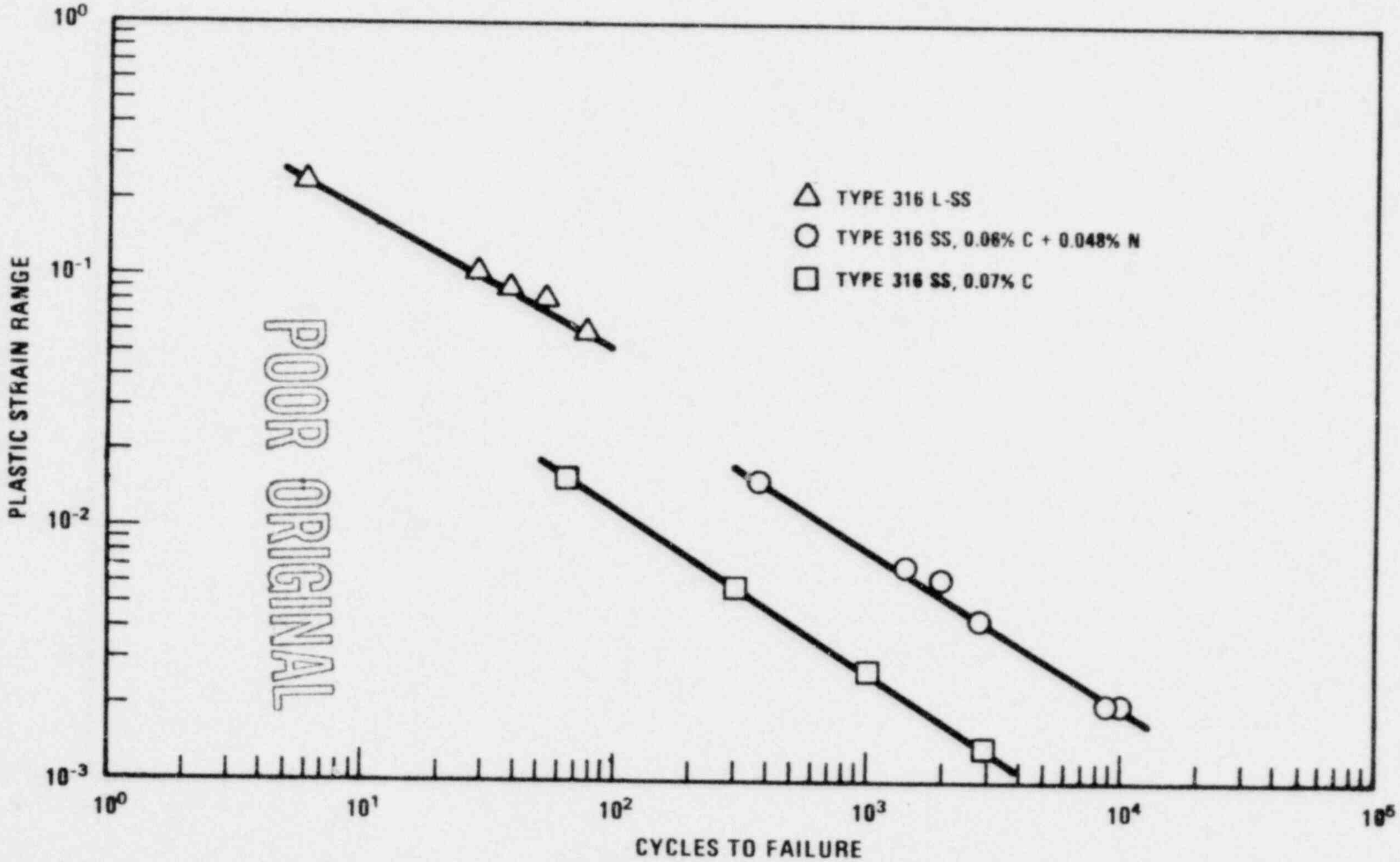

Figure 4.2-48K. Effect of Carbon Concentration on the Fatigue Strength of Type 316 Stainless Steel at 1200° F.

59 FIGURE 4.2-49

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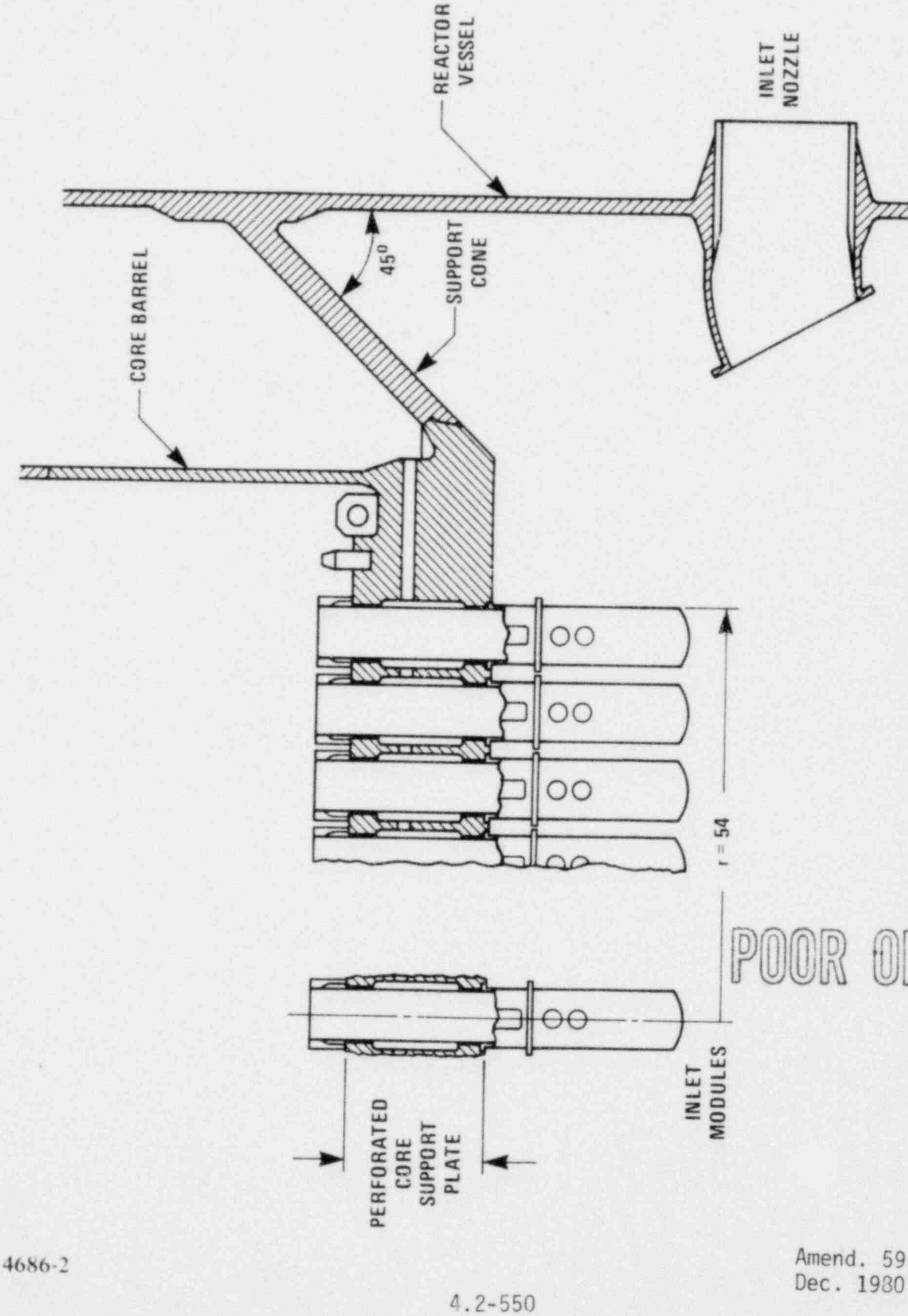

Figure 4.2-50. Core Support Structure (CSS)

4685-5

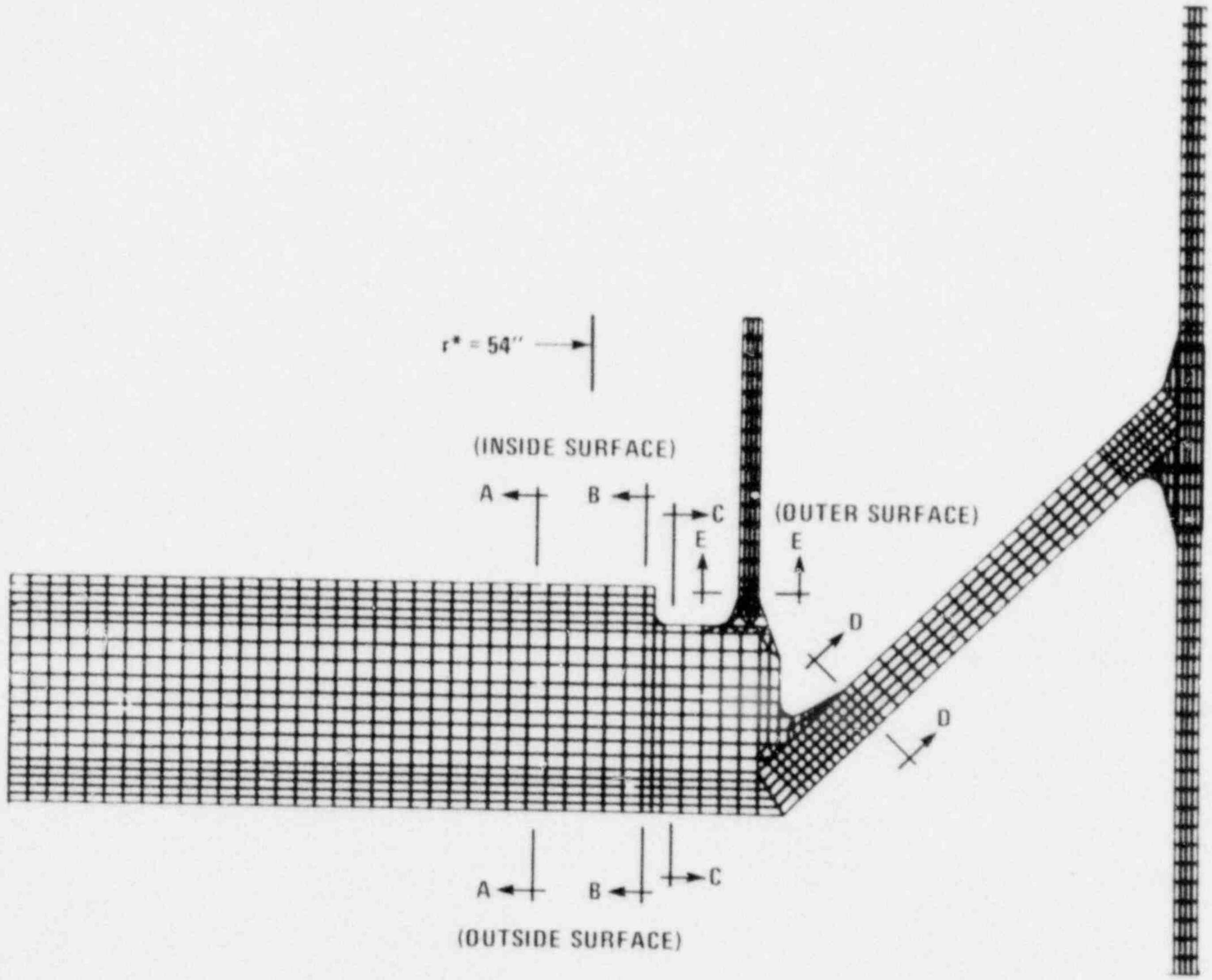

Figure 4.2-51 Core Support Structure Axisymmetric Thermal Stress Model

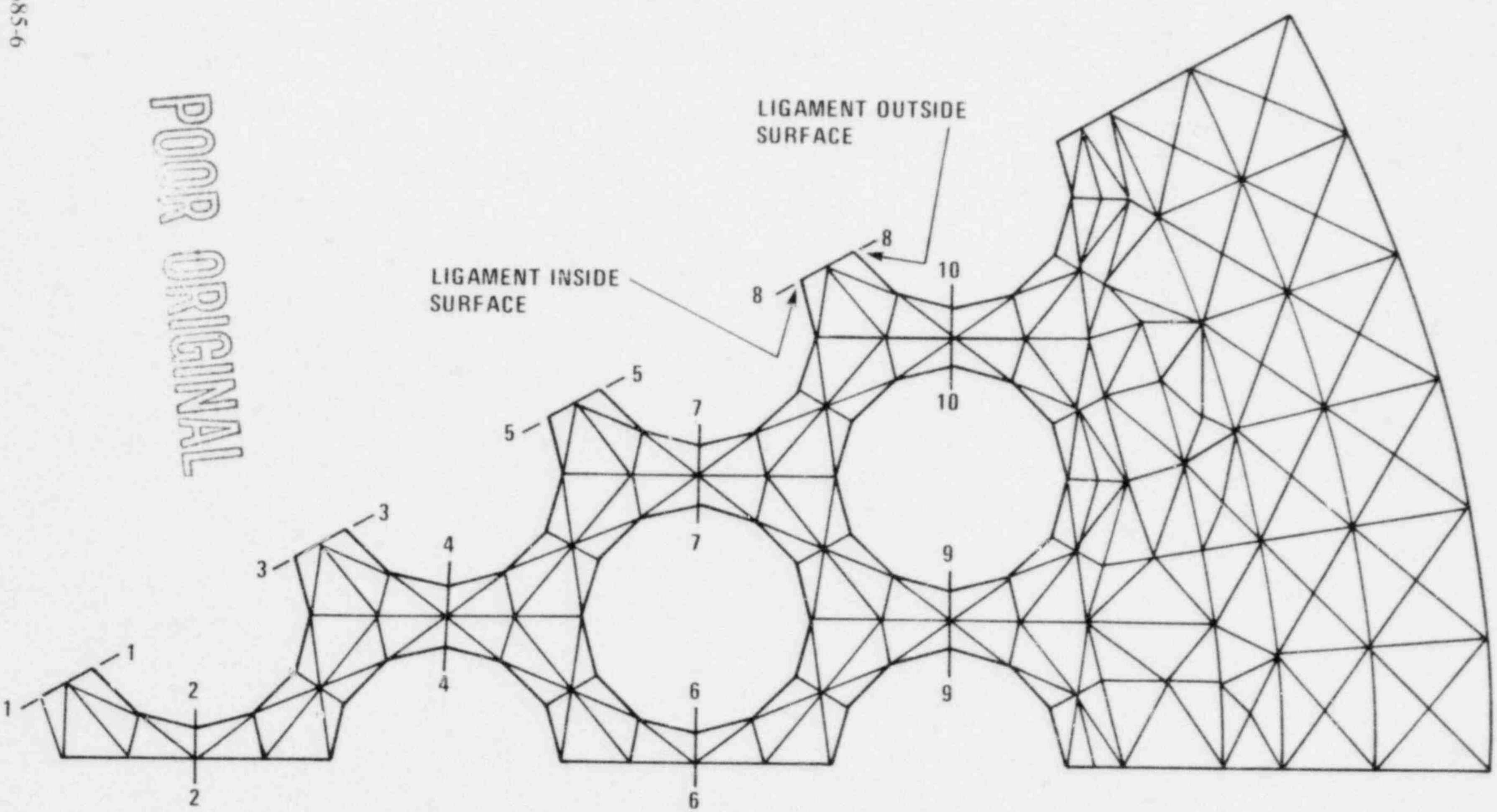

Figure 4.2-52 Identifying Cut Numbers in Sector of the Lower Support Structure Central Region

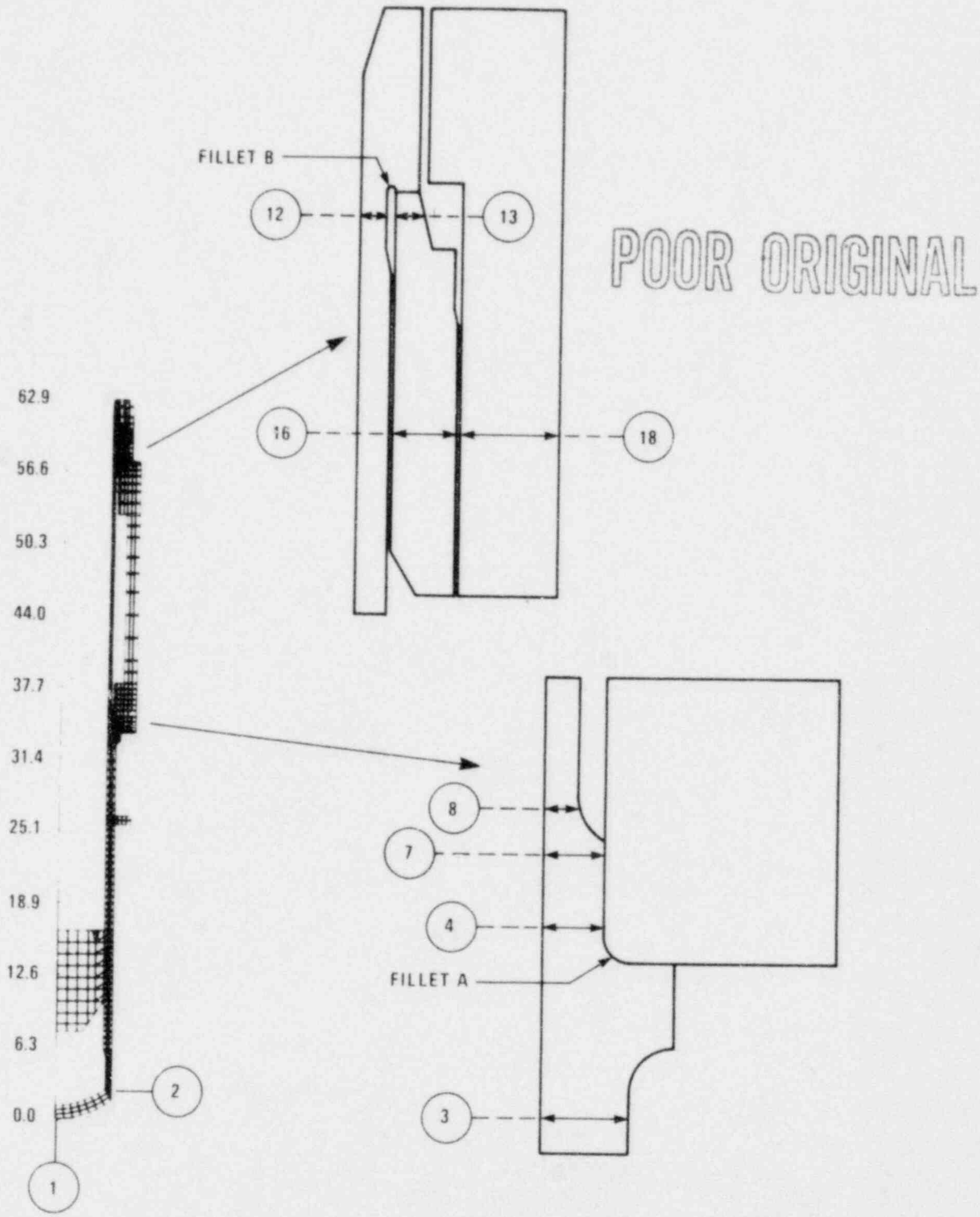

Figure 4.2-53 Lim Liner Finite Element Stress Model with Key Structural Evaluation Sections

4685-7

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Figure 4.2-54 Core Support Structure Seismic and Deadweight Model

4685-8

59 FIGURES 4.2-55 through -63

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51 FIGURE 4.2-64, 65 and 66 DELETED

59 FIGURES 4.2-67 through -74

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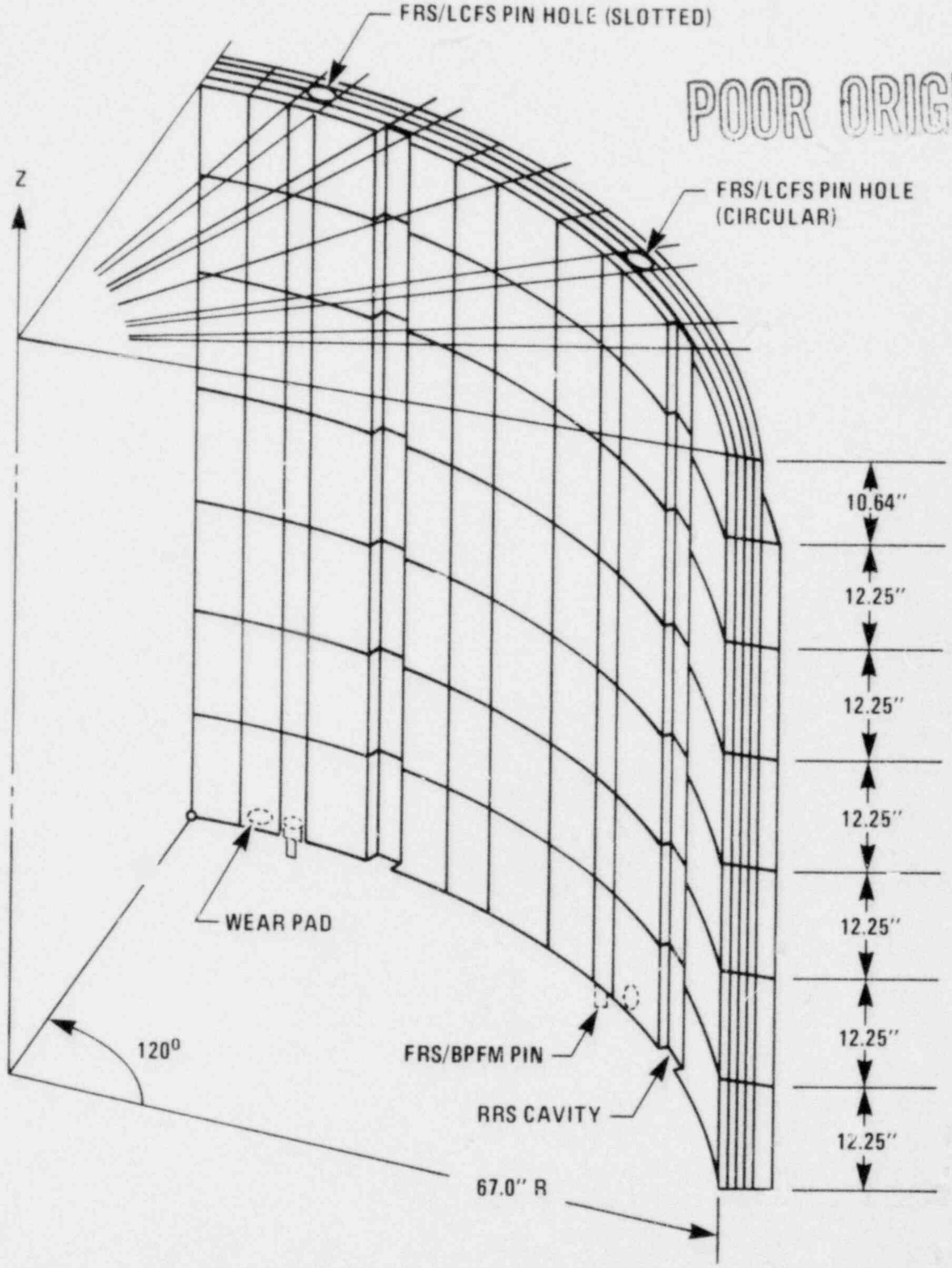

Figure 4.2-75. FRS Computer Model W/8 Node Isoparametric Elements

4756-2

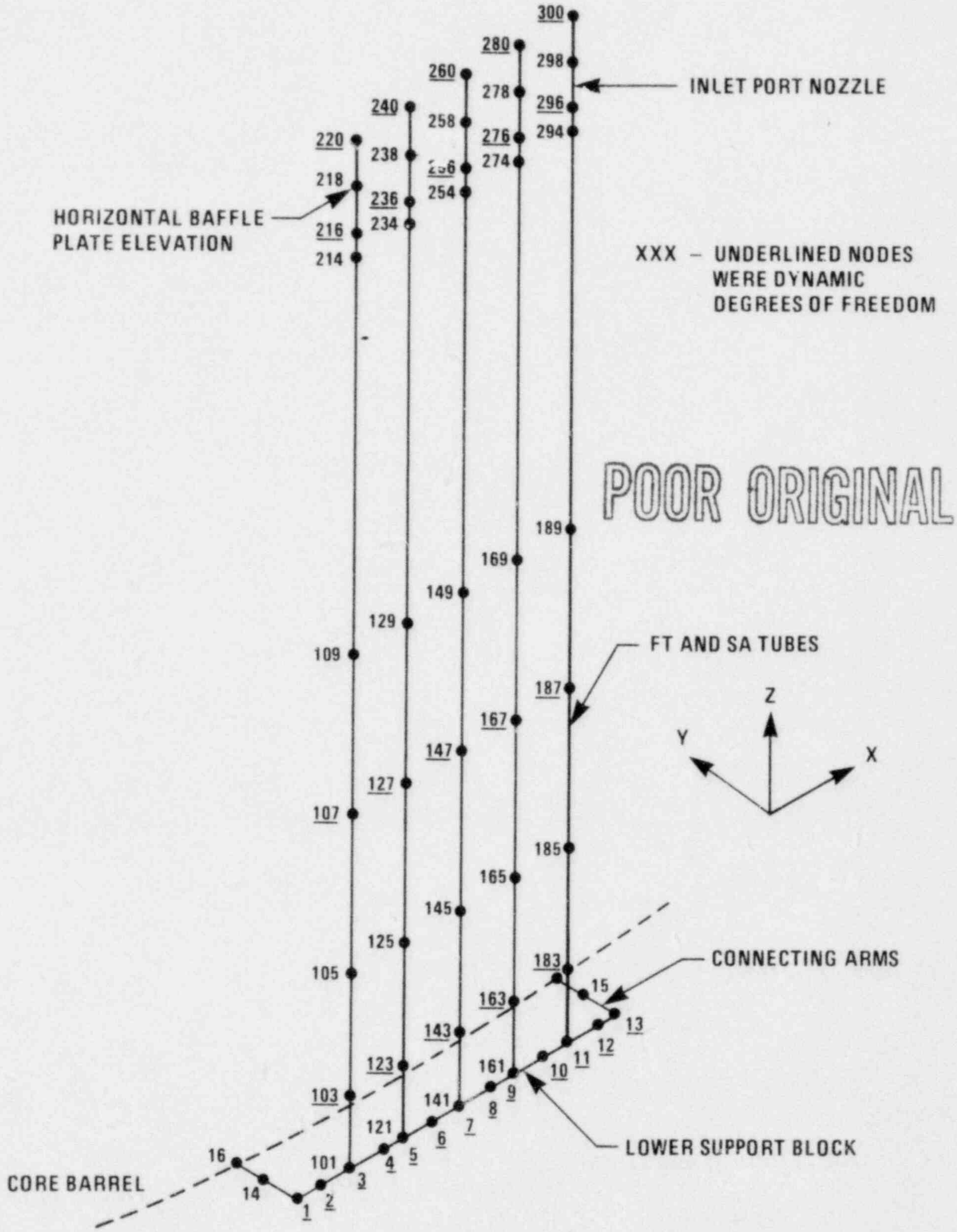

Figure 4.2-76. Finite Element Model of FT and SA Assembly

4756-3

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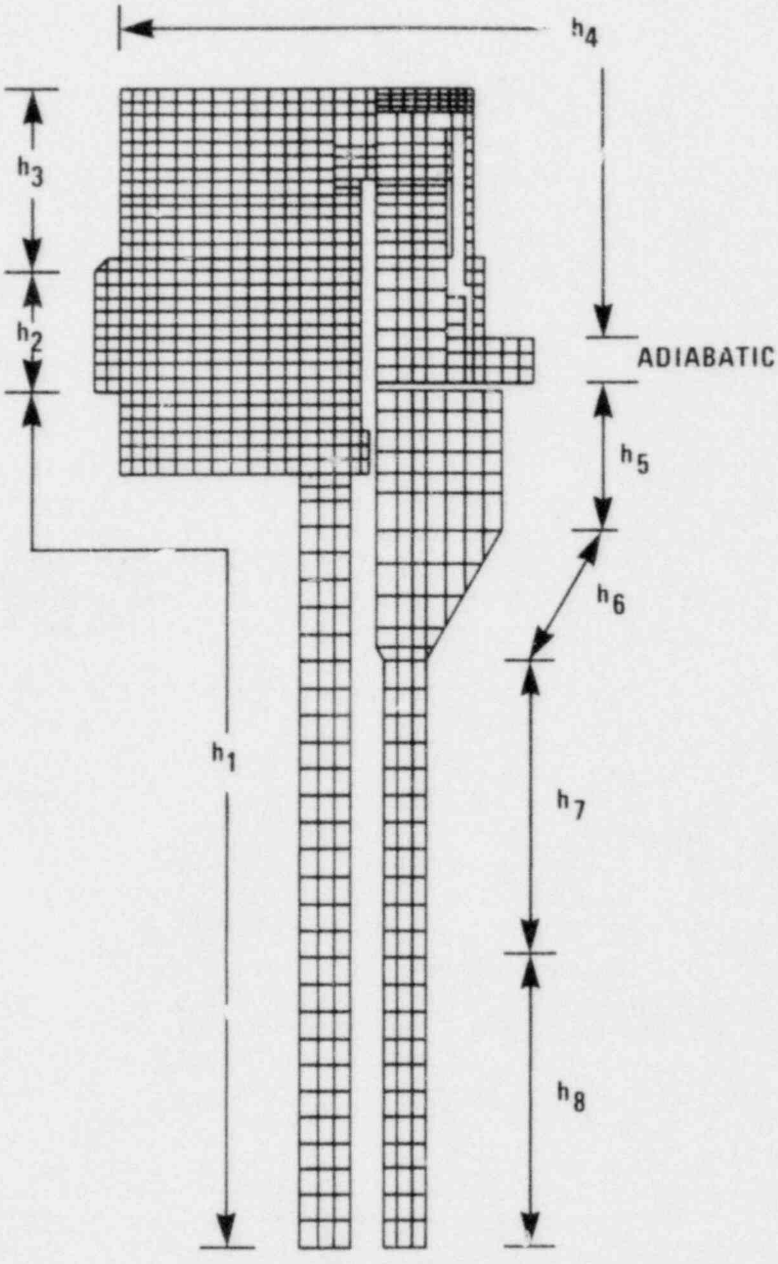

Figure 4.2-77. Heat Transfer Coefficient Regions for the CFS

4756-1

59 FIGURES 4.2-78 THROUGH 4.2-83 DELETED

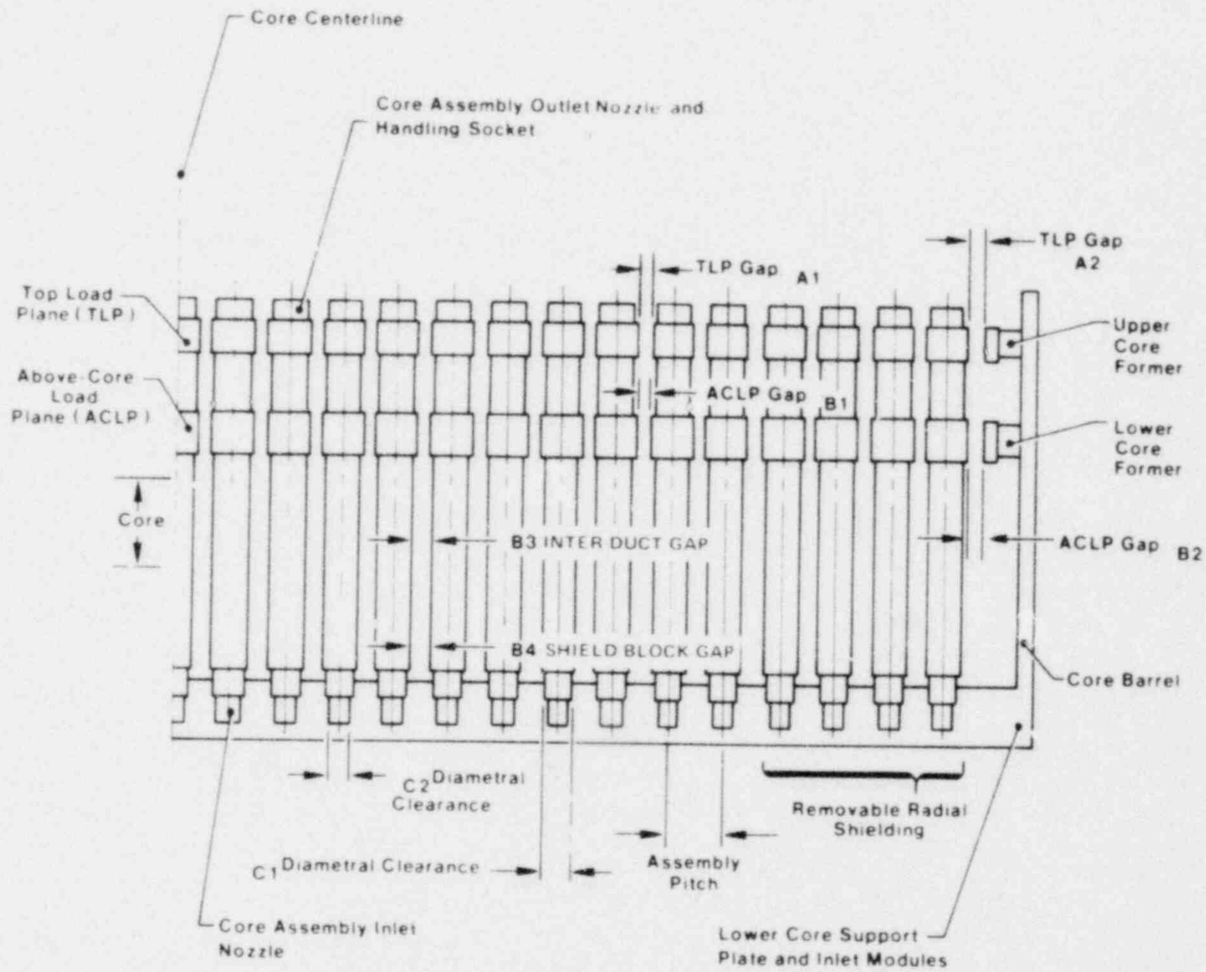

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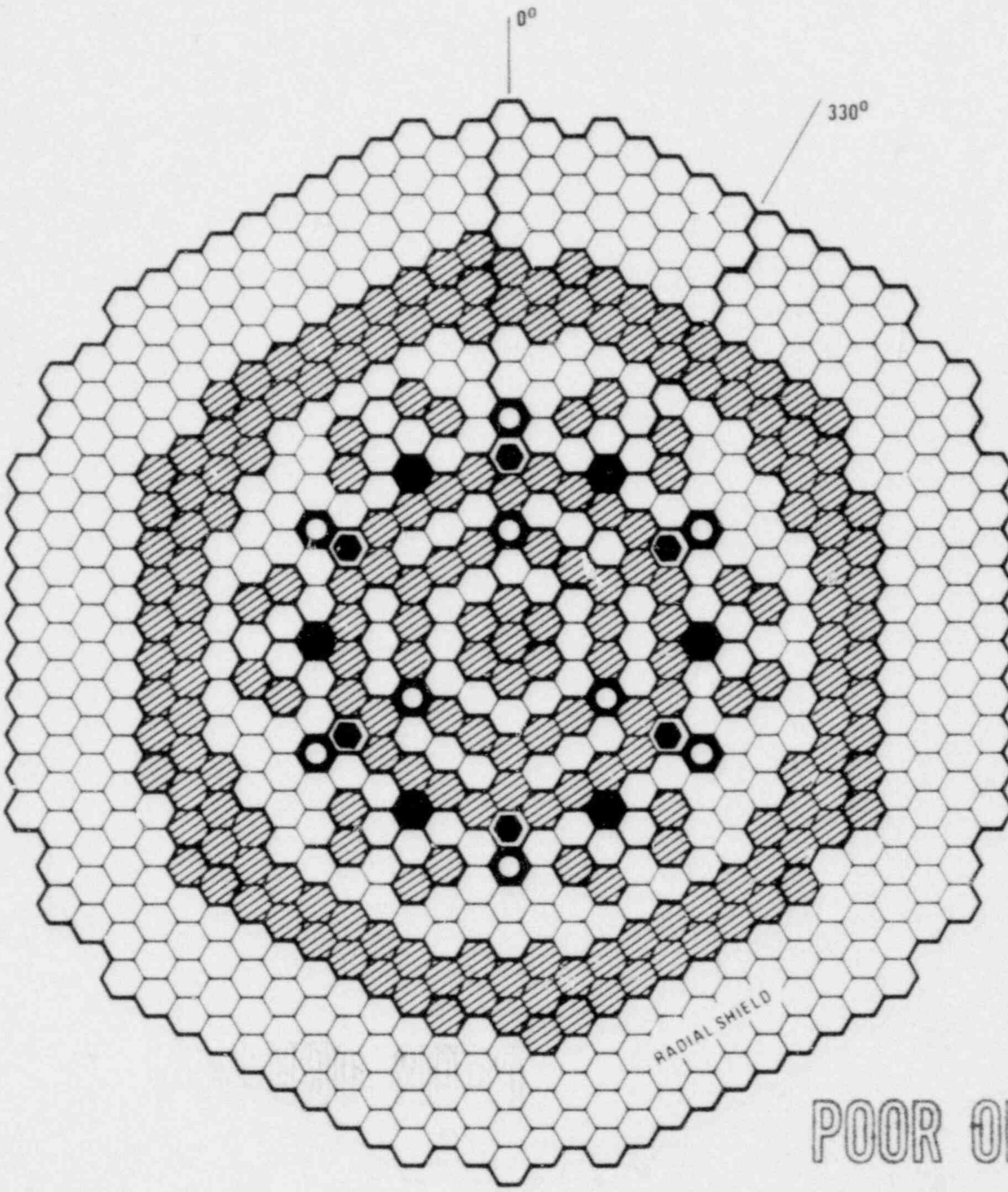
Figure 4.2-84. CRBRP Core Restraint System

4655-4



FUEL ASSEMBLIES

BLANKET ASSEMBLIES

CONTROL ASSEMBLIES

ALTERNATE FUEL BLANKET ASSEMBLIES

ROW NO. = RADIAL ROW FROM CENTER ASSEMBLY (CENTER = ROW 1)

FIGURE 4.2-85 Schematic View of Reactor Core Showing Sector of Core Simulated in NUBOW-3D Model

4655-1

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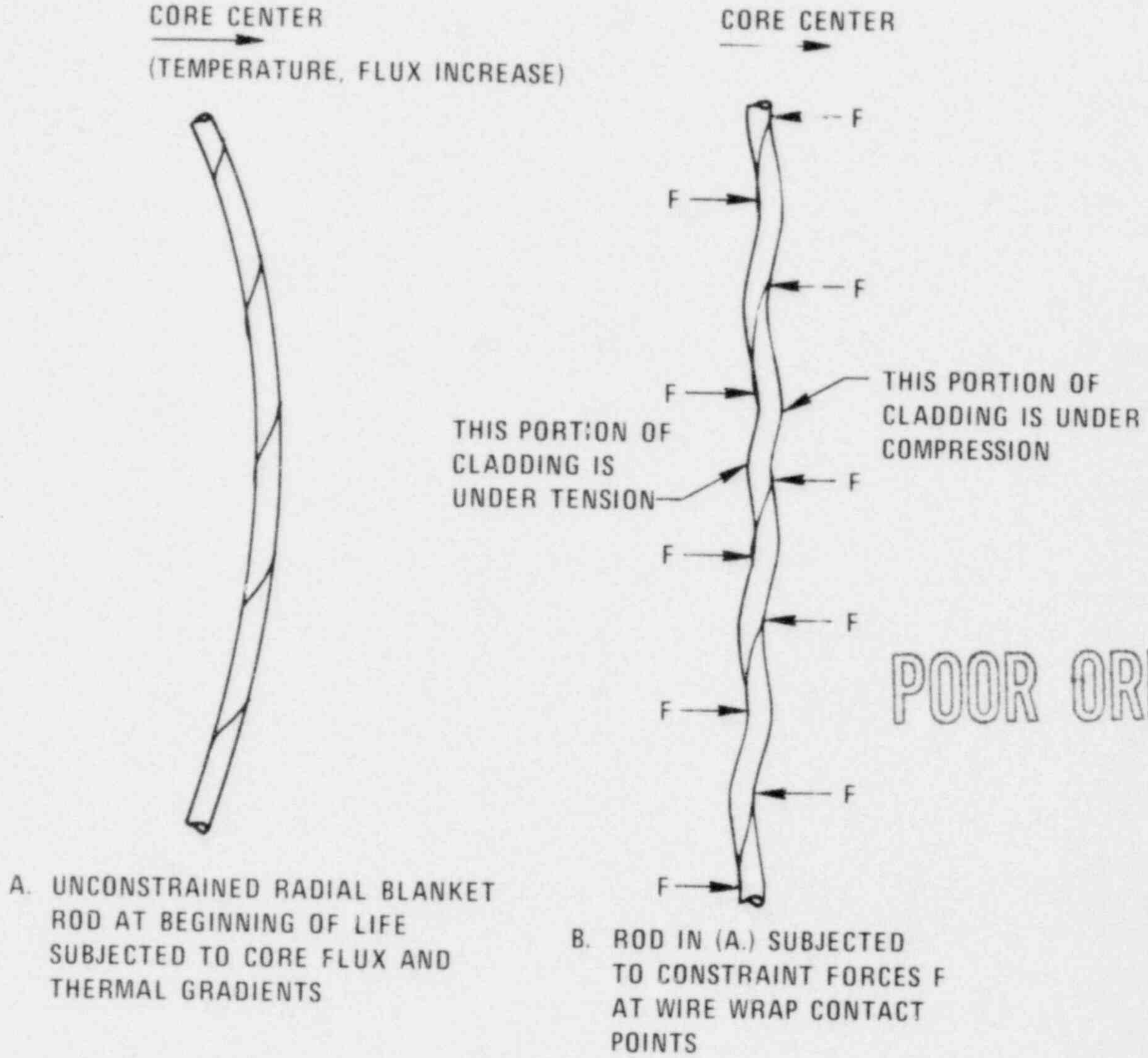

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Figure 4.2-87A. Illustration of Effects of Core Thermal and Flux Gradients in Radial Blanket Rod (not to scale)

7683-217

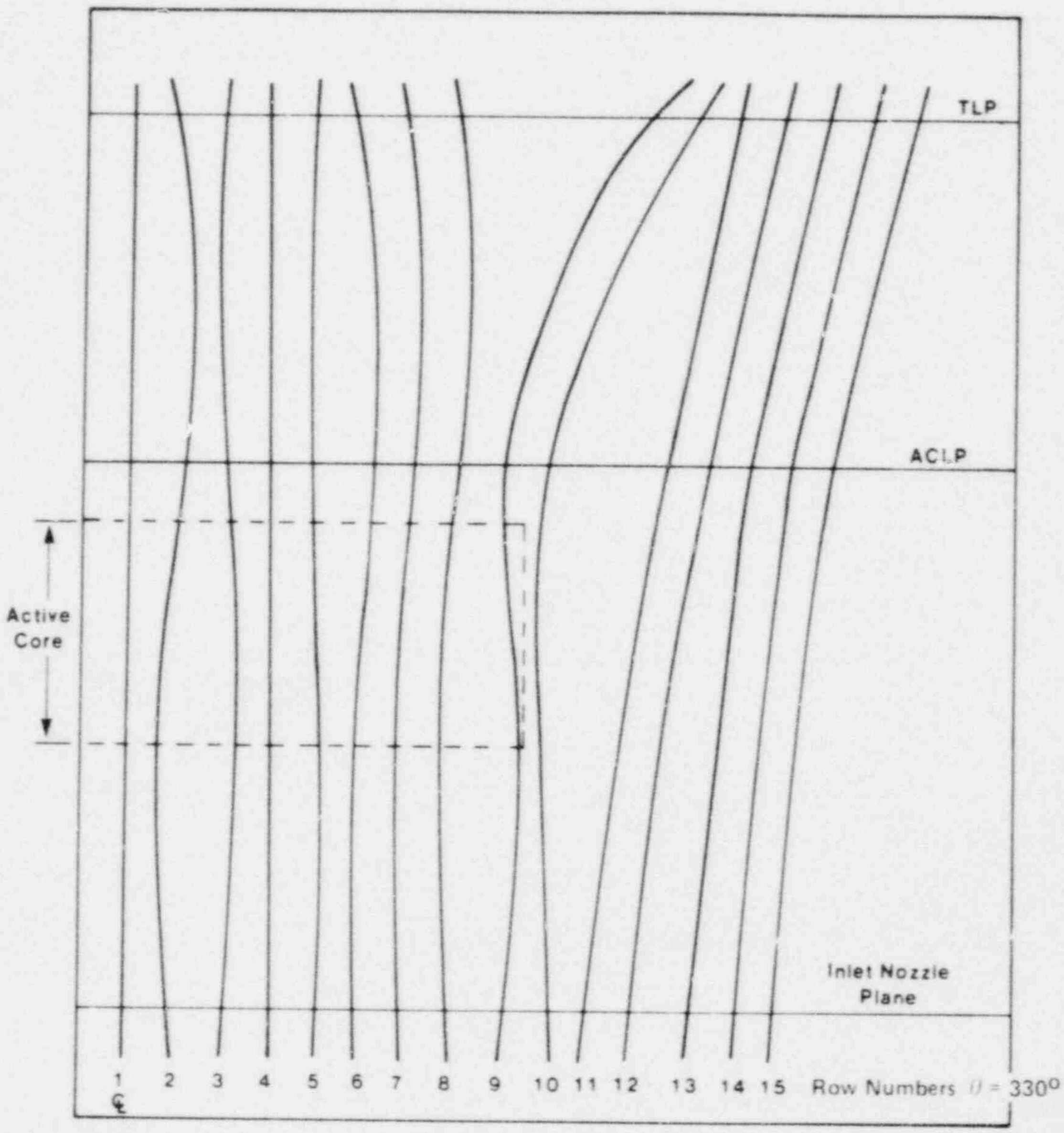

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Figure 4.2-88. On-Power Bow Shapes at SOC1

4655-3

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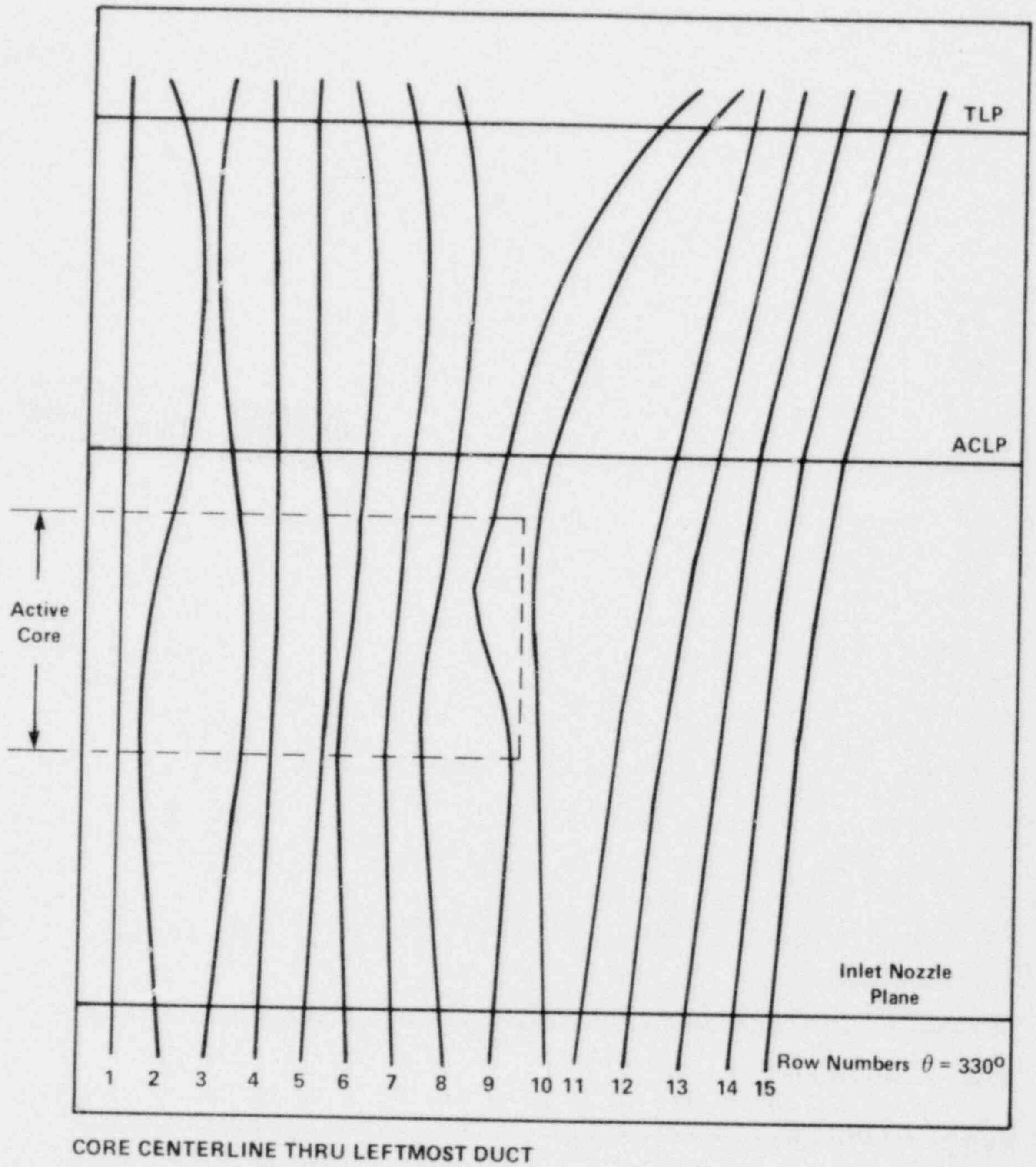

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FIGURE 4.2-89 On-Power Bow Shapes at EOC2

4655-10

Nominal Case, Time = 0.0 Days
Contact Force Level 3, ACLP

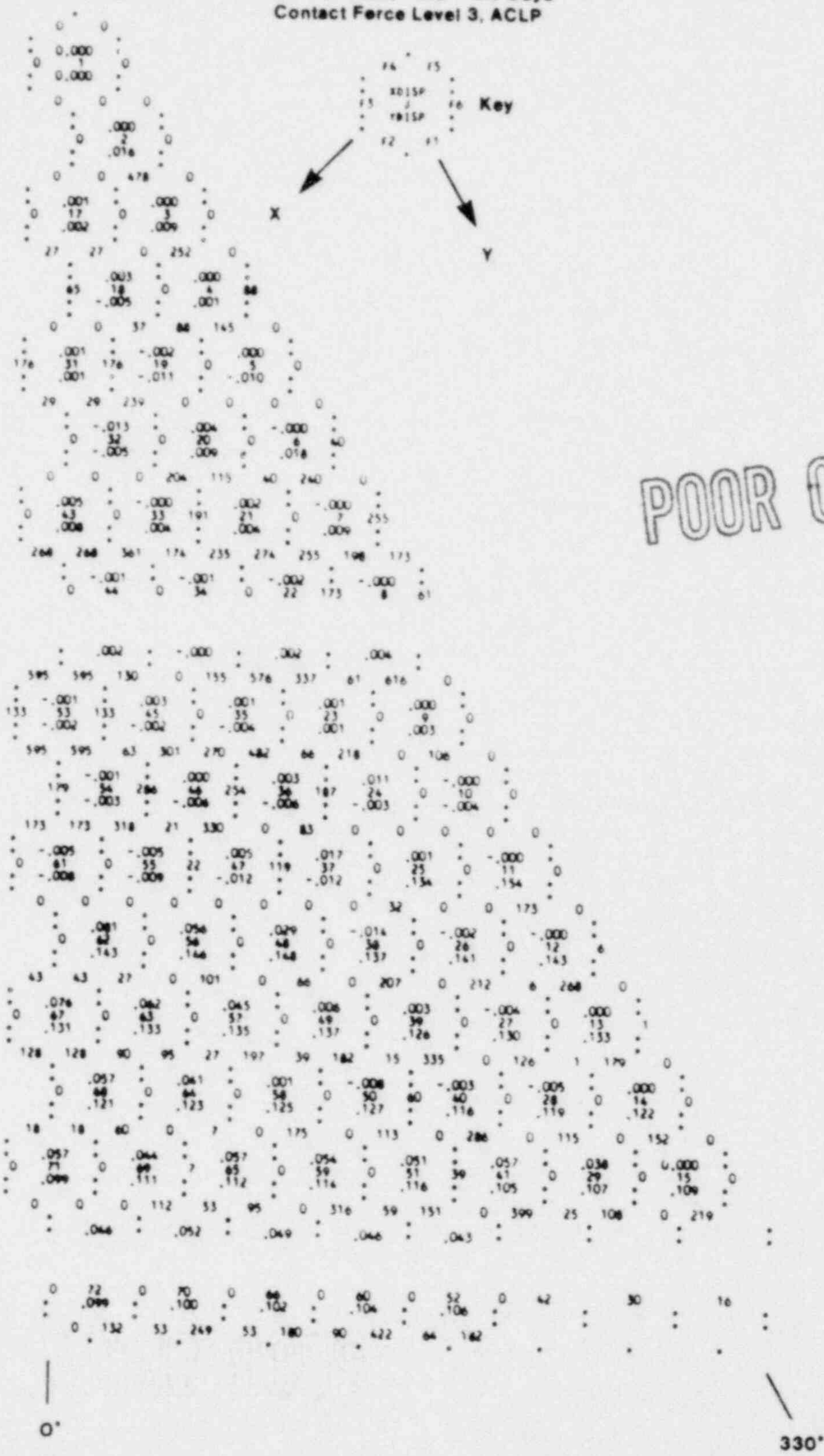

Figure 4.2-90. On-Power Forces and Displacements at the Above Core Load Plane at Start of Cycle 1.

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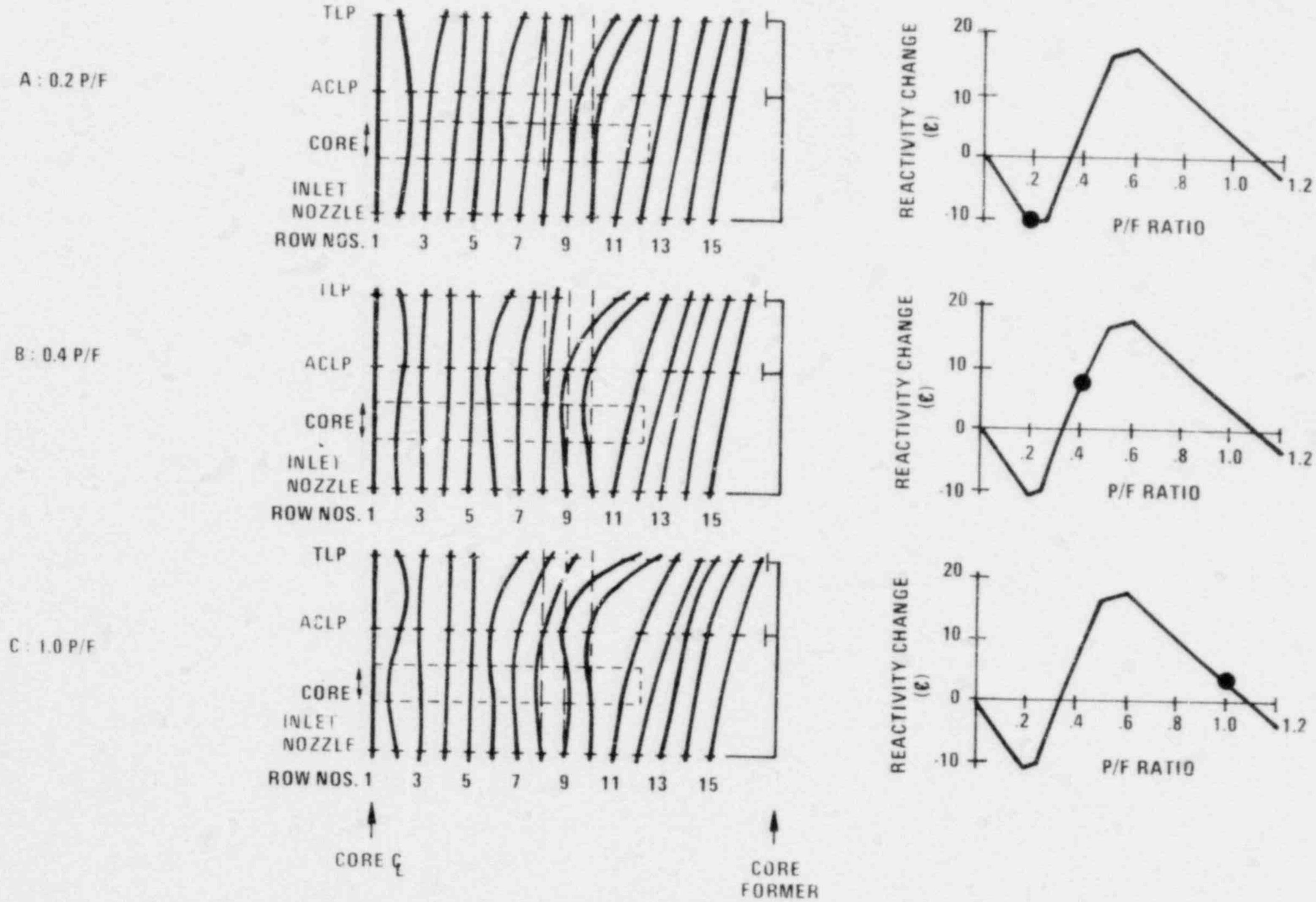

Figure 4.2-92a Bowing Displacements and Reactivity Change Versus Power to Flow Ratio

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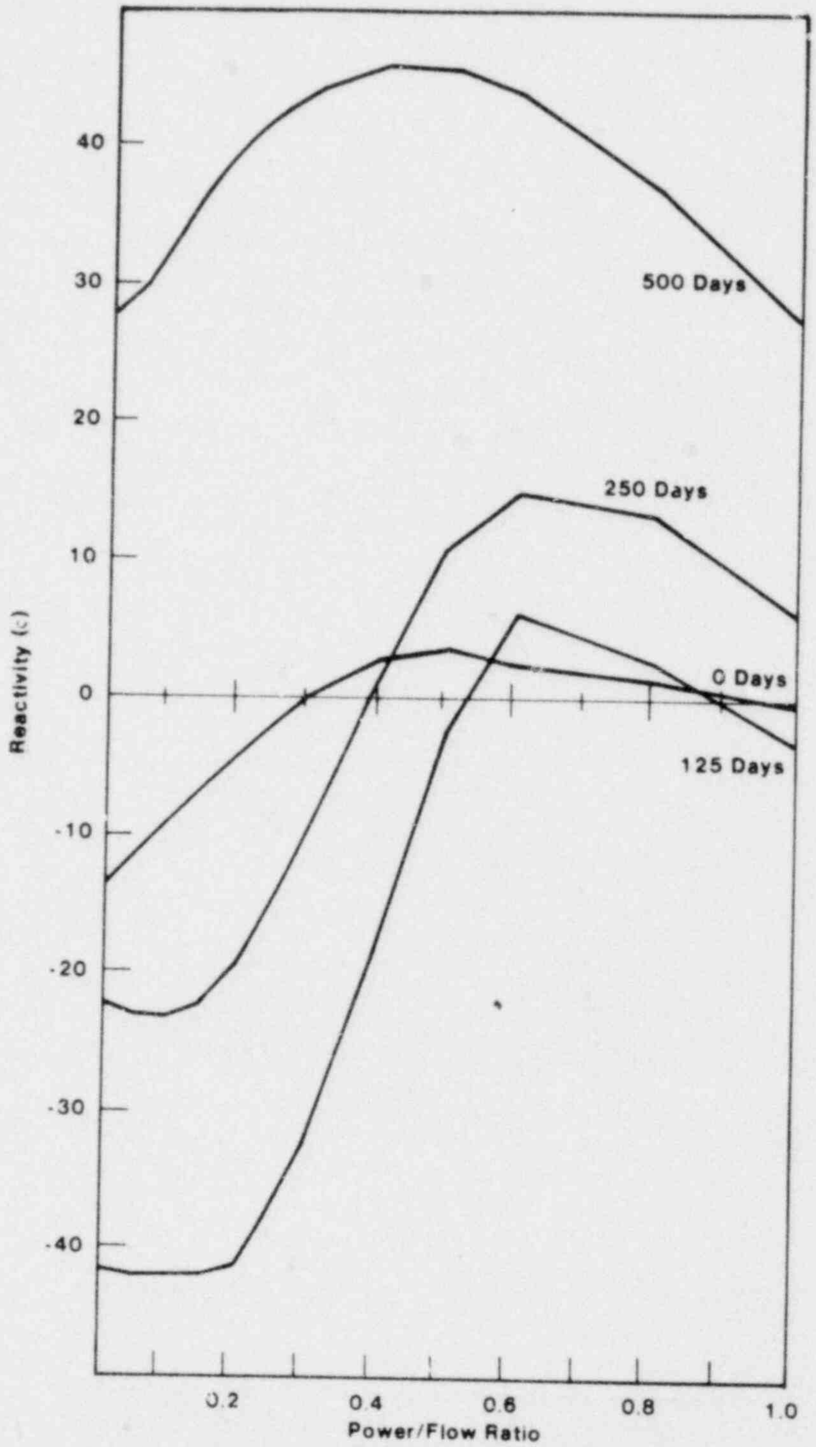

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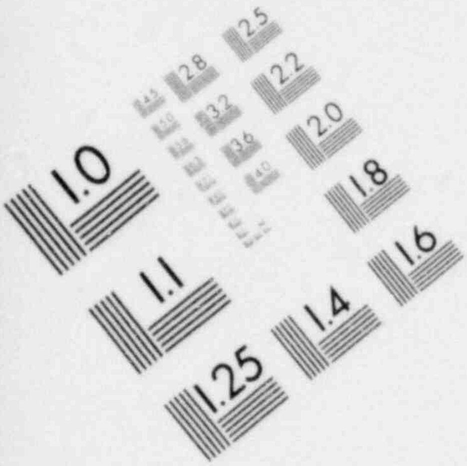
FIGURE 4.2-92B Bowing Reactivity Vs. Power-to-Flow Ratio at 40% Flow at Various Times in Fuel Assembly Life

4655-2

IMAGE EVALUATION
TEST TARGET (MT-3)

6″



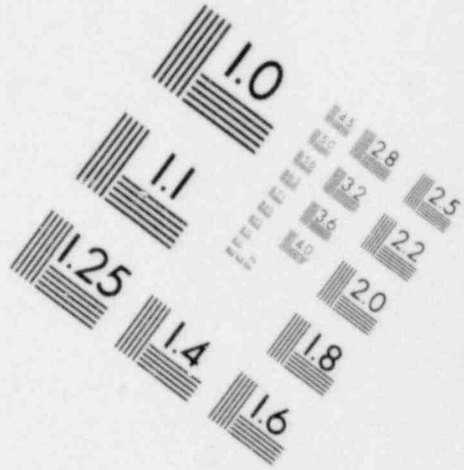
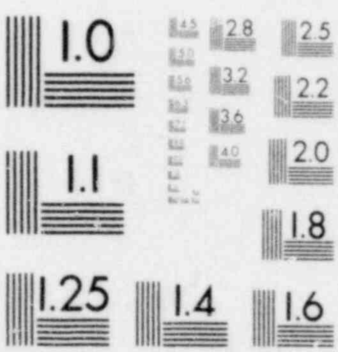


IMAGE EVALUATION TEST TARGET (MT-3)



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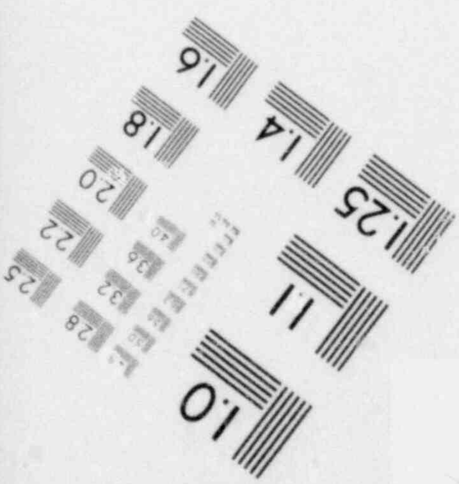

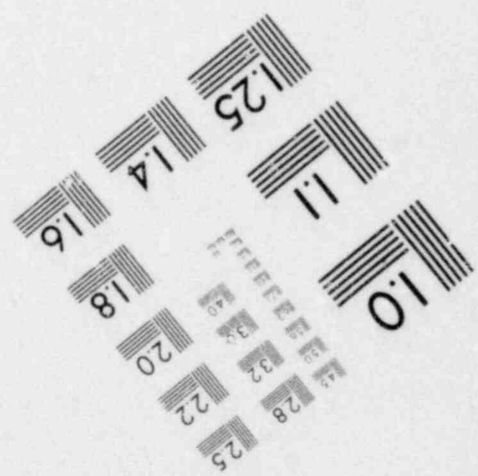

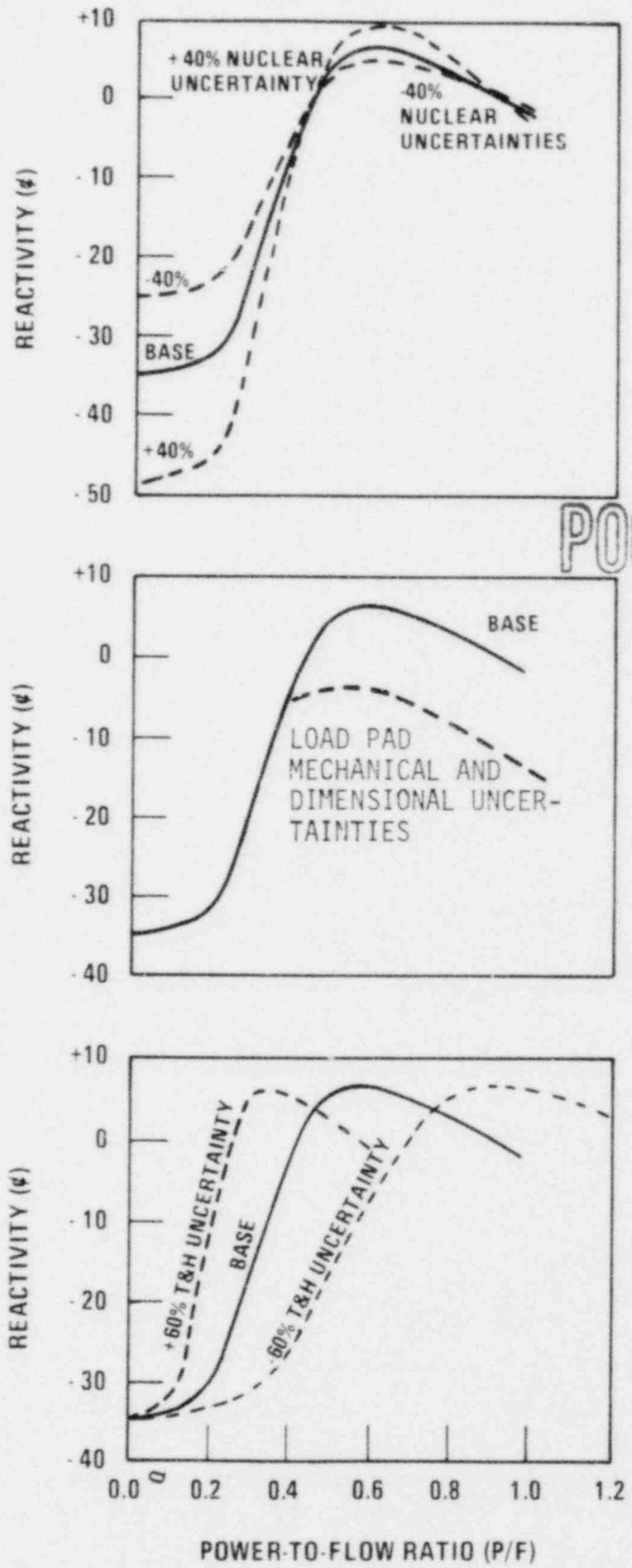

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FIGURE 4.2-92C Bowing Reactivity vs. Power-to-Flow Ratio

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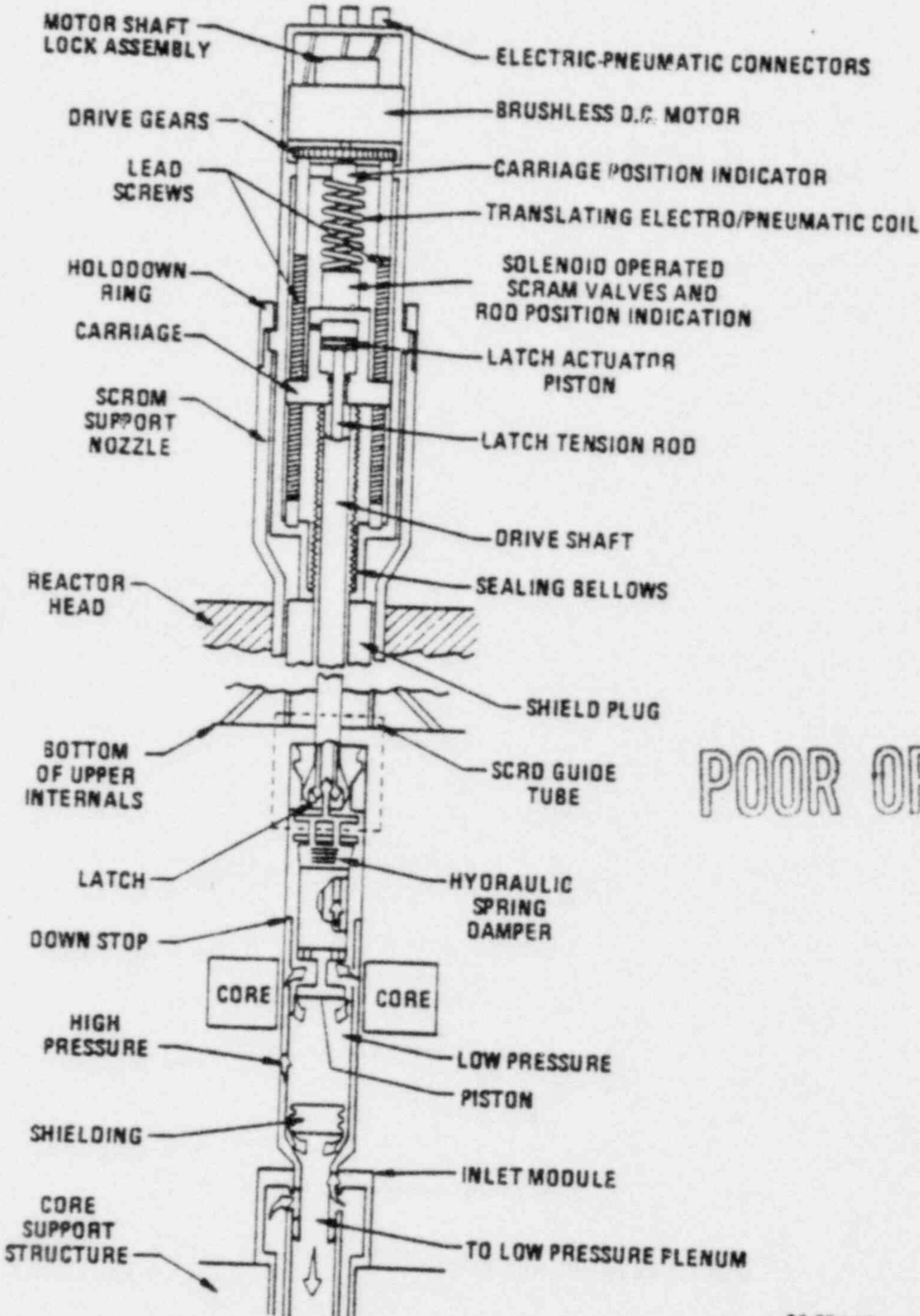

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Figure 4.2-105. SECONDARY CONTROL ROD SYSTEM SCHEMATIC

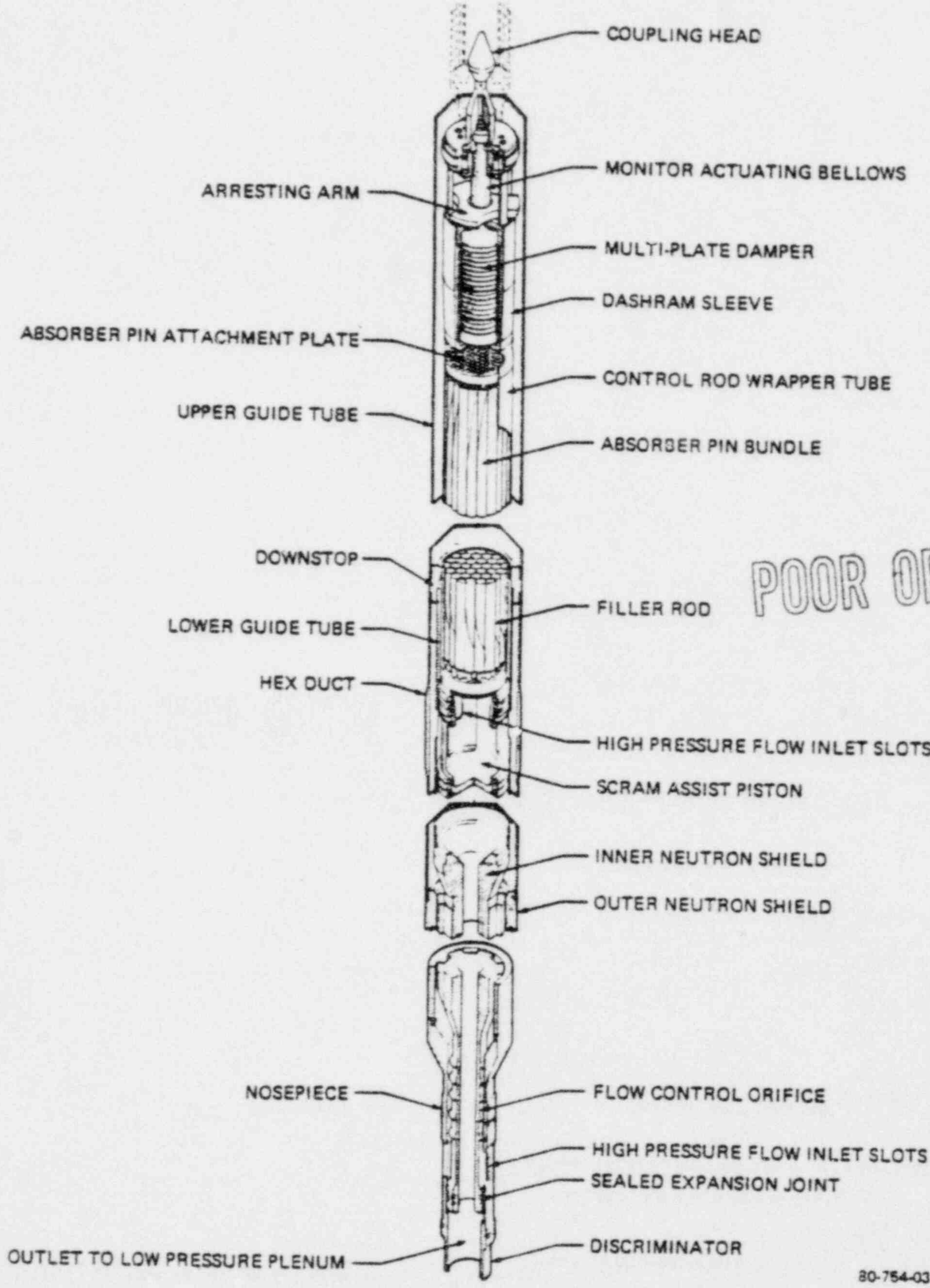

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Figure 4.2-106. SECONDARY CONTROL ASSEMBLY

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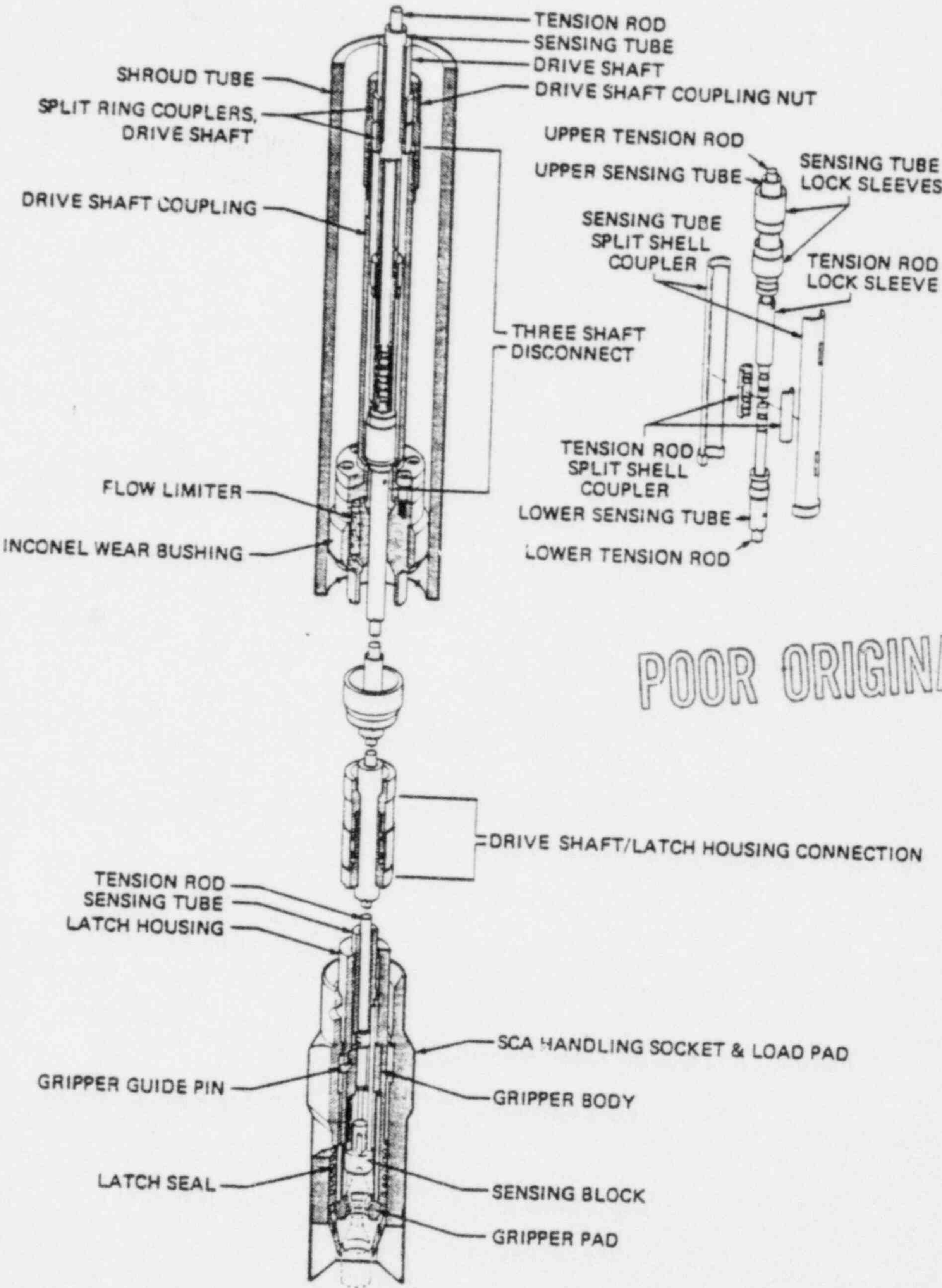

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Figure 4.2-107A. SECONDARY CONTROL ROD DRIVE LINE

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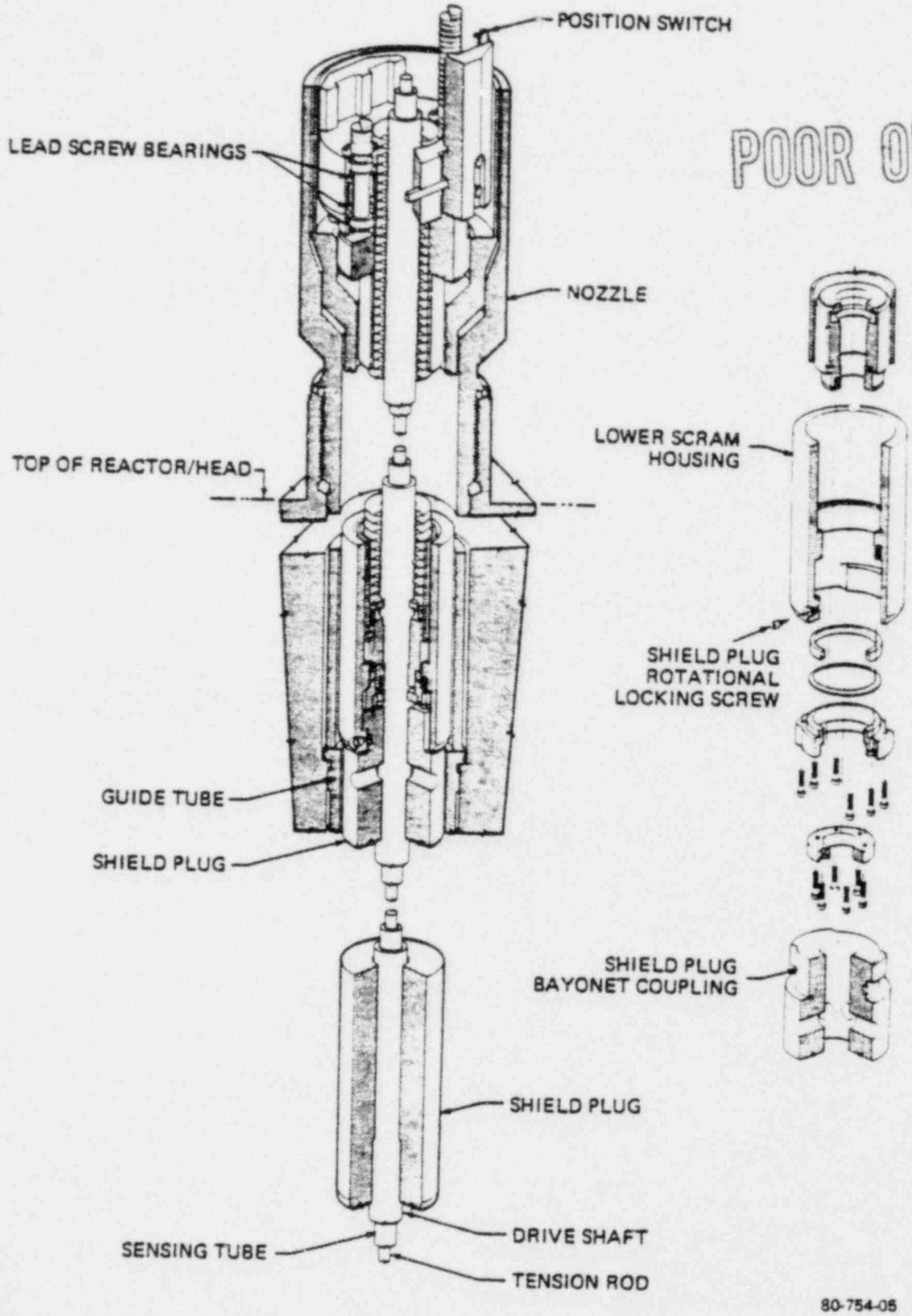

Figure 4.2-107B. SECONDARY CONTROL ROD DRIVE LINE (IN REGION OF LOWER SCRDM, NOZZLE, HOUSING AND SHIELD)

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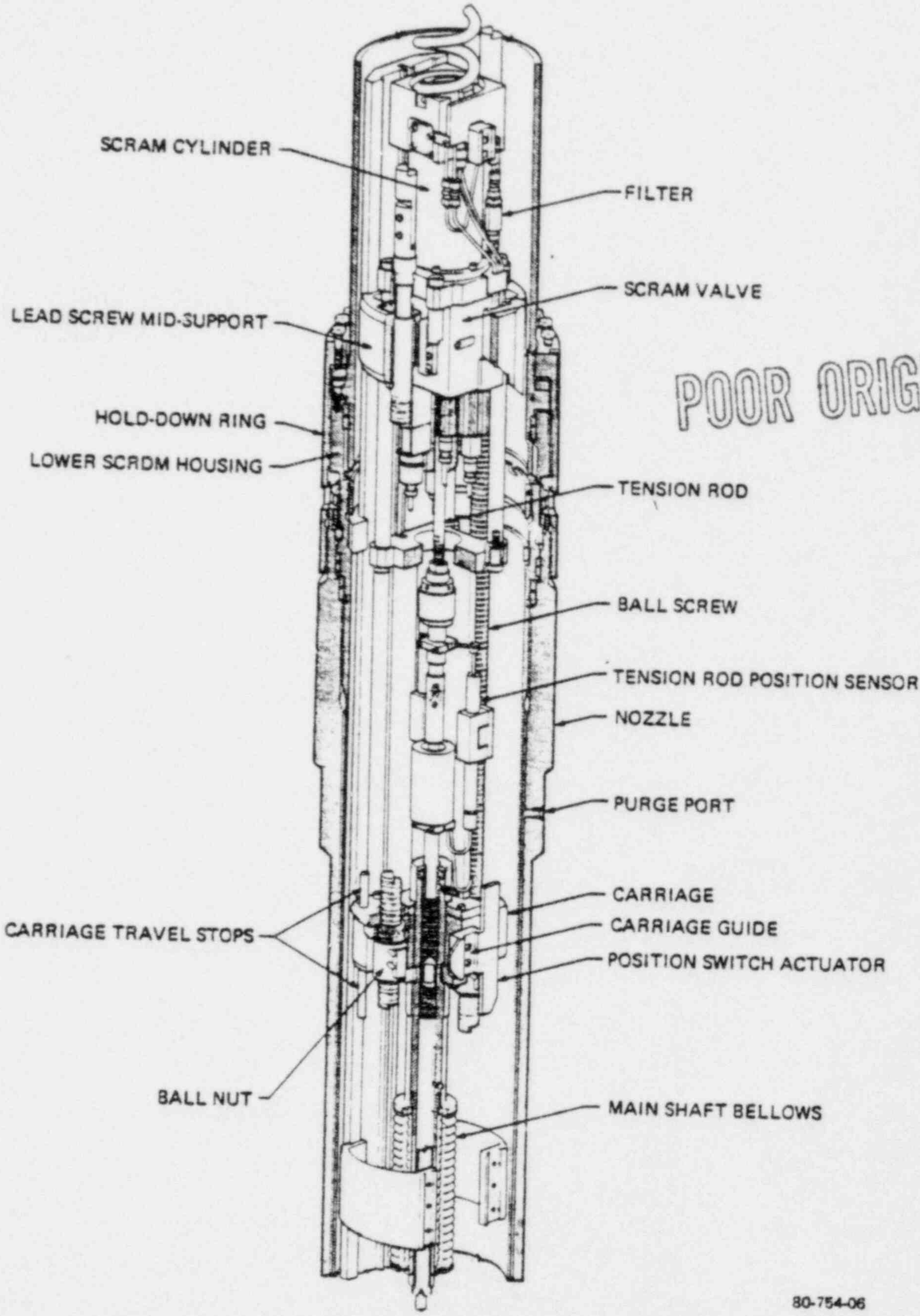

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Figure 4.2-108A. SECONDARY CONTROL ROD DRIVE MECHANISM

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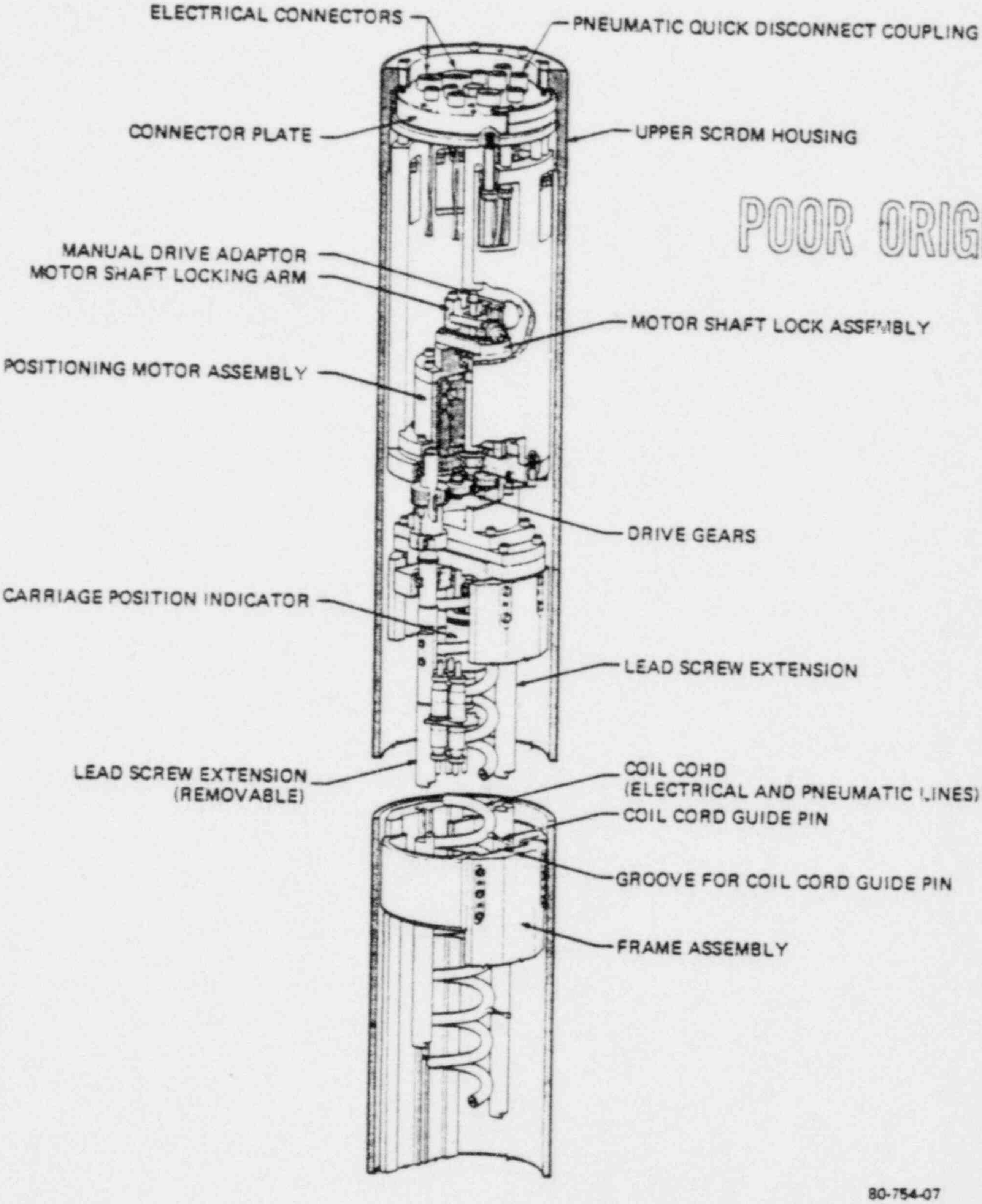

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Figure 4.2-108B. SECONDARY CONTROL ROD DRIVE MECHANISM

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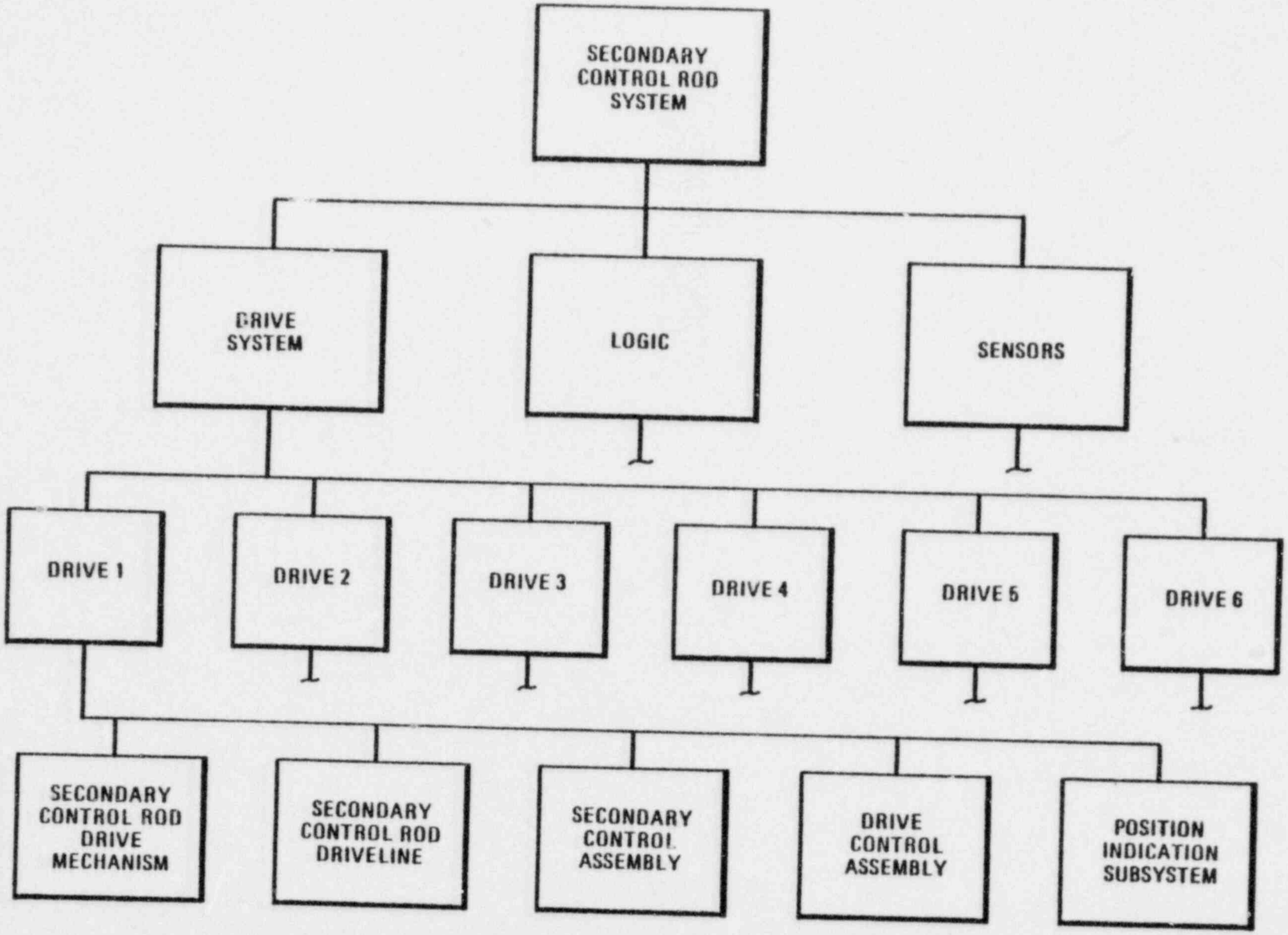

Figure 4.2-123. SECONDARY CONTROL ROD SYSTEM

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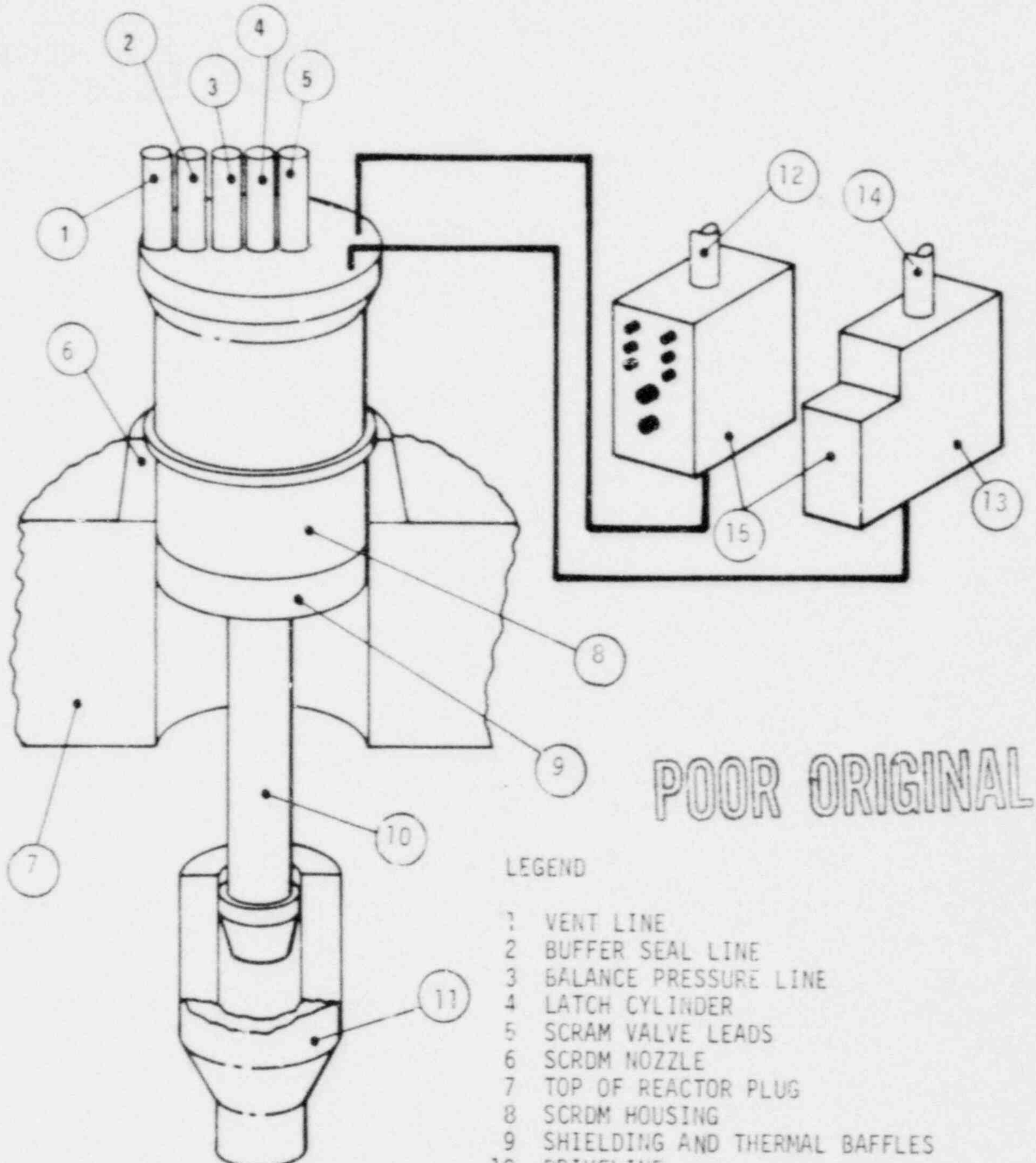

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LEGEND

1 VENT LINE
2 BUFFER SEAL LINE
3 BALANCE PRESSURE LINE
4 LATCH CYLINDER
5 SCRAM VALVE LEADS
6 SCRDM NOZZLE
7 TOP OF REACTOR PLUG
8 SCRDM HOUSING
9 SHIELDING AND THERMAL BAFFLES
10 DRIVELINE
11 SCA DUCT AND HANDLING SOCKET
12 POWER LEAD
13 POSITION INDICATOR MODULE
14 POWER LEAD
15 DRIVE CONTROL MODULE

Figure 4.2-124 Secondary Control Rod System

6664-105

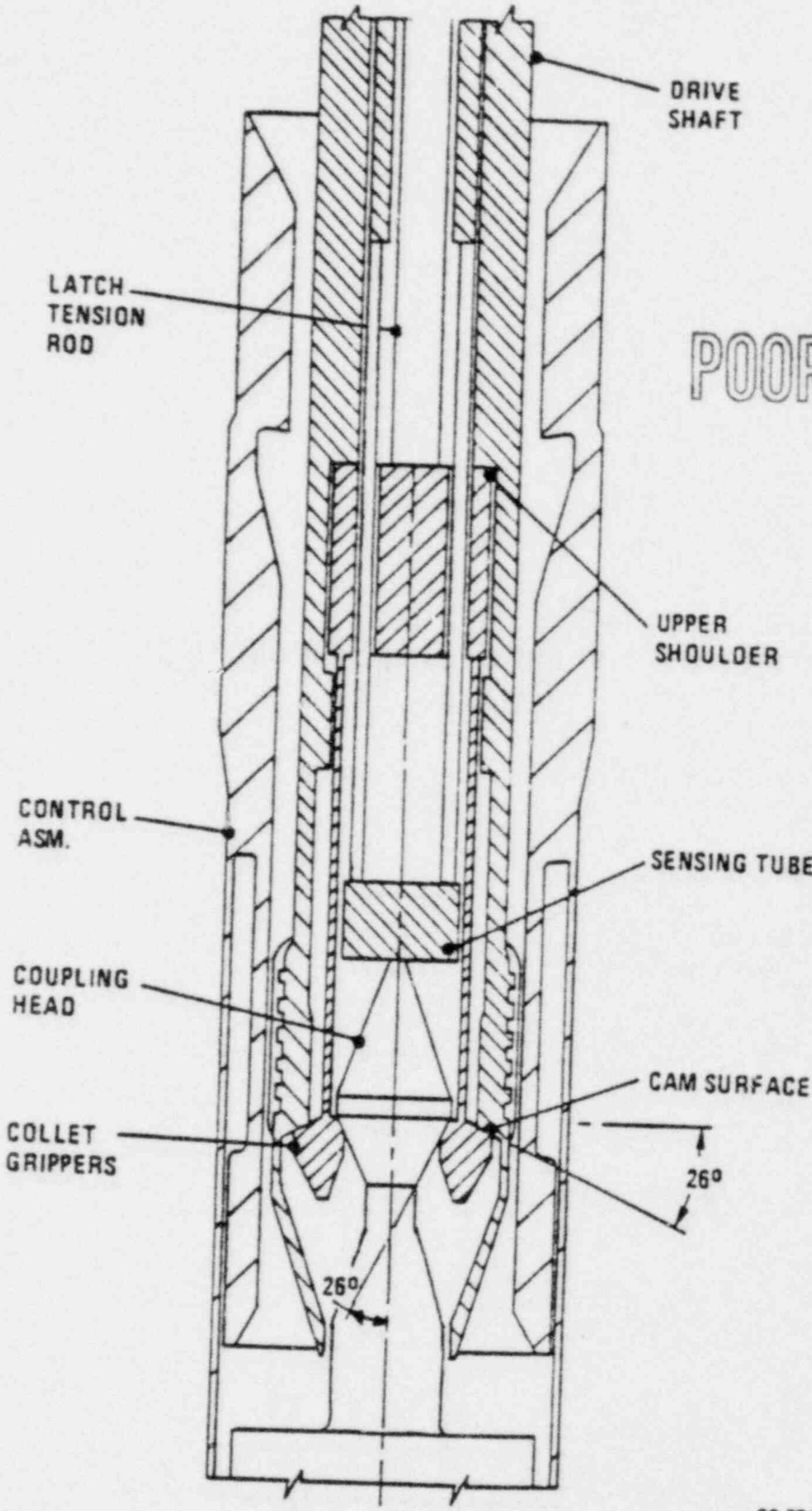

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Figure 4.2-125. COLLET TYPE LATCH

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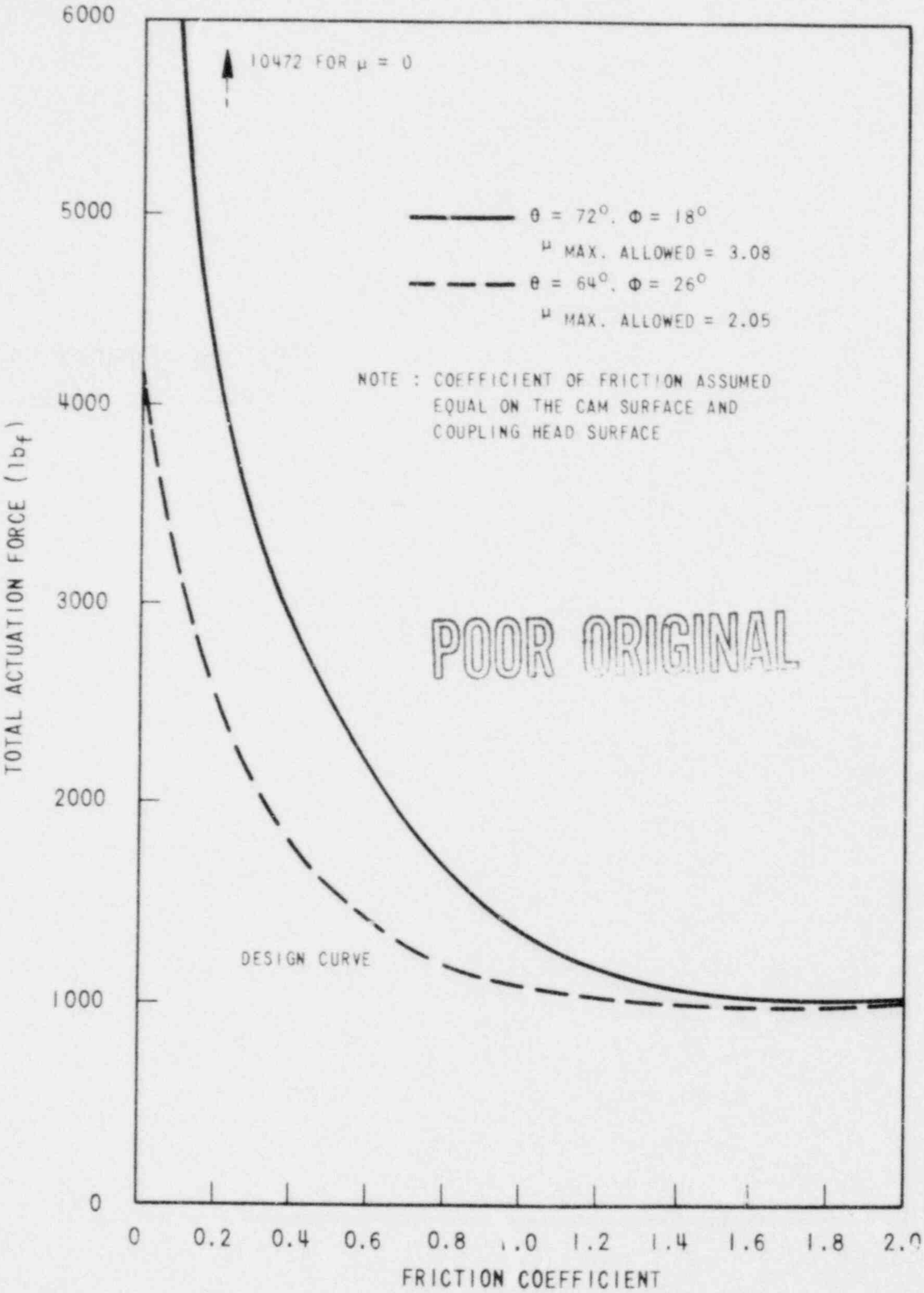

Figure 4.2-126. Total Actuation Force Versus Friction Coefficient

6664-149

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4.3.2.1.4 Multi-Group Nuclear Cross Section Data

A complete description of the method used to generate the multi-group neutron cross sections is given in subsection 4.3.3.2.

4.3.2.1.5 Source Range Flux Monitoring System

The basic function of the source range flux monitor (SRFM) system is to follow the reactor low power neutron flux and to provide that information in a usable form so that a determination of the neutron flux status
54 of the reactor from shutdown to low power (a few Kw) can be made during normal plant operation.

The following two operational requirements have been imposed on the SRFM system to achieve successful operation. First, the instrumentation must provide continuous flux monitoring during low power and normal shutdown operation. Second, during any subcritical operations, other than intentional approach to criticality, it must provide a warning to assure that the reactor does not approach criticality any closer than the worst single refueling error.

The SRFM instrumentation satisfies the first requirement by providing a means of following the reactor flux at both low power and shutdown conditions. This includes the capability for normally measuring the shutdown neutron flux level at all times while fuel is in the core as required for operational control and safe reactor operation. The design and operation of the SRFM will assure that significant changes in the reactor flux level can be detected. The SRFM will also accurately and reliably follow the reactor flux from full shutdown to low power operation. The SRFM provides an output signal that is proportional to the reactor power level from hot standby conditions (zero neutronic power) up to the top end of its sensitivity range.

The second requirement specifies that during refueling, the SRFM must provide a warning to the operator and thereby assure that the reactor does not approach criticality any closer than that level from which criti-
54 cality could be attained by a single refueling error, with adequate margin for the associated uncertainty. An alarm sounds in the control room if the minimum shutdown reactivity based on this criterion is exceeded.

The source range flux monitoring system consists of three redundant channels that will monitor the flux level of the reactor core from shutdown power through low power. The measurements will be performed at the reactor midplane by three redundant BF_3 neutron counters located in
51 graphite blocks and spaced approximately 120 degrees apart in the reactor cavity external to the guard vessel.

The inherent neutron source in the core, due to spontaneous fission and (α,n) reactions, is multiplied within the core and is the source of neutrons at the ex-vessel SRFM locations during shutdown (source multiplication). The count rate at the SRFM is inversely proportional to the subcritical reactivity and directly related to the inherent neutron source strength in the fuel. Consideration will be given to the effect of control rod insertion to achieve shutdown and changes in the spatial source importance during refueling on the proportionality constant (calibration constant) when required. The derivation of the source multiplication equation is given later in this subsection. SRFM calibration measurements using the CRBR control system will be required for relating subcriticality and count rate. This calibration is performed with a control rod of known reactivity worth. The derivation of the inverse kinetics rod drop technique, which is used to establish the worth of the control rod, is also presented later in this subsection.

When fuel assemblies are present in the Fuel Transfer and Storage Assembly (FT&SA) no core component exchanges of any kind are anticipated. Under these conditions, the only requirement placed on the SRFM detectors is that they must provide continuous monitoring of the flux status of the reactor. The FT&SA and SRFM are being designed so that for FFTF grade fuel, the nominal background count rate from one maximum discharge and one fresh fuel assembly in the FT&SA will be one-fourth (1/4) of the nominal foreground count rate from the core at the two SRFMs not adjacent to the FT&SA. For LWR recycle fuel, the nominal background count rate will be one-half (1/2) of the nominal foreground count rate at the same two detectors with the
59 same two assemblies in the FT&SA.

Five primary nuclear characteristics of the SRFM design were investigated to justify its use in the Clinch River Breeder Reactor. These characteristics are:

a. The expected count rate at the ex-vessel location. This count rate is a function of the neutron flux spectrum and instrument sensitivity.

b. The proportional relationship between reactor power and count rate at the instrument location. A principal function of the SRFM is the monitoring of reactor power through neutron flux detection. Non-linearities between neutron flux at the detector location and reactor power could be introduced by the presence of fuel assemblies in the Fuel Transfer and Storage Assembly or control rod movement.

c. The ability to isolate the detectors from neutron flux which may arise from sources other than the reactor core.

51

During periods of reactor operation which required SRFM operability, the Flux Monitor System Mechanical Components (FMSMC) shall provide moderator and shielding to maintain the following neutron foreground-to-background
59 ratios:

o For reactor startup, with no core assembly of any kind resident in the FT&SA; or normal refueling with no core assembly of any kind in the FT&SA, and with any core assembly being exchanged within 64 inches of its fully inserted core lattice position, all three SRFM Detectors shall have nominal foreground-to-background ratios greater than:

"FFTF Grade" Fuel	10:1
"LWR Recycle" Fuel	7:1

o In the fully shutdown condition, with a maximum of one spent and one fresh core assemblies resident in the FT&SA, and with no core assembly being exchanged two-out-of-three SRFM Detectors shall have nominal foreground-to-background ratios greater than:

"FFTF Grade" Fuel	4:1
"LWR Recycle" Fuel	2:1

These foreground-to-background ratios shall be calculated based on the following nominal conditions as applied to the last core assembly interchange during a particular refueling period:

o The discharge fuel assembly with the highest neutron source strength is withdrawn from the central core matrix position 64 inches above its fully inserted position, and

o All the other fuel and blanket positions in the core are loaded with fresh assemblies.

With all the nuclear and shielding uncertainties applied in the most conservative manner, foreground-to-background ratios at the SRFM Detectors as small as 1.4:1 for FFTF grade fuel and 1.0:1 for LWR recycle fuel (all three SRFM Detectors with no core assembly in the FT&SA); and 1.0:1 for FFTF grade fuel and 1.0:2 for LWR recycle fuel (two-out-of-three SRFM Detectors with a maximum of one spent and one fresh core assemblies in the FT&SA) may be possible. These lower ratios will not degrade the reactivity monitoring capability below minimum requirements nor cause spurious alarms for operations allowed under the above stated conditions. This requirement applies to the range of subcriticality below that level defined
59 by the alarm setpoint.

d. The gamma background at the detector location. The gamma background will be a primary factor in determining instrument sensitivity at a given location.

e. The ability of the SRFM to accurately determine the reactivity worth of in-core control rods by means of the inverse kinetics rod drop technique. The application of this IKRD technique at the ex-vessel detector locations is required to properly calibrate the SRFM system for subcriticality determination.

Extensive analyses at ARD and both analyses and experiments at Oak Ridge National Laboratory (Reference 1 and 2) have been performed in support of the ex-vessel SRFM system. Particular emphasis was placed on investigating the five nuclear characteristics listed above. The significant results of the analyses and experiments performed to date are summarized below.

Calculations were performed to assess the magnitude and spectrum of the neutron flux at the ex-vessel SRFM location during shutdown conditions. These calculations were performed for beginning-of-life conditions (all fresh fuel, FFTF-grade plutonium in the core). The minimum shutdown flux at the SRFM locations (beginning-of-life conditions) was calculated to be approximately 0.1 nv, which corresponds to about
54 4 counts per second at each BF_3 proportional counter. The magnitude of this count rate which is smaller than during any subsequent refueling sequence, assures good counting statistics for monitoring subcriticality and refueling operations. Additional calculations have shown that the neutron flux is almost fully thermalized (~85% below 0.1 ev) at the SRFM location, eight inches behind the front face of the graphite block. This enhancement of the thermal flux inside the graphite block has been confirmed by experiments performed by ORNL at the Tower Shield Facility near Oak Ridge, Tennessee (Reference 1).

To investigate the effect of core configuration on count rate, the homogeneous core configuration was modified by employing different banks of control rods to maintain a fixed reactor power level and K_{eff}. For these reactor configurations, the flux level at the SRFM varied by less than 10%. This result shows that the ex-vessel detectors are not sensitive to changes in the homogeneous core configuration during constant power operation. The detector response is proportional to the power level of the reactor. Similar calculations will be repeated for the present heterogeneous core layout and the results will be reported.

Regarding the possibility of the detector monitoring neutron flux from sources other than the core, analyses have shown that the flux monitoring requirements for the SRFM can be satisfied with as many as
51 four fuel assemblies in the FTSA. Shielding configurations are required

54

The count rate due to background is minimized by shielding in the form of boron carbine slabs which surrounds all sides of the graphite block
59 except the front face. The shielding is used to reduce the count rate from neutrons which are scattered into the graphite block from the reactor cavity walls. Normal refueling procedures will require that no assembly of any type be in the FT&SA while any assembly is being inserted the last 64 inches into the core or withdrawn the first 64 inches from the core so that all three SRFM detectors will have an unhindered view
59 of the core. For this case, the requirement of (c) above are imposed.
54

The gamma dose at the SRFM location immediately after shutdown has been analyzed in detail to assure that the sensitivity of the BF_3 neutron counters is not adversely affected. The type of BF_3 neutron
54 detectors to be used in CRBR have a minimum sensitivity of 40 counts per second/thermal equivalent nv for gamma dose rates less than 100 R/hr. When the gamma dose exceeds 100 R/hr the detector sensivitity falls off rapidly. Calculations have shown that the local gamma dose rate at the SRFM location is less than 100 R/hr with appropriate shielding in front of the graphite block and in the other locations as required.

A principal function of the SRFM is to determine the subcritical reactivity of the CRBRP based on proper calibration of the instrumentation near critical. The recommended method for calibrating the SRFM detectors is a two step procedure. First, a known value of negative reactivity must be established. This is accomplished by using the SRFM count rate trace that results from scramming one or more control rods to determine the reactivity worth of the scrammed rods. This is known as the inverse kinetics rod drop (IKRD) technique. Second, the calibration constant, which relates the subcritical reactivity to the count rate, must be determined. This is accomplished by inserting the previously measured reactivity worth (the same control rods described above) and noting the corresponding count rate. This same calibration constant is then used to imply subcritical reactivity when all the control rods are inserted and the reactor is fully shutdown.

This procedure depends strongly on the accurate determination of the negative reactivity worth of the scrammed control rods by means of the IKRD technique. ORNL has performed numerous rod-drop experiments (Reference 2) in the Tower Shield Facility in addition to analytical calculations and both have supported the conclusion that reactivity interpretations, based on the change in count rate at the ex-vessel detectors, are consistent with in-core detectors. The experiments and analyses performed to date have not included the effect of neutron streaming in the reactor cavity. Future analyses will investigate these reactor cavity effects and the results will be included in the FSAR.

The neutron source multiplication technique is employed to monitor the subcritical reactivity state of the reactor during the loading to critical and all subsequent fuel reloadings. The relationship between the steady-state SRFM detector count rate and the subcritical reactivity
51 is derived from the point kinetics equations:

e. Nominal Doppler, Sodium Density, and Axial Expansion Feedback

The transfer function for all the nominal feedback coefficients is shown on Figure 4.3-42, Curve F, which shows excellent, stable frequency response characteristics. This is mainly due to the strong stabilizing effect of the Doppler constant as can be seen by comparing Curve F with Curve E. This case includes all the nominal feedback coefficients (Figure 4.3-41) and represents a high degree of stability.

f. Nominal Doppler Feedback

Figure 4.3-43 shows several transfer functions previously discussed along with the transfer function with Doppler feedback alone. The amplitude scale has been expanded by a factor of 10 to better show the effects of Doppler only, Curve A, compared to the case of Doppler and sodium expansion feedback, Curve B, and the case of all nominal feedbacks, Curve C. These curves clearly show the dominance of the Doppler effect.

g. Reactor Assembly Bowing Feedback

The reactivity feedback due to reactor assembly bowing is a function of power to flow ratio (P/F) as discussed in Section 4.2.2.4.1.8.3 and in Sections 4.3.2.3.4 and 5. From Figure 4.2-92B the slope of the reactivity curve, with worst-case uncertainties, is positive over a limited range, but again turns negative above P/F equal to about 0.7. This implies that the bowing reactivity feedback coefficient is negative above that point. The nominal P/F ratio is maintained at a value of 1.0 above 40% power so that no further bowing occurs. Neglecting the effect of this coefficient is conservative for variations in power of up to 30% or variations in flow up to 42% about nominal reactor operating power and flow levels. The stability predictions for the nominal power operating range are not affected by neglecting bowing feedback.

The results of Figure 4.3-44 are obtained with reactor neutronic parameters that exist at EOC4. The net feedback from sodium density in this case is slightly positive. The counteracting negative feedback components are from the Doppler effect and the axial expansion.

a. Doppler Feedback

The transfer function for Doppler feedback only is shown in Figure 4.3-44, Curve A, to provide a comparison with the other transfer functions.

51

b. Sodium Density (Twice Nominal) and Doppler Feedback (Half Nominal)

For the transfer function of Figure 4.3-44, Curve B, credit is taken for only half of the nominal negative Doppler, whereas the positive sodium density feedback is taken at double its nominal value. The system is stable and the transfer function has acceptable frequency response characteristics. Clearly, the effects of the Doppler feedback, even taken at a value well below the smallest value expected, is dominant.

c. Nominal Doppler, Sodium Density, and Axial Expansion Feedback

The transfer function for all the nominal EOC4 feedback coefficients is shown in Figure 4.3-44, Curve C, which shows excellent, stable frequency response characteristics.

Thus, for a wide range of variation of feedback coefficients, the CRBR is a stable, well-behaved system in the response of the reactor to reactivity perturbations. The main stabilizing feedback is due to Doppler, and even at half-nominal value, in any combination
59 with the other feedback coefficients, a stable system results.

4.3.2.8.2 <u>General Stability Analysis (Bounded Input-Bounded Output)</u>

In the startup range (0^+ to 40% power), P/F varies between 0
59 and 1.0. In this range, the bowing reactivity feedback is nonlinear, varying from positive to negative with increasing power/flow ratio as described in Sections 4.2.2.4.1.8.3, 4.3.2.3.4 and 4.3.2.3.5. Its change with power/flow ratio is sufficiently significant so that its impact on reactor stability must be established. The stability criterion which has been applied in this analysis is the following:

> The system is stable if the system output is bounded for a given bounded input, the system is unstable if the system output is unbounded for a given bounded input (Ref. 18). The system is stable in the practical sense if the system output is bounded within acceptable parameter levels (Ref. 19).

59 Figure 4.2-92C illustrates bowing reactivity responses which were predicted using nominal data and with maximum analytical uncertainties applied. The maximum positive bowing reactivity feedback effect is obtained when a positive reactivity perturbation is introduced at an operating point corresponding to the low end of the positive slope portion of the bowing reactivity vs. power-to-flow ratio curve. To this end, bowing reactivity functions which closely approximated the results in Figures 4.2-92B and 4.2-92C were input into the FORE-2M reactor kinetics
59 51 and feedback model.

59

Figures 4.3-46 through 49 illustrate key reactor responses during an inherent response transient (no control or PPS action) ini-
59 tiated at a reactor startup operating point (9% power; 40% flow) at which the reactivity insertion due to bowing would be maximum. The transient was initiated by a +2¢ step reactivity perturbation.

The responses illustrate that all parameters rise initially due to the dominance of positive bowing reactivity at low power-to-flow ratios. However, when the bowing reactivity coefficient becomes negative at higher power-to-flow/ratios, all parameter responses change slowly and approach a new stable equilibrium state. The final values of the parameters are shown in Table 4.3-34 together with acceptability limits.

It is concluded that if the limits for acceptability (Table 4.3-34) are selected so as to remain below reactor parameter severity levels associated with a major incident (Table 15.1.2-1), and parameter responses remain below acceptability limits, the reactor is stable in the practical sense and inherent reactor protection shall have been demonstrated.

Additional reactor stability (inherent response) transients have been evaluated which were initiated at other operating states (i.e., reactor flow, reactor power, etc.). All of these indicate the response characteristics typically exhibited in Figures 4.3-46 through 4.3-49 and were bounded by the acceptability levels of
51 Table 4.3-34.

As a result of these studies, it is concluded that the reactor is stable given the bowing reactivity characteristics exhibited in
56 Figure 4.2-92a since all transient responses are bounded for a bounded input perturbation. Furthermore, the reactor is stable in the practical sense since the maximum values of key reactor variables are below levels which are considered acceptable for the reactor when responding to its inherent feedbacks. Therefore, reactor inherent protection is demonstrated and Criterion 9 is satisfied.

4.3.2.9 Vessel Irradiation

The spatial and energy dependent neutron flux distributions are utilized in obtaining the irradiated characteristics of the reactor structural materials and components. One application of this flux data is in determining the total and fast fluence received by both replaceable and non-replaceable reactor components. The neutron fluence must be limited so that the end-of-life ductility for structural materials exceeds the specified minimum requirements.

Assembly-by-assembly radial neutron flux distributions (assembly-average in the central 36-inch active core height) are given in Figures 4.3-50 and 51 for core conditions reflecting the beginning of cycle one with the six row 7 corner primary control rods partly inserted and with fresh fuel and clean blankets, and for the end of cycle four conditions with all control rods fully withdrawn and with plutonium burned out of the fuel and built up in the blankets. Values are shown in Figures 4.3-50 and 51 for both the total neutron flux and for the fraction of the flux with an energy greater than 0.11 MeV. The latter reflects the relative spectral behavior throughout the core. The shift in the critical flux shape toward the center of the core with increasing burnup, and the spectral hardening in the blankets, is evident by comparing the fluxes in Figure 4.3-50 at the beginning-of-life with the end-of-life fluxes in Figure 4.3-51. Figures 4.3-52 and 53 show typical axial distributions of the total flux and the fast flux fraction in the core. These axial distributions are normalized to 1.0 over the central 36-inch active core height such that the product of the axial shape factors in Figures 4.3-52 and 53 with the fluxes in Figures 4.3-50 and 51 results in the three-dimensional flux distribution throughout the central core and blankets. The total flux in Figure 4.3-52 exhibits the typical bell-shaped axial distribution. Figure 4.3-53 indicates that the neutron energy spectrum (as measured by the fraction of the flux with an energy greater than 0.11 MeV) is relatively flat throughout the central core and degrades rapidly through the axial blankets. These radial and axial flux and spectrum distributions were obtained from two-dimensional, 21-
51 group diffusion calculations in hexagonal-planar and RZ geometry, respectively.

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TABLE 4.3-33

ZPPR CRITICAL EIGENVALUE (k_{eff}) PREDICTED BY CRBRP DESIGN METHOD AND DATA

	Homogeneous: ZPPR4				Heterogeneous: ZPPR 7 & 8		
	Measured	Calculated	C/E		Measured	Calculated	C/E
ZPPR-4/1	1.00080	0.99527	0.9945	ZPPR-7A	1.00028	0.99019	0.9899
ZPPR-4/2	1.00065	1.00049	0.9998	ZPPR-7B	1.00064	0.98924	0.9886
ZPPR-4/3	1.00088	0.99916	0.9983	ZPPR-7C	1.00002	0.99089	0.9909
ZPPR-4/4	1.00083	0.99390	0.9931	ZPPR-7D	1.00001	0.99348	0.9935
			Mean C/E = 0.9964	ZPPR-7F	1.00058	0.98873	0.9882
			1σ = ± 0.0031	ZPPR-7G	1.00053	0.98858	0.9881
				ZPPR-8F	1.00062	0.99156	0.9901
							Mean C/E = 0.9900
							1σ = ± 0.0019

TABLE 4.3-34

MAXIMUM PARAMETER VALUES DURING A REACTOR INHERENT RESPONSE
59 TRANSIENT AT A REACTOR STARTUP OPERATING POINT (9% POWER,
40% FLOW)

Parameter	Maximum Value	Acceptability Limits
Reactor Power	39%	
Maximum Fuel Temperature	1597°F	Less than fuel melting temperature
Maximum Cladding Temperature	984°F	Less than cladding melting temperature
59 Maximum Coolant Temperature	984°F	Less than sodium boiling temperature

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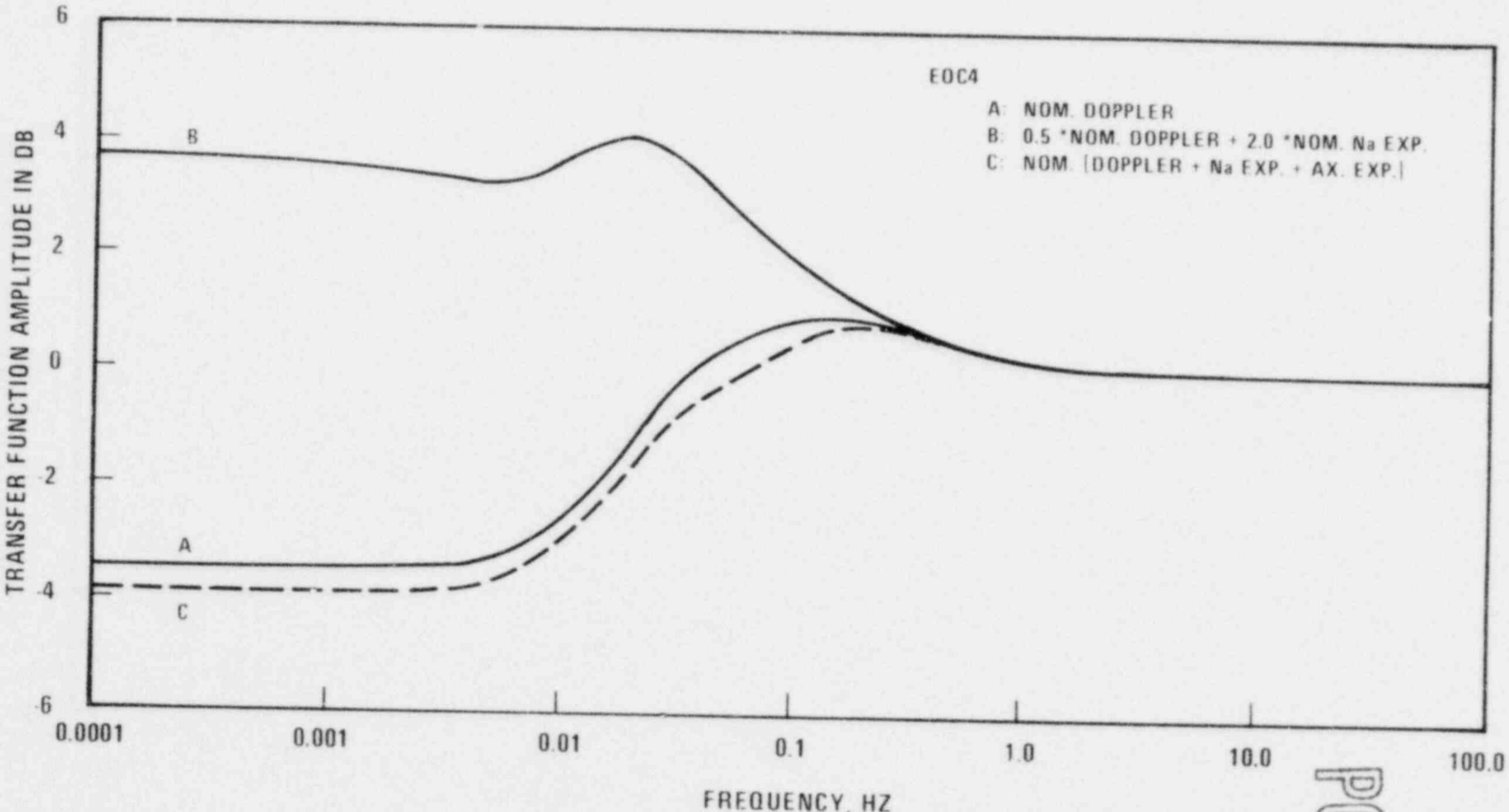

Figure 4.3-44. Transfer Functions for Various Feedback Reactivities (Note Expanded Scale)

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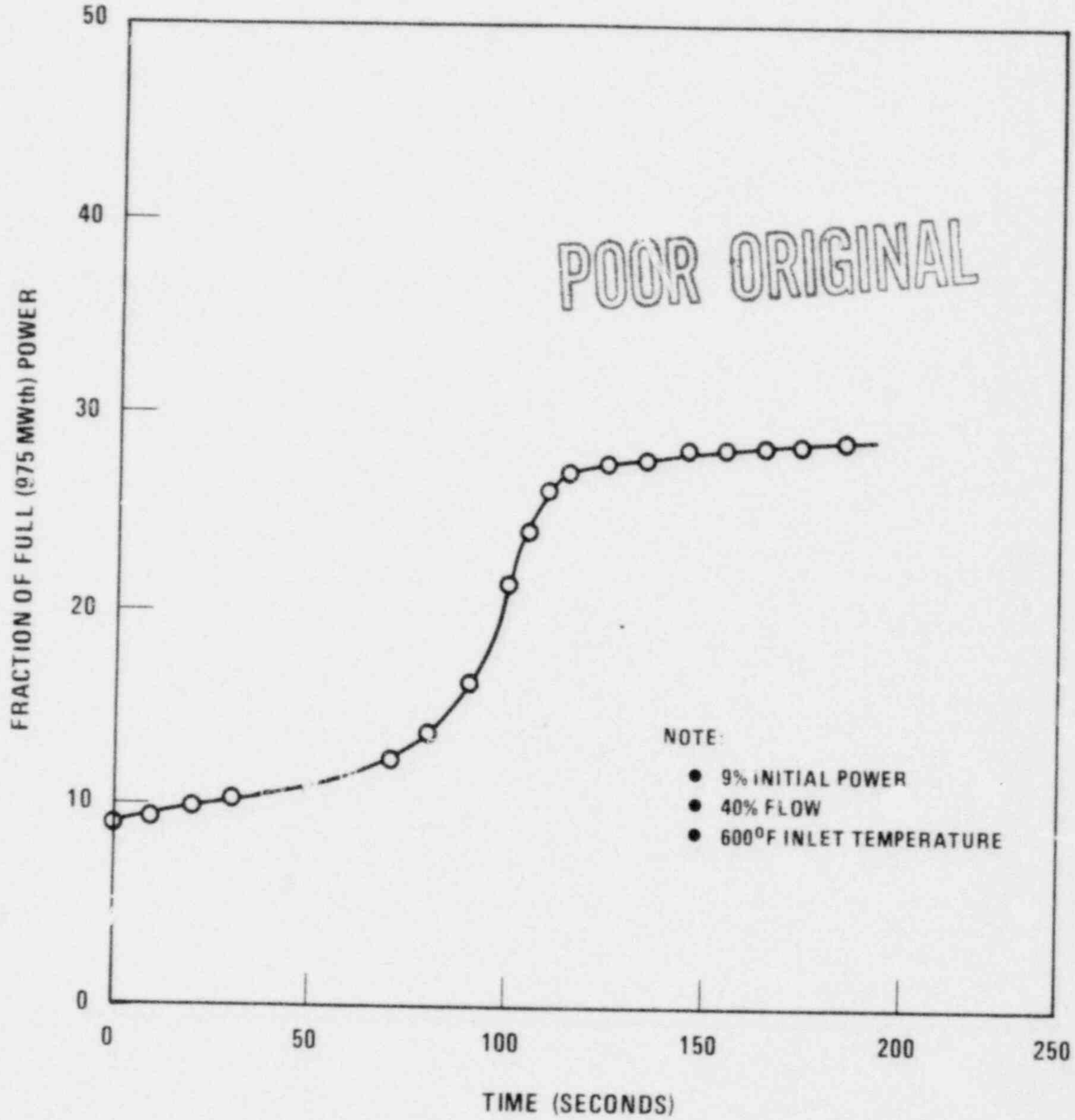


Figure 4.3-46. Reactor Power Response to Inherent Reactivity Feedback Following a 2¢ Step Insertion

4655-8

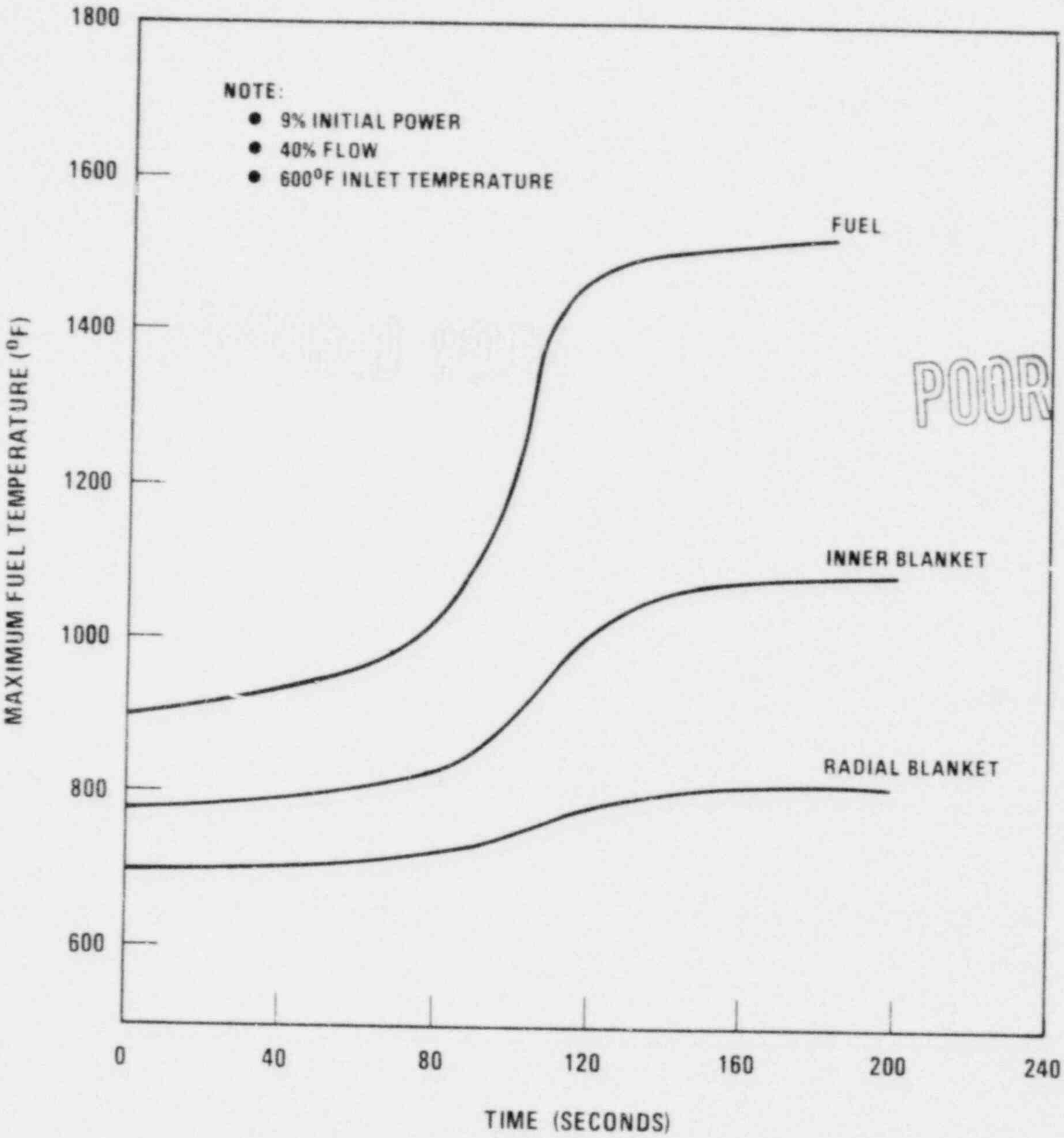


Figure 4.3-47. Base Case Maximum Fuel Temperature Response to Inherent Reactivity Feedbacks Following a 2¢ Step Insertion

4655-7

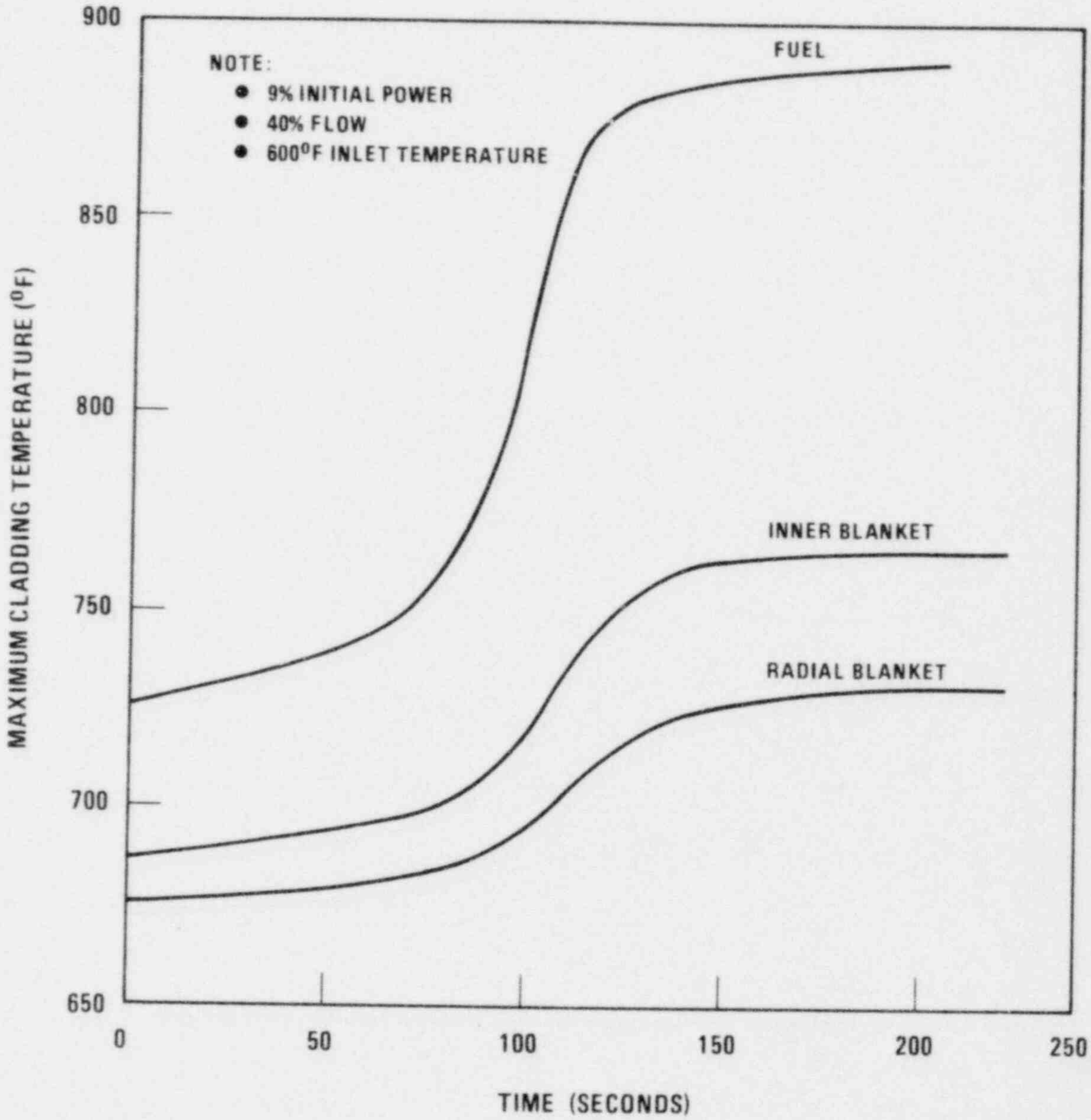


Figure 4.3-48. Base Case Maximum Cladding Temperature Response to Inherent Reactivity Feedbacks Following a 2¢ Step Insertion

4655-6

POOR ORIGINAL

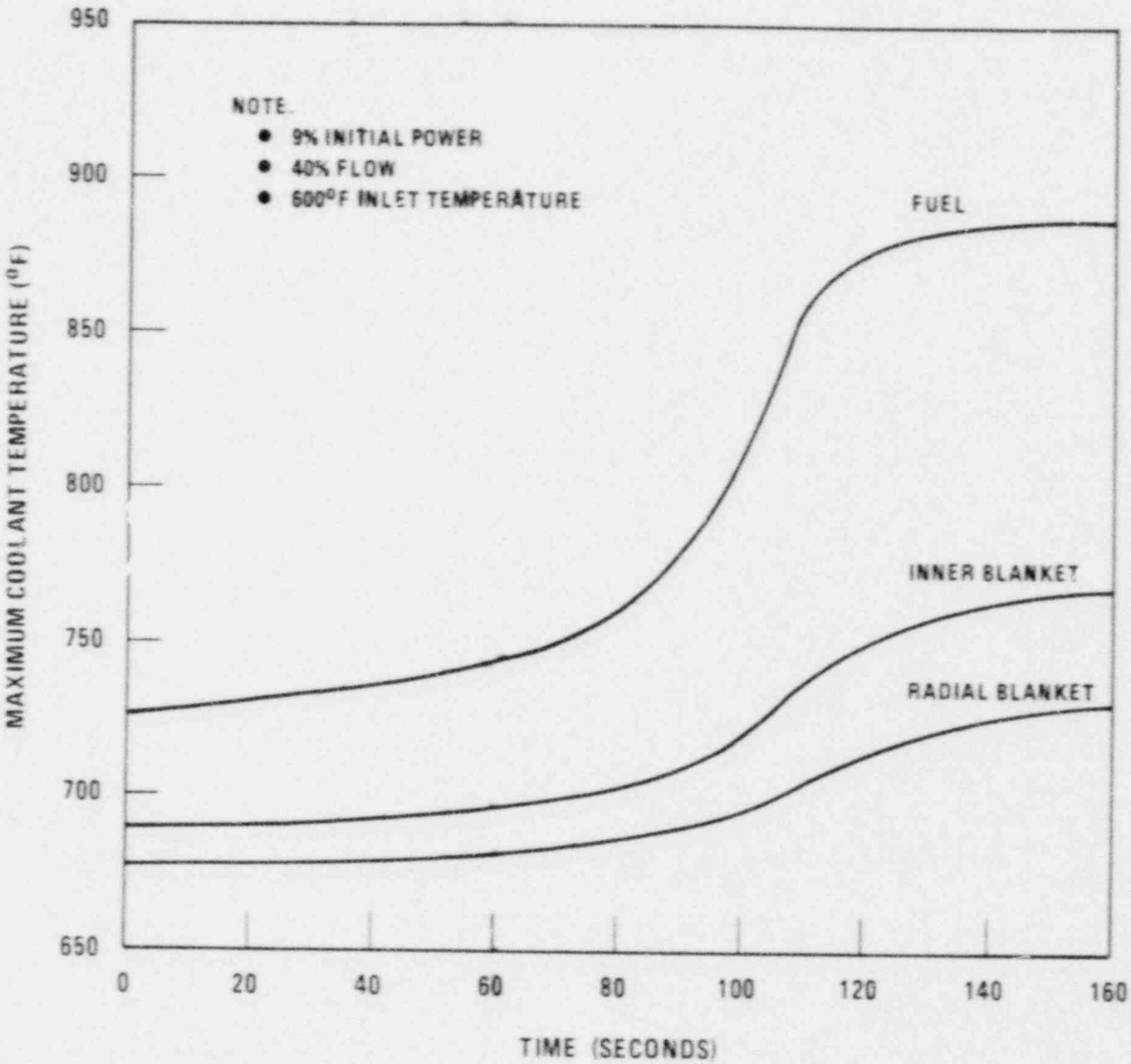
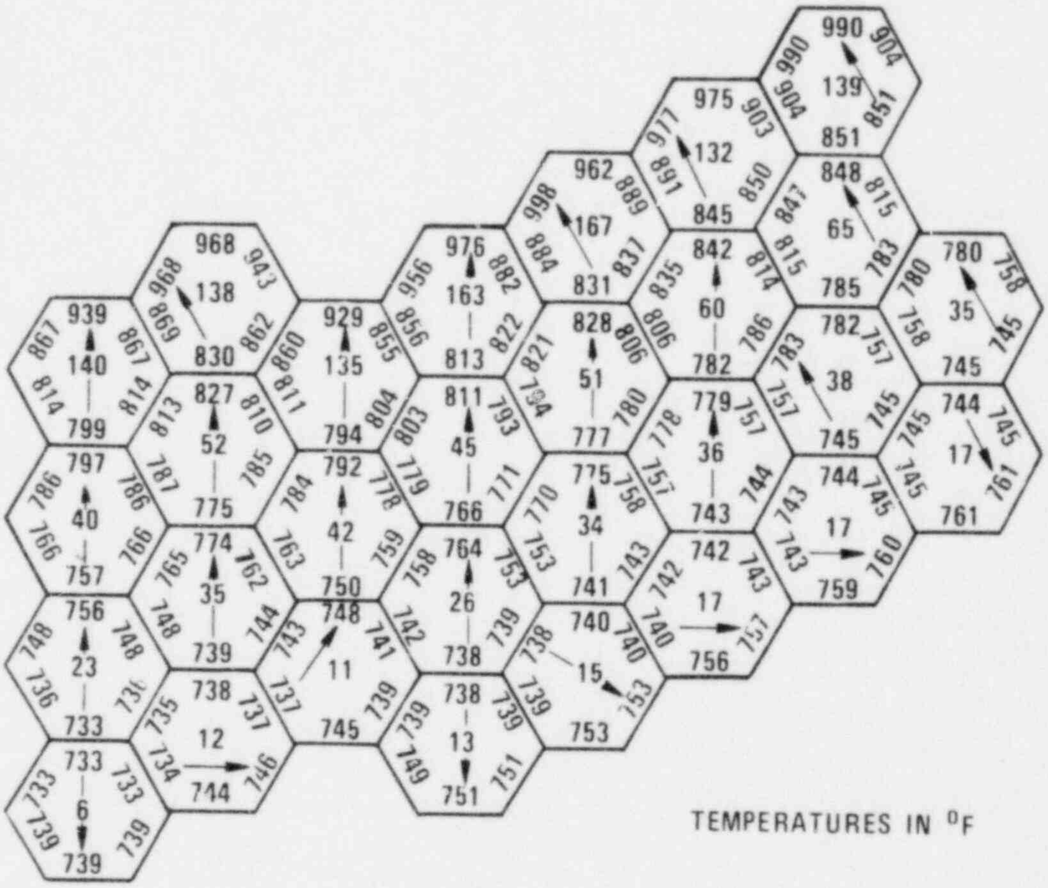


Figure 4.3-49. Base Case Maximum Coolant Temperature Response to Inherent Reactivity Feedbacks Following a 2¢ Step Insertion

4655-5

CONDITIONS: PLANT EXPECTED OPERATING CONDITIONS
EOC5, 714°F INLET TEMPERATURE
+2σ UNCERTAINTIES ON RADIAL BLANKET BOUNDARY TEMPERATURES
−2σ UNCERTAINTIES ON REMOVABLE RADIAL SHIELD TEMPERATURES

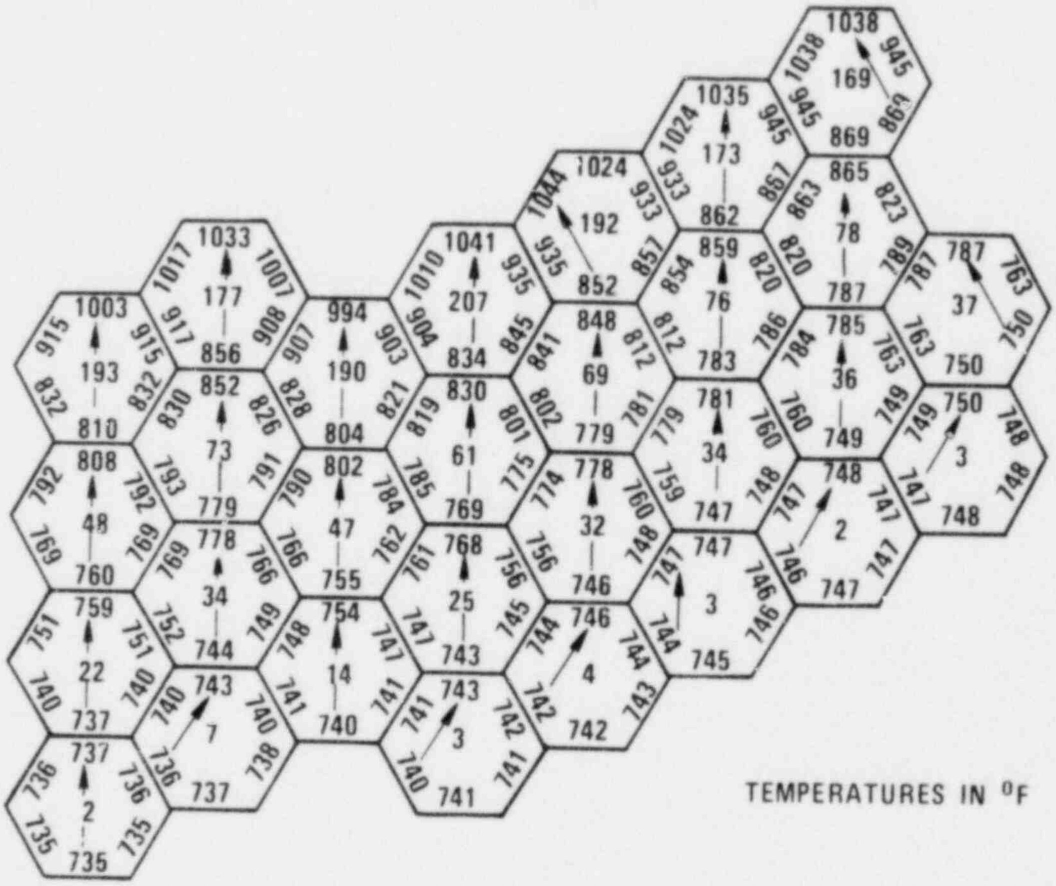


POOR ORIGINAL

Figure 4.4-61 Removable Radial Shield Duct Midwall Temperatures at Above Core Load Pad (−2σ RRS Uncertainties)

4482-3

CONDITIONS: PLANT EXPECTED OPERATING CONDITIONS
EOC5, 714°F INLET TEMPERATURE
+2σ UNCERTAINTIES ON RADIAL BLANKET BOUNDARY TEMPERATURES
−2σ UNCERTAINTIES ON REMOVABLE RADIAL SHIELD TEMPERATURES



POOR ORIGINAL

Figure 4.4-62 Removable Radial Shield Duct Midwall Temperatures at Top Core Load Pad (−2σ RRS Uncertainties)

4482-1

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TABLE 5.1-1

HEAT TRANSPORT SYSTEM THERMAL HYDRAULIC DESIGN CONDITIONS

	Parameter	Thermal Hydraulic Design Value	
	Thermal Power (MWt)	975	
	Primary System		
	Hot leg temperature (°F)	995	
	Cold leg temperature (°F)	730	
45	Flow (per loop) million lb/hr	13.8	
	Pump Flow (gpm @ 995°F)	33,700	
45 36	Pump Head (Ft Na @ Design Flow and Temp)	450	33
	Intermediate System		
45 40	Hot leg temperature (°F)	936	
40	Cold leg temperature (°F)	651	
59	IHTS ΔT (°F)	285	
45 40	Flow (per loop) million lb/hr	12.8	
	Pump Flow gpm @ 651°F	29,500	
45	Pump Head (Ft Na @ Design Flow and Temp)	330	33

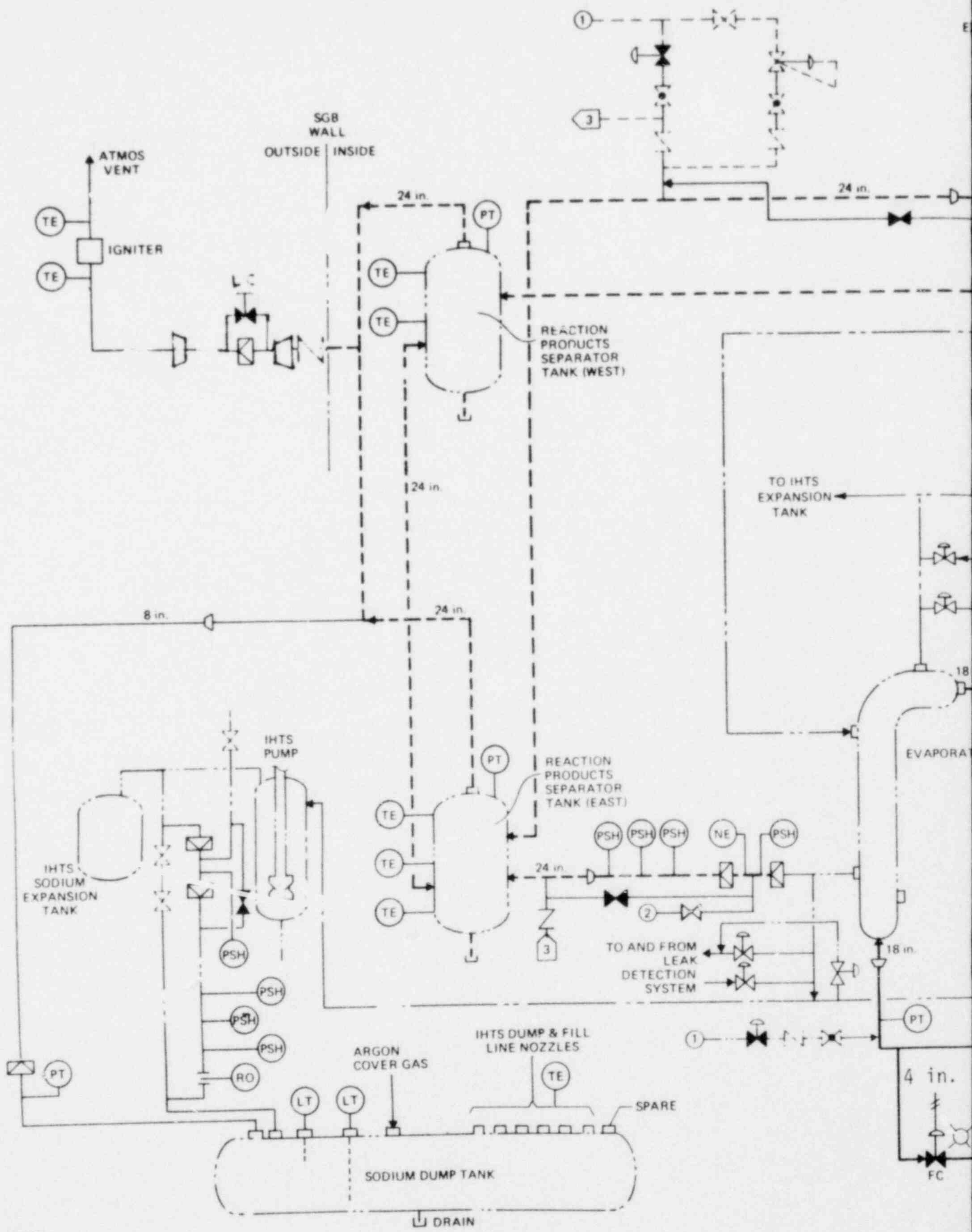
TABLE 5.1-2

PHTS VOLUMES AND VOLUME CHANGES*

Component	Sodium Containment Volume (ft)^3 at Room Temperature	Sodium Containment Volume at Thermal/ Hydraulic Design Conditions	
45			
Primary System			
Primary System			
45 o R.V. to Pump	725, 717, 728	746, 738, 749	
o Pump to IHX	235	242	
o IHX to R.V.	426, 435, 448	434, 444, 457	
43 IHX (Shell Side)	1348	1381	
Pump - Tank, Suction, and Discharge Nozzle at Normal Operating Level	367	378	33
Check Valve	88	90	
Total Volume (Per Loop)	3189, 3190, 3214	3271, 3273, 3297	
43 Three Loop Total	9593	9841	
Reactor Vessel	13629	13961	
45 44 43 Total Primary Volume	23222	23802	

56 Note: Net sodium overflow volume from a system fill temperature of 400°F to the thermal/hydraulic operating condition is 1439 FT^3 when corrected to an assumed 900°F in the overflow tank.

* Where three volumes are given, they refer to loops #1, #2 and #3 respectively.



POOR ORIGINAL

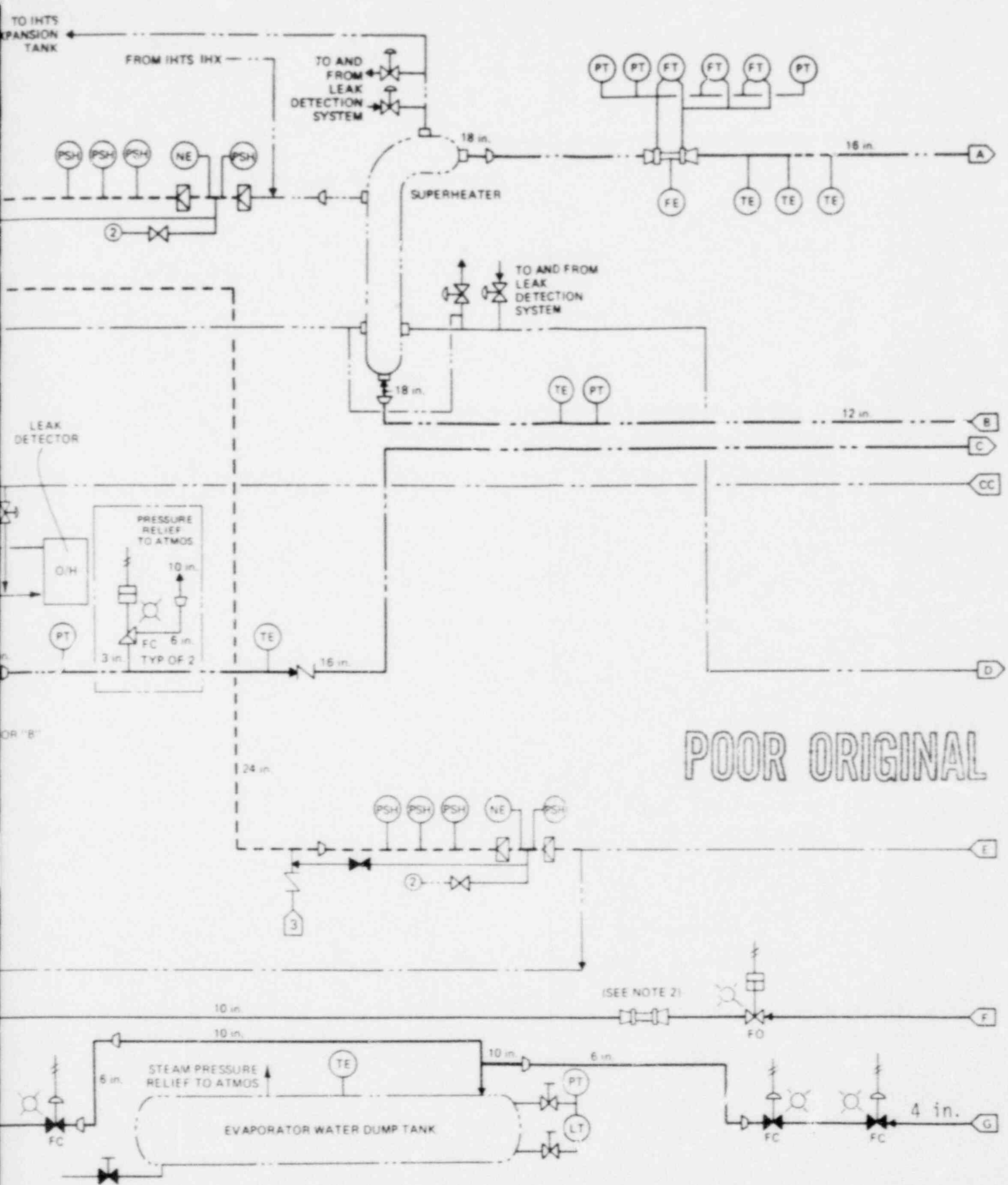
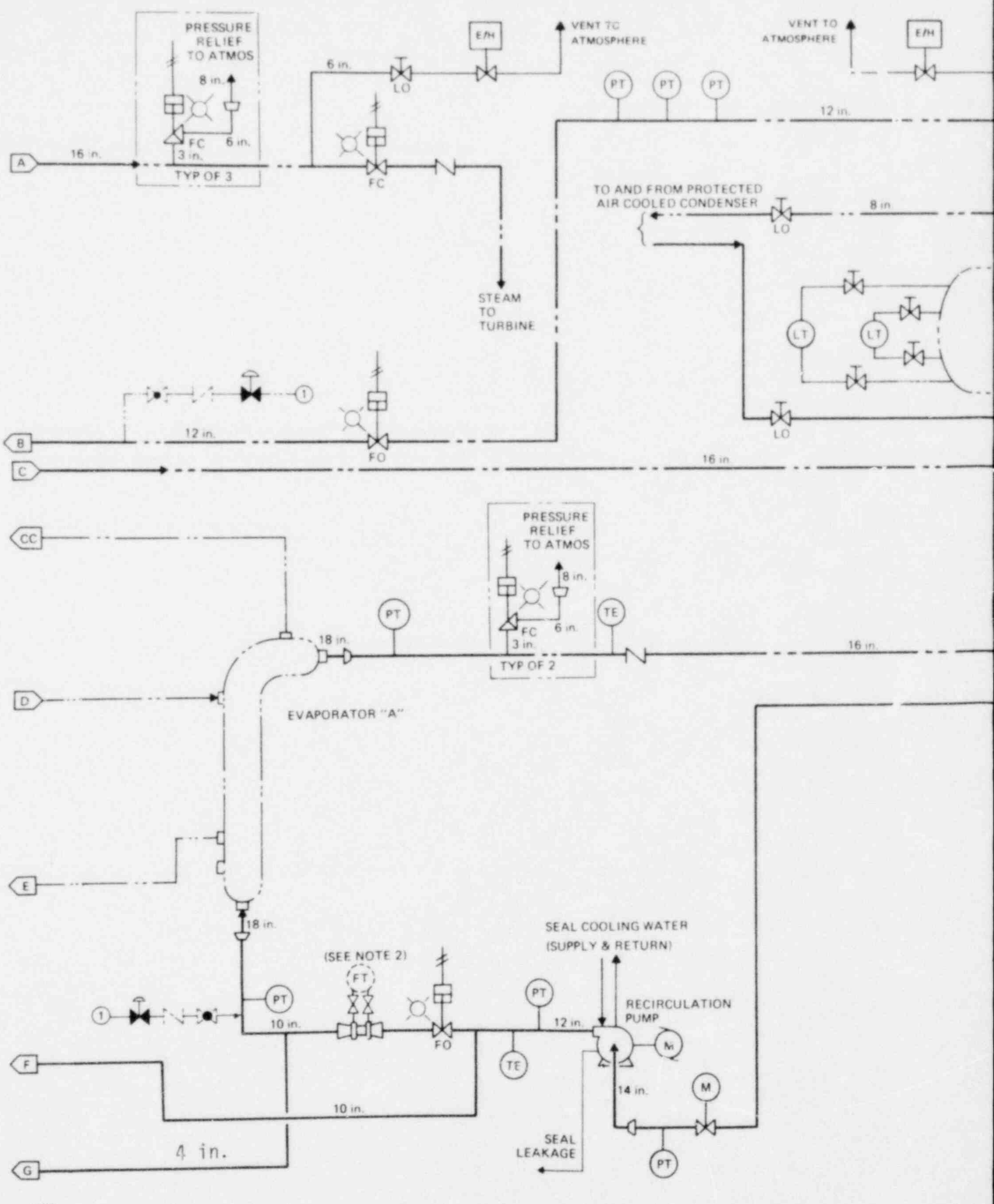
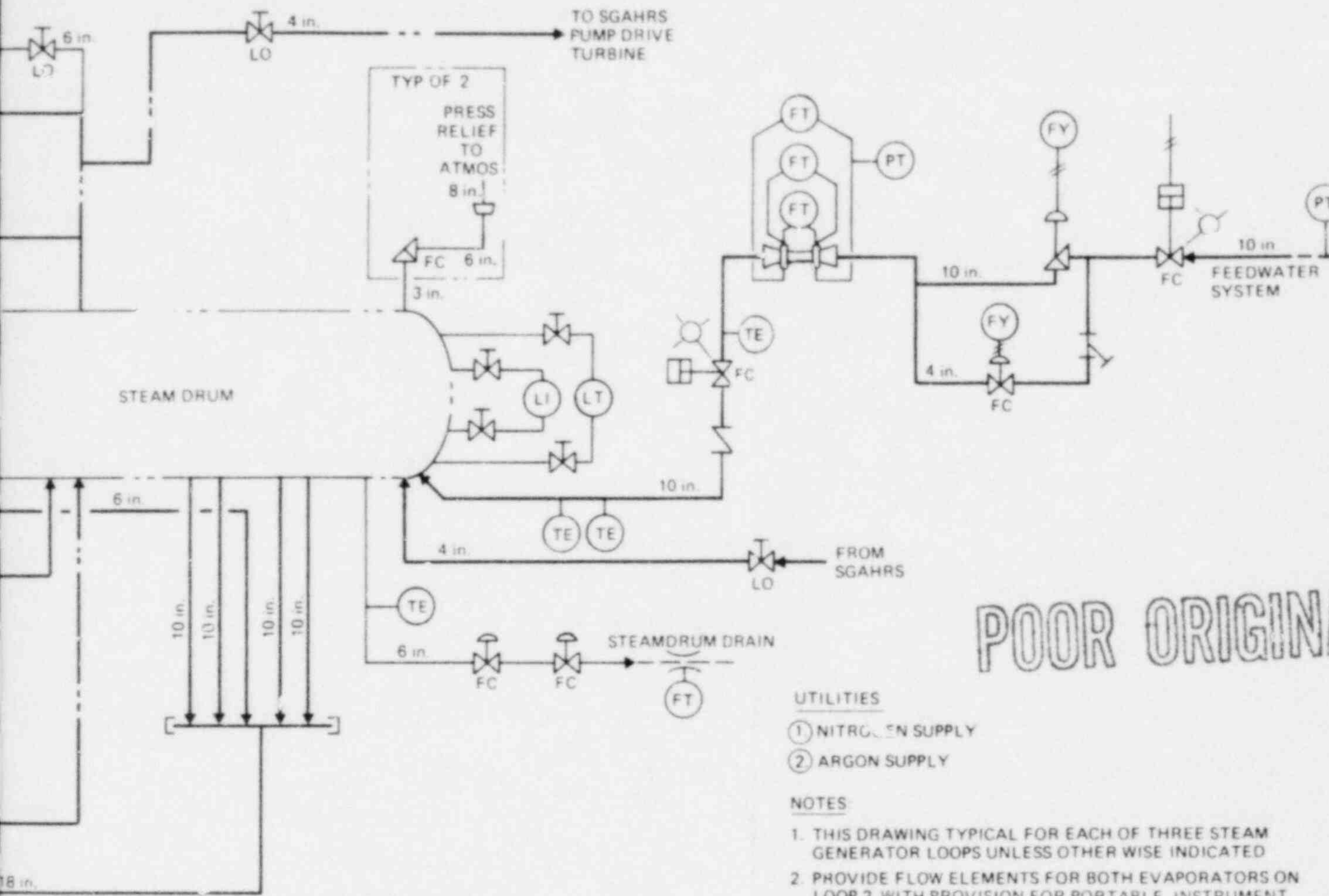


Figure 5.1-4 Steam Generator Schematic Flow Diagram (Sheet 1 of 2)



POOR ORIGINAL



UTILITIES

(1) NITROGEN SUPPLY

(2) ARGON SUPPLY

NOTES

1. THIS DRAWING TYPICAL FOR EACH OF THREE STEAM GENERATOR LOOPS UNLESS OTHER WISE INDICATED
2. PROVIDE FLOW ELEMENTS FOR BOTH EVAPORATORS ON LOOP 2 WITH PROVISION FOR PORTABLE INSTRUMENT HOOKUP

LEGEND

WATER
STEAM
SODIUM (REFERENCE ONLY)
SODIUM WATER REACTION PRODUCTS
INSTRUMENT ELECTRIC LINE
INSTRUMENT AIR LINE
OTHER
NORMALLY CLOSED VALVE
SGAHRS – STEAM GENERATOR AUXILIARY HEAT REMOVAL SYSTEM
SWRPRS – SODIUM WATER REACTION PRESSURE RELIEF SYSTEM
IHTS – INTERMEDIATE HEAT TRANSPORT SYSTEM
FC = FAIL CLOSE
FO = FAIL OPEN
LO = LOCK OPEN

Figure 5.1-4 Steam Generator Schematic Flow Diagram (Sheet 2 of 2)

5.2.2.1 Reactor Vessel and Support

The reactor vessel and support will be constructed mainly of
austenitic stainless and low alloy steels, and consists of six basic
sections: the support ring, the vessel flange, the barrel, the core
support forging and cone, the inlet plenum, and the vessel thermal liner.
59 The support ring is an SA 508 Class 2 steel forgin welded to the vessel
17 flange. A box ring type of reactor vessel support interfaces with the
vessel support ring and the reactor cavity support ledge. Holddown bolts
pass through holes in the vessel support ring, the reactor vessel support
and the support ledge clamping the three together. The vessel support
is a ring structure with a box type cross section. The vertical sides
of the box are Inconel 600 to limit the heat flow from the reactor
vessel. The top and bottom plates of the box cross-section are
59 58 SA 543 Class 2. The bolts are SA 193 Type B7 with 3.50-8UN threads. The
ring supports the reactor vessel and internals and closure head. The vessel
flange is a second SA 508 Class 2 steel ring forging welded to an Inconel
600 transition section. The latter is, in turn, welded to the barrel.
Radiation shielding in the form of a boron carbide collar surrounding the
vessel near the flange is provided in the annulus between the reactor
vessel and the vessel support ledge. The barrel comprises the upper
cylindrical portion of the vessel and has an inside diameter of 243 in.
17 with a minimum wall thickness of 2.38 in. The lower end of the barrel
is joined to the core support forging and cone, which provide support
for the core support structure. The overall height of the reactor vessel
and support is nominally 704 in. (58 ft. 8 in.). The inlet plenum is
designed for 200 psig at 775°F and -15 psig at 600°F, the stainless steel 11
portion of the outlet plenum is designed for 15 psig plus head of sodium
17 at 900°F and -15 psig at 600°F.

Coolant enters the reactor vessel through three 24-inch nozzles located 120° apart in the inlet plenum below the core support structure. Core effluent and bypass flow are mixed in the outlet plenum region above the core, and the

b. After the second cycle or repair in welds that are heat treated after each repair cycle.

Purchaser approval is required for repair of crater cracks restricted to the crater of any weld pass if the third repair cycle results are not acceptable.

5.3.1.5 Leak Detection Requirement

The PHTS Leak Detection Subsystem, part of the Sodium/Gas Leak Detection System described in Section 7.5.5.1, will provide indication and location information to the operator in the event of a sodium leak from the primary sodium coolant boundary, in a timely manner in order that action may be taken before a critical size crack in the primary boundary develops. (A critical size crack is a crack that would bulge open due to operating stresses. See Section 5.3.3.6.)

The detection system sensitivity requirements are discussed in Section 7.5.5.1.

5.3.1.6 Instrumentation Requirements

The primary system is provided with an instrumentation system which monitors the process variables within the PHTS and which provides signals for safety action and operational information. The measured variables and instrumentation provided are discussed in Section 7.5.2.

5.3.2 Design Description

5.3.2.1 Design Methods and Procedures

5.3.2.1.1 Identification of Active and Passive Components which Inhibit Leaks

In the primary heat transport system, the only active component which is considered a part of the PHTS is the primary pump (see Table 5.3-10 for a list of pumps and valves). In the event of pipe leaks, the primary pumps are automatically reduced to pony motor flow following manual reactor shutdown or
59 40 reactor scram due to low reactor vessel sodium level.

In the unlikely event of a primary pipe or component boundary failure, the PHTS has been designed to limit the loss of reactor coolant and assure that for any boundary failure, continued reactor cooling is provided. The PHTS design features which limit loss of coolant and assure reactor cooling are the
56 combined use of elevated piping, use of a guard vessel around major equipment and a five foot pony motor shutoff head. The PHTS guard vessels have been designed such that the tops of the guard vessels are at an elevation which is approximately 9 feet above the tops of the reactor vessel discharge nozzles.
56 This level is based on the combination of the pony motor shut-off head of 5 feet and the minimum safe reactor vessel level which is two feet above the top
40 of the reactor discharge nozzle, plus an additional two feet to accommodate sodium shrinkage and hydraulic uncertainties.

The volume of the guard vessel and the volume of sodium above the minimum safe level (MSL) in the reactor vessel and pump tank have been sized to assure that the guard vessel volume will be less than or equal to the volume loss from the reactor and pump tanks for any leak condition assuming contraction and cooldown after the leak assuming no reactor vessel makeup. At minimum operating sodium level the volume of sodium above MSL in the reactor vessel and pump tanks is 4430 ft^3. The net volume of each PHTS guard vessel is 2700 ft^3. The net
59 volume of the reactor guard vessel is 3100 ft^3.

Continued reactor cooling is provided in the unlikely event of a pipe failure by the PHTS elevated piping arrangement. All PHTS piping is routed at an elevation above the tops of the PHTS guard vessels thereby limiting the loss of coolant in the unlikely event of a pipe failure.

The combination of guard vessel elevation, guard vessel volume, reactor vessel
59 and pump tank sodium inventory above the minimum safe level, pony motor shutdown head and elevated piping assures a limited loss of reactor coolant and continued reactor cooling capability.

Within the PHTS, there are two general types of failures of the pressure
59 containing boundary. They are (1) failures which occur within a guard vessel and (2) failures in elevated piping outside of the PHTS guard vessels.

59

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59 Leak of PHTS Outside Guard Vessels

For a leak in the elevated pipework between the reactor vessel and the pump suction, leaks will stop when the level of sodium in the reactor vessel drops about one foot below the level of the leak, because the syphon effect is broken. At most, in postulated cases where the check valve fails to close, reactor sodium level will drop no further than the frictional head across the reactor with pumps operating at pony motor speed. It should be noted that independent of the check valve the level in the reactor vessel will not fall below the minimum safe level. The unaffected loops continue to operate because, with the low reactor outlet nozzles, the syphon effect in these loops is unimpaired. The pony motor is required only to overcome the frictional losses in the unaffected loops, so they continue to provide decay heat cooling.

For a leak in the elevated piping between pump discharge and reactor vessel inlet, sodium will flow out until the level of sodium in the reactor vessel falls to a level where the head developed by the pump at pony motor speed is insufficient to raise fluid to the height of the break. At this point, the
59 sodium level in the reactor will be approximately one foot above the minimum safe level. If the pony motor in a failed loop is stopped, the loss of sodium will be reduced to that generated by reverse flow from the active pumps. The rate of sodium loss from the defective loops will be restricted by check valve action. However, plant safety is independent of operation of the check valves. For small leaks, check valve action will produce little mitigation of the leak.

The potential for cell atmosphere ingress into the coolant system in the event of a leak exists at any point in the system that operates under a negative pressure relative to atmospheric. Therefore, the potential for cell atmosphere ingress into the coolant system depends on whether the loop is shut down, the pump is on pony motor operation, or the pump is on main motor operation. Each of these three loop conditions produce a unique pressure profile around the 25
loop, and therefore, must be addressed separately.

Loop Shutdown - Pumps Idle

When the loop is shut down, the potential for gas getting into the system
exists in three different places: the elevated section of the 36" hot leg
piping, the upper region of the IHX (above the upper tubesheets), and in the
elevated section of the cold leg piping including the check valve. These three
areas and the connecting vent lines are the only points in the system that are
above the normal sodium level in the reactor. The high point vent lines
59 contain freeze seals backed up by locked closed valves in the Argon cover gas
system.

Loop Operation at Pony Motor Flow

Operation at pony motor flow in the loop eliminates the negative pressure in
the elevated section of the primary cold leg piping (including the check valve)
and therefore, eliminates the potential for cell atmosphere ingress. Negative
pressure does exist in the IHX and the elevated section of the 36" hot leg
piping, and therefore, the potential for gas ingress does exist. The potential
for cell atmosphere ingress into the primary system in the IHX exists at two
separate places; the other annulus to which the IHX to pump tank vent line is
attached, and at the top of the IHX near the bellows seal. Gas ingress near
the bellows seal would be self limiting because any gas in-leakage in this area
would have to travel the entire length of the tube bundle through the annulus
between the intermediate (tube side) downcomer and the inner shroud of the tube
bundle (See Figure 5.3-15) before it could enter the main sodium stream. Gas
entering the main stream from this point would require a pressure of
approximately 10 psi higher than the normal cell pressure. Therefore, the only
part of the IHX that has a potential for gas inleakage is the outer annulus of
59 the IHX to which the IHX vent return line is attached. However, this region of
the IHX is not in the main stream and in-leakage would be self limiting. Any
gas accumulation would be forced to the gas space of the pump upon start-up of
the pumps on main motors.

Loop Operation with Pumps Driven by Main Motors

Operation of the pumps on main motors eliminates negative pressure in all of
the systems downstream of the pump and leaves only the elevated section of the
36" primary hot leg piping at a negative pressure. At full flow in the loop,
the system pressure is negative from the second elbow from the reactor vessel
to the second elbow before the pump (approximately 73 ft. in each loop). This
section of piping has only one penetration (high point vent connection). It
contains no thermowells, connections to pressure transducers or other
penetrations to the coolant boundary. The high point vent contains a freeze
seal backed up by redundant "locked closed" valves. 25

Discussions of protective system action and delay times can be found in Section 7.2.

5.3.2.1.2 Design of Active Pumps and Valves

The applicable ASME Boiler and Pressure Vessel Code, Addenda, Code Cases, and
59 RDT Standards were used as the design basis for the Primary

Heat Transport System (PHTS) sodium pumps. Detailed analyses are being made to
59 40 verify structural performance within the design basis limitations.

The primary pump normally operates in a temperature range between 400°F to 1015°F and a pressure range between 0 to 200 psi. Since the operating temperatures exceed 800°F (for austenitic steels), materials time-dependent behavior must be considered and the high temperature design criteria must be invoked for structural analysis. The analytical criteria to be employed are the Class 1 requirements of the 1974 Edition of the ASME Boiler and Pressure Vessel Code; Section III, Nuclear Power Plant Components with Addenda through winter of 1974, as supplemented by RDT Standard E15-2NB-T with Amendments and addenda through June, 1975. The elevated temperature Code Cases (i.e. 1592
59 37 through 1596) and RDT Standard F9-4T also apply.

59

59 The pump stress analysis is being performed using verified and documented
40 computer programs as well as standard hand calculations. Where possible, the
design of the pumps is such as to keep the membrane stresses in the elastic
range; where this is not possible, inelastic analysis techniques is being
59 used. Vibration of the pump components and the pump's response to seismic
49 excitation are being determined by analysis. Amplitude and frequency limits
imposed are being limited to accumulated fatigue damage and consider proper
function of the pump parts. RDT Pump Standards require that the first rotor
bending natural frequency must be at least 25 percent higher than the maximum
pump shaft speed. This requirement is being augmented with amplitude
59 restrictions on other pump components as required to produce a smooth-running
pump.

40 Pump Operability

The PHTS pump manufacturer is required to assure operability under accident
conditions and during seismic events in accordance with Reference 12, PSAR
44 Section 1.6.

Prototype pump testing includes acceptance testing in water at the pump
suppliers' facility and performance testing in sodium at the Sodium Pump Test
Facility. The prototype pump was tested in water at approximately 97% of full
flow design conditions (drive motor is sized for sodium service; 97% is the
technically approved drive system operating limit of capability for pumping
water) to verify hydraulic and mechanical performance. Sodium performance
testing is being done at full rated flow, at expected operating head and
temperature. Testing in sodium includes mapping of head and flow for both
maximum and minimum plant loop impedences. Testing of the pump's performance
when subjected to fluid borne temperature transients includes the plant
predicted upset and emergency transients up to the capability of the facility.
Operability of the pump during and after the emergency and faulted plant
conditions is being verified by analysis, since comprehensive accident and
seismic qualification testing is not possible due to test facility
59 43 limitations.

43 10

A listing of the analytical methods anticipated to be used in the evaluation of stresses is presented in Section 5.3.3.1.5. 33

5.3.2.1.3 Surveillance and Inservice Inspection

Introduction

The Inservice Inspection Program consists of three types of inspection which are listed in decreasing frequency occurrence.

Visual condition inspection to be conducted during periods of plant shutdown such as for refueling.

Weld inspections to be conducted during periods of prolonged plant shutdown such as for maintenance.

Metallurgical and component internals inspections to be conducted when components are removed for maintenance.

Inservice inspection will be planned and conducted according to the 6
59 appropriate requirements of Appendix G. 33

Visual Condition Inspections

Visual inspections will consist of remote visual viewing in the PHTS cells and pipeways (between the PHTS cell and reactor cavity), and visual examination of the intermediate (excontainment) system. Remote viewing capabilities will include the following:

> Ability to view the primary coolant boundary (in-so-far as practical) in the PHTS cell and pipeway including capability to conduct visual examinations of the annulus between the pump or IHX and the respective pump or IHX guard vessels.

5.3.2.2 Material Properties

The coolant boundary sections of the primary heat transport system are made from unstabilized austenitic stainless steels, viz. Type 304 and Type 316 stainless steel. Depending on the application, minimum lower limits ranging from 0.04% to 0.055% will be placed on the carbon level. In all applications governed by the rules of ASME Code Case 1592, a minimum carbon content of at least 0.04% is specified in the material purchase specification. In certain applications (e.g., primary hot leg piping) where interstitial loss in sodium becomes appreciable, a minimum carbon content of 0.055% is specified in the material purchase specification to minimize the impact of interstitial transfer on design. Specific procedures for accounting for environmental effects on material properties (including interstitial transfer) are provided in an appendix to the Equipment Specifications. In order to minimize adverse effects due to too high a carbon level, an upper limit of 0.08% will be maintained. Various properries of the austenitic stainless steels are described below. 13

5.3.2.2.1 Short Term Tensile Properties

With respect to short term tensile properties it may be shown that both carbon and nitrogen levels have a strong influence and the following equations may be used to estimate the yield strength (y), the ultimate tensile strength (u), and the total elongation ( ) from the carbon and nitrogen contents C and N.

For solution treated Type 304 stainless steel

59 $$y = 89.952 + 181.167\,(C+N) - 0.148\,T\,(C+N) - 5.727\,T^{1/2} + 0.105\,T \quad (1)$$

standard deviation = 2.236

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Values of (C + N) can thus be estimated and these values substituted into the appropriate equations, 1 through 6, in 5.3.2.2.1 in order to determine mechanical property changes arising from interstitial transfer during sodium exposure. 17

5.3.2.3 Component Descriptions

The primary sodium pumps are free surface, single stage, vertically mounted, drawdown type centrifugal pumps driven by a variable speed 5000 hp squirrel cage induction motor. An auxiliary 75 hp pony motor, on each pump unit, provides low flow capability (<10%) for decay heat removal and other low power standby conditions. Variable pump speed is achieved by having the main drive motor supplied with variable frequency power from a fluid coupled MG set. Each
59 pump is designed to deliver 33,700 GPM of 995°F sodium at a 458 foot head. 33

The design envelope for the primary pump is shown in Figure 5.3-14. The pump tank incorporates a 36 inch side suction nozzle. Pump discharge is through a 24 inch nozzle.

Sodium flow enters the pump tank through the horizontal 36" diameter nozzle on the equator of a spherical tank. From the nozzle the flow splits and enters the inlet guide structure into the upward and downward facing impeller inlets. The flow leaving the impeller is joined and the combined flow passes through a
59 triple volute pump casing. The flow is then channeled through the 24"
horizontal exit nozzle on the pump tank equator at a location 90° clockwise 33
from the suction nozzle as viewed from above. A 20 inch diameter balancing pressure port is located on the high pressure side of the pump, 180 degrees away from the discharge opening. This feature is included to reduce side loads exerted by discharging fluid, thereby reducing any creep enhanced rachetting
49 57 and to prevent excessive loss of bearing clearance. Separation of the suction and discharge pressure in the pump tank is achieved by controlled clearances
59 between the impeller, pump casings and ducts within the pump tank sphere.

The impeller and impeller shaft assembly is supported in the pump casing by dual sodium lubricated hydrostatic bearings above and below the impeller. The bearings are supplied with sodium at near pump discharge pressure. The sodium
49 flows through the hydrostatic bearing pads and returns through the bearing clearances to the low pressure side of the pump. 33

Pump design is based upon sodium level control using a standpipe bubbler system
59 in which a continuous flow of argon gas is supplied to the pump cover gas space above the sodium free surface. During pump shutdown or low sodium flow conditions, the gas supply pressure is sufficient to depress the liquid sodium level to the standpipe nozzle elevation and gas will bubble up the standpipe to the reactor cover gas 33

system. During higher sodium flow rates, normal drawdown will almost uncover
41 the standpipe nozzle, purge gas will further lower the sodium level, and gas
will flow (bubble) into the nearly empty standpipe to the cover gas system.

49 The standpipe is essentially one leg of a manometer which automatically
balances the gas pressure in the pump tank. The standpipe-bubbler will
automatically perform the function of a pump cover gas vent and relief valve to
the cover gas system, and will not rely on the signal of a liquid level sensor
49 for level control. Level sensors are available for alarm.

The standpipe bubbler nozzle location is shown on Figure 5.3-14.

41 Since the standpipe bubbler is physically just a pipe connecting the pump tank
to the gas equalization line between the reactor and the overflow tank, there
are no malfunctions which could result in overpressuring the cover gas region
of the pump such that the sodium level in the pump would drop to a point which
59 would permit this gas to enter the hydraulics of the pump.

The vent line from the top of the IHX to the pump tank enters the pump tank at
an elevation about 30 inches below the standpipe bubbler nozzle. A trip in the
IHX vent return line prevents pump cover gas from flowing to the IHX should the
54 pump sodium level drop to the IHX vent return nozzle elevation. During full
52 power operation, 200 gpm of sodium flows from the IHX to the pump to vent any gas tending to accumulate at the top of the IHX through the vent line to the cover gas system of the pump. During rapid flow changes such as a scram, the flow from the IHX to the pump through this line decreases with the speed of the pump. Since the vent line connection to the pump is located above the hydraulics of the pump, any gas entering through this line will bubble up to the free surface rather than enter the main sodium pump. In the case of bubbler system, the standpipe is 6 inches in diameter connected to a 2-inch gas equalization line which is more than adequate to vent all of the gases which could inadvertently enter the primary pump tank. Therefore, there is no way that this gas could be bled to the core. Even if gas entering the pump should get into the pump hydraulics and into the main sodium piping, it would accumulate at the top of the IHX and be vented from the heat exchanger back to the pump gas system via the IHX vent line. Therefore, any gas introduced into the PHTS from the pump is precluded from entering the core of the reactor.

The gas blanket in the pump tank will serve as a damper on level swings - for 25
example during pump trip the sodium level will tend to rise from the 100% flow, drawn-down level to the normal system level. During this event the sodium fluid will rise to a level somewhat above the gas-bubbler nozzle working against the entrapped and compressing gas in the pump upper tank ullage. As the gas supply continues to enter the pump tank, the liquid level will stop its rise, then fall until the bubbling action through the standpipe is re-initiated. 33

Both PHTS and IHTS pumps require a shaft seal to effect a zero leak seal from
59 cover gas to atmosphere. This shaft seal, shown schematically in Figure
5.3-14A, is an oil lubricated, double rubbing face seal. The seal has a shaft
49 driven internal oil circulator and an integral air to oil heat exchanger, with
oil supply to make up for oil leakage past the rubbing faces. Oil leakage
49 from the seal assembly into the sodium coolant is prevented by two barriers.
The first barrier is an oil dam approximately 1.2 inches above the lower face
49 seal. The normal leakage from the lower face seal is diverted by this oil dam
into the oil leakage drain passage into the lower seal leakage collection
reservoir. A second barrier is the collar above the static maintenance seal
located just above the purge labyrinth. This collar extends beyond and over
the labyrinth, thereby shunting any oil to a drain plenum. For oil to
59 penetrate into the sodium, three things must happen:

- Failure of the oil dam

- 59 Failure of the collar to divert oil

- Overflow of the plenum drainage over the static maintenance seal lip

A positive pressure is maintained in the shaft seal oil at all times by means
49 of a pressurized gravity tank which will be pressurized as required by the loop
operating pressure. The oil feed line to the seal will be oriented to preclude
seal drainage in the event of a line break. The seal is capable of many hours
of operation on the self contained fluid.

The oil system supporting the shaft seal contains three tanks, each of which
49 will have a level probe, thereby permitting monitoring of total oil inventory,
its location, and permitting calculation of seal leak rate. The lower seal
leakage collection tank is sized to hold the entire system's oil inventory of
49 approximately 41 gallons.

T1 - Supply reservoir

T3 - Upper Seal Leakage Collection Tank

59 T4 - Lower Seal Leakage Collection Tank

49

33

Oil vapors which may potentially be drawn from the lower seal leakage
collection tank into the tank ullage during draw down (pump speed up) are
retarded from such passage by means of a split flow purge gas feed of recycled
argon into the purge labyrinth. This gas feed splits and flows up and down the
shaft from the feedpoint. This gas input is flow controlled at the inlet, and
49 flow controlled at the discharge from tank T4, the lower seal leakage
collection tank. If feed pressure into the tank is detected to be low (by the
gas feed system) the discharge of gas from tank T4 will be closed. In event of
gas line rupture at the tank T4 discharge, the orificing by the line will
49 59 prevent loss of cover gas pressure. The gas feed to the purge labyrinth will
be made from two supply lines, with an automatic switchover to auxiliary gas,
49 so as to provide dependable gas supply during and after SSE events.

Radioactive vapors from the tank ullage are prevented from escape to the
atmosphere by the two barriers consisting of the gas downflow at the purge
labyrinth and the oil lubricated double shaft seal. Radioactive purging is
continuous by means of the bubbling in the standpipe, which is connected to
33 RAPS.

The kinetic energy of the total rotating mass coupled to the pump impeller following a primary pump trip is as follows:

Pump (impeller, bearing, and shaft)	660,000 ft-lbs
Motor	5,540,000 ft-lbs
33 Total estimated kinetic energy (1116 rpm)	6,195,000 ft-lbs 16

Cracking or internal flaws in the rotating assembly could result in a fracture-type failure; however, the rotational inertia forces are sufficiently low that a structural failure has little potential for serious consequences. It is a design requirement and it has been demonstrated analytically that failure of any part of the pump rotating assembly will not affect the integrity of the pressure boundary. It has been demonstrated analytically that failure of any
59 part of the rotating assembly or the lower region of the inner structure (including the static part of the pump hydraulic system and the shaft lower bearing support) will not reduce the flow path area below a required minimum for natural circulation of the sodium under shutdown conditions.

The pump equipment specification requires the maximum energy of missiles resulting from a fracture of the pump rotating assembly at maximum overspeed to be dissipated before a coolant boundary is breached. An examination of the design of the pump static hydraulic machinery shows that there is a minimum of one and as many as four layers of over one inch thick stainless steel between the impeller and the outer shell of the coolant boundary. It was shown that this structure will easily absorb the energy of a ruptured impeller. The
analysis determined that the maximum kinetic energy available in a ruptured 14
piece from the rotating assembly (at maximum pump overspeed) was less than the energy required to rupture the weakest section of the coolant boundary.

The pump has been designed such that the first bending critical speed of the shaft is at least 125 percent of the design speed. General allowable pump vibrations were specified based on the Hydraulic Institute Standards, 12th Edition. For design conservatism, the vibrational amplitudes of the primary pumps are to be less than those specified by the Hydraulic Institute. Water performance tests of the pump and drive motor assembly substantiated the
59 analytical models used for the dynamics analysis of the pump.

59 The PHTS pump design minimizes the potential for amplification of impeller-volute reaction pressures by acoustic resonance within various portions of the pump and pumping system. Such amplification results from some portion of a pressure pulse arriving at a specific location in phase with other pressure pulses so that the effects are additive. Such an increasing pulse is finally limited by energy dissipation due to flexure of the containment walls, fluid friction, inefficiency of the reflection, etc. Based on fluid borne noise data collected by the pump manufacturer, the fluid borne peak-to-peak pressures have been demonstrated in water tests to be less than 0.5 psi for the PHTS pumps.
59

The design transient descriptions for normal, upset, emergency and faulted plant conditions used in the design of the primary system piping runs are given in Section 5.7.3 and 5.7.4. The process for the selection of transient umbrella events and the design duty cycle is explained in appendix B.

Reactor Cavity Attachment

The primary heat transport system penetrates the reactor cavity at six locations; the 36 inch hot leg between the reactor vessel and the primary pump of each loop and the 24 inch cold leg between the reactor vessel and IHX of each loop. At the penetration the piping is fitted with a "flued head" piping transition piece which becomes an integral part of the piping run. The flued head in turn is welded to a bellows seal which is attached to the cavity wall forming a flexible seal. The bellows, although not code stamped, will be designed to the criteria of ASME Section III, Subsection NF. 30

5.3.2.3.5 Guard Vessels

The IHX's and primary sodium pumps are located in structurally independent, free standing guard vessels. These will be fabricated from Type 304 stainless
41 steel. Location and general arrangement of these vessels is shown in Figure 5.1-3.

The IHX and primary pump guard vessels enclose the component and its primary system inlet and outlet piping up to a level which will preserve the reactor system minimum safe sodium level in the unlikely event of component leakage. The top elevation of each guard vessel is sufficient to prevent spillage of sodium over the lip when the pump is running at pony motor speed and the sodium is at the minimum safe level in the reactor. The guard vessels have segmented covers. These covers do not establish a pressure boundary. The covers contain penetrations for a drain and purge pipe, a gas sample tube and leak detector guide tubes. Additionally, inspection ports are provided to facilitate periscope inspection of the component during refueling, maintenance or other shutdown periods. The maximum pressure which can be experienced by the guard vessels is equal to the sodium head of the filled vessel.

5.3.2.3.6 Insulation

The thermal insulation used for the PHTS is a refractory fiber (alumina silica) in blanket form. The blanket insulation is sheathed in reflective steel. Where in-service inspection requirements are identified the insulation is prefabricated in encapsulated reflective steel modules to facilitate quick removal without disassembly of large sections of insulation. All materials used on the PHTS insulation system are compatible with the PHTS materials.

59

Piping

The PHTS insulation system incorporates a one inch annulus between the pipe and
41 the insulation. The annulus houses trace heaters and stand-off heater
59 retention units. To insure the one inch annulus, there is an inner reflective sheath, then the appropriate thickness of thermal insulation and an outer protective sheath with banding straps. 1

59

The present trace heating design locates the heaters at the bottom of the piping. The heater power is selected so as to produce a specific heatup rate for the piping and to limit the diametral temperature gradient across the pipe. The stresses that result from the diametral temperature gradient on the piping will be combined with other stresses in the piping during the filling condition to ensure that code stress limits are not exceeded. Preliminary evaluations of this condition have shown that the stresses resulting from the temperature
gradients are well within code allowables. 25

Components

For the PHTS components (IHX, pump) the insulation and trace heaters (IHX only) will be supported by a fabricated stainless steel framework supported from the cell structure. Annuli between the vessel surfaces and the insides of the insulation system structures will prevent metal-to-metal contact. For the PHTS component guard vessels (IHX-GV and pump GV) the fabricated insulation system frameworks are supported from appurtenances provided on the vessels and secured to the vessels by banding straps. A network of pins and banding straps will support layers of alumina-silica blanket insulation which can be readily removed so the trace heaters can be serviced if necessary. An outer protective
59 sheath with banding straps will keep the insulation in place.

Thermal Analysis

59 The heat loss per linear foot of piping versus insulation thickness is
calculated using the following equation:

$$\underline{q} = \frac{2\pi r_0\ (\Delta T)}{\frac{r_0}{r_1 h_1} + \frac{r_0 \ln \frac{r_2}{r_1}}{k_1} + \ldots\ldots + \frac{r_0 \ln \frac{r_{x+1}}{r_x}}{k_x} + \frac{1}{h_r + h_c}} \quad \left(\frac{BTU}{hr\text{-}ft}\right)$$

The design requirement for in-containment insulation exterior surface
59 temperature is 140°F maximum with a cell atmosphere temperature of 90°F. Since
the temperature drop from the pipe to the outer surface is proportional to the
59 thermal resistances of the materials, the surface temperatures were calculated
using the following equation:

$$T_{surface} = T_{max} - (\Delta T)\ \frac{R_{insul}}{R_{total}}$$

The insulation thicknesses are, generally 14" on components, 1" on hot leg
pipes and 8" on cold leg pipes. In some limited envelope applications, reduced
59 thicknesses of special high temperature insulation are used. 1

5.3.2.4 Overpressure Protection

The reactor coolant system contains no pressure relieving devices for the sodium coolant. Pressures in the reactor coolant system are normally a function of the reactor cover gas pressure and the pressure developed by the main circulation pumps. Since the system contains no isolation valves, isolation of individual loops or components in those loops is not possible.

The reactor cover gas pressure is protected from pressures in excess of 15 psig by relief valves connected to the cover gas space of the primary overflow tank. The overflow tank gas space is connected to the reactor vessel cover gas space through a pressure equalization line which contains no isolation valves. (See Section 9.5) System pressures with pumps running at classified as faulted for the affected steam generator module. Differential pressure between the primary and intermediate sides during this event is conservatively evaluated by assuming that the primary side pressure is that resulting from pony motor speed (approximately 6 psig). For the rest of the loop, the occurrence is classified as an emergency event.

For the unaffected loops, the event is similar to the reactor trip from full power. Decay heat removal is maintained throughout the two remaining loops. The transient responses of temperature, flow and pressure on both the primary and intermediate side of the IHX in the
56 affected loop are presented in Figures 5.3-18A through 5.3-18G. Particular attention is directed to Figures 5.3-18F and G which show intermediate side short term and long term pressure effects.

In evaluating the structural adequacy of the IHX, with respect to the check valve slam, the dynamic nature of the primary sodium pressure history is being accounted for by using dynamic load factors. The factor will be applied to the maximum primary pressure, which in turn, is used to determine the pressure-induced primary stresses. These primary stresses are limited by the emergency condition allowables of Code Case 1592, Paragraph 3224, as modified by RDT F9-4T. The fatigue damage associated with the cyclic nature of the pressure history will be accounted for per Paragraph T-1400 of Code Case 1592. The description of the pressure pulses for the sodium-water reaction and check valve closure is included in the equipment specification. The curves define the amplitudes, duration and number of cycles.

Rapid check valve closure can only occur as a result of primary pump mechanical failure. The event involves a postulated instantaneous stoppage of the impeller of one primary pump, while the system is operating at 100% power. The failure may be a seizure or breakage of the shaft or impeller. Primary system sodium flow in the affected loop decreases rapidly to zero as the pumps in the unaffected loops seat the check valve (thereby causing a rapid check valve closure or slam). A reactor trip will be initiated by the primary intermediate flow ratio subsystem. Sodium flow in the intermediate circuit of the affected loop decays as in a reactor trip from full power, modified by changes in natural circulation head. The event is characterized by a down transient in the hot leg of the intermediate circuit of the affected loop. The transient responses of temperature, flow and pressure on both the primary and intermediate side of the IHX in the affected loop are presented in Figures 5.3-18H through 5.3-18M. Particular attention is directed to Figure 5.3-18J which shows primary pressure effects.

Both the sodium water reaction and check valve closure events are classified as emergency events for the IHX. As such, the IHX designer is required to determine which of the six emergency events is most severe to the IHX. The selected event is then applied with a periodicity of two consecutive occurrences during the first three years of operation, and thereafter five times over the remaining 27 years (or once every six year period). If vendor analysis indicate either as the most severe event, the occurrence of the two consecutive events will be moved to the most stringent time in the life for the event to occur. The IHX design has not progressed to the point where either the sodium water reaction or check valve closure can be defined as the most severe emergency event. Rather, preliminary analysis indicates that damage from either of these events will be insignificant.

Pump

Inelastic analyses of the pumps was required to demonstrate conformance with
59 27 the ASME Code. Paragraph 4 of RDT Standard F9-5T, Sept. 1974 gives a description of acceptable methods for time-independent elastic-plastic analysis and time-dependent creep analysis. Some of the computer programs listed above have inelastic capabilities, and will be used where applicable.

For the purposes of loads and analysis the pump R-Spec divides the pump into four areas. These are: Subcomponent 1 which consists of the pump tank, Subcomponent 2 which is the upper inner structure including the pressure bulkhead, Subcomponent 3 which is the rotating machinery and Subcomponent 4 which is the static hydraulics.

Subcomponent 1 is designed to the ASME Boiler and Pressure Vessel Code Section III, Subsection NB Class 1 and Code Case 1592 where applicable. The cone and cylinder are designed mainly by dynamic stiffness requirements. These include seismic loads and the necessity of keeping the natural frequency of the structure above the operating speed of the impeller. SAP IV and the "NASTRAN"
59 computer codes were used for this analysis. The analysis has been qualified by comparing the results of one analysis against the other. The sphere sealing ring and cone-sphere support ring are designed by sealing ring leakage which requires elastic response during normal and upset conditions. A failure will reduce pump efficiency below plant criteria. These areas have been analyzed by 3D global analysis using NASTRAN. The nozzles are designed by pressure, pipe
59 nozzle loads, and thermal transients. The failure modes associated are creep and creep fatigue. 2D elastic analysis is required. The design is being made with sufficient space for thermal baffles and liners to keep it elastic as much as is possible. But it may be necessary to qualify it using simple inelastic analysis. Hydraulic leakage test data has been obtained which determined the
51 relation of sealing gap to leakage rate.

59 Subcomponent 2 conforms to the same Code requirements as Subcomponent 1. The upper closure plate and radiation shield are designed by the design pressure and temperature requirements. Elastic failure is the predominant mode. The heat shield has steady state thermal gradients which are determined by a 2D
59 axisymmetric model and stresses are calculated with a 2D stress model. The motor stand has been designed by the stiffness requirements of the motor and seismic loads. The principle failure mode is excess vibration leading to fatigue failures.

59

Subcomponent 3 can be removed and inspected after an emergency or faulted event and repaired before the plant is placed in service again. Therefore, this section was designed and analyzed to the ASME Boiler and Pressure-Vessel Code, Section III, Subsection NB for Class 1 Components and Code Case 159_ where
59 applicable. However for emergency events Code Case 1592 is used and the design rules for load controlled stresses (Section 3227) applies. Strain deformation and fatigue analysis need only be performed up to the emergency event and the limits will apply only to the pumps ability to operate at pony motor speed after the event. This area has been designed by critical frequency
59 requirements, inertial loads, torque and thermal transients. It was analyzed with a 2D axisymmetric model. The loads caused by bearing misalignment were accounted for. A general 1/2 scale model hydraulic performance test was run
26 using water as the pumped fluid. This test provided information on the pump
59

NPSH and internal leakage flows. Inelastic analysis of the upper journal impeller weld region of the rotating assembly using ANSYS, was required to show
59 adequate ratchetting strain margins for various upset events.

Subcomponent 4 consists of the lower removable region of the pump inner structure. It was analyzed to the same Code rules as Subcomponent 3. The
59 principle loads are thermal transients, hydraulic pressure, containment of a failed impeller, reaction loads against the hydraulic machinery due to deformation of the sphere during the thermal transients and bearing loads during asymmetrical heating. Principle failure modes associated are elastic
59 failure, creep and creep fatigue. The hydraulic casting has been analyzed by a 3D global model using NASTRAN. The bearings are fed directly from the pump discharge so they are exposed to thermal transients. They have been analyzed with a 2D axisymmetric analysis to develop loads and stresses. An axisymmetric 2D model was used to calculate the stresses in the static shroud around the impeller. Inelastic analysis was required in the bearing support region using MARC and ANSYS.
59

<u>Piping</u>

The incontainment sodium piping shall be designed and analyzed to the Class 1 requirements of the ASME Code, Section III and Code Case 1592. The piping will be designed to assure that piping stresses, strains and deformations are within the applicable Code criteria and system functional limits. The analyses to satisfy these limits shall reflect both time-independent and time-dependent material properties and structural behavior (elastic and inelastic) by considering all of the relevant modes of failure listed below:

1. Ductile rupture from short-term loadings
2. Creep rupture from long-term loadings
3. Creep-fatigue failure
4. Gross distortion due to incremental collapse and ratchetting
5. Loss of function due to excessive deformation
6. Buckling due to short-term loadings
7. Creep buckling due to long-term loadings

Stresses that result from pressure, dead weight, thermal restraint, thru-the-wall thermal transient, and seismic loadings will be considered in evaluating the failure modes of the piping and piping material.

The types of analysis required to verify the design of the piping will include elastic, simplified inelastic and detailed inelastic. Simplified inelastic and detailed inelastic methods that are to be used will conform to the requirements
26 of RDT Standard F9-4T and the guidelines of RDT Standard F9-5T.

Degradation of material properties over the lifetime of the piping in accordance with the requirements identified in the equipment specification will be accounted for.

26 No structural verification testing is planned for the main sodium piping.

5.3.3.1.6 <u>Analytical Methods for Evaluation of Pump Speed and Bearing Integrity</u>

The primary pump shaft is supported at the impeller end by a sodium bearing. Hard facing material is applied to the bearing and journal surfaces to provide
59 wear-resistant surfaces for startup and shutdown transients during journal lift-off and touchdown. 33

The approach to reduce or eliminate large thermal stresses in the sodium
bearings is to provide two-sided exposure to the thermal transients over most
of the bearing/journal supports and to use baffling to isolate the bearings
from the large enclosed sodium pools. The bearing supports were analyzed for
59 temperature differences between the hydraulic assembly casting and the bearing
supports during thermal transients, including thermal shear loads on bolts. 33

Because the bearing hard face surfaces are to be protected from the rapid
thermal transients, a two-step approach to analysis was taken. The first step
was to determine the transient temperature distribution in the structure.
59 These temperature distributions were then used in the second step to establish
the thermally induced loads.

59 Pump structural lateral and vertical natural frequency and critical speed
analyses were performed using SAP IV and STARDYNE computer programs. The shaft
torsional natural frequency and critical speed analyses was performed using SAP
59 IV. These analyses included calculation of the natural frequency by eigen-
value extraction from the dynamical matrix based on the standard finite-element 33
stiffness formulation. In addition, time history and response spectrum dynamic
response calculations were performed.

59

Amend. 59
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exit velocity (11 ft/sec) at design flow from the reactor vessel. Gas
entrainment within the pump is minimized by designing and testing, to ensure
that all pump parts which need to be submerged during operation at pony motor
speed or restart to pony motor speed are located below the minimum sodium
59 level.

The maximum oxygen content in the primary system sodium is specified to be ≤ 2
ppm at 800°F or above and ≤ 5 ppm below 800°F. This level of sodium impurity
will not affect the pump operating characteristics.

The biological shielding for the PHTS sodium pumps as well as the pump
59 assemblies is designed to withstand the loadings associated with the Safe
Shutdown Earthquake (SSE) and the transient overpressures for extremely
unlikely plant conditions. The analyses required to demonstrate this treated
59 the PHTS pump as a Class I component in accordance with the rules of the ASME
Code Section III and modifying RDT Standards.

The biological shielding for the PHTS sodium pumps is provided by (1) an
annular shield tank which surrounds the pump shaft, (2) the pump shaft itself
which is designed to provide an integral part of the shield requirements, (3)
the pump support structure which is part of the operating floor, and (4)
59 special precautions to preclude streaming along the instrumentation
penetrations. The annular shield assembly is integral with the top closure
flange of the pump pressure boundary containment vessel which is designed in
accordance with the ASME Code for Class 1 nuclear components. The pump shaft
supporting assembly and the annular shield structural assembly are supported on
the pump tank flange which is mounted on a pump support ledge designed into the
operating floor pump motor well. The design of this joint provides for the
dual function of resisting static and dynamic loads and permitting final seal
welding containment of the pump atmosphere boundary and the vault atmosphere
boundary.

43 For the PHTS pump SSE seismic analysis, a 2% damping value was used. The SSE
loadings were considered to occur in conjunction with a plant trip. Following
59 the SSE, the Intermediate Heat Transport System, Steam Generator System, and
Steam Generator Auxiliary Heat Removal System must provide for removal of
59 stored and decay heat. The primary pump is designed to maintain pony motor
flow without loss of structural integrity after the SSE. Computer programs,
such as SAP IV and ANSYS, will be utilized to perform seismic analyses on the
primary pumps. Descriptions of these computer programs can be found in
Appendix A.

33

This page was deleted in Amend. 33

flow are 13.7 psi nominal for the shell (primary) side and 8.5 psi nominal for
59 the tube (intermediate) side.

During full power operation, the thermal center of the unit is 15 feet above the core midplane. (The geometric center of the tube bundle will be approximately 16'-3" above the core midplane). The thermal center is that point along the axis of the tube bundle at which half of the heat transfer has taken place; that is, the enthalpy at the thermal center is the average of the inlet and exit enthalpies. During a transient, the thermal center shifts. When the pumps are tripped, the primary and intermediate flows tend to collapse together. However, the intermediate flow decreases more rapidly (proportionately) because there is less momentum stored in the intermediate system. This "mismatch" of flows causes a slight lowering of the thermal center in the IHX. A preliminary analysis indicated that the thermal center would drop 1 foot in the first 10 seconds and an additional 3 feet in the next 90 seconds. As natural circulation continues, the thermal center gradually moves towards the top of the tube bundle because under steady state natural circulation conditions, there is more flow in the intermediate loop than in the primary loop. This condition arises primarily as a result of the smaller system flow impedance of the intermediate system and its larger elevation difference by comparison to the primary system. These effects are accounted
15 for in DEMO, the system transient analysis code (see Appendix A).

Internal convection within the unit is not expected to be significant. The shell side of the unit is baffled to create crossflow in addition to axial flow, and this feature is expected to minimize the tendency to develop maldistribution of flow even at low flows. Tube to tube flow variations on the intermediate side would be expected to be self-correcting due to buoyancy 1
effects.

The design transients for normal, upset, emergency and faulted plant conditions are described in detail in Appendix B of this PSAR. Also, factored into the design is the effect of corrosive oxide on heat transfer.

The level of sodium oxide and other corrosive impurities, in the primary system, will be maintained and controlled by continuous cold trapping at 60 gpm. as discussed in Section 9.3. The present oxide levels during normal system operation is maintained below 2 ppm.

As mentioned above, the heat transfer area includes an allowance for fouling. This allowance is consistent with a 9% degradation of the overall heat transfer coefficient due to mass transfer deposits which is based on experimental data
reported in Reference 2a. 1

The IHX is designed to use tubes with 0.045 inch -0 wall thickness (i.e., 0.045 inch min.). Allowing for 0.001 inch corrosion on either side of the tube wall, and 0.005 inch for scratches on the surface, the minimum available wall thickness for analysis is 0.038 inch.

Analysis per ASME code for 200 psi design pressure at 775^{o}F requires a minimum w~~l~~l of 0.030 inch. The available wall thickness of 0.038 inch would perm.: an external pressure of 280 psi (per code) with an inherent

5.3.3.10.3.2 Mass Transfer of Radioactive Species

The radioactive aspects of mass transfer in the reactor coolant from the reactor to the Heat Transport System are discussed in detail in Section 11.1. These aspects, in themselves, do not affect the structural integrity of the HTS.

5.3.3.10.4 Compatibility with External Insulation and Environmental Atmosphere

Within the heat transport system the reactor vessel, pumps, and intermediate heat exchangers are enclosed by guard vessels. Between these components and the guard vessels a semi-inert gaseous atmosphere of 0.5-2% oxygen/nitrogen is maintained. The piping and the upper portions of the components containing sodium external to the guard vessels are also insulated to minimize heat loss to the PHTS cells. The thermal insulation consists of alumina silicate blanket
53 material manufactured under controlled conditions to minimize the pickup of halogens and/or moisture. The insulation is protected from halogen pickup during shipping, storage and installation. The insulation has an inner liner and is installed on standoffs to provide an annulus for heaters and leak
59 53 detection equipment and, therefore, does not directly contact piping or components. No field compounded thermal insulation materials are used. This will minimize any potential contamination of the piping by corrosive elements in the insulation. Most piping is also exposed to the 0.5-2% oxygen/nitrogen atmosphere. 1

Sodium leaks into the guard vessels, should they occur, are unlikely to be self sealing in view of the low oxygen content. Small leakages will be contained within the guard vessel. With respect to the piping (except that which is situated within the guard vessels) any sodium leakage will react with oxygen, nitrogen, and thermal insulation. No comprehensive data appear to be available to evaluate the reaction in detail but available information from experimental sodium loops indicates that the leaking sodium will form a sodium oxide (and very likely sodium nitride) "growth" beneath the insulation at the point of
53 leakage. For temperatures below about 1000°F no self sealing of the leak is usually observed. Studies were conducted to evaluate the nature of sodium leakage through precracked austenitic stainless steel piping into a 1.2 v/o oxygen/98.7 v/o nitrogen atmosphere. Materials of Construction are listed in Tables 5.3-4 thru 5.3-9.

5.3.3.10.5 Chemistry of Reactor Coolant

The heat transport system sodium chemistry is selected to minimize corrosion. A periodic analysis of the coolant chemical composition is performed to verify that the coolant quality meets the specifications.

56

Sodium purification capability is provided through the use of cold traps.

Capabilities are provided both for "in-line" primary and
56| intermediate sodium purity determinations (sodium plugging temperature indicators) and for direct sampling and laboratory analysis to monitor impurities. The systems are described in Section 9.3.2.

TABLE 5.3-11

$S_t(2 \times 10^5 h)$ AND S_m VALUES FOR AUSTENITIC STAINLESS STEELS*

TEMPERATURE (°F)	S_t (ksi)				S_m (ksi)			
	304 L	304 and 304 H	316 L	316 and 316 H	304 L	304 and 304 H	316 L	316 and 316 H
100	N/A	N/A	N/A	N/A	16.6	20.0	16.6	20.0
200	N/A	N/A	N/A	N/A	16.6	20.0	16.6	20.0
300	N/A	N/A	N/A	N/A	16.6	20.0	16.6	20.0
400	N/A	N/A	N/A	N/A	15.7	18.7	15.5	19.2
500	N/A	N/A	N/A	N/A	14.7	17.4	14.4	17.9
600	N/A	N/A	N/A	N/A	13.9	16.4	13.5	17.0
700	N/A	N/A	N/A	N/A	13.4	15.9	12.8	16.3
800	N/A	20.5	N/A	21.0	13.0	15.1	12.3	15.8
900	N/A	16.6	N/A	19.6	N/A	14.6**	N/A	15.7**
1000	N/A	10.0	N/A	14.9	N/A	14.0**	N/A	15.5**
1100	N/A	6.1	N/A	8.5	N/A	13.3**	N/A	14.8**
1200	N/A	3.7	N/A	4.8	N/A	12.7**	N/A	14.6**

*Data are from ASME Boiler and Pressure Vessel Code, Section III and ASME Code Case 1592, and apply to Class 1 components.

**For Type 304 and Type 316 grades only.

N/A = Not applicable.

TABLE 5.3-12

PRIMARY HEAT TRANSPORT SYSTEM CODE
AND SEISMIC CATEGORY MATRIX

Primary System Component	RDT Component Standards[1]	ASME Code Section/Class	Seismic Category	F9-4T Req'ts For Nuclear Components At Elevated Temp.
Primary Pump	E3-2T June 1974	III/1	I	Note 2
IHX	E4-6T April 1975	III/1	I	Note 2
Check Valve	E1-18T May 1975	III/1	I	Note 2
Flow Meter	C4-5T April 1974	III/1	I	-
27 Guard Vessels	E10-2T July 1973	III/2	I	Note 3
Piping & Fittings	N/A	III/1	I	Note 2
27 Pipe Hangers Supports & Snubbers	E7-6T May 1972	III/ Subsection NF	I	-
59 Thermal Insul.	N/A	N/A	III*	-
Trace Heating	N/A	N/A	I	-

Notes:

1. Component standards used as guidance in equipment specification preparation, not invoked.

2. This standard, modified only as indicated in the equipment specification, is to be applied in its entirety to all structures in the component.

3. Constructed to rules of Class 1 but not hydrostatically tested or code stamped. An elevated temperature supplement to the equipment specification, equivalent to RDT Standard F9-4 with modifications and code cases, will be used.

59 *Thermal insulation is functionally seismic Category III, however, it is designed to the requirements of Seismic Category I, utilizing static analysis rather than dynamic analysis.

TABLE 5.3-13

COLD LEG CHECK VALVE CHARACTERISTICS

Requirement	Units	Design Value
Design Flow Rate (at 730°F)	lbs/hr	13.82×10^6
Flow Range at Normal Operating Conditions	% of Design Flow	40 - 100
Max. Shutoff, P Imposed Across Seat		
Steady State	psi	160
Pressure Loss at Design Flow	psi	≤10
Pressure Loss at Pony-Motor Flow Conditions of 2500 gpm at 600°F	psi	≤0.20
Pressure Loss at Natural Circulation Flow Conditions of 670 gpm at 730°F	psi	≤0.03
Temp. at Which Design Flow Pressure Loss is Calculated	°F	730
Allowable Leakage in Reverse Direction at Shutoff at 730°F	gpm	21
Pressure Difference for Allowable Leakage, Reverse Direction	psi	50

Closure Characteristics

The maximum steady state reverse flow allowed by the check valve shall be less
than 1100 gpm. The valve shall not require a pressure differential across the
59| disk greater than 1.75 psi to shut. Closing time shall be 12 seconds maximum
(after flow reversal) with a resultant pressure surge of less than 50 psi under
the specified reverse flow conditions.

56 53 TABLES 5.3-14 thru 5.3-22

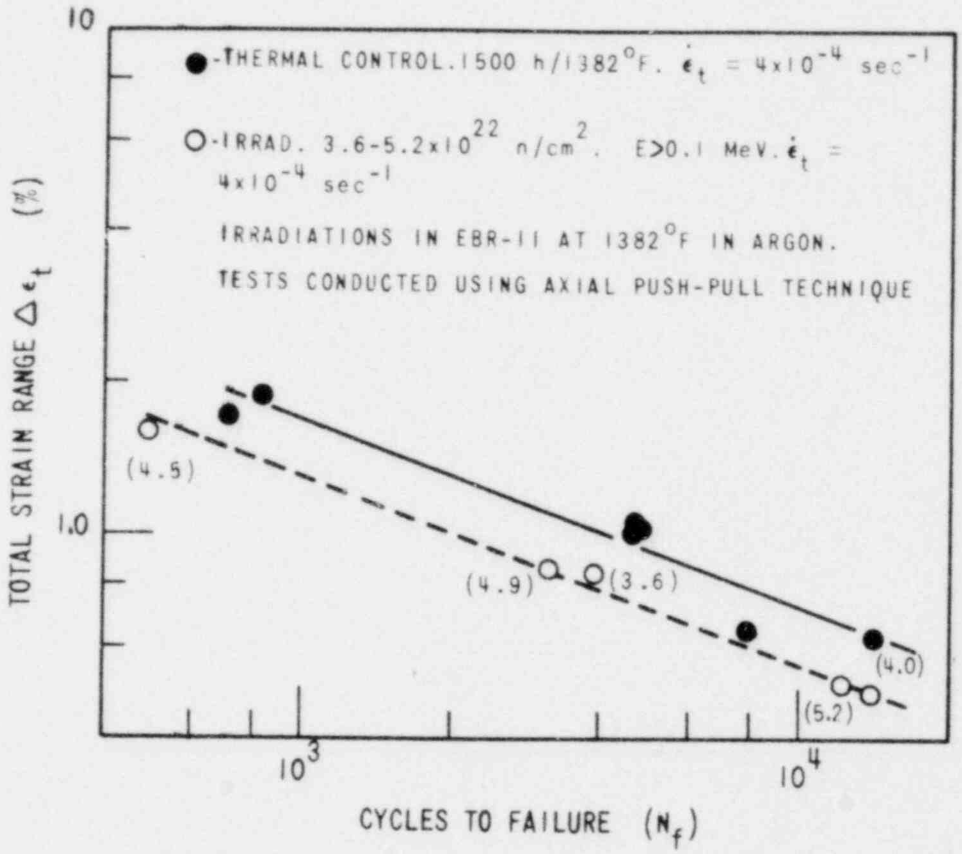
HAVE BEEN DELETED

(next page is 5.3-97a)

TABLE 5.3-23

PHTS and IHTS Pump Generated Frequencies

Condition	Structural Vibration (Frequency - hz)	Pressure Pulse: Enter Impeller (Frequency - hz)	Pressure Pulse: Leave Impeller (Frequency - hz)	
Primary Coolant Pump				
Pony Motor Flow	1.87	19	56	
40% Rated Flow	7.4	74	223	
80% Rated Flow	14.9	149	447	
59 100% Rated Flow	18.6	186	553	
Intermediate Coolant Pump				
Pony Motor Flow	1.6	16	48	
40% Rated Flow	6.4	64	192	
80% Rated Flow	12.8	123	384	
59 100% Rated Flow	16.0	160	480	25



POOR ORIGINAL

Figure 5.3-13. Effect of Neutron Irradiation on the Fatigue of Type 304SS at 932°F

* Taken from Reference 47. | 25

6669-16

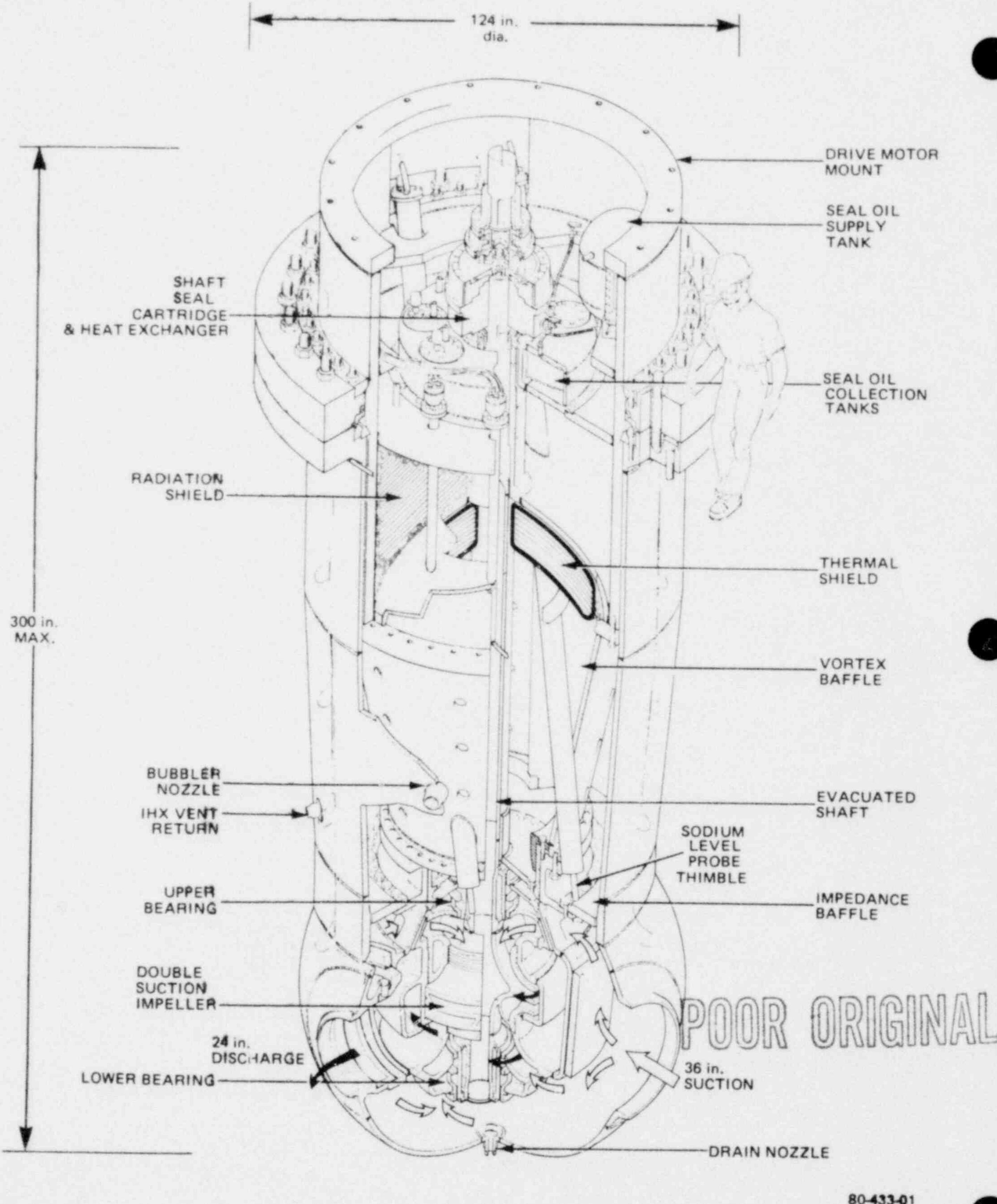
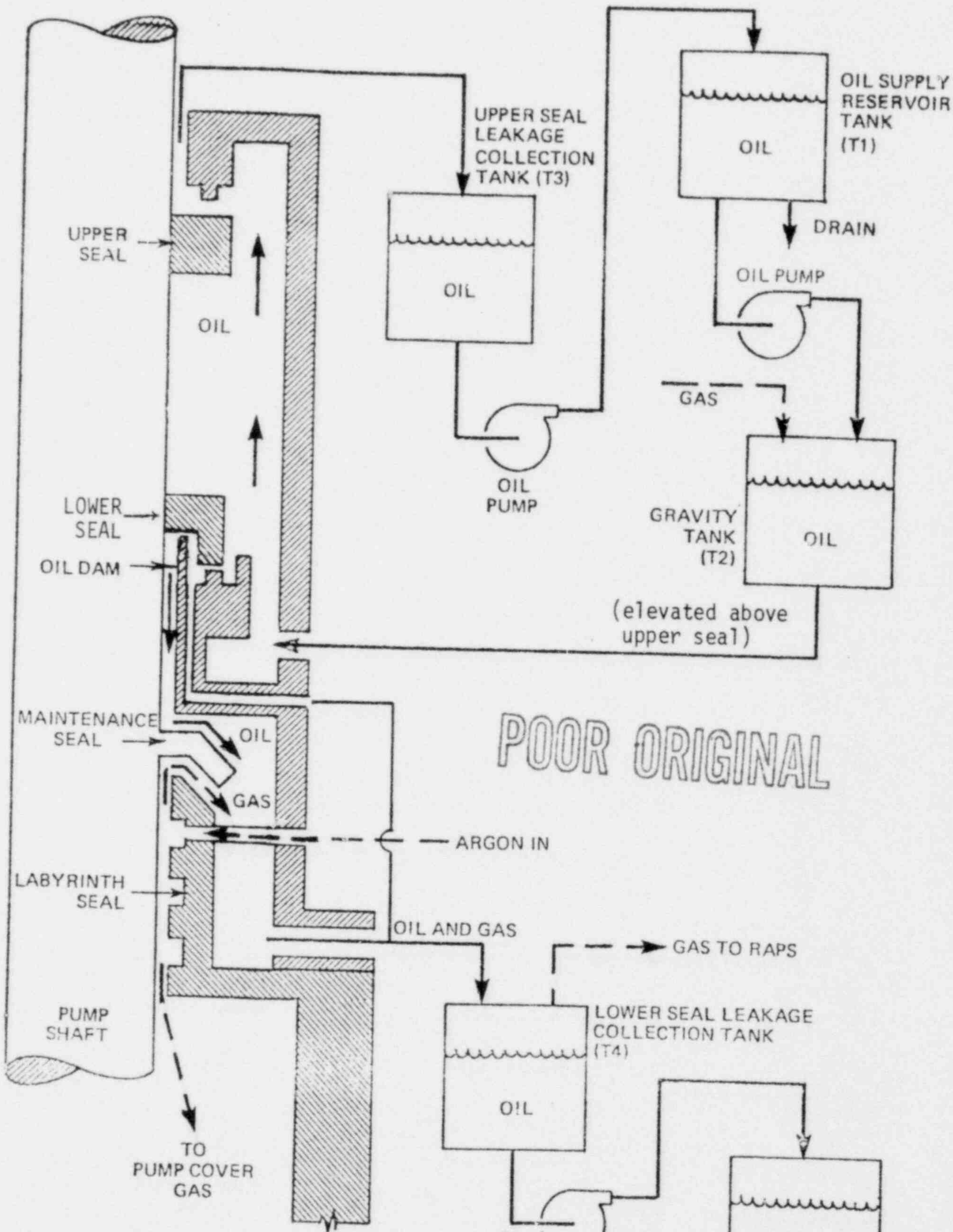


Figure 5.3 - 14 PRIMARY PUMP ISOMETRIC



POOR ORIGINAL

Figure 5.3-14A Schematic of Typical Seal Oil System

Amend. 59
Dec. 1980

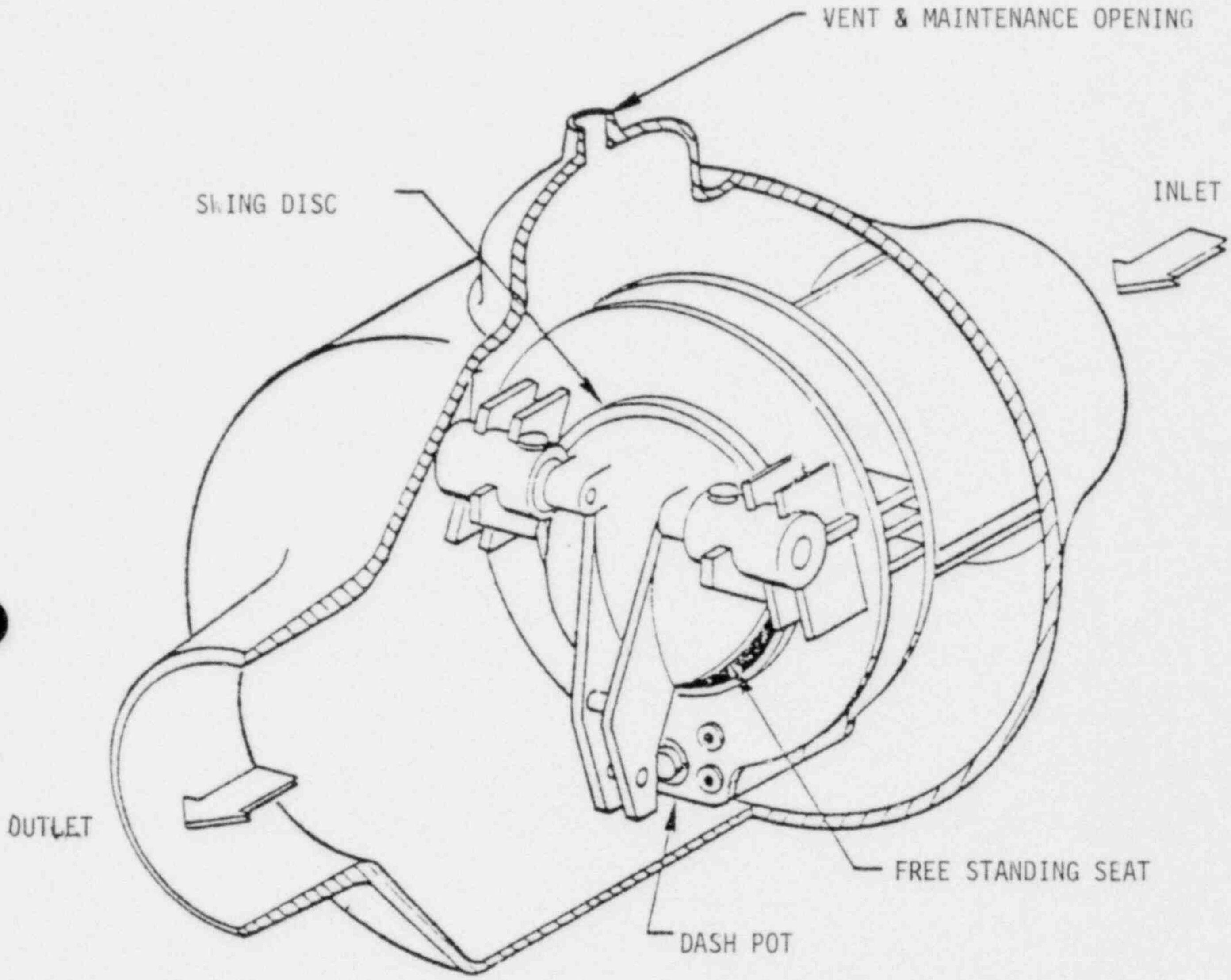


Figure 5.3-16 Preliminary Cold Leg Check Valve Concept

	SECTION													
	AREAS CONSIDERED FOR ANALYSIS													
I. ANALYSIS PREPARATION														
A. REVIEW DESIGN SPECIFICATION														
B. REVIEW DESIGN CONCEPT AND MAKE FAILURE MODE ANALYSIS														
C. PREPARE STRUCTURAL EVALUATION PLAN (SEP)														
II. CLASS OF ANALYSIS AND REPORT PREPARATION														
A. PRELIMINARY														
B. INTERIM														
C. FINAL														
D. FORMAL														
III. OPERATING CONDITIONS														
A. DESIGN														
B. NORMAL														
C. UPSET														
D. EMERGENCY														
E. FAULTED														
F. TESTING														
G. SPECIAL														
IV. TYPE OF ANALYSIS														
A. HEAT TRANSFER														
(1) TRANSIENT METAL TEMPERATURE														
(2) STEADY STATE METAL TEMPERATURE														
B. MECHANICAL STRESSES														
(1) PRESSURE LOADS														
(2) DEADWEIGHT														
(3) PIPING LOADS														
(4) EXTERNALLY APPLIED LOADS														
(5) SEISMIC LOADS														
(6) DYNAMIC LOADS														
(7) VIBRATORY LOADS														
C. THERMAL STRESSES														
(1) RADIAL GRADIENT														
(2) LONGITUDINAL GRADIENT														
(3) THERMAL DISCONTINUITY														
D. BUCKLING														
(1) GROSS INSTABILITY														
(2) LOCAL INSTABILITY														
(3) CREEP BUCKLING														
(4) THERMAL WRINKLING														
E. DEFORMATION AND STRAIN LIMITS														
(1) FUNCTIONAL LIMITS														
(2) STRUCTURAL LIMITS														

Figure 5.3-17 Sample Structural Design and Analysis Checklist (Continued on Next Page)

6669-9

POOR ORIGINAL

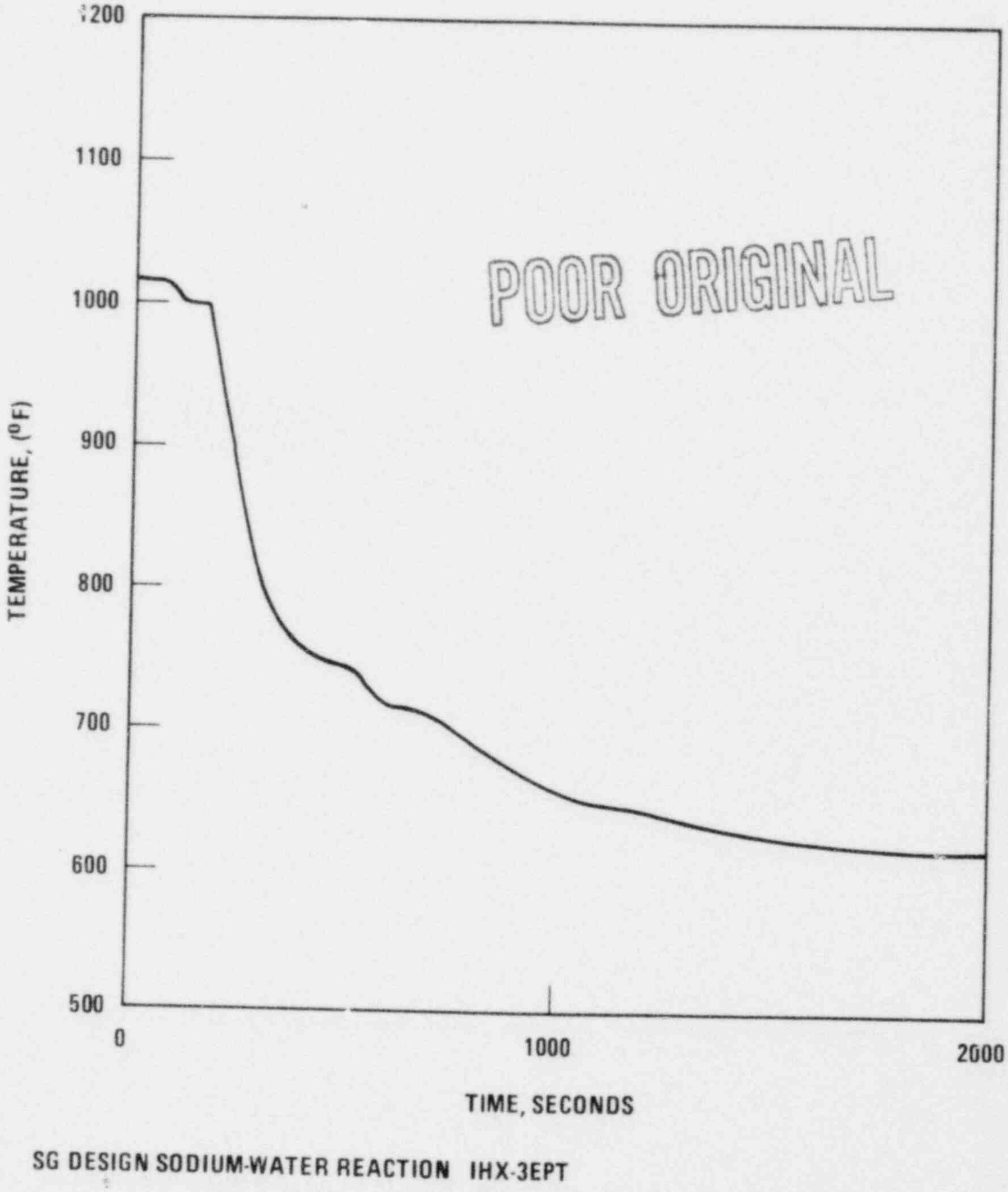
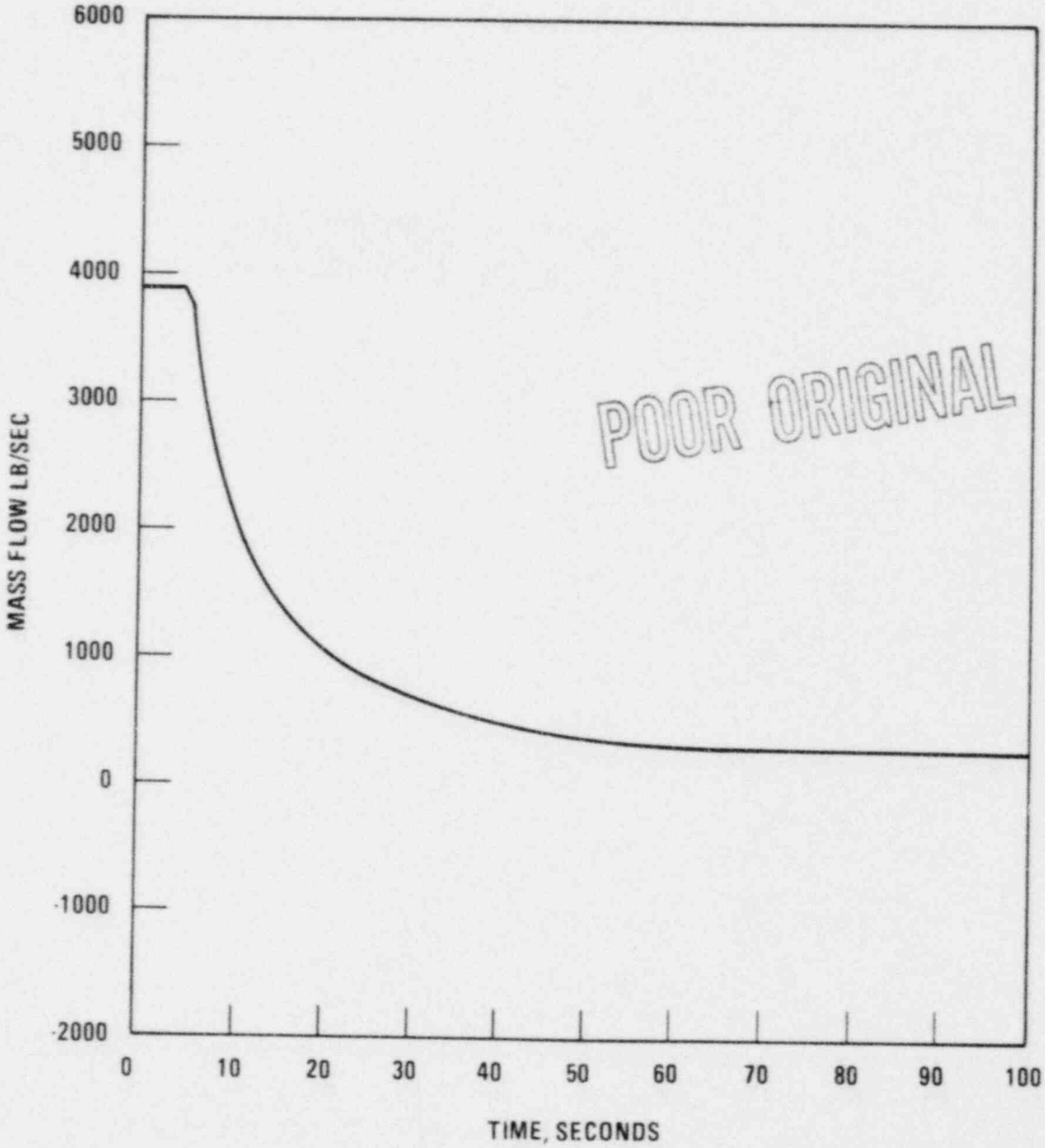


Figure 5.3-18A. Intermediate Heat Exchanger, PRI IHX Inlet Nozzle Temperature-vs-Time

4786-3



SG DESIGN SODIUM-WATER REACTION IHX-3EPF

Figure 5.3-18B. Intermediate Heat Exchanger, PRI Pump Ex Massflow-vs-Time

4786-1

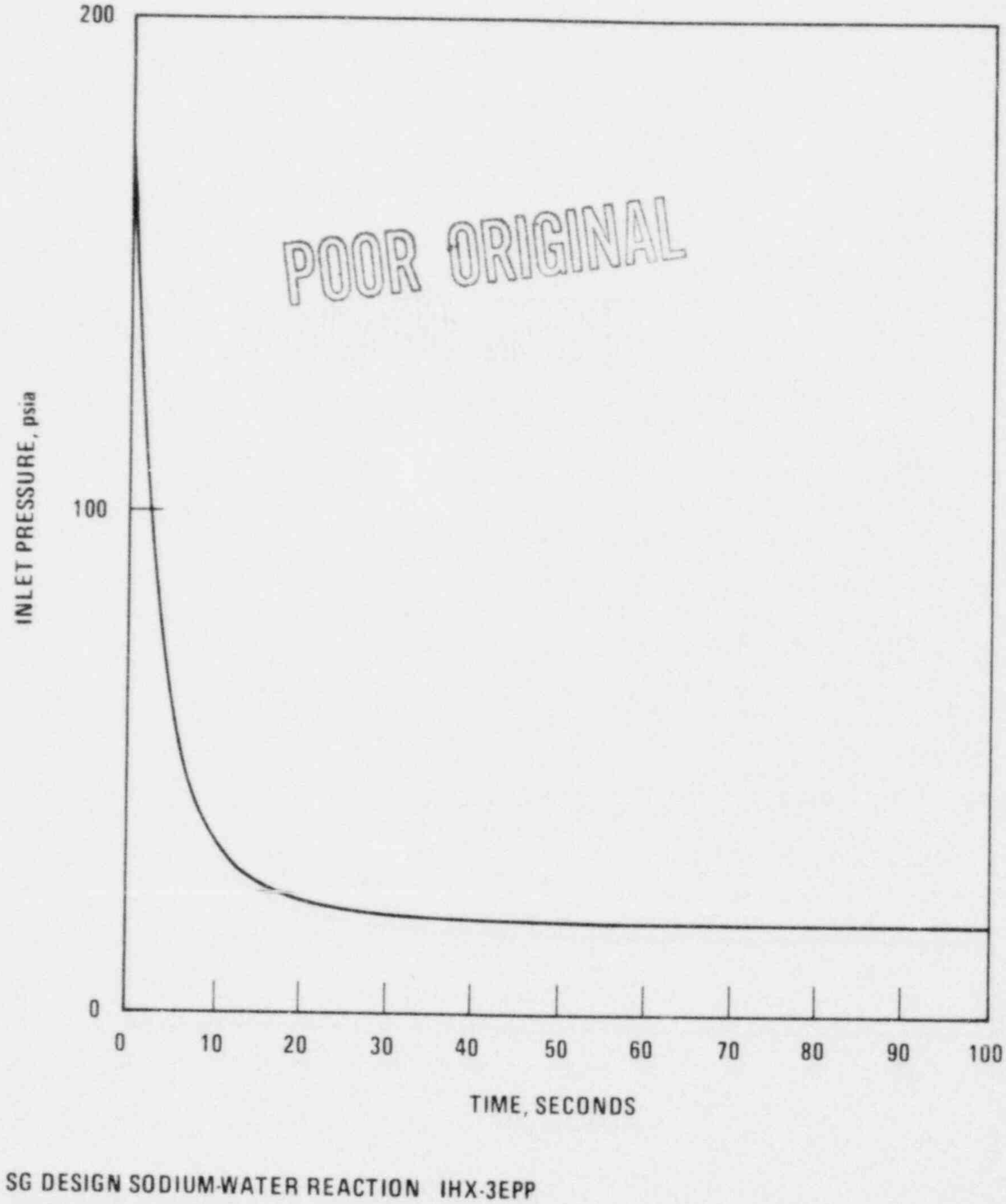


Figure 5.3-18C. Intermediate Heat Exchanger, Primary IHX Inlet Pressure-vs-Time

4786-2

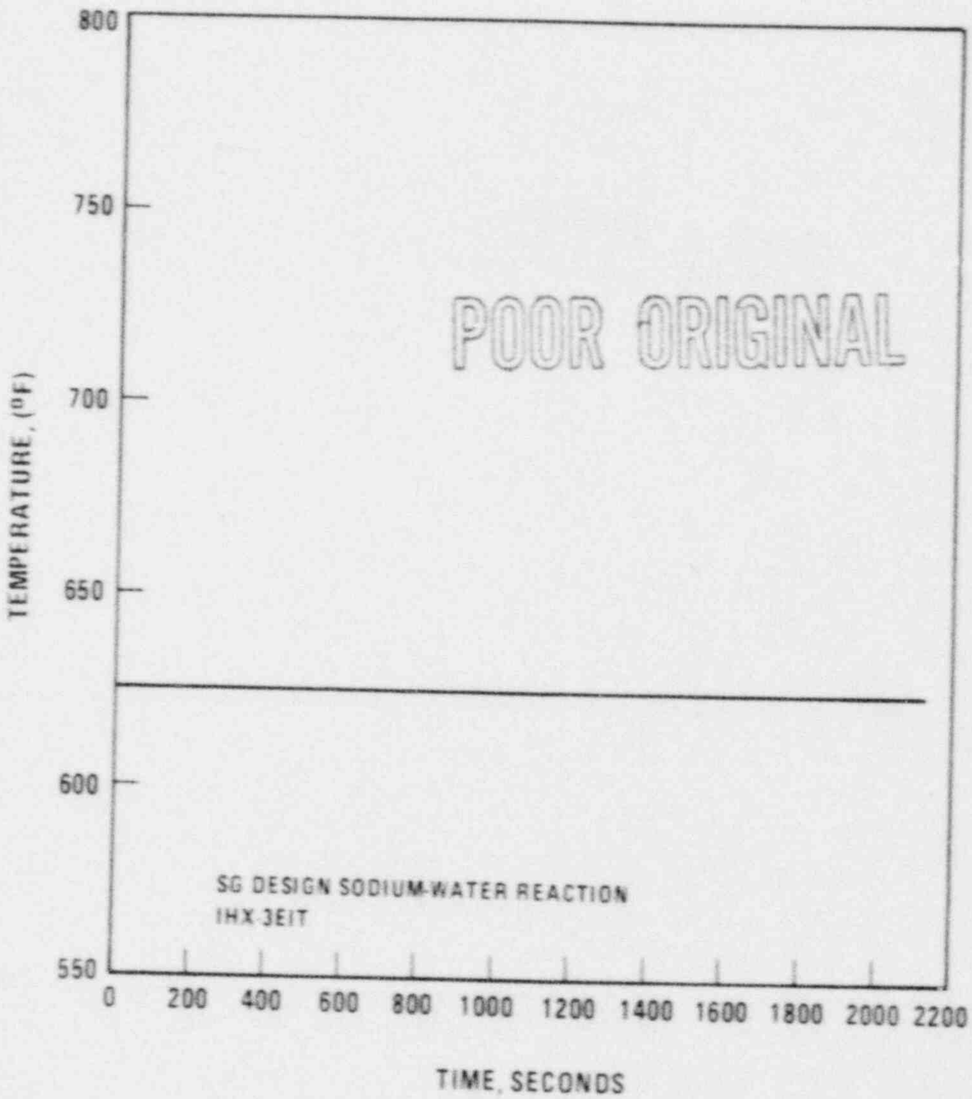
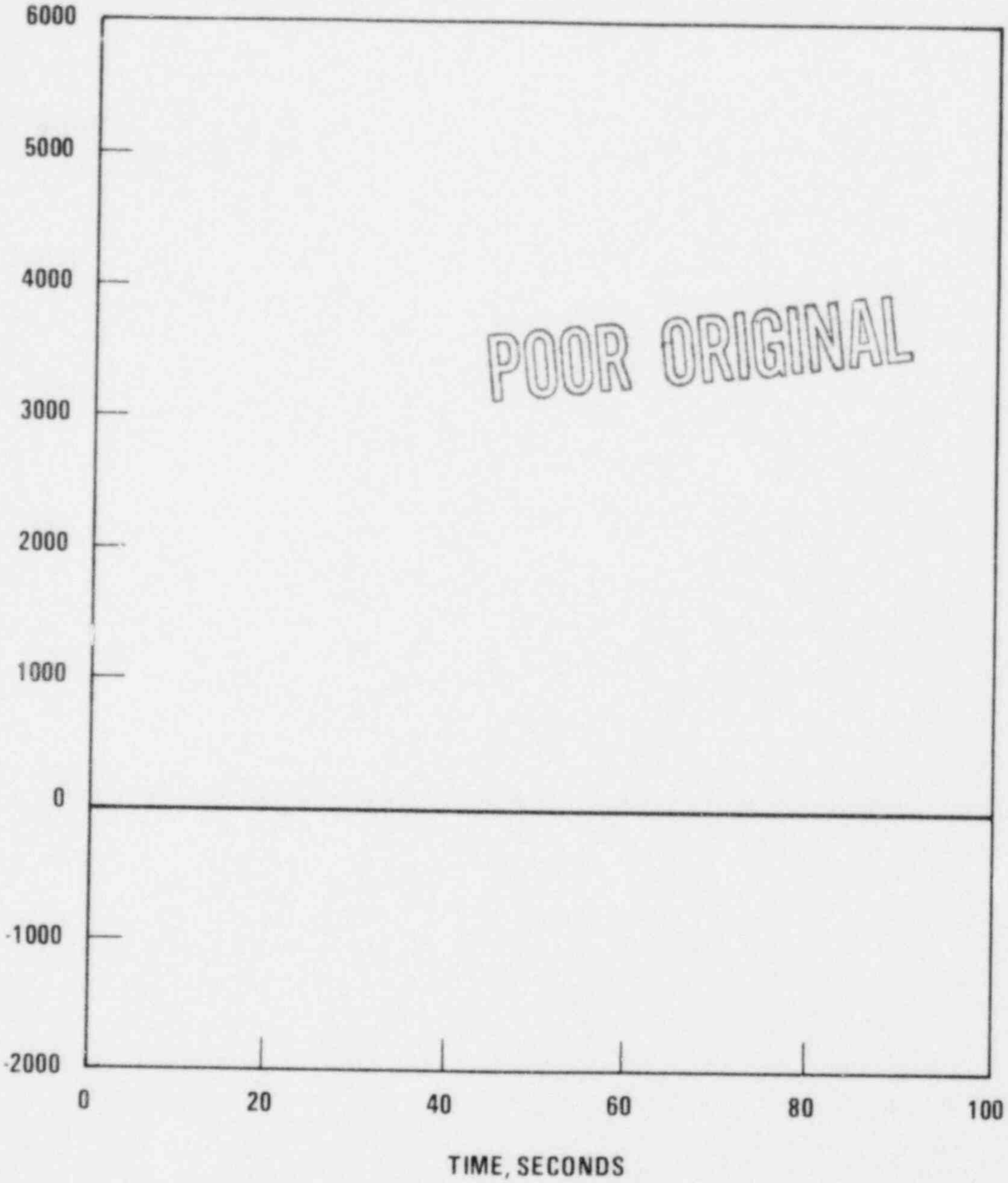


Figure 5.3-18D Intermediate Heat Exchanger.
INT IHX Inlet Nozzle Temperature-vs-Time

7683-168



SG DEISGN SODIUM-WATER REACTION IHX-3EIF

Figure 5.3-18E. Intermediate Heat Exchanger, INT IHX Inlet Nozzle Flow-vs-Time

4786-6

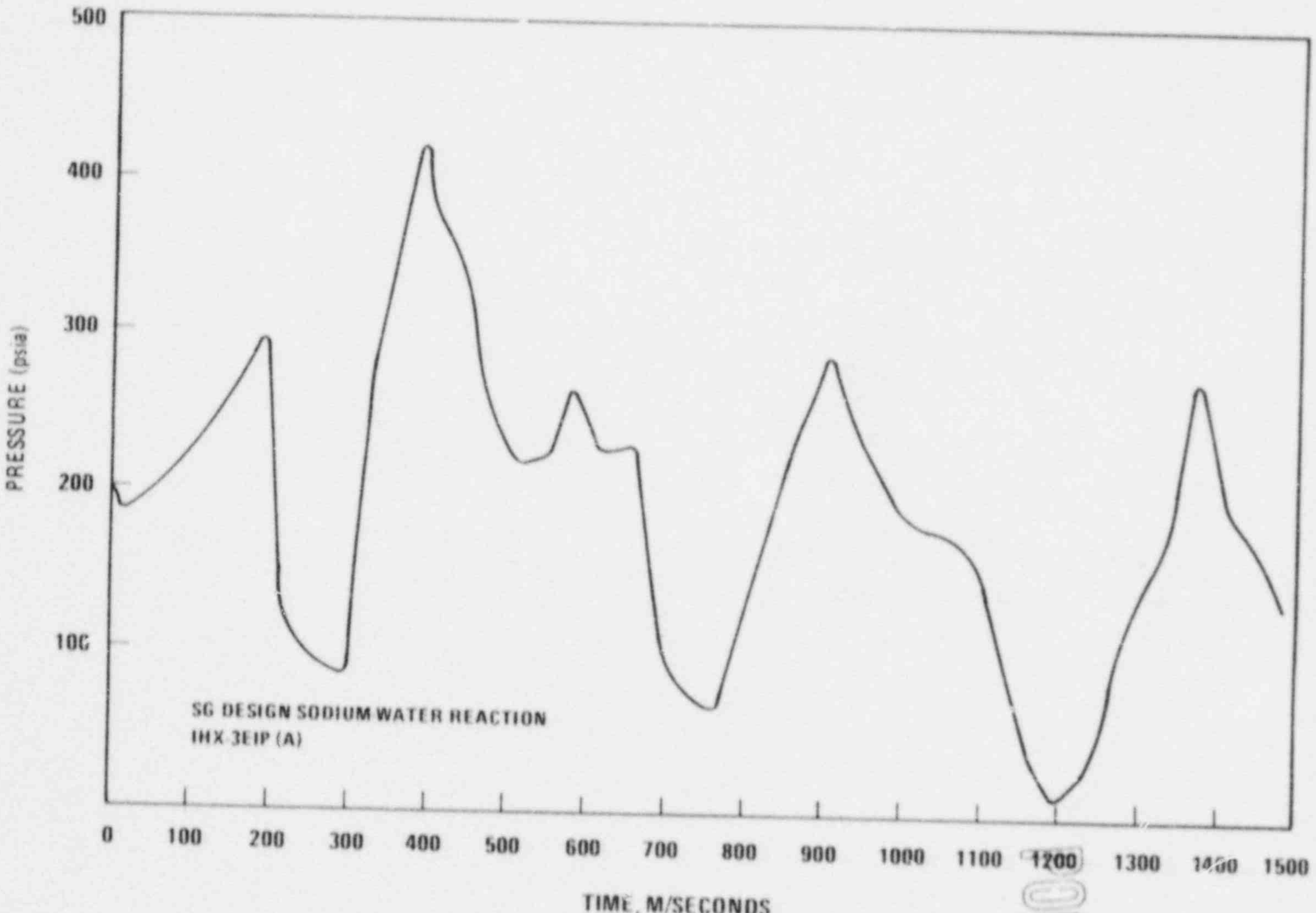
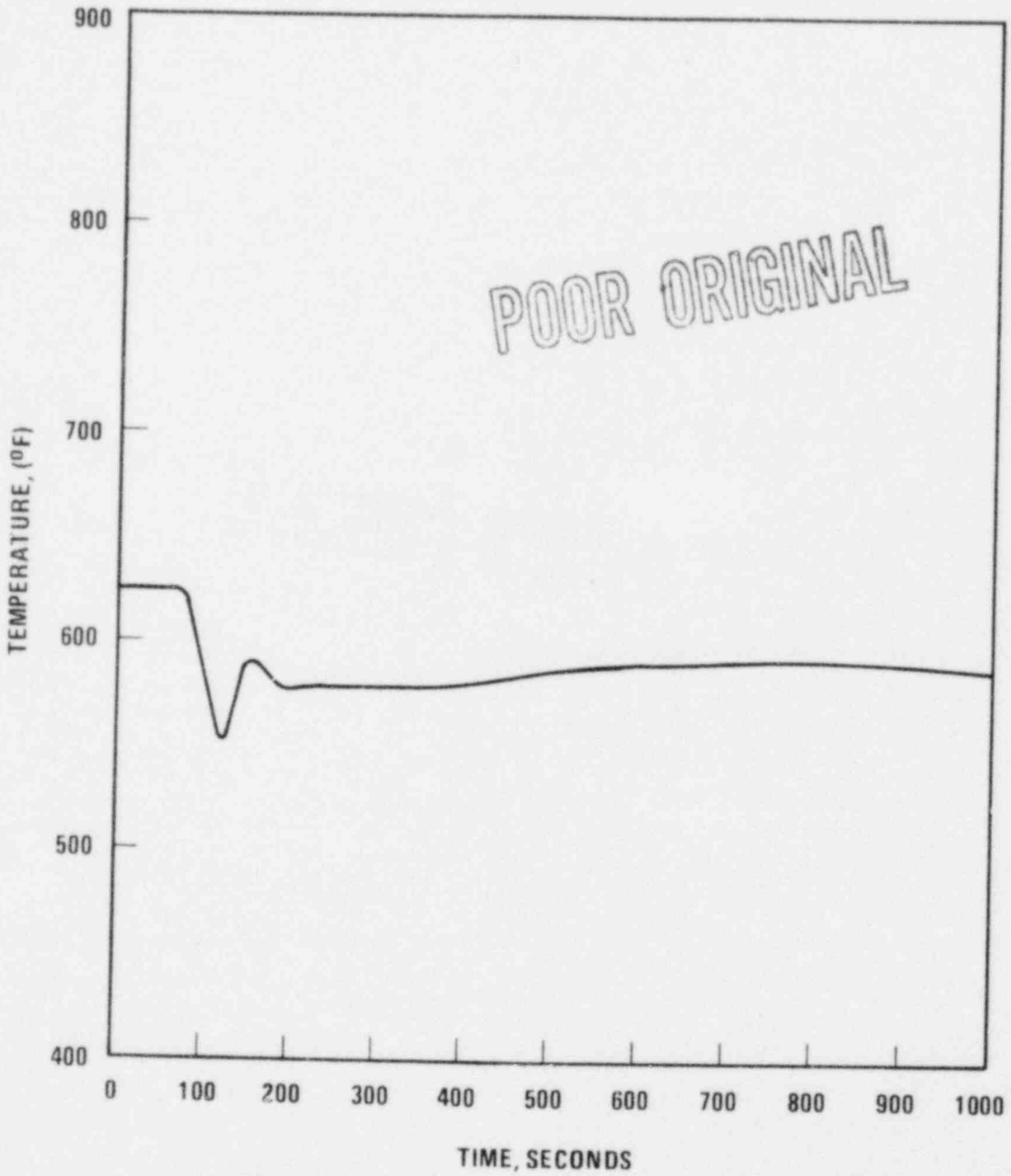


Figure 5.3-18F Intermediate Heat Exchanger, INT IHX Inlet Pressure-vs-Time

7683-170

POOR ORIGINAL

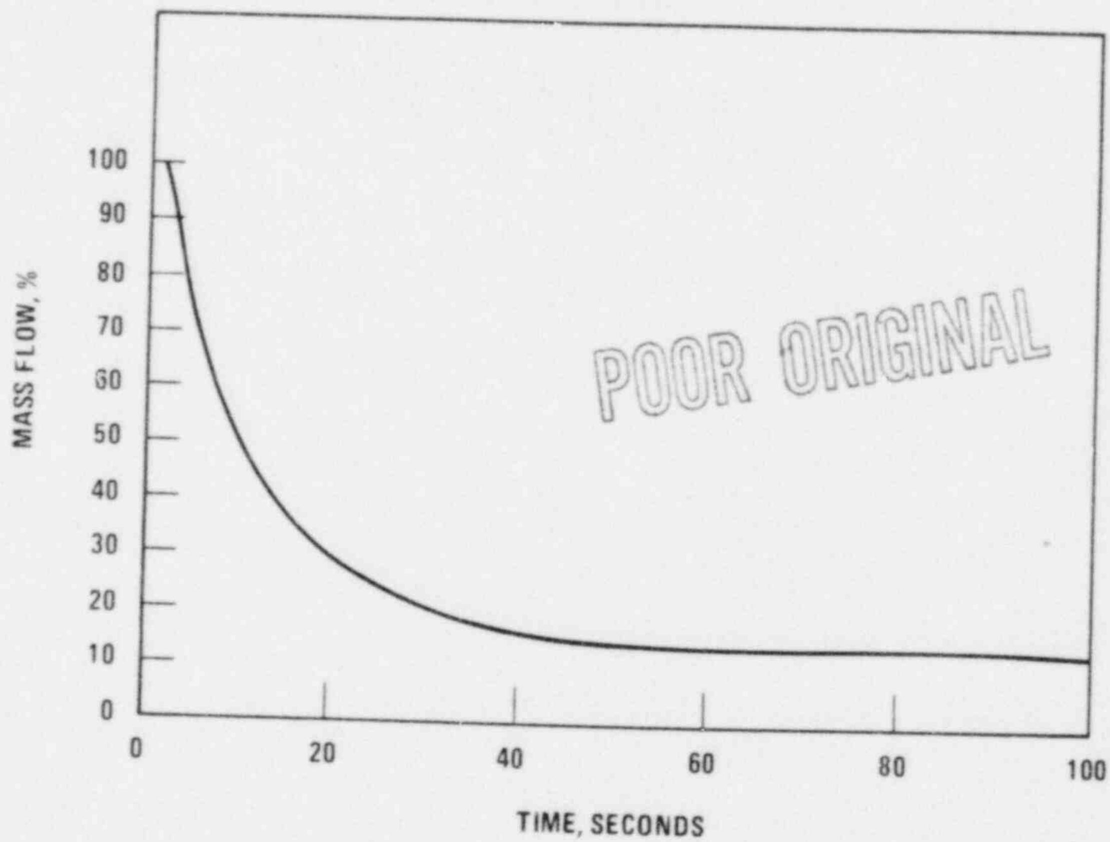


E-1 PRIMARY PUMP MECHANICAL FAILURE.
IHX-6EIT

Figure 5.3-18K Intermediate Heat Exchanger, INT IHX Inlet Nozzle Temperature-vs-Time

7683-175

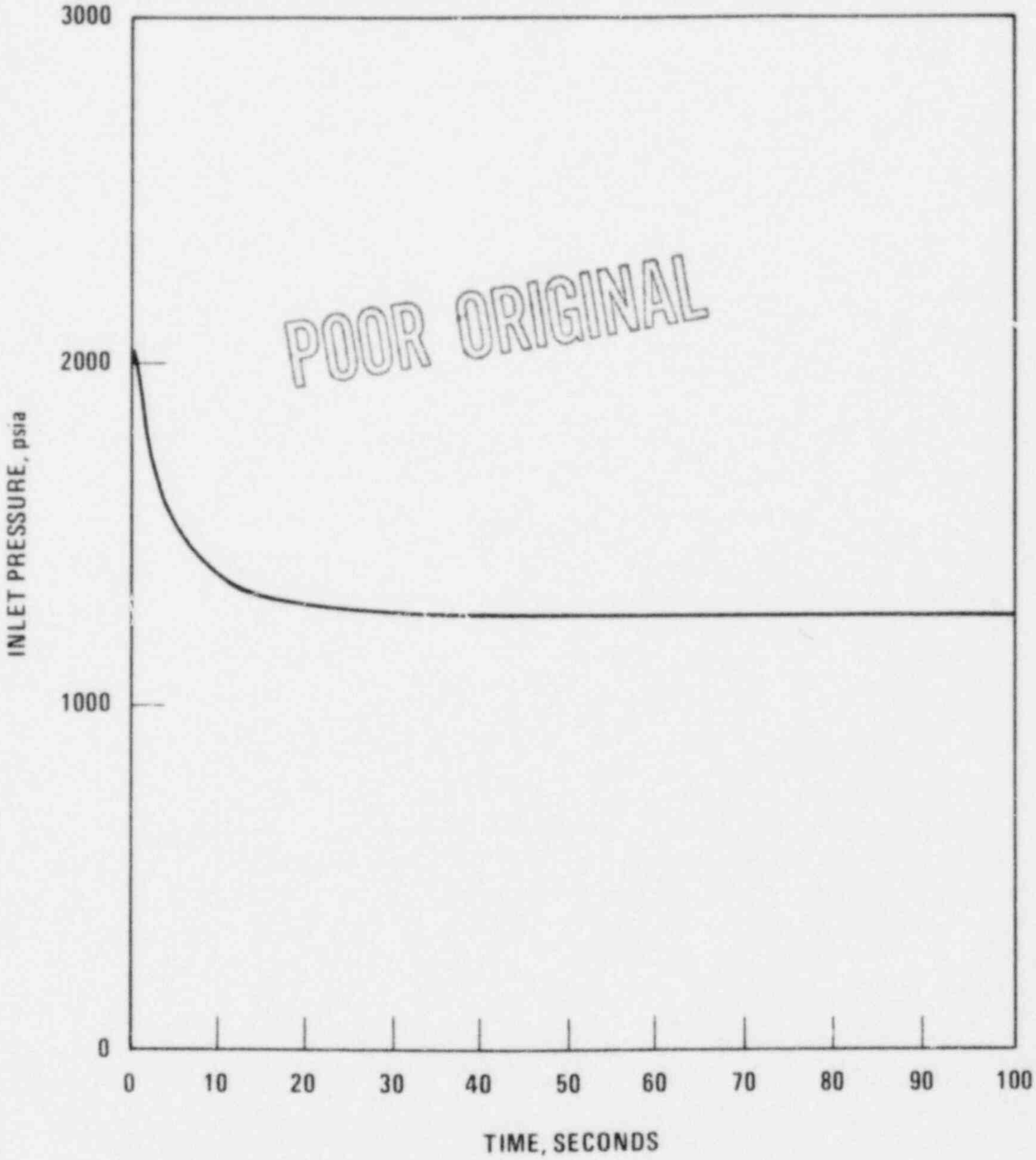
Amend. 16
Apr. 1976



E-1 PRIMARY PUMP MECHANICAL FAILURE. IHX-GEIF

Figure 5.3-18L. Intermediate Heat Exchanger. INT Loop Massflow vs. Time

4786-4



E-1 PRIMARY PUMP MECHANICAL FAILURE. IHX-GEIP

Figure 5.3-18M. Intermediate Heat Exchanger, INT IHX Pressure vs. Time

4786-5

Amend. 59
Dec. 1980

Table 5.3-18n has been intentionally deleted.

b. The system shall be designed such that a normal or upset event does not adversely affect the useful life of any IHTS component.

c. Following an emergency condition, resumption of operation must be possible following repair and re-inspection of the components, except that the intermediate coolant pumps (damaged or undamaged) must maintain capability to provide pony motor flow following all emergency events except in the affected loop for a pump mechanical failure, steam generator leak, or inadvertent dump of intermediate sodium.

d. Following a faulted condition, the intermediate heat transport system must remain sufficiently intact to be capable of performing its decay heat removal function, including maintenance of intermediate coolant pump pony motor flow.

The structural design parameters of the IHTS and individual components are listed in Table 5.4-1, IHTS Design Parameters. The thermal and hydraulic
59 design parameters are given in Table 5.1-1.

Seismic Loads

The intermediate heat transport system components under the jurisdiction of the ASME Code, Section III, Nuclear Power Plant Components, shall be designed to accommodate the load combinations prescribed therein without producing total combined stresses and strains in excess of those allowed in the Code. No component of an individual loading condition shall be included which would render the combination non-conservative. Transient loadings shall be included as required by the Code. For elevated temperatures, Code Case 1592 supplemented by RDT Standard F9-4T will apply.

Details of seismic loading combinations and analysis are provided in Section 3.7.

5.4.1.2 Applicable Code Criteria and Cases

The IHTS pressure containing components shall be designed, fabricated, erected, constructed, tested, and inspected to the standard(s) listed below:

Component	Applicable Standard and Class
IHTS Pump	ASME III, Class 1
IHTS Expansion Tank	ASME III, Class 1
IHTS Dump Valves	ASME III, Class 1
IHTS Piping	ASME III, Class 1
IHTS Flowmeter	ASME III, Class 1
IHTS Thermowell	ASME III, Class 1
IHTS Pressure Taps	ASME III, Class 1
44 33 29 IHTS Dump/Fill and Drain Lines (Downstream of First Valve)	ASME III, Class 2

All intermediate heat transport system components except the dump lines shall be analyzed as Class 1, nuclear components in accordance with the following rules:

29 a. The 1975 Edition of the ASME Boiler and Pressure Vessel Code and addenda through and including Winter, 1975, Section III.

b. ASME Code Case 1592, "Class 1 Nuclear Components in Elevated Temperature Service."

c. RDT E15-2T (Supplement to Section III).

29 d. RDT F9-4T (Supplement to Code Case 1331-8).

The "Liquid Metal Fast Breeder Reactor Materials Handbook", (Ref. 1), shall be used to obtain material properties data not available from the above sources. As required in RDT F9-4, the use of additional or alternative material properties*shall require the approval of the purchaser. Code Case 1521, "Use of H-Grades of SA-240, SA-479, SA-336, and SA-358, Section III," may be used for H-Grades of Type 304 and 316 austenitic stainless steels. RDT-F9-5 (Section 6) provides alternative procedures for satisfying the strain limits of Code Case 1592 which are acceptable to the purchases. Section 6 of RDT F9-5 also provides time temperature limits below which the primary plus secondary and peak stress limits of Section III may be used in place of the limits of Code Case 1592. The scope of the analysis of Code Case 1592 shall be used even if the limits from Section III are used. For example, the primary plus secondary stress intensity range due to Emergency as well as Normal plus Upset Conditions is limited.

In addition, Code Class 1593, for fabrication and installation of elevated temperature components, 1594 for their examination, 1595 for their testing; and 1596 for their overpressure protection shall apply for the intermediate heat transport system components.

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* The term "alternative material properties" refers to the material property data used alternatively to the data of the same property contained in the authoritative sources of ASME Code (Section III and Code Case 1592), RDT Standards (E15-2 and F9-4), and the LMFBR Materials Handbook. The intent of the statement is to make clear that alternative material property data, which may be more conservative or less conservative than the data supplied in the authoritative sources, cannot be used without the approval from the purchaser. The purchaser will only approve the less conservative property data after obtaining permission to use it from the ASME Code and RDT Standards Committees, and will approve the more conservative property data upon valid justification by the user. The data approved in this manner will be incorporated by the purchaser in the design specification for alternative use. 17

5.4.1.4 Material Considerations

5.4.1.4.1 Basis for High Temperature Design Criteria

The basis for high temperature design and analysis of all HTS (PHTS and IHTS) Class 1 components are given in Section 5.3.1.4.

5.4.1.4.2 Materials of Construction

The materials to be used for the intermediate heat transport piping and pump system will be specified the same as the PHTS materials described in Section 5.3.1.4.2 and Tables 5.3-6 and 5.3-7. The valves and expansion tank will utilize specifications shown in Table 5.4-3. The discussion in Section 5.3.1.4.2 on selection of material and specifications apply to all IHTS material.

Selection of alternate materials shall be based on the mechanical properties, metallurgical stability, sodium compatibility and response to radiation under the applicable design and environmental conditions. When recommending the use of alternate materials, the supplier shall document the justification which shall include, as a minimum, a summary of available test or experience data and a discussion of the adequacy of the recommended materials relative to the
36 materials specified in Table 5.4-3.

5.4.1.4.3 Additional Requirements

The additional requirements described in Section 5.3.1.4.3 for the PHTS materials apply to the IHTS materials.

5.4.1.4.4 Welding

The discussion in Section 5.3.1.4.4 concerning the welding of PHTS components shall apply to the IHTS components. The welding filler materials and specifications are given in Table 5.4-4. Welding of the trimetallic transition joints (ferritic - Alloy 800H - austenitic with ErNiCr-3 as the weld filler material between ferritic and Alloy 800H, and 16-8-2 stainless steel filler
59 between Alloy 800H and austenitic) is covered in Section 5.5.3.11.2.

36

5.4.1.5 Leak Detection Requirement

The IHTS leak detection subsystem (which is part of the Leak Detection System discussed in Section 7.5.5) will provide indication and location information to the operator in the event of a sodium leak from the IHTS to a cell atmosphere.

Leakage from the intermediate system to the primary system will be detected by volume changes within the IHTS as discussed in Section 7.5.5.2. Leakage from the primary system to the intermediate system which may occur under accident conditions will be detected by a radiation monitoring system as discussed in Section 7.5.5.2. 30

The leak detection system sensitivity requirements are discussed in Section 7.5.5.

5.4.1.6 Instrumentation Requirements

The intermediate system is provided with an instrumentation system which monitors the process variables within the IHTS and which
49 provides signals for safety action and operational information and control. The measured variables and instrumentation provided are discussed in Section 7.5.2.

5.4.2 Design Description

5.4.2.1 Design Methods and Procedures

5.4.2.1.1 Identification of Active and Passive Components which Inhibit Leaks

Table 5.4-5 lists those active and inactive components of the IHTS and their normal operating mode.

29 The IHTS is composed of three independent and physically separated loops, contained within individual cells. Each cell is designed to contain the maximum expected sodium spill and is designed such that the adjacent loops are not affected by a disabled loop.

In the event of a pipe leak in an operating loop, the IHTS is designed to provide shutdown and decay heat removal capabilities with one of the two remaining loops.

Small pipe leaks in the IHTS can be detected by sodium leak detectors. Leak detection ability allows for operator action to manually shut down the plant and drain the affected loop. The leak detection system is described in Section 7.5.5.

A pipe leak in the IHTS would result in a level decrease in the expansion tank and IHTS pump, which activates an alarm in the control room. The operator can then take appropriate action. In addition, a reactor scram can be initiated by the primary to intermediate flow ratio and pump speed ratio trips and by the high primary cold leg temperature trip dependent upon the location and size of the break. These events are discussed in more detail in Section 15.3 and 5.5.3.6.

5.4.2.1.2 Design of Active Pumps and Valves

The IHTS sodium pumps will be designed, analyzed, manufactured, tested, and shipped as described for the PHTS in Section 5.3.2.1.2.

5.4.2.1.3 Surveillance and Inservice Inspection Program

59 An inservice inspection program for the IHTS will be implemented and conducted in accordance with the intent of the ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear 1 Reactor Coolant System. The inservice inspection program will include all IHTS components such as pressure vessels, piping, pumps and valves.

To facilitate the inspection program, it is a design goal that all IHTS sodium welds be accessible for inspection after insulation and heater removal. Where necessary, hand held optical aids or remote devices such as periscopes will be used for inspection.

5.4.2.1.4 Protection Against Accelerated Corrosion and Material Degradation

The materials of construction of the IHTS are protected against stress corrosion and intergranular corrosion from purchase through the operating life of the plant using the same methods as employed for the PHTS described in Section 5.3.2.1.5.

5.4.2.1.5 Material Inspection Program

The material inspection program for the IHTS is the same as that identified for the PHTS in Section 5.3.2.1.5.

5.4.2.2 Material Properties

The coolant boundary sections of the intermediate heat transport system are
fabricated from unstabilized austenitic stainless steels (Type 304, Type 304H
59 and Type 316H), ferritic steel (2-1/4 Cr - 1 Mo), and Alloy 800H. Properties
of the austenitic stainless steels used in the intermediate heat transport
system are similar to those described in Section 5.3.2.2 for the primary heat
59 transport system. Properties of the ferritic steel and Alloy 800H used in the
intermediate heat transport system are similar to those described in Section
36 5.5.3.11 for the steam generator system.

5.4.2.3 Component Descriptions

5.4.2.3.1 Intermediate Coolant Pumps

The intermediate sodium pumps are free surface, single stage vertically
mounted, drawdown type centrifugal pumps driven by a variable speed 5000 Hp
29 squirrel cage induction motor. An auxiliary 75 Hp pony motor on each pump
provides low flow capability (7.5 percent) for decay heat removal and other
low power, hot standby conditions. Variable pump speed is achieved by the main
drive motor supplied with variable frequency power from a fluid coupled MG set.
The intermediate pump will be operated to provide a flow of 29,500 GPM of 651°F
33 sodium at a 336 foot head.

The intermediate pump is identical to the primary pump (Section 5.3.2.3.1) except for the following significant differences in requirements.

- o No radiation shielding is required.
- o Head and flow requirements are different.
- o Transition piping is required to connect the 36 inch pump suction nozzle to the 24 inch IHTS cold leg piping.
- o The intermediate pump does not require a stand pipe-bubbler for sodium level control.
- 99 o The argon and oil vapors are vented from the seal leakage reservoir to
33 the atmosphere rather than to RAPS.

The design envelope for the intermediate pump is shown in Figure 5.4-1.

It is a design basis that the PHTS pumps operate with 10" W.G. (0.36 psig) cover gas pressure. Since the PHTS and IHTS pumps are being
49 procured as identical units, the IHTS pumps could also operate with essentially atmospheric cover gas pressure without cavitation even though this is not a design basis. This type of IHTS operation is not anticipated, however, because the IHTS pump cover gas pressure requirement is 100 psig during normal operation (10 psig at pony motor flow) based on the requirement to maintain a positive pressure on the IHTS side of the IHX with 25
49 respect to the PHTS side of the IHX.

Section 5.4.1.1 specifies performance requirements of the IHTS following a faulted condition. These are applied to the pumps as follows:

The IHTS coolant pumps are required to remain operable only at pony motor speed following all system-related emergency conditions and the faulted conditions. For pony motor operation, the IHTS pumps require (1) unimpaired pump shaft rotation, (2) a physically intact shaft seal lube oil system (lube oil pump operation not required; however, lube oil containing boundary must retain its integrity, (3) continuous lube oil cover gas supply, and (4) uninterrupted power supply to the pony motors. These functional requirements on the pumps are in addition to the requirement that the IHTS sodium boundary retain its integrity. The decay heat removal function can be accomplished with as few as one of
35 the three IHTS loops operating. 25

Free surface level changes in the IHTS pumps are significant only for system temperature changes. The minimum sodium level corresponds to the level of
29 sodium in the system after filling at 400°F. As the system is heated to operating temperatures, the sodium level rises about 3' 3". The expanded volume of sodium is contained in both the pump tank and a connecting intermediate sodium expansion tank. An additional free volume of 1' 11" above the normal liquid level is provided to accommodate potential abnormal operating conditions.

The kinetic energy of the total rotating pump mass at 960 rpm following an intermediate pump trip is as follows:

Pump (impeller, bearing, and shaft)	487,000 ft-lbs
Motor	4,100,000
33 Total estimated kinetic energy	4,587,000 ft-lbs 16

Design techniques used in the prevention of fracture-type failure and prevention of oil leakage in the Intermediate Coolant Pump are the same as those identified for the Primary Coolant Pump in Section 5.3.2.3.1.

59 The IHTS sodium pumps are designed to withstand the loadings associated with (1) the extremely unlikely plant condition occurring from a design basis leak in a steam generator, and (2) the Safe Shutdown Earthquake (SSE). The analyses
59 required to demonstrate this treats the intermediate pump and its hydraulic assembly as a Class 1 component in accordance with the rules of the ASME Code Section III and modified RDT Standards as defined in Section 5.4.1.1.

The preliminary design basis sodium-water reaction produces a pressure transient at the IHTS pump suction nozzle. This transient is described in Section 5.5.3.6. The calculated pressure transient arriving at the
59 intermediate pump tank is considered an emergency condition for the purpose of evaluating the pump tank.

Dynamic analyses of the pump tank was performed to determine the structural response. Simplified models such as beams, frames, and plates were used if results could be shown to be conservative. Finite element shell models were used if simplified models could not be shown to give adequate results.

In the final code analysis for the stress report, if the pressure load is a transient, it and the dynamic load will be treated as time dependent dynamic loadings. This analytical method may amplify or mitigate the stresses. In
59 either case, the dynamic analysis will be used in the final design. 19

Stress analysis of the pump tank will be performed according to the rules of ASME Code Section III (see Sections 5.4.1.1.1 and 5.3.2.1.2 19
for applicable classification and Code Cases and RDT F9-4). Stresses due
to thermal transients occurring simultaneously will be combined with the 33
primary load set. The IHTS pump tank will be designed to meet the limits of ASME Code Section III, 1974

59 Edition up to and including Winter 1974 addenda, applicable to emergency conditions, for the combined load set. The effects of cyclic pressure loading during the transient will be included.

5.4.2.3.2 Expansion Tank

An expansion tank is provided in each IHTS loop to accommodate the sodium volume change in excess of that accommodated in the IHTS pump tank, due to the thermal expansion associated with normal and off normal conditions. The pump tank and expansion tank cover gas volumes are connected by a gas service line. The expansion tank sodium line is connected to the main IHTS cold leg piping just upstream of the IHTS pump suction nozzle.

5.4.2.3.3 IHTS Piping and Support

The IHTS piping conducts sodium in a continuous loop to transport reactor heat to the Steam Generation System. An isometric drawing of the IHTS piping and components in a typical steam generator cell is shown in Figure 5.4-2.

Each IHTS piping run is provided with the necessary elbows, tees and reducers to provide adequate loops for thermal expansion. While each loop contains the necessary appendage piping and associated fittings, there are no valves in the main sodium flow piping. Each loop has similar components, although the loop piping differs in length and configuration because of the differences in distance between the IHX units and steam generator modules.

The hot leg piping is 24" OD x 1/2" wall Type 316H stainless steel and extends from the IHX outlet nozzle through the reactor containment penetration to the superheater inlet nozzle. From the superheater outlet nozzle, the piping consists of two parallel runs of 18" OD x 9/16" wall 2-1/4 Cr - 1 Mo ferritic steel and extends to each of the two evaporator inlet nozzles. The cold leg piping commences at the evaporator outlet nozzles and consists of two parallel runs of 18" OD x 1/2" wall Type 304H stainless steel pipes. The lines are joined together through expanders at a 24" x 24" x 24" Type 304 H mixing tee and continue as a single run of 24" OD x 1/2" wall Type 304H stainless steel pipe which is connected to the 36" diameter IHTS coolant pump suction through a seven foot long diffuser. The cold leg continues as a 24" OD x 1/2" wall Type 304 H stainless steel pipe from the pump discharge through the reactor containment penetration and completes the loop at the IHX inlet nozzle. The Type 316H and Type 304H stainless steel pipes are joined to the 2-1/4 Cr - 1 Mo
59 ferritic steel superheater and evaporators, respectively, through Alloy 800H
35 transition spool pieces.

The auxiliary IHTS piping includes appendage piping for instrumentation, system high point vents and low point drains, fill lines, sodium dump lines and service connections for sodium purification.

The IHTS piping except for the dump lines, will be designed in accordance with Section III, Class I of the ASME Boiler and Pressure Vessel Code. The dump/fill and drain lines downstream of the first dump valve and the gas
29 equalization line downstream of the first valve and the rupture disc assembly
49 will be designed in accordance with Section III, Class 2. The material used in piping is in accordance with the applicable code and will be of all welded construction.

All IHTS piping and components, except the dump /fill and drain
piping downstream of the first dump valve in series, the gas equalization
line downstream of the first valve in series and the gas equalization by-
44 pass line downstream of the rupture disc assembly are required to be
Safety Class 2 as a minimum and therefore, ASME Boiler and Pressure Vessel
Code, Section III, Class 2, as a minimum; however, they are being
optionally upgraded and will be designed, constructed and code stamped in
44 accordance with ASME B&PV Code, Section III, Class I. The dump/fill and
drain piping downstream of the first valve in series, the gas equalization
line downstream of the first valve in series, and the gas equalization
by-pass line downstream of the rupture disc assembly are required to be
Safety Class 3 as a minimum, and therefore, ASME B&PV Code, Section III,
Class 3 as a minimum; however, they are being optionally upgraded and will
be designed, constructed and code stamped in accordance with ASME B&PV
35 Code, Section III, Class 2.

The information is an approximation and is based on pump vendor test data from similar pump designs and on data from Reference 5.

Since the intermediate pump hydraulics are identical to the hydraulics of the primary pump, the locked rotor characteristics are as shown in Figure 5.3-20. The following expressions for intermediate pump locked rotor impedance are based on Figure 5.3-20:

Forward flow $\quad H = 184\,\bar{\omega}^2$

Reverse flow $\quad H = 235\,\bar{\omega}^2$

where, H = pressure drop through pump, feet

$\bar{\omega}$ = normalized flow, i.e., the ratio of actual system flow to design flow

59 The pump drive system incorporates separately powered pony motors for 1
circulating intermediate sodium at low flow rates during startup, testing and stand-by operations following accident conditions. Basically, the requirement 33
is to provide approximately 7.5 percent flow (2210 gpm) at 400°F to 650°F at a
29 head of about 3.0 feet of sodium. Following reactor trips, the pony motors will maintain flow for decay heat removal.

There will be no vortexing or gas entrainment at flows in the operating range of the IHTS pumps. All parts which need to be submerged during operation at
59 pony motor speed or restart to pony motor speed are located below the minimum sodium level.

The oxygen concentration in the intermediate system sodium will be maintained
59 below 2 ppm during normal operation. This level of sodium impurity does not affect the pump operating characteristics.

Pump Integrity

The IHTS pump shaft supporting assembly and inner structure are supported on the pump tank mounting flange at the IHTS pump support structure. Circumferential bolts secure the pump tank head to the supporting assembly and 33
inner structure which is in turn secured in a similar fashion to the pump tank. The pump tank is circumferentially bolted to the support structure.

59 43 For the IHTS Pump SSE analysis, a 2% damping value was used. The SSE loading
is considered to occur in conjunction with a plant trip. Following the SSE,
the Intermediate Heat Transport System, Steam Generation System, and Steam
Generator Auxiliary Heat Removal System must provide for removal of stored and
59 decay heat. The IHTS pumps are designed to maintain pony motor flow without
loss of structural integrity after SSE. Computer programs, such as SAP IV and
59 ANSYS, were utilized to perform seismic analyses on the intermediate pumps.
Descriptions of these computer programs can be found in Appendix A of the PSAR.
59 The intermediate coolant pump and its hydraulic assembly are treated as a
Class 1 component for the seismic analysis.

5.4.3.4 Valve Characteristics

There are no valves in the IHTS main sodium piping.

5.4.3.5 Evaluation of Steam Generator Leaks

The large sodium water reaction evaluations covering a range of leak sizes are given in Section 5.5.3.6. Large sodium water reactions generate acoustic pressure pulses which propagate through the IHTS and also result in long term pressurization of the IHTS. The IHX has been identified as the critical component in the IHTS for these overpressure effects since the IHX tubes are the boundary between the primary and intermediate sodium. For the entire range of large leaks considered in Section 5.5.3.6, peak pressures in the IHX do not result in primary stress levels exceeding the stress allowable for emergency conditions as given in the ASME Code, Section III.

5.4.3.6 IHTS Coolant Boundary Integrity

The integrity of the IHTS will be evaluated in a manner similar to that described for the PHTS in Section 5.3.3.6.

The highest quality of engineering, fabrication, installation, and inspection will go into the IHTS piping. The IHTS piping is a Class 1 item and will require a detailed stress analysis as required by the ASME Code.

The design basis and analyses as described in Section 5.3.3.6 for PHTS piping have direct application for the IHTS piping within the inerted HTS cells of the Reactor Containment Building. That is, the analysis methods and computer codes, the fracture mechanics analysis, the corrosion effects and sodium leak detectability will be considered as for the PHTS piping and the coolant
59 | boundary integrity.

For the IHTS sodium piping in an air environment within the Reactor Containment Building Intermediate Bay and the Steam Generator Building, the approach for proving pipe integrity as presented in Section 5.3.3.6 is applicable with supplemental consideration of corrosion effects and leak detection capability. Separate discussions on corrosion rates and sodium leak detection for the IHTS piping within the air atmosphere are presented in Sections 1.5, 5.4.3.6.3 and 7.5.5.

5.4.3.6.1 Design and Quality Assurance

5.4.3.6.1.1 Design Assurance

As for the PHTS piping, the main IHTS piping will be designed as an ASME Code Class 1 system and the applicable Code Sections, Addenda, Code Cases, and RDT Standards will be used as the design bases. Detailed loads and resulting stresses will be obtained for each segment and component of the piping system. The detailed results of the stress analysis will enable a comprehensive assessment of the structural capability of the IHTS piping, and will be issued as a formal stress report as required by the ASME Code.

Stress Analysis Methods

The stress analysis procedures to be used in the integrity analysis of the IHTS are those as described in Section 5.3.3.6.1.1 for the PHTS piping. The same or equivalent computer codes will be used to ensure that the stress and strain evaluation of the IHTS piping is of sufficient detail to demonstrate structural integrity.

Piping System Design Basis

The bases and criteria for design of the IHTS piping system will be included in the HTS Piping Design Specification. The Piping Design Specification will be for the complete HTS system which includes both the PHTS and IHTS piping systems. This will ensure that the details of the design criteria, loadings and conditions for design of the IHTS will be of the same detail as for the PHTS piping.

The operating conditions for the IHTS piping are similar to those discussed for the HTS piping in Section 5.3.3.6.1.1. The IHTS piping system is a low pressure system operating in the temperature range of $650^{o}F$ to $965^{o}F$. Due to the low pressures, (<225 psia) the system is constructed of thin-walled piping operating at low primary (or load-controlled) stresses.

The highest stressed piping components of the intermediate system are in loop 1. The loop 1 hot leg (reactor containment penetration to superheater inlet) has the highest thermal expansion stresses. The elbows, the SWRPRS tee and the 2 1/4 Cr-1 Mo stainless steel transition piece are the critical items in the pipe line. The integrity of this piping during the service lifetime of the plant is ensured by stringent quality assurance, conservative design practice, and leak detection equipment which would identify small through-wall leaks and an inservice inspection program, following the intent of Section XI of the ASME
59 Code, Rules for Inservice Inspection of Nuclear Power Plant Components. 25

The specified loadings on the IHTS piping will be categorized into design, normal, upset, emergency and faulted conditions and organized into a load histogram. The design stress analyses will be performed using these loading categories and histograms. Pressure surge, vibration and temperature fluctuation effects will be assessed with the usual effects of internal and external pressure, deadweight, support reactions, thermal expansion, seismic and thermal transient gradients. As specified by Regulatory Guide 1.48, the IHTS piping will be designed for two levels of seismic acceleration spectra, one for an Operating Basis Earthquake and one for a Safe Shutdown Earthquake.

The piping stress analysis will be performed using verified and documented computer programs. Detailed stress maps for each IHTS piping segment will be obtained using the computer code ELTEMP which is described in Section 5.3.3.6.1.1. These stress maps will be the basis of the fracture mechanics analysis to be carried out on the IHTS piping elbows and components discussed in Section 5.4.3.6.2.

5.4.3.6.1.2 Quality Assurance

The quality assurance program for the IHTS system piping, in conformance with RDT Standard F2-2, is designed to provide the highest possible system integrity. Fabrication, installation, and inspection procedures, in accordance
31 29 with Class 1 requirements of Section III of the 1975 Edition of the ASME B&PV Code, will be used as well as supplemental requirements as imposed by applicable RDT Standards to assure that the IHTS piping will be able to function properly and safely throughout the projected service life of the plant. A detailed discussion of the quality assurance procedures is provided in Chapter 17.0.

Piping Fabrication

29 59 The IHTS hot leg (IHX-to-superheater) will be made from 24 inch O.D, 0.50 inch nominal wall thickness, Type 316H stainless steel, ASME Class 1 welded pipe, according to design specifications and the supplementary requirements imposed by RDT Standards M3-7T and M2-5T. In accordance with these two standards, the weld filler material will conform to RDT Standards M1-1T and M1-2T.

The IHTS cold leg will be made from 18 inch O.D. (evaporators-to-mixing tee) and 24 inch O.D. (mixing tee-to-intermediate pump-to-IHX), 0.50 inch nominal wall thickness, Type 304H stainless steel, ASME Class 1 welded pipe. The applicable RDT Standards for the hot leg piping will be applied to the cold piping also.

The piping between the superheater and evaporator modules is 18 inch O.P., 0.562 inch nominal wall thickness, 2 1/4 Cr-1MO ferritic steel, ASME Class 1. 29
Other aspects of the piping fabrication discussed in Section 5.3.3.6.1.2 for the PHTS piping will be the same for the IHTS piping system.

Mixing Tee Fabrication

The IHTS mixing tee joins the outlet sodium flows from the two evaporators before the flow enters the IHTS pump. The tee geometry was tested at ANL and
59 ORNL. The development program was aimed at optimizing the design of the tee to assure long life and reliable service. 29
The basic design is a tee 24 x 24 with two 18" - 24" expanders located upstream from the tee in the 18" diameter lines coming from the evaporators. The material is 304HSS. Development work was 29
59 required for the mixing tee since under transient operation, outlet temperatures may vary between the two evaporators. The mixing tee must be able to accommodate this temperature fluctuation within fatigue limits. 2

Elbow Fabrication

The elbows for the IHTS piping system will be procured to RDT Standard M2-51 and will be of welded, stainless steel, Types WP304 and WP316 materials. Other 29
aspects of the elbow fabrication discussed in Section 5.3.3.6.1.2 for the PHTS elbows will also be applied to the IHTS piping elbows.

In-Service Inspection

In-service inspection shall be performed on the components of the IHTS to provide a continuing assurance that these components can perform their functions safely. The examinations will include continuous monitoring by liquid metal-to-gas leak detection systems and visual inspections of the IHTS components. Dissimilar metal welds will be inspected by volumetric methods. Hangers and snubbers will be visually inspected. Snubbers will also be
59 functionally testad to assure continued operability.

5.4.3.6.2 IHTS Piping Failure Potential Due to Fatigue Crack Growth

In this section a postulated defect much larger than the QA allowable size was analyzed for fatigue growth. This analysis employed the materials properties and applied stresses typical of those in the IHTS piping system along with the postulated crack size. The defect was assumed to exist in a highly stressed elbow in the hot leg of IHTS piping. A preliminary estimate was made of the fatigue crack extension of the hypothetical flaw. As the analysis of the IHTS piping system progresses, other piping locations will be examined in detail using the methods presented here.

Along with crack growth considerations, preliminary calculations were made to determine the critical crack size for elbows in each leg of the IHTS piping system. These crack sizes were calculated assuming through-the-wall flaws with operating and design pressures. The purpose was to show that starting with a flaw that is much larger than any flaw allowed by the RDT Standards and applying the cyclic load history, the flaw will not grow to the critical crack size.

Based on preliminary results obtained from the analysis of the one elbow using the analytical procedures described in 5.3.3.6, the crack extension is negligible and the critical crack length will not be. It is expected that the analyses to be done on the other sections of IHTS piping will also result in the conclusion that the growth of a postulated large defect would be negligible and that the potential for pipe failure is negligibly small. Details of the crack growth calculations for the IHTS elbow are given below.

The crack growth calculations for the IHTS elbow are representative of the IHTS austenitic steels (304, 304H and 316H) only and are not necessarily applicable to the ferritic steel (2-1/4 Cr - 1 Mo) and Incoloy materials. Crack growth
35 will be investigated for these additional materials.

Material Behavior Considerations

The materials properties data given in Figures 5.3-26 thru 5.3-28 are representative of the behavior in a sodium or air environment of the IHTS piping material, as discussed in Section 5.3.3.6 for the PHTS piping, and may be used for IHTS crack growth calculations. This data can be conservatively used for both IHTS Types 304 and 316 stainless steel.

Stress Analysis Considerations

The loadings to be considered in the final crack growth analyses for all the sections of the IHTS piping will include all anticipated normal, upset and emergency events of stress magnitude sufficient to cause fatigue-crack extension. The preliminary estimate of crack propagation for the most-highly-loaded hot leg elbow is based on the loadings for the reactor scram event applied for the total number of expected upset and emergency events expected during the life of the plant. This event is described in Figure 5.4-6 and consists of a transition from refueling at 400°F to normal operating temperature (965°F) followed by a scram which returns the system to 400°F. This event was selected for the preliminary crack growth analysis, since it is one of the most severe upset events and it makes up more than 400 of the 600 events to be considered. The final analyses for crack growth calculations for the IHTS piping system will consider each loading cycle separately for each highly stressed region in the IHTS piping system.

Analyses for the load cycle of Figure 5.4-6 have identified the hot leg elbow adjacent to the HTS cell penetration to be the region of highest stress from a fatigue standpoint. Circumferential and axial stress components were calculated for the inside surface, midplane and outside surfaces at various points around the circumference. The maximum values are given in Table 5.4-6 (also see Figure 5.3-30). These values are used for a preliminary estimate of crack propagation.

In general, it was found that the circumferential stress components are greater than the axial components. Therefore, the analysis assumed the existence of an axial crack (acted upon by circumferential stresses) on the inside elbow surface at the location of maximum circumferential (hoop) surface stress in the middle of the analyzed elbow. The stress values used in the calculation of the crack growth are given in Table 5.4-6.

Other aspects of the stress analysis considerations for crack growth predictions are discussed in Section 5.3.3.6 for the PHTS piping system and are directly applicable to the IHTS piping system.

The preliminary tensile stresses of Table 5.4-6 are below the 0.2% offset yield strength (approximately 17,200 psi for Type 316SS at 965°F). Therefore, the use of linear-elastic fracture mechanics in this case is justified.

Initial Crack Size Considerations

The postulated initial defect was hypothesized to have a length (2c) of 1.5 in. and a depth (a) of 0.125 in. This is considerably larger than the size of the largest crack that is estimated might be missed by non-destructive evaluation which has 2c = 1.0 in. and a = 0.015 in. The postulated crack is illustrated schematically in Figure 5.3-31 and the ratio of crack depth to nominal wall thickness (a/t) is 0.250.

	18-in. O.D.	24-in. O.D.
Ult. Strength	54,000 psi	54,000 psi
Flow Stress	28,720 psi	28,720 psi
Operating Pressure	127 psi	225 psi
Design Pressure	325 psi	325 psi
Pipe Thickness	0.4375 (Min.)	0.4375 (Min.)
Pipe Mean Radius	8.75 in.	11.75 in.

Using the parameters listed above, assuming an axial crack, and using the model results of Figure 5.3-36, critical through-the-wall crack sizes were determined for the hot and cold legs of the main coolant IHTS piping at operating pressure and the design pressure, respectively. These critical crack sizes (a_{cr}) are listed in Table 5.4-7.

The margin of safe life for the IHTS piping can be evaluated by taking the assumed crack size (established by either inspection limits or hypothesized), adding on the expected crack growth and comparing it to the critical crack size. In addition, crack extension due to sodium-leakage-induced corrosion should be considered. In equation form, this may be stated as follows;

Initial Crack Size

$$\left.\begin{array}{ll} \text{(a)} & \text{Determined by} \\ & \text{inspection limits} \\ & \quad\text{or} \\ \text{(b)} & \text{Hypothetical} \\ & \text{size} \end{array}\right\} + \begin{array}{l}\text{Fatigue Crack}\\ \text{Growth}\end{array} + \begin{array}{l}\text{Crack extension}\\ \text{due to corrosion}\end{array} \leq a_{cr}$$

For the case of an inert atmosphere the margin of safe life for piping is very large because, (1) the critical through-the-wall crack is several times larger than the postulated crack which was shown to grow a negligible amount during the service life and (2) if a through-the-wall crack or leak could develop, the leaking sodium would be detected by the sensitive leak detection system before the crack could reach critical crack size (see Section 7.5.5 for a discussion on the leak detection system).

For the case of the IHTS piping in the air environment, crack extension due to sodium-leakage-induced corrosion must be considered. Section 5.4.3.6.3 discusses sodium induced corrosion in an air environment and the testing planned to quantify it. In any case, the margin of safe life for the IHTS piping in the air environment has been shown to be large because (1) the postulated crack would not grow to a through-the-wall crack during the plant service life and (2) the critical crack size is so large that a through-the-wall crack, enlarged by sodium, will produce such a large sodium spill that the leak detection system will detect the leakage before further significant crack growth could occur.

5.4.3.6.3 Sodium-Leakage-Induced Corrosion

If it is postulated that a through-the-wall crack could occur in the IHTS piping system, then sodium leakage could occur. Section 7.5.5.1 describes the several diverse leak detection systems that will provide alarms for a wide range of leakage rates. Experimental investigations characterizing sodium leaks to inerted gas and air environments are also discussed in that section.

In order to demonstrate that a small sodium leak does not create a safety related problem and to show that there is a large margin in the design against potential adverse consequences of small leaks, experiments as described in Section 1.5 are being conducted to investigate the consequences of small leaks in an inerted gas environment and an air atmosphere. A major goal of these experiments is the measurement of the rate of corrosion of austenitic stainless steel pipe due to the reaction products of sodium, gas and air environments.

The experiments to date have been carried out at FFTF operating conditions. Section 1.5.2 discusses the experiments being done at CRBRP operating conditions.

The available data are discussed in Section 5.3.3.6.3. For the case of sodium leaking into an inerted gas environment, the resulting corrosion rate of the stainless steel pipe at FFTF operating conditions is small amounting to the order of one-thousandth of an inch per month. It is believed that similar corrosion rates will be demonstrated for CRBRP conditions for the case of the PHTS and IHTS piping in the inerted gas environment.

Additional corrosion tests identified in Section 1.5.2 are in progress to provide additional data for both the cases of the CRBRP inert gas environment and an air atmosphere to better define the processes involved.

Based on the data described in Section 5.3.3.6.3, the sensitivity of the leak detection system discussed in 7.5.5.1, and the magnitude of critical crack sizes determined in the previous section, it concluded, subject to verification by the tests described in Section 1.5.2, that more than one of the several leak detection methods for monitoring the IHTS piping system would detect the leak before it could grow to a large size.

The corrosion data for austenitic stainless steels are not necessarily
59 applicable to the ferritic steel (2-1/4 Cr - 1 Mo) and Alloy 800H material used
35 in the IHTS. Corrosion will be investigated for these additional materials.

5.4.3.7 Inadvertent Operation of Valves

The major IHTS piping contains no valves. The valves listed in Table 5.4-5 are discussed in other sections as noted in the table.

5.4.3.8 Performance of Pressure-Relief Devices

Devices used to protect the IHTS from overpressurization are evaluated in Section 5.5.3.9.

5.4.3.9 Operation Characteristics - Design Transients

The overall plant duty cycle list, which includes the event classification according to the ASME Section III categories of normal, upset, emergency faulted, and the event frequencies are given in Appendix B. Events in the plant duty cycle list with similar thermal transient characteristics for the IHTS components have been grouped under a reduced number of "umbrella" transients to reduce the amount of structural analysis required. The umbrella transients for these IHTS components will be used to construct component histograms for use in the structural analysis.

The design loading combinations and the associated stress or deformation limits to be used are those specified in the ASME Section III, Nuclear Component Code in the following figures:

A. Figure NB-3222-1, Stress Categories and Limits of Stress Intensity for Normal and Upset Operating Conditions.

B. Figure NB-3224-1, Stress Categories and Limits of Stress Intensity for Emergency Conditions.

5.4.3.10 Material Considerations

The material considerations for the austenitic stainless steel portion of the IHTS are the same as those for the PHTS described in Section 5.3.3.10 except for design temperatures. The highest design temperatures for the IHTS are 775°F for Types 304 and 304H stainless steels and 1015°F for Type 316H stainless steel. Creep is not significant for Types 304 or 304H stainless steels at 775°F.

59 The material considerations for the 2-1/4 Cr - 1 Mo ferritic steel and Alloy
800H used in the IHTS are the same as those for the steam generator system
35 described in Section 5.5.3.11.

5.4.3.11 Protection Against Environmental Factors

Protection for the principal components of the IHTS against environmental factors is provided by the structural integrity of the Reactor Containment and Steam Generator Building. Environmental factors to be considered include the following:

Fire Protection - See Section 9.13.

Flooding Protection - See Section 3.4.

Missile Protection - See Section 3.5.

Seismic Protection - See Section 3.7.

Accidents - See Section 15.6.

References to Section 5.4

26 * 1. Hanford Engineering Development Laboratory, Liquid Metal Fast Breeder Reactor Materials Handbook", HEDL-TME-71-32.

26 * 2. W. H. Yunker, "Standard FFTF Values for the Physical and Thermophysical Properties of Sodium", WHAN-D-3, July 6, 1970.

3. "Flow of Fluids, Tube Turns", Bulletin TT725, Louisville, Kentucky.

26 * 4. L. F. Moody, "Friction Factors for Pipe Flow," Transactions of the American Society of Mechanical Engineers, Volume 66, November, 1944; pages 671 to 678.

26 * 5. Stepanoff, A. J., "Centrifugal and Axial Flow Pumps", 2nd Edition, John Wiley and Sons, Inc., New York, New York, 1975.

6. P. C. Paris and G. C. Sih, "Stress Analysis of Cracks" in Fracture Toughness Testing and its Applications, pp. 30-83, ASTM, Philadelphia, Pa., 1965.

* References annotated with an asterisk support conclusions in the Section.
26 Other references are provided as background information.

59 TABLE 5.4-2

HAS BEEN DELETED

TABLE 5.4-3

IHTS MATERIAL SPECIFICATIONS

PRODUCT FORM	MATERIAL	ASME	RDT STD	SPECIFICATION
Pipe	304H SS 316H SS	SA-358	M3-7T	Electric-Fusion-Welded Austenitic Chromium - Nickel Alloy Steel Pipe for High Temperature Services
		SA-376	M3-3T	Seamless Austenitic Steel Pipe for High Temperature Central-Station Service
	2-1/4 Cr-1 Mo	SA-155	M3-11T	Electric Fusion-Welded Pipe for High-Pressure Service
		SA-335	M3-12T	Seamless Ferritic Alloy Steel Pipe for High-Temperature Service
Fittings	304H SS 316H SS	SA-403	M2-5T	Wrought Austenitic Stainless Steel Pipe Fittings
	2-1/4 Cr-1 Mo.	SA-234	M2-3T	Pipe Fittings of Wrought Carbon Steel and Alloy Steel for Moderate and Elevated Temperatures
Plate	304 SS 304H SS 316 SS 316H SS	SA-240	M5-1T	Heat Resisting Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Fusion-Welded Unfired Pressure Vessels
	2-1/4 Cr - 1 Mo	SA-387	M5-22T	Pressure Vessel Plates, Alloy Steel, Chromium-Molybdenum
Bars and Shapes	304 SS 304H SS 316 SS 316H SS	SA-479	M7-3T	Stainless and Heat-Resisting Steel bars and Shapes for use in Boilers and Other Pressure Vessels.
Castings	304 SS	SA-351	M4-2T	Ferritic and Austenitic Steel Castings for High-Temperature Service.

35

Amend. 35
Feb. 1977

TABLE 5.4-3 (Cont)

IHTS MATERIAL SPECIFICATIONS

PRODUCT FORM	MATERIAL	ASME	RDT STD	SPECIFICATION
Forgings	304 SS 304H SS 316 SS 316H SS 2-1/4 Cr - 1 Mo	SA-182	M2-2T	Forged or Rolled Alloy-Steel Pipe Flanges, Forged, Fittings, and Valves and Parts for High-Temperature Service
	2-1/4 Cr - 1 Mo	SA-336	M2-4T	Alloy Steel Forgings for Seamless Drum, Heads, and Other Pressure Vessels
59 35	Ni-Fe-Cr (Alloy 800H)	SB-407	M3-9T	Nickel Alloy for Transition Joints

TABLE 5.4-5

59 CLASSIFICATION OF IHTS PUMPS AND VALVES

	Component	Classification	Normal Operating Mode +
	IHTS Pump	Active	100% of Rating
1	Sodium Dump Valves		
	Pump Discharge	Inactive	Isolation - NC
	Superheater Inlet Line	Inactive	Isolation - NC
59	Superheater Outet Line	Inactive	Isolation - NC
	Evaporator Outlet	Inactive	Isolation - NC
49	Exp/Dump Tank Eq. Valve	Inactive	Isolation - NC
1	Hydrogen Detector Valves*	Inactive	Isolation - NO
	IHX Vent Valves*	Inactive	Isolation - LC

+ NC = Normally Closed

NO = Normally Open

LC = Locked Closed

*These valves form part of the intermediate coolant boundary, but are not actually parts of the IHTS. The valves are discussed in Sections 5.5, 9.3, and 9.5.

Amend. 59
Dec. 1980

TABLE 5.4-6

SUMMARY OF STRESSES IN MOST HIGHLY LOADED IHTS HOT LEG ELBOW

Condition	Location φ	Hoop Stresses			Axial Stresses		
		Inside	Mid	Outside	Inside	Mid	Outside
Refueling	0°	-410	4827	10,062	1950	3521	3506
	7.5°	709	4787	8865	3437	4661	5884
	37.5°	7415	4873	2331	4969	4207	3444
	90.0°	5335	4657	3979	2545	2342	2138
Normal Plus Scram	0°	-4958	3990	12,940	5603	4023	2970
	7.5°	-2498	4025	10,548	9022	7241	5460
	37.5°	15,980	4793	-6394	14,350	7256	162
	90.0°	11,847	4657	-2533	7812	1917	-3978

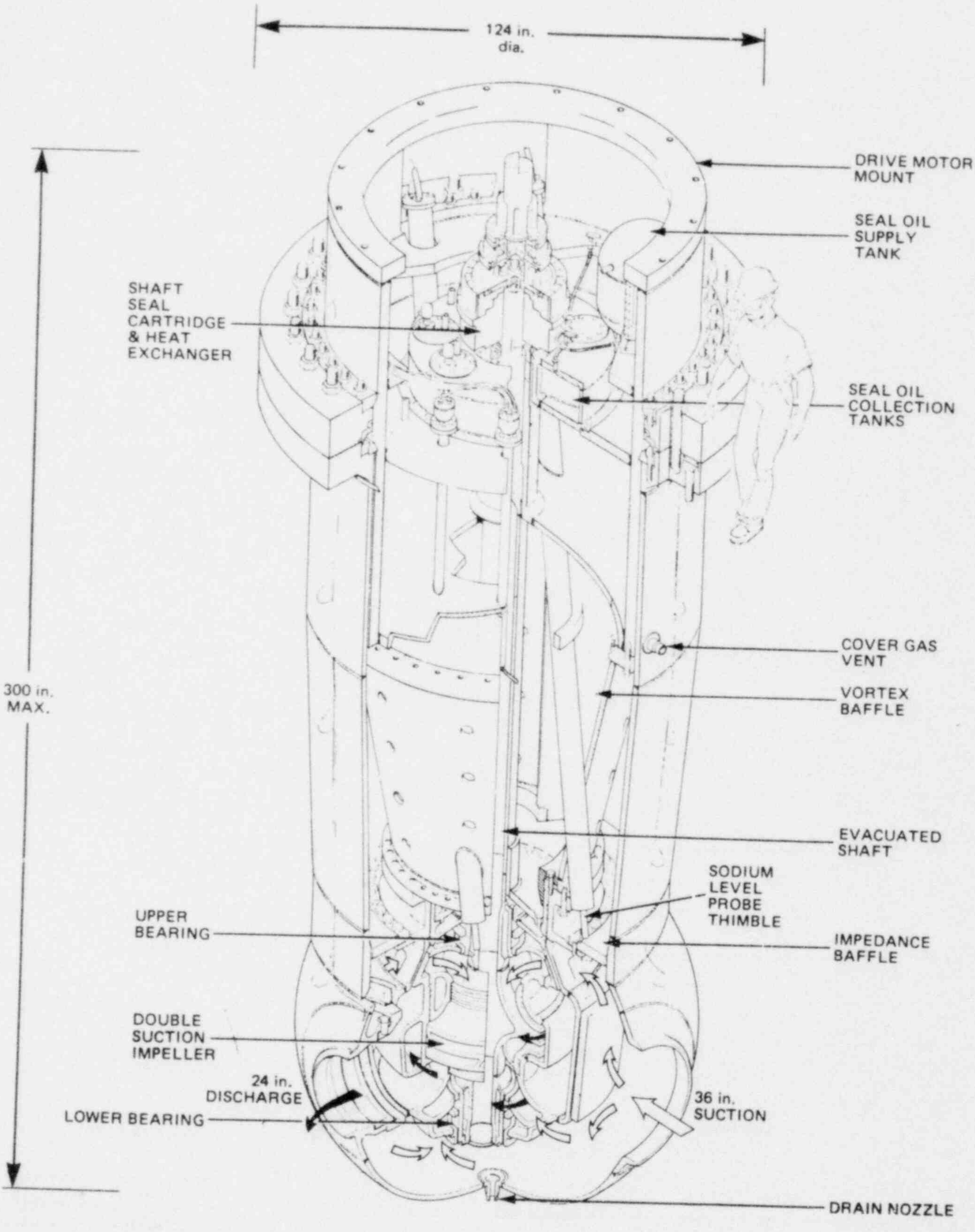

Figure 5.4 - 1 INTERMEDIATE PUMP ISOMETRIC

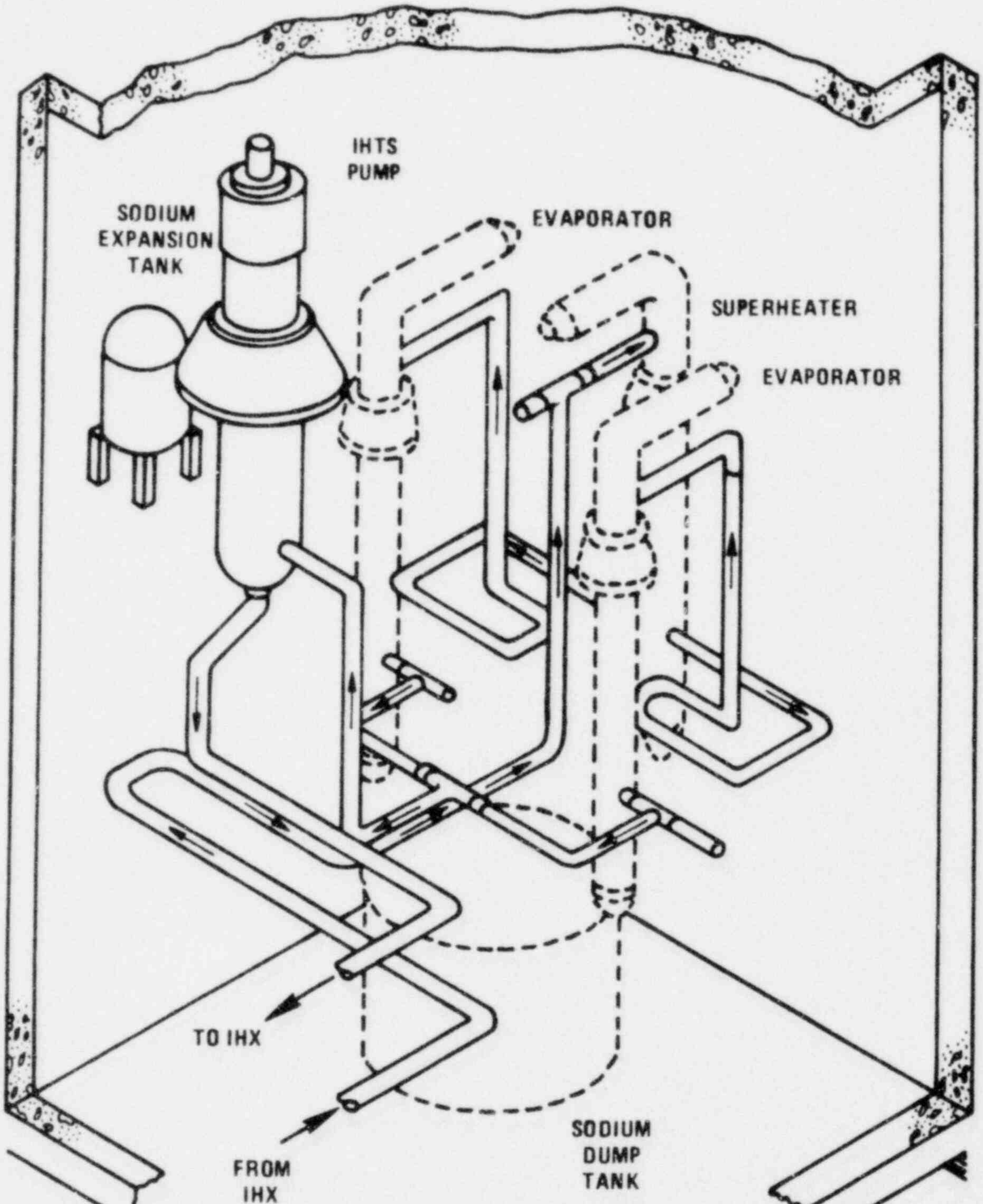

Figure 5.4-2 IHTS Components and Piping(Steam Generator Cell) POOR ORIGINAL

6670-16

5.5 STEAM GENERATION SYSTEM (SGS)

5.5.1 Design Basis

5.5.1.1 Performance Objectives

Steam Generator and Water Steam Subsystem

The objectives of the steam generators and water steam subsystems are to:

a. Transfer reactor generated heat (975 MWt) from the intermediate sodium to the water/steam in the steam generator modules at the rate required
41| to maintain the intermediate sodium cold leg temperature within its operating limits.

b. Provide superheated steam at the temperature, pressure, and flow rate required by the turbine.

41| c. Regulate the feedwater flow rate in response to plant process control over the full operating power range of 40 to 100 percent reactor thermal power.

d. Remove plant sensible and reactor decay heat from the IHTS (1) following reactor shutdown from rated power with one, two or three
59| 41| loops using pony motor flow on the sodium side and forced or natural circulation on the water/steam side; or with two or three loops using natural circulation on the sodium side and forced or natural circulation on the water/steam side; (2) following reactor shutdown from approximately 2/3 rated power with one or two loops using forced or natural circulation on either or both the sodium and water/steam sides.

e. Contain intermediate sodium and maintain a safe boundary between the sodium and the water/steam in the steam generator modules.

f. Prevent the water or steam side pressure from exceeding a safe value.

g. Transfer reactor generated heat from the intermediate sodium to the water/steam in the steam generator modules with two loops at nominally two-thirds rated power output.

Sodium Water Reaction Pressure Relief Subsystem (SWRPRS)

The objectives of the SWRPRS are to:

a. Prevent the pressure generated by a sodium water reaction within a steam generator module from reaching a value which would violate the integrity of the IHX primary to intermediate boundary.

b. Control the solid, liquid, and gaseous products of a large sodium-water reaction so that the solids and liquids are contained in appropriate vessels and the gas (hydrogen) is released to the atmosphere in a safe manner.

The SWRPRS is to be designed to accommodate the sodium-water reaction which would result from the design basis leak for a steam generator module. This event is assumed to start with an equivalent of a full guillotine rupture of a heat transfer tube in the most unfavorable location in the unit, which causes
59 the equivalent of two additional guillotine tube ruptures. Justification for the selected design basis leak is given in Section 5.5.3.6. Water or steam flow is injected into the sodium mass from both ends of the tubes, causing high pressures and acoustic waves in the intermediate sodium system. The pressures will burst the rupture discs to the SWRPRS, permitting sodium, sodium-water reaction products, and steam and water to be ejected into the SWRPRS. 9

Leak Detection Subsystem

The objective of the leak detection subsystem is to detect and alert the operators of water-to-sodium and steam-to-sodium leaks in the steam generator modules and to identify the module in which the leak has occurred. See Section 7.5.5.3 for further details.

Sodium Dump Subsystem

59 The objective of the Sodium Dump subsystem is to provide drainage capability and storage for the IHTS sodium, which may be contaminated with sodium-water reaction products following a sodium-water reaction within a steam generator.

The Sodium Dump Subsystem for each Intermediate Heat Transport System (IHTS) shall meet the following design basis:

a. Each subsystem shall accommodate and store all IHTS sodium during
59 drainage.

b. Design shall provide capabilities for the Intermediate Sodium Service System to clean the sodium from the Sodium Dump Subsystem and transfer or fill the IHTS loop, if necessary.

During a rapid dump, the sodium from the steam generators is drained through the dump lines in less than twenty minutes. When the IHTS sodium is dumped in this fashion, the IHX remains essentially filled with sodium because drainage is through the IHTS piping which enters and leaves at the high point of the IHX. 9

59 Water Dump Subsystem

The Water Dump Subsystem for each SGS loop shall meet the following design bases:

a. The Water Dump Subsystem in combination with the power relief valves will be designed to reduce the evaporator operating pressure to about
59 35 300 psig within 30 seconds.

b. Each Water Dump Subsystem shall have sufficient capacity to store the
41 liquid water dumped after isolation of both evaporators in one loop.

c. The water from the Water Dump Subsystem will be drained for reuse or disposal.

d. No single failure of the isolation and dump equipment shall cause the loss of shutdown heat removal capability. In addition, no single failure shall cause the two water dump valves in the same dump path to
59 open.

SGS Design Parameters

Table 5.5-1 lists the structural design temperature, pressure and minimum test pressure for the SGS components. The specified operating transients for SGS components are given in Appendix B of the PSAR. Refer to Section 3.7 for discussion of the input criteria for seismic design of Category I structures, systems and components.

Design requirements for the SWRPRS rupture disks are to assure that the disks will rupture at differential pressures low enough to prevent a loss of integrity of the IHTS and IHX as a result of over-pressures produced by a large sodium-water reaction, and high enough to maintain system operability under all other normal, upset, emergency and faulted plant conditions.

The features which protect the principal components of the SGS against environmental effects are discussed in Section 5.5.3.11.

5.5.1.2 Applicable Code Criteria and Cases

The Steam Generation System will be constructed in accordance with Section III of the ASME Boiler and Pressure Vessel Code and applicable code cases. The
59 steam generator modules are classified as Class 2 components but shall be
91 constructed to Class 1 rules, while the remainder of the SGS Components and
59 Piping shall be designed as Class 2 or 3. Class 2 and 3 components will use supplemental Code Cases 1606 and 1607 as applicable. Steam Generation System Piping Classifications are given in Section 5.5.2.3.3.

Construction of all ASME Section III components will be supplemented with appropriate sections of RDT standards (see Sections 5.5.2.3.4, 5.5.3.1.5, and 5.5.3.11.2). Mandatory application of these RDT standards to the steam
41 generator module will be limited to the water/steam to sodium boundary materials.

59 The SGS modules have normal operating temperatures exceeding those for which allowable stress values are given by Section III. For these high temperature components, the design rules provided in several code cases will be made mandatory. A list of the mandatory code cases is given in Table 5.5-2 with a brief description of their applications. If the rules provided in these code cases are not complied with, justification for using less stringent rules will be provided.

5.5.1.3 <u>Surveillance Requirements</u>

Surveillance requirements for the SGS are defined by the appropriate Section III (Class 1, 2 or 3) of the ASME Code.

5.5.1.4 <u>Material Considerations</u>

<u>High Temperature Design Criteria</u>

41 The high temperature design bases for the steam generator modules of the Steam Generation System are the same as those identified for the Primary Heat Transport System in Section 5.3.1.4.

<u>Material Specifications</u>

A list of material specifications for the SGS vessels, piping, pumps and valves is given in Table 5.5-3. Corresponding weld material specifications are also listed in Table 5.5-4.

5.5.1.5 <u>Leak Detection Requirements</u>

The steam generator leak detection system provides early detection of possible water to sodium leaks in the steam generator modules. For small leaks, operator corrective action is taken, for large leaks, automatic corrective action is taken.

For further details, see Section 7.5.5.3.

5.5.1.6 <u>Instrumentation Requirements</u>

Section 7.5.2 provides the instrumentation requirements for the Steam Generation System.

<u>Control Systems</u>

<u>Feedwater Flow Control</u>

Instrumentation and control equipment is provided for the automatic regulation of feedwater flow to the Steam Generation System. Details of this control system are discussed in Section 7.7.1.5.

<u>Recirculating Water Flow Control</u>

The recirculation pump runs at constant speed for the full range of power operation and no automatic control system is required. See Section 7.7.1.6.

Class 2 and 3 valves. All materials, exclusive of seals and packing, shall be designed for a 30-year plant life under the environmental conditions applicable to the particular system.

Power operators shall be sized to operate successfully under the maximum differential pressure determined in the design specification.

41 The main steam isolation valves (superheater outlet isolation valves) are capable of being closed to stop the venting of steam into the steam generator or turbine buildings in case of a steam line pipe break downstream of the isolation valves. The maximum steam flow rate is expected from a steam line break immediately downstream of the isolation valve. The disc and stem will be designed to withstand the forces produced when closing the valve under choke flow conditions.

Figure 5.5-2A shows a main steam isolation valve. It is a conventional gate
59 valve to provide a minimum resistance flow path when the valve is wide open.
An air cylinder, supplied by plant or accumulator stored air, is provided at
the top of the operator assembly for opening and closing the valve during
normal operation or during valve exercising. Four sets of coil springs are
mounted below the air cylinder to automatically close the valve when there is a
complete loss of air to the valve operator. An oil dashpot mounted on the
valve stem extension provides a means of adjusting the closing speed of the
valve. The rate of closure of the valve is preset and held constant,
regardless of the differential pressure across the valve seat, by two pressure
compensated flow control valves. This will prevent damage of the seat and/or
valve body by limiting the impact force at valve closure. 14

Each valve used in the SGS will be evaluated as to its performance relative to plant safety and mode of operation in the event of failure (fail open, fail closed, etc.). As part of these evaluations, the need for a pneumatic accumulator adjacent to a valve and solenoid requirements for emergency operation will be determined.

Tests and Inspections

Line valves will be shop tested by the manufacturer for performance according
41 to the design specifications for leakage past seating surfaces and for
integrity of the pressure retaining parts. Selected line valves will be
manually operated during loop shutdown periods to assure operability.

5.5.2.3.2 Recirculation Pumps

35 The recirculation pump will be a single stage, centrifugal type, driven by a
59 constant speed, 4.0 KV, 1000 HP motor. It will take suction from the steam
drum, and provide 2.22×10^6 pounds of water per hour to the evaporators.

The pump and its support will be designed and fabricated per ASME Section III,
35 Class 3 as shown in Table 5.5-6.

5.5.2.3.3 Steam Generation System Piping

Design Basis

Steam Generation System (SGS) piping for each loop shall meet the following design bases:

1. The design shall accommodate operational stresses, such as internal pressures and safe shutdown earthquake loads without failure.

2. The piping will accommodate the worst possible loading from the duty cycle (Appendix B) according to the design requirements for the water/steam side SGS piping given in Table 5.5-7.

3. The design of sodium piping shall be the same as for the intermediate heat transport system as described in Section 5.4.2.3.4.

41 14

Functionally, the drum receives a saturated water/steam mixture from the evaporators and subcooled feedwater and produces saturated steam of low moisture content for the superheater and subcooled water of low steam content for the recirculation pump. The water/steam mixture from the evaporators enters the drum through the water/steam nozzles and flows into an annular volume along the sides of the inner drum wall created by a girth baffle
59 extending along the side of the drum for the length of the cylinder. Centrifugal steam separators mounted along the length of the drum draw from this annular volume, separate the mixture into phases, and direct the steam upward and the water downward into the inner volume of the drum. The main 6
28 feedwater enters the drum through a single nozzle which feeds two distribution pipes through a "Y" connection inside the drum. The feedwater is distributed along the length of the drum by rows of orifice holes in the two pipes which are located along each side of the drum beneath the steam separators. The auxiliary feedwater enters through a separate nozzle and is distributed along the length of the drum by two rows of spray nozzles in a single distribution
59 pipe located above the water level in the drum. Feedwater mixes with the water from the separators and is drawn downward and out through the water outlet nozzles by the recirculation pump. The steam passes upward through chevron type dryers in the upper portion of the drum and out through the steam outlet nozzles to the superheater. The dryers remove all but the last fractional percent of the moisture from the steam and drain this moisture back to mix with the resident drum water. Drum drain piping, located along either side of the drum in the region where the water from the separators enters the drum inner volume, draws water of high impurity concentration from the drum. 6

41 5.5.2.4 Overpressure Protection

Location of Pressure Relief Devices

Safety/power relief valves are located in the steam generation system to:

41 1. Prevent a sustained pressure rise of more than 10 percent above system
59 design pressure at the design temperature within the pressure boundary
41 of the system protected by the valve under any pressure transients anticipated; and

2. Provide steam generator module blowdown capability.

Installation of the valves will comply with the requirements as specified in Section 3.9.2.5. Safety/power relief valves are installed on the outlet lines 17
from each evaporator to provide venting capability and a portion of the
41 required safety/relief capability. Safety valves are installed on the steam
46 drum to provide the remainder of the safety capability for the recirculation loop. Additional safety/power relief valves are installed on the steam exit line from the superheater because the steam lines to and from the superheater have isolation valves. The P&ID for the Steam Generation System, Figure 5.1-4 shows the locations of these safety/power relief valves. Additional details of sizes and pressure rating are given in Table 5.5-8.

Pressure Relief Devices

Water/Steam Side

46

Each safety relief valve on the evaporator outlet piping provides a saturated
59 steam (100% quality) relief capacity of 430,000 lb/hr, or 39% of the rated
steam generating capacity of the recirculation loop. Each safety relief valve
59 on the steam drum provides a saturated steam relief capacity of 410,000 lb/hr,
or 37% of the rated steam generating capacity of the recirculation loop. The
difference in rated capacity of these valves is due to the difference in the
valve set pressure. The combined relief capacity of the six valves for the
59 recirculation loop is therefore 230% of the rated steam generation capacity.
This generous margin is provided for two reasons: (1) the capacity required to
relieve most of the overpressure transients in the recirculation loop can be
satisfied by opening one or both of the steam drum valves, relieving the system
with dry steam rather than wet steam; (2) the capacity of the evaporator relief
41 valves is based on the capacity required to achieve rapid blowdown of the
evaporator modules following a water to sodium leak.

Three safety/power relief valves installed on the exit line from the
41 superheater provide a relief capacity of 75% of rated superheater steam flow at
a pressure of approximately 1800 psig and temperature of 900°F. The remaining
25% of rated flow is relieved by the steam drum valves.

Settings for the safety/power relief valves are in accordance with Code requirements. Setting presently selected are shown in Table 5.5-8.

The capacity requirements stated for the safety/power relief valves are based upon the steady-state plant ratings. During the plant design period, it was
59 determined that the relieving capabilities required by the steady-state conditions are adequate for all plant unbalanced or transient conditions.

The safety/power relief valves and the piping for these valves will be subjected to sizable impulse forces during operation of these valves. The relief/safety valve reaction forces, due to dead weight, seismic events, discharging fluid, thermal expansion, and the dynamic effects of valve opening or closure, will be accommodated with the use of supports on the valve discharge piping and proper sizing of the inlet nozzle to limit the stress in the valve and nozzle within Code allowables. As details of the installation of 25
these valves and the piping were established, the magnitudes of these loads were determined and suitable restraints for the valves and piping were provided. The discharge from the valves are routed to a suitable location on the roof where plant personnel will not be exposed to hazards from the
59 discharge.

Sodium Side

The IHTS is protected by pressure-relief devices in the steam generator IHTS sodium piping and in the expansion tank/dump tank gas equalizer line. Each device consists of two rupture discs installed in series that relieve to the
41 Reaction Products Separation Tank and sodium dump tank respectively. Between each pair of rupture discs a sodium leak detector is installed so as to give an alarm in the event the sodium-containing rupture disc develops a small leak. A leak developed by the downstream rupture disc would not affect the safety function of the pressure-relief assembly. Three pressure sensors are located
41 immediately downstream of the IHTS sodium piping rupture disc assemblies. A coincident signal from any two of the three pressure sensors initiates an automatic loop shutdown and water/steam side pressure reduction. See Section 7.5.6 for plant control actions following rupture of a rupture disc.

Rupture disc assemblies are located on the 24-inch sodium inlet piping to the superheater, each evaporator 18-inch outlet piping, and in the IHTS expansion
41 tank to sodium dump tank gas equalizer line. Their capacity and the magnitude and application of any reactive forces generated on the system or its components will be included in the description of the Steam Generation System.

5.5.2.5 Leak Detection System

The steam generator leak detection system description is provided in Section 7.5.5.3.

5.5.2.6 Sodium Water Reaction Pressure Relief System (SWRPRS)

35 The SWRPRS includes piping, reaction products separator tanks, rupture disc assemblies, stacks and non-return valves, and ignitors. The system for each of
54 the three heat transfer loops is identical.

Dual rupture disc assemblies are installed on the intermediate sodium piping
adjacent to the steam generator units and in the IHTS expansion tank to sodium
dump tank gas equalizer line. For large sodium-water reactions excessive
42 pressure in the intermediate sodium system will burst the rupture discs,
59 41 dumping sodium and reaction products to the separator tanks. Gaseous reaction
products and any entrained sodium and liquid or solid particulate where the
major portion of the extrained particulate matter will be removed. The gaseous
41 reaction products are then directed to the flare stack where an igniter is
installed. The dual rupture disc configuration is used to avoid complete
dumping of the intermediate sodium system into the SWRPRS in case of a leak in
the first disc.

Intermediate size steam or water leaks in the steam generator modules up to
approximately 2 lbs/sec can be accommodated without failure of the SWRPRS main
rupture discs. This is accomplished by relieving IHTS system pressure at the
expansion tank through the expansion tank/dump tank gas equalization line
41 rupture disc to the sodium dump tank.

A large sodium-water reaction may inject slugs of sodium into the piping from the rupture discs to the separator tank. These slugs can be accelerated rapidly by the hydrogen gas pressure resulting from the sodium-water reaction, resulting in high velocities for these slugs while in the pipes and high reaction forces in the piping. The results of the TRANSWRAP analyses for various configurations of piping and sodium-water reactions will be used to optimize the piping sizes and system from the rupture disc assemblies to the separator tank.

Inconel 600 was selected as the rupture disc material based on an extensive
review of material properties and past experience. The rationale for its
35 selection is that at the design temperature, I-600 has: 1) high creep
41 strength; 2) mechanical properties are not significantly changed with long term
aging; 3) demonstrated good resistance to sodium corrosion; 4) minimal change
in strength with increasing temperature in the temperature range of interest
35 and 5) expected low carburization rate in sodium (See Ref. 3 & 4). In
59 41 addition, I-600 was used as the disc material for the rupture disc on the
Modular Steam Generator (MSG) during the testing of that unit in the Sodium
Components Test Installation (SCTI) at the Liquid Metal Engineering Center
(LMEC). The discs performed satisfactorily.

The basis for selecting carbon steel as the SWRPRS piping and equipment
material was to minimize the potential for caustic stress corrosion failure
from exposure to sodium hydroxide following a postulated large sodium-water
59 41 reaction in a steam generator module. Austenitic stainless steels are known to
be very susceptible to caustic stress corrosion failures. 19

Amend. 59
Dec. 1980

54 The reaction products separator tanks (two per loop) are used to separate most
of the liquid sodium and solid sodium-water reaction products from the gases,
hydrogen and steam, which are ejecting the sodium from the failed unit. The
relief piping enters the separator tank tangentially, near the top of the tank,
causing rotation of the materials in the tank. This rotation creates a
centrifugal force which assists in the separation of the liquid and solid
54 materials from the gases. The tanks are sized to hold the total volume of
sodium in the three steam generator modules, the volume of sodium in the
59 intermediate sodium pump and expansion tank, and the volume of sodium which
would be expected to spill from the remaining secondary sodium system into the
superheater unit because of the momentum of the fluid in the loop. The
superheater sodium inlet and intermediate sodium pump inlet are high points in
the intermediate sodium system, such that sodium will not drain by gravity from
the remainder of the system into the reaction products separator tanks. The
54 gases separated from the other materials in the reaction products separator
tanks are discharged through central nozzles at the top of each tank.

Gases, which will be primarily nitrogen used to inert the SWRPRS and hydrogen
resulting from the sodium-water reaction, but may include some steam, are piped
to a stack where the hydrogen, when of a combustible concentration, will be
35 ignited and burned. Burning the hydrogen prevents the postulated creation of a
hydrogen "bubble" in the atmosphere which could mix with oxygen and create a
potentially explosive region in the atmosphere. A non-return valve is located
in the line between the tanks and the stack to reduce backflow of air into the
54 system.

The entire SWRPRS is normally filled with an inert gas to avoid the possibility of a hydrogen explosion in the system after activation of the system. The inert gas is maintained above atmospheric pressure so that oxygen will not leak into the system and to verify that the inert gas atmosphere is being maintained in the SWRPRS. To maintain an inert gas in the system, a low pressure rupture disc is required in the line to the stack. This rupture disc is quickly broken or ejected after a small pressure increase in the SWRPRS.

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

54

The Reaction Products Separator Tanks are sized to accommodate the maximum amount of sodium and liquid or solid sodium reaction products which can be ejected or drained from the IHTS into SWRPRS during and following a sodium-water reaction. Regions of the IHTS and components in that system which might drain into SWRPRS during and following a sodium-water reaction were determined by study of the hydraulic profile of the IHTS. It was assumed that all of the rupture discs in the main rupture disc assemblied would be broken by the
35 sodium-water reaction. An additional capacity of about 25% above that determined by evaluation of the hydraulic profile of the IHTS was then added to
establish the capacity of the Reaction Products Separator Tanks. 19

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

<u>Tests and Inspections</u>

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

5.5.2.7 Sodium Dump Subsystem

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

A sodium dump tank is provided within each sodium dump subsystem to serve as a:

59 a. Sodium dump for sodium at operating temperatures

b. Intermediate sodium storage

c. Sodium fill

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

The Sodium Dump Subsystem will be used for normal drainage or for drainage of sodium from each IHTS circuit after the sodium has been contaminated by sodium-water reaction products from a large sodium-water reaction occurring within an evaporator or superheater module. Upon manual initiation, rapid drainage of IHTS sodium and $Na-H_2O$ reaction products will be accomplished by lines which are connected at different locations within the IHTS sodium circuit. An equalizer line to IHTS sodium expansion tank is provided.

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

59 In the event that a sodium-water reaction occurs, sodium contaminated with sodium-water reaction products would be dumped into the dump tank and maintained in a molten state by trace heaters on the tank. Later, the contaminated sodium in the dump tank will be cleaned by circulation through the

Intermediate Sodium Processing System (see Section 9.3) and then transferred to the IHTS sodium loop. The dump tank could then be cleaned and the Sodium Dump Subsystem be made ready for use again.

In the event of a large sodium-water reaction event within a Steam Generator Module, sodium and sodium-water reaction products will be discharged through connecting piping to the Reaction Products
41 Separator Tank, where most of the sodium and liquid or solid reaction
35 products will be accumulated. The detailed methods and processes to be used to remove the reaction products have not been chosen, however, studies are underway to evaluate potential alternatives. It has been recognized that in addition to removal of the affected Steam Generator Module, replacement of the Reaction Product Separator Tanks may be necessary following a large sodium-water reaction event. Since most of the liquid and solid reaction products will end up in the tanks, replacement of the tanks will remove most of the reaction products from the system.

The Reaction Products Separator Tanks provide sufficient capacity so that the solid reaction products from a sodium-water reaction may be left in the tank following the incident if the tanks have not sustained mechanical damage from the incident which would render them unsuitable for further service. Sodium would be melted and removed from the tank, leaving the solidified reaction products in the tank.

Contamination of the remainder of the IHTS with sodium-water reaction is expected to be limited to the piping between steam generator modules and the piping between the evaporator modules and the sodium pump. The sodium-water reaction products are not expected to reach the pump. (See Section 5.5.3.6.2).

As a result of a water to sodium leak, caustic stress cracking can occur under certain reasonably well defined conditions of stress, temperature, and concentration of aqueous sodium hydroxide. It is important to note that the sodium hydroxide must be aqueous to cause cracking and that the lifetime of aqueous sodium hydroxide in sodium is very short. The backflow of sodium from the pump is expected to transport the reaction products in the IHTS piping upstream of the pump to the SWRPRS relief lines and RPST's. Therefore, the residence time of the reaction products in the IHTS piping and the potential for caustic stress corrosion are minimized. Any reaction products contained in the IHTS sodium remaining in the loop will be drained into the sodium dump tank following water side depressurization of the Steam Generator Modules to 300 psig. Solid reaction products, if any, will be removed from the IHTS sodium contained in the dump tank by circulation through the Auxiliary Liquid Metal
System. 19

59 Later, when the HTS loop is ready for operation, the water from the dump tank will be drained, either for reuse or for disposal, thus making the Water Dump Subsystem available for future use.

The Water Dump Subsystem components are designed in accordance with ASME Code Section III Nuclear Power Plant components.

The dump tank and its associated piping are designed to satisfy the
59 41 requirements of Seismic Category 2. (See Section 3.7.)

41 The water dump tank is relieved to the atmosphere through an open pipe to limit the maximum internal pressure and to assure the integrity of its components.

Tests and Inspection

The water dump tank shall be designed such that all normal inspection, maintenance, and repair can be performed during normal shutdown periods or after the removal of water following an evaporator blowdown.

59 41 The inspection of the dump tank, tank supports, and other components will be conducted in accordance with manufacturer's instructions. Water level indication, water and other system component temperatures and pressures will be available or displayed in the control room.

5.5.3 Design Evaluation

5.5.3.1 Analytical Methods and Data

The design of the SGS relative to sodium flow and water or steam flow in the interconnecting piping is based upon technology attained during the development, design, construction, and operation of sodium and water or steam systems of similar type. The design of the steam generator components is based
41 on available technology, testing of similar components, model testing, and development testing. The design of the Sodium-Water Reaction Pressure Relief Subsystem (SWRPRS) is based on experimental data and analysis.

Sodium properties are based on the report "Standard FFTF Values for the Physical and Thermo-physical Properties of Sodium". (Ref. 1)

Water or steam properties are based on the "1967 ASME Steam Tables".

The pressure losses are calculated using established equations and values available in textbooks and piping catalogs.

41 49 5.5.3.1.1 Structural Evaluation Plan (SEP)

The procedure for developing SEP's for SGS components is the same as that
41 described for the PHTS in Section 5.3.3.1.1.

5.5.3.1.2 Stress Analysis Verification

The analytical requirements given in the ASME Section III Code for Class 1, 2,
and 3 components under normal, upset, emergency, and faulted operating
59 conditions were used as design limits for the SGS. The appropriate figures and
sections in Section III, that contain the analytical requirements corresponding
to the appropriate design class and operating conditions, are given below.
Accidents classified as faulted will satisfy the code requirements referenced
below for faulted operating conditions.

For components in which temperatures exceed those provided for in Section III,
the rules of Code Case 1592 provide analytical requirements for Class 1
41 components.

5.5.3.1.3 Compliance with Code Requirements

Regulatory Guide 1.48 delineates design limits and appropriate combinations of loading associated with normal operation, postulated accidents, and specified seismic events for the design of Seismic Category I fluid system components. The SGS will be treated as a Seismic Category I component and will comply with the intent of Regulatory Guide 1.48 as described in Section 3.9.

5.5.3.1.4 Category I Seismic Design

The limits of required Category I design for the SGS are shown in Figure 5.1-4.
The components and supporting structure for the major components will also
comply with the Category I requirements. (See Table 3.3-2). Diagrams of
preliminary seismic models of SGS components are provided in Figures 5.5-6 17
through 5.5-9.

5.5.3.1.5 Analytical Methods for Pumps, Valves, and Heat Exchangers

59 Standard text book techniques along with finite element computer methods were
used to evaluate the stresses in the SGS. For these components, elastic,
inelastic, and dynamic analysis methods were utilized. A brief description of
59 the types of analysis, along with a list of computer codes used, are given
below. Analysis of a particular component was made by appropriate selection of
59 the computer codes listed below.

59 5.5.3.1.5.1 Steam Generator

The steam generator is a Class 2 vessel but is designed as a Class 1 vessel in
accordance with Section III of the ASME B&PV, RDT E-15-2NB and Code Case 1592
as supplemented by RDT F9-4 (see Section 5.5.1.2). Non-pressure boundary
41 components are designed to the intent of Code Case 1592.

The failure modes which are expected to dominate the steam generator design are identified below. The loadings which could contribute to such failures are also identified.

a. Rupture of steam generator tubing. The design recognizes that the tubing is subject to an internal, high water/steam pressure loading, augmented by steady and cyclical temperature gradients through the tubing wall and by point loadings at support plate contact points.

b. Water/steam leak through tubing or tube-to-tubesheet weld. This failure mode is identified more with material degradation situations than with loading conditions. It can result from water/steam leakage paths caused by weld inclusions or porosity, waterside metal corrosion on cyclic fatigue, sodiumside metal wear at support points, sodiumside metal corrosion by adjacent tube leak.

41 c. Sodium-to-air boundary rupture. To preclude this eventuality, the design recognizes large sodium/water reaction pressure pulse loadings resulting from rupture of water/steam tubing, low pressure loadings from the sodium system and transient thermal gradients during duty
59 cycle events.

Tests which are being performed or planned to support the steam generator design are: 26

a. Hydraulic Test Model (HTM)

Objectives

The objectives of the shell-side flow/tube vibration test utilizing the HTM were to (1) verify the absence of damaging tube vibration for CRBRP steam generators, and (2) verify uniform flow distribution for CRBRP steam generators.

Test Facility

High Flow Test Facility, Rockwell International, Rocketdyne Division, Canoga Park, California.

Component Characteristics

The HTM was a full-scale model of the CRBRP steam generators, except that the active length was shortened about 20 feet. The HTM incorporated all of the internal elements (baffles, tube supports, shrouds, flow distributors, etc.) required to simulate the sodium-side flow system. The inlet and outlet plumbing duplicated CRBR sodium piping for about 10 feet on either side of the nozzles to insure proper simulation of exit and entrance conditions. Since the CRBR superheaters and evaporators are identical in internal design, the model could be tested in either mode. The model was fabricated from carbon steel rather than the 2-1/4 Cr - 1 Mo steel specified for the CRBR steam generators and was designed to evaluate sodiumside flow characteristics using water as the test fluid.

Test Description Summary

The HTM was tested for response to flow-induced vibrations and to determine its flow distribution characteristics. The vibration tests included
36 measurements of:

Group I tests will be on Electroslag Remelted (ESR) base metal tubing to determine rupture life at 510°C (950°F). Approximately twenty more base metal tests will follow for Group 2 to characterize actual CRBRP ESR tubing. A complete range of stresses will be studied for both decarburizing and control tests. Approximately 16 elevated temperature tensile tests will be run to determine the influence of pre-exposure time to decarburizing sodium on tensile properties at 510°C (950°F). Analytical work to assess decarburization profiles will be performed. Decarburized layers will be machined from specimen surfaces and analyzed for carbon by combustion and wet chemistry methods. Results will be compared with probe profiles. Specimens will be exposed at 510°C (950°F) for 2000 hours. A facility to determine ratchetting in the presence of a surface carbon gradient (decarburization) will be prepared, and tests will be performed.

The objective of these experiments is to develop an analytical model to predict the mechanical behavior as a function of the time and temperature of sodium exposure.

4. Effects of IHTS Sodium Environment on Mechanical Behavior of Transition Metal Joints

Uniaxial creep tests and fatigue crack growth tests identical to those described in item 3 will be performed on 2 1/4 Cr - 1 Mo/182/A800 and A800/16-8-2/316SS weld joints. Experience has indicated that the 2 1/4 Cr - 1 Mo/182 transition is the more critical of the four transitions and thus, will receive the highest priority in the testing schedule. Testing will be performed at 510°C (950°F) to simulate IHTS conditions.

Schedule

Preliminary data from this program will be available for the CRBRP Steam Generator Final Design Review and Final Fabrication Release. The program,
36 as planned, will be essentially completed by 1980.

43 41 Computer codes; MARC, TAP-4F, and DRIPS are used in the design of the steam generator and are described in Appendix A of the PSAR.

5.5.3.1.5.2 Valves

The steam generator system control valves shall be designed to the alternative rules defined in ND 3512 of the ASME Code, Section III. In addition, thermal transient stress analysis, transient pressure analyses, and seismic response analyses were performed for appropriate valve components and, as applicable, for the valve operators. The analyses, which demonstrated that the valve assembly will function as designed and in accordance with the criteria specified in the ASME Code and the valve equipment specification, were provided by the valve manufacturer, after review and approval of their analytical 26
59 methods.

Inelastic methods were used by the manufacturers to verify that stresses were within code allowables.

Included in the failure modes to be considered were:

59 1. Rupture of valve assembly components from short term loadings such as pressure (static or dynamic), and seismic loading. 26

2. Loss of valve function due to excessive deformation of the valve assembly components, including air supply connections, because of seismic loading and/or thermal distortion.

3. Structural integrity damage from cyclic loadings.

4. Stem deformation due to excessive loading by the valve operator.

59 No structural tests of the valves were required to support the design analysis.

59 Computer programs used to verify vendor analysis, are those identified below.

59 The steam generator system pressure relief valves were designed to two different criteria, depending on the valve location in the steam generator system. For the valves located in the recirculation loop and on the steam drum, the design requirements of the ASME Code, Section III, ND 3511 and ND
59 7000 were used. For the relief valves subjected to elevated temperatures on the superheated steam line, the design requirements of the ASME Code, Section III, ND 3511 and ND 7000 were used. In addition, thermal transient stress
59 analysis, transient pressure analyses, and seismic response analysis were provided for the appropriate valve components and the valve operator, as applicable. As with the control valves, all of the analysis was provided by the valve manufacturer after approval of the analytical methods, which included
41 59 those methods outlined below. Also, where it was necessary to use inelastic analyses, they conformed to the guidelines of RDT Standard F9-4T and RDT Standard F9-5T, as applicable to Class 3 components at elevated temperatures.

59 41 Included in the failure modes considered for all relief valves were:

59 1. Rupture of valve assembly components from short term loadings such as seismic loadings; system pressure (static and dynamic), and loading due to discharging fluid.

2. Loss of the safety function due to excessive deformation of the valve assembly components under seismic loading.

3. Structural integrity, valve stem, and valve spring damage from cyclic loadings.

59 4. Loss of function due to cyclic loading on the pilot valve spring. 26

59

No structural tests were required to support the design analysis. 26

Isolation valves which are required to close under normal as well as pipe rupture fluid conditions (identified per BTP APCSB 3-1 as Essential Components) were analyzed for the resulting stresses. The valve velocity at closure was limited to a value which will result in negligible impact effects, based on past tests and/or analysis of similar valve configurations. To maintain a relatively constant valve closing velocity regardless of the closing forces due to normal and pipe rupture conditions, hydraulic flow control device is provided on the valve operator. The stresses in the valve components, at valve closure, determined using applicable static loads and the design is such that structural response will be within the elastic limits for the material thereby satisfying functional deformation requirements. For the stem, the actuator thrust/force was used. For the disk, valve seat, and valve body, the load due
59 to the differential pressure and actuator thrust was used.

<u>Elastic Analysis</u> 30

Standard text book methods (theory of elasticity, strain energy theory, etc.), including the finite element methods used in the computer codes listed below, were used for elastic analysis.

<u>Inelastic Analysis</u>

59 Inelastic analyses were required in some cases to demonstrate conformance with ASME and RDT Standards. RDT F9-5T gives a description of

59 | acceptab'e methods for time-independent elastic-plastic analysis. Some of the computer programs listed below have inelastic capabilities, and were used where applicable.

Dynamic Analysis

59 | Dynamic analyses were required for evaluation of the seismic response of the SGS components. The computer programs listed below employ several acceptable finite element methods used exclusively for the dynamic analyses of the SGS.

Computer Codes

44 | 41 | SAP IV

MARC

41 | ELTEMP

Abstracts of these codes are provided in Appendix A of this PSAR.

59 | 5.5.3.1.5.3 Analytical Methods for Evaluation of the Recirculation Pump

The analytical methods and computer software discussed in paragraph 5.5.3.1.5 were used to evaluate the SGS pumps during transient conditions. Standard techniques were used to evaluate the stresses in the recirculation pumps. Elastic and dynamic analyses methods were utilized. Brief descriptions of the
59 | types of analyses used are given below:

Elastic Analysis

59 | Standard methods (theory of elasticity, strain energy theory, etc.) were used for elastic analysis.

Dynamic Analysis

59 | Dynamic analysis was required for evaluation of the dynamics and the seismic response of the recirculation pumps. SAP-IV and ANSYS are computer programs which employ several acceptable finite element methods used for the dynamic analyses of the pumps.

See Appendix A for Code Abstracts.

59 5.5.3.1.6 <u>Operation of Active Valves Under Transient Loading</u>

Active SGS valves relied upon for operation under transient loading are the feedwater flow control valves, the superheater outlet isolation valve, the
59 feedwater inlet isolation valve, the safety/relief valves, and the drum drain valves. To ensure proper valve operation during their design life, an in-service testing program will be implemented for the SGS.

59

After a valve and/or its control system has either been replaced, repaired or undergone maintenance that could affect its performance, and prior to the time it is returned to service, it will be tested as necessary to demonstrate that the performance characteristics are within acceptable limits.

The remote position indicators will be visually calibrated at the same frequency as scheduled refueling outages, but not less than one observation every two years, to confirm that remote valve indications accurately, reflect valve operation.

The SGS active valves will be moved through their full-stroke during each
59 shutdown. During plant operation the valves will be exercised by part stroke operation at specified surveillance intervals.

<u>Feedwater Flow Control Valves</u>

The performance of the feedwater flow control valves has been demonstrated in other plants using similar valves under nearly identical feedwater system
59 operating conditions.

44 <u>Superheater Outlet and Feedwater Inlet Isolation Valves</u>

These isolation valves, which are required to operate during transient
59 conditions, and whose functional capabilities may be affected by the abnormal ambient pressure and temperature associated with the transient, will be tested as part of the vendor's qualification test program. Functional requirements will be verified throughout the test sequence. Components tested will be fully representative of production components.

<u>Safety Relief Valves</u>

The safety relief valves will be subjected to tests that simulate conditions experienced during service life.

Drum Drain Valves

Drum drain valves (which permit continuous blowdown flow during operation) will
59 be tested by full stroke operation at specified surveillance intervals.

Valve Requirements

To assure operability of active valves under the transient loadings to be experienced during plant service life, design specifications include the following requirements:

1. Valve bodies will be qualified to withstand seismic forces, and those
59 valves with extended proportions will have minimum frequency-of-vibration requirements.

2. Valve operators will be sized to open or close under the maximum differential pressure across the valve seat that is dictated by the transient service conditions.

3. Valves will be fully cycled at the vendor's shop before delivery to substantiate the vendor's guarantee that they will operate under actual service pressure conditions.

59

5.5.3.1.7 Analytical Method for Component Supports (Vessels, Piping, Pumps and Valves)

In accordance with the ASME Code, component supports have the same code classification as the components they support. Design of each component
41 support complies with the ASME Section III, Subsection NF design rules
59 corresponding to the component support classification. In order to provide assurance that the component support stresses comply with limits specified in
59 paragraph 5.5.3.1.2, analysis of each component support was performed. The analytical techniques and applicable computer codes discussed in paragraph
59 5.5.3.1.5 also apply to detailed analysis of support components.

5.5.3.2 Natural Circulation

The sodium side of the steam generators is vented to the IHTS Expansion Tank. Following a normal scram the main sodium pump motors will coast down to pony motor speed, where a clutch engages and drives the pump to produce about ten percent of design flow. This flowrate occurs approximately 30 seconds after scram initiation. The vent lines characteristic flow curves are similar to those of the main system and as such will result in a similar decrease to ten percent flow at pony motor speeds.

If the pony motor should fail to operate in a given loop the system flow will decrease to the natural circulation level of approximately 6% of full design flow. The vent line flow will also follow a similar characteristic curve.

Since the total vent flow from the modules is only 1/30 of the main loop flow, the vent flow will not significantly affect either pony motor or natural circulation main loop behavior.

During either natural circulation or pony motor operation, flow in the vent lines will be maintained and the vent lines will remain submerged in the expansion tank, preventing the cover gas from entering and inhibiting flow in 25
the vent lines.

Calculated operating values for the important parameters applicable to natural circulation conditions have shown that the SGS will function satisfactorily in the natural circulation operating mode. The anticipated range of sodium flows is from 2 to 15% for SGS inlet temperatures of 650-850°F. For this same temperature range, the recirculation system natural circulation flow is expected to be in the range of 9 to 14.5%. The hydraulic profiles for water and steam side of the SGS and for the IHTS are shown in Figures 5.5-1 and 5.1-3, respectively.

59

Tests and Inspections

In-service inspection of the steam generator modules is discussed under In-Service Inspection Program, Section 5.5.2.1.3.

Part Load Operation

Part load operation curves over the range of steam flows from 40 to 100 percent are presented in Section 5.7.2.

Design module heat transfer length were used with nominal values of sodium, water or steam, and tube heat transfer correlations for purposes of this analysis. This implied excess area, therefore, results in sodium operating temperatures in the evaporator lower than those used for design. Design heat transfer areas are determined by adding sufficient margin to the module length to permit operation with fouled tubes at 100% power for nominal sodium conditions. The margin calculated is 10% and is arrived at considering the error-band in heat transfer coefficients and tube wall thickness. Also, included in the 10% margin is a 5% surface allowance made for tube plugging.

The steam flow rate is defined by turbine conditions, power level, and feedwater temperature.

The water or steam side temperature and flow rate are essentially the same for both clean operation and fouled operation. However, the presence of fouling will cause an increase in the required sodium operating temperatures and flow compared to clean operation.

For power levels below about 40 percent a good portion of the inlet sodium end of the superheater and the outlet sodium of the evaporator will operate close to isothermal temperatures with small sodium to water temperature differences. This is because most of the heat transfer takes place in other portions of the modules.

5.5.3.6 Evaluation of Steam Generator Leaks

A primary design objective for the steam generators is that they be of sufficiently high quality that leaks in the sodium/water boundary will not occur. Careful design and close quality control of materials and manufacturing processes are expected to yield units which are free of common defects, and the probability of a leak in a steam generator tube is expected to be quite small. A Steam Generator Leak Detection System, described in Section 7.5.5, has been provided to allow operator action to limit the consequences of a leak. The leak detection system will alert the operator to the existence of a leak rate as low as 2×10^{-5} lb water/sec, which will allow sufficient time for operator action to prevent a significant increase in the leak rate for a broad spectrum
59 of leak rates.

As a final level of protection against tube leaks in a steam generator, the steam generators and the IHTS are being designed to withstand the effects of a large sodium water reaction (SWR). The ASME Code categories being applied in the design of the steam generators and IHTS piping and components for the large SWR event are given in Table 5.5-10.

The design basis leak (DBL) for the CRBRP was selected based upon examination of the physical processes which exist for leak initiation and growth. The conservatism of this postulated DBL will be confirmed through the LLTR test program (Ref. 12).

Two types of tests have been reported which provide information on the leak growth mechanism - small scale tests which model effects of a SWR on materials, and large scale tests which model a large water leak in a model of a steam generator. Smaller scale sodium-water reaction tests have been done to develop an understanding of the effect of a SWR on neighboring tubes in a steam generator. Three mechanisms have been identified for leak growth: self-wastage, impingement, and overheating (mechanical damage from pipe whip, although extremely unlikely, could be considered another mechanism, as
46 discussed later in this section). Self-wastage has been shown to occur for very small leaks in the range of 10^{-6}-10^{-3} lb/sec (Ref. 13). The process is depicted in Figure 15.3.3.3-1. The result of this process is a leak size of the order of 10^{-3} to 10^{-2} lb/sec. which can produce wastage on another tube in
59 the vicinity of the leaking tube.

Wastage can occur on the outside of a steam generator tube from a leak in another tube in the vicinity. Tests of this mechanism have typically been done by using a water jet directed through sodium to a target material sample. Water injection rates of approximately 10^{-4} lb/sec to 1 lb/sec have been tested. The wastage mechanism results in erosion of the target material at maximum rates of 0.001 to 0.005 inches per second (Ref. 13, 14). The wastage rate is found to be a function of the water injection rate, tube spacing, sodium temperature and leak geometry. Wastage occurring on the surface of a CRBRP steam generator tube at these rates could cause a secondary water leak from tube penetration. However, this would require at least 20 seconds to penetrate the 0.109 inch thick tube wall assuming an initiating leak of the proper characteristics to produce maximum wastage.

The size of a secondary water leak resulting from wastage is difficult to quantify since wastage tests are typically done on materials samples rather than pressurized tubes. The wastage areas observed in tests have ranged from 0.1 in^2 to 1.5 in^2. Failure areas corresponding to the highest observed wastage areas would result in water leak rates corresponding to that of a double-ended guillotine tube failure. However, the entire wastage area would not be expected to blow out. The wasted areas are typically pit-shaped with the area of the pit decreasing with depth. It would be expected that the small area at the bottom of the pit would fail, yielding a return water leak which halts the wastage. Therefore, while the

59

size of a secondary failure caused by wastage is difficult to predict, it is expected to be smaller than the leak rate corresponding to a double-ended guillotine failure.

The third mechanism for failing a tube is overheating from the thermal effects of a SWR caused by a leak in an adjacent tube. This could cause heatup in the adjacent tube and a decrease in tube strength until the tube bursts from the internal pressure. The time for tube failure was analytically investigated using boundary conditions of a temperature of 2700°F (the adiabatic SWR temperature) and heat transfer coefficients as high as 10,000 Btu/hr-ft^2°F. The minimum computed time for failure was 0.4 seconds for an evaporator tube and 0.3 seconds for a superheater tube. Measurements during large SWR tests show peak temperatures of 2300°F and heat transfer coefficients of approximately 2000 Btu/hr-ft^2°F (Ref. 15). These measured conditions, if applied uniformly around the circumference of a tube and over a significant longitudinal area, would require longer times than computed above to produce an overheating failure. Further, establishment of the conditions that could cause
such a secondary failure require a large initiating leak. 9

It i. concluded that all three leak growth mechanisms require time to develop;
hours for small leak self-wastage, tens of seconds to minutes for wastage, and
59 tenths of seconds to seconds for overheating. The secondary failures that
could potentially result from these mechanisms are expected to yield water leaks considerably less than that of a single double-ended guillotine failure. An estimate of the worst plausible leak development sequence would be as follows. A small leak (less than 10^{-3} lb/sec) is postulated to develop. This leak will be assumed to not be detected by the operators through readings from the hydrogen leak detectors. This leak then grows by self-wastage to a leak of 10^{-2} or 10^{-3} lb/sec (a leak size that gives maximum wastage rates on an adjacent tube). This leak continues, if no mitigating action is taken by the operators, until the adjacent tube wall is penetrated. The second leak is assumed to be small enough to not activate the SWRPRS by blowing out rupture disc, but large enough to yield a SWR which overheats an adjacent tube to the point of failure. This final overheating failure is assumed to activate the SWRPRS. The result of this worst plausible sequence is leaks for which the total leak flow rate is not expected to exceed that of a single double-ended
guillotine failure. 9

The Design Basis Leak is not intended to represent a realistic sequence, but, rather, to provide a conservative basis for computing design loads. The DBL is defined as a equivalent double ended guillotine (EDEG) failure of a steam generator tube which is followed by two additional single EDEG failures, spaced at 1.0 second intervals, to a total of 3 EDEG. This sequence is superimposed on a system which has been pressurized by an undetected moderate sized leak to
59 just below the rupture disk burst pressure.

The design basis leak for the CRBRP steam generators contains safety margins in the timing and also the magnitude of the assumed failures. Experimental data
59 (Ref. 15, 16, 17, 24) on the consequences of a sudden, single tube rupture
event, justify the design basis leak postulated for the CRBRP steam generator.
59 Although much of the data is not strictly prototypic of the CRBRP steam
generator modules, the results demonstrate the effects of SWR reactions in
LMFBR steam generators using similar materials, tube wall thickness, pressures, 9

water injection rates and sodium temperatures. Japanese, German and US large leak SWR tests have produced no secondary failures.

The Japanese have conducted seven large leak SWR tests ranging from seven to
ten seconds. The Germans have conducted five tests of durations 4 to 9
seconds. Six tests (in near-prototype configurations) have been conducted in
the U.S. The U.S. test have ranged from 3 to 40 seconds in duration.
Significant wastage was observed in only one U.S. test in which one tube in the
59 leaksite region exhibited a 0.016 inch reduction in wall thickness. This
corresponded to a wastage rate of 0.016 inch/sec.

British and Russian tests have demonstrated that an initial tube failure can
59 produce secondary tube failures. In the 15 tests of the NOAH series (Ref. 20)
there were three tests in which pressurized Cr-Mo secondary tubes failed. No
more than one tube failed in any test. Times to failure varied from 2.5 to 4.5
seconds. In three Super-NOAH tests (Ref. 21) there was one secondary failure
attributed to wastage/overheating. Time to failure was 14.5 seconds. Dumm's
(Ref. 15) sodium-water reaction test number 5 had pressurized secondary tubes
41 none of which failed.

The Russians (Ref. 17) report a sodium heated steam generator leak test in which a small leak (13.2×10^{-3} lb water/sec injected through a 0.030" hole) caused two tubes to fail in a period of 100 seconds. The first failure was attributed to direct wastage (from the jet impinging on the tube). The second (large) leak caused a blowout type (overheating) failure of the second tube.

In every case, the mechanism producing failures has been of the type which
requires a substantial time lag between the occurrence of the primary event and
the initiation of secondary SWR, and the magnitude of the secondary failure was
59 less than the equivalent of one double-ended guillotine failure. This delay
or lack of coherence in propagation and the limited extent of secondary tube failures has been a significant finding. These results demonstrate that water/steam injection from any secondary failures occurs at a time when potential pressure or shock effects are mitigated by the sequence of conditions resulting from the primary rupture event, i.e., by the production of a large volume of reaction product gas within the steam generator shell, and by actuation of the SWRPRS rupture discs. Also, while high frequency (1000 Hz) acoustic pressure pulses typically occur in the first few milliseconds of a guillotine-type rupture event, it has been determined that they produce no shell loading problems due to their low energy content (Ref. 15). Therefore, on the basis of large SWR experience to date, no mechanism has been found effective in causing even a single secondary failure on a time scale rapid enough to contribute to substantial IHX loadings. In addition, the number of equivalent double-ended guillotine secondary tube failures that have occurred in any test (approximately one - Ref. 16) is significantly less than the two assumed for the DBL.

In the case of small primary (initiating) leaks, the leak growth mechanisms
identified through tests do not cause instantaneous secondary failures and do
not cause secondary water leaks equivalent to a double-ended guillotine tube
break. Any delay in time to fail the additional tubes would reduce the
pressures resulting from this event. Based on existing data and analyses, the
design basis leak of a 1-tube double-ended guillotine failure followed at 1.0
second intervals by single EDEG failures to a total of 3 EDEG failures 30
41 equivalent will result in a conservative IHTS and SGS design. Analysis (Ref.
25) of data from LLTR test series (Ref. 24) has verified analysis methods used
59 in assessing the conservatism of the DBL.

59 The three tube event is not intended to represent a realistic, mechanistic sequence, but rather it provides a basis for calculating loads for the design of components and piping which are believed to be conservative for the large number of mechanistic sequences involving secondary failures which can be postulated.

59 Preliminary analyses have indicated that adjacent tubes would not fail when subjected to the peak calculated pressures for the double ended guillotine failure of one tube. Mechanical failure of adjacent tubes due to whipping of the initially failed tube is also considered unlikely. In a series of tests it was demonstrated

that, for the range of pipe sizes and schedules considered, that whipping pipes will not shear off other stationary pipes of at least the same size and schedule. A tube whip analysis is presented in Section 5.5.3.6.1. 9

The method of analysis and the calculated consequences of the design basis leak are given in Section 5.5.3.6.2. The same sequences are also postulated for the 9
59 superheater.

5.5.3.6.1 Steam Generator Tube Whip Analysis

Tests were performed using a 2.275 inch OD pipe in the configuration shown in Figure 5.5-5 as reported in Reference 10. The tests were for a more severe case of piping whip in which a steam propelled pipe whipped crosswise onto an anchored pipe. Test conditions included:

a) Steam reservoir pressure of 2000 psia

b) Steam discharge at 90° angle to axis of test section of whipping pipe.

c) The whipping pipe was designed to impact a stationary pipe.

In none of the cases of blowdown-induced dynamic bending did either the whipping pipe or the target pipe show any evidence of wall penetration. The tests showed that a whipping pipe is flexible at impact and absorbs energy in both gross bending and local deformation. In addition, the degree of distortion of the target pipes was consistently less than that of the corresponding whipping pipe. Although the range of pipe sizes and schedules in the test did not include steam generator tubes, theoretical extrapolation technique enable extrapolation of results to include the CRBRP steam generator tubes.

An analytical model was developed from the above tests:

$$\frac{KE}{V_{IP}} = (1.8 \quad \theta) \frac{P\bar{R}_{IP}}{h_{IP}} \left(\frac{\bar{R}_{WP}}{\bar{R}_{IP}} \right)^2$$

where: KE = Kinetic energy for unit length of whipping pipe measured at linear velocity of pipe end 9

V_{IP} = Volume of impacted pipe per unit length

θ = Angular displacement of whipping tube.

P = Steam reservoir pressure

$\bar{R}_{IP}$ = Mean radius of impacted pipe

$\bar{R}_{WP}$ = Mean radius of whipping pipe

h_{IP} = Thickness of impacted pipe

Applying the above equation to a whipping tube in the steam generator is as follows: The first term, angle of displacement of the ruptured pipe in the steam generator, is expected to be small, since the steam generator tubes are essentially parallel during impact. The second term is a constant since only the pressure is a variable and both the impacted and whipping tubes are equal. The third term is unity. Therefore, the kinetic energy, which is a measure of the damage potential of a whipping pipe, will be small. It is concluded that a whipping pipe in the steam generator will not damage adjacent pipes. ANSI N176, "Design Basis for Protection Against Pipe Whip" (Reference 11) provides 9
additional support for this conclusion by stating "The energy level in a whipping tube may be considered insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

59 The LLTR test completed in 1978 (Ref. 24) have strengthened the above conclusion.

5.5.3.6.2 <u>Analysis of Effects and Consequences</u>

<u>Methods, Assumptions and Conditions</u>

The large SWR events have been analyzed using the TRANSWRAP (Transient Sodium Water Reaction Analysis Program) code. The following set of events is modeled by TRANSWRAP. Immediately following the sudden introduction of water into sodium via a large leak (e.g., a full-guillotine tube rupture), accoustic waves of relatively large amplitude and short duration are generated. These acoustic waves may contain sufficient energy to activate the relief system rupture discs. The acoustic waves are generated within the first few milliseconds following the tube rupture and through reflection and reinforcement can cause peak pressures in the sodium that are above the steam-side pressure. In a time frame of milliseconds after the initial tube rupture, the sodium-water reaction creates an expanding hydrogen bubble which begins to eject the sodium from the faulted unit through the burst rupture discs and into the SWRPRS piping: Water/steam continues to enter the hydrogen bubble, and the sodium/water reaction front propagates with the surface of the hydrogen bubble. Liquid sodium is ejected through the relief lines to the reaction products separator
41 tank.

c. Pressure and volume in cover gas spaces are computed.

d. Rupture disc bursting and subsequent flow of sodium into the relief system are computed.

e. Pressure pulse reflections and resultant reinforcements and
41 rarefactions are treated.

f. The motion of the bubble-sodium interface is computed.

g. Friction effects are included for all components and piping. Resultant acoustic wave attenuation is computed.

h. Pressure pulse attenuation due to flow into tanks and reservoirs is computed.

Important conditions and assumptions built into TRANSWRAP include:

a. Instantaneous conversion of 65% of the injected water to hydrogen gas. The 65% yield has been determined (Ref. 25) to be conservative through analysis of U.S. large leak test data (Ref. 24). The British (Ref. 20) and Germans (Ref. 15) have inferred, respectively, 55% and 50% hydrogen
59 yields from their large leak tests.

b. The Na_2O remains in the H_2 bubble but has no effect on pressure or volume of the bubble.

c. The reaction products are in thermal equilibrium.

d. The effective hydrogen bubble temperatures is 1700°F. This has been determined (Ref. 25) to be conservative through analysis of U.S. large leak test data (Ref. 24).
59

e. The pump cover gas experiences isentropic compression (expansion) as a perfect gas.

f. The rupture discs are represented by dynamic models which have been conservatively calibrated against prototype rupture disc tests.

The TRANSWRAP code has been validated through analysis (Ref. 25) of the LLTR
59 Series I test data (Ref. 24).

The expanding hydrogen bubble is treated within the TRANSWRAP code as a continuum in which there are no gradients. The mass of hydrogen contained in the bubble increases as the reaction proceeds and the bubble interacts hydrodynamically with the flowing sodium at its boundaries. The bubble is considered to be a perfect gas. The sodium-water reaction is assumed to be instantaneous with a 65% yield of hydrogen.

59 Prediction of local hot spots is not within the scope of the TRANSWRAP Code. The tube over-heating analysis presented earlier in this section is believed to be a conservative bounding estimate for local hot spot effects. The probability of local hot spots is believed to be small considering the excessive amounts of high conductivity sodium and the high degree of turbulence which can be expected at the leak site.

Analytical Model

Each component in the IHTS is represented in the TRANSWRAP model as an equivalent circular cross section pipe (or a combination of equivalent pipes) of specified length, diameter, elasticity, resistance, and associated initial conditions of flow rate and pressure distribution. The model employed for the large SWR analysis is represented in Figure 5.5-3.

The method of characteristics as developed in References 6 and 7 is incorporated in the TRANSWRAP Code. The equations solved between nodal points on the characteristic grid are (p. 23, Reference 6).

$$\frac{g}{a}\frac{dH}{dt} + \frac{dV}{dt} + \frac{fV|V|}{2D} = 0$$

$$\frac{dX}{dt} = + a$$

$$-\frac{g}{a}\frac{dH}{dt} + \frac{dV}{dt} + \frac{fV|V|}{2D} = 0$$

$$\frac{dX}{dt} = - a$$

where H = pressure head; g = gravitational constant
V = fluid velocity, a = acoustic velocities
f = friction factor; D = pipe diameter
t = time; X = distance (axial)

6

Attenuation of head or acoustic pulse between nodal points, i.e., through straight runs of pipe, is thus implicitly recognized through the friction factor f, which is input for each calculational segment within the model. Three types of junction points or joints connecting straight runs of pipe are available within TRANSWRAP; butt joints, tee joints and end joints (dead ends). Attenuation of acoustic pulses in passing through each type of junction is incorporated through the boundary conditions imposed on the above equations. The boundary conditions, as derived in Reference 7, for each junction type are given below.

(1) Head due to reflections at a dead end:

$$H = 2F(t - \frac{L}{a})$$

F = travelling wave function evaluated from boundary conditions

L = length from disturbance to dead end

The meaning of this expression is that at a dead end, the reflected wave is equal to the incident wave and is of the same sign.

(2) At a butt joint with change of diameter, both reflection and transmission are recognized through the factors:

$$\text{Reflection Factor} = \frac{(A_1/a_1) - (A_2/a_2)}{(A_1/a_1) + (A_2/a_2)}$$

$$\text{Transmission Factor} = \frac{2A_1/a_1}{(A_1/a_1)+(A_2/a_2)}$$

where A is cross-sectional flow area and a is acoustic velocity.

(3) Transmission and reflection factors at a tee joint are:

$$\text{Transmission Factor} = \frac{2(A_1/a_1)}{(A_1/a_1)+(A_2/a_2)+(A_3/a_3)}$$

$$\text{Reflection Factor} = \frac{(A_1/a_1)-(A_2/a_2)-A_3/a_3)}{(A_1/a_1)+(A_2/a_2)+(A_3/a_3)}$$

This treatment of attenuation through friction factors and boundary conditions is generally conservative in that the one-dimensional model cannot account for multi-dimensional attenuation-effects, e.g., in the transmission of an acoustic wave around an elbow. Preliminary measurements in water (Reference 8) indicated a reduction in acoustic wave magnitude of from 10 to 30% in passing through an elbow. This is about an order-of-magnitude higher than what is computed in TRANSWRAP using one-dimensional fraction factors. 6

Results

TRANSWRAP calculations will be performed for both evaporator and superheater tube failures for the Design Basis Leak. Results of the analysis will be
59 provided when they become available.

For the Design Basis Leak sequences in the superheater, the peak pressures in
59 all the major components are lower than for the DBL in the evaporator leak
sequence except in the IHX. The lower pressures are the result of the lower
water mass flowrate throughout the transient.

59 The steady-state sodium flow rates and pressure drops throughout the IHTS prior
to the steam generator tube rupture are represented in the TRANSWRAP model.
Following the tube rupture, the expanding bubble of sodium/water reaction
products is treated as a continuum of perfect gas which interacts
hydrodynamically with the flowing sodium. For the Design Basis Leak in an
evaporator, the bubble is predicted to expand away from the leak in both
directions, i.e., the sodium which at steady state flows into the evaporator is
reversed while the sodium flowing out of the evaporator is accelerated. As the
sodium originally within the evaporator below the leak site is displaced by the
reaction products, it is driven through the evaporator outlet tee. The bulk of
the sodium and reaction products are expelled through the interconnected SWRPRS
relief line. However, a portion is also predicted to flow towards the pump.
59 The SWRPRS relief line is cleared of liquid sodium after about 1 second. Gas
blowdown through the cleared relief line decreases the bubble pressure.
Therefore, peak system pressures occur during the first second of the event.
59 It is expected that the sodium flow in the pump suction line will reverse
before the gas bubble reaches the pump. The sodium will then drain back toward
the relief line (low point in the system). Loop draining will be completed by
manual opening of the sodium dump valves. 14

In the unlikely event that the flow in the IHTS does not reverse and the gas bubble reaches the pump, no damage to the coolant boundary of the pump is expected. It is conservatively assumed that the sodium/gas interface reaches the pump inlet about 8 seconds after the SWR is initiated. However, all PHTS and IHTS pumps are tripped by the PPS by approximately 1 second after SWR initiation. Per the specified pump transient, by seven seconds the pump inlet pressure is reduced to the order of 50 psi, and the pump speed will be reduced to the order of 40% full speed.

Since the pump main motor is tripped long before the bubble could arrive at the
pump inlet, there is no possibility of pump overspeed and subsequent missile
generation. Uneven hydraulic loads and loss of sodium would eventually result
in bearing damage and seizure of the pump. 31

For the Design Basis Leak in the superheater, the reaction products bubble is predicted to expand away from the leak site in both directions also. Reaction products above the leak site and sodium which normally flows into the superheater are expelled through the SWRPRS relief line connected to the superheater inlet tee. Since flow reversal in the superheater sodium inlet line does not occur because of continued flow from the pump, reaction products cannot enter the IHTS hot leg. Reaction products below the leak site are predicted to accelerate downstream toward evaporator relief lines. The sodium drains from the
loop through the relief lines and the sodium dump lines similar to that described 14
above for the evaporator event.

Conclusions

Based on the information provided above, it is concluded that systems
43 and components designed to the ASME Section III categories given in Table 3.2-5
6 using the loadings given in Table 5.5-11 will maintain their

integrity for a large SWR event. Therefore, the integrity of the barrier separating the primary radioactive sodium from the non-radioactive intermediate sodium is maintained preventing the release of radioactive material to the environment.

5.5.3.7 Pipe Leaks

Pipe leaks are categorized into two categories. Category 1 is labeled
identified leakage and includes all leakages into a closed system. Identified
leakage includes piping seal or valve packing leaks that are directed to a
collection tank or a pump. These leaks occur in a system where it is not
practical to make the components 100% leaktight. The existence of identified
leakage is known in advance and is provided for in the system design. Category
2 is labeled unidentified leakage and encompasses all leakage that is not into
a closed system. CRBRP piping systems are designed to preclude unidentified
leaks, in that, the piping is of all welded construction and all welds are
59 inspected and leak tested prior to putting the piping into service. Pipe leaks
were postulated in each piping run of the Steam Generator System and 13
investigated to identify those areas where unexpected loading and possible
damage may result. Section 3.6 presents a discussion of the criteria for
postulating leaks, their effects on nearby equipment and the design measures to
be used to prevent propagation of damage. Section 15.3 provides discussion of
15 the effects of such leaks on the core cooling function. Utilizing three
independent loops in separate cells precludes propagation of a pipe leak
failure between loops; each loop alone is adequate for decay and sensible heat
rejection. The common points of the loops, the main steam header and the
feedwater header, are beyond isolation valves in each loop (see Figures 10.3-1
and 10.4-4); hence a single failure cannot cause a loss of function in more
15 than one loop.

Small unidentified leaks present no special problems to the steam plant. Water
or steam leaks will be detected visually during routine inspection of the
plant. In addition, experience with sodium and steam piping has resulted in
expected failure rates (external leakage) on the order of 1 per 10^6 hr per loop
for steam piping. These failure rates are based upon data reported in 25
References 5.3-18 and 19.

5.5.3.8 Inadvertent Operation of Valves

A discussion of the design basis events and their appropriate limits for this plant is given in Section 15.3. As described in Appendix B of this report, the events in Chapter 15 have been selected to envelope the most severe change in critical parameters from events which have been postulated to occur during planned operation.

5.5.3.9 Performance of Pressure-Relief Devices

The safety-relief valves installed in the water-steam system of the steam generation system will be as used in conventional water-steam service. The service conditions are not expected to be significantly different from those encountered in a fossil-fired power plant with similar pressures and temperatures. In case of a major steam generator failure which results in a large sodium-water reaction and activation of the evaporator water dump system, inert gas pressure is introduced into the failed unit as the pressure drops toward the sodium pressure. This inert gas is at a pressure higher than the sodium pressure to prevent entry of sodium into the water-steam system of the evaporator. In addition, the failed steam generator is isolated from the rest of the steam generation water-steam system during such an incident so that, even if the inert gas is not fully able to prevent entry of sodium into the water-steam side of the failed evaporator unit, the sodium cannot get into the rest of the water-steam system loop.

To ensure proper performance of the pressure-relief devices protecting the IHTS, several methods are utilized to minimize the effects of oxide concentrations, including:

a. The pressure-relief device consists of two rupture discs in series, eliminating potential malfunction of moving parts.

b. The upstream pressure relief device is located in liquid sodium and is not exposed to sodium vapor. The downstream relief device is located in an inert atmosphere free of sodium vapor.

c. The pressure relief devices are located near the main high temperature sodium flow stream and trace heating will be provided for the devices and adjacent piping.

A more detailed description of these devices is contained in Section 5.5.2.4.

5.5.3.10 Operational Characteristics - Design Transients

The overall plant duty cycle list, event classification according to the ASME Section III categories of normal, upset, emergency and faulted, and the event frequencies, is given in Appendix B of this document. Further information as to application of the transients to component design is contained in Section 5.7.3.

5.5.3.11 Material Considerations

5.5.3.11.1 Structural Materials for Elevated Temperature Service

Extrapolation of Creep-Rupture Data

The data base used for extrapolation of creep-rupture properties is that employed in ASME Code Case 1592. For the Steam Generator Modules materials,
59 | 35 the maximum design temperatures are $965^{\circ}F$ for 2-1/4 Cr-1 Mo steel, and $925^{\circ}F$ for 304 SS. The 304 SS components for which design requirements have been established are associated with the $Na-H_2O$ leak detection subsystem and operate at low pressures. Therefore, creep is not a major consideration for 304 SS in the SGS.
59 | 35

Appropriate allowable stress values for 2-1/4 Cr-1 Mo are available in the Code Case 1592 out to 300,000 hours. The allowable stresses for decarburized 2-1/4 Cr-1 Mo have been modified to account for lower carbon content based upon test data. These allowables provide a conservative basis of design for the steam generator modules.

From steps 1 and 2, the transition configuration evolved presently is ferritic - Alloy 800 - austenitic steel with ER Ni Cr-3 as the weld material between
59 ferritic/Alloy 800 and 16-8-2 stainless steel filler between Alloy 800/ stainless steel. Step 3 is being taken at ORNL. The hot-wire Gas Tungsten Arc welding technique with necessary equipment is being developed at GE as part of satisfying the steps 4, 5 and 6. For step 7, two test programs are planned:

(a) Creep-rupture properties of the specimens containing transition welds will be determined and compared with those of the base and weld materials. One group of specimens will include ferritic/ER Ni Cr-3
59 82/Incoloy 800 joints and the other Alloy 800/16-8-2 SS Filler/ austenitic stainless steel and

(b) a test plan is being developed to test a large size (about 10 inches) spool with the chosen transition configuration under thermal cycling and long time high temperature heat conditions. Under Steps 5 through
41 8 the transition welds will be given at least two independent (i.e., radiographic and ultrasonic) examinations and the transition spools will be finished machined on the inside and outside to eliminate notches and other discontinuities due to shrinkage and distortion. The welding and finishing procedures will be carefully developed and incorporated in the transition joint Equipment Specification.

The ferritic/austenitic joints located presently in each loop are as follows:

(1) Main Sodium Piping

(a) Hot Leg - A 316SS/2 1/4 Cr - 1 Mo joint is located upstream of Superheater inlet nozzle, has size of 26 inches O.D. and is ASME Class 1.

(b) Cold Leg - Two 2 1/4 Cr - 1 Mo/304SS joints are located downstream of each of evaporator outlet nozzles, have size of 18 inches O.D. and are ASME Class 1.

(2) Auxiliary Sodium Piping (Size 6 inches or less O.D.)

(a) Cover gas equalization line from expansion tank to sodium dump tank - One 304SS/carbon steel joint is located near the dump tank nozzle and is ASME Class 2.

(b) Dump line from hot leg to sodium dump tank - One 316SS/carbon steel joint is located near the dump tank nozzle and is ASME Class 2.

(c) Dump line from cold leg to sodium dump tank - One 304SS/carbon steel joint is located near the dump tank nozzle and is ASME Class 2. 17

(d) Dump line from superheater dump nozzle to sodium dump tank - One 2 1/4 Cr - 1 Mo/304SS joint is located near the Superheater dump nozzle and is Class 1; another 304SS/carbon steel joint is located near the dump tank nozzle and is ASME Class 2.

(e) Dump line from two evaporator dump nozzles to sodium dump tank - Two 2 1/4 Cr - 1 Mo/304SS joints are located near evaporator dump nozzles and are ASME Class 1; and one 304SS/carbon steel joint is located near the dump tank nozzle and is ASME Class 2. The two evaporators are connected to a tee which connects to a dump tank nozzle, the connecting piping being 304SS.

(f) Vent line from two evaporators to expansion tank - One 2 1/4 Cr - 1 Mo/304SS joint is located near the expansion tank nozzle and is ASME Class 1.

(g) Vent line from superheater to expansion tank - One 2 1/4 Cr - 1 Mo/316SS joint is located near the superheater vent nozzle and is ASME Class 1. 17

5.5.3.11.3 Austenitic Stainless Steel

43 The materials presently expected to be used in the SGS are: Type 304 austenitic stainless steel, 2-1/4 Cr-1 Mo steel, and carbon steels. The material of which each component is fabricated is shown in Table 5.5-3.
43

Cleaning and Contamination Protection Procedures

The steam generators will not be chemically cleaned once fabrication is started. Only local wiping with a rag and solvents such as acetone is approved. The 2-1/4 Cr-1 Mo ferritic components in the completed steam generators will be protected against rusting by inerting the tube side and shell side with dry nitrogen gas at a positive pressure after fabrication and hydrostatic testing, and during shipment to the site.

Rust prevention methods for components prior to assembly will be provided following review and approval of supplier proposed techniques. The methods under consideration for the steam generator tubing are as follows:

a. Sealing individual tubes in plastic bags with dry nitrogen.

b. Vapor phase inhibitor (VPI) inserted within the tubing.

c. Generation of a protective oxide layer by a final furnace treatment. 14

Solution Heat Treatment Requirements

Solution heat treatment requirements for the unstabilized austenitic
43 stainless steel to be employed in the SGS are as described in 5.3.3.10.2.

Control of Delta Ferrite

Control of delta ferrite content in austenitic stainless steel welds will be as described in 5.3.3.10.2.3 of this PSAR, in compliance with ASME Code Case 1592. 25

Corrosion allowances for both steam and sodium side 2-1/4 Cr-1 Mo steel will be included in the design. These corrosion allowances are based on recommendations from the steam generator module designer (Ref. 3).

No specific protection is required for protecting Type 304 SS or 2-1/4 CR-1 Mo steels against intergranular attack, stress-corrosion or general corrosion, provided that specified sodium purity is maintained.

In water or steam, carbon steel and 2-1/4 Cr-1 Mo steel are susceptible to caustic gouging and possibly caustic stress corrosion cracking. Maintaining the feedwater and steam drum purity levels as stated below will prevent these forms of localized attack. For normal operation other than start-up conditions, the feedwater purity at the drum inlet will be specified as follows:

	Feedwater Impurities	Steady State
	Suspended solids ppm max.	0.016
	Dissolved oxygen ppm max.	0.007
	Silica, ppm max.	0.02
	Iron as Fe, ppm max.	0.01
46	Copper as Cu, ppm max.	0.0015
	Hydrazine (residual) ppm max.	0.015
	Conductivity (cation) @ 77°F-micro-mho/cm max.	0.3
45	pH at 77°F for austenitic heaters	8.7-9.1
	Chlorides, ppm max.	0.009
	Sodium, ppm max.	0.001

49 46

2 59 Limited duration operation with impurity levels above specified limits is allowable for periods not to exceed 24 hours in special instances. These special instances are defined to include condensate polishing system perturbations, such as those immediately associated with a termination of regeneration.

41 59

Corrosion impurities may enter the feedwater system through condenser leakage and/or poor makeup water. To guard against damage from such sources, the feedwater and steam drum water are maintained at levels within stated limits by full flow demineralization and continuous steam drum drainflow (blowdown) at
59 a nominal rate of 10% of full power steam flow (See Section 10.4.7).

To determine the feedwater and recirculation water quality, in-line analyses for conductivity and sodium content are performed for the feed water entering
59 the steam generator system and for the evaporator inlet water. The steam drum water is monitored at the drain or blowdown line for the same chemicals. The condenser hot-well is monitored for conductivity and sodium ions to guard
48 against condenser leakage. The demineralizer effluent is guarded against impurities break-through by in-line measurements of silica, conductivity and sodium. Finally, the feedwater train is monitored downstream of the deaerator for pH and oxygen content to prevent potential corrosion of this portion of the steam system. An alarm is coupled with the most critical in-line measurements
59 to signal departure from specified levels.

5.5.3.11.5 Compatibility with External Insulation and Environmental Atmosphere

Compatibility of austenitic stainless steel with external insulation is assured as set forth in 5.3.3.10.4. Strict control of halide contents in insulation materials is required. Carbon steels and 2-1/4 CR-1 Mo are compatible with
59 external insulation during normal operation in the absence of excessive moisture. Excessive moisture is prevented by quality controlled installation and operating procedures.

5.5.3.12 Protection Against Environmental Factors

Protection for the principal components of the SGS against environmental factors is provided by the structural integrity of the Steam Generator Building. Environmental factors to be considered include the following:

Fire Protection - See Section 9.13.
Flooding Protection - See Section 3.4.
Missile Protection - See Section 3.5.
Seismic Protection - See Section 3.7 and 3.8.
Accidents - See Section 15.6.

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* 1. W.H. Yunker, "Standard FFTF Values for the Physical and Thermophysical Properties of Sodium" WHAM-D-3, July 6, 1970.

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26 * References annotated with an asterisk support conclusions in the Section. 9
Other references are provided as background information.

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26 * References annotated with an asterisk support conclusions in the Section. Other references are provided as background information.

TABLE 5.5-1

STRUCTURAL DESIGN TEMPERATURE, PRESSURE AND MINIMUM TEST

PRESSURE FOR SGS COMPONENTS

COMPONENT	DESIGN TEMP (°F)	DESIGN PRESSURE (PSIG)	MINIMUM HYDRAULIC TEST PRESSURE (PSIG)
Evaporator Module			
Shell	885	325	406
Tubes	885	2400	3000
Superheater Module			
Shell	965	325	406
Tubes	965	2200	2750
Steam Drum	650	2200	3300
SGS Water/Steam Piping			
59 Feedwater Inlet to Drum Iso. Valve	500	3350	5025
Drum Iso. Valve to Drum	650	2200	3300
Drum to Pump	650	2200	3300
Pump to Evaporator	650	2450	3675
Evaporator to Drum	650	2200	3300
Drum to Superheater	650	2200	3300
Superheater to Isolation Valve	935	1900	2850
Recirculation Pump	650	2450	3675
Leak Detection Subsystem Piping	985	325	487
Sodium Dump Tank and Piping	700	55	83
Water Dump Tank Piping	420	300	450
Reaction Products Separation Tank 41	800*	125*	188

* Design pressures and temperatures shown are not coincident in time.

TABLE 5.5-1 (Continued)

COMPONENT	DESIGN TEMP. (°F)	DESIGN PRESSURE (PSIG)	MINIMUM HYDRAULIC TEST PRESSURE (PSIG)
SWRPRS Piping			
Steam Generator to Reaction Products Separation Tank	800*	300*	450
Reaction Products Separation Tank to Centrifugal Separator	800* max.	125*	188
41 Centrifugal Separator to Vent Stack	200* max.	100*	150

*Design pressures and temperatures shown are not coincident in time.

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TABLE 5.5-5

SGS PUMP AND VALVE DESCRIPTION

	PUMPS	ACTIVE	INACTIVE	ACTUATING SIGNAL
	Recirculation Pump		X	N/A
	VALVES			
59	Pump Suction Isolation		X	Manual (Remote)
	Evaporator Inlet Isolation		X	SWRPRS
	Evaporator Inlet Water Dump		X	SWRPRS
41	Evaporator Outlet Relief	X		SWRPRS**,High Pressure-SWRPRS** or High Pressure Evaporator (Steam)
59 46	Steam Drum Relief	X		High Pressure - Steam Drum
	Superheater Inlet Isolation		X	SWRPRS
	Superheater Relief	X		SWRPRS**,High Pressure SWRPRS** or High Pressure Superheater (Steam)
41	Superheater Outlet Isolation	X		SWRPRS**, OSIS or Low Superheater Outlet Pressure
	Steam to SGAHRS HX		X	Manual (L.O.)*
59	Water from SGAHRS HX		X	Manual (L.O.)*
	Steam to SGAHRS Auxiliary FW Pump		X	Manual
	Feedwater from SGAHRS		X	Manual (L.O.)*
59	Main Feedwater SGB Isolation	X		SWRPRS**, High Steam Drum Level, Low Steam Drum Pressure, Cell Temp and Humidity
43 41	Main Feedwater Drum Isolation		X	High Steam Drum Level
	Main Feedwater Check Valve		X	Simple Check
43	Main Feedwater Control	X		High Steam Drum Level, Cell Temp and Humidity
59 41	Startup Feedwater Control	X		High Steam Drum Level, Cell Temp and Humidity
	Evaporator Outlet Check Valve		X	Check Valve
	Superheater Outlet Check Valve		X	Check Valve
41	Steam Drum Drain Isolation	X		SWRPRS**, SGAHRS Initiation, Low Steam Drum Pressure

* L.O. - Locked open

41 ** This function is not safety active

TABLE 5.5-5 (Continued)

Valves	ACTIVE	INACTIVE	ACTUATING SIGNAL
SWRPRS Stack Check Valve		X	Check Valve
SWRPRS Atmospheric Seal Bypass		X	Manual
41 Sodium Dump Tank Pressure Relief		X	High Sodium Dump Tank Pressure
59 Evaporator Water Dump Tank Drain		X	Manual

42

TABLE 5.5-6

SGS LOADING CONDITIONS

35 ASME III Code Class 3 system components will be designed considering the following load combinations:

1 Pumps (Recirculation Loop)

Operating Condition	Component	Load	Stress Limit
See Note 1	Pump Case	Design Pressure	Section III
		Design Temperature	Allowable Stress
	Cover	Design Pressure	Section III
	Bolting	Design Temperature	Allowable Stress
		Safe Shutdown Earthquake	
		Pump Thrust	
		Weight	
		Gasket Loads	

Note 1: Design pressures and temperatures of the recirculation system components are established using pressures and temperatures occurring during emergency and faulted transients. The design temperature is not exceeded during these transients. The design pressure may be exceeded by not more than 10% during these transients. Normal and upset conditions are not controlling.

2 Valves (Recirculation Loop and Main Water/Steam)

35 The valve pressure retaining parts designed to ASME - III Class 3 will withstand seismic forces and pipe loads of the SSE as well as design pressure and temperatures. On other parts, if earthquake needs are to be considered, the following applies:

Operating Condition	Loads
Upset	1. Normal Operating
	2. OBE
Faulted	1. Normal Operating
	2. SSE

TABLE 5.5-7

SGS PIPING AND THEIR DESIGN CHARACTERISTICS

	PIPING AND HEADERS	COMPONENT SIZE	NO. PER LOOP	NO. PER PLANT	ASME CODE SEC. III CLASS	DESIGN REQUIREMENTS
	1. Steam Generator Subsystem & Feedwater Subsystem					
59	SGB Wall to Drum Feedwater Isolation Valve	10", sch. 160	1	3	3	3350 psig, 500 F
59	Feedwater Drum Isolation Valve to Steam Drum	10", sch. 160	1	3	3	2200 psig, 650 F
	Drum to Pump Inlet Header	10", sch. 140	4	12	3	2200 psig, 650 F
	Pump Headers (Inlet)	18", sch. 140	1	3	3	2200 psig, 650 F
	Pump Inlet Header to Pump	18", sch. 140	1	3	3	2200 psig, 650 F
	Pump to Pump Discharge Tee	12", sch. 160	1	3	3	2450 psig, 650 F
	Pump Discharge Tee	12", sch. 160	1	3	3	2450 psig, 650 F
	Pump Discharge Tee to Evaporator Isolation Valve	10", sch. 160	2	6	3	2450 psig, 650 F
	Evaporator Isolation Valve to Evaporator	10", sch. 140	2	6	3	2400 psig, 650 F
59	Evaporator to Drum	16", sch. 140	2	6	3	2200 psig, 650 F
	Drum to S.H.	12", sch. 140	1	3	3	2200 psig, 650 F
	S.H. to Isolation Valve	16", sch. 160	1	3	3	2200 psig, 650 F
	Isolation Valve to SGB Wall	16", sch. 160	1	3	3	1900 psig, 935 F
59	Startup Feedwater Control Valve Piping	4", sch. 160	1	3	3	3350 psig, 500 F
	2. SWRPRS					
	Sodium Rupture Disc Discharge Lines to Separator Tanks	24", sch. XS	3	9	3	300 psig, 800 F
59	Separator Tanks to SGB Roof	16", sch. 40	1	3	3	125 psig, 200/800 F
41	SGB Roof to Flare Tip	16", sch. 40	1	3	ANSI B31.1	125 psig, 100°F
54						

TABLE 5.5-7 (Continued)

SGS PIPING AND THEIR DESIGN CHARACTERISTICS

	PIPING AND HEADERS	COMPONENT SIZE	NO. PER LOOP	NO. PER PLANT	ASME CODE SEC. III CLASS	DESIGN REQUIREMENTS
54	Separator Tank Equalizer	24", sch XS	1	3	3	125 psig, 800 F
	3. Sodium Dump Subsystem					
	Sodium Dump Tank Inlet Piping f/Dump Valve	4", sch. 40	5	15	2	50 psig, 965 F
	Sodium Dump Tank Vent Line to Stack	8", sch. 40	1	3	3	*
	Sodium Dump Tank Equilizer Gas Line to Isol. Valve	6", sch. 40	1	3	2	5[illegible] [illegible]sig, 700 F
	4. Water Dump Subsystem & Relief Lines					
	Steam Drum Relief Valve Discharge Piping	6", sch. 120	2	6	3	900 psig, 535 F
		8", sch. 40	2	6	3	300 psig, 420 F
	Superheater Relief Valve Discharge Piping	6", sch. 80	3	9	3	900 psig, 840F
	Evaporator Relief Valve Discharge Piping	10", sch. 40	2	6	3	300 psig, 420 F
		8", sch. 40	2	6	3	300 psig, 420 F
		6", sch. 80	4	12	3	900 psig, 535 F
	Evaporator Water Dump Valve Inlet Piping	3", sch. 160	[illegible]	12	3	2400 psig, 650 F
	Evaporator Water Dump Valve Discharge Piping	6", sch. 80	2	6	3	900 psig, 535 F
		10", sch. 80	2	6	3	900 psig, 535 F
	Water Dump Tank Discharge Piping	6", sch. 80	1	3	3	300 psig, 420 F
59	Water Dump Tank Inlet Piping	10", sch. 80	1	3	3	900 psig, 535 F

41 * Design requirements will be provided after final evaluation of transients.

TABLE 5.5-7 (Continued)

SGS PIPING AND THEIR DESIGN CHARACTERISTICS

	PIPING AND HEADERS	COMPONENT SIZE	NO. PER LOOP	NO. PER PLANT	ASME CODE SEC. III CLASS	DESIGN REQUIREMENTS
59	5. Drum Blowdown Drum to SGB Wall	6", sch. 160	1	3	3	2200 psig, 650 F
59 47	6. Na-H_2O Leak Detectors IHTS to Leak Detection Sodium Isolation Valves	3/4"	4	12	2*	325 psig, 985 F
47	Isolation Valves to Leak Detection Modules	1/2"	4	12	3	325 psig, 985 F
41						

* Designated ASME III, Class 2, optionally upgraded to Class 1.

TABLE 5.5-8

SGS SAFETY RELIEF VALVES

	VALVE LOCATION	SIZE (Effective Flow Area)	NO. PER LOOP	NO. PER PLANT	DESIGN REQUIREMENTS	SAFETY RELIEF SETTING
46	Steam Drum *	4.18 in^2	2	6	2200 psig, 650° F	2170, 2220 psig
	Evaporator	4.18 in^2	4	12	2200 psig, 650° F	2250, 2280 psig
59	Superheater	4.18 in^2	3	9	1900 psig, 935° F	1800, 1850, 1900 psig
41						

46 * Safety Function Only

TABLE 5.5-9

35 THERMAL/HYDRAULIC DESIGN OPERATING CONDITIONS

	Sodium	
41	Superheater Inlet Temperature, ^{o}F	922
41 35	Evaporator Outlet Temperature, ^{o}F	651
	Flow per Loop, lbm/hr	13.49×10^{6}
	Water/Steam	
	Turbine Steam Pressure, psia	1465
	Turbine Steam Temperature, ^{o}F	900
	Turbine Steam Flow Per Loop, lbm/hr	1.11×10^{6}
35	Feedwater Temperature, ^{o}F	468
	Recirculation Ratio	2:1
	Steam Piping Pressure Drop	
27	Superheater-Turbine, psi	85
41	Steam Drum-Superheater, psi	32
35	Evaporator-Steam Drum, psi	7

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TABLE 5.5-10

SWR DESIGN BASIS

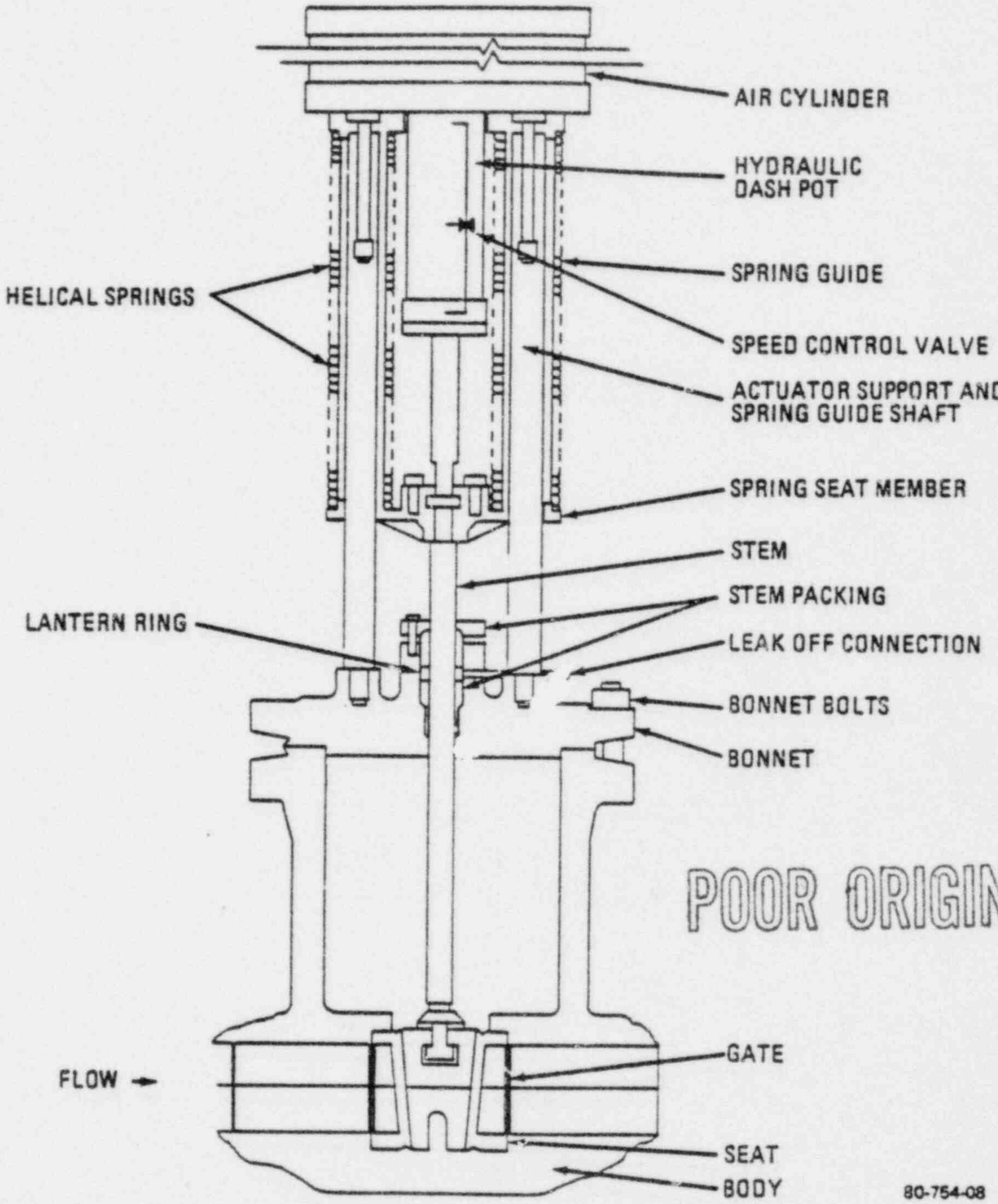
LEAK DESCRIPTION(1)	FAILED STEAM GENERATOR	AFFECTED RPST	OTHER STEAM GENERATORS AND IHTS EQUIPMENT IN THE AFFECTED LOOP
	ASME CODE CATEGORY		
Small Leak in One Tube	Upset	Normal	Upset
44 One Guillotine Tube Failure Followed by Six DEG* (Equivalent) Secondary Tube Failures	Faulted	Faulted	Emergency

(1) See Section 5.5.3.6 for detailed descriptions and basis

* Double ended guillotine

59 TABLE 5.5-11

HAS BEEN DELETED



POOR ORIGINAL

Figure 5.5-2A. MAIN STEAMLINE ISOLATION VALVE (SUPERHEATER ISOLATION VALVE OUTLET)

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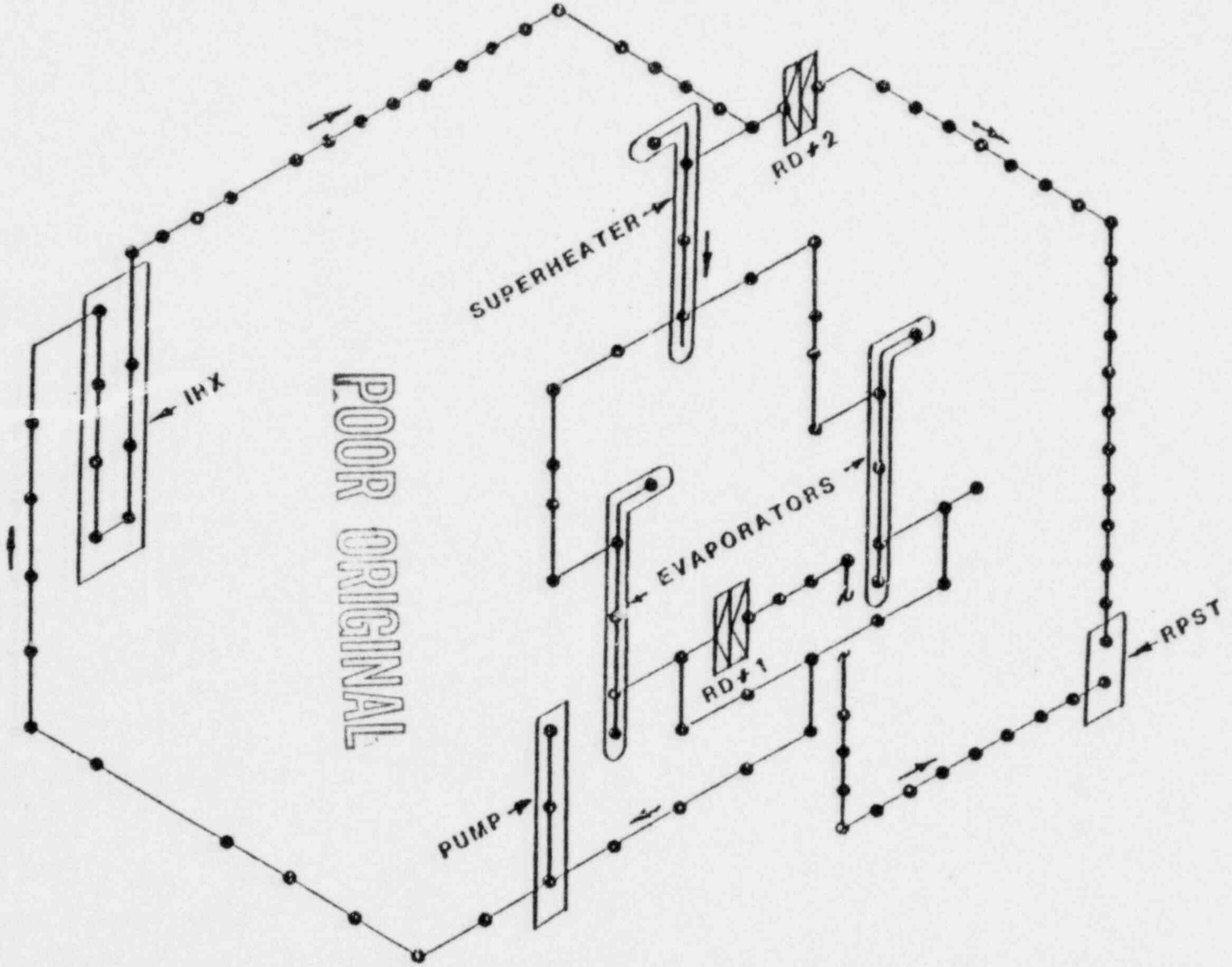
FIGURE 5.5-3 TRANSWRAP Model of IHTS and Relief System



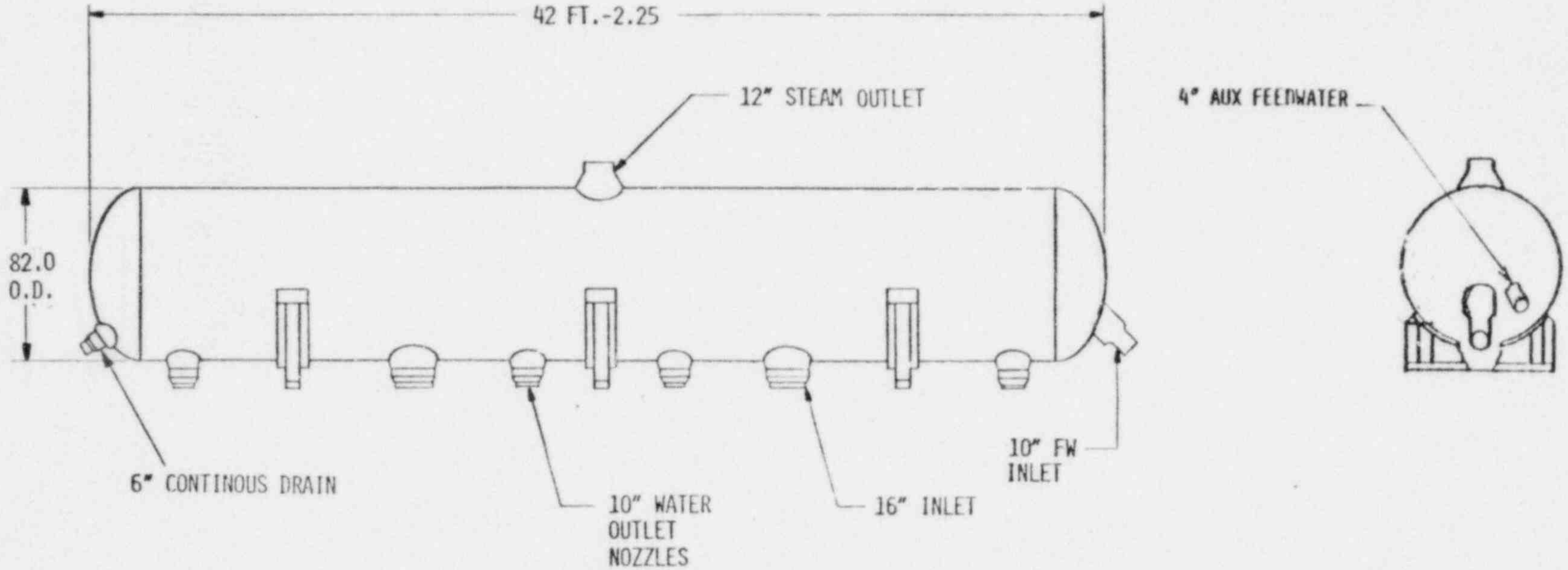


Figure 5.5-4 Steam Drum Outline

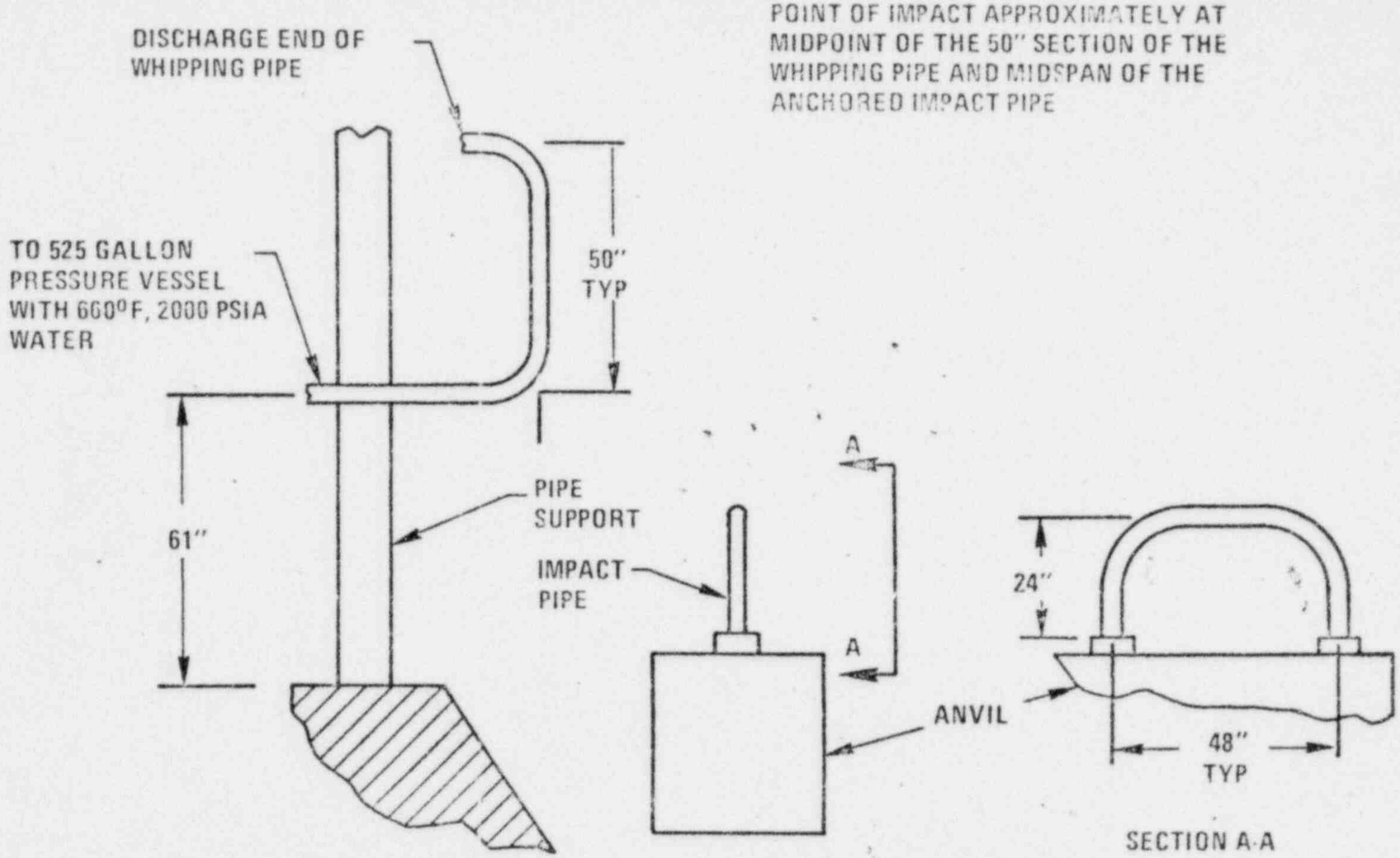


Figure 5.5-5. Pipe Whip Test Configuration, Tests 3 And 4 Of Reference 9

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Protected Water Storage Tank (PWST)

The PWST will provide sufficient water so that the heat removal function can be accomplished by venting of steam until such time as the heat load is reduced to a level which permits using the Protected Air Cooled Condensers (PACC's) as the heat sink.

Sufficient water is provided to accomplish the venting, plus an additional amount to assure that valve leakage and losses due to postulated pipe failures do not result in premature depletion of the protected water supply. Postulated pipe failures, including full ruptures, can be accommodated with operator initiated isolation of the break. Valve leakage makeup for 30 days can be accommodated in addition to recirculation pump seal leakage for a 24-hour
17 period.

Protected Air-Cooled Condenser (PACC)

PACC will provide the long term heat removal capability to remove reactor decay heat when the SGAHRS is activated following a reactor shutdown. The capacity of the PACC on a single loop will be sufficient to assume the total decay heat load prior to the time that the PWST water is depleted below the level required to provide the 30-day emergency makeup water supply with heat removal being
17 accomplished via the PACC.

The PACC shall provide capability for removing reactor decay heat during a long term planned outage.

Design Parameters

The design pressure, temperature and other significant parameters for each of the SGAHRS components are listed in Table 5.6-1. The SGAHRS will be routinely tested during plant life and prior to installation as required to meet the
17 intent of Section XI of the ASME Code. The SGAHRS is Seismic Category I and will be operative following any anticipated, unlikely, or extremely unlikely plant condition. Seismic loading conditions are specified in Section 3.7.

5.6.1.1.2 Applicable Code Criteria and Cases

The SGAHRS pressure vessels and appurtenances, piping, pumps, valves and other components shall be constructed in accordance with Section III of the ASME Code
43 as indicated in Table 5.6-2. Code Cases 1606 and 1607 will be applied to Class 2 and 3 piping and vessels subjected to the SGAHRS operating conditions. Code Cases 1729 and 1739 will be applied to AFW pumps. Code Case 1797 will be
43 17 applied to the finned tubing in the PACC. Effective dates of Codes and Code Cases will be those that are in effect at the time the contract is let.

5.6.1.1.3 Surveillance Requirements

The only toughness degradation phenomenon which is a consideration is strain aging. Since the degradation is anticipated to be small in magnitude and localized in nature, and since post fabrication stress relief will be employed if necessary (see Section 5.6.1.3.10.2) fracture toughness surveillance specimens are not considered to be required for the SGAHRS.

5.6.1.1.4 Material Considerations

The following information pertains to materials which will be used in the SGAHRS.

High Temperature Design Criteria

Code Case 1481 will be applied to the superheater vent valves and the superheater vent valve piping since they have a design temperature above 800°F. The remainder of the SGAHRS components and piping have a design temp-
26 erature below 800°F so no high temperature design criteria are necessary.

44

5.6.1.2.1.3 Surveillance and In-service Inspection

The SGAHRS system will be inspected in accordance with the intent of
17 58 Section XI Division 1 of the ASME Code.

5.6.1.2.1.4 Protection Against Accelerated Corrosion and Material Degradation

In water and/or steam, carbon steels are susceptible to pitting in the presence of chloride and oxygen. Furthermore, below 550°F, these materials are susceptible to caustic gouging and, perhaps, caustic stress corrosion cracking. Maintaining the water purity consistent with the requirements for chlorides, caustics and oxygen for short term operation will prevent these forms of localized attack.

Carbon steel is also susceptible to hydrogen embrittlement under SGAHRS operating conditions. However, maintaining the specified water purity will prevent this occurrence. Administrative procedures will be established to assure that water purity will be maintained.

5.6.1.2.1.5 Material Inspection Program

The SGAHRS material inspection program will be based on the require-
58 ments of the ASME Code, Section III, for carbon steel and 2¼ Cr 1mo. steel.

5.6.1.2.2 Material Properties

The materials used in the SGAHRS are described and discussed in
58 Section 5.6.1.1.4.

5.6.1.2.3 Component Descriptions

The major SGAHRS components have been designed with sufficient margin to assure that they will provide adequate cooling after a plant shutdown from power operation up to 115% of rated power. The decay heat levels shown in Figure 5.6-6 were used for component sizing and system response calculations for SGAHRS. 25

5.6.1.2.3.1 Protected Air Cooled Condensers (PACC)

Component Description

The condenser is a tube configuration manifold of once-through steam flow. A unit consists of two half-size carbon steel tube bundles, each having its own, separately controlled air supply. Steam/water flow is accomplished by natural circulation. Air flow is provided by fans (one per bundle) with variable position louvers for flow control. The air flow control range is from 10% to 100% of rated capacity, with control accomplished by sensing and maintaining steam pressure at the desired set point. The louvers are designed to fail open and capacity is provided for manual remote and local override.

Design Data

Design Conditions:

Pressure	2200 psig
Temperature	650°F

Thermal Hydraulic Performance:

Heat Removal	15 MWt (7.5 MWt per tube bundle)
Steam Pressure	1450 psig
Steam Temperature	592°F
Moisture	0%
Condensate Temperature	592°F
Air Temperature	100°F
Air Pressure	14.3 psia

Design Criteria

The power supplies to the PACC fans, instrumentation and controls are Class IE. The Instrumentation and Control System is a safety related system and as such will meet the requirements of the regulatory guides and standards as listed in Tables 7.1-2 and 7.1-3 of the PSAR. The means of compliance are described in Section 7.1.2.

Three PACC units are provided, one for each heat transport loop, each capable of removing the total decay heat approximately 1 hour after shutdown. Each unit is single active failure proof in that no single active failure will result in the loss of more than 50% of heat removal capability. This is provided by utilizing two tube bundles, two fans, etc., such that at least half capacity is retained following the failure. The PACC unit is a Seismic Category I design, hardened against tornado missiles and designed to withstand the pressure loads from tornados. The PACC tube bundle design is based upon standard techniques for steam-to-air heat exchangers. 22

5.6.1.2.3.2 Auxiliary Feedwater Pumps (AFWP)

59 The AFWP will be a multi-stage, centrifugal pump selected from a commercial vendor's equipment line. No special requirements should be necessary since these pumps have been proven to be reliable in commercial applications. The turbine driven pump will be sized to deliver a 1116 GPM flow rate at 4540 feet developed head, and the two motor driven pumps will be sized to deliver one-half of this flow rate each at the same head.

59 AFWP Motor Drives

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

59 AFWP Turbine Drive

This component will be obtained from an experienced vendor and will be sized to
17 produce 1960 horsepower. The turbine will be constructed with 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
17 designed to Seismic Category 1 requirements. Allowable stress limits are specified in Table 3.9-3 and pressure limits are specified 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
17 be constructed to the requirements for an ASME Section III/Class 2 vessel and it will operate at low temperature (<200°F) and low pressure (<15 psig).

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
58 operated, locked open valve and an electrically operated,
normally open isolation valve. These lines are Class 2
49 from the PWST to the electrically-operated isolation valve
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 feedwater and condensate
58 system 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 dis-
charge points, one to each steam drum feedwater supply loop.
a 6 inch carbon steel line from each motor driven pump feeds
43 17 into a 6 inch header which also has three discharge points, one
to each drum.

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Analysis Methods

In order to size critical components of SGAHRS, such as AFW pumps, PACC's and the protected water storage tank (PWST), the transient heat load on SGAHRS must be known. Plant stored heat and reactor decay heat must be removed through SGAHRS initially over a period of several hours, and then decay heat must be removed through PACC's for an indefinite period. The DAHRS (Demo Auxiliary Heat Removal Simulation) computer code (Appendix A) is used to compute the heat load on SGAHRS after shutdown, while accounting for both sources of heat.

To provide an example of normal SGAHRS response to abnormal plant operation, DAHRS was used to evaluate the heat load following a loss of off-site power. Reactor decay heating and flows (sodium, main feedwater, and recirculation water) are inputs; system temperatures and AFW flow are computed values.

Analysis Input Assumptions

Initial conditions are assumed to be at 15% above rated power (the "stretch" condition), with system temperatures adjusted to the maximum possible within estimated instrument error. The higher initial temperatures are conservative for heat loading on SGAHRS. Sodium flows are assumed to be at their maximum reasonable values above rated, in order to assure conservative heat loads in
17 SGAHRS sizing calculations. Specific values are as follows: 8

Power: 1121 Mwt

Flows: Primary--4030 lb/sec/loop
Intermediate--3910 lb/sec/loop
Recirc H_2O--617 lb/sec/loop
Main FW--355 lb/sec/loop 8

Temperature: Pri Hot Leg--1015°F
Pri Cold Leg--725°F
Int. Hot Leg--960°F
Int Cold Leg--662°F
Main FW--465°F 8

Response to the initiating events was computed with the following assumptions:

- All three heat transport loops behave identically (i.e., no additional failures).
- Sodium flows coastdown to pony motor values (384 lb/sec/loop and 426 lb/sec/loop for primary and intermediate, respectively).
- Main FW goes to zero; recirculating water flow coasts down to 20% of initial.
- Venting at 1450 psia occurs at the superheater steam outlet
- AFW at 125°F is supplied to match vented steam mass.
- PACC heat removal begins 4 minutes after shutdown and is 15 MWt/
17 loop after 1 additional minute. 8

Upon plant shutdown, steam is vented until the heat load on SGAHRS can be removed by the PACC alone. Thereafter, pressure is assumed to remain constant and PACC heat removal is assumed to be equal to the decreasing heat load to the steam generator. As the transient proceeds, the plant cool down will be controlled by adjusting the PACC airflow which will control the heat removal rate.

Reactor decay heat is taken as 125% of nominal. Stored heat in the metal throughout the plant includes an appropriate margin:

- 110% of nominal for piping and components,
- 125% of nominal for reactor vessel region.

Analysis Results

Results of DAHRS analysis of a loss-of-offsite-power event are presented in Figures 5.6-1, 5.6-2 and 5.6-3. These results will serve to explain SGAHRS operation in response to the event. Transient heat load on SGAHRS is plotted in Figure 5.6-1. Several features of this figure are noteworthy:

Heat load on SGAHRS peaks at about 0.07 hours after shutdown, when the PACC's are started. The peak at 0.15 hours is due to the positive temperature transient in the IHTS hot leg reaching the steam generators (see Figure 5.6-3).

Heat removed by the superheater is seen to be the difference between the two curves, peaking at the same time as heat load on the evaporators and then falling to zero as steam venting
17 subsides. 8

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 within 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 concentrations of smoke, toxic chemicals and unsafe radiation levels,
49 annunciate the presence of the hazard and automatically 22
transfer the HVAC system to its accident mode of operation.

6. Appropriate fire suppression equipment.

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

8. An amply stocked inventory of emergenc, 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 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 operation and during accident
59 conditions. 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 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 basis.

Another important feature of the Control Room Habitability System is the HVAC system. The HVAC system for the Control Room contains
59 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 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 filter units that purify both make-up and partially recirculated air flows during postulated accidents. Each filter unit in the HVAC system contains bag filters,
59 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
49 Steam Generator Building Auxiliary Bay at approximately elevation 858'. This along with instrumentation provided, allows selection of the cleaner air source during such periods.

A third important feature of the Main Control Room Habitability System is its low leakage enclosure. This enclosure is formed by:

1. Monolithic reinforced concrete floor, walls and roof described in Paragraph 3.8.4.1.2.

2. Low leakage seals for all electrical lines penetrating the enclosure. 22

3. Low leakage double doors and door seals.

4. Low leakage ventilation system isolation dampers. 22

A fourth important feature of the Control Room Habitability
System is the capability for operating in two different operating modes
and still maintain a positive air pressure level in the control room at all
59 times. The normal operating mode will be the mode utilized during all
normal plant operations and during most minor accidents. The accident mode
of operation will be initiated by the receipt of a containment isolation
signal through the plant protection system or by the detection of smoke,
toxic chemicals or high levels of radioactivity in the control room air
49 intake ducts. These detectors and controls will be designed and installed
in accordance with engineered safety feature standards. The fire detection
instrumentation is discussed in Section 9.13. Detailed information on
design bases and location of radiation monitors for the Control Room is
presented in Section 12.1.4 and Table 12.1-48. Detailed information on
airborne radioactivity monitors for the Control Room is presented in
Section 12.2.4 and Table 12.2-3. Fixed-type airborne radioactivity
monitors will be provided to continuously monitor the control room breathing
air for airborne radioactivities including particulate, gaseous, and iodine
activities. 22

Offices, living accommodations and emergency equipment and supplies are also important features in the Control Room Habitability System.

39 Fire protection for the main control room is provided by the use of non-combustible equipment in the room and by administrative control over the use of papers, manuals, and log sheets for day-to-day operations. Protection is also afforded by small hand fire extinguishers for local fire protection.

Face masks and self contained breathing apparatus will be provided to permit emergency operation. Smoke detectors will be installed in the control room. Upon detection of smoke in the control room, an alarm will be sounded. The detectors do not affect operation of the ventilation system.

The operator is responsible for taking appropriate action to
49 extinguish a fire. If he is unable to do so, he may transfer control at
the backup control area. Safe shutdown can be achieved and maintained
from the backup control area even with the Control Room completely
evacuated.

All equipment in the Control Room is designed to operate normally in the temperature range $50^{\circ}F$ to $104^{\circ}F$. The safety-related equipment will be designed to operate in the temperature range from $40^{\circ}F$ to $120^{\circ}F$. The operator is responsible for judging whether or not fire equipment requires transfer of control to the backup control area.

6.3.1.3 Design Evaluation

The Control Room Habitability System has several features that collectively provide the capability needed to satisfy CRBRP General
49 Design Criterion 17. These features are:

1. Shielding enclosing the Control Room Analyses presented in Subsection 12.1.2 show that this shielding reduces the control room personnel doses from external sources postulated to be associated with thermal margin beyond design basis requirements to a small fraction of that permitted.

2. Low leakage enclosure for the main Control Room. Such an enclosure provides the capability for keeping a positive pressure within the main Control Room at all times.

3. Positive pressure level maintained in the main Control Room at all times. Such a capability assures that in-leakage of contaminated air is minimized.

4. Widely separated intakes, one located at the SW corner
49 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'
are provided for pressurizing the Control Room during
accident conditions. During such conditions, air will
be supplied from the less contaminated intake. Radiation
49 monitors indicate the radiation levels in both pressurizing
air streams, thus allowing the selection of the cleaner
of the two air streams for continued operation. The
selection of intakes will allow a cleaner air mass into
the Control Room and will reduce contamination within
the Control Room significantly. 22

5. Properly sized emergency pressuring air flow rate. The air intakes are designed to be sufficient for main Control Room pressurization during all operating modes.

6. Filters and adsorbers incorporated into the HVAC system.
59 The filter units will help provide the capability for
reducing the radiation exposures below the limits of the
CRBRP General Design Criterion 17. The 30 day accummulated
doses to individual control room personnel for the whole
49 body and critical organs are: 22

6.3.1.5 Instrumentation Requirement

Several kinds of instrumentation are utilized in the Control
Room Habitability System. Redundant radiation monitors, toxic gas detectors,
59 and smoke detectors, are installed in the outside air intake ducts to
detect harmful concentrations of these airborne hazards. Thermostats
and humidistats are positioned in the main control room to control HVAC
system operations. Static pressure differential sensors are installed
59 in the filter units to measure the pressure change across each air filter
bank. Temperature sensors are utilized for duct heater element control
to keep the incoming air above specified limits. Flow sensors are installed
59 downstream from each control room air handling unit to sense air flows
and initiate startup of the standby redundant HVAC train. Details of
this instrumentation are provided in Sections 9.6, 9.13, 12.1.4, and
12.2.4.

6.3.1.6 Effects of Sodium Combustion Products or Other Toxic Gases on the Habitability System

The Control Room operators are protected from the effects of sodium combustion products or other toxic gases listed in Table 6.3-3, by the Control Room Habitability System.

6.3.1.6.1 Sodium Combustion Products

The release of sodium and subsequent sodium fires resulting in the release of Na_2O, Na_2O_2, NaOH, NaH and Na_2CO_3 to the atmosphere is described in Section 7.2.1.1 of the Environmental Report. The effect of the release of sodium combustion products to the atmosphere on the habitability of the Control Room will be analyzed in accordance with Regualtory Guide 1.78. The results of this analysis will be presented in the FSAR.

All stationary sodium and NaK piping, tanks and other components are located within Seismic Category I, tornado hardened buildings. Due to the above arrangement, sodium/NaK pipe or tank rupture caused by external missiles for these components (tornado or turbine) is extremely unlikely and not considered.

Any release of sodium or NaK combustion products to the atmosphere is preceded by a sodium/NaK leak and a subsequent fire. The sodium leak detection system, as described in Section 7.5.5 of the PSAR provides the first early warning to the Control Room of a probable sodium/NaK combustion product release to the atmosphere. If the sodium-NaK leak occurs in a cell containing an air atmosphere and the leak starts a fire, the sodium fire detection system will provide the second early warning to the Control Room of a probable sodium/NaK combustion product release to the atmosphere.

59

During the lifetime of the plant it is expected that in a limited number of occasions Na or NaK will be transported to the site for refill. Sodium is transported to the site in the solid state contained in tank cars. At the sodium unloading station the sodium is melted and pumped into storage tanks located in tornado hardened buildings. NaK is transported to the site in the liquid state, contained in drums and stored on the site in tornado hardened buildings. During transportation or sodium unloading a liquid metal (Na or NaK) release can be postulated due to external missiles or tank or drum failures. The probability of this accident is very low, since liquid metal transportation to the site will take place only a few times during the life of the plant. The detectors located inthe Control Room air intakes will be able to detect sodium or NaK combustion products if an accident occurs, resulting in a release of liquid metal combustion products to the atmosphere.

To prevent the accumulation of higher than premissible sodium/NaK combustion product concentrations in the Control Room, the HVAC System air intake will be isolated. The HVAC System will operate in a recirculating mode and the minimum outside air, required for the Control Room pressurization, will be taken through the Control Room Filter Unit. The isolation of the Control Room and the start of the HVAC system emergency operation will be automatically initiated by one of the above warning signals. The selection of the proper signal will be based on the Regulatory Guide 1.78 evaluation and will be presented in the FSAR.

6.3.1.6.2 Toxic Gases

The Control Room is classified as Type B construction in accordance with Regualtory Guide 1.78 and is constructed such that the leakage rate is about 0.06 air changes per hour. The Control Room is pressurized to maintain a minimum of 1/4 inch water positive pressure. The Control Room HVAC System air intakes will be provided with redundant, Seismic Category I detectors for those toxic materials which can enter the Control Room air intake in unacceptable concentrations. The Control Room HVAC System air intake and discharge ducts are provided with redundant quick acting isolation valves. The closing time of the isolation valves is less than 4 seconds. The closing of the isolation valves automatically sets the Control Room HVAC system into a recirculating mode. Two redundant Control Room filter units (as described in Section 9.6.1 of the PSAR) are provided for the maintenance of the Control Room positive pressure during isolation periods.

The toxic materials stored on the site or in onsite facilities are identified in Table 6.3-3. Storage vessels containing toxic materials are provided with instrumentation that will detect a leak, set off an alarm in the Control Room and automatically isolate the Control Room HVAC System air intake and discharge ducts, and set the Control Room HVAC System into a recirculation mode if the extent of the leak warrants it. The effects of accidents resulting in the release of toxic materials from the outside storage facilities on the Control Room habitability will be evaluated on the basis of the above described Control Room features, in accordance with Regulatory Guide 1.78, and will be presented in the FSAR.

Nearby facilities within a 10-mile radius of the plant are described in Section 2.2.1.1 of the PSAR. Early detection and telephone warning, as necessary, will be provided between these facilities and the CRBRP Control Room. Upon a significant toxic gas release and warning from the nearby 25

LIST OF TABLES

LIST OF FIGURES

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TABLE 7.1-1

SAFETY RELATED INSTRUMENTATION AND CONTROL SYSTEMS*

Reactor Shutdown Systems

Includes all RSS sensors, signal conditioning calculation units, compara-
tors, buffers, 2/3 logic, scram actuators, scram breakers, scram solenoid
57 valve power sources, control rods, HTS shutdown logic, coolant pump
breakers, and mechanical mounting hardware (equipment racks).

Containment Isolation System

Includes radiation monitoring sensors, signal conditioning, comparators, 2/3 logic, containment isolation valve actuators and valves.

57 Shutdown Heat Removal System Instrumentation and Control System

Includes initiating sensors, signal conditioning, calculation units, compa-
rators, logic, auxiliary feedwater pump actuators and controls including
feedwater turbine pump, PACC DHX actuators and controls, steam relief
valve actuators and valves; sensors, signal conditioning, logic and
26 57 actuators related to shutdown heat removal functions of DHRS including
control of sodium and NaK pumps and air blast heat exchangers; and sensors,
signal conditioning, logic and actuators related to removal of heat from
the EVST.

Other Safety Related Instrumentation and Control

Includes Instrumentation and Controls for portions of the following functions to assure the plant is maintained in a safe shutdown condition:

- Emergency Chilled Water System
- 33 Emergency Plant Service Water System
- Instrumentation necessary to assure plant is maintained in safe shutdown status (See Table 7.5-4)
- 59 49 Heating, Ventilating, and Air Conditioning System
- Recirculating Gas Cooling System 1

*The Clinch River Breeder Reactor Plant (CRBRP) structures, systems, and components important to safety are to be designed to remain functional to the event of a Safe Shutdown Earthquake (SSE). These plant features are also designated as safety-related features in the SAR. These include, but are not limited to, those structures, systems and components which are necessary:

a. To assure the integrity of the Reactor Coolant Boundary;
b. To shut down the reactor and maintain it in a safe shutdown condition;
c. To prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposures of 10CFR100.

NOTE: Class IE equipment loads are identified in Chapter 8. 1

TABLE 7.1-2

LIST OF REGULATORY GUIDES APPLICABLE TO SAFETY RELATED INSTRUMENTATION AND CONTROL SYSTEMS

1.6	Independence Between Redundant Power Sources and their Distribution Systems (as discussed in Sections 8.3.1.2 and 8.3.2.2)
1.12	Instrumentation for Earthquakes
1.17	Protection of Nuclear Power Plants Against Industrial Sabotage
1.22	Periodic Testing of Protection System Actuation Functions
1.28	Quality Assurance Program Requirements (Design and Construction)
1.29	Seismic Design Classification
1.30	Quality Assurance Program Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment
1.32	Use of IEEE Std 308-1971 "Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations"
1.40	Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water Cooled Nuclear Power Plants
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems
1.53	Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems
1.62	Manual Initiation of Protective Actions
1.63	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants
1.64	Quality Assurance Program Requirements for the Design of Nuclear Power Plants
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants
1.75	Physical Independence of Electric System
1.78	Control Room Habitability During Chemical Release (as discussed in Section 6.3).
1.89	Qualification of Class IE Equipment for Nuclear Power Plants (as discussed in Section 7.1.2.5).

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- <u>Sodium-Water Reaction</u>

57 The Secondary RSS uses three redundant pressure sensors located in the reaction products vent line immediately downstream from each rupture disk to detect if the rupture disks have blown. See Section 7.2.1 for details.

The configuration of the instrumentation in the protective subsystems is described in Section 7.2.1.2.

57 <u>Primary Reactor Shutdown System Logic</u>

The Primary RSS logic is implemented using integrated circuits to minimize the scram delay time. Other advantages include minimizing power consumption and space required and maximizing testability, maintainability
57 and reliability. The Primary RSS logic is arranged as shown in Figure 7.2-3.

57 In each logic train, twenty-four 2/3 coincidence logic circuits feed a 1/24 module, whose output is coupled to the final actuation logic and rod actuators by a transistorized power amplifier. When only one comparator of any or all protective functions is tripped, the logic signal output remains positive (reset). When any two comparators of a protective function trip and provide a negative logic signal to the protective logic, the output of the corresponding 2/3 module also trips to a negative logic signal. This negative logic signal in turn trips the 1/24 logic module which outputs a negative logic signal to the final actuation logic and removes power from the scram
57 breaker undervoltage coil.

Light emitting diodes and phototransistors are utilized to provide complete electrical isolation at strategic points through the Primary RSS
57 logic. There is no electrical connection between the comparator output and protective logic input. Consequently, an internal electrical fault in a single instrument channel or comparator cannot propagate to the other channels, protective functions, or logic trains of the protective system. Each logic train is electrically isolated from the other so that protective action can be initiated regardless of any internal electrical fault in a single logic train.

The equipment needed to implement the 24 protective subsystems of the Primary RSS includes the sensors, signal transmitters and amplifiers or equivalent, calculational units, comparators, logic isolators, 2/3 logic modules,
57 1/24 logic modules, logic drivers, final scram actuation circuitry and breakers,
43 buffers, permissives and bypasses. A three section equipment cabinet is used to house the equipment for each of the three instrument channels including the calculational units, comparators, power supplies and buffers. A two section equipment

cabinet is used to house the equipment for each of the three logic trains and single equipment cabinets house signal conditioning equipment for each channel. This arrangement of equipment within cabinets provides the necessary mechanical
36 separation of redundant equipment.

Secondary Reactor Shutdown System Logic

The Secondary RSS logic consists of the 16 protective subsystems arranged in a general coincidence configuration, as shown in Figure 7.2-4. In this arrangement, the outputs of instrument channel A comparators are directly coupled to a 1/16 logic circuit in logic train A, as are the outputs of instrument channel B with logic train B and the
57 outputs of instrument channel C with logic train C.

When the sensed parameter in an instrument channel exceeds its setpoint (trips), the comparator outputs a zero (trip) signal to the 1/16 logic module, which in turn outputs a zero (trip) signal to the scram latch and scram solenoid valves (see Figure 7.2-2D). The 1/16 logic module output voltage changes to zero regardless of the output of the other comparators. The output of the 1/16 logic modules are combined in a 2/3 coincidence by the
59 3 solenoid valves located within the Secondary RSS rod. Electrical
43 57 isolation of the logic output to the solenoid drivers and Heat Transport System shutdown logic (HTS pump breakers) is shown in Figure 7.2-2D. Redundant isolated outputs are provided from each Secondary RSS logic train to the Secondary RSS pump trip logic where they are combined in a 2/3 logic. Trip signals are provided to the HTS pump breakers when 2/3 of the redundant
57 Secondary RSS channels are in a tripped condition.

57 The equipment of the Secondary RSS includes sensors, signal conditioning equipment (transmitters), calculational units,
43 comparators, 1/16 logic modules, solenoid drivers, secondary final actuation logic and actuators, buffers, permissives and bypasses. The equipment is designed using hardware which is diverse from that used in the Primary RSS. Since each instrument channel is uniquely associated with a logic train, a four section equipment cabinet houses each of the instrument
43 57 channel comparators, logic trains and solenoid drivers. Single equipment cabinets are used to house signal conditioning equipment for each channel. This arrangement of equipment within separate, completely metallically enclosed cabinets provides the necessary mechanical separation between redundant equipment.

41

Channel Output Monitoring

Channel output monitoring is included to provide the operators with early indication of anomalous instrumentation performance. This equipment is not safety related. If the output of one channel differs from either of the
57 redundant channels by more than a preset amount, the channel output monitoring circuitry alarms this condition.

7.2.1.2 Design Basis Information

57 The RSS initiates and carries to completion Reactor, Heat Transport and Balance of Plant Shutdown if any of the off-normal plant conditions listed in Table 7.2-2 occur. The table also shows the frequency classification of the postulated fault, and the first Primary and Secondary
57 RSS subsystems which act to terminate the fault. As detailed in Chapter 15,

Steam-Feed Flow Mismatch

57 The Steam-Feed Flow Mismatch subsystem (Figure 7.2-7) initiates reac-
tor trip to prevent continued operation with large imbalances between the
steam and feedwater flow for each HTS loop. One of these subsystems is
included in each HTS loop. These subsystems protect the steam generators and
drums against unacceptable thermal transients. As shown in the figure, each
subsystem compares the steam flow and feedwater flow, both of which are mul-
tiplied by appropriate constants, in two individual comparators. If the
difference between the two values exceeds the setpoint in either of the com-
parators, a trip is initiated. Increasing steam flow and decreasing feed-
water flow fault events are sensed by the first comparator. The second com-
parator senses decreasing steam flow and increasing feedwater flow fault
events. Analysis of this function is based upon worst case parameter values.
This subsystem must be bypassed for plant startup. A permissive is included
which allows manual bypass of this subsystem for nuclear power less than 10%.
Two loop bypass provisions are also included for the shutdown loop. Two loop
bypasses are also under permissive control. Nuclear flux must be less than
10% of full power flux at the time of instating and the primary HTS pump
speed in the shutdown loop must be less than 15% of the speed producing
57 nominal full flow or the two loop bypass is automatically removed.

IHX Primary Outlet Temperature

The IHX Primary Outlet Temperature subsystem (Figure 7.2-7) compares the sodium temperature in the primary cold leg of each IHX to a fixed set point. A reactor trip is initiated if the sodium temperature exceeds this set point. These subsystems assure that temperature increases in an intermediate loop sodium resulting from steam side fault events or intermediate flow reductions do not increase the reactor coolant temperature. There is one IHX primary outlet temperature subsystem per HTS loop. These subsystems are never bypassed.

57 7.2.1.2.2 Secondary Reactor Shutdown System Subsystems

Modified Nuclear Rate

The Modified Nuclear Rate subystems (Figure 7.2-8) initiate trip for
rapid sustained reactivity disturbances which occur in the load range. Two sub-
systems are provided. One for positive flux rates and one for negative flux
rates. These subsystems prevent undesired thermal transients caused by
rapid changes in power with flow held constant. The reactor trip is based
on flux rate measurements from the fission counters. A permissive is included
which allows manual bypass of the negative rate subsystem for nuclear power
57 less than 10%. The positive rate subsystem is never bypassed.

Flux-Total Flow

The Flux-Total Flow Subsystem (Figure 7.2-8) provides protection against increasing and decreasing flow and power events over the 40 to 100% load range. The primary flows of the three HTS loops are summed and multiplied by an appropriate gain. A nuclear power signal obtained from the fission counters is subtracted in the comparator from the total flow value and this difference is compared to a fixed set point. If the difference exceeds the set point, then a reactor trip is initiated. Analysis of this subsystem is based on worst case parameter values, including a 500 msec. time delay for the flow detectors. This subsystem is never bypassed.

Startup Nuclear

The Startup Nuclear subsystem (Figure 7.2-8) obtains a wide range log
57 channel measurement of nuclear power from the fission counters and compares
it to a fixed-set point. If nuclear power is greater than the set point,
a reactor trip is initiated. A permissive module is provided which allows
manual bypass of this subsystem upon the verification of the operation of
57 the wide range linear channel. This subsystem provides protection against
positive reactivity disturbances occurring during startup.

Primary to Intermediate Flow Ratio

The Primary to Intermediate Flow Ratio subsystems (Figure 7.2-8) protect against an imbalance in the heat removal capability of the primary and intermediate loops. The heat removal capability of a particular loop is
determined by measurement of the sodium flow within the loop. The Secondary
57 RSS includes two of these subsystems, Primary Flow High and Primary
Flow Low. In the Primary Flow High subsystem, the output of the high
primary flow auctioneer is compared to the summation of the outputs from the
low intermediate flow auctioneer and a signal proportional to the total
primary flow. When the high primary flow auctioneer signal exceeds the low
intermediate flow auctioneer signal by an amount proportional to the total
primary flow, a reactor trip is initiated.

Similarly in the Primary Flow Low subsystem, a comparison is made
between low primary flow and high intermediate flow. When the high
intermediate flow auctioneer signal exceeds the low primary flow auctioneer
signal by an amount proportional to the total primary flow, a reactor trip
is initiated. These subsystems are manually bypassed during plant startup.
41 The permissive signal used is based on reactor power. If reactor power is
less than 10%, the subsystems can be manually bypassed.

Steam Drum Level

57

The Steam Drum Level subsystem (Figure 7.2-9) measures steam drum
water level and compares it to a fixed setpoint. A reactor trip is initiated
whenever the drum water level decreases below this fixed setpoint. There
are three of these subsystems, one per HTS loop. Analysis of these sub-
systems are based upon worst case perimeter values. For two loop operation,
a manual bypass is instated under administrative control by changing the
hardware configuration. Two loop bypasses are also under permissive control.
Nuclear flux must be less than 10% of full power flux at the time of
instating and the primary flow in the shutdown loop must be less than 15%
57 of full flow or the two loop bypass is automatically removed.

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7.4 INSTRUMENTATION AND CONTROL SYSTEMS REQUIRED FOR SAFE SHUTDOWN

The Instrumentation and Control Systems necessary for safe shutdown are those associated with monitoring of core criticality, decay heat removal (SGAHRS portion), outlet steam isolation, and control room habitability.

Monitoring of core criticality is effected by the Flux Monitoring System (Section 7.5.1). The control room habitability is covered in Chapter 6. Thus, this section treats the control and instrumentation needs for decay heat removal by the Steam Generator Auxiliary Heat Removal System (SGAHRS) and outlet steam isolation by the Outlet Steam Isolation System
26 (OSIS); control and instrumentation for Direct Heat Removal Service (DHRS) is discussed in Section 7.6.

7.4.1 Steam Generator Auxiliary Heat Removal Instrumentation and Control System

7.4.1.1 Design Description

7.4.1.1.1 Function

The SGAHRS (fluid system and mechanical components as described in Section 5.6.1, and electrical components as described below) provides the
54 heat removal path and heat sink for the nuclear steam supply system following upset, emergency, or faulted events which render the normal heat sink unavailable.

The SGAHRS Instrumentation and Control System in conjunction
54 with the PPS detects the need for, initiates, and controls the alternate heat removal path when the normal heat sink is unavailable.

54

7.4.1.1.2 Equipment Design

The mechanical system for which the SGAHRS I&C is provided is briefly described below.

When actuated, the SGAHRS draws water from a Protected Water Storage Tank and pumps it to each steam drum. Two supply lines are provided for each steam drum. One line is supplied by two half-sized, motor-driven feedwater pumps while the other is supplied by a full-sized, turbine-driven pump. Each supply line provides a flow control valve and an isolation valve at the inlet to each steam drum. The isolation valves are provided to isolate the auxiliary feedwater system from the steam generator system during power operation and to provide leak isolation during SGAHRS operation.

In addition, a Protected Air Cooled Condenser (PACC) supplied with each steam drum is placed into operation. This system rejects heat
54 to the atmosphere via convection. Saturated steam is supplied to the

condenser from the steam drum and saturated water is returned. This steam and water loop is driven by natural circulation. Each PACC unit consists of two tube bundles, two sets of louvers and two fans. The fans are used to force air across the condenser tubes. Control of the air flow is accomplished by a combination of fan blade pitch position control and air inlet louver position control.

The arrangement of SGAHRS equipment is shown in Figure 5.1-5
(SGAHRS P&ID). Instrumentation and controls are provided for the components
54 described below:

- Auxiliary Feedwater Pump Control - Upon receipt of the SGAHRS initiation signal, (see Section 7.4.1.1.3), the two motor driven pumps are started, resulting in both pumps coming on line and operating at constant speed. In addition, the isolation valves in the steam supply lines from the steam drums to the turbine driven pump are opened. At the turbine inlet a pressure regulating valve reduces the steam supply pressure to the 1000 psig required by the turbine drive. The turbine drive mechanism is equipped with a governor to provide speed regulation. Each auxiliary
58 feedwater pump can also be actuated manually at the operator's
discretion.

  Each pump control includes a "Normal Long Term Cooldown (LTC)" mode selector. In "normal" mode, the pumps start on SGAHRS initiation. In the "LTC" mode, the operator may shutdown any or all AFW pumps provided the steam drum water level is above the trip point setting. When in the "LTC" mode, the pumps come on line automatically when the steam drum water level drops to
59 a low level trip point.

- Auxiliary Feedwater Flow Control - The Auxiliary Feedwater Isolation Valves are opened upon receipt of the SGAHRS initiation signal. During SGAHRS operation, these valves close automatically upon indication of a sodium/water reaction, a high steam drum level, a steam drum pressure less than 200 psig, or AFW flow greater than 150% of full flow for 5 sec. This automatic closure occurs only in the affected loop. If the valves are closed by a high drum level signal they will reopen automatically when the drum level falls to the low drum level trip point. The flow to the steam drum is controlled with a control valve that is positioned by a single controller. Manual control of the Auxiliary Feedwater Flow Control valves is provided at the main control panel and at the local SGAHRS panel.

- Protected Air Cooled Condenser Control - The Protected Air Cooled Condenser louvers are opened and the fans started upon receipt of either a SCRAM or the SGAHRS initiation signal.
54 Air flow through the condenser is controlled by a combination

of fan blade pitch and inlet louver position. The fan blades and inlet louvers are positioned by automatic controllers. Manual control of the inlet louver position and fan blade pitch is
54 provided. Manual controls are also provided for the blower motors. The outlet louver is interlocked with the inlet louver. It opens automatically when the inlet louver actuator is energized. If a high concentration of sodium aerosol in each PACC cell is
59 detected, redundant trip logic generates trip signals to shutdown the affected PACC system for approximately 1½ hours.

7.4.1.1.3 Initiating Circuits

The Reactor Shutdown System (see Section 7.2) provides redundant
54 48 primary and secondary initiation signals to SGAHRS to sequentially start
the three Auxiliary Feedwater Pumps and the three Protected Air Cooled
Condensers when either a low steam drum level or high steam-to-feedwater
47 flow ratio occurs in any one of the three Steam Generator System (SGS)
loop subsystems. In each subsystem, the three trip signals for low
steam drum level and the three trip signals for high steam-to-feedwater
flow ratio are each isolated and input to redundant two out of three
logic networks. The outputs from the redundant logic networks are each
isolated within the SGAHRS divisional control system and combined in a
48 one-of-four logic to initiate SGAHRS. If two of three trip signals
occur in any subsystem, the SGAHRS is initiated. The sequence of decay
heat removal events is shown in Table 7.4-1. The scheme used for initiating
the SGAHRS is shown in Figure 7.4-1.

Since the automatic activation and control of auxiliary feedwater flow is necessary to assure decay heat removal, provisions are included in the design to assure that the automatic initiation takes precedence. A startup signal to the feedwater pumps overrides a manual control signal. Similarly, a signal to open the isolation valves overrides a manual closure signal.

7.4.1.1.4 Bypasses and Interlocks

Bypasses are required on the steam to feedwater flow mis-
match and steam drum level subsystems to allow system reset and
reactor startup without initiating SGAHRS. These bypasses will be
48 implemented as described in the Reactor Shutdown System (Section 7.2).

The following are interlocks provided in the SGAHRS components control circuits:

(a) Each auxiliary feedwater pump (preferred) inlet valve may be closed only after the associated alternate inlet valve has been fully opened. The preferred inlet valve will open automatically anytime the alternate inlet valve starts to close.

(b) A switch is provided on the back panel to permit the operator to bypass the sodium aerosol protection circuit of the PACC. The bypassed position is annunciated on the Main Control Room panel.

(c) The PACC outlet louver opens automatically whenever the inlet louver
59 is not fully closed. When the outlet louver is fully open,
the PACC blower may be started either automatically or manually.

7.4.1.1.5 Redundancy/Diversity

The SGAHRS (fluid system and mechanical components) is designed with suitable redundancy and diversity so that it can perform its safety functions following a single failure of an active component for anticipated, unlikely and extremely unlikely plant conditions. The design of SGAHRS relating to these objectives is discussed in Section 5.6.1.

Redundancy and diversity are also provided within the initiating circuitry of the SGAHRS control system. As shown in
54 Figure 7.4-1, the system is actuated on two-out-of-three trip signals from either low steam drum level, or high steam-to-feedwater flow ratio.

7.4.1.1.6 Actuated Devices

All automatic valves and motors in the SGAHRS are provided with remote manual control capability, so that the entire system can be operated from the control room or the remote shutdown panels.
54

All isolation valves within the SGAHRS utilize an electro-hydraulic actuator. All isolation valves are designed to fail
54 to the position of greater safety upon loss of electrical power.

All required components of the SGAHRS instrumentation and control system operate on a vital electrical bus.

7.4.1.1.7 Testability

Instrumentation and controls for the SGAHRS are designed and arranged to allow for complete testability during reactor power operation. Bypassing of the actuated components (i.e., isolation valves and motors) is not required during testing as operation of these components during power operation poses no penalty on plant operation.

7.4.1.1.8 Separation

The SGAHRS instrumentation and control system, as part of
54 the Decay Heat Removal System, is designed to maintain required isolation and separation between redundant channels (see Section 7.1.2.2).

7.4.1.1.9 Operator Information

Indicators and alarms are provided to keep the plant operator informed of the status of the SGAHRS. The following items are located
58 on the Main Control Panel for operator information.

Analog Indication

- Protected Water Storage Tank Level
- Protected Water Storage Tank Temperature
- Auxiliary Feedwater Flow (each loop)
- Auxiliary Feedwater Pump Discharge Pressure
- Drive Turbine Steam Inlet Pressure
- 54 Drive Turbine Speed
- 58 PACC Outlet Air Temperature
- PACC Outlet Water Flow and Temperature
- PACC Inlet Louver Position
- 54 PACC Fan Blade Pitch Position
- Steam Drum Pressure and Water Level

Indicating Lights

- 54 PACC Outlet Louver Position
- Position of all Isolation and Control Valves
- Operating Status of all Motors
- 48 SGAHRS Initiation Logic Reset

Annunciators

- Low Protected Water Storage Tank Level
- Low Low Protected Water Storage Tank Level
- High PWST Temperature
- Simultaneous Opening or Closure of the AFW Pump Inlet Valve and the AFW Pump Alternate Inlet Valve
- Flow Limiting of AFW
- High AFW Supply Temperature
- High/Low Drive Turbine Speed
- High Drive Turbine Steam Inlet Pressure
- Drive Turbine Group Alarm (Bearing and Lube Oil System)
- 58 AFW Pump Group Alarm (Bearing Temperature)
- High Motor Bearing Temperatures
- 54 Transfer Switches on Local
- 48 SGAHRS Initiation Logic Trip
- Na Aerosol Concentration High
- Na Aerosol Control Bypassed
- 59 PACC Start-up on Reactor Trip "On Test"

Additional indicators and alarms are provided at the local instrumentation and control panels. Most information is also available to the operator via the Plant Data Handling and Display System (PDH&DS).

54

7.4.1.2 Design Analysis

To provide a high degree of assurance that the SGAHRS will operate when necessary, and in time to provide adequate decay heat removal, the power for the system is taken from energy sources of high reliability which are readily available. As a safety related system, the instrumentation and controls critical to SGAHRS operation are subject to the safety criteria identified in Section 7.1.2.

Redundant monitoring and control equipment will be provided to ensure that a single failure will not impair the capability of the SGAHRS Instrumentation and Control System to perform its intended safety function. The system will be designed for fail safe operation and control equipment where practical and will, in the event of a failure, assume a failed position consistent with its intended safety function.

Because there are three redundant decay heat removal loops, the instrumentation and controls associated with each individual loop (e.g., auxiliary feedwater flow and air cooled condenser control systems) do not independently meet single failure criteria. However, when taken collectively as a system, they provide the single failure capability required.

7.4.2 Outlet Steam Isolation Instrumentation and Control System

7.4.2.1 Design Description

7.4.2.1.1 Function

The Outlet Steam Isolation Subsystem (OSIS) provides isolation of steam system pipe breaks. Steam system isolation is a necessary function for safe shutdown in those pipe break conditions

TABLE 7.4-1

SEQUENCE OF DECAY HEAT REMOVAL EVENTS FOR POSTULATED LOSS OF FEEDWATER

Event	Time From Previous Event (1)	Auto/Manual	Controlled Parameter	Controlling Signal
Scram Reactor & Trip Turbine. 59 Initiate Opening of AFW Isolation 58 Valves, SGAHRS Vent Control Valves, AFW Turbine Drive Steam Supply Isolation Valves, and Drive Turbine Pressure Control Valve. Start AFW Pump Motors. Initiate Start of Air Cooled Condensers.	0 Sec.	A	Initiate SGAHRS & OSIS	Plant Shut Down System High Steam to Feed Water Flow Ratio or Low Level
Steam Generator System Relief Valves Open. (2)	1 Sec.	A	SG System Pressure	High Steam System Pressure
AFW Drive Turbine Steam Supply Isolation Valve Full Open	2 Sec.	A	------------------	----------------------
Electric Motor Driven AFW Pumps Attain Rated Head	7 Sec.	A	------------------	----------------------
AFW Control Valves Regulate FW Flow	----	A	AFW Flow	Steam Drum Level
Vent Control Valves Regulate Steam Drum Pressure	----	A	Steam Venting	Steam Drum Pressure
Turbine-Driven AFW Pump Attains Rated Head	17 Sec.	A	Pump Speed	Pump Speed
Stop 2 of 3 AFW Pumps. (3)	Optional	M	------------------	----------------------
Vent Control Valves Close and PACCs assume complete 54 heat load	~1 Hr.	A	Steam Venting	SG System Pressure

TABLE 7.4-1 (Continued)

Event	Time From Previous Event(1)	Auto/Manual	Controlled Parameter	Controlling Signal
PACC Fans and Louvers Stabilize PACC heat rejection rate	---------	A	Air Cooled Condenser Heat Rejection	Steam Drum Pressure
Stop AFW Pump	Optional	M	-------------------	-------------------
Decrease Steam Drum Pressure Setpoint for PACC	Optional	M	-------------------	-------------------
Plant Reaches Isothermal Conditions Corresponding to PACC Setpoint	--------	A	-------------------	-------------------

(1) For loss of off-site power event startup of AFW pumps initiated 20 sec. after loss of power.

(2) The SG System Relief Valves are open only a short time (<15 sec.) due to their high setpoint. As the system pressure falls, these valves will close first and the system pressure will be controlled by the SGAHRS Vent Control Valves only.

54 (3) Redundant pumps stopped when one motor-driven pump can assume the entire AFW flow requirement.

to reactor low power (a few Kwt) operation. The sensitivity of the
54 BF_3 counters will be maintained at a minimum of 40 cps/thermal equivalent
57 nv at a gamma background count rate of approximately 1 cps. Shielding will be
provided to limit the gamma dose rate at the detectors due to prompt gammas
from the core and from decay of activation isotopes in the sodium coolant
and structural materials to less than 100R/hr. This will apply over the
operating range of the channel at all times when operation is required.
The BF_3 counters and associated cables will be designed to operate under
normal environmental conditions of 170°F maximum and atmospheric pressure
and under emergency conditions of 260°F maximum and 12 psig during con-
57 tainment testing. The design life goal for the counters is 3 full power
years based on a total fluence of 10^{18} nvt. In order to achieve this life-
time without retracting the counters, the operating voltage will be removed
and the anode of the counter shorted to ground through an appropriate
resistance when the flux level is above the operating range of the channel.

The output pulses from the detector assembly will be amplified by a dual
section pre-amplifier mounted in the head access area and routed to the signal con-
ditioning equipment in the control room racks. This equipment will consist
of dual amplifiers with pulse shaping networks to minimize gamma pulse pile-
up followed by discriminators to reject amplifier noise and gamma pulses.
57 The summed output of the discriminators will drive a count rate circuit followed
by a scaling amplifier which will produce an analog output signal pro-
portional to the logarithm of the input pulse rate. The analog signal
will be displayed on a log scale countrate level meter calibrated from
1 to 10^6 cps. The accuracy and repeatability of the channel will be $\pm$ 3%
and $\pm$ 1%, respectively, of linear equivalent full scale under the 17
worst case environmental conditions of temperature, pressure and input
power fluctuations to be encountered while channel operation is required.
The response time of the count rate circuits will vary with count rate,
being on the order of 30 seconds at the lowest expected average count
rate of approximately 4 cps which will occur during refueling at beginning of core
59 life conditions. The 1σ statistical counting error will be approximately $\pm 7\%$
54 at the 4 cps rate. The response time will decrease with increasing count
rate to approximately 0.1 seconds at a count rate of 3×10^5 cps produced
57 at a few kilowatts. The analog output signal of the scaling
amplifier also drives a differentiating circuit to produce a rate of
change of level signal which is displayed on a linear scale meter from
-1 to 0 to +3 decades per minute. Power supplies mounted in the equipment
drawers will provide detector excitation high voltage and low voltage 17
instrument power. Individual scaler timers will be provided in each
59 channel which will be driven from a buffered output of the discriminator summer
which precedes the countrate circuit. The scaler timers will be programmable
to count for a short preset time, stop, transfer the accumulated count to tempo-
rary storage for PDH&DS readout, immediately restart the count and repeat the
57 sequence to provide an accurate record of the count rate trace to implement the
inverse kinetics rod drop technique used to establish control rod worth. The
57 scaler-timers will also provide for counting of signal pulses for longer time
periods to accurately determine the system calibration constant which relates 17
subcritical reactivity to count rate.

A visual/audio linear count rate circuit will be provided for
operations at shutdown. This circuit will be provided with a switch

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through which it can be connected to the summed discriminator output of any of
57 the three Source Range Channels, one at a time. The visual portion provides expanded scale indication of the flux level. The audio portion provides tone bursts whose rate changes (increase and decrease) with increasing and decreasing flux level.

Each channel will be provided with built-in counting level and rate of change calibration circuits for channel alignment and preoperational testing to assure that the instrumentation circuitry is functioning properly.

The Source Range Channels are used to monitor the shutdown and startup flux only, no signals are provided to the PPS. Bistable comparators in each channel will activate individual annunciators in the control room to provide alarms if the specified minimum shutdown reactivity is exceeded during refueling or if detector excitation or instrument power is lost in any channel. To improve the operational effectiveness of the source range shutdown reactivity monitoring function, the reactivity alarm is inhibited during core assembly movement which could cause an erroneous alarm. The inhibit circuit is continuously monitored for proper operation and any malfunction of the inhibit circuit will activate a separate control room annunciator. An alarm will also be provided in the control room via the PDH&DS if any one channel deviates from the other two by a preset amount. The shutdown margin alarm bistables and the PDH&DS channel deviation alarm will operate off of the buffered outputs of the log count rate scaling amplifiers. The bistables which provide alarms upon loss of detector excitation voltage as loss of instrument power will operate off of H.V. sense signals developed by voltage divider networks corrected to the preamplifier H.V. outputs to the detector assemblies and off
59 of instrument power supply output voltages, respectively.

The operation of the source range channels will be under manual control of the reactor operators. These channels will be in operation continuously during reactor refueling and other shutdown conditions. During reactor startup, the source range channels will be used to monitor the core flux level until a predetermined overlap between the source range channels and wide range log count rate channels is obtained. The high voltage will then be removed from the source range detectors by the operator. He will do this by actuating a momentary contact pushbutton switch on the main control panel. When actuated, this switch will remotely interrupt the input power to the detector high voltage supplies, and ground the detector anodes through an appropriate resistor. The relay control circuit established by the momentary contact of this switch will also illuminate a green indicator light located adjacent to the switch. Upon reactor shutdown, the operator will interrupt the relay control circuit by actuating a second momentary contact pushbutton switch when the wide range log counting channels indications fall to a predetermined level within the source range channels operating range. This action will remove the ground from the detector anodes, restore the detector excitation voltage, extinguish the green indicator light and illuminate a red indicator light. Inadvertent removal of, or failure to restore the operating voltage when needed will be prevented by procedural control, utilizing the separate on/off switches and with monitoring through the color of the illuminated indi-
57 cator light.

be displayed on the related meters. Five power level meters, five selector switches and three rate of change of power meters will be provided for the operator.

- The source range level will be indicated in logarithmic counts per second and rate of change of level in decades per minute. Linear count rate will be provided at shutdown at the refueling console and at the FMS system panels in the control room. Audible count 57 rate indication will be provided in the control room and in containment at the refueling console.

- The wide ranges will be indicated as follows:

  Counting channels - Logarithmic percent power level and decades per minute rate of change.

  MSV channels - Logarithmic percent power level and decades per minute rate of change.

  DC channels - Linear percent power level.

- The power range will be indicated in linear percent power level.

Preliminary Failure Mode and Effect Analysis results applicable to the FMS have been determined in an analysis of possible failure modes and their effects on the Reactor Shutdown System performance and are presented in Tables C.S. 1-4 and C.S. 1-5.
41

7.5.2 Heat Transport Instrumentation System

7.5.2.1 Description

The Heat Transport Instrumentation System provides sensors, associated signal conditioning equipment and controls other than Plant Control, for the Primary Heat Transport, the Intermediate Heat Transport and the Steam Generator. The signals from the sensors are conditioned and then supplied 59 to the Reactor Shutdown System logic, the Plant Control System, the Plant Data Handling and Display System, and the Plant Annunciator System as appropriate. The location of the Heat Transport Instrumentation is provided in Figures 5.1-2 and 5.1-4 (P&ID's).

7.5.2.1.1 Primary and Intermediate Sodium Loops

Reactor Inlet Pressure

The measurement is made by pressure elements installed in the cold leg of the primary loop piping just before it enters the reactor vessel. 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.

Each pressure transducer consists of a diaphragm which moves in response to pressure changes in the NaK filled capillary, a strain gage which converts diaphragm motion to resistance change, and a bridge and amplifier to convert strain gage resistance change to a standard signal.

Since pressure element replacement requires plant shutdown, two pressure elements per loop are provided.

The signals from the six, two per loop, pressure measurements are transmitted to the control room in three separate isolated PPS channels for use in the Reactor Shutdown System logic. The Reactor Shutdown System sup-
59 plies buffered signals to the PDH & DS.

Primary and Intermediate Loop Flow

The flow measurements are made by a permanent magnet flowmeter located in the cold leg of each primary and intermediate loop (except
49 for intermediate loop 2, which has the PM flowmeter in the hot leg). The magnet assembly is in the shape of an inverted "U" which is suspended around the pipe. The magnet assembly is mounted rigidly to the building structure and is physically separated from the pipe.

Type K thermocouples are installed in the magnet structures to moni-
59 tor the magnet temperatures. The signals from these thermocouples are routed to a local panel. Provisions will be made to permit periodic monitoring of the magnetic flux of the flowmeters without disassembly or entrance into HTS cell. This is also accomplished at the local panel.

Four independent pairs of 3/8" (approx.) electrodes are attached to the pipe. The electrodes are of the same composition as the pipe so that thermal potentials are not developed. Three pairs of electrodes are connected to the conditioning equipment. The fourth pair is available for a portable measuring instrument or as an installed spare.

Flexible mica, polyimide and fiberglass insulated cables in separate conduits to meet PPS separation requirements are used to bring the four signals from each flowmeter assembly out of the Heat Transport System cell. The signals are then routed to signal conditioning equipment.

From the signal conditioning equipment, the signals are sent to the control room for the Reactor Shutdown System logic which in turn supplies
59 buffered signals to the PCS and the PDH & DS.

IHX Primary Outlet Temperature

The IHX primary outlet temperature measurement is made by three Chromel/Alumel thermocouples per loop installed in thermowells in the elevated section of the HTS cold leg piping nearest the IHX primary outlet. The thermocouples are 1/8" insulated junction swaged to 1/16" at the tip to

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
59 buffered signals to the PCS and PDH & 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
43 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
59 for transmission to the PDH & 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 PDH & DS. Since this pressure element is located in an inerted cell and replacement would require entry into the cell and draining of the
59 loop, two pressure elements per loop are provided.

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 PDH & DS.

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.
59 The conditioned signal is supplied to the PDH & DS.

Intermediate Expansion Tank Level

Two separate level measurement channels are provided; both channels are used for indication in the control room and PDH & 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 PDH and DS uses measurements for intermediate loop sodium
59 inventory (see also Section 7.5.5). The level probes are designed to be replaceable.

Evaporator Sodium Outlet Temperature

59 Three thermocouple (as described above in the paragraph on IHX outlet temperature) 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
59 signals to the PDH & 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
59 signal is also provided to the PDH & DS where it can be used in calculation of sodium inventory.

Primary and Intermediate Pump Speed

Five primary and intermediate pump speed signals are provided. Three for the Reactor Shutdown System and two for the speed control system. These signals are obtained from the shaft of the vertical drive motor.

Three redundant signals are produced for each pump conditioned and transmitted to the control room as the three channels required for the Reactor Shutdown System. These signals are buffered and supplied to the PDH & DS.

Two redundant speed signals are provided to the pump speed control
59 equipment where it is used as a feedback signal.

29 59

Pony Motor Running

A signal is provided to the control room indicating that the pony motor is running.

Diagnostic Instrumentation

In addition to the instrumentation described above, diagnostic instrumentation is provided.

7.5.2.1.3 Steam Generator

Sodium Flow

Venturi flowmeters are provided, one loop only, to accurately measure the sodium flow rate through each of the superheater outlet ports. The accurate flow data is used for determination of the performance characteristics typical of the superheaters and evaporators.

Sodium Temperature

The evaporator and superheater outlet temperature is monitored, on all three loops, by Resistance Temperature Detectors (RTD). The superheater inlet is monitored, on one loop only, also by an RTD for purposes of steam generator performance evaluation. These temperature sensors provide signals
59 for the PDH & DS. The evaporator bulk outlet temperature is measured with three thermocouples and are part of the Reactor Shutdown System.

Sodium Pressure

For the purpose of steam generator performance evaluation, pressure is measured, in one loop only, at the superheater inlet, superheater outlet (both legs) and evaporator outlet. The type of pressure sensor used is the same as the one for Intermediate pump inlet pressure. These pressure mea-
59 surements provide pressure signals to the PDH & DS.

Steam and Water Flow

- Feedwater Mass Flow - sensed by three differential pressure elements across one venturi in the inlet line to each steam drum.

- The temperature corrected feedwater flow signals are supplied
to the Reactor Shutdown System logic. The Reactor Shutdown
59| System provides buffered signals to PCS and PDH & DS.

- Steam Mass Flow - sensed by three differential pressure elements
across one venturi in the outlet of each superheater. The temper-
ature and pressure corrected mass signals are supplied to the
Reactor Shutdown System logic. The Reactor Shutdown System pro-
59|29 vides buffered signals to PCS and PDH & DS.

- Steam Drum Blowdown Flow - sensed by flow orifice (differential
59| pressure) in the blowdown line for each steam drum. The signal is
provided to the PDH & DS.

- Evaporator Inlet Flow - sensed by a differential pressure element across a venturi in the inlet line to one of the evaporators in one loop only. This is to aid in the performance evaluation of a typical evaporator module.

Steam and Water Temperature

- Feedwater Temperature - sensed by three resistance temperature
detectors in the steam drum inlet line. The signal provides
temperature compensation for the feedwater flow signal. Buffered
59| signals are supplied to the PDH & DS.

- Recirculating Water Temperature - sensed by a thermocouple detector
in the recirculation pump discharge header. The signal is pro-
59| vided to the PDH & DS.

- Saturated Steam Temperature - sensed by a thermocouple detector in
the outlet header from the steam drum. The signal is provided
59| to the PDH & DS.

- Superheat Steam Temperature - sensed by three resistance tempera-
ture detectors in the superheater outlet line. The signal pro-
vides temperature compensation for the steam flow. Buffered sig-
59| nals are supplied for PCS and PDH & DS.

- Evaporator and Superheater Inlet and Outlet Temperature - sensed by RTDs located at the inlet and outlet nozzles for one evaporator and superheater in one loop only. Used for performance evaluation for a typical generator module.

- Steam Drum Blowdown Temperature - sensed by a thermocouple
located on the blowdown line. The signal provides temperature
compensation for the steam drum blowdown flow and is also supplied
59| to the PDH & DS.

Steam and Water Pressure

- Feedwater Pressure - sensed by a pressure element located on the inlet line to each steam drum. The signal is used for PCS and is
59 supplied to the PDH & DS.

- Steam Drum Pressure - sensed by three pressure elements located on an appendage from the steam drum. The signal provides pressure
59 compensation to steam drum level. Buffered signals are used for PCS and PDH & DS.

- Recirculation Pump Discharge Pressure - sensed by a pressure element located on an appendage from the recirculation pump inlet header.
29 59 This measurement is required for recirculation pump protection and performance analysis by the PDH & DS.

- Superheat Steam Pressure - sensed by three pressure elements located on the loop output steam line. The signal is supplied to
59 the Reactor Shutdown System logic. Buffered signals are supplied to PCS and PDH & DS.

- Evaporator and Superheater Inlet Pressure - sensed by pressure elements located off the inlet nozzles for one evaporator and superheater in one loop only. The signal is supplied to the
59 PDH & DS for performance evaluation of a typical steam generator module.

Evaporator and Superheater Outlet Pressure

Sensed by pressure elements located off the outlet nozzle of each evaporator and superheater. The signal is used for SWRPRS Control and is
59 supplied to the PDH & DS for the performance evaluation of a typical steam generator module.

Steam Drum Level

Sensed by three differential pressure elements measuring the differential pressure between a reference column and the water head in the steam drum. This measurement is density compensated. The signal is supplied to the Reactor Shutdown System logic. Buffered signals are supplied to PCS and
59 PDH & DS.

Impurity Monitoring

Hydrogen and oxygen concentration monitors are utilized on the sodium side of the steam generators for detection of water-to-sodium leaks. The equipment is discussed in detail in Section 7.5.5.

Indication and Alarm

Indication and alarms are provided to keep the operator informed of the status of system and equipment and to quickly determine location of malfunctioning equipment. All of the measurements discussed in this section are processed by the Plant Data Handling and Display System. In addition, the following parameters are continuously monitored in the main control room:

- Feedwater pressure, temperature, and flow.
- Steam drum pressure, temperature, and level.
- Superheat steam pressure, temperature, and flow.
- Turbine inlet pressure, themperature, and flow.
- 29 Recirculation Pump Discharge Pressure.

7.5.2.2 Analysis

Instruments part of the Reactor Shutdown System will comply with the PPS Design Requirements (See Section 7.1.2 and 7.2.1). The design analysis for the Reactor Shutdown System applies (Section 7.2.2), and a Failure Mode and Effects Analysis is performed as shown in Table C.S. 1-2 for PM flow meters; Table C.S. 1-6 for Sodium Pressure Sensors; Table C.S. 1-7 for Tachometers; Table C.S. 1-8 and C.S. 1-9 for the Pressure Sensors and Associated Temperature Sensors for Flow Measurements; and Table C.S. 1-10 for Steam Drum Level Instruments.

41 Non-PPS instruments will be duplicated or triplicated either to obtain more representative values of the parameter measured, or to permit continued parameter measurements upon failure of one instrument, when sensor replacement requires plant shutdown.

Built-in calibration means have been provided to ensure reliable and accurate monitoring throughout the plant life, e.g., the permanent magnet flowmeters can be calorimetrically recalibrated by the hot and cold leg temperature measurements after initial plant startup calibration.

Seismic considerations have been also included in the design of the Heat Transport Instrumentation, e.g., the clearance between the pipe and the magnet structure of the permanent magnet flowmeters is such that contact is prevented under all postulated conditions of vibrations, including a Safe Shutdown Earthquake (see Sections 3.7 and 7.2.1 for Seismic Design).

The design includes the requirements of sodium coolant contingencies by providing a double barrier against sodium spillage for the temperature and pressure sensors installed in the Primary and Intermediate Heat Transport coolant boundaries.

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

Amend. 50
June 1979

13

Operation Requirements

o 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
High	$> 5 \times 10^{-3}$	Confirm leak Initiate rapid 47 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

The Sodium-Water Reaction Pressure Relief System (SWRPRS) Instrumentation and Control System detects the inception of a large or intermediate water to sodium leak in any of the steam generator modules (see Section 5.5.2.6).

For a large leak, three 1E pressure sensors (nine per loop) are provided immediately downstream from each pair of rupture disks in the superheater and evaporator's (two) reaction products vent line. The signals are transmitted to the PPS Secondary Shutdown System which initiates a reactor trip and PHTS and IHTS sodium pump trip. Buffered signals are transmitted to the SWRPRS trip logic which isolates the affected loop. A group alarm is transmitted to the Plant Annunciation System (PAS).

For intermediate leaks, three pressure sensors are provided in the IHTS sodium expansion tank equalization line to the sodium dump tank, downstream of the rupture disks. These signals are transmitted directly to the SWRPRS trip logic via a two-out-of-three coincidence logic which isolates the affected loop. Reactor trip and trip of the PHTS and IHTS sodium pumps is initiated via the PPS Primary Shutdown System as a result of a high steam-to-feedwater flow in the affected loop.

SWRPRS TRIP LOGIC

There are these separate SWRPRS trip logics, one each loop. Thus,
59 only the affected loop will be isolated leaving the other two loops for shut-

down heat removal. The SWRPRS trip logic (Figure 7.5-6) and the remainder of this discussion addresses one loop only.

In parallel with sending signals to the PPS for reactor and sodium pump trip for large leaks, the SWRPRS instrumentation send buffered signals to the SWRPRS trip logic. The trip circuit develops a two-out-of-three coincidence logic from each steam generator module (one superheater and two evaporators). Each module is combined in a one-out-of-three coincidence logic which in turn is then combined in a one-out-of-two coincidence logic with the intermediate leak logic.

Upon receiving a signal from the large leak detection circuit, the intermediate leak detection circuit or a manual trip from the control room the following simultaneous actions occur in the faulted loop.

a. The HITS sodium pump pony motor of the affected loop is tripped by deenergezing the contactor coil (large leak detection circuit only).

b. The SGS recirculating pump motor is tripped off the line by energizing the switchgear's tripping circuit.

c. The water/steam side of each evaporator and superheater is individually isolated by closure of their respective isolation valves. The main feedwater, auxiliary feedwater, and steam drum inlet and drain isolation valves are closed.

d. Water is removed from the evaporators by opening the valves between the evaporator inlet and the water dump tank. Power relief valves on the outlet line of each evaporator and the superheater are opened to provide a steam vent to the atmosphere.

e. Water dump and steam vent action is terminated by closure of all steam power relef and water dump valves when the units have been de-pressurized to 250 psig.

f. The water-steam side is then inerted by opening of the nitrogen purge valves which provide nitrogen to both units in the affected loop. A regulator on the nitrogen supply maintains the pressure at 200 psig. In the event of continued pressure buildup, the steam vent power relief valves will open at 300 psig and provide for another depressurization to 250 psig.

g. SWRPRS piping is purged by nitrogen following bursting of SWRPRS main rupture disks.

All isolation, dump, power relief, and purge valves are provided with controls and status indication in the Main Control Room to provide manual control at the plant operator's discretion. Alarms are provided in the PAS for the SGS isolation, dump and pressure relief valves to warm the operator
59 of inadvertent off-normal operation.

40

59

7.5.6.1.3 Bypasses and Interlocks 40

The control logic for the actuation of the Sodium-Water Reaction Pressure Relief System will be designed to insure reliability and freedom from spurious operation. A discussion of the bypasses and interlock functions will be provided as detail system design and analysis progresses.

7.5.6.1.4 Sodium Dump 40

No automatic action is associated with the removal of sodium from the affected loop. However, sodium dump valves are provided for draining 40 of sodium to the sodium dump tank, and can be initiated by operator action.

Drain valves are located in five piping runs between the IHTS sodium 40 loop and the sodium dump tank. Each piping run contains a pair of drain valves arranged in series. Controls and indications for all these valves are located on the Main Control Board.

7.5.6.1.5 Monitoring Instrumentation 40

In addition to the instrumentation required for the initiating circuitry, the following parameters are measured to aid the plant operator in assessing the performance of the Sodium-Water Pressure Relief System:

- Pressure in the gas space between each pair of rupture disks is monitored to detect leakage, or failure of the sodium side rupture disk. Spark plug leak detectors are also provided in the gas space to detect rupture disk failure.
- Thermocouple elements are provided for monitoring surface temperatures of the reaction products separator tank, centrifugal separator, centrifugal separator drain tank, and the hydrogen igniter.

29

REACTION PRODUCTS VENT FLOW DETECTION

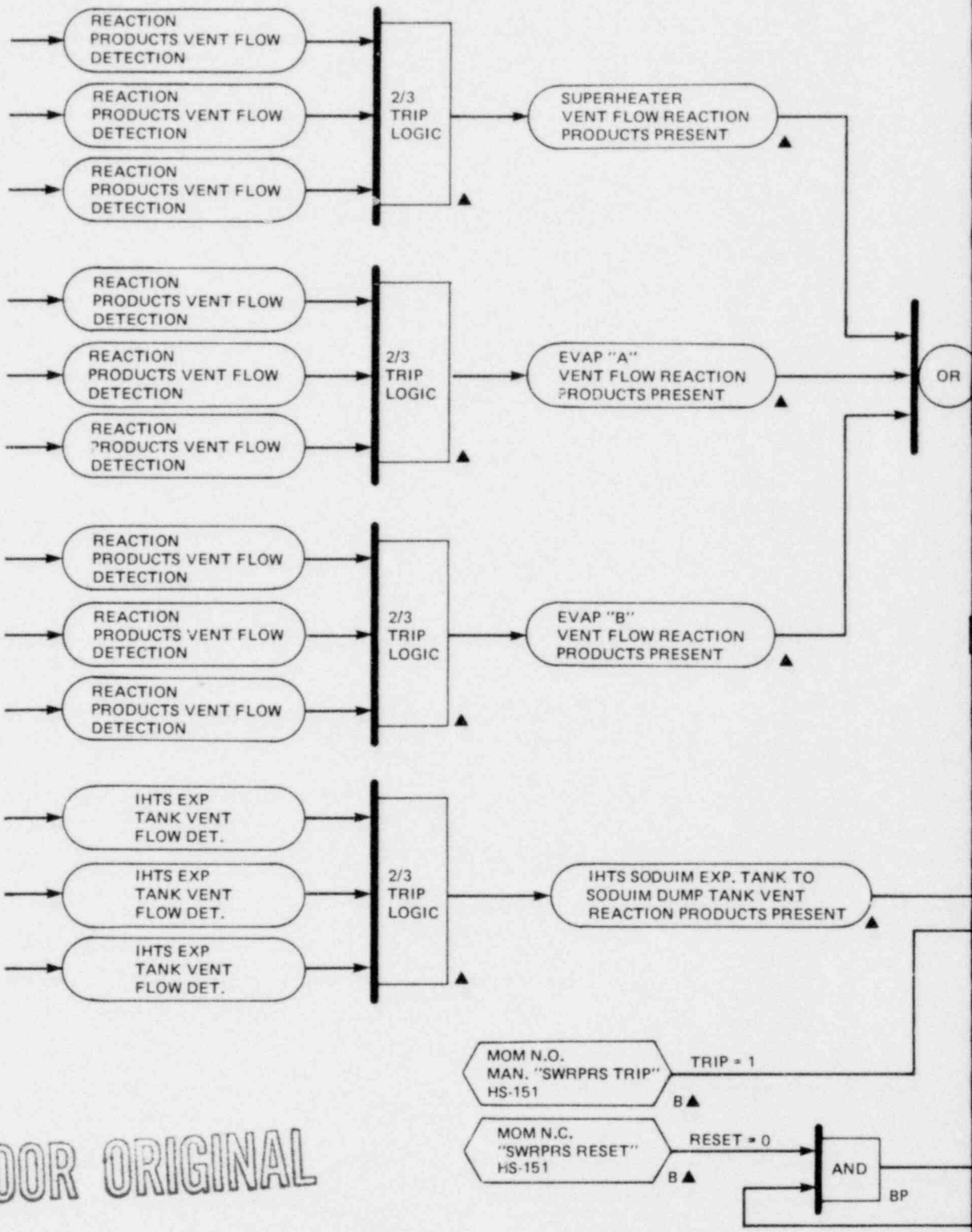
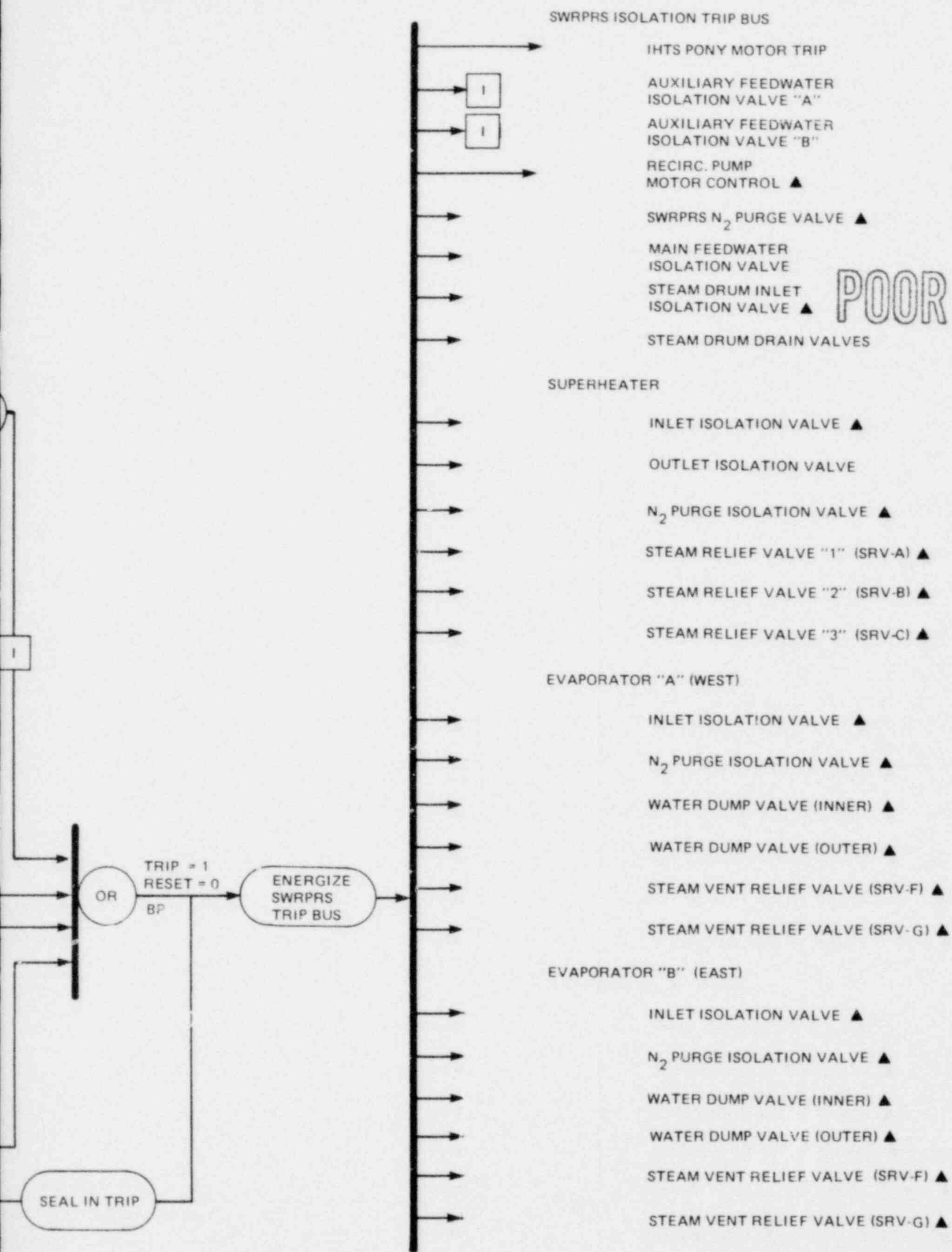


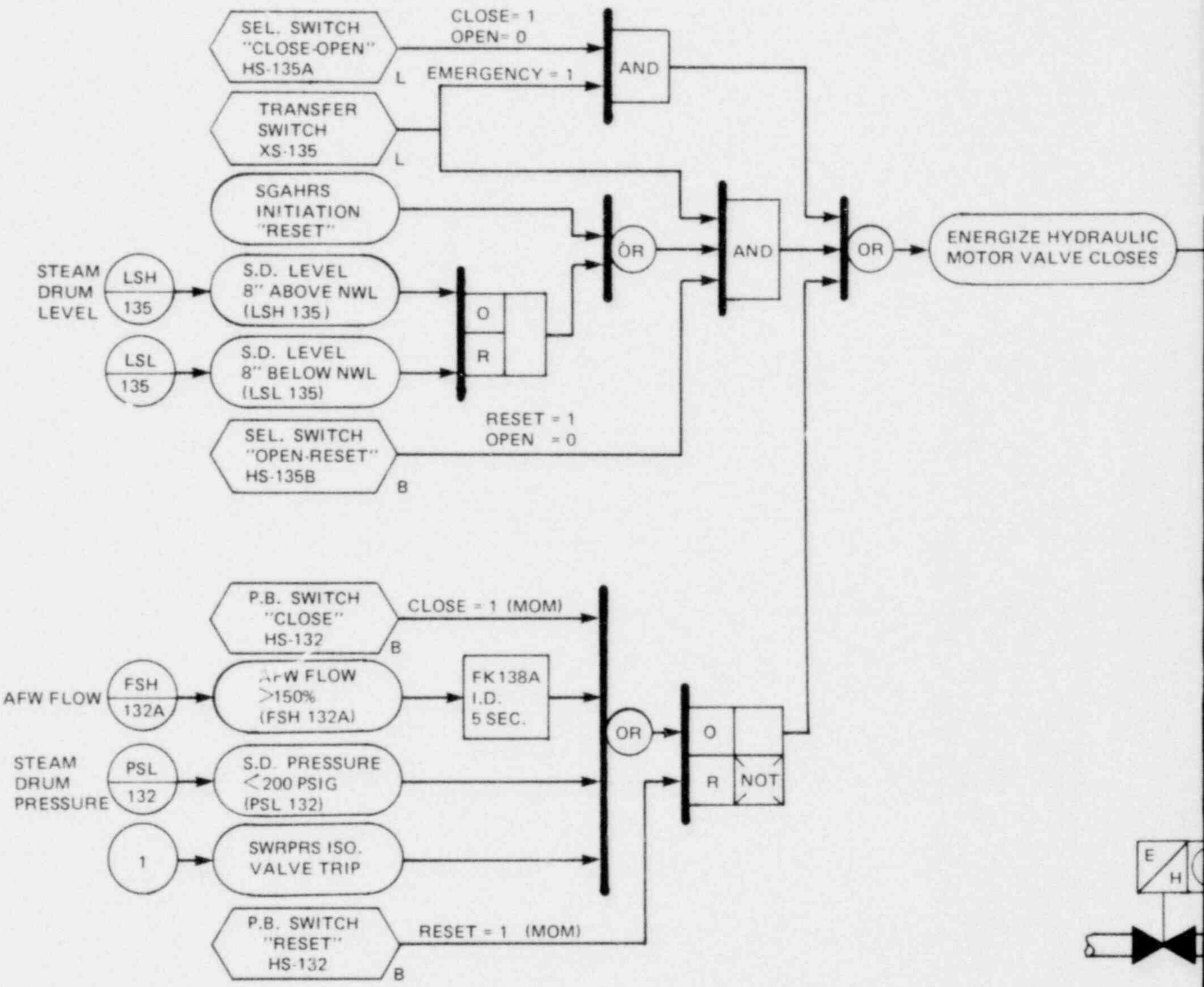
Figure 7.5 - 6 SWRPRS TRIP AND SWRPRS CONTROLLED ISC



POOR ORIGINAL

LATION VALVES CONTROL LOGIC DIAGRAM

SHEET 1 OF 6
80-433-08

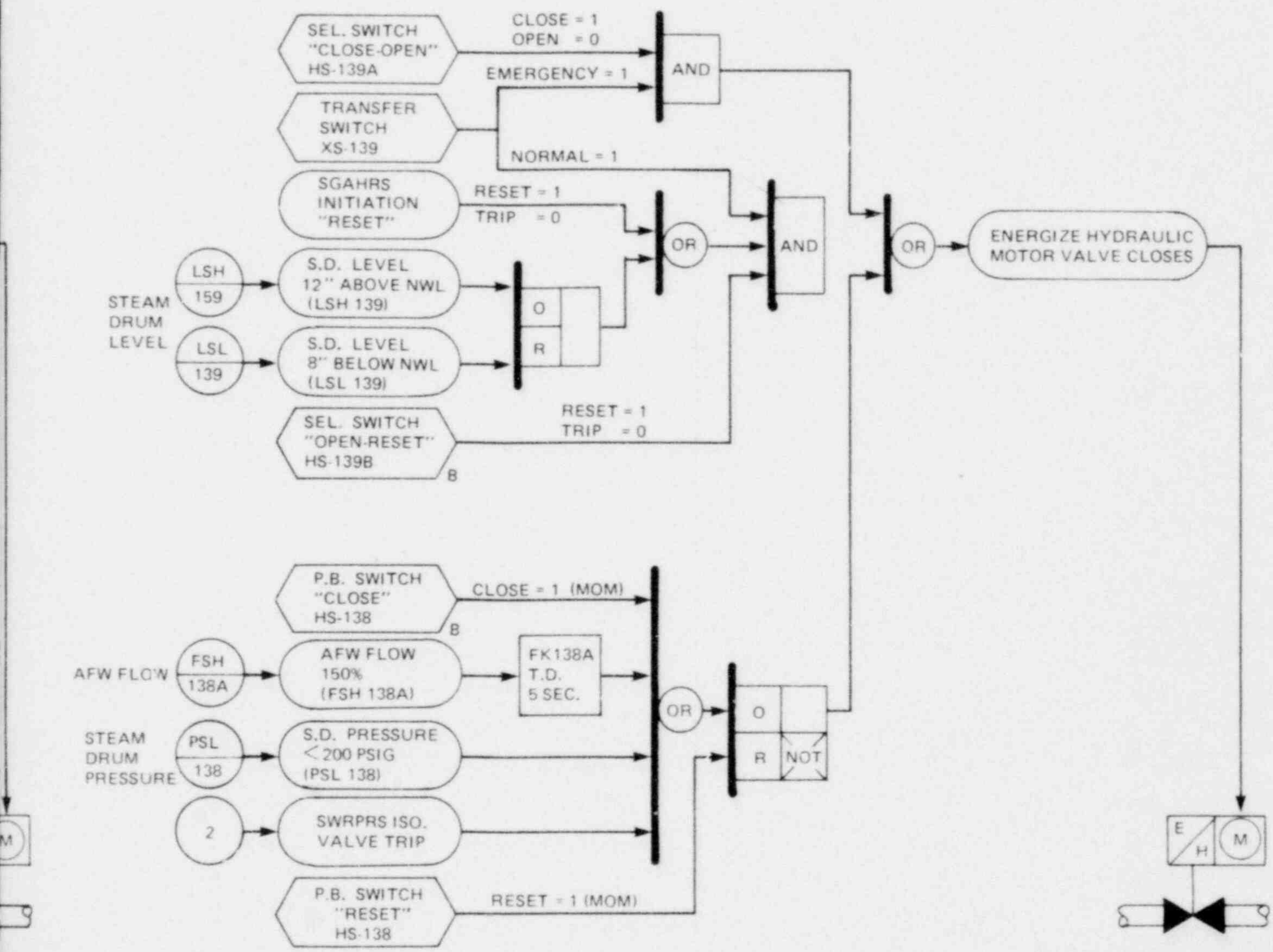


AUXILIARY FEEDWATER ISOLATION VALVE "A"

Figure 7.5 - 6 SWRPRS TRIP AND SWRPRS CON

POOR ORIGINAL

POOR ORIGINAL



AUXILIARY FEEDWATER ISOLATION VALVE "B"

SHEET 2 OF 6
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TROLLED ISOLATION VALVES CONTROL LOGIC DIAGRAM

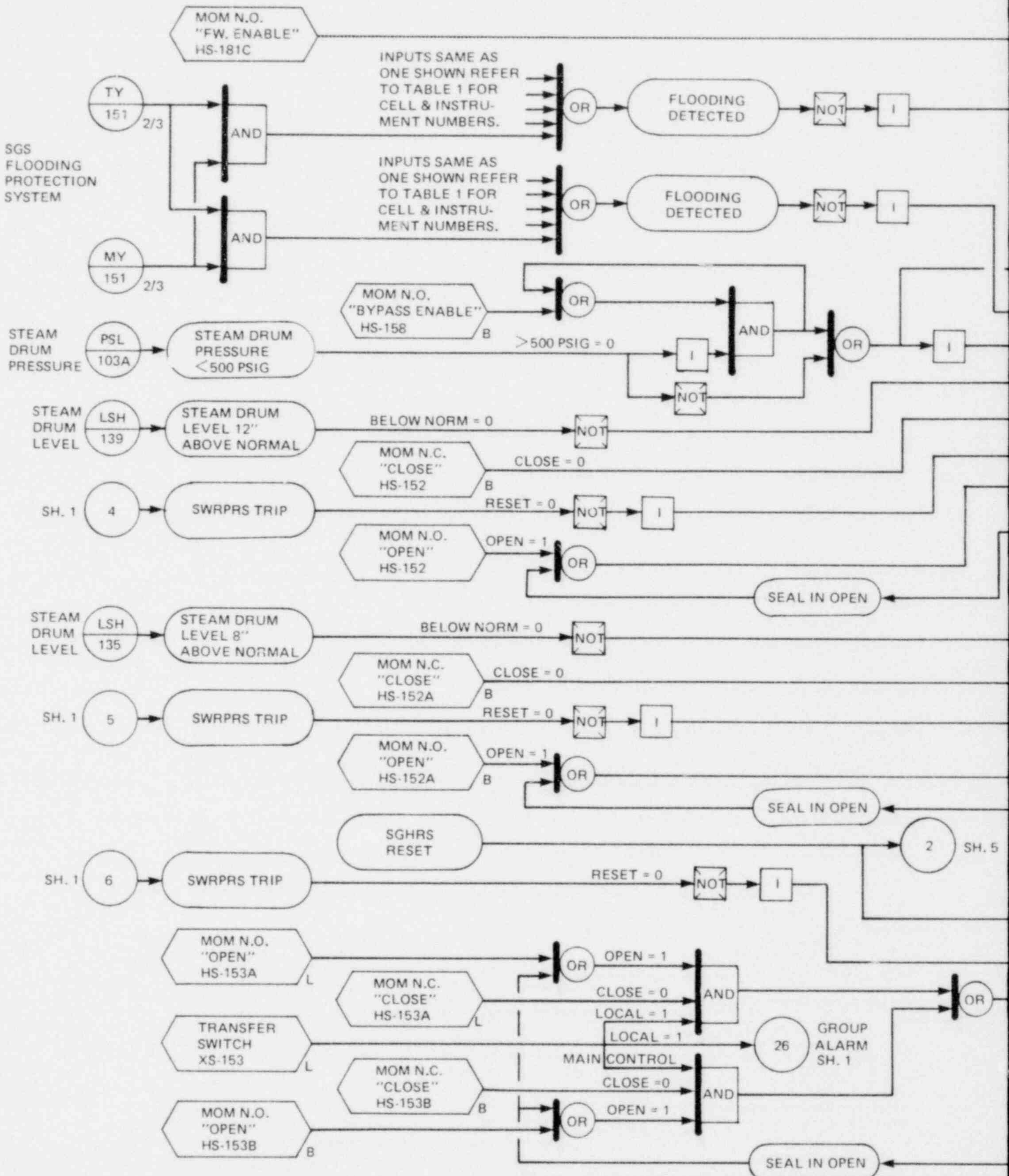
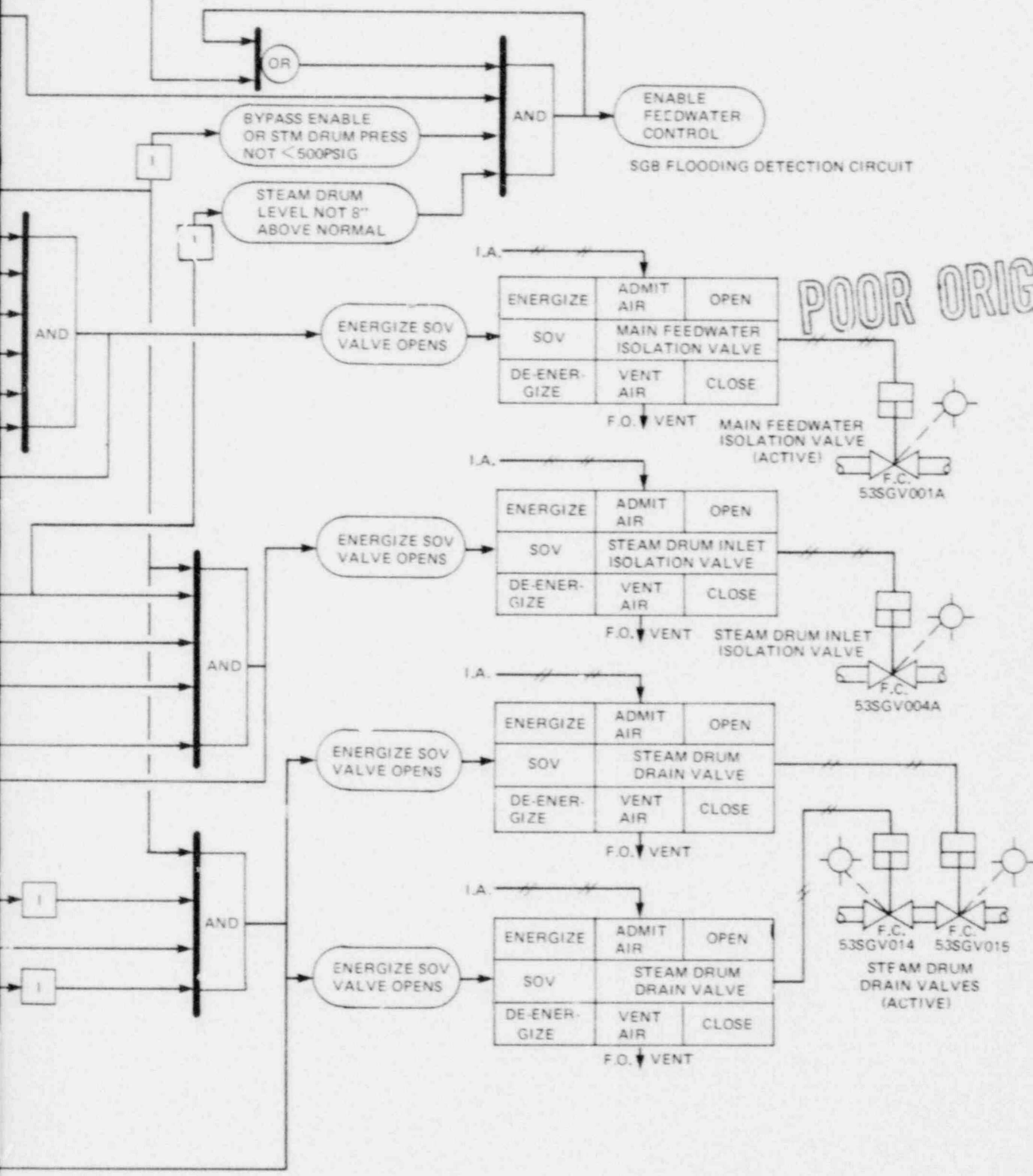


Figure 7.5 - 6 SWRPRS TRIP AND SWRPRS CONTROLLED IS

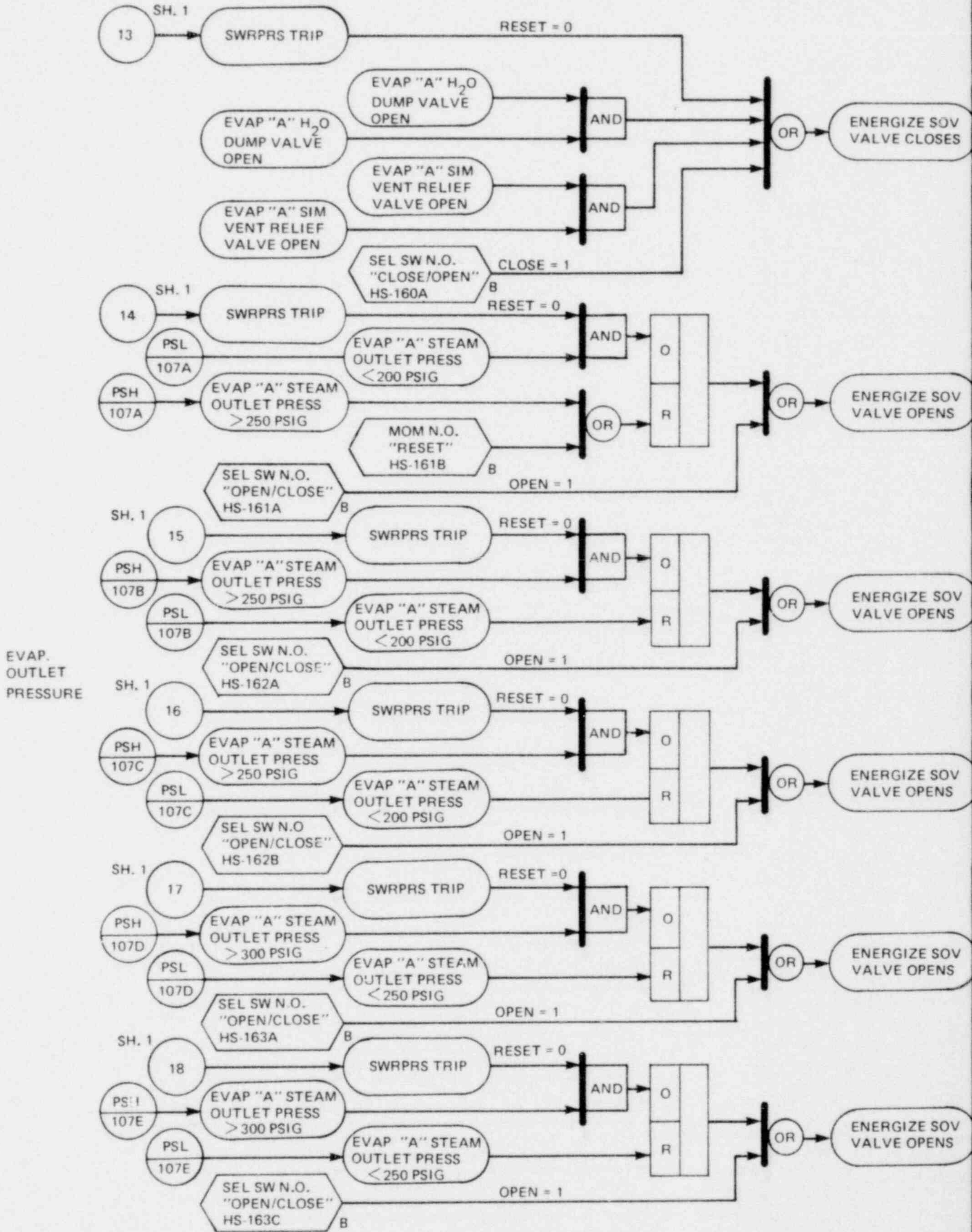
POOR ORIGINAL

TABLE 1												
LOOP NO.	1				2				3			
CELL NO.	241	221	224	207	242	222	225	208	243	223	226	209
INSTRUMENT NO.	151	152	153	154	251	252	253	254	351	352	353	354



OLATION VALVES CONTROL LOGIC DIAGRAM

SHEET 3 OF 6
80-433-06

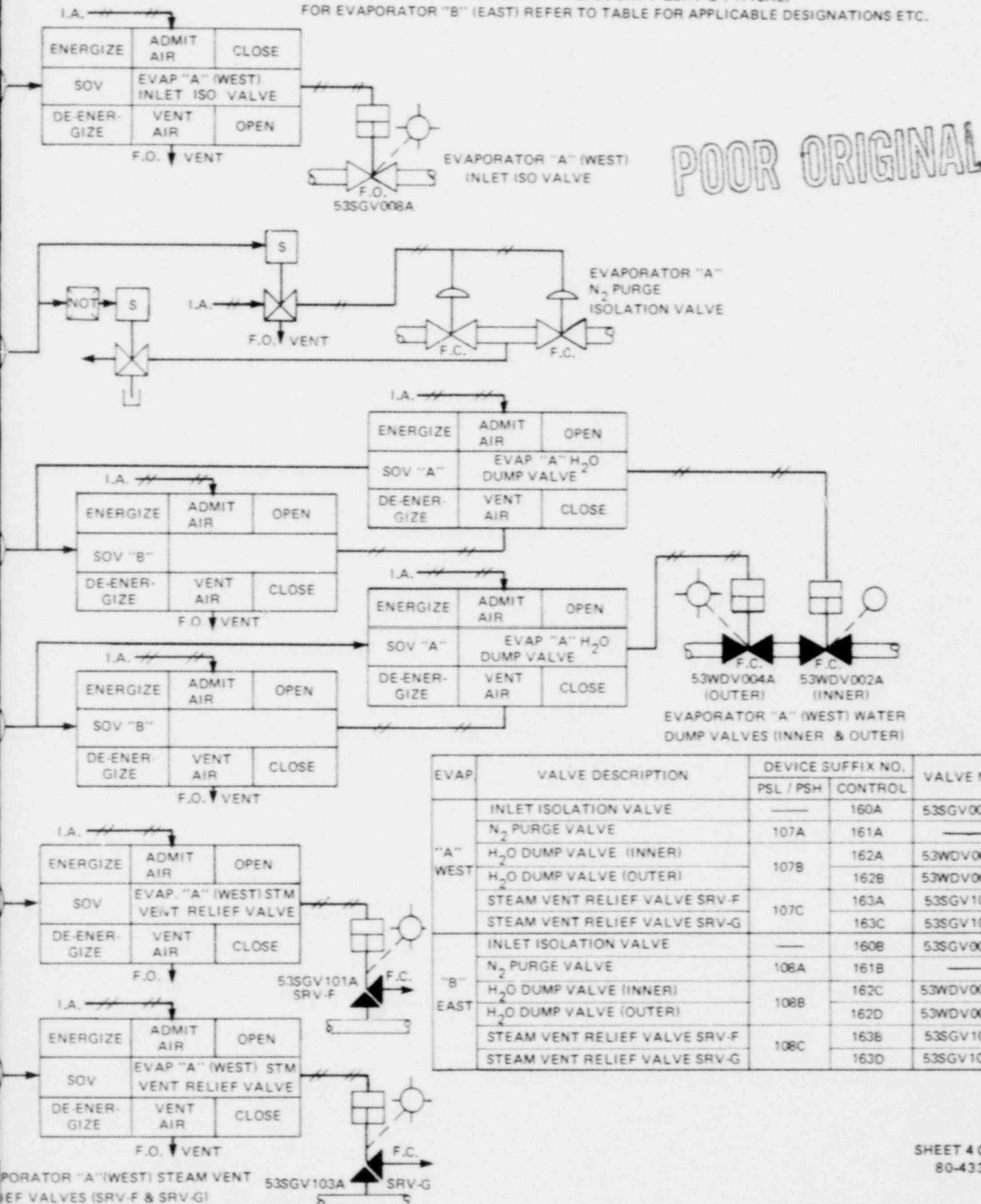


EVA
REL

Figure 7.5 - 6 SWRPRS TRIP AND SWRPRS CONTRO

POOR ORIGINAL

EVAPORATOR "A" (WEST) INLET ISO. VALVE, N_2 PURGE ISO. VALVE, H_2O DUMP VALVES INNER & OUTER AND STEAM PWR RELIEF VALVES SRV-F &SRV-G TYPICAL.
FOR EVAPORATOR "B" (EAST) REFER TO TABLE FOR APPLICABLE DESIGNATIONS ETC.



EVAP.	VALVE DESCRIPTION	DEVICE SUFFIX NO.		VALVE NO.
		PSL / PSH	CONTROL	
"A" WEST	INLET ISOLATION VALVE	——	160A	53SGV008A
	N_2 PURGE VALVE	107A	161A	——
	H_2O DUMP VALVE (INNER)	107B	162A	53WDV002A
	H_2O DUMP VALVE (OUTER)		162B	53WDV004A
	STEAM VENT RELIEF VALVE SRV-F	107C	163A	53SGV101A
	STEAM VENT RELIEF VALVE SRV-G		163C	53SGV103A
"B" EAST	INLET ISOLATION VALVE	——	160B	53SGV007A
	N_2 PURGE VALVE	108A	161B	——
	H_2O DUMP VALVE (INNER)	108B	162C	53WDV001A
	H_2O DUMP VALVE (OUTER)		162D	53WDV003A
	STEAM VENT RELIEF VALVE SRV-F	108C	163B	53SGV100A
	STEAM VENT RELIEF VALVE SRV-G		163D	53SGV102A

PORATOR "A" (WEST) STEAM VENT
IEF VALVES (SRV-F & SRV-G)

SHEET 4 OF 6
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POOR ORIGINAL

OLLED ISOLATION VALVES CONTROL LOGIC DIAGRAM

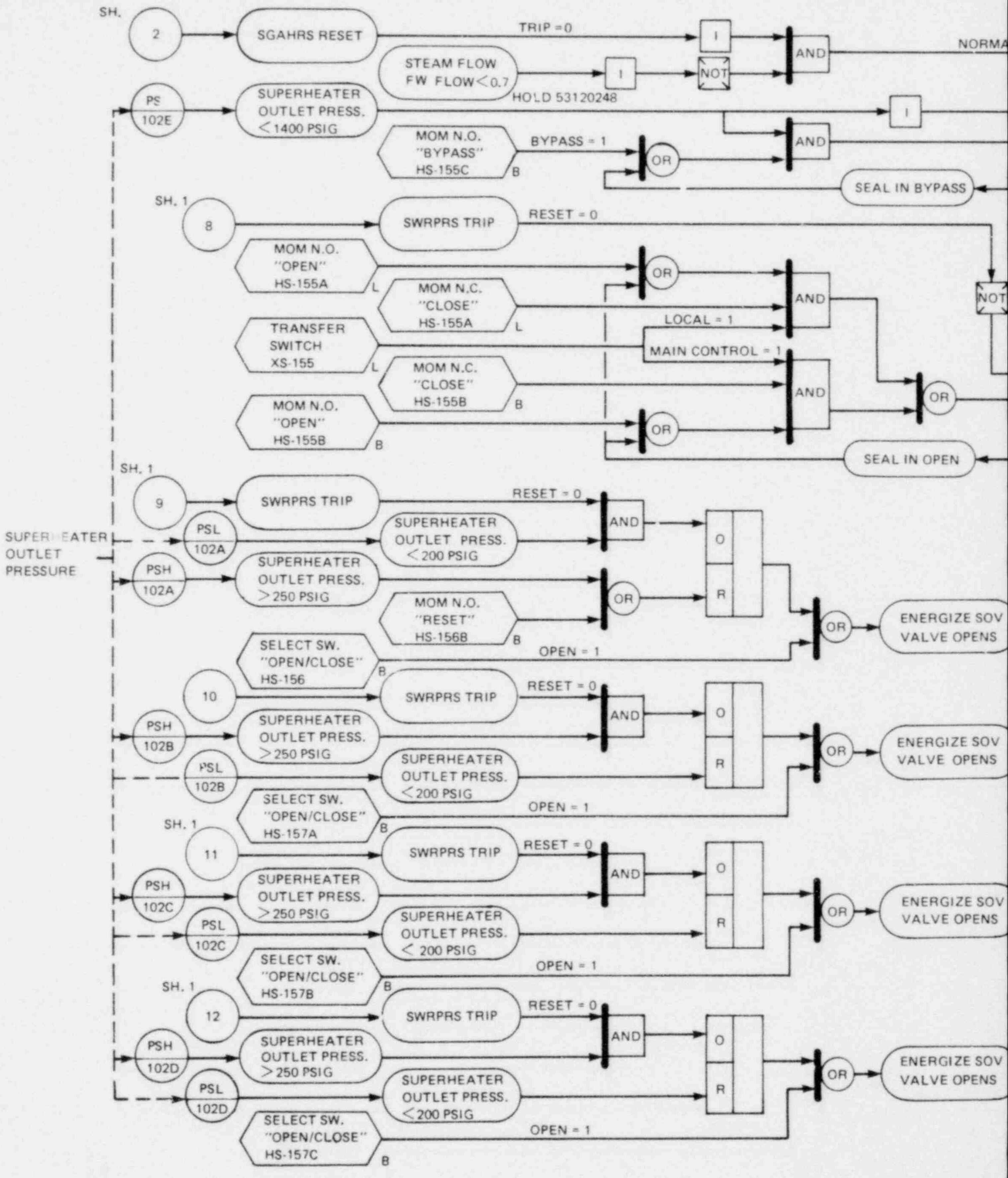
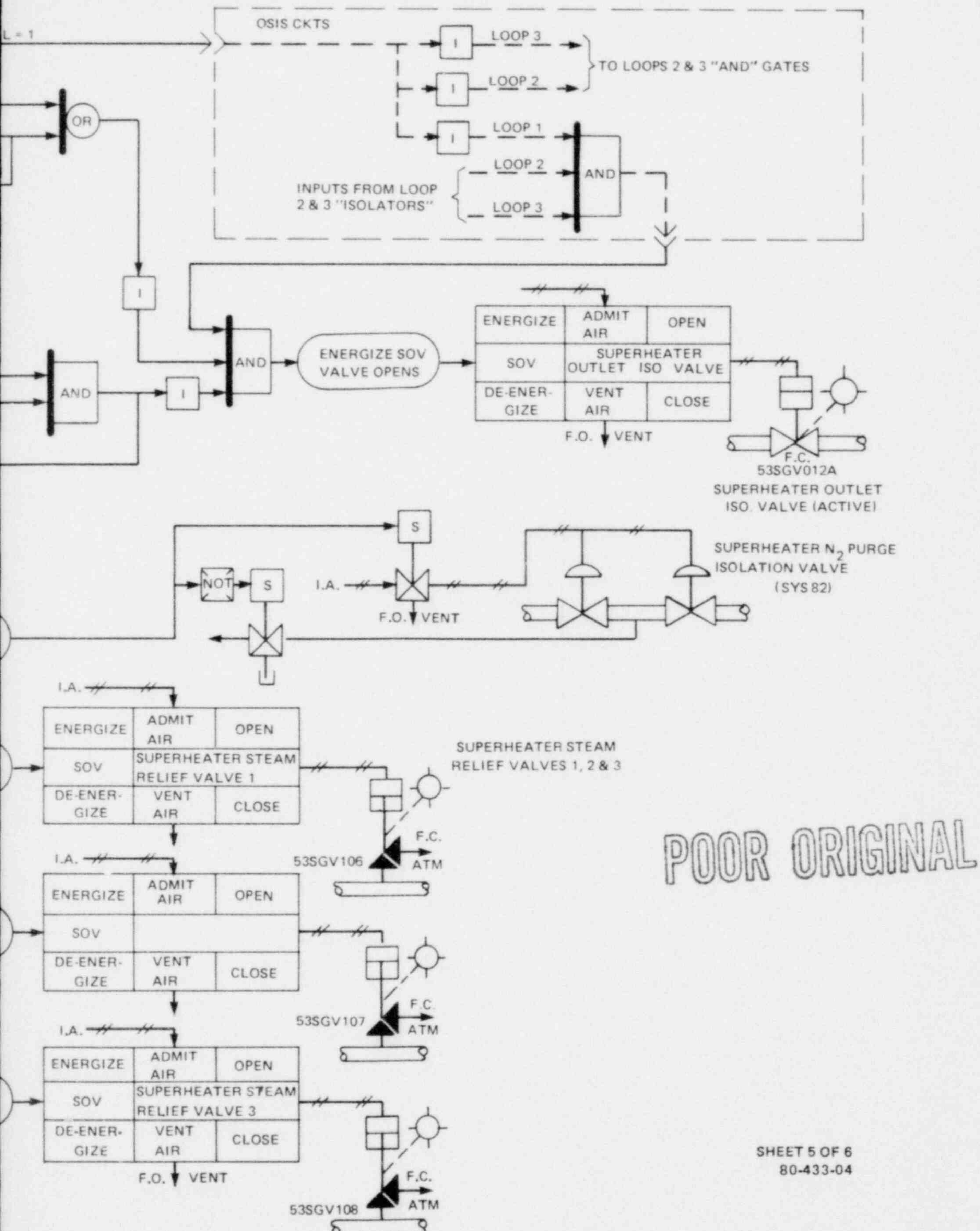


Figure 7.5 - 6 SWRPRS TRIP AND SWRPRS CONTROLLED ISOL



ATION VALVES CONTROL LOGIC DIAGRAM

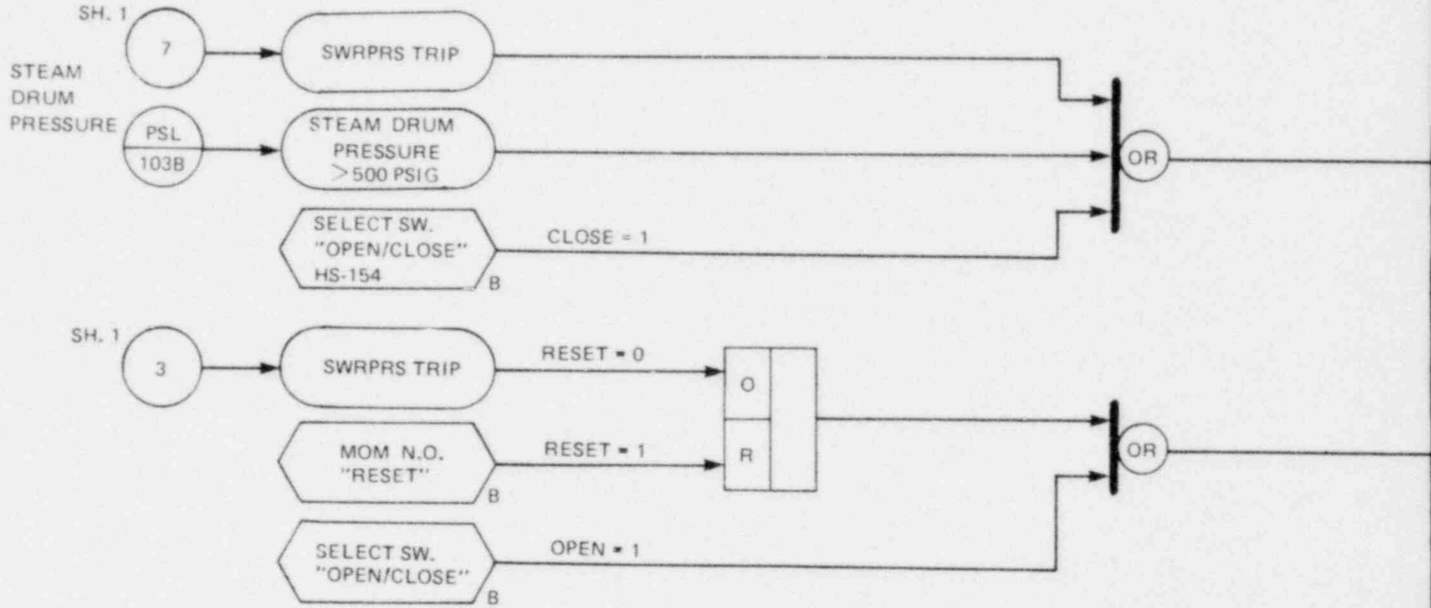


Figure 7.5 - 6 SWRPRS TRIP AND SWRPRS CONTRO

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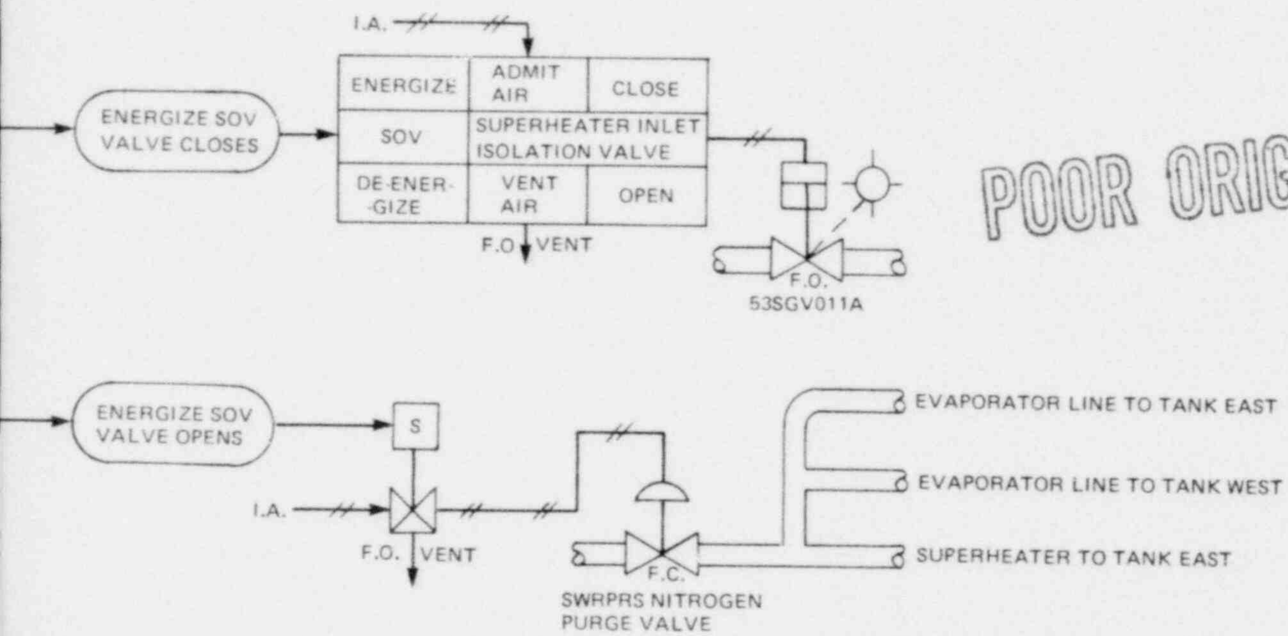

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LLED ISOLATION VALVES CONTROL LOGIC DIAGRAM

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7.6 OTHER INSTRUMENTATION AND CONTROL SYSTEMS REQUIRED FOR SAFETY

The additional instrumentation and control systems required for safety which have not been discussed earlier in Chapter 7 are identified as the Plant Service Water and Chilled Water Instrumentation and Control Systems, the Fuel Handling and Storage Safety Interlock System, and the Direct Heat Removal Service Instrumentation and Control. The Radiation Monitoring System also contains safety related components which are discussed in Chap-
33 ter 11. The Cooling of Structural Concrete is addressed in Chapter 3A. The Normal and Emergency Plant Service Water and Chilled Water Systems, Fuel
26 Handling, and DHRS I and C are discussed in this Section.

7.6.1 Plant Service Water and Chilled Water Instrumentation and Control Systems

7.6.1.1 Description

Those portions of the Plant Service Water and Chilled Water Systems which are required for safety, include the Emergency Chilled Water System, and the Emergency Plant Service Water System (see Sections 9.7.2, 9.9.2).
33 15 Instrumentation and control for these systems will include the necessary redundant instrumentation, control and indicating circuits and devices required for operation of the system.

7.6.1.2 Analysis

As required by IEEE Standard 279-1971, redundant monitoring and control equipment will be provided to ensure that a single failure will not impair the capability of the Emergency Plant Service Water and Chilled Water Instrumentation and Control Systems to perform their intended safety functions. The systems will be designed for fail safe operation and control equipment, were practical, will assume a failed position consistent with its intended safety function.

To comply with CRBRP General Design Criterion 19, adequate instrumentation and control in the control room and locally, will be provided for those portions of these systems which are required for safety.

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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.

7.6.5.3.1 Instrumentation

Instrumentation provided for this subsystem consists of Class 1E temperature, and moisture transducers. In addition, non-Class 1E level transducers are provided. The transducers and associated control logic are located in the SGB cells containing main feedwater or recirculation piping. Three independent moisture and temperature measurements in each cell are utilized for identifying a major water/steam line rupture. Water level measurements in each cell confirm a flooding condition and are annunciated in the main control room.

7.6.5.3.2 Controls

Each heat removal loop isolates the main feedwater supply upon detection of a major pipe rupture. The start-up and main feedwater control valves close upon activation by a two-out-of-three logic using measurements of moisture and temperature in each cell. The main feedwater isolation valve is independently closed upon activation by a two-out-of-three logic using the same three moisture and temperature measurements from each cell. Separation and isolation is maintained between the control valve and isolation valve activation logic.

Small water/steam leaks are identified in each SGB cell by measuring water level. Manual corrective control of flooding is initiated by the operator upon annunciation in the main control room. 45

TABLE 7.6-1

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7.7.1.3 Primary and Secondary CRDM (Control Rod Drive Mechanism) Controller and Rod Position Indication

The Primary Control Rod Drive Mechanism Control System transforms the bulk 3 phase power into the pulsed DC necessary to operate the Control Rod Drive Mechanism in response to input commands from the Reactor Control System. Interlocks and permissives are provided to prevent operating sequences of the control rods which would damage the equipment, and assure that the rods are maintained in the banked configuration required to maximize core performance. Rod Position Indication is provided redundantly for each rod to permit the operator to verify the reactivity status and operation of the control system. The Secondary CRDM Controller and Rod Position Indicators are described in Section 4.2.3.

7.7.1.3.1 Primary CRDM Control

The control rod drive mechanism is actuated by a 4 pole, 6 winding reluctance stepping motor. The mechanism lead screw has a thread pitch of 0.6 inch, and moves 0.025 inches for each pulse to the drive stator. A block diagram of the drive system is shown in Figure 7.7-4. Driving power is supplied from the site power system through redundant motor-generator sets, Reactor Shutdown System scram breakers, a 3 phase to 6 phase transformer, and banks of silicon controlled rectifiers (SCR's) in the individual controllers to the stator windings of the CRDM.

The primary rods are divided into 2 groups. One group of 3 startup
59 rods responds only to single rod manual control and during
reactor operation is normally fully withdrawn. A group of 6 control rods
respond to manual control or to an analog signal from the Reactor Control
57 System.

Rod speed demand limits are included in the reactor controller as well as rod speed limits in the individual Primary CRDM Controllers. Rod block interlocks are included in the "OUT" demand input as shown on Figure 7.7-5.

When the Reactor Shutdown System initiates a scram, the "Scram Breakers" open and interrupt the power to the Primary CRDM stator coils; the rotor collapses and disengages the rollers from the lead screw; and the CRDM drive train falls under the force of gravity and the scram assist spring to insert the control rods into the core. Failures within the sequence and controller units cannot prevent removal of the power required to hold the CRDM's in the withdrawn position. The components are described below. 19

Motor-Generator Set

57 Dual M-G sets provide the 3 phase power for CRDM operations.
When a latch signal is received at the voltage control, the output voltage of a generator is increased. The M-G sets for the primary rods use a 200 Hp motor and a 150 Kw generator.

Mechanism loads are shared by the two M-G sets; however, either M-G set has the capacity to power the entire load of the primary system. Controls are provided to synchronize the two M-G sets. The motor-generator sets are designed to provide sufficient inertia and voltage control to prevent rods dropping in the event of power dips of 0.3 seconds or less.

Generator output circuit breakers provide the necessary electrical
57| protection for the generators and for system maintenance

Power Supplies and Transformers

57| Since the CRDM controllers use 6 phase AC power, one 3 phase to 6 phase transformer is provided for the primary rods. The transformer includes appropriate secondary side surge protection.

Control of the argon supply and vent valves is accomplished by an "on-off" type pressure controller which cycles the supply and vent valves to maintain the cover gas pressure between the lower and upper limits. Sufficient dead band is provided between lower and upper limit operation to prevent undue cycling of the supply and vent valves.

Operational Considerations

The pressure controllers for the sodium dump tanks are located at the local control panel in the Steam Generator Building. However, manual overrides for the supply and vent valves are provided in the main control room and may be utilized at the plant operator's discretion.

High and low pressure alarms alert the operator to off-normal conditions which may result from a malfunction of the pressure control system. Pressure data is provided to the Data Handling and Display System and is available for display upon call by the operator.

7.7.1.8 Steam Dump and Bypass Control System

The Steam Dump and Bypass Control System provides the necessary control and instrumentation hardware to operate the Turbine Bypass System as described in Section 10.4.4 and shown in Figure 10.3-1.

Redundant interlocks are provided to prevent bypass operation in the event the condenser is unable to accept steam flow (e.g., high condenser back pressure or loss of circulating water flow).

Independent instrumentation and control channels are provided for the detection or load rejection (e.g., reactor power exceeds net generation by 20%). These two (2) channels are arranged for 2/2 coincidence and failure of one channel in any mode will not cause the operation of the Steam Dump and Bypass Control System. Time response will be such that the proper Bypass valve(s) lift will be obtained within three (3) seconds of system actuation to maintain bypass steam flow, approximately proportional to reactor power.

A pressure control channel is provided for the regulation of main steam pressure following reactor trip, during decay heat removal operation and during turbine standby, loading and unloading operations. The pressure control mode is automatically selected for reactor power levels below 40% and will operate two (2) of the four (4) bypass valves.

At reactor power levels above 40%, the Steam Dump and Bypass Control System automatically positions bypass valve(s) to regulate bypass steam flow approximately proportional to reactor power, however; the pressure control mode may be manually selected by operating personnel at any power level.

7.7.1.9 Fuel Handling and Storage Control System

The Fuel Handling and Storage (Reactor Refueling) Control System consists of electronic, electrical and mechanical hardware and software integrated into a coordinated system through a single common center utilized primarily for control of the refueling equipment used to load and unload core components and move them in and out of the plant as required. Emphasis is placed on optimizing control of the machines and facilities used repetitively during the annual refueling cycle [i.e., In-Vessel Transfer Machine (IVTM), Ex-Vessel Transfer Machine (EVTM) Reactor Rotating Plugs (RRP) and Ex-Vessel Storage Tank (EVST)]. See Section 9.1 for description of refueling equipment. However, control equipment for the other refueling equipment (i.e.,
20 Auxiliary Handling Machine (AHM) and Fuel Handling Cell (FHC) is also provided. Secondary functions of this system are: (1) data acquisition as required by the Plant Data Handling and Display System, (2) inventory of all core components in the plant and their locations, (3) providing system status indication and display to the refueling supervisor, (4) and monitoring of refueling equipment for out-of-limit conditions.

The Refueling Control System provides an efficient means of controlling the fuel handling machines used for reactor refueling. The control system design makes use of computer control for simple linear and rotary motions which are repeated many times during a refueling cycle. This automation reduces the possibility of operator error and provides for automatic inventory control.

The following definitions are important to an understanding of the Fuel Handling and Storage Control System design:

- EVENT - A change of state that occurs in essentially zero time, e.g., the closing of a limit switch.
- ACTION - A change of state that occurs between two events, e.g., the motion of a gate valve between the open and closed position.
- SEQUENCE - A series of actions which are always performed in the same order; for example, the various actions associated with pressurizing and depressurizing the seals associated with closing the EVTM closure valve constitute part of the closure valve sequence.

Three levels of control and automation are included in Fuel Handling and Storage Control System:

- Manual Systems (little or no automation) in which all actions are initiated individually by an operator. This is accomplished with pushbuttons, handcranks, or other appropriate means. Sufficient
59 information is displayed to the operator for verification that the preconditions for the next action have been satisfied.

- Semiautomatic systems in which each sequence is initiated manually by an operator, but actions within each sequence are initiated automatically by the control system. In each case, prior to initiating an action, the control system must have the capability for verifying that the previous action in the sequence has been successfully completed, and that any other preconditions for initiation of the next action have been satisifed. Information must be displayed to the operator for verification that the preconditions for sequence initiation have been satisfied.

- Moderately automated systems in which all actions within each sequence are initiated automatically, as in semiautomatic systems, and in addition, some sequences are initiated automatically. For automatic initiation of sequences, the control system must have the capability for determining which sequence should be initiated following completion of a previous sequence, as well as being capable of determining that all necessary conditions are met prior to sequence initiation.

59 Interlocks as described later in this section are incorporated to prevent the operator or automatic control equipment from initiating actions or sequences that result in equipment damage.

The following definitions are used to denote equipment location:

- CENTRAL - The control equipment is situated in a control room known as the Communications Center located at the far end of the gantry rails in the Reactor Service Building (RSB).

- LOCAL - The control equipment is situated in control consoles or racks located on or near the applicable refueling machine.

The degree of automation and centralization are interrelated. All manual systems are also local systems. Semiautomatic systems are local. Moderately automatic systems have central sequence and local action initiation. Whenever automatic control is used, a local manual backup is provided for use in case of failure in the automatic control equipment.

Figure 7.7-8 is a block diagram showing the interconnection of the control hardware and the man-machine interfaces. The EVTM, IVTM and EVST all have automatic local control of sequences with manual local backup control.

44 The Central Computer in the Communications Center handles the following items: Core component inventory control (both by core component location and by storage location contents), system level data acquisition from all

refueling machines, data storage and transmission to the Plant Data Handling and Display System, over-all system status information acquisition and display as required for Communications Center operating personnel, and monitoring of all interlock and annunciator status.

The EVTM computer functions as follows: receives and processes data from instrumentation and controls on the EVTM, and passes data to the Central Computer as required; provides action initiation for the EVTM; and displays data required for the EVTM operator.

The IVTM computer functions as follows: receives and processes data from instrumentation and controls on the IVTM, and passes data to the Central Computer as required; provides action initiation for the IVTM from sequence initiation commands given by the IVTM operator, processes data for core component identification done by the IVTM, and passes this data to the Central Computer.

Control consoles are located in the Communications Center and at or near all the machines. These consoles provide input for command and data output for the computers, display of required instrumentation and control data, manual backup to automatic control for the machines, annunciator displays, instrumentation signal conditioning and power distribution to the machines. Where required, key switches will be provided to override inter-
59 locks for maintenance, initial set-up, and calibration. In order to override these interlocks, it will be necessary for a keyswitch to be activated by the responsible operator; this action will also be annunciated in the Communications Center. All signals from control computers to control equipment can be manually initiated from the local console in the event of a computer failure.

Instrumentation and control cables are run from the machine locations to the local console and computer system and in some cases back to the Commu-
59 nications Center. Synchronous data links are run between the EVTM Computer and Central Computer, between the IVTM computer and Central Computer, and between the Central Computer and the Plant Data Handling and Display System. Cables are run through a flexible tray system along the gantry rails to the EVTM; and through flexible trays on the reactor head rotating plugs to the IVTM. Cables to the EVTM floor valves are run through the EVTM. All cables will be appropriately shielded and routed so as to minimize noise pickup from other cables or from other equipment (welding, etc.) which might be in operation during the refueling period.

The method of computer application (computer-generated setpoint control) used in the Fuel Handling and Storage Control System has several features which will tend to reduce operator errors and which facilitate the use
59 of an interlock system as described later in this section. Among these features is the computerized fuel inventory which enables cross-checking of every core

component movement prior to initiation. The type of core component is checked for compatibility with the intended destination. The destination for the core component is checked for occupancy and readiness to receive a particular core component. Core components can be identified by the IVTM to verify the type of core component prior to any movement into the reactor core or removal from the Reactor Vessel. The Central Computer monitors the operation of the other refueling machines and incorporates a software operational alarm system to add further depth to the design for operation without errors. The use of setpoint generation rather than direct digital control permits the IVTM and EVTM computer commands to be passed through a permissive hard-wired interlock system only if proper preconditions are met. In addition, the Central Computer monitors annunciator status and alarm failures. An alarm log can be displayed at all local computer CRT terminals.

Finally, a complete manual control capability is provided which also must work through the refueling interlock logic.

The analysis of the consequences of specific fuel handling events given in Section 15.5 has not identified a requirement for any specific safety interlocks.

Some interlocks are included in the design to preclude the possibility of major machine damage.

Typical interlocks are given below and in Table 7.7-1.

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

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 there-
59 fore concluded that the RRS interlocks should not be designated as safety interlocks.

7.7.1.10 Nuclear Island Auxiliary Instrumentation and Control Systems

A number of Instrumentation and Control Systems, not discussed in Section 7.0, are provided in the plant to support various auxiliary systems. These systems do not perform a safety-related function, nor would their failure prevent the functioning of a safety-related system. These instrumentation systems, discussed in other sections of this report are:

System	Section
Recirculating Gas	3.A.1, 3.A.2
Auxiliary Cooling Fluid	9.7.5
Heating and Ventilating	9.6
Sodium Fire Protection	9.13.2
Inert Gas Receiving and Processing	9.5
Impurity Monitoring and Analysis	9.8.5
Auxiliary Liquid Metal	9.3

7.7.1.11 Balance of Plant Instrumentation and Control Systems

A number of Instrumentation and Control Systems are provided to support various Balance of Plant Systems. These systems do not perform a safety-related function, nor would their failure prevent the functioning of safety-related systems.

7.7.1.11.1 Treated Water Instrumentation and Control System

The Treated Water System includes the Portable Water System, the Normal Plant Service Water System, the Secondary Service Closed Cooling Water System, The Emergency Plan Service Water System, the Normal and Emergency Plant Chilled Water Systems, and the Makeup Water Treatment System.

The Treated Water Instrumentation and Control System provides the instrumentation, control and indicating circuits and devices required for operation of the system. Section 9.9 covers this equipment.

Only the Emergency Plant Service and Chilled Water Systems are required for emergency safe shutdown after an accident or loss of AC power. Safety-related instrumentation for the Treated Water System is discussed in Section 7.6.1.

7.7.1.11.2 Waste Water Treatment Instrumentation and Control System

The Waste Water Treatment System provides for the treatment of non radioactive contaminated waste water, sanitary waste water and post-treatment facilities.

The Waste Water Treatment Instrumentation and Control System provides the instrumentation, control and indicating circuits and devices required for operation of the system.

7.7.1.11.3 Remaining Systems

Instrumentation and Control for the following systems are described elsewhere:

System	PSAR Section
Turbine Generator Control System	10.2.2
Feedwater and Condensate	10.4.7
Main Steam Supply	10.3
Circulating Water System	10.4.5
River Water Service System	9.9.7
Compressed Gas System	9.10
BOP Heating and Ventilating	9.6
BOP Fire Protection	9.13.1

7.7.2 Design Analysis

As described in Section 7.2.2, the Plant Control System is separate from the Plant Protection System; any failure of the Plant Control System will not affect the capability of the Plant Protection System but may result in a plant trip. Section 7.7.1 explains the interrelationship between the supervisory control and each of the plant sub-loop controls.

Where instrumentation signals are provided to the Plant Control System by the Plant Protection System (e.g., Nuclear Flux), multiple failures of the PPS sensors could cause the loss of instrumentation channels in both the Plant Protection and Control Systems. This could cause a Control System action that

7.7.2.4 Steam Generator Feedwater Flow Control System

The Steam Generator Feedwater Flow Control System contains a feedwater flow controller (typical of three) and feedpump controller. The failure of either controller would result in improper actuation of the Feedwater Control Valve and, consequently, an undesirable feedwater flow. This, in turn, would generate a change in drum level and a mismatch in steam flow to feedwater flow. The Plant Protection System has protective subsystems for each of these anomalies.

7.7.2.5 Balance of Plant Instrumentation and Control

The Balance of Plant Instrumentation and Control Systems described in Section 7.7.1.11 do not perform safety related functions, nor would their failure prevent the functioning of a safety related system.

The most severe transient introduced because of these controls is a turbine trip; the transient is described in Section 15-3.

Plant Fire Protection Instrumentation and Control System failure would not prevent the functioning of a safety-related system. Accident analysis for conventional and maximum fires is provided in Section 15.6 and 15-7.

59 TABLE 7.7-1

USE OF REFUELING INTERLOCKS

Event*	Cause	First Line of Defense	Second Line of Defense
44 Rotation of one or more rotating plugs with assembly partially inserted into core or storage location	Operator error when in manual control	Interlock prevents energizing rotating plug drive motors unless IVTM grapple and holdown sleeve are fully raised.	Redundant switch independently prevents energizing rotating plug drive motors unless IVTM grapple and holddown sleeve are fully raised.
59 Rotation of one or more rotating plugs while EVTM is over reactor	Operator error when in manual control	Interlock prevents energizing rotating plug drive motors if EVTM is within 14 ft of the reactor transfer port position, as indicated by rail switches.	Redundant interlock prevents energizing rotating plug drive motors if EVTM is within 14 ft. of the reactor transfer port position, as indicated by rail switches.

44 *Occurrence of Event could cause machine damage, but no unacceptable hazard to public health and safety.

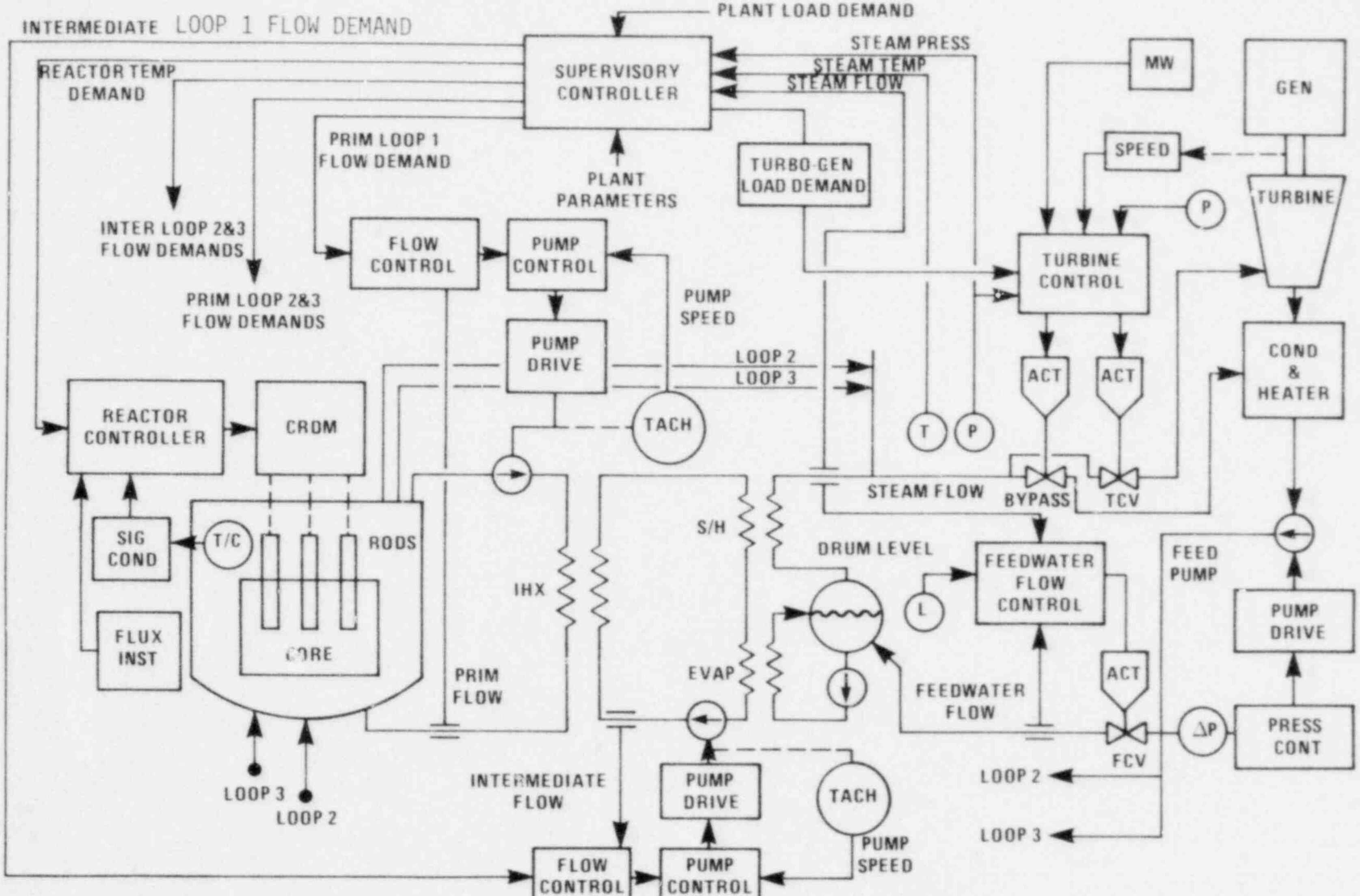

Figure 7.7-1. Plant Control System

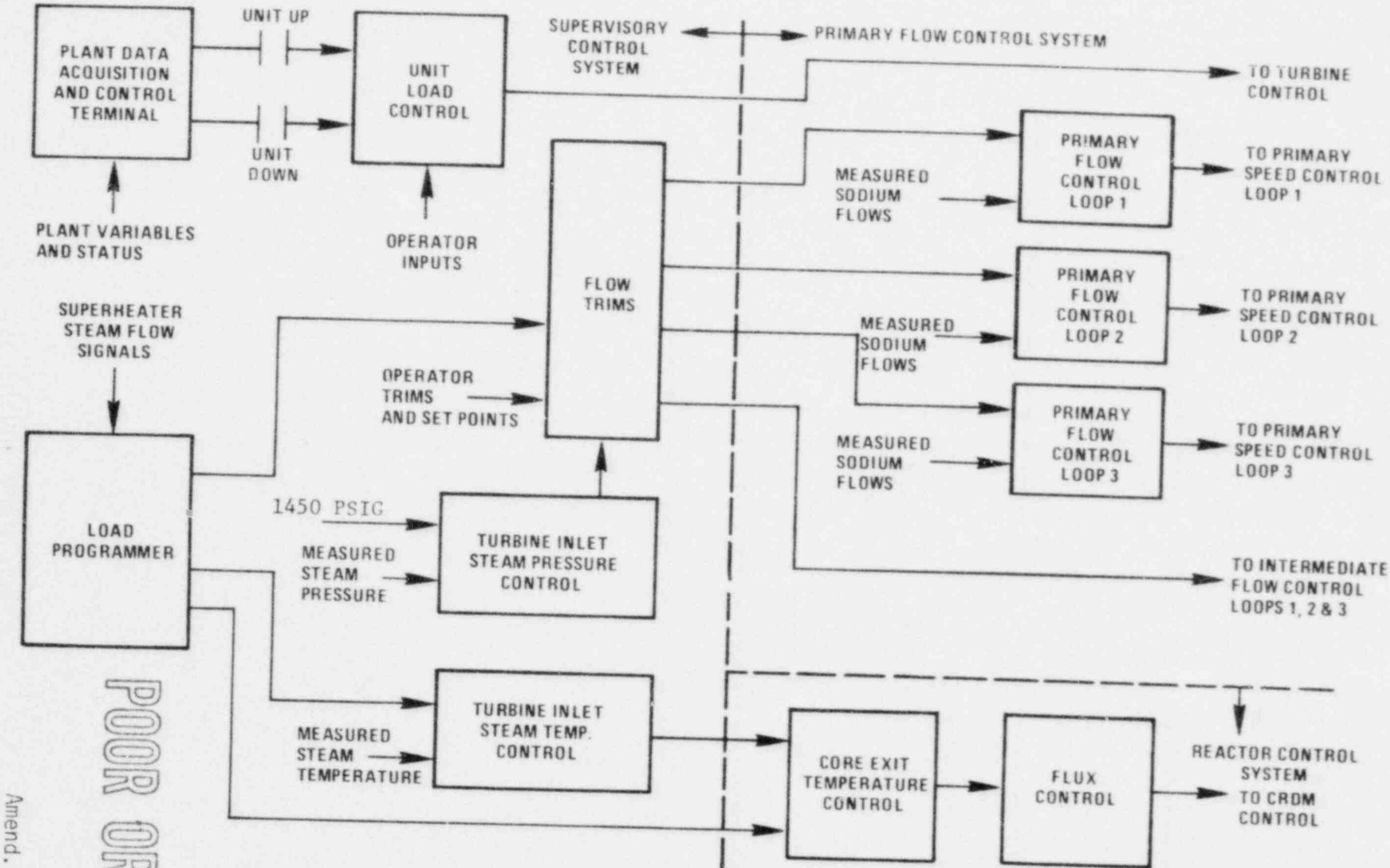

Figure 7.7-2 Supervisory Control System

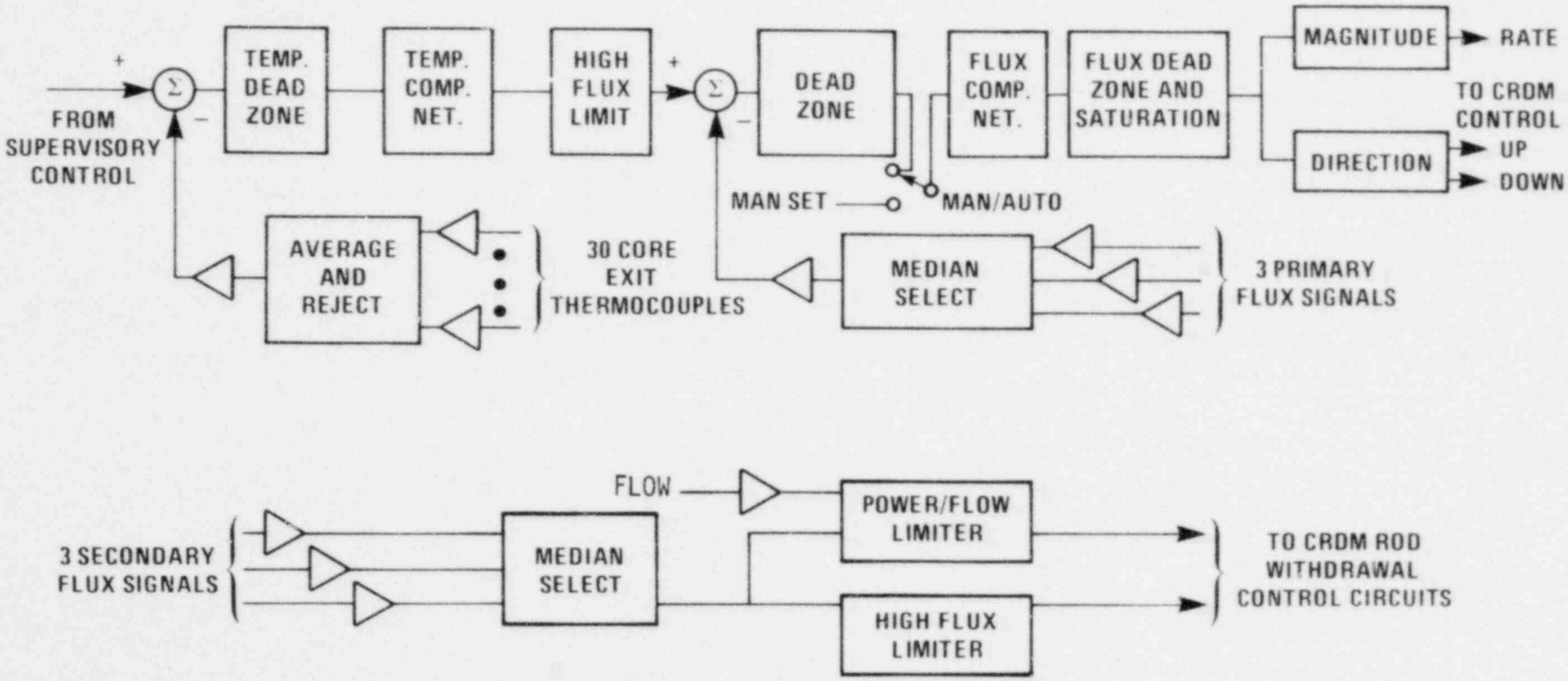

Figure 7.7-3. Block Diagram of Reactor Control System

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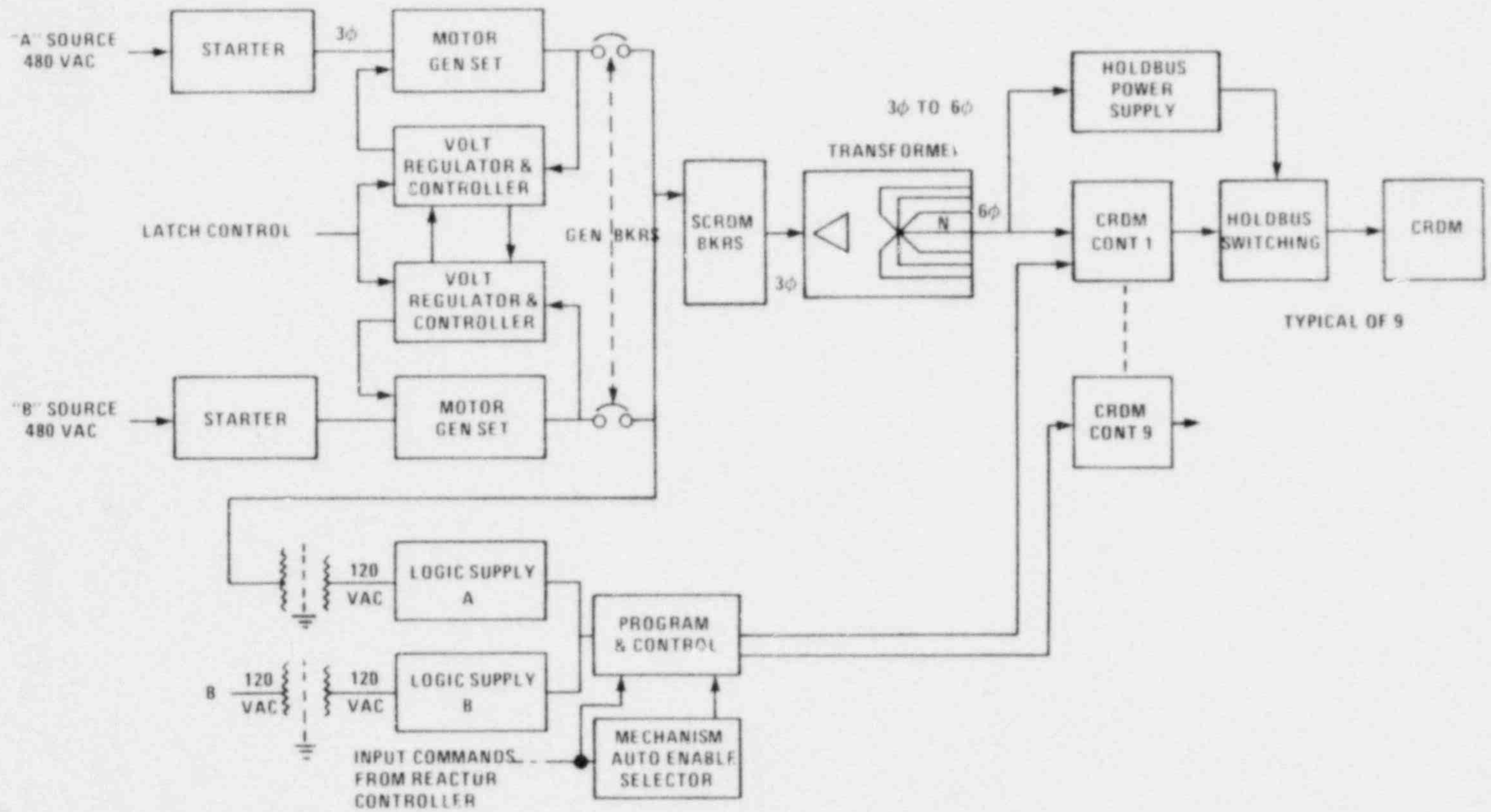

Figure 7.7-4. CRDM Controller and Power Train for Primary Rods

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9.1 FUEL STORAGE AND HANDLING

The Reactor Refueling System provides the means of storing, trans-
20 porting, and handling core assemblies and core special assemblies within the
44 CRBRP.* The following are defined as core assemblies:

1) Fuel assemblies
2) Inner blanket assemblies
3) Radial blanket assemblies
4) Control assemblies
5) Removable Radial shield assemblies

44 59 Of the new core assemblies arriving at the CRBRP, only new fuel assemblies require shielding and criticality control, and present any potential radiation safety hazard. However, after irradiation in the reactor core, all core assemblies require shielding and removal of decay heat in
44 various degrees. Irradiated core assemblies containing fuel also require criticality control and containment of fission gas in the event of leaks in fuel rods.

The Reactor Refueling System consists of the facilities and equipment needed to accomplish the normal scheduled refueling operations, and all
44 other functions incident to handling of core assemblies.

Its primary functions are as follows:

44 1) Receive, inspect, store, and prepare new core assemblies for insertion in reactor

44 2) Transfer core assemblies between facilities (i.e., ex-vessel storage tank, reactor, fuel handling cell and new fuel unloading stations)

44 3) Transfer core assemblies between the core and in-vessel storage or transfer positions, or between core positions

44 4) Provide storage for irradiated core assemblies

20 *The purpose of core special assemblies is to facilitate core loading and unloading. Due to the tight tolerances of the discriminator post with
20 respect to the socket, the vertical position of a core special assembly has a limited deviation from the vertical center line when it is "standing
44 freely." This makes insertion or withdrawal of neighboring core assemblies
20 easier during core loading or unloading, respectively. Core special assemblies do not contain any fuel and are removed prior to reactor startup. They have the same handling socket, except for identification number, as the
44 core assembly they replace.

5) Examine and prepare irradiated core assemblies for shipment

44 6) Provide inventory control of all core assemblies.

The major equipment and facilities needed to perform these functions are as follows:

44 Function 1) New fuel shipping containers, new fuel unloading stations, and ex-vessel storage tank (EVST)

20 Function 2) Ex-vessel transfer machine (EVTM)

Function 3) In-vessel transfer machine (IVTM)

Function 4) EVST

44 Function 5) Fuel handling cell (FHC)

Function 6) Refueling instrumentation and control system, FHC and IVTM.

Major equipment and facilities needed to perform functions incident
44 to the handling of core assemblies and their functions are as follows:

1) Auxiliary handling machine (AHM)-handle equipment needed in fuel handling and serve as a maintenance cask for some components

44 2) RCB miscellaneous storage facilities and RSB plug storage facilities — store port plugs when removed from the reactor, EVST, or FHC

3) Floor valves — seal port openings in the reactor, EVST, and FHC when port plugs are removed

4) Fuel transfer port cooling inserts — provide cooling for removal of decay heat if an irradiated fuel assembly becomes immobilized in a port during transfer.

Figures 9.1-1 and 9.1-2 show the arrangement of major fuel handling and storage equipment in the Reactor Containment Building (RCB) and Reactor Service Building (RSB). Figure 9.1-1 also shows the flow path of new and spent core assemblies. New core assemblies enter the RSB by truck, are unloaded from their shipping containers, inspected and transferred by the EVTM to the sodium filled ex-vessel storage tank for preheating and storage. After reactor shutdown for refueling, the ex-vessel transfer machine transfers new core assemblies from the EVST to the reactor, and spent core assem-
44 blies from the reactor to the EVST, on a one-for-one basis. The in-vessel transfer machine, in conjunction with the reactor rotating plugs, transfers
44 spent core assemblies from the core to the in-vessel transfer positions, and

44 new core assemblies from the in-vessel transfer positions to the core, on a
one-for-one basis. After completion of refueling and suitable period for
44 decay, the spent core assemblies are removed from the EVST to the fuel handling cell, examined if desired, and loaded into the spent fuel shipping cask for shipment to a fuel reprocessor.

The design bases, description and safety evaluation of fuel handling and storage equipment and facilities are given in 9.1.1 (new fuel storage), 9.1.2 (spent fuel storage), 9.1.3 (spent fuel cooling) and 9.1.4 (fuel handling equipment). Section 9.1.4 also describes the movement of fuel assemblies through the plant in more detail.

44 All design bases were developed to conform with the CRBRP De-
sign Criteria 53, 54, and 55 described in Section 3.1, and with the intent
44 of Regulatory Guide 1.13. The Reactor Refueling System is designed 36
to reduce the probability of operator mishandling or of maloperations that
could cause fuel damage and potential fission product release, while limit-
ing the in-plant buildup of airborne radioactivity during normal plant oper- 36
ations, so that the exposure to plant operators is minimized. In addition,
specific attention was given in the selection of equipment design bases to
ensure that no single active component failure can result in a loss-of-
safety function. Additional margin is provided for protecting the public
during refueling and fuel handling operations by the low leakage design of
the RSB, and maintaining the RSB and RCB when the refueling hatch is open at
a minimum of 1/4" W.G. negative pressure with respect to the outside atmos-
phere, with the exhaust discharged through a high efficiency filter train
capable of efficiencies as high as 99% adsorbent efficiency and a 99.9% com-
bined HEPA filter efficiency. For design basis events, the efficiencies of 36
Section 6.2.6.2 are applied.

9.1.1 New Fuel Storage

59 New fuel is stored within the Reactor Service Building (RSB) in the ex-vessel storage tank (EVST), containing sodium. This storage facility contains both new and spent fuel. In addition to the EVST, new fuel assemblies are also temporarily retained in shipping containers on the RSB operating floor and below the operating floor in two new fuel unloading stations. Each of these two new fuel unloading stations can temporarily contain one shipping container with a new fuel assembly until unloaded. A conceptual drawing is provided in Figure 9.1-3.

This section will cover only new fuel storage in the unloading stations and in the shipping containers. Storage of new fuel in the EVST will be covered in Section 9.1.2 under spent fuel storage.

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The new fuel shipping containers (NFSC) will be licensed separately; therefore, the following paragraphs are intended to assure that operations performed within the RSB do not exceed those for which the container is designed.

59

9.1.1.1 Design Basis

The conditions to which the new fuel shipping containers are exposed within the RSB will not exceed those of the applicable regulations to which containers must be designed.

9.1.1.2 Design Description

Each container can only hold 1 core assembly. The container measures about 20 in. in diameter; a conceptual design drawing is shown in
59 Figure 9.1-5. The containers are removed from the truck one at a time by the RSB crane and lowered into a new fuel unloading station. The shipping cover is not removed from the container until it is installed in the unloading station.

Two new fuel unloading stations are located between the EVTM gantry-trolley rails in the RSB. Each station supports and positions, in a vertical orientation, a new fuel shipping container to permit loading or unloading of a new core assembly by the EVTM. The unloading station unloads the fuel assembly from the end in a vertical orientation.

59

Each station consists of a pit in the operating floor with a structural steel support for the shipping container. An adaptor and valve assembly is mounted on the top of the unloading station to provide an interface between the unloading station and the EVTM. The adaptor mates with and seals to the EVTM closure valve. Argon gas to inert and purge the shipping containers is provided from a nearby floor service station.

9.1.1.3 Safety Evaluation

44 The outer diameter of the shipping container limits the center-to-center separation, and keeps the NFSC array subcritical, even if a large number of containers were as close together as physically possible and submerged in water. Each fuel assembly is removed from the shipping container by the EVTM only after the container is installed in the unloading
44 20 ing station.

9.1.2 Spent Fuel Storage

Spent fuel is stored within the Reactor Service Building (RSB) in
59 44 two areas; the ex-vessel storage tank (EVST) and the fuel handling cell (FHC).

The ex-vessel storage tank (EVST) provides safe and controlled storage under sodium for both new and spent fuel. New fuel is stored in the EVST under sodium, after an initial preheating in an argon gas environment in one of the EVST's 24 preheat positions. Temporary storage of spent fuel occurs
44 in a small transfer station in the fuel handling cell (FHC). The FHC is a sub-floor hot cell in which spent fuel assemblies are removed from Core Component Pots (CCP), inspected and measured, if desired, and transferred into the Spent Fuel Shipping Cask (SFSC).

This section covers the safety aspects of new and spent fuel stored in the EVST and spent fuel stored in the FHC. The SFSC itself will be
59 44 licensed separately. (Reference 2)

The design bases, description, and safety evaluation of the spent fuel storage will cover the following safety items: prevention of criticality, provision for adequate shielding protection against radioactivity release, prevention of mechanical damage, and fuel storage monitoring. Spent fuel decay heat removal is discussed in Section 9.1.3.

9.1.2.1 Safety Aspects of New and Spent Fuel Storage in the EVST

The EVST performs the functions of preparing new core assemblies for insertion into the reactor, and providing safe, controlled storage for irradiated and new core assemblies.

The EVST capabilities required to implement its functions are as follows:

1) Preheating of new fuel assemblies

2) Radiation shielding

3) Storing new and spent fuel assemblies under sodium

20 4) Cooling spent fuel assemblies (See Section 9.1.3)

5) Containing cover gas

6) Providing structural support and physical separation of fuel assemblies to maintain their subcriticality

26

The EVST is a sodium-filled storage facility with a two-tier
rotating turntable and a fixed head with fuel transfer ports (see
44 Figure 9.1-6). One EVST is provided, and is located in the RSB between
the EVTM gantry rails in a nitrogen-filled concrete vault. The major
EVST components are: (1) storage vessel, (2) guard tank, (3) closure
head assembly (4) rotating turntable, (5) support structure, and
(6) drive and controls.

The design of the EVST is similar to the FFTF intermediate
59 decay storage facility (IDS). The storage vessel is supported at its
upper flange, suspended into the surrounding guard tank. The guard tank is
bottom supported. The turntable is supported by a bearing and seal configuration
above the storage vessel flange. It holds storage tubes which provide approx-
imately 650 storage positions in two tiers. The entire turntable (except for
20 the approximately upper 11 ft. of flanged support) with the storage tubes is
59 under sodium.

9.1.2.1.1. Design Bases

59 Criticality of new and spent fuel assemblies stored in the
EVST is prevented by physical separation. The center-to-center spacing
between fuel assemblies in storage positions in combination with several
51 permanently installed neutron absorber assemblies is sufficient to
maintain the array, when fully loaded with new fuel of the highest
44 anticipated enrichment and immersed under sodium, in a subcritical con-
dition with the K_{eff} less than 0.95. The EVST design considers all
normal loadings in combination with the loads from a safe shutdown
earthquake (SSE) in maintaining the necessary physical separation.
The EVST head is designed to absorb the load of the heaviest piece of
equipment handled by the RSB bridge crane over the EVST: (a) for the
main hook, lowered at the maximum crane speed (5 fpm), and (b) for the
auxiliary hook, accidentally dropping from the maximum handling height
to which it is raised, onto the center of the striker plate without
44 affecting the integrity of the fuel separation lattice. The EVST is
20 located such that heavy equipment not belonging to the fuel handling
and storage system is not carried over it.

Shielding is provided in the EVST and in its containment vault for radiation protection to meet the radiation protection requirements specified in 10CFR20, to ensure that the integrated dose is below 125 mrem/quarter, and to meet the radiation zone criteria of Section 12.1.

The EVST is designed for containment of radioactive fission
gases. Radiation doses due to leakage and diffusion through seals and
penetration are limited to well below those of the guidelines of 10 CFR
100 when the entire fission gas inventory of one fuel assembly is
44 released into the EVST cover gas.

The EVST is so designed that movement of the rotating turntable will not occur while a core component pot is being withdrawn or inserted. This design condition prevents mechanical damage to the CCP or its contents.

Monitoring instrumentation will be provided for the EVST and its associated areas for conditions that might result in a loss of the capability to remove decay heat and to detect excessive radiation levels.

9.1.2.1.2 Design Description

The EVST is shown in Figure 9.1-6. The fuel assemblies are
separated from each other by the storage positions which are part of
20 the turntable. The storage positions within the EVST are cylindrical
59 tubes restrained and supported by a steel gridwork. Each
storage tube is held in place such that all fuel assemblies have a
center-to-center distance of 9 in. or more. Each storage tube holds
two core component pots containing either new or spent fuel assemblies
44 in a vertical array, i.e., one above the other in any order. A maxi-
mum of 650 new and spent fuel asssemblies can be stored in the EVST in
nine circular rows, although under typical refueling conditions, there
44 20 will be 162 fuel assemblies and fewer than 150 radial blanket assemblies,
all stored in the upper tier. Nine storage positions in various rows
of the upper tier, and the nine corresponding positions in the lower tier
59 contain B_4C-filled neutron absorber assemblies and are inaccessible
for fuel storage. The purpose of the absorber assemblies is to limit
51 the value of K_{eff} for the EVST to < 0.95

The closure head assembly consists of a 6.5 inch-thick steel
striker plate, a 12-inch thick steel closure head, and a multi-sheet
steel reflective thermal insulation. The closure head seals the upper
end of the storage vessel and provides containment for the EVST cover gas.
Striker plate and closure head are designed to support normal structural
44 loads as well as the accidental impact loads given in the design bases.

The EVST is contained in a 26 ft square by 57 ft deep nitrogen-
filled concrete vault located in the Reactor Service Building (RSB),
as shown in Figure 9.1-1. The EVST vessel is filled with sodium,
20 topped by argon cover gas. A guard tank, which precludes the loss of
sodium, surrounds the storage vessel. The design and construction of
the EVST is in accordance with applicable codes, standards, and speci-
fications listed in Section 3.8 for Seismic Category I structures.

The top of the EVST is located at the operating floor level
of the RSB, as shown in Figure 9.1-2. Sufficient axial shielding must
therefore be provided so that the radiation level at the top of the
EVST does not exceed 0.2 mrem/hr, (see plant radiation zone criteria
presented in Section 12.1). This shielding is provided by 22 in. of
steel in the closure head assembly and 87 in. of sodium above the fuel.
Nine fuel transfer ports penetrate the head, as shown in Figure 9.1-6.
Each fuel transfer port is provided with a shielded cooling sleeve
(see Section 9.1.4.7) which extends from the closure head to a point
between the thermal insulation and the sodium level. Lead shielded
20 collars around the ports are located in the space between the head and
striker plate. A floor valve adaptor is inserted into the fuel transfer

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

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

The fuel transfer port plugs in the EVST head have double seals with capability for convenient periodic leak testing. Large diameter seals are between the storage vessel and the closure head. The operating floor striker plate has a seal at its mating surface with the side wall vault lining. In general, all seals in the EVST are double with
20 capability for convenient leak testing.

59 The EVST is designed with sensors and interlocks to prevent
20 any unscheduled movement of the turntable while the EVTM is mounted on the EVST. The interlock allows the turntable to rotate only when the EVTM grapple is in the full up position. The EVST is designed to prevent
44 20 excessive relative motion between the head and turntable during an SSE.

Temperature instrumentation and sodium level sensing probes will monitor cooling capability. High EVST sodium outlet temperature, and high or low sodium levels will sound an alarm. Other monitors will be
20 provided in the EVST cooling system (see Section 9.1.3). Sodium leak detectors will monitor the space between the storage vessel and the guard tank. An argon cover gas activity monitor will be provided. An area monitor of the gamma scintillation type will measure the gamma radiation on the RSB operating floor above the EVST.

9.1.2.1.3 Safety Evaluation

The minimum center-to-center separation distance between storage
20 tubes and the 9 storage positions permanently filled with B_4C will keep the storage array subcritical even if the EVST were completely loaded
51 with new fuel assemblies of the highest reactivity. The B_4C neutron absorbers are designed such that they cannot be removed inadvertently,
59 i.e. cannot be removed with the normal refueling equipment. Based on the calculations reported below the K_{eff} of this array, either with sodium or void of sodium, will be less than 0.95, as required.
44 27

27

Calculations of the criticality of the EVST void of sodium were made using a two-dimensional, multigroup, diffusion method (DOT codes, see Appendix A). A k_{eff} value of 0.69 ± 0.04 was obtained. Virtually the same k_{eff} would be obtained if argon, nitrogen, or air were to replace the sodium. The RSB design and elevation of the operating floor prevent the entry of flood or rainwater into the building. No significant amounts of fluids having greater moderation than sodium are used in either the EVST or its sodium
44 cooling system.

The storage tubes are held in place at the upper end by a top grid plate which precludes insertion of any other new fuel assembly between storage tubes. Seismic forces will not change the separation distance between fuel
44 assemblies due to the restraints between storage tubes. The EVST closure head assembly has sufficient strength to absorb the heaviest loads carried above
20 it without changing the separation lattice of the storage tubes.

Adequate shielding for radiation protection on the operating floor is provided by the EVST design. Overall dose rates on the operating floor are less that 0.2 mrem/hr which permits routinely occupied area access
49 (see Section 12.1).

Transient dose rates due to high-powered spent fuel assemblies are less than 200 mrem/hr at the surface of the striker plate port. The inte-
49 grated doses are below the limits for unrestricted access (see also subsection 9.1.4).

Radiation shields near the EVST bearings and seals limit the dose to 125 mrem/quarter integrated over the time required to perform bearing or seal maintenance. Additional safety factors are provided by the fact that maintenance is not expected to be required more than once in three years. The streaming dose rates from gaps around fuel transfer port plugs are less than 2 mrem/hr, in accordance with the shielding criteria of 12.1.2.1. The
44 vault walls and floor provide radial and bottom shielding, as described in
49 20 Section 12.1.

The EVST has adequate seals to prevent excessive radioactive emissions into other areas of the RSB. Section 15.5.2.4 analyzes a limiting case and shows it to be acceptable. Radioactivity released from the EVST does not exceed the limits specified in Sections 12.1.1 and 12.1.2. The
49 RSB has radioactivity monitors above the EVST to monitor radioactivity levels to
59 detect accidental release and to sound alarms.

The design prevents movement of the turntable sufficient to cause
20 failure of the CCP or to damage a new or spent fuel assembly due to either a seismic event or inadvertent rotation.

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59 PAGES 9.1-10 through 9.1-19

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9.1.3 Spent Fuel Cooling and Cleanup System

There are two locations in the plant where spent fuel is stored.
They are the Ex-Vessel Storage Tank (EVST) and Fuel Handling Cell (FHC),
44 both of which are located in the Reactor Service Building.

The description of the spent fuel cooling and cleanup systems associated with these spent fuel storage systems is presented in this section.

9.1.3.1 Ex-Vessel Storage Tank Cooling and Cleanup System

Cooling and purification of the sodium within the Ex-Vessel Storage Tank is accomplished by the EVS Processing System, shown in Figure 9.3-3. This system is a subsystem within the Auxiliary Liquid Metal System, the balance of which is described in Section 9.3. The design bases, design description, safety evaluation and inspection and test requirements discussed in this section are applicable to the sodium cooling aspects of the EVST and to the EVS Processing System.

9.1.3.1.1 Design Bases

Sufficient cooling capacity is provided in the EVST to handle
46 the unlikely event of a complete core unloading occurring with two annual
loads of spent fuel assemblies already in the EVST.

The EVST is designed to maintain the sodium coolant level at a
height which permits continued cooling of the spent fuel assemblies under
normal conditions and in the extremely unlikely event of a primary vessel
or cooling system leak or rupture. The system provides three independent
means of heat removal, each of which can provide the required cooling.
The system is designed such that no single failure, or operator error,
59 44 can result in loss of two heat removal paths.

Either of the EVS Sodium Processing System's two normal forced
cooling circuits provide the capability to remove 1800 kW of heat while
46 maintaining an EVST exit sodium temperature of approximately 510°F. The
20 third (backup) cooling circuit provides the capability to remove 1800 kW
while maintaining an exit temperature less than 775°F.

In the extremely unlikely event that both normal EVST heat re-
moval circuits are unavailable due to a combination of an initiating
event (active or passive failure) followed by an active failure, heat
44 will be removed by a third (backup) natural convection heat removal
circuit.

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

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

9.1.3.1.2 Design Description

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

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

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

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

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

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

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

The entire EVS processing system includes the following
44 components:

EVST Sodium Pumps (2)

EVST Sodium Coolers (2)

44 EVST Backup Sodium Cooler (1)

EVST NaK Pumps (2)

EVST NaK Airblast Heat Exchangers (2)

EVST NaK Natural Draft Heat Exchanger (1)

EVST NaK Expansion Tanks (3)

44 EVST NaK Diffusion Cold Traps (3)

26 DHRS NaK Expansion Tank

EVST NaK Storage Vessel

44 EVS Sodium Cold Trap

Interconnecting Piping and Valves

All pumps, both for sodium and NaK service, are electromagnetic pumps. Heat exchangers are all-welded units. All pressurized fluid containment boundary components are of 300 series stainless steel. The normal cooling circuits plus the sodium loop of the backup cooling circuit are designed and fabricated to the requirements of Section III of the ASME Boiler and Pressure Vessel Code for Class 2 components. The backup cooling circuit EVST NaK expansion tank, the natural draft heat exchanger, and the NaK piping are designed and fabricated to the requirements of Section III of the ASME Boiler and Pressure Vessel Code for Class 3 components. All components are classified Seismic Category I and are housed in hardened
44 structures.

Each of the cooling circuits (the sodium and its associated NaK loop) are physically separated to preclude common failure due to potential accidents such as liquid metal leakage or fire. In addition, the sodium loop within each circuit is enclosed in an inerted cell to avoid possibility of a significant radioactive sodium fire. Shielding is provided to permit access for inspection or maintenance of any loop while the other loops remain operational. The cold trap is shielded separately from all three loops (see Section 12.1).

44 a) Normal System Operation for Cooling of the EVST – The principal function of the EVS Processing System is to provide cooling for the EVST. Since its operation to provide reactor core heat removal is expected to occur, if at all, only once during plant life, EVST cooling is the normal mode of system operation. During normal EVST cooling operation
44 sodium is circulated at 400 gpm through one of the two normal sodium loops from the EVST to the EVST sodium pump, through the NaK-cooled EVST sodium cooler, and back to the EVST
44 (see Figure 9.1-10). A bypass flow of 60 gpm is circulated through the EVS sodium cold trap for purification, and 3 gpm through the plugging temperature indicator (provided by the Impurity Monitoring and Analysis System, described in Section 9.8).

At the design flow rate of 400 gpm, the normal loop is
designed to remove 1800 kW with EVST sodium inlet and
46 20 outlet temperature of 400 and ~510^{o}F, respectively. The
EVS sodium cold trap can be valved into either of the normal
loops to provide essentially continuous EVST sodium purifica-
tion. On the basis of anticipated in-leakage, the cold trap
will maintain the oxygen content below 5 ppm in the EVST.
44

Normally, heat from EVST sodium is transferred in the EVST sodium
cooler to the associated NaK loop. NaK is pumped through the shell-
46 59 side of the sodium cooler and then to the airblast heat exchanger and
20 returned to the sodium cooler. With an EVST heat load of 1800
kW, NaK is circulated at a flow rate of approximately 400 gpm.
As the EVST heat load decreases with time, the ABHX flow rate
is decreased; air flow is controlled to maintain the EVST
sodium inlet temperature at 400^{o}F. The NaK cold leg temper-
ature, exiting from the airblast, is maintained below 400^{o}F by
controlling either the airblast fan speed or the setting of
air inlet dampers. NaK volumetric changes from temperature
44 variation are accommodated in the NaK expansion tank connected
59 to the high point of each loop. Oxide level in each NaK loop is
minimized by the diffusion cold trap.

On a periodic basis, EVST cooling is manually transferred
from normal Circuit No. 1 to Circuit No. 2 in order to equalize
operating time. The circuit not in use (both sodium and NaK
loops) is maintained full and at operating temperature, ~400^{o}F
44 to be ready for immediate use, and is called the standby circuit.

During normal EVST cooling operation, the "crossover piping"
shown in Figure 9.1-10 is isolated by closed valves at each
of the EVST loops.
59

b) <u>Off-Normal System Operation for Cooling of the EVST</u> - In the
<u>extremely unlikely event of loss of both normal cooling</u> cir-
cuits, the dampers on the natural-draft heat exchanger will
47 be manually opened to initiate natural draft air flow. This
will induce NaK flow through the tubes of the natural-draft
heat exchanger and the shell of the backup sodium cooler.
This in turn induces sodium flow from the EVST through the
tubes of the backup sodium cooler and back to the EVST.
44

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The third (backup) cooling circuit is designed to remove
1800 KW of heat while maintaining sodium temperatures within
the EVST below 775^{o}F. The damper position on the natural-
draft heat exchanger is adjusted to control EVST sodium tem-
perature under various heat loads. NaK volumetric changes
from temperature variations are accommodated in the NaK ex-
pansion tank connected to the high point in the NaK loop.
Oxide level in the NaK loop is minimized by the diffusion
cold trap.

The backup circuit is maintained in a preheated condition
and is ready for immediate operation. A small sodium and
NaK flow is induced by slightly opening the dampers of the
natural-draft heat exchanger and removing a minimum amount
of heat during normal EVST cooling by circuits 1 or 2. Trace
heaters are also provided for the sodium piping and components.

44 59 26

44 c) Operation During Reactor Decay Heat Removal — Reactor decay
heat removal is accomplished by using the combined heat
removal capability of both of the normal NaK loops in the
44 EVS Processing System. In conjunction with the primary
sodium overflow heat exchanger, each of the NaK loops cir-
culate approximately 400 gpm through each airblast heat ex-
changer. The circuits are interconnected during this oper-
ating mode to provide a total NaK flow of 800 gpm which is
routed through the shell-side of the overflow heat exchanger
in the Primary Sodium Storage and Processing System (de-
scribed in Section 9.3). When the DHRS is initiated one-
20 half hour after reactor shutdown, the NaK exits from the overflow
overflow heat exchanger at a maximum temperature of approxi-
mately 1000^{o}F. At this temperature, the NaK airblast heat
exchangers have a combined heat removal capacity of approxi-
mately 11-1/2 MW, which is sufficient to remove decay heat
20 from both the reactor and the EVST.

44

During the DHRS mode of operation, one of the normal Na/NaK
loops remains in use for EVST cooling. The NaK in this loop
is circulated (400 gpm) from the airblast heat exchanger
through the EVST sodium cooler prior to its flow to the over-
flow heat exchanger. The EVST, in effect, is cooled in
series with the overflow heat exchanger. In this flow pat-
tern, the EVST is located in the cold leg of the loop in
order to minimize temperature rise of the EVST sodium. The
sodium and NaK flow path in this mode of operation is shown
schematically in Figure 9.1-11. Switchover from normal
cooling (EVST) only to reactor decay heat removal (DHRS)
is done remotely from the control room. Switchover is accom-
plished by opening the isolation valves at the connections

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

9.1.3.1.3 Safety Evaluation

The EVST cooling capability can be provided by either of two iden-
46 tical, forced convection cooling circuits, each of which can remove 1800 kW
44 20 while maintaining a maximum EVST sodium outlet temperature of ~510°F.

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

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

The peak fuel cladding temperature is approximately 100°F greater than the sodium outlet temperature shown in Table 9.1-1. Hence, no damage to the stored fuel assemblies will occur.

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

Each of the th--e sodium cooling loops is designed against the possibility of common-- failure. Two pump suction lines are provided within the EVST for no sodium circuit No. 2. The open end elevation of each is different, or , one low. Each of the two lines is separately valved externa the EVST. After the initial fill of the loop, the isolation valve i the low suction line is locked closed and remains closed (except for periodic testing) throughout the plant life. This low suction line is used only in the event of a major loop or vessel rupture. One pump suction line is provided within the EVST for normal cooling circuit No. 1. The open end elevation of this line is between those for circuit No. 2. This line is valved externally to the EVST, and is called a "high" pump suction line. During normal system operation, one of the normal cooling loops is operated using the "high" pump suction line. The suction line(s) in the standby normal loops are closed. In the event
44 of a major failure (rupture) of the operating normal sodium cooling loop, the isolation valve in the pump suction line is closed by operator action from the control room, signalled by concurrent alarms, indicating low level

in the EVST and a sodium leak within the cooling loop cell. If the isola-
tion valve should not be closed the EVST sodium level could only be siphoned
44 to the (high) pump suction outlet within the tank. Siphoning from the re-
turn line is prevented by an antisiphon vent in this line within the EVST.
If a failure of normal cooling loop occurs, as described previously, the
44 standby normal cooling circuit can be immediately activated, by valving in
its lower pump suction and increasing pump flow to the design rate of
44 400 gpm.

In the extremely unlikely event that the second normal loop cannot
be activated after the first loop has experienced a failure, the third
(backup) circuit will be brought into operation. One suction line is pro-
vided within the EVST for the backup cooling circuit. The open end eleva-
tion of this suction line is below the lower suction line of normal cooling
circuit No. 2. Flow back to the EVST is through the fill/drain line.
44 Siphoning from this return line is prevented because the entire backup
loop is elevated above the sodium level in the EVST.

Failure of any component, in any of the sodium or NaK loops,
can cause loss of only the circuit in which it is located. The normal
44 standby or backup cooling circuit can then be put into operation within
minutes to provide essentially continuous cooling of the EVST sodium. The
potential radiological consequences of an extremely unlikely release of
44 59 EVST sodium to an inerted cell is described in Section 15.

44 All components of the normal sodium and NaK loops which require
electrical power are on the Class IE power system, to ensure continuous
EVST cooling and reactor decay heat removal. In the event of complete loss
of external power to the plant, power to both of the normal cooling circuits
44 is provided by the plant diesels. Immediate activation of the diesel-
powered supply is not necessary for the EVST sodium pumps since the sodium
volume within the EVST provides a heat sink to minimize sodium temperature
46 rise during loss of circulation. Sodium circulation can be lost for approxi-
44 mately 2 hours before the maximum sodium temperature in the upper portion of
the EVST reaches 600°F. Activation of the emergency power supply to the
59 NaK pumps and airblast fans is required within ½ hour, however, to ensure the
availability of DHRS for reactor decay heat removal.

44 The only "active" component in the backup loop is the damper on the
46 26 natural draft heat exchanger. It is operated manually and, therefore, does
not require connection to the emergency power system

44

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

Isolation valves are provided in the suction and return lines to
44 each of the normal sodium loops, to permit loop isolation for inspection or
maintenance. Prior to personnel access to a cell, the isolation valves will
be closed and, if necessary (depending on Na activity level), the loop may
26 be drained. The loop isolation valves and loop high-point vents are loca-
ted higher than the sodium level in the EVST. Thus, once the loop is vented
44 and drained, siphoning of the EVST cannot occur even if the isolation valves
are accidentally opened during a maintenance operation.

The high-point vent in the sodium loop of the backup cooling circuit allows sodium to drain back to the EVST for inspection and maintenance. Since the entire backup loop is located higher than the sodium in the EVST,
44 siphoning cannot occur.

Instrumentation is provided to monitor and alarm off-normal con-
44 ditions in the sodium and NaK systems. The off-normal conditions include high temperature, low flow, and external leak detection. The operating pressure of the NaK system is maintained higher than that of the sodium system. Leakage of NaK-to-sodium is monitored and alarmed by abnormal level indication in the NaK system expansion tank, in conjunction with the level in the EVST.

9.1.3.1.4 Inspection and Test Requirements

Leak checks will be made on all of the systems prior to filling
59 with Na or NaK. Prior to spent fuel loadings, the system will be operationally tested to determine that the system will perform within design limits.

The equipment containing Na will be placed in inert atmosphere
59 cells that will be accessible for inspection. The three independent cooling loop components are separated by shield walls so that inspection and maintenance can be performed with the other loops remaining operational.
44

An in-service inspection device will be used to periodically check the structural integrity of the EVST vessel. Space for such a device is provided by allowing sufficient clearance between the storage vessel and guard vessel.
44 59

The two NaK airblast heat exchangers and NaK Natural Draft Heat
44 Exchanger will be located in air atmosphere cells and will be available for periodic visual inspection.

9.1.3.1.5 Instrumentation Requirements

Instrumentation and controls (I&C) are provided for operation, performance evaluation, and diagnosis of the EVS Sodium Processing System. These functions are required for off-normal, as well as for the full range 1

of normal operation. Details of the I&C for the subsystem are shown in the
piping and instrumentation diagram, Figure 9.3-3. DHRS instrumentation is
discussed in Section 5.6.2.1.6. The following I&C is required to ensure
44 safe operation of, and to prevent damage to, the EVS Sodium 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 are
provided with winding temperature measurements and winding coolant low flow
46 indication. These measurements are alarmed for off-normal conditions, and
interlocked to automatically shutdown the pump to prevent damaging it.

The EVST and the NaK expansion and storage tanks are provided with
44 level measurements, which are alarmed for abnormal low and/or high level.
59 This information, in conjunction with leak detection data, is utilized to
diagnose external liquid metals leaks. The operator is alerted to NaK to
59 46 sodium leakage by NaK expansion tank high-low level alarms. A differential
pressure sensor and flow meter 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. Accessories (solenoids, pressure gauges, 1
I/P converters, etc.) for remotely operated valves installed in inerted
54 cells are installed off of the valves in adjacent accessible air cells.
The stem portion of the sodium valve is monitored and alarmed for low
temperature, to ensure free operation and protect the valve sodium seal from
46 damage. 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. 1

9.1.3.2 <u>Fuel Handling Cell Spent Fuel Storage Cooling System</u>

Decay heat generated by spent fuel assemblies stored in the spent
fuel transfer station within the FHC is removed by natural convection heat
44 transfer to the FHC argon atmosphere.

When fuel is transferred using the crane, a gas cooling grapple
is used to remove decay heat from the assembly. The design basis, description,
evaluation, and inspection and test requirements for these systems
are discussed in the following sections.

9.1.3.2.1 <u>Design Bases</u>

Sufficient cooling capacity will be provided in the spent fuel
20 transfer station to remove a maximum heat load of 24 kW from a total of
three fuel assemblies in sodium-filled core component pots. During normal
44 operation, a maximum of two fuel assemblies will be present in the FHC; the

third position is used for transfer and will normally be empty or contain an empty CCP. Normally, the decay power of fuel assemblies handled in FHC will be limited to <8 kW each. However, under unusual conditions, it may be desired to examine a short-cooled assembly, i.e., with a decay power greater than 8 kW, but less than 15 kW. Under this condition, no more than a single fuel assembly shall be permitted in the FHC. The transfer station will be designed to cool a single fuel assembly of up to 15 kW decay heat without exceeding the normal cladding temperature limit.

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

9.1.3.2.2 Design Description

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

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

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

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

44

Cooling is provided by a Recirculating Gas System consisting of a
fan, gas heat exchanger and a piped distribution system. The heat exchanger
removes heat from the argon gas and rejects it to the recirculating Dowtherm J
which rejects it to the Chilled Water System (Normal or Emergency as applicable) which in turn rejects the heat to the ambient air through the Emergency
Cooling Tower in the emergency mode and through the Normal Cooling Tower in
the Normal mode. The Argon Gas Cooling System operates during normal plant
operation, accident conditions, and periods of normal electrical power
failure. The Low Pressure Argon Gas Distribution System including the shell
44 of the cooler is designed to ANSI B31.1, and Section VIII of the ASME

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9.1.4 Fuel Handling System

The following subsections discuss the refueling procedure and the safety aspects of fuel handling equipment not discussed in the preceding sections of 9.1. The two major machines discussed are the ex-vessel transfer machine (EVTM), and the in-vessel transfer machine (IVTM). The design bases of fuel handling equipment cover the following specific safety aspects: prevention of criticality, sufficient cooling, protection against radioactivity release, prevention of mechanical damage, and provisions for adequate shielding.

9.1.4.1 Refueling Procedure

The sequence for handling fuel begins with receipt of new fuel in
the hardened portion of the RSB by truck. The truck to be used for shipping
new fuel assemblies will be a Safe Secure Trailer. The fuel assemblies
are transported in shielded New Fuel Shipping Containers (NFSC), with six
49 containers per shipment. Each NFSC contains one new fuel assembly (see
Figure 9.1-5) and is designed to withstand the hypothetical accident of 10
CFR71, characterized by a 30-ft drop onto a flat, unyielding surface.
(When handled by the RSB bridge crane, a drop of 30 ft. is not possible
since the maximum crane hook height above the RSB operating floor is 41
ft. and an NFSC is hooked on the end and is about 17 ft. long.) Shielding
59 in the container is provided for the NFSC to accommodate the future use of
49 LWR recycled fuel plutonium, which will be radioactive. The RSB crane is
20 used to lift one NFSC at a time from the trailer and place it in a laydown area.

The new fuel shipping containers are checked for damage.
One shipping container is placed in one of the two new fuel unloading stations
by the RSB crane. The stations are located between the EVST and FHC and are
within the coverage of the EVTM. A check is made to verify that the correct core
59 assemblies have been received.

49 The shielded new core assembly inspection equipment is installed
over the unloading station. The fuel assembly is lifted from the container, inspected, and returned to the container.

The inspection consists of visual and mechanical identification, checking for shipping damage, and other visual and dimensional checks. While these operations are taking place, a second shipping container may be placed in the second unloading station to expedite the inspection operations. The inspection equipment could then be moved to the second station to complete inspection of the second assembly. After both assemblies are inspected, the inspection equipment is moved out of the way.

The atmosphere of the shipping container is changed from air to
argon to be compatible with the EVTM and EVST. The EVTM is mated with the
shipping container through an adaptor assembly. The EVTM removes the new
44 fuel assembly and transfers it to the EVST. The fuel assembly is lowered

into one of the argon filled preheat stations in the EVST. In the pre-
heat station the new fuel assembly is slowly heated by the hot argon to
14 approximately the same temperature as the sodium in the EVST storage
vessel. After preheating, the assembly is transferred to a sodium-filled
core component pot in one of the storage positions in the EVST. This
transfer is accomplished by means of the EVTM. All new core assemblies
44 are thus transferred to the EVST prior to reactor shutdown for refueling.
All operations up to this point are performed while the reactor is
operating and the equipment hatch between the RCB and RSB is closed. A
minimum 1/4" W.G. negative pressure is maintained in the normal atmos-
pheric areas of the RSB during the fuel handling operation. During RSB 36
open hatch refueling operations, the outside air dampers for the RSB HVAC
is manually closed and reopened when the 1/4" WG negative pressure in the
RSB is achieved.

Refueling, operations of the reactor begin with reactor shutdown
and cooldown to refueling temperatures. The control rod drive lines (CRDL)
are disconnected from the control assemblies which are fully inserted
in the core. The drive lines and the upper internals are then raised to
clear the parting plane to allow the reactor head plugs to rotate. Concurrently,
the reactor cover gas is purged and purified to reduce radioactivity levels
in the gas to a very low level. The equipment hatch between the RCB and RSB
is opened while these operations are in progress. In order to provide a
minimum 1/4" W.G. negative pressure in the RCB and RSB after the containment
vessel refueling hatch is opened, the RSB HVAC system fresh air inlet
damper is closed and the RCB supply and exhaust fans are stopped. Once
the required negative pressure for the RSB and RCB with open hatch is 36
achieved, the RSB fresh air intake supply will be opened and the RCB supply
59 fan is started to supply the required ventilation rate.

20 Two adapters and floor valves are installed on the reactor head.
Then the fuel transfer and in-vessel transfer machine (IVTM) port plugs are
removed using the EVTM and the auxiliary handling machine (AHM) respectively.

The AHM then installs the in-vessel section of the IVTM into the
IVTM port in the small rotating plug (SRP) of the reactor head. Following
this, the IVTM adapter and floor valve are removed and the drive section of
the IVTM is installed with the RCB polar crane and connected to the in-vessel
section. The reactor is now ready for refueling.

The EVTM and the IVTM work in conjunction during refueling to
44 exchange spent core assemblies in the reactor with new core assemblies stored
in the EVTS. Two operations occur simultaneously - the EVTM removes a new

core assembly in a sodium filled core component pot (CCP) from the ex-vessel
44 storage tank, while the IVTM removes a spent core assembly from the core and
deposits it in a CCP located in a transfer position outside the core, but
still within the reactor vessel. The EVTM then places the CCP containing the
44 new core assembly in the second transfer position inside the vessel, the
rotating guide tube rotates to the first transfer position, and the EVTM
44 removes the CCP containing the spent core assembly from the first transfer
position. During this time, while the EVTM is at the reactor, the IVTM and
rotating plugs must remain stationary. The EVTM then moves away from the
44 reactor to deposit the CCP with the spent core assembly in the EVST, while
the IVTM, operating in conjunction with the rotating plugs, installs the new
44 core assembly in the open core lattice position. These operations are
repeated until the refueling is completed.

51 20

Spent blanket and control assemblies are exchanged first to increase
59 51 44 core negative reactivity. Then spent fuel assemblies are exchanged in inverse
order of decay power. As a consequence of this refueling sequence, time is
permitted for decay, thus reducing the decay heat which must be removed by the
EVTM during fuel handling. This sequence reduces the maximum spent fuel
assembly decay power handled by the fuel handling equipment to ~15 kw compared
to the 20 kw capability.

59 51 44

After refueling is completed, the IVTM is removed from the reactor and stored in the RCB. The fuel transfer and IVTM port plugs are reinstalled in the reactor head and both floor valves and adapters removed to their respective storage areas in the RCB. The reactor upper internals are lowered, and CRDL's are reconnected to the control assemblies. Finally, the hatch between the RCB and RSB is closed and leak checked, the RCB and RSB HVAC systems are returned to the normal operating mode, and the reactor is ready for startup.

After spent fuel has decayed for ~100 days in the EVST, it may be loaded into the spent fuel shipping cask. Control, radial shield, and some low-power blanket assemblies can be shipped offsite before the 100-day

59 cooling period, but fuel and high-power blanket assemblies are held until
20 they decay to less than 6 kw.

44 Spent assemblies to be loaded into the spent fuel shipping cask are transferred in sodium-filled CCP's from the EVST to the fuel handling cell by the ex-vessel transfer machine. In the FHC, they are stored temporarily in a three-position spent fuel transfer station. The assem-
44 blies are removed, one at a time, from the transfer station by a gas cooling grapple, the exterior is dimensionally and visually examined if desired, and residual sodium is drained prior to loading.

The spent fuel shipping cask (SFSC) is brought onsite by a special railroad car. The cask is removed from the railroad car, lowered down a shaft onto a transport dolly by the Reactor Service Building crane, and the outer containment shipping cover removed. The dolly moves the cask under the fuel handling cell floor, where it is sealed to the bottom of the cell. An access plug in the floor of the cell is removed by an in-cell crane, and spent
59 fuel assemblies are loaded into the cask. The cask is then decoupled from the FHC, the shipping cover is reinstalled, the cask is removed from the FHC shaft, loaded onto the rail car and checked for radioactive contamination prior to shipment. The cask is then shipped to the reprocessor.

The reprocessed fuel material is provided to the fuel fabricator, who eventually returns new fuel assemblies to the CRBRP, thus completing the fuel cycle.

44 51 There are 385 core special assemblies provided in the CRBRP. They are used for:

1) Initial reactor core loading,

2) Reactor core unloading,

44 3) Lower inlet module replacement (see PSAR Section 4.2.2.1.1.2), and

4) Removing jammed fuel assemblies (see PSAR Section 9.1.4.4.3).

44 Prior to the first reactor startup, the core special assemblies are
59 replaced on a one-for-one basis by new control, blanket, and fuel assemblies.
44 If the core is unloaded, the core special assemblies are inserted for spent core assemblies on a one-for-one basis. During initial core loading the new
44 and clean core special assemblies can be installed using the RCB polar crane since the reactor core is open to the RCB air atmosphere and does not contain sodium at this time. At any other time, they must be handled by the reactor refueling system equipment. 25
44

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After removal from the reactor core, the sodium wetted core special assemblies are first brought to the EVST, and are later transferred into the
44 51 FHC. 162 of these core special assemblies simulate fuel assemblies in the core and have full-flow filters. Some of these assemblies are partially disassembled in the FHC and made ready for sodium removal performed in the large
44 59 component cleaning vessel. All other core special assemblies are only inspected.

44 Whenever core special assemblies are handled by refueling equipment, they are accounted for using the same inventory control system as "real" core assemblies. Before entering and after leaving the reactor core lattice they are electromechanically identified by the IVTM using identifica-
44 tion notches (see 9.1.4.4.2). In the FHC, core assemblies are identified and differentiated both visually and electromechanically. The core special
44 assemblies leave the FHC in a poly-film wrapped transfer rack. The outer surface of this poly-film wrap is checked for radioactivity immediately after
44 59 sealing and leaving the FHC port. The core special assemblies are transferred from the FHC to the Large Component Cleaning Vessel (LCCV) located in the RCB
44 for sodium removal. Cleaned core special assemblies are packaged in polyethylene bags, loaded into holding transfer racks, and transferred to a
59 storage area.

44 The physical difference of identification marks between special and real core assemblies, the positive identification of core assemblies at two
44 locations, and the radiological monitoring of core special assemblies before cleaning them are regarded as sufficient safeguards to insure that no real
44 fuel assemblies are mistaken as special ones, and stored in a storage facility not designed to receive them. 25

44 9.1.4.2 Deleted

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

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

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

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

A perspective cutaway view of the EVTM is shown in Figure 9.1-14. The EVTM
is similar in concept to the FFTF Closed Loop Ex-Vessel Machine (CLEM), but
44 is considerably shorter. The EVTM is 35 ft high and weighs 240 tons, includ-
ing the gantry. The span of the gantry is 30 ft between the rails.

The EVTM is modular in construction to limit the size and weight of
the subassemblies for better maintainability and fabricability. Major subas-
semblies are: grapple drive system, several cask body modules which provide
structural support and shielding, service platforms, a view port assembly to
permit visual inspection of CCP's, port plugs, or drip pans, a drip pan
assembly to collect sodium drippage, an extender which raises and lowers the
closure valve, a closure valve which mates with the floor valve and provides
44 the gas tight seal, the cold wall assembly and electrical, instrumentation,
and control equipment.

Irradiated core assemblies are handled by the EVTM only in sodium-
filled CCP's. New core assemblies are handled both in CCP's (between the
reactor and the EVST) and bare (between the new fuel unloading stations and
the preheat stations in the EVST and between the preheat stations and the
44 storage stations in the EVST). Different grapples are required for the bare
assembly and CCP. Grapple changing is fairly straightforward, and is done in
the FHC using the designs and techniques developed for the FFTF CLEM. The
CLEM, in turn, is similar to fuel handling machines used successfully at
Hallam Nuclear Power Facility and at Fermi-1 (the replacement machine).

The EVTM is connected at each fuel transfer port by lowering the
closure valve and mating with the floor valve. The sealing at this inter-
face is accomplished by double seals on the top of the floor valve. The
space between the seals is then pressurized with argon buffer gas. The air
trapped within the sealed interface space between the valves is purged by
pressure-vacuum cycling. First, vacuum is drawn by the plant radioactive
vent system, then pressure is applied from the argon supply system. This
cycling is repeated a sufficient number of times to reduce the oxygen con-
44 centration to an acceptable level. The closure valve and floor valve are
then opened, and the drip pan pot is rotated out of the way. The grapple
is then driven down to pick up a CCP or a bare core assembly.

59 The grapple is lowered into the handling socket of the CCP by
the chain drive system. The grapple fingers are actuated, engaging the lip
on the handling socket. The CCP containing the new or irradiated core assem-
44 bly is then raised into the EVTM cold-wall assembly. After the CCP is fully
raised, the drip pan is closed to prevent sodium drippage on the closure
valve or floor valve. The closure valve and floor valve are then shut, and
44 radioactive gas trapped between the valve interface seals is purged by the
vacuum-pressure cycling described previously. The extender is then retracted,
permitting the EVTM to move to another location. The closure valve and drip
pan module are based on the same design and operating concept as the floor
20 valve described in Section 9.1.4.6.

The safety aspects of the EVTM that will be covered in this section
are: radiation shielding, prevention of radioactive releases, cooling of
spent fuel, and prevention of mechanical damage.

9.1.4.3.1 Design Basis

Adequate shielding for radiation protection is provided in the design of the EVTM to meet the radiation protection requirements of 10 CFR 20.

59

The activity released from a damaged or leaking spent fuel assembly while in the EVTM is contained in the EVTM by proper sealing or welding of penetrations and openings. Radioactive leakage and diffusion through seals
44 are well below the limits specified in 10 CFR 100.

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

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

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

9.1.4.3.2 Design Description

Axial and radial shielding is provided in the EVTM to limit the
49 44 dose rate to less than the criteria given in Sections 12.1.1 and 12.1.2 at the surface of the cask body. Shielding is provided over the entire length of the EVTM and is graduated in thickness, being thinnest at the upper end, where the radiation source from the spend fuel assembly being handled is least. Approximately 11 in. of lead shielding is provided at
44 the lower end of the EVTM.

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

44 leak checked. The space between the closure valve and the floor valve is
purged by the plant argon supply and vent system, through hose connections
made after the EVTM has been mated to a refueling station. All elastomer
seals in the fission gas containing boundary are double, with dynamic and
inflatable seals having pressurized buffer gas between seals to detect
44 leakage.

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

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

44 The EVTM CCP grapple has an interlocking finger design such that,
with the CCP engaged with and supported by the grapple fingers, the fingers
cannot be retracted even if the entire weight is supported by the finger-
actuating chain. Redundant support chains are utilized to insure component
safety in the event of a single chain failure. The EVTM is transported and
positioned by a trolley, traveling on a gantry. The gantry, in turn, travels
on rail tracks (see Figure 9.1-13) secured to the RSB/RCB floor. Trolley and
gantry wheel truck structures incorporate anti-lift-off and overturning
restraints. Trolley and gantry rails are equipped with rail stops plus shock
absorbers as positive travel limitations. Collisions between the EVTM and

other equipment is prevented by a combination of hard stops (with shock
absorbers), soft stops (limit switches which interrupt power to the gantry
under all conditions), interlocks (which interrupt power unless certain pre-
specified conditions are met), administrative procedures, and an operator
59 riding on the trolley-mounted cab. Permanent hard and soft stops are
44 installed at both ends of the gantry rails. Temporary
hard and soft stops are installed on the RSB side of the RCB equipment hatch
after it is closed following refueling. Interlocks are provided to assure:
(1) that the EVTM is centered on its gantry prior to passage through the
equipment hatch, (2) that the IVTM is rotated to the farthest position away
from the reactor fuel transfer port, (3) the reactor fuel transfer port is
44 rotated over a fuel transfer and storage position, and (4) power to the reac-
tor rotating plugs is off before the EVTM can approach the reactor. The hard
and soft stops and interlocks are all backups for, or are backed up by, the
operator and normal administrative procedures.

The EVTM containment (pressure) boundary is constructed in accord-
ance with the rules described in Section III of the ASME B&PV Code for
Class 3 vessels. The EVTM pressure boundary support is constructed accord-
ing to the rules described in ASME III NF3 up to the gantry-trolley interface.
The gantry-trolley design is governed by CMAA 70 (Crane Manufacturers Asso-
ciation of America, Specification 70) and by the AISC standards, consistent
with LWR refueling machine and gantry-crane industry practice. The AISC
design rules are essentially identical to those of the ASME Code, Subsec-
tion NA, Appendix XVII for ASME III, Classes 2 and 3 linear-type support
structures. Both the EVTM, and its gantry-trolley are Seismic Category I 43
structures.

The EVTM drip pan pots are inside the EVTM containment and are at
all times surrounded by an inert atmosphere. Three drip pan pots and a
throughport are mounted on the EVTM drip pan turntable. It can be rotated by
a motor to position a pot under the EVTM cask barrel after a CCP has been
hoisted. The EVTM extender module and closure valve are below the drip pan
module. Before the EVTM is decoupled from a floor valve over a fuel transfer
44 port, the closure valve will be closed, and no exchange of RSB/RCB atmosphere
with the inert EVTM atmosphere can take place. After coupling the EVTM to a
floor valve and opening both closure and floor valves, the EVTM inert gas
atmosphere is communicating with the cover gas of the reactor, EVST, or FHC.
The valve sequencing, inert gas purging of the valve interfaces, and the
coupling and decoupling sequences are interlocked in such a way that no air
can enter into the EVTM or into the fuel transfer port.
59

Each drip pan pot has three level indicator posts mounted on the
bottom which indicate the 20, 40, and 60% full positions. The sodium level
relative to the posts is remotely viewed by the EVTM operator using an opti-
cal system. This optical system consists of a remotely rotatable TV camera,
a heated 45° - mirror, a heated viewing window, and a light source with fiber 25
optics light guide.

After one drip pan pot is 60% full, another (empty) drip pan pot is rotated into its place. Each drip pan pot is not filled completely to permit pickup by the EVTM grapple and avoid covering the grapple with an excessive amount of sodium. The purpose of providing three drip pan pots in the EVTM is to minimize the refueling time penalty due to drip pan pot exchange in the FHC. It has been estimated that the EVTM requires at the most four trips to the FHC for drip pan pot emptying during one refueling. (This conservatively assumes failure of the CCP siphon; normally there will be only one emptying at
44 the end of refueling.)

When all three drip pan pots in the EVTM have been filled to 60%, the EVTM travels to the FHC and mates to the FHC maintenance port floor valve. After opening the closure and floor valves, the CCP grapple of the EVTM 25

lowers one drip pan pot into an empty, heated position of a rotary table,
directly below the maintenance port in the FHC. The rotary table has 6 posi-
59 tions, three of which are heated and three are not heated. The grapple and drip
pan pot dwell for a short time in the heated rotary table position until any
44 frozen sodium on the grapple has melted, and the grapple fingers are able to
retract. The grapple is released and raised a short distance. Next, the
rotary table rotates and brings the unheated position, containing three empty
drip pan pots, under the EVTM grapple. The grapple is lowered, engages an
empty drip pan pot, and is hoisted into the EVTM. There the empty pot is
deposited in the drip pan assembly. The same operation is repeated twice
more until the EVTM contains three empty pots, and the FHC three full pots.
The EVTM then uncouples from the FHC and resumes its refueling operation.

The three drip pan pots with molten sodium in the FHC are picked up
by the powered manipulator, one at a time, and poured into a waste container.
The FHC operators observe that each drip pan pot has only a minimum of resid-
ual sodium left before returning the pot to the unheated position in the
rotary table. The drip pan pots are not decontaminated after each emptying
since they are used on a repetitive basis. The container with frozen, pos-
44 sibly contaminated sodium, is later transferred out of the FHC and turned
over to the Radioactive Waste System for further processing and disposal.
Procedures for handling and disposing of radioactive metallic sodium are
discussed in PSAR Section 11.5.3. 25

9.1.4.3.3 Safety Evaluation

The dose rate from the highest powered spent fuel assembly is lim-
49 44 ited to less than the limits given in Sections 12.1.1 and 12.1.2 at the
surface of the EVTM cask body. A significant dose rate exists only during
the time when a spent fuel assembly is located in the machine. Under normal
44 conditions, this amounts to less than 1 hr. per assembly for a maximum of
59 162 fuel assemblies. In addition, the closest locations where personnel
can be exposed to the radiation source are 10 ft. from the cask body for
44 normal operation and 1.5 ft. for infrequent service operations. These distances
result in an attenuation of personnel doses by more than a factor of three,
59 so that the integrated dose to personnel is less than the maximum allowable
dose.

The EVTM has adequate seals to prevent excessive radioactive emis-
sions to the operating floor of the RSB or RCB. Radioactivity released from
the EVTM will not exceed the limits set forth in Section 12.1 when combined
with normal releases from all other sources in the RCB or RSB. The RCB and
RSB have radioactivity monitors to detect accidental releases and to sound
alarms. Leakage through the seals has been evaluated in Section 15.5.2.3 for
the case of 100% release to the interior of the EVTM of all fission gas in a
high powered spent fuel assembly and there is no hazard to the public.
59

Assessment of the physical constraints to both horizontal and ver-
44 tical motion of the EVTM with relation to the floor valve and closure valve

indicates adequate assurance for both an OBE and SSE that: (a) the composite component will reseat from much greater than maximum anticipated vertical motion; (b) the clamps will prevent disengagement of the extender from the closure valve; and (c) the lip on the closure valve will limit horizontal motion to one inch. The latter is more than adequate to prevent contact with or damage to a CCP that might be in transit through the plane of the slip joint. The seals between the extender and the closure valve and between the closure valve and the floor valve ensure cover gas containment under normal
44 and seismic conditions.

The EVTM cooling capacity of 20 kW is adequate to provide a substantial margin above the maximum normal heat load expected, which is 15 kW The active portion of the cooling system, the blower, is capable of providing the specified cooling without exceeding normal temperature limits. In case
44 of failure of the blower or loss of all AC power, completely passive cooling is automatically provided by natural convection. In this case, cladding temperature is maintained to less than the limit for unlikely events. Some cladding failures, resulting in the release of fission gas to the interior of the EVTM, might occur; but as shown in Section 15 such a release is well well within acceptable limits.

43

59

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

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

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

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

<u>Full-Scale Heat Transfer Tests</u>

Full-scale tests (Reference 7 of Section 1.6) were performed in a HEDL test facility design to simulate the cooling systems of the CLEM for the FFTF and the EVTM for the CRBRP. The fuel assembly was simulated by a full scale, 217-pin, electrically heated "fuel" bundle in a hexagonal duct. The fuel assembly was contained in a sodium filled core component pot (CCP), surrounded by an inert gas filled annulus, and cooled by the concentric cold wall. The test facility and test article design assured that accurate extrapolation could be applied to test data for either refueling machine. Major test results showed the following: 15

position. The EVTM grapple has a conical-shape to provide lead-in for the CCP handling socket and could pick up the dropped CCP.

Even in the dropped position, the entire heat generating region of the spent fuel assembly in the CCP would be within the EVTM cold wall. Conservative heat transfer analysis results have shown that steady state temperatures of the seals near the bottom of the cold wall would be below 320°F. The steady state peak temperatures of the fuel assembly cladding and of the sodium in the CCP would be below 1300°F. The analysis assumed a CCP containing a 20-kw spent fuel assembly with part of the heat generating region of the fuel assembly positioned at a level below the cold wall.

The consequences of an accident involving a CCP dropped within the EVTM can be summarized as follows:

1) The dropped CCP can be picked up again by the EVTM grapple.

2) If pickup were delayed, the cladding of the highest powered spent fuel assembly would reach a steady state peak temperature below 1300°F, which is well below the boiling point of sodium.

3) EVTM seal temperatures at the worst location would be low enough not to affect the integrity of elastomeric seals.

Therefore, the event will not result in any impact on the health and safety of the public. 25

Dropping of a spent fuel assembly onto the operating floor is pre-
44 cluded by the following design features and interlocks:

1) Interlocks prevent release of grapple fingers if load cells indicate that the CCP is supported by the grapple.

2) Grapple design makes release of CCP under load impossible.

44 3) Drip pan prevents a CCP from dropping more than 4.4 feet within the EVTM.

4) Interlocks prevent opening of EVTM closure valve unless EVTM is mated to a floor valve at a fuel transfer port.

Concurrent or sequential failure of all these design features and interlocks is considered to have a hypothetically remote probability, but would be necessary in order to drop a CCP onto the operating floor. However, even in this case the CCP would remain vertical within the EVTM, with only
20 the lower 2 feet exposed, and it would be adequately cooled.

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9.1.4.4 Safety Aspects of the In-Vessel Transfer Machine (IVTM)

The IVTM is an under-the-shield refueling machine that is mounted on the Small Rotating Plug (SRP) of the reactor head only when the reactor is shut down for refueling. The primary function of the IVTM is to move core assemblies between their normal positions in the reactor core and the transfer positions outside the core, but within the reactor vessel. In the transfer positions, the core assemblies are accessible for transfer in or out of the reactor vessel by the EVTM. The IVTM executes only vertical movements, all horizontal movements are performed by rotation of the triple rotating plugs in the reactor head. The machine is depicted in Figure 9.1-16. The IVTM is based on a concept which has been under development in the LMFBR Base Program for several years. Major subassemblies (e.g., grapple, core assembly identification system) are very similar to comparable subassemblies of the FFTF in-vessel handling machine (IVHM). 15

The IVTM implements its function by the following operational capabilities:

1) Grapple and release core assemblies

2) Raise and lower core assemblies

3) Provide holddown of adjacent core assemblies

4) Uniquely identify core assemblies

44 5) Orient and center core assemblies for insertion in the core.

The IVTM is a rising stem type design and consists of two principal
44 subassemblies, the in-vessel section, and the ex-vessel or drive section. The in-vessel section contains the grapple, centering device, identification mechanism, the holddown mechanism, and the seals which contain the reactor
20 cover gas. It is installed into the SRP of the reactor vessel head with the
44 auxiliary handling machine. The drive section contains the drive equipment which powers the in-vessel section and is installed with the polar crane. The in-vessel section is designed to operate within the sodium filled envi-
44 ronment of the reactor during shutdown, and the drive section transitions the machine motions to the air filled environment where normal power equipment such as electric motors and pneumatic cylinders can be located.

44 The grapple and drive mechanism raises or lowers core assemblies inside the reactor vessel under sodium. By moving the reactor plugs, core
44 assemblies grappled by the IVTM can be translated to and from a core position and an in-vessel fuel transfer position. Figure 9.1-16A shows the IVTM grapple in a core assembly socket. 14

Safety aspects of the IVTM that will be covered in this section are radiation shielding, prevention of radioactive releases, prevention of
44 mechanical damage, prevention of improper core assembly insertion or removal, and removal of jammed fuel assemblies.

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9.1.4.4.1 Design Basis

Adequate shielding for radiation protection is provided in the design of the IVTM. The shielding meets the radiation protection requirements specified in 10 CFR 20 and ensures an integrated dose below 125 mrem/quarter.

Release of radioactive cover gas through the IVTM during refueling
59 is prevented by buffered seals or welding of penetrations. Radioactive leakage and diffusion through seals is less than the limits specified in Sections
49 12.1.1 and 12.1.2.

44 Mechanical damage to core assemblies could potentially be caused by the IVTM due to the following events:

44 1) Exertion of excessive push or pull forces to core assemblies

2) Unscheduled movement of the IVTM grapple or the reactor plugs

44 3) Dropping of a grappled core assembly.

Upward or downward vertical forces on core assemblies during insertion or removal from the core by the IVTM grapple drive mechanism are limited by the load control system to 1000 LB push or pull loops for normal operations,
44 and up to the maximum allowable push and pull loads on a core assembly of 3000
59 lb and 4300 lb, respectively, for moving a stuck core assembly.

Vertical movement of the grapple is prevented by interlocks until positive grapple engagement or disengagement is indicated. These interlocks
44 are discussed in Section 7.7.1.9.

Horizontal translation of the grapple by reactor plug rotation is prevented by interlocks until the grapple, the grappled core assembly and the
44 holddown sleeve are fully raised to clear the upper core internals.

The design of the IVTM grapple, in conjunction with the handling lip on core assemblies and suitable interlocks, prevents dropping of core
44 assemblies.

Improper core assembly insertion in the core is prevented by limit switch interlocks and the configuration of the core assembly discrimination post and its receptacle. Verification of proper seating is accomplished by satisfaction of all interlocks, and by the IVTM position indication and load
44 control systems.

Improper core assembly insertion is prevented by IVTM design features permitting unique identification of each core assembly inside the reactor vessel. Core assemblies are also identified immediately after removal
44 from a core position to ensure that the proper core assembly has been removed.

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9.1.4.4.2 Design Description

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

44 The IVTM is sealed to the SRP by 3 elastomer O-ring seals with buf-
fer gas between them. Between each set of seals, a leak detector port is
provided to enable connection of a sensor for monitoring the seals integrity
before refueling. This arrangement is typical for other seal sets that
44 involve dynamic motion such as the grapple stem and holddown drive shafts.
Static seals or seals with very small displacements, such as those for the
grapple finger actuator shaft are double, with capability for convenient
44 20 periodic leak testing.

The IVTM drive mechanism has been designed to exert an upward or downward load of 5000 lb maximum. Incorporated into the design is a pneumatic load control system and a load cell system that limits the load exerted on a core assembly to 4300 lb pull and 3000 lb push.

The load in the pneumatic load control system will be set to provide a norm[illegible] or pull force on a core assembly of 1000 lb. The pressure of the loa[illegible]l system may be adjusted to provide higher load capability up to the pu[illegible] or pull load limits.

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

1. The primary limitation is provided by the pneumatic load control system.
The pressure in the pneumatic load control cylinders limits the load
59 applied by the electromechanical actuator to the driven core assembly.
When the core assembly insertion resistance load exceeds the pressure
setting in the system, the core assembly stops moving, but the
electromechanical actuator continues driving the pistons for about
0.25 inches. This differential travel trips a set of limit switches
which automatically stop the electromechanical drive. Self-contained
hydrauluc dashpots prevent sudden actuator movements due to sudden
changes of the frictional resistance at the load.

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

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

59 Grapple vertical motion is possible only when interlock switches indicate either grapple fingers are retracted, or grapple fingers are extended. Schematics of the IVTM interlocks are shown in Figure 9.1-16B and 9.1-16C. 16

The grapple and grappled core assembly must be raised a minimum of 209.8 in., and the holddown sleeve a minimum of 40 in. above the core plane to clear all mechanical interferences before the interlocks permit horizontal translation.

There are three cam operated fingers for picking up a core assembly.
The cams are mounted on a rod that is operated by an air cylinder mounted at
the top of the grapple stem in the drive section of the IVTM. When the rod
is down, the fingers are extended in the pickup mode; when the rod is raised
(2.75 in.) the fingers are retracted in the release mode. The design and
spacing of the cams on the rod provide for positive engagement or disengagement
59 such that a core assembly cannot be partially grappled.

A mechanical interlock and redundant interlock switches prevent actuation of the IVTM grapple fingers during insertion of a core assembly into the core until the core assembly is 1.32 inches from being fully seated in the core. Within the 1.32-inch distance, proper engagement of core assembly discriminator post to its mating receptable is ensured, and the core assembly can be released after the switches of the load control system are tripped, thereby shutting off the electromechanical drive.

Since the core assembly release can be initiated anywhere within the 1.32-inch distance (core assembly insertion may be terminated in this region as a result of high core friction) assurance that the core assembly is resting at the bottom of its receptable can be verified by the IVTM position indication system within the uncertainty due to the core-assembly-to-core-assembly tolerance, axial core assembly dilation and position indication system inaccuracy. However, the 1000 lb normal push force applied by the grapple is considered to be sufficient to ensure that the core assembly is fully seated in its mating receptacle.

When removing an irradiated core assembly from its reactor core
44 location, the assembly is raised into the IVTM in-vessel section by the IVTM
grapple. In this position, the core assembly is rotated by rotating the grapple stem. The rotational position of the grapple stem is checked by an
encoder, driven by the rotating stem. Each core assembly has an orientation
notch and uniquely defined identification notches on the outside diameter of
44 its handling socket. During rotation of the core assembly these are felt by
an identification and orientation pawl, and transmitted by a shaft to an
external encoder that supplies electrical signals to the IVTM instrumentation
and control system.

44 Each irradiated core assembly is thus uniquely identified immediately following removal from its core location. Prior to any further changes

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in core geometry, the identification number of the irradiated core assembly
has to match that specified in the refueling computer program (when operating
44 in the automatic mode) or the operator's instruction (when operating in the
manual mode). Failure to match will automatically halt refueling when in the
automatic mode . The interruption in the manual control mode is accomplished
59 up by the operator.

44 Each new core assembly is likewise identified by the same method,
after the IVTM grapple has removed it from one of the in-vessel transfer
44 positions and before it is oriented and inserted into an empty core location.

All transfer operations of the IVTM involving core components are
performed entirely under sodium. More than 20 ft of sodium is above the top
of the reactor core and the fuel transfer positions, as shown schematically
69 in Figures 9.1-16B and 9.1-16C. The fueled region of the fuel assemblies is
covered by approximately 9 ft. of sodium during all transfer operations
involving the IVTM, as stated above.

When a spent fuel assembly in a CCP is hoisted from the reactor
fuel transfer position into the EVTM cask, it is cooled along its path of
vertical travel first by the reactor sodium, then by heat transfer to an air-
cooled insert in the reactor fuel transfer port (see Section 9.1.4.7), and
finally by heat transfer to the air cooled EVTM cold wall.

There is about 7 ft of travel between the sodium surface in the
reactor vessel and the bottom of the cooling insert in the reactor fuel
transfer port, and about 11 ft between the top of the cooling insert and the
bottom of the EVTM cold wall. A fuel assembly in a CCP traveling through
these two sections is only cooled by natural sodium and argon convection,
conduction, and radiation. Normally that part of the CCP containing the heat
generating part of the spent fuel assembly and the hot CCP sodium above it
passes through these two zones of reduced effective cooling in 1 min. and
1.7 min., respectively. In case of a malfunction of the EVTM, the CCP can be
either manually lowered or raised at a reduced speed, bringing it into a
position where it can be adequately cooled. In such an event the CCP would
travel through the two zones in a time conservatively estimated to be 16 min.
and 20 min., respectively. Transient heat transfer analyses have shown that
the peak cladding temperature of a 20 kw spent fuel assembly in a CCP reaches
about 820°F after 20 min. and 940°F after 30 min. when the CCP is immobilized
in a location where no forced air cooling or sodium pool cooling is available.
The peak metal temperatures in the vicinity of the nearest containment seals
were calculated to about 160°F. The initial temperatures of the fuel assem-
bly and the CCP sodium are those of the reactor sodium during reactor shut-
44 down for refueling, i.e., about 400°F.

It is concluded that this event will not produce radioactivity
release into the EVTM and reactor vessel and/or result in a breach of EVTM/
reactor vessel containment. 25

44 If a core assembly is found to be jammed during an attempt to with-
draw it from the core, i.e., the normal IVTM pull load of 1000 lbs is not
44 sufficient to remove it, additional pull load may be exerted with the

approval and cognizance of the refueling supervisor. The pull load can be increased as required by increasing the pressure setting to the load control
44 system up to a limit of 4300 lb. Maximum worst friction loads are currently calculated to be less than 3000 lbs. If the maximum allowable pull load of 4300 lbs is reached and the assembly cannot be removed, the surrounding core assemblies (which may be bowed) will be removed and replaced one at a time,
44 as necessary, with either new core assemblies or core special assemblies (which are straight and have greater clearance than fuel assemblies). This procedure reduces interaction forces from core distortions until withdrawal forces are lowered sufficiently to remove the jammed assembly.

Fuel assembly bowing increases the contact force between assemblies. Because of the resulting inter-assembly frictional force increment, the force required to remove spent fuel assemblies is also increased due to fuel assembly bowing. In general, assembly bowing, and thus assembly withdrawal loads, increase with increasing burnup. Also, withdrawal loads are higher for assemblies on the core perimeter than at the center because assembly bowing is typically larger at the core perimeter.

The spent fuel assemblies can withstand the maximum applied withdrawal pull of the IVTM (4300 lbf) with a margin of safety greater than 4, based upon prevention of yielding in the critical duct welds. The fuel rod bundle is not mechanically attached to the duct. Thus, the force transferred to the fuel rods due to relative rod bundle-duct motion would be due only to frictional forces between the rod bundle and the duct wall. These forces are not applied directly to the fuel rod cladding, but rather to the wire wrap spacer. The magnitude of the rod bundle-duct frictional force depends upon the fuel rod bundle porosity, burnup, and compressibility at end of life. The component of these frictional forces which are applied to the fuel rod cladding, depend on the tension in the wire wrap at end of life. These parameters are currently being analyzed and correlated with experimental results. The actual maximum withdrawal force the fuel rod bundle can withstand at end of life is also currently being analyzed; it is a function of end-of-life cladding materials properties, rod configuration, and rod bundle-duct frictional forces. These parameters will be determined as part of the test program described in the response to NRC Question 001.283. The consequences of extensive fuel rod failure in a spent fuel assembly during refueling were examined in Section 15.5.2.1 in connection with the analysis of a dropped fuel assembly. This analysis concluded that fuel rod cladding rupture within the reactor vessel during refueling can be accommodated from the standpoint of both fission gas release and fuel particle dispersal. 19

9.1.4.4.3 <u>Safety Evaluation</u>

Adequate shielding is provided by the IVTM's shield column to limit the integrated radiation dose to personnel from the IVTM to less than the
59 49 maximum allowable dose rate.

The IVTM, when mated to the SRP, has adequate seals to prevent excessive radioactive release due to cover gas leakage into the RCB. Radioactivity

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

The drive mechanism and grapple design, together with load controls,
limit switches, and interlocks will prevent exertion of excessive forces on
44 core assemblies, unscheduled vertical movement of the grapple or disengage-
ment of the grapple fingers except in the full-down position of the grapple.
59 44 Limit switches and interlocks (see Section 7.6.2) will also prevent inadvertent
rotation of the reactor plugs. The extremely unlikely accident of a fuel
assembly dropped from the IVTM is discussed in Section 15.5.2.1.

44 The combination of position interlock switches and the switches of
the load control system in conjunction with the design of the discriminator
post and matching receptacle ensures proper seating of core assemblies. The
44 fuel and inner blanket assemblies are divided into 8 groups, each having a
different configuration of discrimination post inner and outer diameters fitting
into corresponding receptacles. This ensures that a fuel assembly of one group
59 44 can only fit into its corresponding receptacle, see also Section 4.2.1.2.3.

Core assembly identification by the IVTM identification and orien-
tation system prior to core assembly insertion into the core, and the posi-
tion indicator system of the reactor rotating plug, will ensure core assembly
insertion into a correct core location.

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

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

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

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Measures to ensure proper indexing of the IVTM relative to the core are taken prior to refueling start and are as follows:

1) The refueling program is prepared and verified in advance.

44 2) The refueling program is checked out on a simulator.

An additional safeguard against gross misloading errors is provided
44 by the low neutron monitoring system which will detect any significant deviation from planned changes in reactivity.

The IVTM, in conjunction with the reactor head rotating plugs and new core assemblies or core special assemblies, is capable of safely per-
44 forming the necessary operations to remove a jammed core assembly without special equipment.

9.1.4.5 Safety Aspects of the Auxiliary Handling Machine (AHM)

The primary function of the auxiliary handling machine (AHM) is to install the in-vessel section of the IVTM in the reactor head prior to refueling, and to remove it afterward. The AHM also serves a plug handling
44 function, in removing and reinstalling the IVTM port plug from the reactor head. Additionally, the AHM supports maintenance functions of other systems, by removing and installing control rod drive lines and core inlet modules using special tools, grapples, adapters, etc., provided by the Reactor Enclosure System and the Reactor System.

The AHM has the following capabilities:

1) Grapple and release components

2) Raise and lower components

3) Transportable by the RCB polar crane (see Section 3.8)

4) Maintain argon environment

5) Provide sealing and containment of the reactor cover gas when mated to the reactor

6) Provide radiation shielding

7) Collect sodium drippage from components handled with surfaces
20 wetted by sodium.

The AHM is a shielded, single-barrel handling machine which is transported by the RCB polar crane. The general arrangement of the machine is depicted in Figure 9.1-17. The machine is about 60 ft high and weighs
44 about 100 tons. The AHM is similar in concept to the EVTM, but it has been

simplified, due to the less demanding requirements of the components it handles (as compared to the irradiated core assemblies in sodium-filled core component pots handled by the EVTM). The major differences are: (1) no
49 radioactive decay heat, (2) lower radiation source strength, (3) less sodium
drippage and (4) lower use factor.

The major subassemblies of the AHM consist of the following: the
44 grapple and hoisting system; the cask body assembly, which provides structural support and shielding; the service platforms; a closure valve and drip
44 20 pan assembly; an extender assembly; a cask containment barrel; a gas service system, which provides gas for process systems and also monitors seal leakage; and lastly, electrical instrumentation and control equipment.

Operations with the AHM begin by connecting the crane hook to the lift structure, then disconnecting the AHM from the AHM storage facility support structure. The polar crane moves the AHM to the proper location, and raises the AHM to place the retracted extender at 9 inches above an AHM floor
59 44 valve (see 9.1.4.6) installed at the location. The extender lowers a mating
flange onto the floor valve. After sealing and purging (see 9.1.4.3), the closure and floor valves are opened. The grapple is driven down, the fingers
20 are actuated, and the component is lifted into the AHM cask. The combination drip pan - closure valve is closed first, then the floor valve is closed.
44 After purging the space between the valves, the extender is raised, and the AHM is moved away by the crane.

The design bases, description, and safety evaluation of the AHM discussed below cover radiation shielding, prevention of radioactivity release, and prevention of mechanical damage to the reactor.

9.1.4.5.1 Design Basis

Shielding for radiation protection is provided in the design of the AHM. The shielding meets the radiation protection requirments specified in Sections 12.1.1 and 12.1.2 and ensures that the integrated dose outside the
49 AHM is less than 125 mrem/quarter.

Radioactive releases from the AHM during installation or removal of the components handled by the AHM are prevented by proper sealing or welding of penetrations. Radioactive leakage and diffusion through seals are limited to meet the radiation requirements of Sections 12.1.1 and 12.1.2.
49

Mechanical damage to the reactor could potentially be caused by the AHM due to dropping of a grappled component or toppling over or dropping of the AHM. Dropping of components handled by the AHM is prevented by the design of the grapple fingers, the grappling lip of the component grappling sockets, and by suitable interlocks. The AHM handling lugs and lifting eyes are designed to assure positive engagement and accommodate all normal and SSE loads. A single failure of the lifting structure does not lead to a drop of
44 the AHM. The potential drop height of the AHM is limited, and the RCB polar

crane lowering speed is limited. The AHM is supported from the polar crane during the entire time when it is in the vicinity of the reactor head and is securely attached to its parking structure when not in use.

If a seismic event occurs while the AHM is mated to the floor valve at the reactor, the design limits the transmitted structural loads, such that the reactor head is not damaged. The ability to operate the AHM floor valve after the seismic event is maintained, and no damage of major equipment located in the head access area is caused by the AHM during or after an OBE or SSE. The AHM is installed on the reactor only when the radioactivity concentration of the reactor cover gas is below the limit given in the Technical
44 Specification, Section 16.3.10.

9.1.4.5.2 Design Description

Figure 9.1-17 shows the AHM shielding arrangement. The cask body is of steel construction with different thicknesses (largest at the bottom) of steel to provide the required shielding. The extender section that mates to the AHM floor valve includes lead shielding. Dose rates at the cask body
49 surface are less than the limits specified in Sections 12.1.1 and 12.1.2. The
44 maximum radiation source is the IVTM port plug.

The AHM is coupled to the floor valve with a mating flange located
44 at the bottom of the AHM extender prior to handling components. The mating surfaces have double seals with leak check capability to prevent radioactive releases. The only source of gaseous activity in the AHM is reactor cover
20 gas containing fission products.

The AHM, like the EVTM, has an interlocking finger design such that, when the handled component is engaged with and supported by the grapple fingers, the fingers cannot be retracted even if the entire weight is supported by the finger-actuating chain. Redundant support chains are used to prevent component dropping in the event of a single chain failure.

The AHM is moved between its storage facility, the reactor head and the RCB plug storage facility by the RCB polar crane at a maximum height of 2 ft. above the operating floor. When the AHM is positioned over a
59 reactor port or plug storage facility port, it is entirely supported by the
44 single failure proof RCB polar crane. The crane raises the AHM to a height of 9 inches above the floor valve and the AHM extender is lowered to mate with the floor valve. Administrative control of this operation prevents accidental lowering of the AHM onto the reactor head. The crane speed is
59 limited to less than 5 fpm. A complete discussion of a lowering event is
44 presented in Section 15.5.2.5.
20

Handling lugs, crane rigging, and lifting eyes are designed to accommodate all normal dynamic and static loads, in addition to loads caused by SSE, to prevent dropping of the AHM. The handling bail is attached to the
44 polar crane hook by a large diameter clevis pin. Wire ropes and spreader

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44 assemblies provide a second, redundant load path from the handling bail to
the polar crane hook. When not in use, the AHM is stored at a parking sta-
59 20 tion located in the northeast quadrant of the building.

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

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

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

9.1.4.5.3 Safety Evaluation

The radial and axial shielding provided by the AHM limits the integrated
59 49 dose to personnel to less than the maximum allowable dose rate during the instal-
lation or removal of the components handled by the AHM. As with the EVTM
(see 9.1.4.3) the radiation source in the machine is intermittent and short
term.

59

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

49

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

If a seismic event were to occur when the AHM is mated at the reactor SRP, shear pin severance between the AHM and the floor valve would leave the AHM free to swing as a pendulum on the RCB polar crane. The pendulum action of the AHM after decoupling would not induce damage to other equipment, since there is at least 7 inches vertical clearance from any adjacent hardware. The closest item is the space allocation for plumbing and wireways to the control rod drive lines. The AHM is only installed on the reactor when the reactor is shut down, and the control rods are fully inserted and disconnected from their drive lines. The automatic rise of the extender after shear pin severance ensures that the extender cannot drop to its maximum extended position, i.e., to a level of approximately 5 inches below the upper floor valve flange.

If the AHM were to break away during a seismic event coinciding with an open floor valve, reactor cover gas could be released into the RCB. The probability for the combination of these events to occur simultaneously is small, (unlikely incident), since the AHM does not spend more than about 13 hours per year at the reactor and the floor valve is only in the open position about 9 hours per year. The normal time from reactor shutdown to reach a configuration with the AHM installed on the floor valve on top of the reactor SRP, and the floor valve gate being open, is more than 3 days. An unlikely event discussing accidental cover gas release from the reactor
59 58 hours after shutdown is discussed in 15.5.2.4. The accident was found to result in a site boundary dose well below the values given in 10 CFR 20 for unrestricted areas. This dose is considered very low, since the applicable guide line values for this event are those of 10 CFR 100. The cover gas radioactivity concentration will be continuously monitored before and after the AHM is installed. AHM installation on the reactor, and subsequent opening of the floor valve gate, will only be carried out if the cover gas activity is below the limit set by a Technical Specification, see Section 16.3.10. In the unlikely event that all cover gas is released into the RCB, operators could work in the RCB for a sufficiently long time to manually close the floor valve, including preparation time, without exceeding the allowable quarterly dose for restricted areas of 1.25 rem given in 10 CFR 20.

In the extremely unlikely event such that the IVTM port plug, the IVTM in-vessel section, or a control rod drive line (CRDL) were to be in transit past the AHM extender/floor valve interface at the moment of an earthquake, and the AHM broke away, sufficient relative motion could occur to damage these components. Due to the relatively small diameter of the CRDLs, damage would not prevent subsequent removal of the grappled components and closure of the floor valve. With the larger diameter IVTM port plug and the
59 IVTM in-vessel section, the effect could be to key the AHM to the floor valve, negating the break-away feature. Overstressing of the AHM, adaptor and SRP could then result, and damage sufficient to jam the IVTM port plug or the IVTM in-vessel section could be possible. In this event, the floor valve could not be closed. Cover gas leakage would be much slower than the immediate release described above. This would permit time to purge the reactor cover
44 gas, suit-up, and take emergency corrective action with no public safety consequences.

59 The total time interval for all grappled components passing through the AHM extender/floor valve interface is less than one hour per year, the above described events are considered to be extremely unlikely.

44 59

9.1.4.6 Safety Aspects of Floor Valves

The functions of a floor valve are to provide shielding and sealing against radioactivity when the plug is removed from a port in the reactor, EVST, or FHC, thus isolating the contaminated gas and radiation environment of these facilities from the ambient air of the RCB and RSB.

The floor valves are portable rotatable disk-type valves used to seal the reactor, EVST, and FHC ports during various phases of the refueling operations when the port plugs are removed. Two basic sizes of floor valves
44 are used: one for the EVTM for the transfer of core assemblies, and another with a larger opening, used with the AHM for the transfer of the in-vessel section of the IVTM and other components handled by the AHM. Both types have the same basic design as depicted in Figure 9.1-18, except for interior opening size and differences in thickness of the radial shielding surrounding the port.

44 The floor valve design concept is similar to the AHM and EVTM closure valve design. It is also similar to the floor valve developed for the FFTF program. Floor valves are installed by the building crane and bolted down. The AHM or EVTM is moved into position over the floor valve, and the closure valve extender is lowered onto the floor valve at the reactor, the EVST or FHC. After sealing the closure and floor valves are opened. The port plug is removed and the valves are closed. The extender is raised, and
20 the AHM or EVTM is moved away by the crane or the gantry, respectively.

This section covers the safety aspects of the floor valves, i.e., shielding and prevention of radioactive releases.

9.1.4.6.1 Design Basis

The design bases for shielding and radioactive release are the same as for the AHM (see 9.1.4.5.1). In addition the appropriate zone shielding
44 criteria of Section 12.1 apply.

9.1.4.6.2 Design Description

The shielding design of the floor valve which is used with the AHM (23-inch diameter port opening) is an asymmetric cylindrical annulus of steel with a minimum radial shielding thickness of 10 in. The vertical height of the floor valve provides a lead shielding thickness of approximately 9-1/2 in. The floor valve used in conjunction with the EVTM is larger in overall diameter (78 in. as compared to 68 in. of the AHM floor valve), and in height (17 in.
44 20 versus 15 in. for the AHM floor valve). The EVTM floor valve port has an

~11 in. diameter opening. The increased thickness of the EVTM floor valve in radial and axial direction provide the additional shielding required for the much higher radiation source which passes through an EVTM floor valve (spent
20 fuel assembly) compared to an AHM floor valve (IVTM port plug).

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

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

9.1.4.6.3 Safety Evaluation

The radial shielding limits the dose rate on the floor valve surface to less than the criteria in Sections 12.1.1 and 12.1.2 during transfer of the
44 49 highest powered spent fuel assembly (for the EVTM floor valve). The floor valve is considered a piece of equipment whose main function is to permit transfer of radioactive components, both fueled and non-fueled, between a machine and a facility. The radiation source is transient and short term (less than 1 min per transfer) in nature. Hence, it results in a low integrated dose.

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

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

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As discussed in Section 15.5.2.4, an unlikely accident releasing radioactive cover gas from the reactor leads to a site boundary dose well below the guideline value of 10 CFR 20.

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

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

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

9.1.4.7.1 Design Basis

44 The design bases for shielding and radioactive release of the fuel transfer port adaptor are the same as for the EVTM (see 9.1.4.3.1). The reactor, EVST, and FHC fuel transfer port cooling inserts have the capacity
44 59 to remove decay heat from 20 KW irradiated core assemblies in sodium-filled CCP's to prevent exceeding the 1500°F spent fuel cladding temperature limit specified for unlikely or extremely unlikely events (Table 9.1-2).

9.1.4.7.2 Design Description

The reactor fuel transfer port adaptor extends from the upper surface of the fuel transfer port in the reactor head to the operating floor, see Figure 9.1-19. The upper surface of the reactor fuel transfer port adaptor consists of a flange which is bolted to the floor valve and provides the
44 sealing surface for the double seals on the lower surface of the floor valve. Shielding is provided by a thick, annular lead cylinder surrounding the adaptor cover gas containment tube over its entire length to limit the dose rate at the shield surface to less than the limits given in Sections 12.1.2 and
49 44 20 12.1.2. The lower part of the adaptor is bolted to the reactor head during refueling only.

44 The reactor fuel transfer port cooling insert extends from the top flange of the adaptor to the fuel transfer port nozzle. The cooling insert uses a cold wall cooling concept, similar to the EVTM. The CCP containing a spent fuel assembly is cooled by thermal radiation and conduction across the argon gas gap to the cold wall which forms the confinement barrier for the reactor cover gas. Ambient air is blown down the outside annulus of the
44 cooling insert, and discharges into the reactor head access area. Air flow from the blower is adequate to limit the cladding temperature of a 20 KW fuel assembly to less than 1500°F.
44

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9.1.4.7.3 Safety Evaluation

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

Cooling is adequate to meet the core assembly cladding temperature
limit of 1500°F for unlikely or extremely unlikely events of the fuel assembly
59 being immobilized in the fuel transfer port. The clearances between the CCP
44 and the port wall are generous to prevent hangup of the CCP. Also, the EVTM
can be used to lower the CCP back into sodium or raise it into the EVTM; in
44 either location it can be cooled normally before emergency cooling is needed.
In case of loss of power to or failure of the EVTM drive motors, this raising
or lowering can be done manually.

9.1.4.8 Spent Fuel Shipping Cask

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

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

9.1.4.8.1 Design Basis

The free fall impact energy of the 72 ft SFSC drop to the bottom of
the cask handling shaft shall be limited to an amount less than that experi-
20 enced in a hypothetical cask accident specified in 10 CFR 71.

9.1.4.8.2 Design Description

The SFSC is handled within the RSB and lowered and raised in the cask shaft by the double reeved RSB bridge crane using rigging specially designed and tested for the SFSC. Preliminary analysis indicates that a

20 72 ft drop of the SFSC onto the concrete floor of the cask shaft would result in a peak deceleration of 92 g, i.e., less than that produced by a 30-ft drop onto an unyielding surface. The reason for the lower deceleration lies in the difference between an unyielding surface and a concrete slab surface.

If the cask corridor transporter is in place at the bottom of the shaft when the SFSC is dropped, it will function as an energy absorber, further reducing the deceleration loads.

9.1.4.8.3 Safety Evaluation

A drop of the SFSC down the shaft onto the concrete slab will produce a decelerat·'n which is smaller than that of a hypothetical 30-ft drop onto an unyielding surface. If the cask falls on the cask corridor transporter, part of the drop energy will be absorbed by the transporter structure and only the remaining energy will be transmitted to the SFSC. In either case the resulting loads will be substantially less than those for which the
59 cask is designed.

An extremely unlikely accident of a SFSC drop with resulting fission gas release is discussed in Section 15.7.3.2. This accident produces a site boundary dose well below the 10 CFR 20 guidelines.

9.1.4.9 Safety Aspects of the Rotating Guide Tube (RGT)

The function of the Rotating Guide Tube (RGT) is to provide a means for the EVTM to insert a CCP containing a core assembly into one of four fixed in-vessel transfer positions, and to remove a CCP-contained spent core assembly from an adjacent transfer position, all without requiring an EVTM decoupling-recoupling procedure.

The RGT is shown in Figure 9.1-20. It consists of a straight tube with an eccentric lower end. The RGT is permanently mounted on the top of the reactor fuel transfer port nozzle and remains in the reactor during operation. During the refueling sequence, the RGT is rotated 180° to locate its eccentric lower end over either of two adjacent transfer positions. Following refueling, the EVTM is used to insert a plug into the RGT. Sealing and plug holddown are completed by a cap attached to the reactor nozzle. This
44 cap also serves to secure the RGT during reactor operation. For refueling the cap is first removed, and then the reactor fuel transfer port adapter is mated to the RGT prior to removal of the port plug.

The safety aspects of the RGT that will be covered in this section are: radiation shielding, prevention of radioactive releases, and prevention of mechanical damage.

9.1.4.9.1 Design Basis

Adequate shielding for radiation protection is provided in the
20 design of the shield plug to meet the requirements of Sections 12.1.1 and
49 12.1.2 and to keep

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

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

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

9.1.4.9.2 Design Description

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

Hermetic sealing is provided by both plug seals and seals in the RGT cap. A means to purge the cap-plug interface volume before removal of the cap is also provided.

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

9.1.4.9.3 Safety Evaluation

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

Double seals and a capability of purging the cap-plug interface volume ensure that gaseous radioisotope leakage from all sources to the head
49 59 area will never cause a dose rate in excess of that given in Section 12.

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

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

59 20 2) Motion of the RGT with a CCP or grapple extending below the base of the RGT.

59

59 9.1.4.10 Safety Aspects of Spent Fuel Storage in the Fuel Handling Cell (FHC)

The primary functions performed in the FHC are to: (1) Receive irradiated core assemblies from the EVTM, (2) provide interim storage for these irradiated
59 assemblies during transfer operations, (3) examine selected irradiated assemblies, and (4) load irradiated core assemblies into casks for shipment off site. Other functions, also performed by the FHC, are to provide service and maintenance of radioactive fuel handling equipment (e.g., grapple replacement and drip pan change-out for the ex-vessel transfer machine). The FHC also provides contingency-storage for low-heat producing core assemblies, in the event
44 of a complete core unloading (i.e., blanket assemblies, control assemblies and removable radial shield assemblies that produce little decay power and are
53 59 coolable by natural circulation in argon).

These functions are implemented by the following features of the FHC:

1) Radiation shielding

59 2) Inert gas (argon) atmosphere

3) Viewing capabilities

44 4) Remote manipulation and handling of core assemblies and other components

59 5) Cooling of spent fuel assemblies (described in Section 9.13)

44

20 6) Packaging of liquid and solid radioactive waste.

The FHC (located as shown in Figure 9.1-2) is a shielded, inerted, alpha-tight hot cell facility located between the EVTM gantry rails below the operating floor of the RSB.

The cell design is based on similar facilities used on other programs (e.g., the FFTF inspection, examination, and maintenance (IEM) cell, and the National Reactor Test Station hot fuel examination facility (HFEF)
59 cell).

The main equipment groups of the facility as shown in Figure 9.1-7 are (1) a spent fuel transfer station for interim storage of up to 3 (2 during normal operation) spent fuel assemblies, (2) a gas cooling grapple for handling bare spent fuel assemblies, (3) a maintenance and service station and
20 pit, (4) a spent fuel examination station, (5) waste container set-down space, (6) CCP storage racks (with no fuel), and (7) a spent fuel shipping cask loading station. In addition, provisions are made around the walls of the FHC to
53 44 store low-heat-producing core assemblies.

59 44 1. FHC Normal Fuel Handling

In a typical spent fuel handling sequence, a spent fuel assembly in a core component pot is lowered through the fuel transfer port (see Figure 9.1-7) by the EVTM, into the spent fuel transfer station directly below the port. A lazy susan assembly, with three transfer positions supported by a stainless steel gridwork, provides the storage locations. Each position holds one fuel assembly, in a sodium-filled core component pot. Decay heat is
59 44 20 removed by natural convection to the FHC atmosphere.

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

44

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

59 2. Spent Fuel Examination

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

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

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

59

1) Visual inspection of all exterior surfaces

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

3) Measurement of the fuel assembly bow

44 25

44 3. FHC Maintenance

In general, standard maintenance techniques will be used to maintain FHC equipment as have been successfully applied at Argonne West's Hot Fuel Examination Facility (HFEF) at the Idaho Nuclear Engineering Laboratory,
59 and at AI's SNAP reactor test facilities.

Most equipment in the FHC is either modularized and sized such that it can be disassembled and positioned for removal from the cell exterior or repaired in the cell utilizing the remote manipulators.

59 44 Several access modes are available for the FHC including:

1) Removal of all or part of the roof closure gives access to the FHC from the RSB operating floor. 25

2) 28" dia. Floor Valve

59 3) 60" dia. Fuel Transfer Port

44 4) An FHC transfer drawer (gas lock) connects the FHC containment to the FHC operating gallery.

Large equipment (e.g., the crane bridge or trolly) can be removed, i.e., bagged out, from the FHC through the FHC roof closure port using the RSB bridge crane. Once outside the cell the equipment can be transferred to the Large Component Cleaning Vessel (LCCV) for cleaning, if required prior to inspection or maintenance. An alternative is to erect a "green house" (plastic tent) over the FHC roof and part of the RSB operating floor. Large equipment could be removed from the FHC through the roof closure port and set down nearby on the RSB operating floor, within the green house. Through an airlock, suited-up personnel with appropriate respiratory protection could enter the green house and performed the required maintenance tasks.

Suited-up personnel could also enter the FHC to repair nonremovable items or to refurbish the cell. If such an occasion would rise, careful plans would be made in advance for all work to be done after entry to minimize occupancy time in the FHC. All fuel would be removed from the FHC, and the cell would be thoroughly cleaned using the powered manipulator with brushes, dustpans, and vacuum cleaners equipped with HEPA filters before personnel could enter. The personnel would wear appropriate respiratory protection and clothing consistent with personnel safety requirements.

Certain consumable material and spare parts (like motors, switches, light bulbs, etc.) will be supplied for permanent storage in the FHC. Small parts can be easily removed through the transfer drawer and can be moved off the facility by bagging. 25

59 4. Contamination Control

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. If excessive contamination buildup on the FHC liner is indicated, cleaning measures will be initiated. The steel liner on the FHC walls facilitates decontamination.

One method of minimizing the radioactive spread of alpha contamination
49 tion 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 an HEPA filter bank installed upstream of the FHC argon gas circulation blowers.

5. Grapple Cleaning Operations

The maintenance-service station contains equipment for alcohol cleaning EVTM and FHC grapples The grapple cleaning equipment is designed to contain and control the supply of alcohol prior to cleaning and the alcohol-sodium products after the cleaning. The EVTM grapple enters the FHC through the maintenance valve in the FHC roof and is inserted in the cleaning tank. Approximately 13 gallons of ethyl alcohol are transferred from the fill tank, located in the operating gallery, to the cleaning tank in the FHC. The alcohol fill tank is a closed tank designed to operate with argon cover gas. Any sodium adhering to the grapple is removed by reaction with the alcohol. Following cleaning, the alcohol is drained from the cleaning tank and stored in a closed-top drum until ready for removal from the FHC
to be treated as liquid radioactive waste. 25

The grapple cleaning operations take place in the argon atmosphere of the FHC, eliminating flammability problems due to alcohol vapor or hydrogen which is relased during the cleaning process. There is no air in the FHC and oxygen is limited to 75 ppm, which is several orders of magnitude too low to support combustion. The hydrogen generation is very small, about 0.08 lb per grapple cleaning based on removing 2 lb of sodium. The argon circulation rate of 8000 cfm through the FHC and the inlet and exhaust location in the cell assure mixing and dilution of the hydrogen
44 with argon. In addition, a flammable vapor detector is provided in accordance with RDT F5-9T (Sodium Removal Process) requirements. The Inert Gas Receiving and Processing System maintains acceptable hydrogen (and nitrogen) levels in the FHC by purging cell gas to CAPS when required. Part of the hydrogen-argon mixture will react in the CAPS catalytic oxider and be converted to water. The remaining hydrogen will be purged with CAPS-processed cell argon to the HVAC vent system.

59 9.1.4.10.1 Design Bases

Adequate shielding is provided in the FHC containment structure for
9 radiation protection outside the cell, to meet the requirements and radiation 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 criteria of 10CFR100.

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

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

The spent fuel transfer station within the FHC is designed so that
44 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.

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

59 9.1.4.10.2 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
49 floor are shielded by conventional concrete to protect the neighboring 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.

The spent fuel transfer station within the FHC is shown in Fig-
44 ure 9.1-8. A maximum of 3 spent fuel assemblies in CCP's can be stored in this interim storage; however, in normal operation, a maximum of 2 fuel assemblies will be stored (1 storage position left empty). The storage positions within the transfer station consist of short, tapered, cylindrical inserts at the bottom, the middle, and the top of a substantial supporting steel grid structure. Each CCP is held in place by the three cylindrical sections. The center-to-center distance of the three storage positions is about 25 in. Each position can hold only one spent fuel assembly in a CCP.
44

During loading of the spent fuel shipping cask (SFSC), a cask-FHC seal assembly forms a gas-tight extension of the FHC containment to the cask interior. The SFSC is, therefore, connected to the FHC atmosphere and is separated from the air atmosphere of the cask corridor. 25

44

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

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

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

The spent fuel transfer station occupies an approximately 68-in.
square corner of the larger FHC maintenance pit. The pit covers an open
area approximately 9.5-ft by 12-ft on the FHC floor, and extends 18 feet
44 deep below the FHC floor. The entire pit is lined with a stainless steel
liner.

The design and construction of the spent fuel transfer station are
in accordance with codes, standards, and specifications listed in Section 3.8
44 for Seismic Category I structures.

Sodium leak detectors monitor the space below the spent fuel
44 transfer station. An argon gas radio activity monitor is provided. An area
monitor measures gamma radiation activity on the RSB operating floor above
the FHC.

59 59 9.1.4.10.3 <u>Safety Evaluation</u>

Adequate shielding for radiation protection on the operating floor
of the RSB, in the operating gallery, and below the FHC floor in the Spent
Fuel Shipping Cask (SFSC) handling corridor are provided by the FHC design.
49 Overall dose rates in these locations are less than 0.2 mrem/hr., permitting
routinely occupied area access (see radiation zones in Section 12.1).
44

Transient dose rates due to high-powered spent fuel assemblies are
49 less than 200 mrem/hr at the surface of the fuel transfer port in the FHC
roof plug. The statements made in 9.1.2.1.3 regarding the acceptability of
integrated doses resulting from the allowed transient dose rates are also
applicable for the fuel transfer port in the FHC roof plug.

The shielding provided to limit streaming dose rates from gaps around
the fuel transfer port plugs in the FHC roof and floor reduce the integ-
rated dose rate to 2 mrem/hr, in accordance with the shielding criteria of
4420 12.1.2.1. FHC shielding is described in Section 12.1.

The FHC has adequate seals and liners to prevent excessive radioactive emission into other areas of the RSB. Section 15.5.2.4 analyzes a limiting case and shows it to be acceptable. Radioactive releases of fission gas due to diffusion through elastomeric seals do not exceed the criteria of Sections 12.1.1 and 12.1.2.

The small number of fuel assemblies which can be present at any
49 time in the FHC (3 at the most) and the large separation distance of fuel
assemblies in the spent fuel transfer station ensures subcriticality in
the FHC at all conditions. The penetration of assembly positions through
the top grid plate preclude any other spent fuel assembly insertion into
59 44 the spent fuel transfer station, and the position's diameter
permits only one fuel assembly per position. More than 30 closely
spaced fuel assemblies are required in order to approach criticality,
as shown in Figure 4.3-30.

Seismic restraints of the spent fuel transfer station structure prevent any shifting within the vault during an SSE seismic event. Seismic forces will not change the separation distance between spent fuel assemblies due to the gridwork restraints between storage positions. The design prevents movement of the lazy susan sufficient to cause failure of the CCP or
44 damage a spent fuel assembly from either a seismic event or from inadvertent rotation.

44

The RSB design, elevation of the operating floor, and the welded steel liner of the FHC prevent the entry of flood or rainwater into the FHC. No significant quantities of moderating fluids are used in either the FHC or its argon cooling system.

44

In spite of the design consideration (see Section 9.1.2.2.2) a hypothetical heavy weight due to maintenance operations has been postulated to
44 drop on the FHC roof closure. The load limit of the RSB bridge crane auxiliary hook was selected as the hypothetical weight for this "umbrella" event. A drop height of two feet above the FHC roof closure was assumed. (The FHC roof closure is at the same elevation as the operating floor.)

A stress analysis of the composite roof closure structure was performed to determine maximum deflections and stresses due to the postulated impact load. The analysis was based on the following ground rules:

44 1) The load drops onto the center of the larger of the two closure plugs; i.e., the north closure plug. Deflections and stresses in the north closure plug would be larger than those in the
44 smaller, stiffer, south closure plug.

2) Conventional formulas for a peripherally supported rectangular plate under concentrated load were used to determine elastic deflection and spring constant of the reinforced concrete slab which forms the upper part of the closure plug.

3) Conventional formulas for end supported beams were used to determine elastic deflection and spring constant of the steel shield plates underlying the reinforced concrete slab.

4) The deformations of the reinforced concrete slab and the underlying steel plates were equalized, since it is a composite structure.

The results of these analyses are as follows:

The combined concrete slab and steel shield plates deform plastically due to the impact load. Based on effective elasto-plastic load versus deflection curves for reinforced concrete and ASTM-A.283GRD steel, the impact energy is absorbed by a 2.15-inch deflection of the composite structure. 20

Fictitious elastic impulse stresses were calculated for the reinforced concrete slab and related to qualitative degrees of concrete damage, using an elastic design index. The relation between elastic design index and degree of damage is based on test data with 104 reinforced concrete beams of varying reinforcement and varying impact velocities. The damage factor scale ranges from 0 to 5, and is defined in Table 9.1-3.

The postulated impact load resulted in a damage factor of 4 for the reinforced concrete. This implies heavy damage but no gross structural failure of the reinforced concrete.

The deflection of the steel shield plates(underlying the reinforced
44 concrete slab)due to the postulated impact load results in a ductility factor of 2.1. The ductility factor is defined as ratio of maximum dynamic deflection to deflection at the effective elastic limit, and is a measure of the degree of plastic deformation. Reference 2 reports that ductile structures can withstand impact loads with ductility factors in the order of 20 to 30 without failure. A ductility factor of 2.1 means that about one-half of the deflection due to the impact load is in the elastic range, the other half is in the plastic range. This implies some permanent deformation but no loss of structural integrity of the steel shield plates.

From the above considerations, it was concluded that the accidental drop of the heaviest object carried over the FHC roof could not lead to a gross structural failure of the roof plug, and could, therefore, not lead
44 to a change in lattice spacing of the fuel storage positions.

Such a drop load could, however, lead to a failure of the FHC roof closure seal. This could result in some air intrusion into the FHC since the
44 cell is at a slight negative pressure. After pressure equalization, radioactive argon gas could escape from the FHC into the RSB. The radioactivity of the FHC atmosphere is continuously monitored and controlled to a low level,
44 as specified in PSAR Section 16.3.10.3.1. A complete, instantaneous release of all FHC argon gas into the RSB operating area would result in a site boundary dose less than the guideline values of 10CFR20.

No safety consequences would ensue if the postulated impact accident were to occur while a spent fuel assembly was being handled by the FHC crane with cooling grapple. Both the grapple and crane are designed according to Seismic Category I requirements. The FHC crane bridge and trolley have seismic restraints which prevent their "jumping off" from the track due to seismic or transmitted impact loads. A spent fuel assembly held by the cooling grapple will not drop or jump off the grapple during the postulated impact accident.

59 20

Amend. 59
Dec. 1980

References

59

1. "Design of Structures to Resist Nuclear Weapon Effects," ASCE Manual of Engineering Practice No. 42, 1964 Edition. 20

44 2. Safety Analysis Report for LMFBR Spent Fuel Shipping Cask Model I-(182)-1, Aerojet AMCO Report AMCO-02-R-107, as revised.

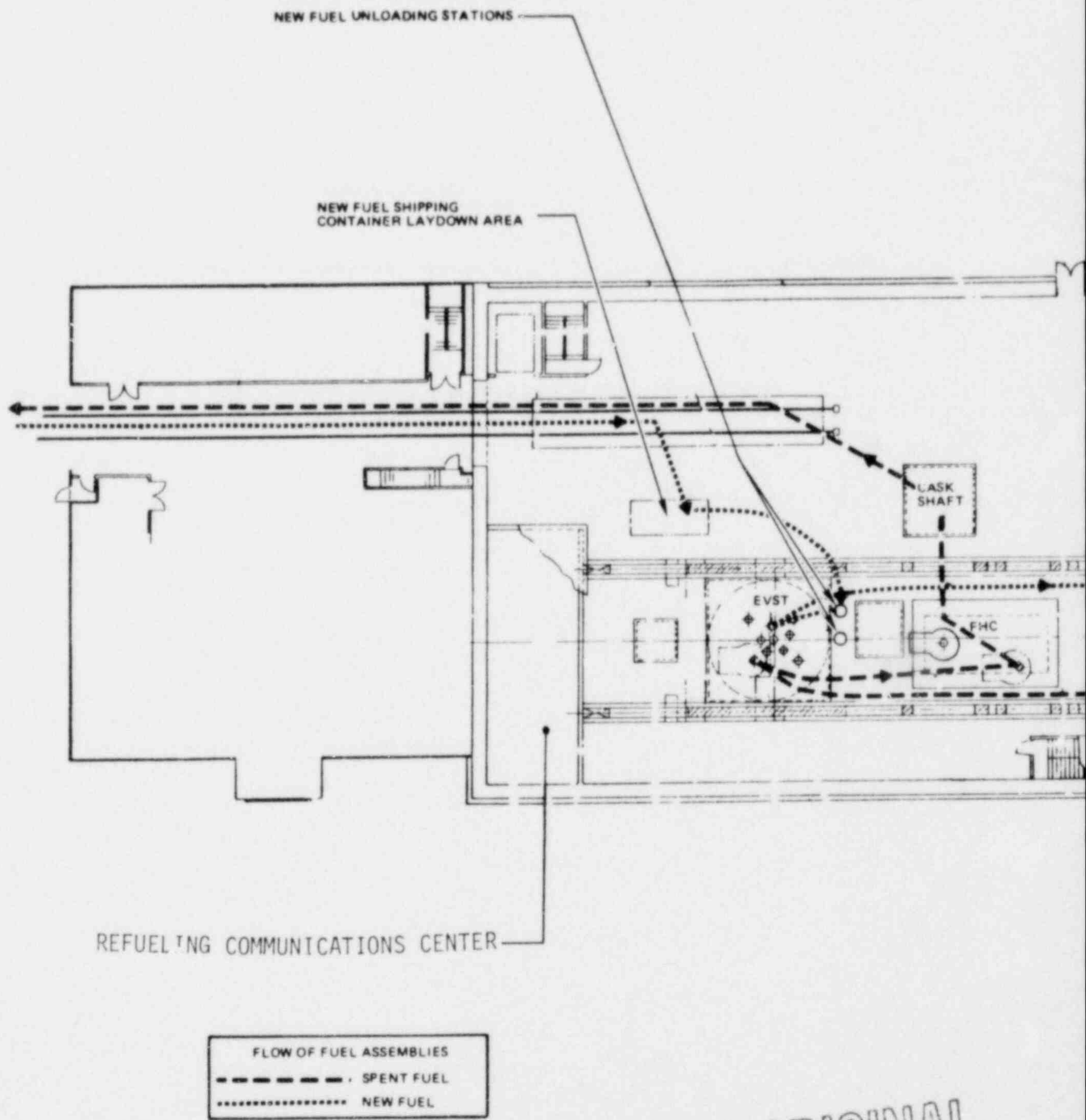
TABLE 9.1-1

44 EVST DECAY HEAT LOADS AND SODIUM OUTLET TEMPERATURES

	Condition	Heat Load (kw)	Sodium Coolant Outlet Temperature (^{o}F)*
	Design		
44 20	Maximum load	1800	$\sim$510
46 44 20			
	Emergency		
	Both normal loops failed	1800	< 775

44 *Sodium inlet temperature for all conditions is 400^{o}F

Amend. 46
Aug. 1978



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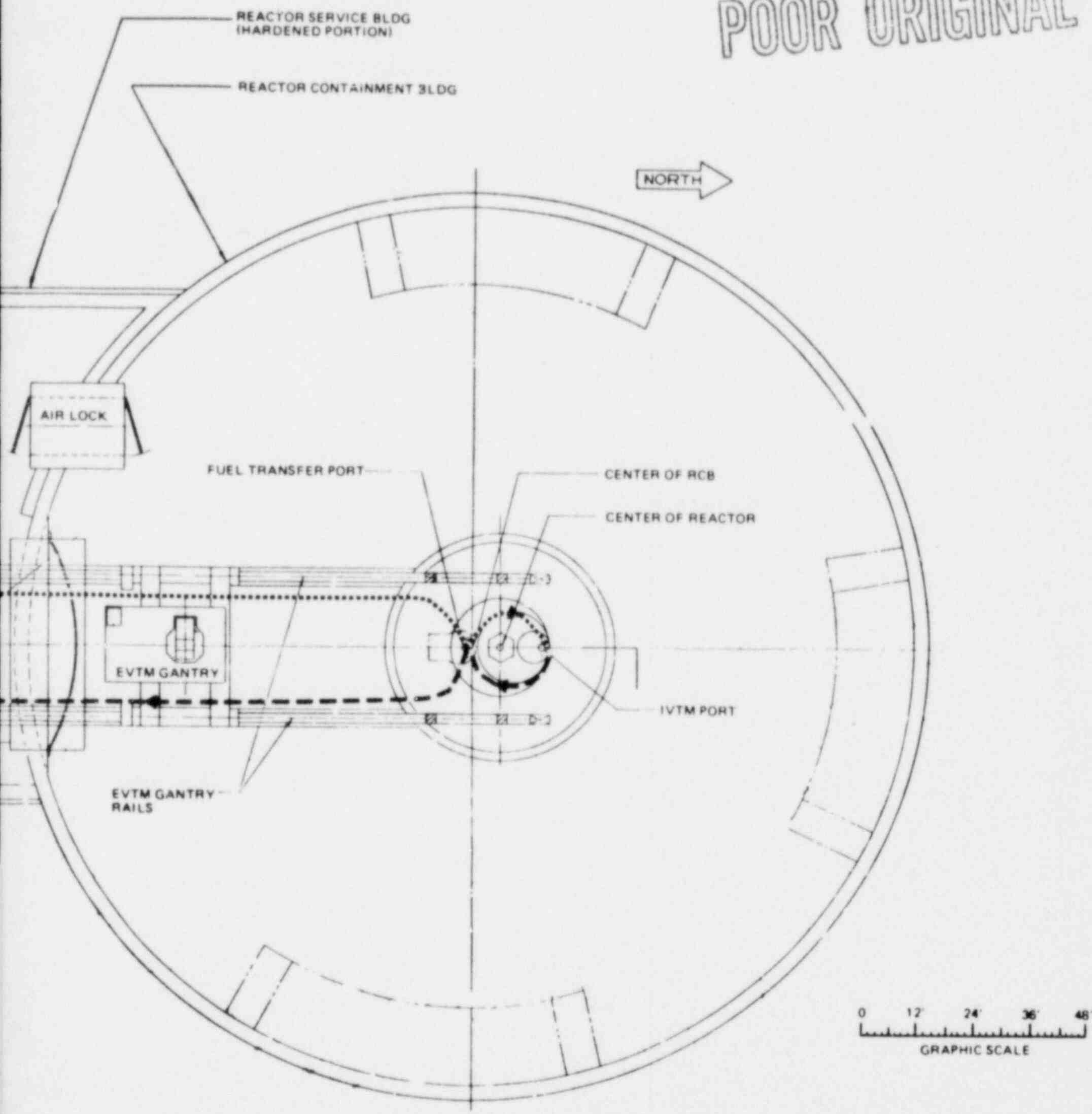

Figure 9.1-1 General Arrangement Plan of Fuel Storage/Handling Equipment and Facilities

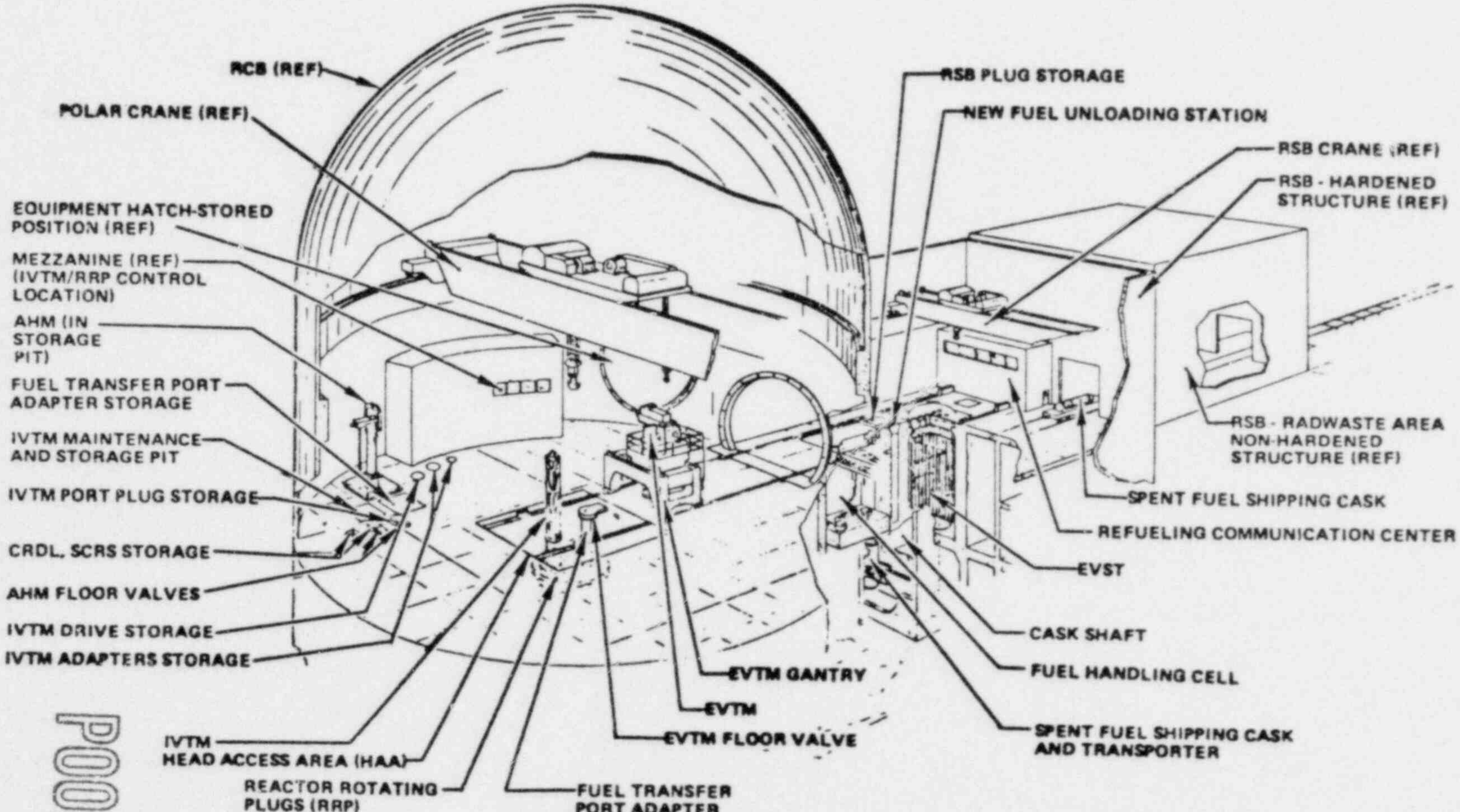

FIGURE 9.1-2 Arrangement of Fuel Storage/Handling Equipment and Facilities (Perspective View)

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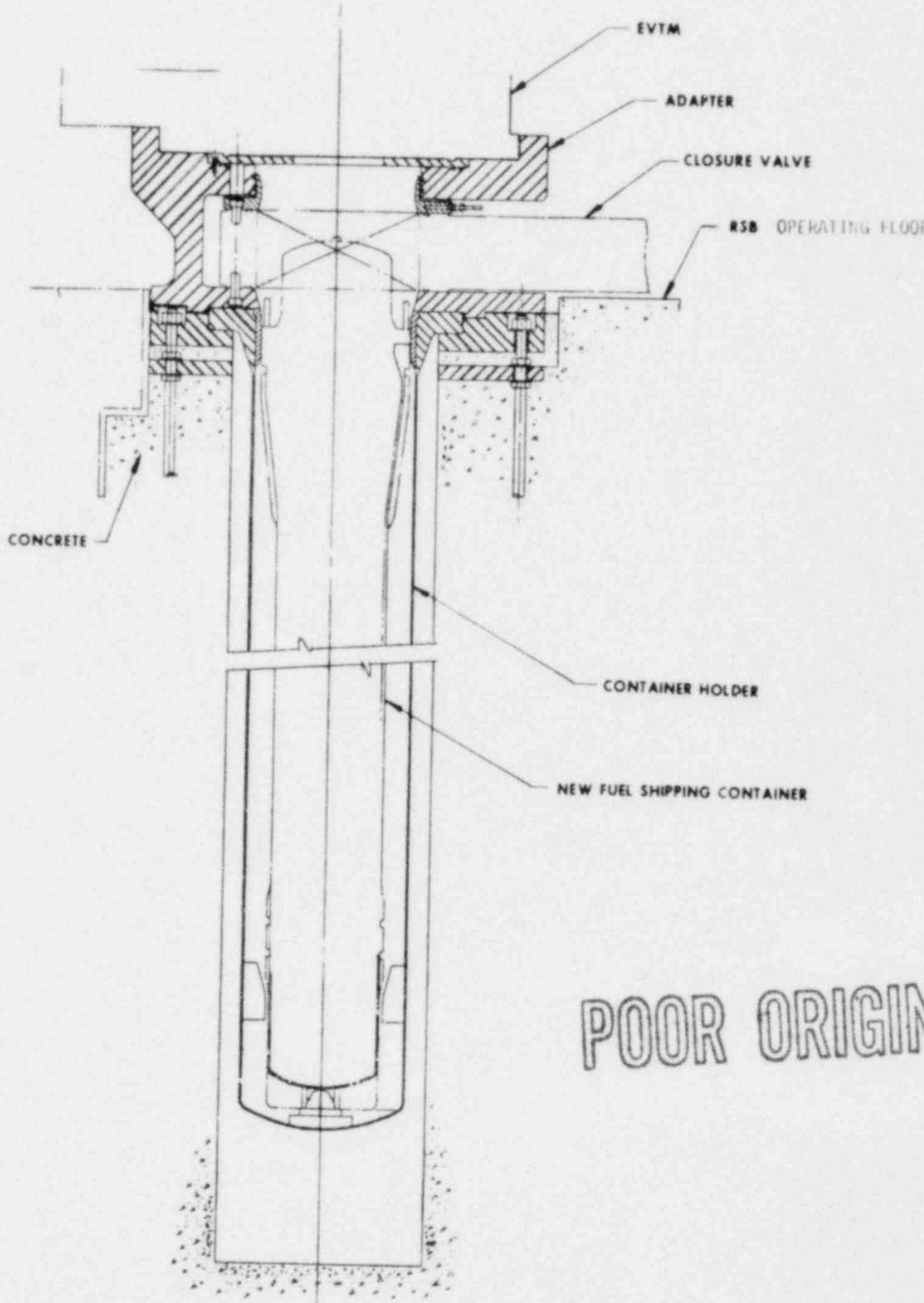

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FIGURE 9.1-3. NEW FUEL UNLOADING STATION

Amend. 59
Dec. 1980

44 Figure 9.1-4

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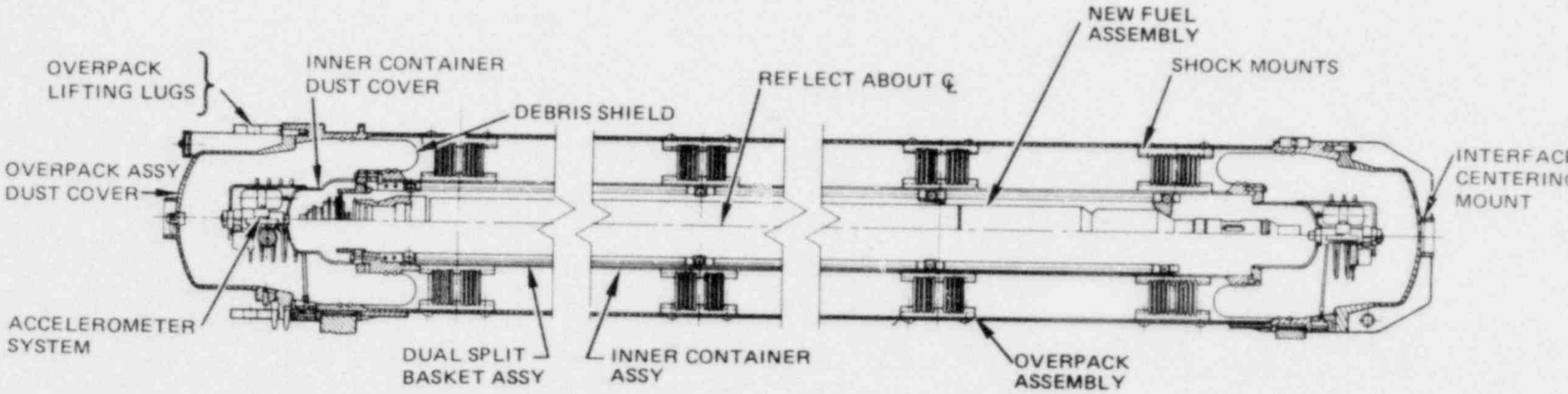

FIGURE 9.1-5 New Fuel Shipping Container Concept

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Dec. 1980

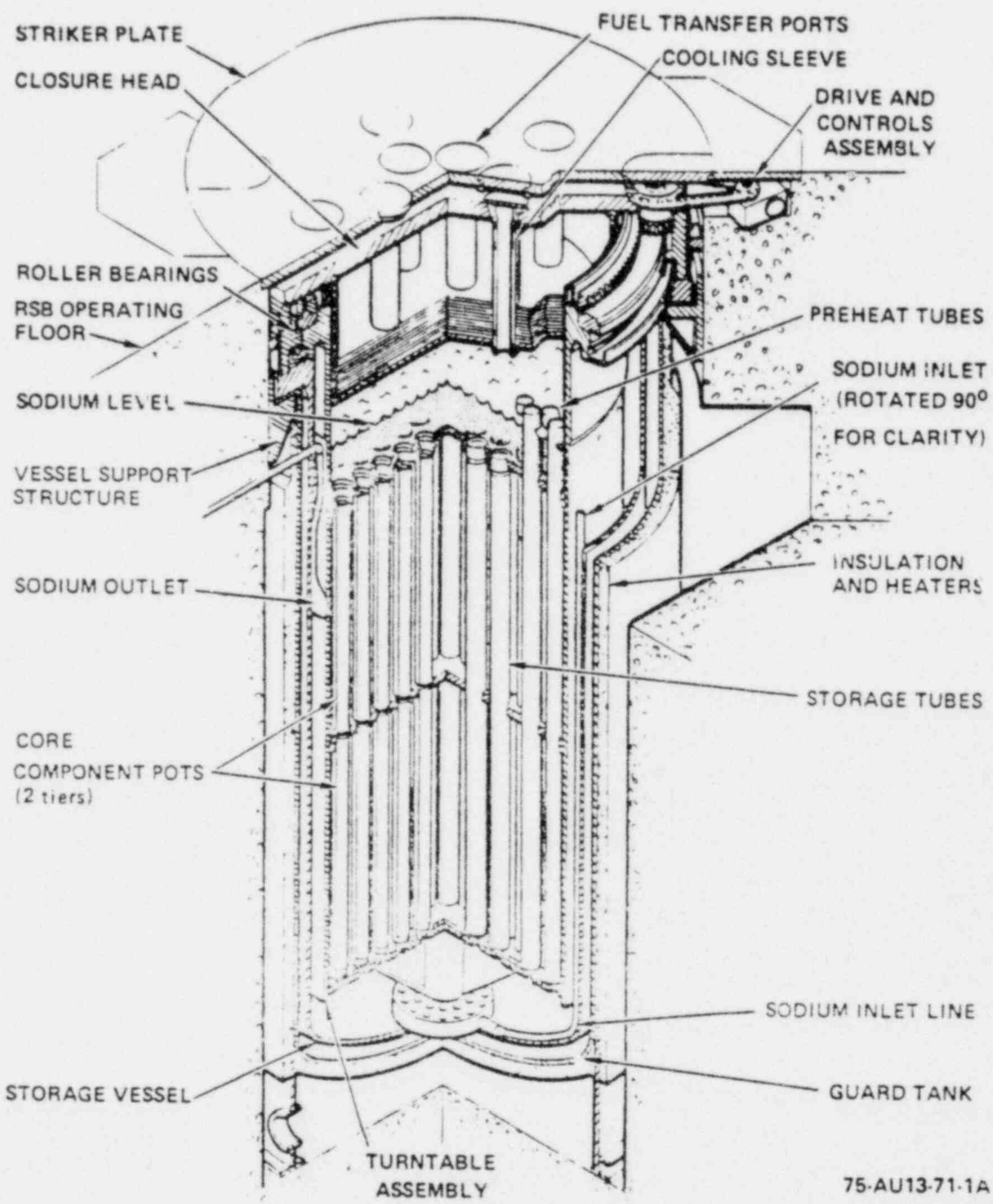

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Figure 9.1-6 Ex-Vessel Storage Tank (EVST)

FUEL HANDLING CELL (FHC)

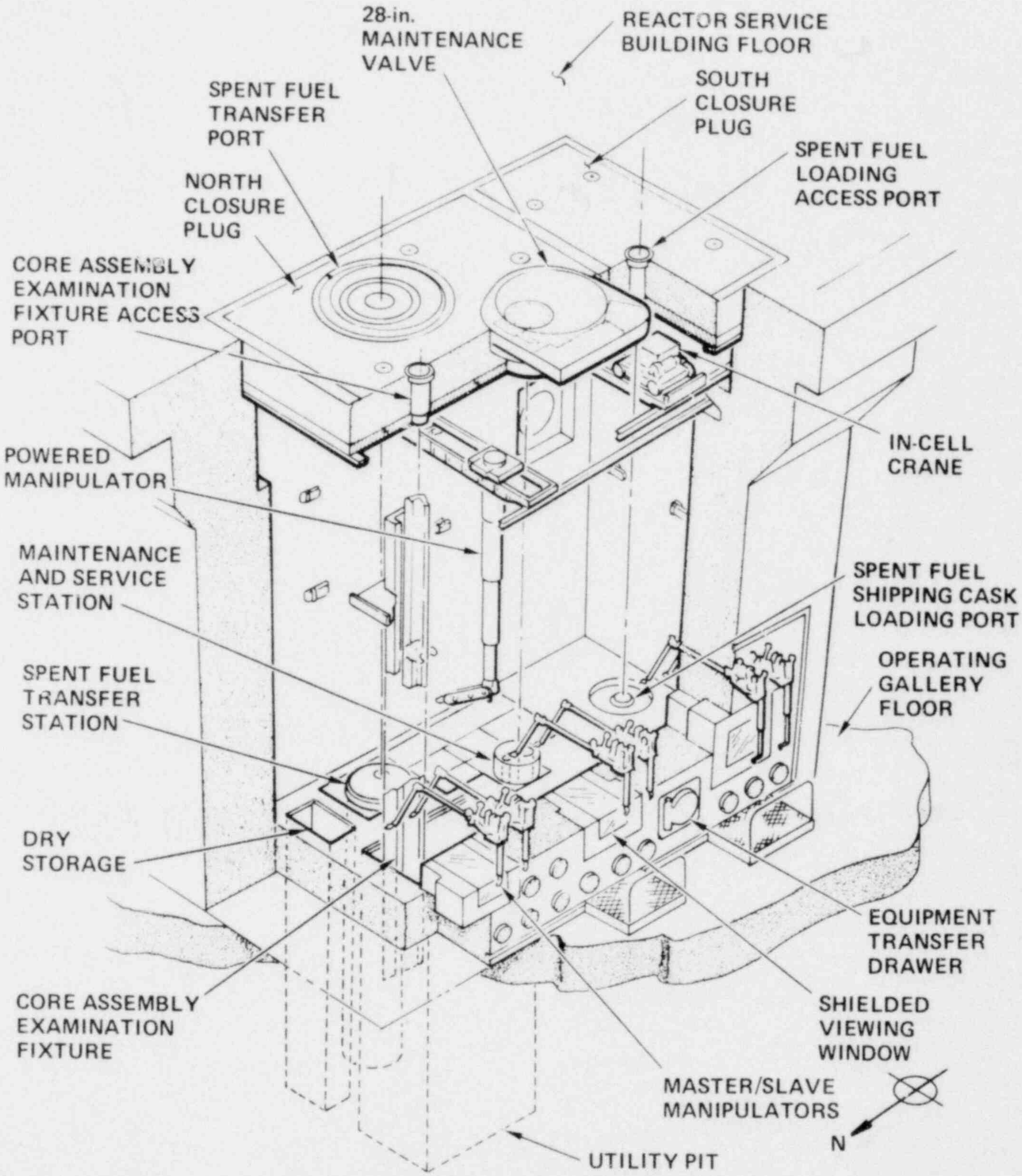

FIGURE 9.1-7 Fuel Handling Cell

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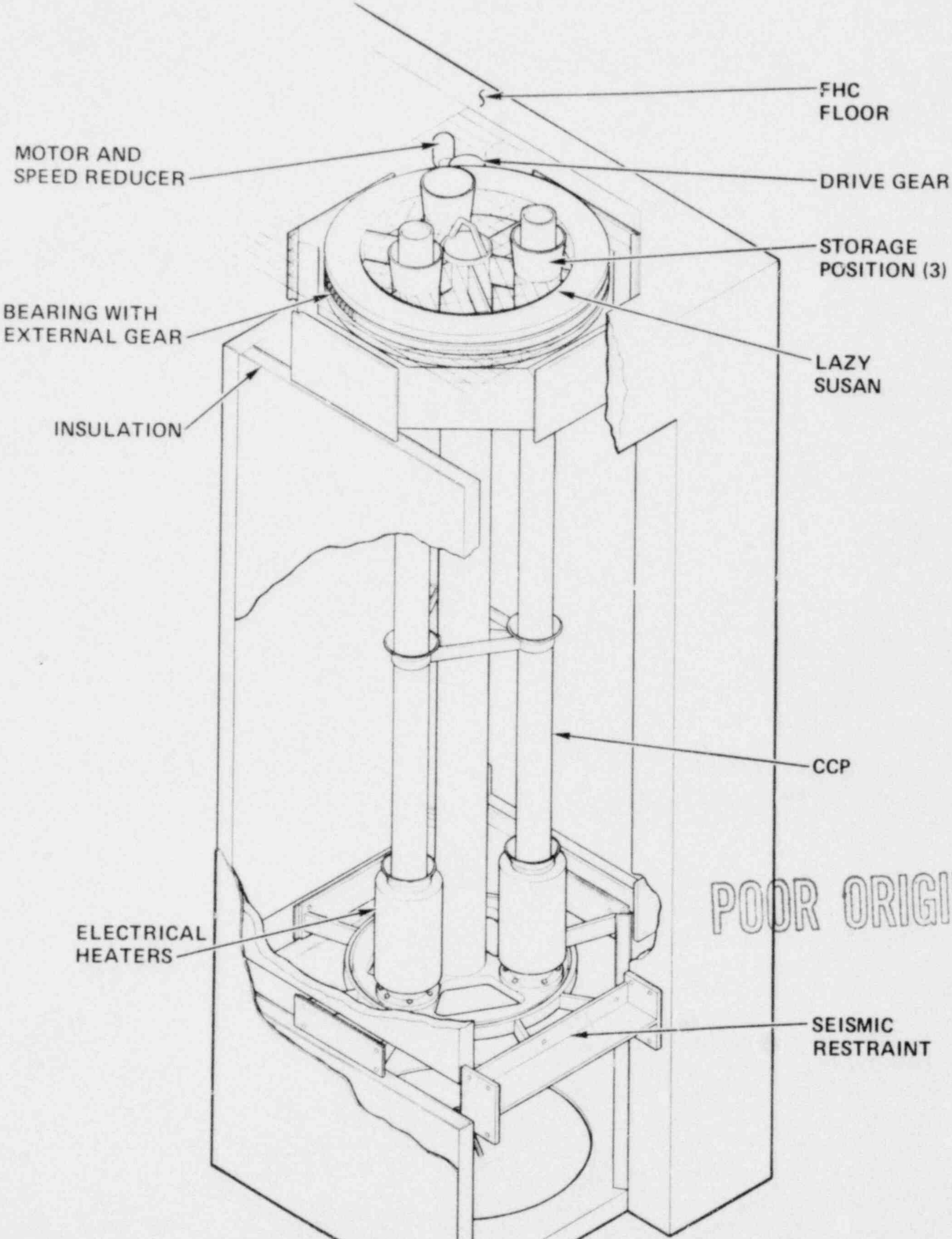

Amend. 44
April 1978

FIGURE 9.1-8. FUEL HANDLING CELL SPENT FUEL TRANSFER STATION

FUEL HANDLING CELL (FHC)

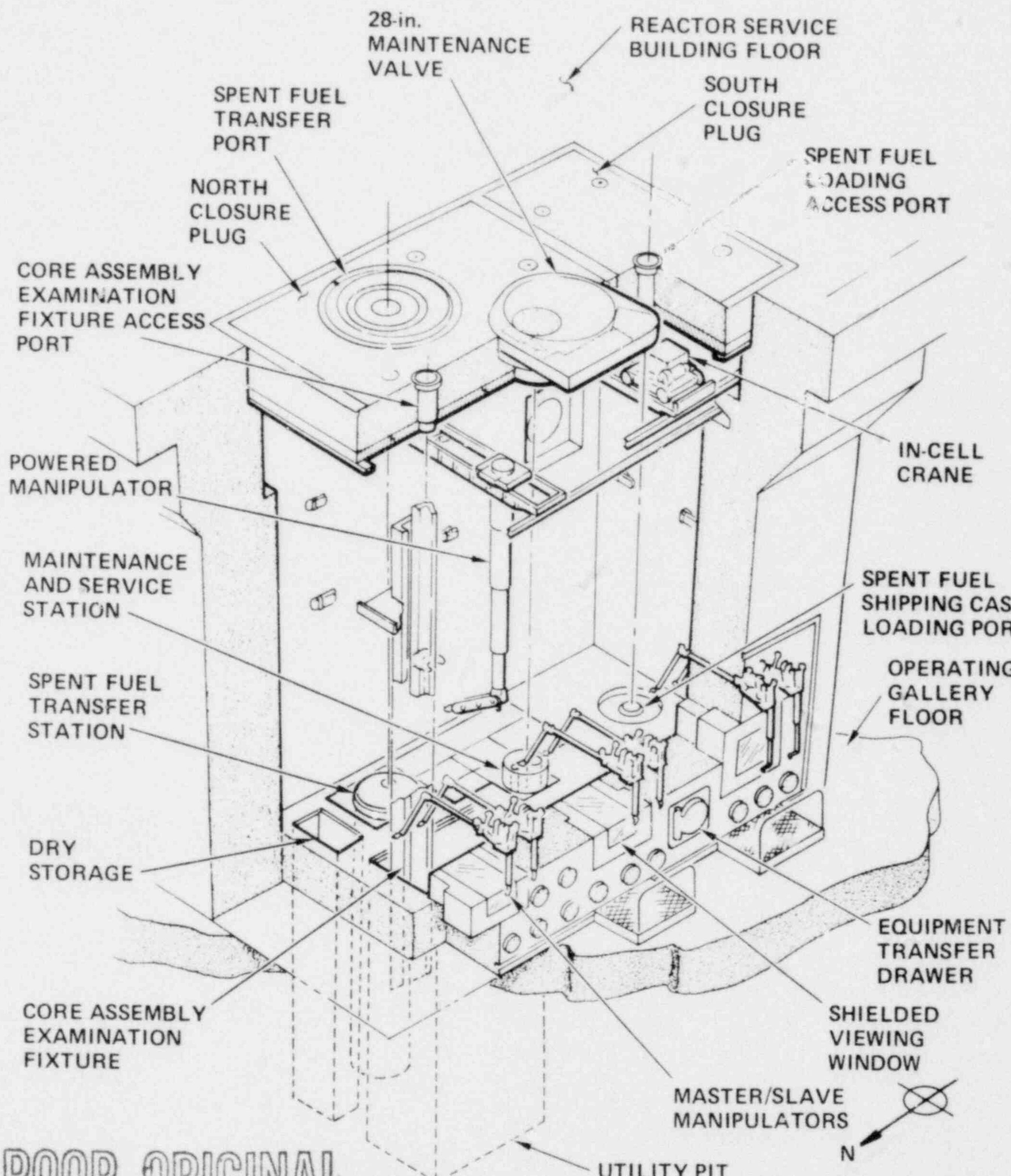

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FIGURE 9.1-7 Fuel Handling Cell

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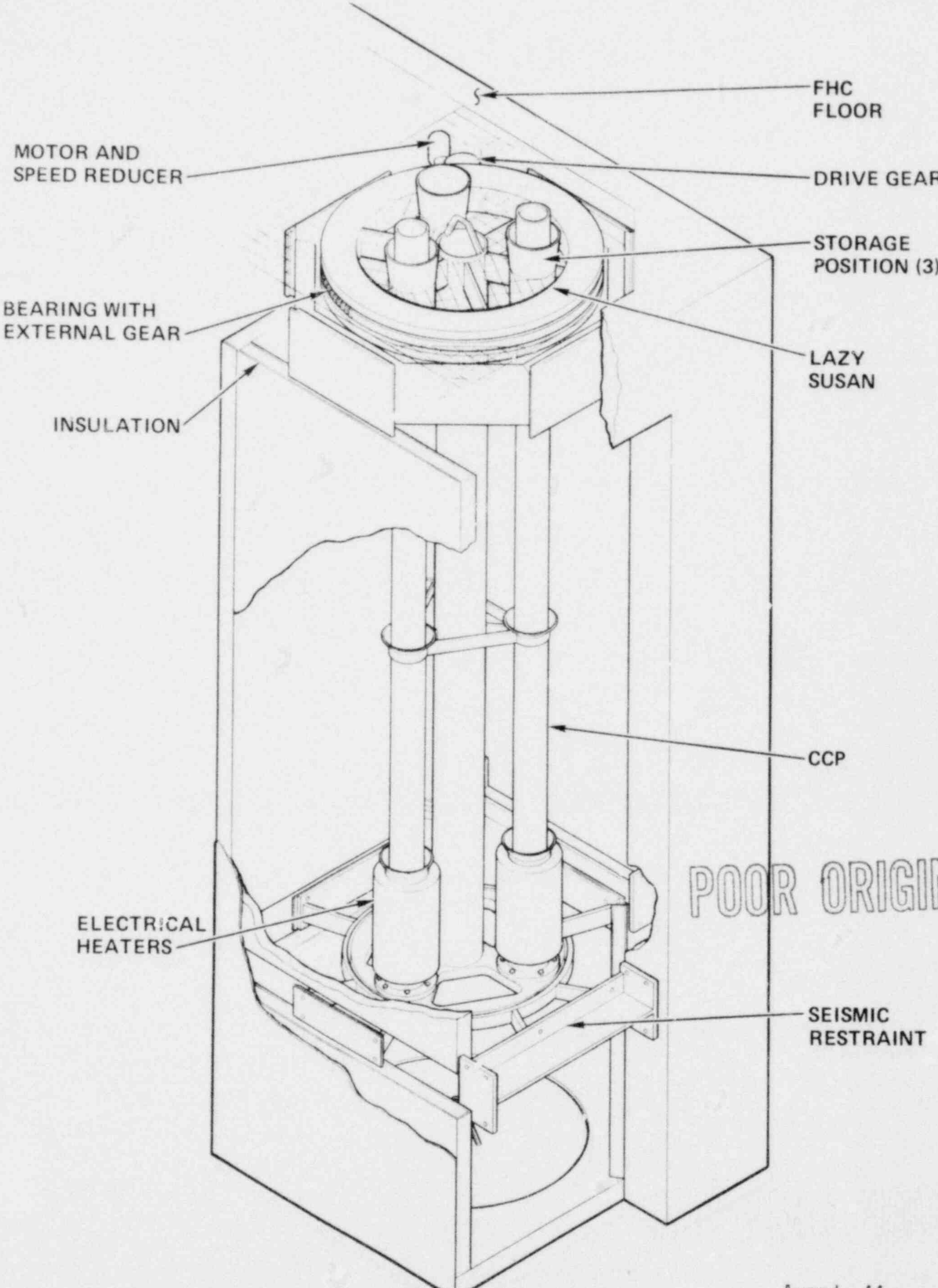

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Amend. 44
April 1978

FIGURE 9.1-8. FUEL HANDLING CELL SPENT FUEL TRANSFER STATION

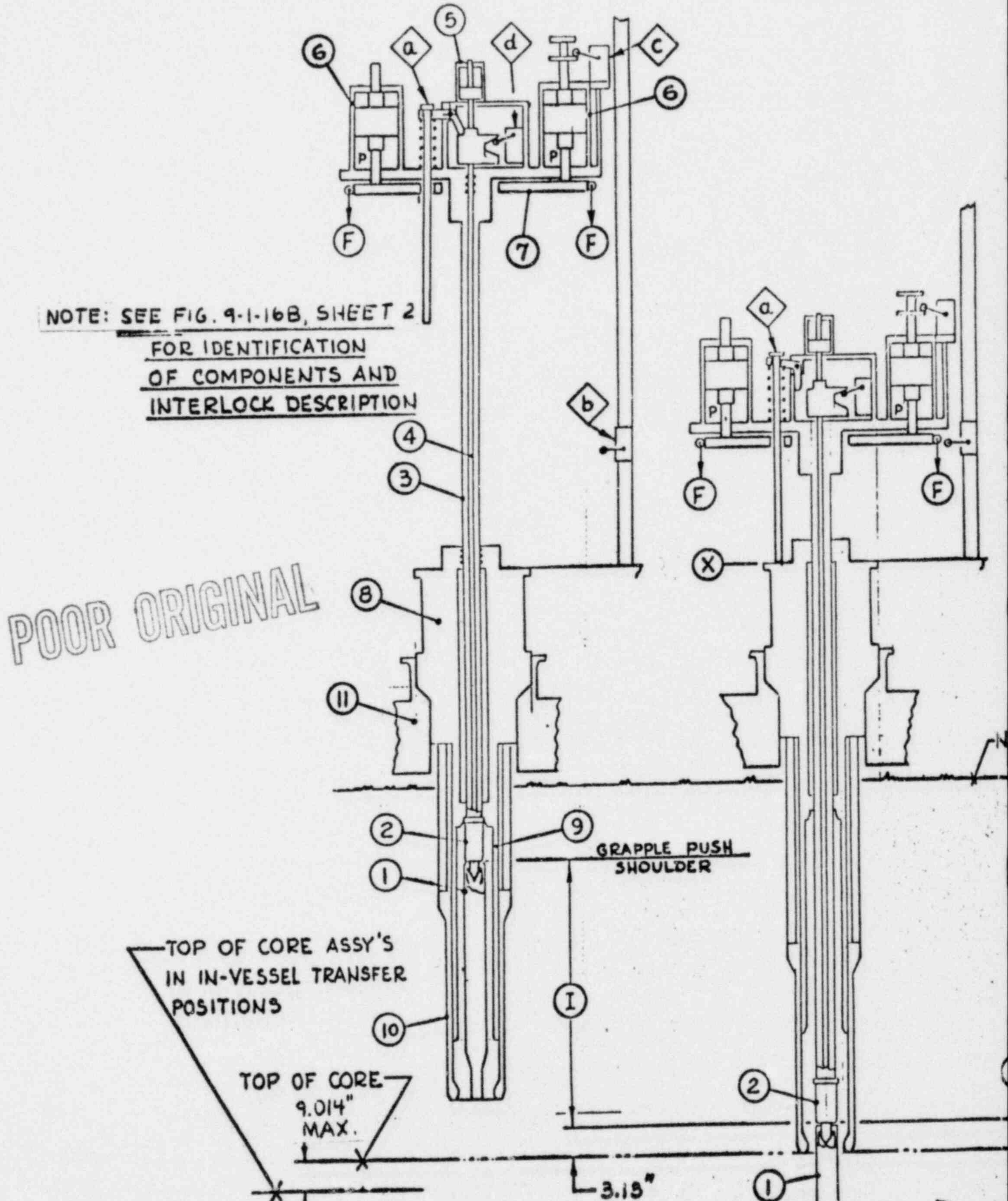

NOTE: SEE FIG. 9-1-16B, SHEET 2
FOR IDENTIFICATION
OF COMPONENTS AND
INTERLOCK DESCRIPTION
GRAPPLE PUSH
SHOULDER
TOP OF CORE ASSY'S
IN IN-VESSEL TRANSFER
POSITIONS
TOP OF CORE
9.014"
MAX.
3.15"
a
b
c
d
F
X
P
I
1
2
3
4
5
6
7
8
9
10
11

FIG. A

FIG. B

Figure 9.1-16B. Schematic of IVTM Interlocks
Over Core Positions

(SHEET 1 OF 2)

Fig. A. - IVTM shown in fully retracted position beginning to descend towards the core for core assembly insertion.
Fig. B. - IVTM shown at instant mech. interlock a has been relieved.
Fig. C. - IVTM shown at instant electrical interlock b has been relieved.
Fig. D. - IVTM shown in down position (core assembly fully seated in the core). Electrical interlock c has been relieved allowing grapple finger retraction. IVTM vertical motion is prevented unless electrical interlock d indicates grapple fingers have been retracted.

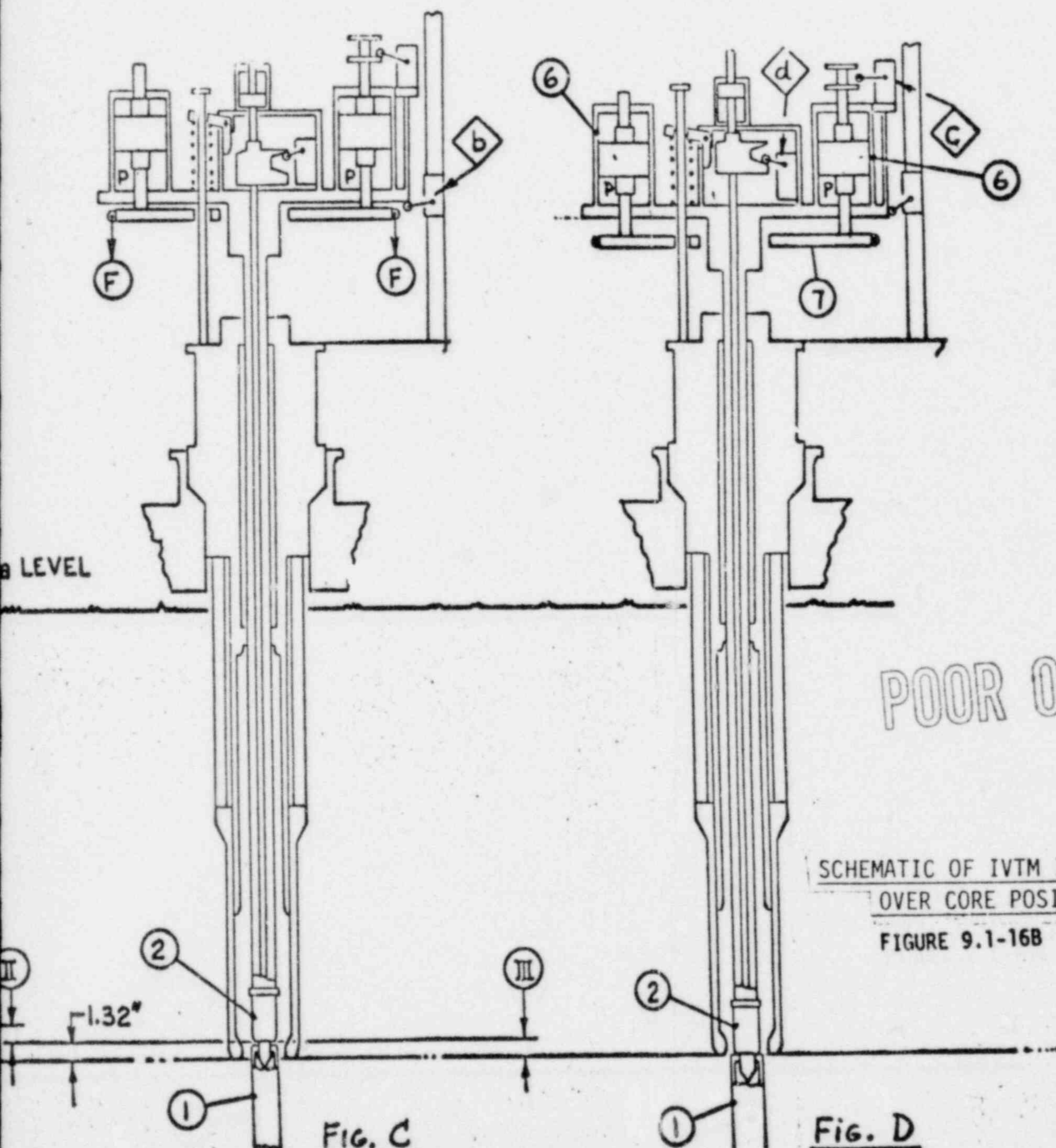
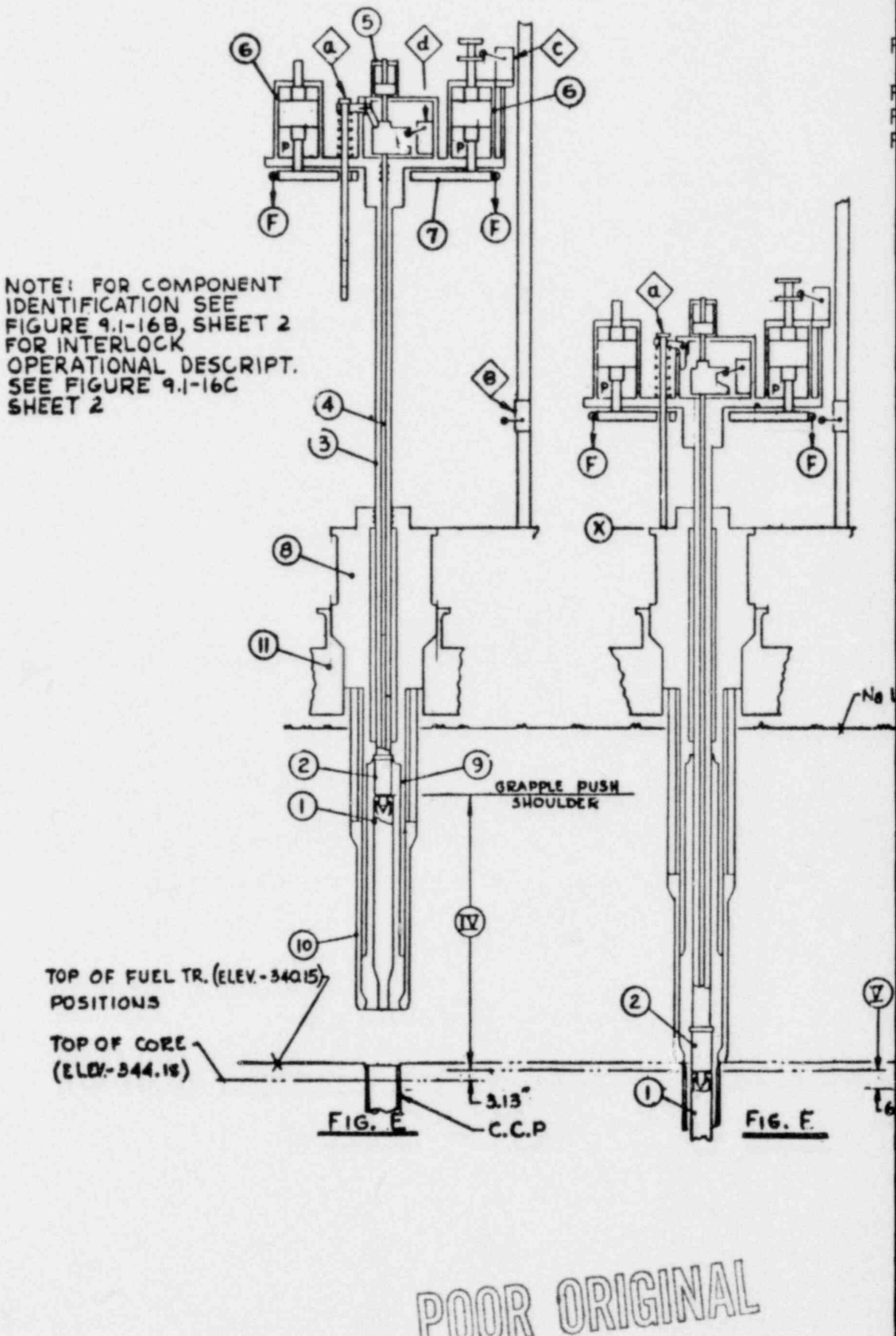


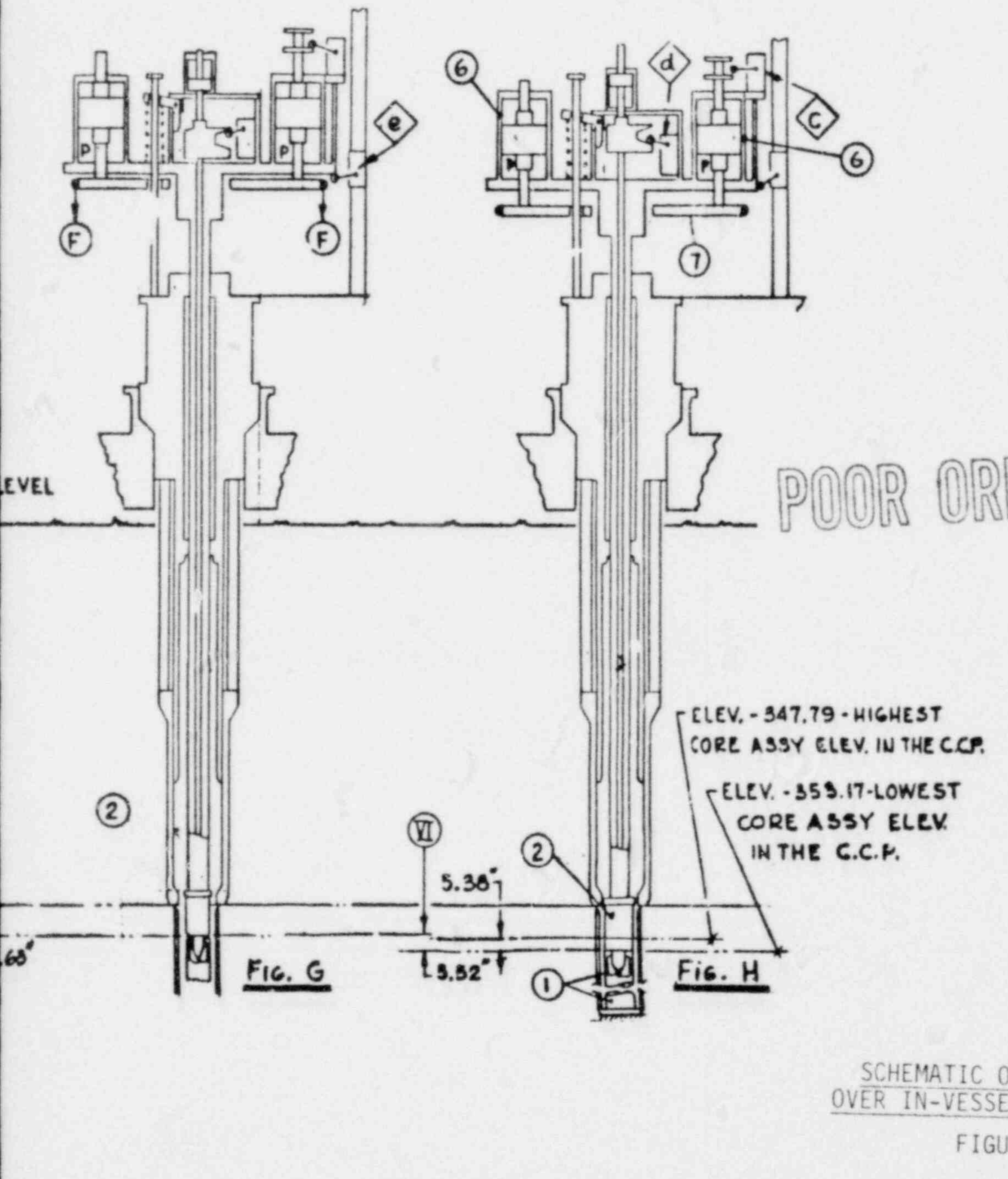
FIG. C

FIG. D

SCHEMATIC OF IVTM INTERLOCKS
OVER CORE POSITIONS
FIGURE 9.1-16B


NOTE: FOR COMPONENT IDENTIFICATION SEE FIGURE 9.1-16B, SHEET 2 FOR INTERLOCK OPERATIONAL DESCRIPT. SEE FIGURE 9.1-16C SHEET 2
GRAPPLE PUSH SHOULDER
TOP OF FUEL TR. (ELEV.-340.15) POSITIONS
TOP OF CORE (ELEV.-344.18)
3.13"
C.C.P
FIG. E
FIG. F

ig. E. - IVTM shown in fully retracted position beginning to descend towards the CCP for core assy insertion.
ig. F. - IVTM shown at instant mech. interlock ⟨a⟩ has been relieved.
ig. G. - IVTM shown at instant electrical interlock ⟨e⟩ has been relieved.
ig. H. - IVTM shown in down position (core assembly fully seated in the CCP). Electrical interlock ⟨c⟩ has been relieved allowing grapple finger retraction. IVTM vertical motion is prevented unless electrical interlock ⟨d⟩ indicates grapple fingers have been retracted.



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SCHEMATIC OF IVTM INTERLOCKS OVER IN-VESSEL STORAGE POSITIONS

FIGURE 9.1-16C

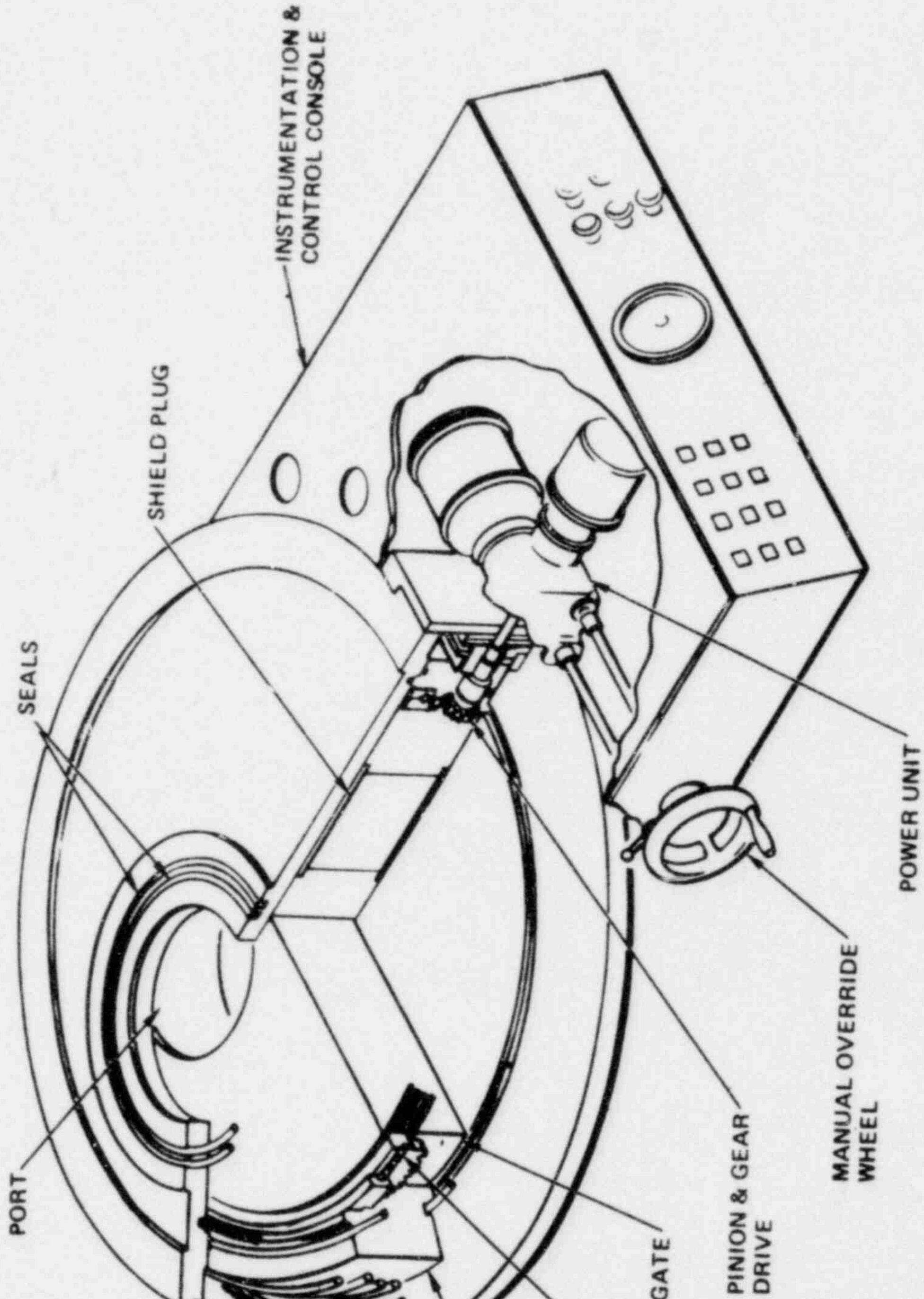

Figure 9.1-18 Floor Valve

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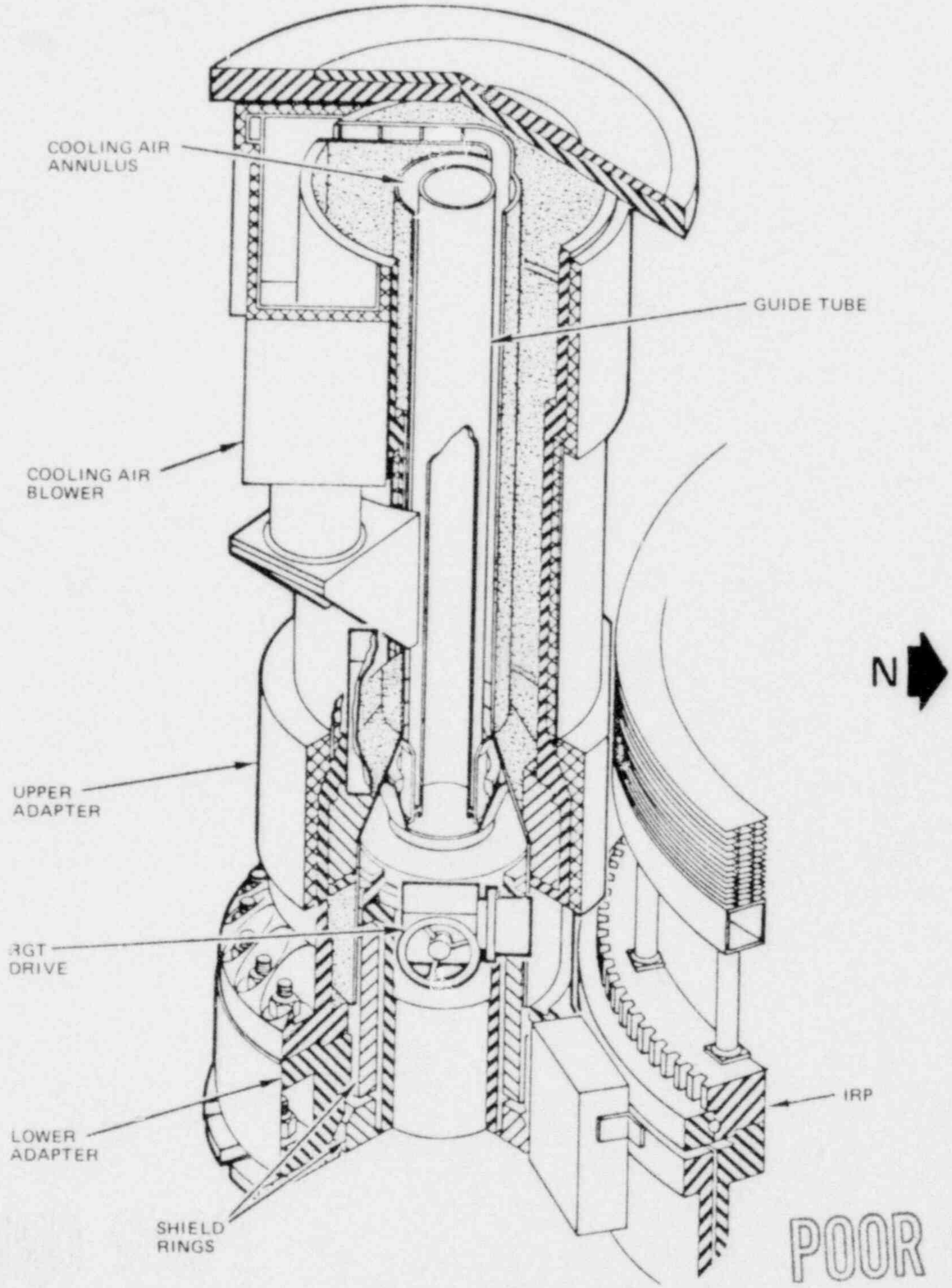

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FIGURE 9.1-19 RFTP Adapter Assembly

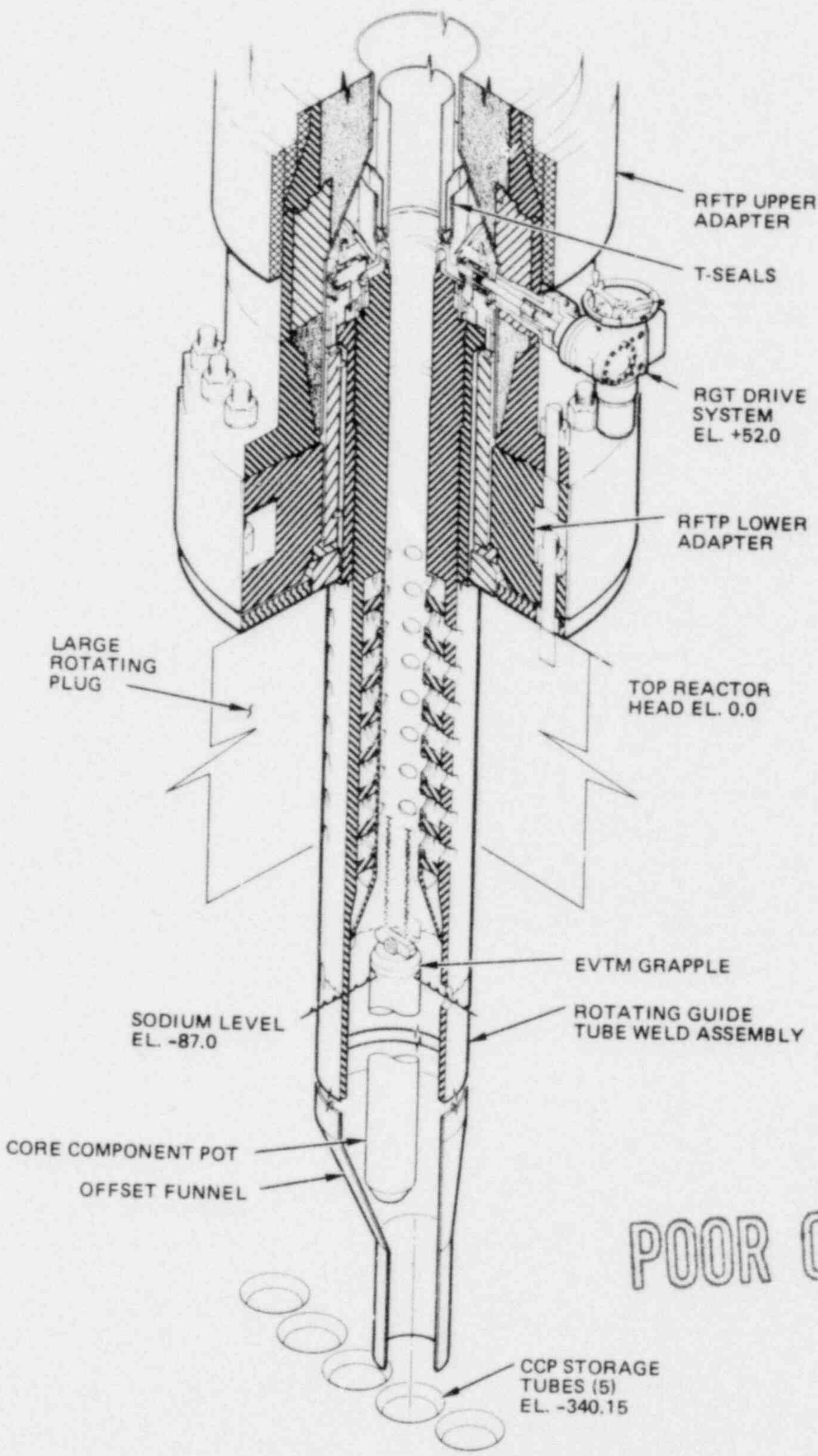

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FIGURE 9.1-20 Rotating Guide Tube

9.4 PIPING AND EQUIPMENT ELECTRICAL HEATING

9.4.1 Design Basis

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

Reactor Enclosure
Reactor Refueling (Storage Tank)
Reactor Heat Transport (Primary and Intermediate) Systems
47| Steam Generation System (Dump Tanks and Sodium Water Reaction Product Tanks)
Auxiliary Liquid Metal System
Inert Gas Receiving and Processing System
Sodium Impurity Monitoring System

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

To perform the dry heat-up function, the Electrical Heating System shall be capable of preheating the sodium process systems from ambient temperature (70°F) to any temperature between ambient and a maximum of approximately 450°F before the system is filled with sodium, at a rate determined by the particular sodium process system requirements.

The Electrical Heating System shall also be capable of providing the applicable heatup rate for the particular system or components when filled with sodium, and of holding process system temperatures when filled with sodium. Heat provided by this system can be used to melt frozen sodium in piping or components. Freezing of sodium in major systems or components is considered unlikely and is an abnormal event. Melting of frozen sodium is not safety related.

The heater physical mounting arrangement and the electrical protection of the heater circuitry shall be designed to preclude damage to the components being heated.

Heaters and the associated mounting hardware that are applied to components which are safety related shall be designed not to impair the ability of these components to perform their safety function during or after a design basis event. Those safety related components which require heaters are listed in Table 9.4-1.

9.4.2 System Description

The electrical heating and control system provides power to the butular heaters or minieral insulated (MI) heating cable mounted on the piping and/or components of the systems indicated in Section 9.4.1.

The heat rates required by different components are controlled by using thermocouples to monitor piping and component temperatures and to adjust the power supplied to the heaters, by means of 3 mode proportional temperature controllers and solid state relays.

Heat is applied by resistance type heaters consisting of nickel-chromium alloy resistance wire insulated from its containing metal tubular sheath by tightly packed Magnesia (MgO) powder. Several inches on each
end of tubular heaters are unheated, leaving a heavry low-resistance
59 conductor connected to the electrical termination.

47 The heaters will be stood off from the sodium containing metal
boundary for the safety-related piping and equipment which is listed in
Table 9.4-1. The heater sheath is not electrically insulated from the
59 metal boundary. For systems and components not listed in Table 9.4-1, the
47 heaters may be primarily standoff or applied directly in contact with the
46 sodium containing metal boundary dependent upon stress limitations. A
ground fault interrupt feature is provided to remove power within two
cycles to any heater which develops a short to ground. This value is
low enough to preclude any metal damage.

Chromel-alumel thermocouples are used throughout the systems for controlling the operation of the electric heaters and for monitoring the temperature of the metal boundary of the sodium containing piping and equipment. Thermocouple compensation is provided for all thermocouples.

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

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

The controller compares the temperature control setting (ramp
rate in heat-up mode and setpoint in hold mode) as set by the plant operator,
59 to the actual temperature of the sodium process metal, as measured by a

chromel-alumel thermocouple, and generates an error signal. The error signal is converted into a corresponding "on" to "off" ratio of output voltage which is applied to a solid state relay which controls the AC power to the heaters.

The required power is controlled by conducting a fraction of the cycles of the 60 hertz. For example, 50 percent power would be conducting every other cycle, 90 percent power would be omitting one cycle of every ten, 10 percent power would be conducting every tenth cycle only.

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

The temperature feedback thermocouple is located in a representative position within the pipe run or area within the heated zone.

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

Loss of off-site AC power is the only abnormal condition for electircal heating. It might, if of sufficient duration, and lacking other heat sources (e.g. decay heat) result in sodium freezing in pipes and components. Melting of frozen sodium will necessitate the sequential operation of heaters from free sodium surfaces in order to avoid undesirable pressure build-up in piping or components.

A dedicated, pre-programmed direct digital control system is provided, on an individual loop basis, for the Reactor Containment Building, Steam Generator Building, and Reactor Service Building. The system is modular, to permit physical distribution of the various functional components and to facilitate future expansion and upgrading.

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

9.4.3 Safety Evaluation 19

As defined in PSAR Section 3.2, the piping and equipment electrical heating and control system is classified as non-safety (NSC). The heating 6
system is not essential for the safe shutdown and isolation of the reactor,

nor will failure of the system result in a release of radioac .e material. In those cases where heaters are applied to safety related components, the heaters are not required for the component or the associated system to perform its safety function.

Heaters and supports affixed to Category I components are qualified to that category to assure physical integrity.

Heater failures originate at the heater element wire, promoting arcing which propagates outward to the heater sheath. This is the worst single failure mode for the System. The failure mechanisms that can cause this failure mode can be attributed to operational and design factors.

Operationally, the failure mechanism can be caused by (1) excess current application, (2) cross over in mounting of adjacent heaters, and (3) improper setting of protective devices. For the design factor, the failure mechanism can be caused by improper heating wire design, fissures in the magnesium oxide and minimum bend radii. The effect of the failure will not affect the system it is heating. 6

In the event a trace heating element in the HTS or auxiliary system fails, a thermal gradient inthe piping or component will be produced. This thermal gradient would then induce a thermal stress in the piping or component. The design process will evaluate these potential thermal stresses in order to identify trace heater locations so that a loss of one heater will not result in unacceptable stress levels. The design process will also identify the number of trace heaters that could fail before corrective action is required. The design evaluation will include dry heat up, heat up with sodium, and normal operation for both sodium containing and any normally dry portions of the system. 17

In order to prevent a heater failure from propogating to the piping or equipment to which it is attached, the following operational criteria are used:

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

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

(3) The cold ends of the heaters are bent 90° and a spacing maintained between adjacent heaters to prevent cross over of heaters and significant mutual heating by radiation.

(4) The proper setting of the GFI units will be set at installation and based on prior tests.

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

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

9.4.4 Tests and Inspections

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

9.4.5 Instrumentation Application

Instrumentation application is discussed in Section 9.4.2.

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*Redundant heaters are non-operating installed spares which duplicate the function of operating heaters.

Table 9.4-1

46 SAFETY-RELATED COMPONENTS REQUIRING STAND-OFF ELECTRIC HEATERS

Component

Reactor Guard Vessel

Ex-Vessel Storage Tank Guard Vessel

Primary Heat Transport System

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

Intermediate-Heat Transport System

Main Piping
Pumps
Expansion Tanks
Appendage Piping

Steam Generator System

Appendage Piping

Auxiliary Liquid Metal System

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

Inert Gas Processing System

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

Impurity Monitoring and Analysis System

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

Amend. 46
Aug. 1978

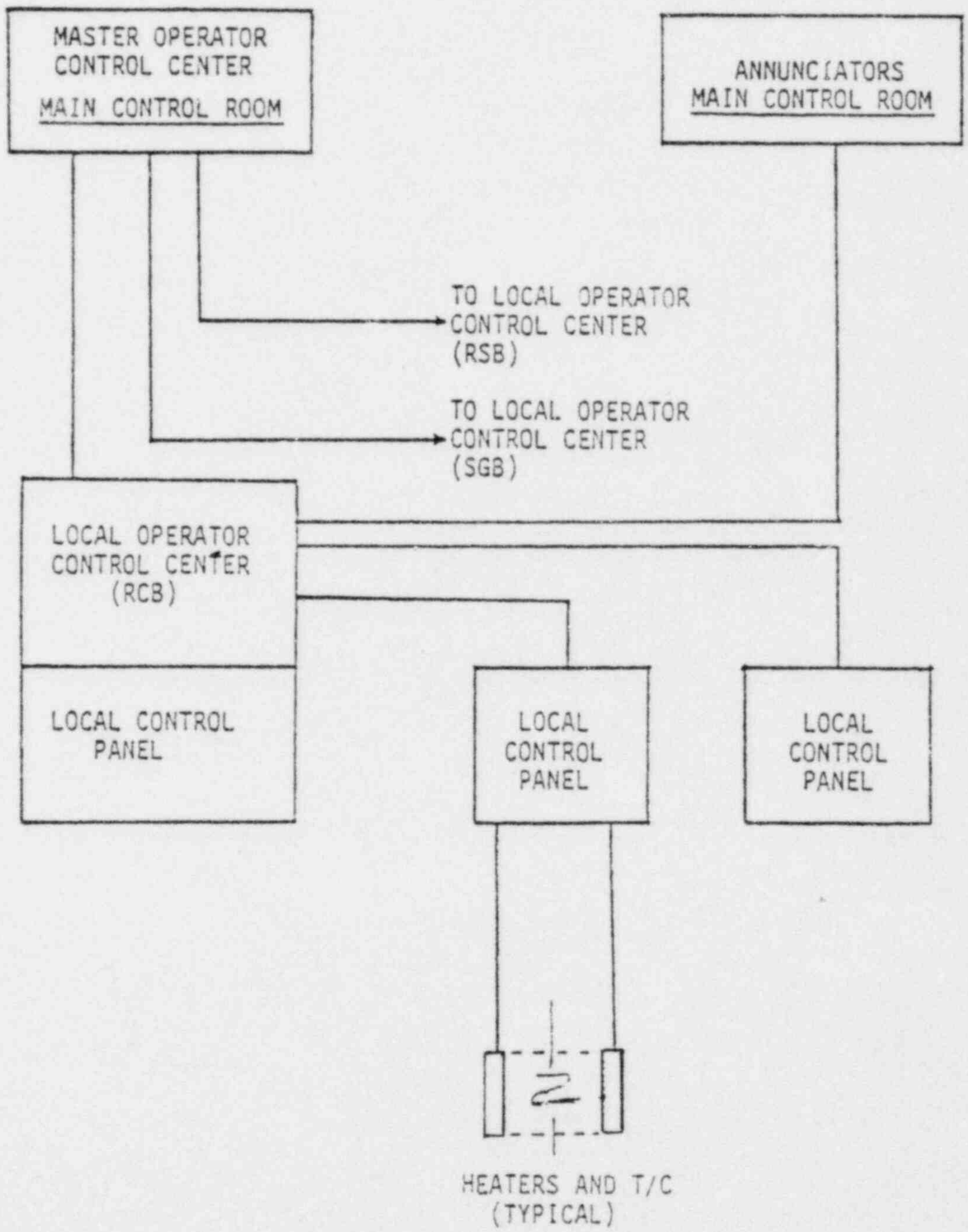


Figure 9.4-1
Power and Control System Arrangement

Amend. 59
Dec. 1980

9.5 INERT GAS RECEIVING AND PROCESSING SYSTEM

The Inert Gas Receiving and Processing (IGRP) System consists of the following four subsystems: (1) Argon Distribution Subsystem, (2) Nitrogen Distribution Subsystem, (3) Radioactive Argon Processing Subsystem (RAPS), and (4) Cell Atmosphere Processing Subsystem (CAPS).

The Argon Distribution Subsystem (Figures 9.5-1 through 9.5-3) provides cover
gas to all free liquid metal surfaces and to component and reactor head seals.
Argon is used initially to inert all the liquid metal vessels and RAPS and
maintains the inert atmosphere in the Fuel Handling Cell (FHC) during
59 operations.

The Nitrogen Distribution Subsystem (Figures 9.5-4 through 9.5-10) provides
inerting gas for cells containing primary sodium components, cover gas for
auxiliary coolant surfaces, inert gas for maintenance operations and gas for
59 operating pneumatic valve operators in inerted cells.

Figures 9.5-1 through 9.5-10 summarize the argon and nitrogen gas services provided by the IGRP System to interfacing systems.

The Inert Gas Receiving and Processing System has several vessels that contain
gases under pressure; these are listed in Table 9.5-2 which identifies their
names, the contained gas, the design, operating, maximum pressures, the
operating temperatures, the vessel volume, and the maximum available stored
energy (PV product) for the maximum pressure. Table 9.5-3 is a summary of the
59 locations of the vessels. The cell and building walls provide the required
protection of equipment essential for a safe reactor shutdown. Figure 9.5-11
shows the locations and arrangements of the equipment items located by the
item numbers in Table 9.5-3.

The Argon Distribution Subsystem also provides evacuation service to vessels and piping that are being filled with sodium or argon.

59 The atmosphere purification (APU) unit of the Argon Distribution Subsystem
removes water vapor and oxygen from the recirculated argon atmosphere of the
FHC and maintains these impurities within specified levels.

The RAPS subsystem processes primary heat transport system cover gas
(particularly reactor cover gas), removes radioactivity and provides a source
59 of purified gas for recycle back to the reactor and the primary sodium pumps.

The CAPS subsystem processes gas exhausted from the cell atmospheres and from
other locations within the reactor complex and ensures that effluent gases
released from the CRBRP have radioactivity levels that are as low as
48 reasonably achievable.

Both stainless steel and carbon steel are used in the IGRP System. All piping and components that are exposed to sodium vapor are fabricated from Type 304 or 316 stainless steel. All piping and components used in cryogenic services are made of Type 304 or other low-temperature alloy steel. The RAPS subsystem piping is made of Type 304 stainless steel. The remaining piping and components are made of carbon steel. All the gas vessels are to be made of carbon steel, as is much of the argon and nitrogen gas distribution subsystems.

After these vessels have been fabricated, their interiors are to be cleaned with abrasives and solvents to remove rust and scale. The tanks are then to be evacuated and back-filled with a dry inert gas. The tanks are to be maintained at this positive pressure until they are installed. Following installation and leak-testing of the weld joints, the tanks are again to be evacuated and filled with a dry inert gas.

The active valves are listed in Table 9.5-4. These valves must be operable during and after design events, such as Safe Shutdown Earthquake (SSE).

The following sections describe in detail the Argon Distribution Subsystem and Nitrogen Distribution Subsystem. The RAPS and CAPS subsystems are described in detail in Section 11.3.

9.5.1 Argon Distribution System

9.5.1.1 Design Basis

Argon is to be supplied for the liquid metal system cover gas spaces for purging, filling, and draining the liquid metal systems, for buffered and inflatable head seals, for the atmosphere in the Fuel Handling Cell, and for services connected with fuel handling, sampling, and maintenance operations.

The reactor closure head plug riser seals consists of a combination of seals
in series to limit radioactive cover gas leakage to the HAA. External to the
dip seals are the inflatable elastomer seals pressurized with fresh argon,
normally at 14 psig with buffer purge monitored to be within 0.05 to 0.1 scfm.
Elastomer and metallic "O" ring seals provide additional series seal
redundancy. Thus, the seal design basis shall not contribute more than one-
tenth the MPC_{40} (maximum permissible concentration for the first 40 hour work-
week) concentration in the HAA. The argon feed and bleed system will maintain
the reactor cover gas pressure at 6 ± 2 in. w.g. during steady-state reactor
51 operation and between 0 to 12 in. w.g. during normal reactor transients.

59 Fresh argon is used as required, for the PHTS and associated equipment
purging, filling, and draining services in the RCB and RSB and for the Fuel
59 48 Handling Cell argon supply to CAPS.

59 The use rate of RSB/RCB argon by these services is variable and is dependent
on operator options. Under start-up conditions, the flow will be maximum, and
59 a minimum automatic supply capability of 94,000 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 for
59 transfer operations and for loop pressure control during normal operations and
during all the postulated IHTS design events.

9.5.1.2 <u>Design Description</u>

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
59 distribute and vent the argon to meet the requirements described in Section
9.5.1.1.

9.5.1.2.1 <u>Recycle Argon Distribution</u>

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
59 of 6 in. w.g. by a constant purge plus a feed and bleed control system.

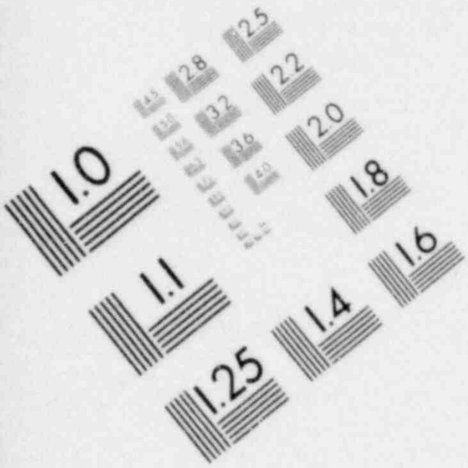
There is a continuous transfer of argon cover gas from the reactor and the
primary pumps via the equalization line to the primary sodium overflow vessel
and then through a 5-scfm vapor trap that removes sodium vapor. This vapor
trap consists of a vapor condenser and two parallel aerosol filters (one
59 redundant). A 1-scfm sample of cover gas is taken from the equalization line
and is passed through a 1-scfm sodium vapor trap to the Failed Fuel Monitoring
59 50 System. This gas and the reactor cover-gas purge gas and the primary pumps
59 purge gas flow through RAPS for processing.

9.5.1.2.2 <u>Fresh Argon Supply at RSB</u>

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 hand switch control override
allows drawing on both Dewars simultaneously. When the switchover takes
place, an alarm signals the operator and he is then required to initiate
48 action to fill the nearly empty dewar.

IMAGE EVALUATION
TEST TARGET (MT-3)

6″



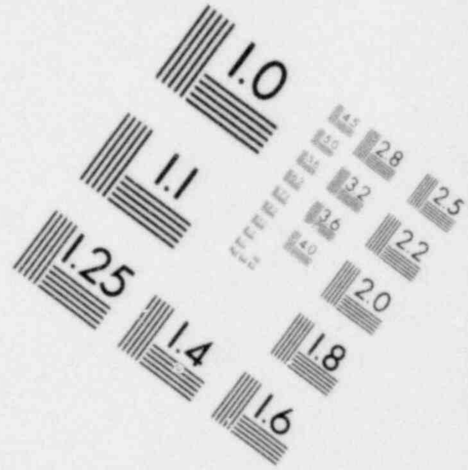
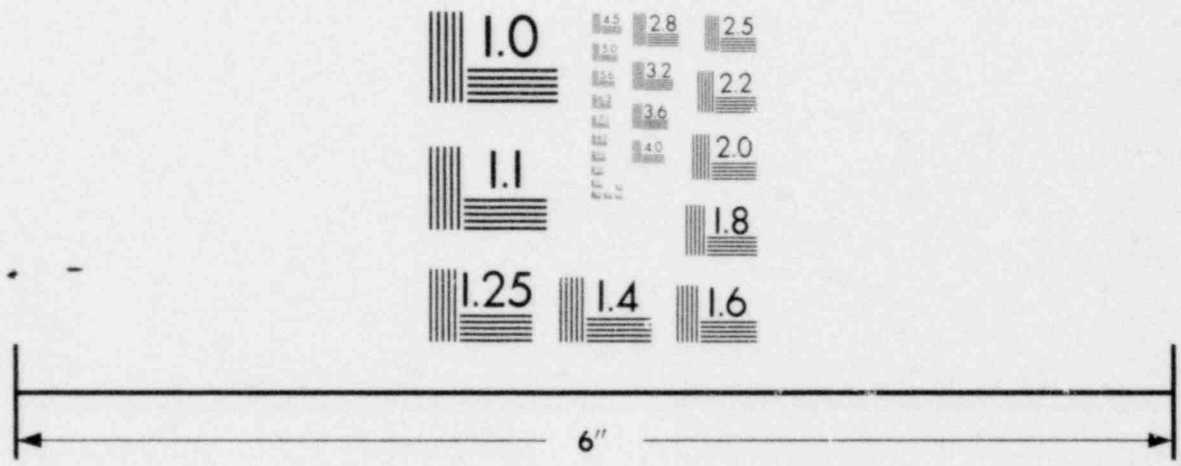
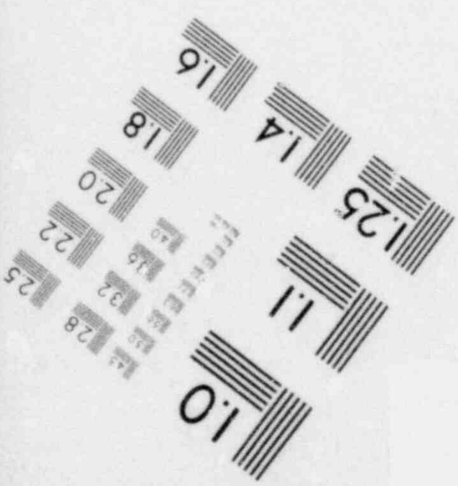
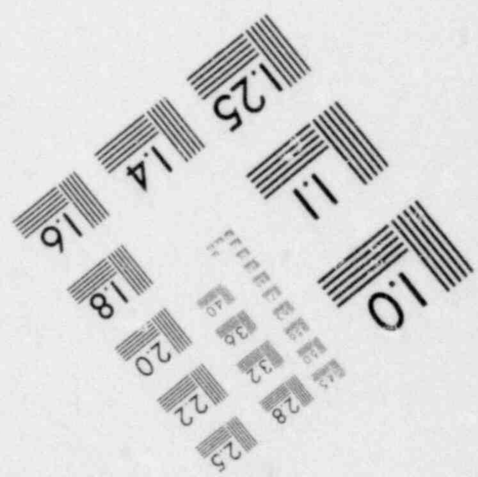


IMAGE EVALUATION
TEST TARGET (MT-3)







Two ambient-air vaporizers on each Dewar can evaporate the liquid argon at a
nominal maximum gas flow rate of 2000 scfh each, at 200 to 235 psig. With
59 both Dewars on-line, therefore, approximately 8000 scfh of argon gas can be
delivered.

The argon from the Dewars passes through a filter and is reduced in pressure to 175 psig. It is then piped to various points in the RSB and the intermediate bay before it is reduced to the required pressure upstream of each interface. Before entering the containment, the fresh argon header is
59 reduced in pressure to 50 psig.

9.5.1.2.3 Fresh Argon: RCB Distribution

The RCB fresh argon header enters the building with isolation valves on each side of the penetration. Valve status is shown by indicating lights on the Containment Isolation System (CIS) panel and the Inert Gas Receiving and Processing System local panel.

Upon entering the containment, this argon header manifolds into a number of branches supplying fresh argon through individual feed and bleed control arrangements to the primary sodium storage vessel, head access penetration G-6, IVTM storage pit, makeup pump drain vessel, primary cold trap NaK storage vessel, primary pump oil supply tanks, to various freeze vents, to the primary sodium plugging temperature indicator (NPTI) and sodium sampling package, and
59 to the RCB floor/wall service stations.

9.5.1.2.4 Fresh Argon: IB Distribution

A 175 psig pressure argon supply line is routed toward the ex-containment primary sodium storage vessels in the intermediate bay. The argon is reduced in pressure before entering the storage vessels, normally to 2 psig. Alternately, the pressure can be reduced to 40 psig for pressurized sodium transfer. These vessels are vented through a vapor trap and a pressure control valve to the Cell Atmosphere Processing System (CAPS). Additionally,
59 they can be evacuated via a vacuum station to CAPS.

9.5.1.2.5 Fresh Argon: RSB Distribution

The RSB header supplies argon at the required pressures to the fission gas monitor module and the gas sampling trap. A branch line allows argon purge of
59 the RAPS process train in the cold box.

Another 175 psig line supplies argon through regulators to the Auxiliary
59 Liquid Metal System EVS sodium and NaK components and to the EVS sodium PTI
and EVS sodium sampling package. The sodium lines have freeze vents that are furnished with argon during startup, maintenance, and sodium drain and fill
59 48 operations at a nominal pressure of 2 to 5 psig.

The RSB argon supply is reduced in pressure in three stages to satisfy the
interface requirements at the ex-vessel storage tank (EVST) and at the fuel
handling cell (FHC). Other reactor refueling system components serviced in
this area are the RSB plug storage facility, RSB floor service stations, EVST
59 seals, FHC operating gallery, and FHC conditioning loop filters and blowers.

9.5.1.2.6 <u>Fresh Argon Supply at the Steam Generator Building (SGB)</u>

Argon for the Steam Generator Building (SGB) is stored as liquid in two Dewars
located on the SGB pad. These Dewars have a capacity of 1500 gal. each and
are equipped with fill and vent lines. Normally only one Dewar is in
operation. When it is nearly empty, a low-level instrumentation signal
operates automatic controls to shut off that Dewar and to open a full Dewar to
the supply header. When the switchover takes place, an alarm signals the
operator who is then required to initiate action to fill the nearly empty
59 Dewar. A control override allows drawing on both Dewars simultaneously.

Two ambient-air vaporizers on each Dewar can evaporate the liquid argon at a
nominal maximum gas flow rate of 250 scfm each, at 200 psig. With adequately
sized piping and regulation, approximately 500 scfm of argon gas, at 93 psig
59 can be delivered to an intermediate loop expansion tank.

The argon flow from these Dewars passes through a filter and into a main
header. Branch lines serve the sodium receiving station and the incoming
sodium drum sodium sampling packages.

9.5.1.2.7 <u>Fresh Argon: SGB Distribution</u>

The argon flow from the main header leaving the SGB dewars is divided into
several branches and routed toward the three IHTS loops in the SGB. Each loop
59 supply services the following components: line vents (freeze vents), rupture
disc spaces, intermediate sodium characterization packages, intermediate
sodium pump seal purge and oil gravity tank, sodium dump tank, and the
pressure equalization line between the intermediate sodium pump and
59 intermediate sodium expansion tank, providing cover gas for both. Radioactive
purged gas from the sodium pump oil leakage collection tank and oil gravity
tank passes through an oil vapor trap before release to the atmosphere outside
of the SGB.

9.5.1.2.8 <u>Vacuum Services</u>

The argon distribution subsystem incorporates permanently installed vacuum
pumps. Several locations are provided for movable pumps that may be
59 temporarily connected.

59 9.5.1.2.9 <u>Atmosphere Purification Unit</u>

59 The atmosphere purification unit continuously processes a side-stream of argon
gas drawn from and returned to the FHC gas cooling stream. The unit contains
two parallel gas-treating trains, each basically consisting of a copper bed to
9 48 remove oxygen and a molecular sieve dryer.

While one of the loops of this unit operates, the other loop is regenerated by
59 flowing mixed argon-5% hydrogen gas through the molecular sieve dryer and then
through the copper bed to reduce copper oxide. The water produced by this
59 purge and reaction is removed by the unit vacuum pump to CAPS.

9.5.2 Nitrogen Distribution System

9.5.2.1 Design Basis

Nitrogen is to be supplied for (1) cooling and inerting the atmospheres of the
cells and pipeways containing radioactive sodium and the Control Rod Drive
Mechanism, (2) actuating pneumatically-operated valves in the inerted cells,
(3) cover gas for the Dowtherm tanks in the chilled water system, (4) purging
the IHTS steam generators and evaporators in the event of a sodium-water
reaction, (5) primary Na removal and autoclave operations, (6) purging of the
RAPS and CAPS cold boxes, (7) a cover gas for the Sodium Water Reaction
Pressure Relief System (SWRPRS), and (8) miscellaneous handling and
59 maintenance services.

The SGB nitrogen supply for the sodium-water reaction purge is sized to
59 provide 250 scfm of nitrogen for a maximum of 12 hours.

59 The SGB nitrogen supply rate to be available for the RCB and RSB cell purge
requirements is to be 250,000 scfd.

To meet these limits the nitrogen subsystem contains two sampling and analysis
units, one for the RSB and the other for the RCB which periodically samples
the gas in each nitrogen-inerted cell and analyzes its atmospheres for
radioactivity, oxygen, and water vapor content. The cell is purged
automatically by fresh nitrogen whenever the oxygen level exceed 2% or the
water vapor concentration exceed 1000 vppm (one of these, by operator
selection) as monitored by the respective sampling and analysis unit. If, as
the result of purging to reduce the water vapor level, the oxygen
concentration falls below 0.5%, dry oxygen from a gas supply bottle will be
59 introduced manually into the affected cell at a tap provided for this purpose.
The RSB sampling unit causes the cell exhaust gases to be diverted to CAPS if
they are radioactive, or to be diverted to heating and ventilating if they are
59 55 not radioactive. All RCB inerted cells are normally exhausted to CAPS, and an
alarm is sounded when a high amount of radioactivity is detected.

The oxygen content of a nitrogen inerted cell is to be limited to 0.5 to 2.0%,
and the water vapor concentration to less than 1000 vppm. The oxygen limits
are chosen to provide enough oxygen to prevent nitriding of the steel, and yet
not exceed a fire-limiting concentration of oxygen. The water vapor is
48 limited in order to assure early detection in the event of a small sodium
59 leak.

9.5.2.2 Design Description

The Nitrogen Distribution Subsystem comprises three sets of liquid nitrogen supply dewars, with vaporizers to provide gaseous nitrogen, two banks of auxiliary gaseous nitrogen supplies (pressure bottles), and the necessary valves and piping for distribution of the gas to usage points. The nitrogen
59 48 gas and liquid are of commercial-grade purity.

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 to a pre-set level. When the switchover takes place an alarm signals the operator who is then required to
59 initiate action to fill the depleted dewar. A control override, however, allows the option of simultaneously supplying nitrogen from both tanks so that
59 increasing 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. Oxygen or water vapor levels are controlled by the sampling and analysis unit as described in
59 Section 9.5.2.1.

55 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 due to residual moisture expulsion from pipe insulation, this water will
59 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
48 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 ot 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

59 The 150 psig RSB header, after providing a side stream for inerted cell valve operating, 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
50 Heating, Ventilating and Air Conditioning 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
59 located within the RSB. A controlled pressure N_2 supply is provided separately to the autoclave.

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

9.5.2.2.5 Nitrogen: RSB Auxiliary Supply

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

provides nitrogen to ensure the uninterrupted operation 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 which
isolates the valve circuit precludes auxiliary supply blowdown to the
remainder of the failed circuit.

9.5.2.2.6 <u>Nitrogen Supplies at SGB</u>

The normal-use nitrogen supply for the SGB is stored as liquid nitrogen in two
Dewars, with 3000 gal. capacity each, on the SGB pad. The liquid nitrogen is
converted to gas by an ambient-air vaporizer (at 15,000 scfh nominal rating)
for each Dewar. Normal usage is supplied from one Dewar, with a level sensor
automatically switching tanks upon depletion to a pre-set level. When the
switchover takes place an alarm signals the operator, and he is then required
59 to initiate action to fill the depleted Dewar. A control override allows the
option of simultaneously supplying nitrogen from both tanks, so that
59 increasing the flow rate to meet abnormal demands is possible.

The nitrogen supply for sodium-water reaction control is stored as liquid
nitrogen in one Dewar of 3000 gal. capacity, also located on the SGB pad. The
liquid nitrogen from this Dewar is vaporized by 3 vaporizers (total capacity
750 scfm). Under normal conditions, the vaporizers can be maintained at
temperature and also provide a small flow to maintain positive pressure in the
Inerted Sodium-Water Reaction Pressure Relief System (SWRPRS). Nitrogen gas
at 200 psig is provided during normal use and during a sodium reaction
accident.

9.5.2.2.7 <u>Nitrogen: SGB Distribution</u>

Headers branch from the normal-use supply to provide nitrogen for service
stations in the SGB, sodium maintenance area, and hot shop. Effluent gases
from these operations will be discharged to CAPS. Provision is made to supply
nitrogen for inerting the ex-containment PSST cell in the IB when sodium is
present.

59 9.5.2.2.8 <u>Nitrogen: Sodium-Water Reaction</u>
Nitrogen gas is provided to the SWRPRS to maintain a positive pressure in the
inerted atmosphere. In the event of a sodium-water reaction, nitrogen purge
will be initiated to prevent the establishment of explosive mixtures of
hydrogen within the SWRPRS.

A nitrogen gas supply at a minimum flow rate of 150 scfm and 190 psig is
59 provided automatically to the water-steam side of the Steam Generator System
following system blowdown. This prevents sodium from entering the water side
in the event of a leak in any one of the nine sodium-water heat exchangers.
Additionally, nitrogen can be manually supplied to fill the system during
59 48 water drainage.

9.5.3 Safety Evaluation

An evaluation of the design of the IGRP System must include the functions and operations of the RAPS and CAPS subsystems, as well as those of the argon and nitrogen subsystems. As has been noted above, the design discussion of the RAPS and CAPS subsystems is presented in Section 11.3. The following evaluation draws on the information in that section.

The evaluation of the design of the IGRP System is based on the degree to which the system meets its major objective, that the radioactivity release to the environment be as low as reasonably achievable, and the corollary objectives that all requirements for the use and control of inert gases, both for normal and off-normal conditions, be satisfied. The following questions are addressed.

9.5.3.1 Control of Radioactive Gases

The principal safety consideration in the design of the IGRP is that the leakage of discharge of radioactive gases to both restricted and unrestricted areas must not only be lower than the maximum permissible concentration (as given by 10 CFR 20 under normal conditions) but must also be as low as is reasonably achievable.

The RAPS subsystem (Section 11.3), by means of a cryogenic still, reduces and maintains the radioactivity in the recycle argon cover gas at a steady-state level such that the piping that distributes the gas to the reactor head and to the primary pumps does not present a radioactivity hazard to operating or maintenance personnel. RAPS reduces the cover gas activity concentrations for most of the radioisotopes by many orders of magnitude, the average decontamination factor being approximately 1,000*. This is particularly important because a large contribution to the total plant release of radioactivity is the diffusion of cover gas through the reactor head seals. A second, but very much smaller contribution to the plant release of radioactivity is the leakage of recycle argon gas from the buffered reactor head seals.

This gas, which originates as RAPS subsystem effluent, also leaks from the seals into the reactor head access area and is released to the atmosphere through the RCB heating and ventilating system. The performance of RAPS is sufficiently effective that these seal leakages result in site boundary dose rates that are a small fraction of the normal background dose rate. Site boundary doses presenting specific values are given in Section 11.3.7.

The small but finite expected leakage or diffusion of cover gas through piping and components into the primary sodium system cells is another source of potential radioactivity release to the environment. In order to prevent the direct release of this activity, purged cell atmospheres containing leaked radioactivity are processed in CAPS to remove gaseous fission product activities. The two delay beds of CAPS provide a decontamination factor of about 62, averaged for all the radioactive isotopes processed. This capability is more than adequate to handle expected normal leakage.

* Average decontamination factor is the total influent radioactivity divided by the totalled effluent radioactivity for all radioisotopes.

48

If the CAPS effluent is of sufficient radioactivity which if released accidentally (to the site boundary) would result in a site boundary dose of 2.5 rem, then a radiation detector located in the effluent line would sound an alarm requiring the operator to initiate action to control the effluent radioactivity. Radiation monitors in the CAPS effluent line will be set to automatically prevent the effluent from resulting in a site boundary dose greater than 10CFR50, Appendix I allowable limits, by sounding an alarm and
59 closing the effluent line.

All penetrations of containment by IGRP System piping are protected by double isolation valves (one inside, one outside) that prevent the escape of contamination through the pipes and out of containment when high activity levels exist in containment. Specific details of containment isolation are presented in Section 6.2.4.

The effects of off-normal events that cause RAPS piping or vessel ruptures outside of containment are discussed in Chapter 15. In brief, the system design lends itself to procedural actions that will contain gaseous radwaste within leak-tightness-specified equipment cells under the worst circumstances. This delay in releasing gaseous radwaste very significantly mitigates the effect of an accident by allowing the decay of radioactivity within the cell, before initiating cleanup procedures.

The fuel handling cell will be well sealed, so that the in-leakage rate of air and its moisture will be small. The presence of sodium vapor in the cell will tend to reduce the concentration of the water vapor by reaction to form NaOH. In addition, the NaOH "smoke" that settles out in the cell will also be a getter for water vapor because of the hygroscopicity of NaOH(s).

59 The Argon Purification Unit (APU) is to be procured from a supplier as a unit having specified and demonstrated performance capabilities. The specifications for the unit are to be developed in the course of system engineering, during which period the parameters that affect the oxygen and water vapor concentrations will be established. The selected fabricator will be required to prepare and submit a design for approval prior to beginning fabrication of the FHC-APU. The fabricated unit will then, as a condition of acceptance, be required to demonstrate that it meets the specified performance requirements (75 volume ppm maximum water vapor and oxygen).

9.5.3.2 Availability of Inert Gases

The supply of inert gases to other systems in the CRBRP is ensured by the installation of a complex of liquid gas storage vessels and distribution systems. Both argon and nitrogen are available at the RSB and the SGB. In addition, there is an independent installation of liquid nitrogen for accident
59 48 control uses in the SGB.

The liquefied gas stations are also fitted with equipment sized to provide gas at flow rates, pressures, and durations in excess of the minimum requirements, thus providing a margin of safety.

The supply of gas for the essential function of valve operation is ensured by the installation of supplementary high-pressure gas bottles in the RCB and in the RSB. These serve as safe shutdown protections in the event of an interruption or loss of the principal gas supply.

9.5.4 Test and Inspections

The components and piping of the IGRP System meet the requirements of the applicable sections of the ASME Boiler and Pressure Vessel Code and ANSI Code B31.1.0. NEMA Standards are applied to the electrical equipment. The system design, procurement, manufacturing, construction, and installation conform with the quality assurance requirements of 10 CFR 50, Appendix B, and RDT F 2-2.

9.5.5 Instrumentation Requirements

9.5.5.1 General System Requirements

The following instrumentation requirements are common to all of the IGRP subsystems:

a. Functions

The Inert Gas Receiving and Processing Instrumentation System shall perform the following functions:

1) Monitor process parameters and positions of selected valves

2) Maintain process parameters within normal prescribed operating ranges

3) Provide for overriding the normal control loops in the event of abnormal conditions of pressure, temperature, or gas analysis, including both venting and blocking-off of subsystems

4) Provide for automatic isolation of all process gas lines entering or leaving the Reactor Containment Building

5) Provide for the remote manual operation of valves (both proportional and on-off control, as required)

6) Provide local, centralized, and main control room I&C panels to accomplish the above.

b. Indicators

The following process variables will be logged, indicated,
48 recorded, or alarmed, as is appropriate:

1) Gas pressures and temperatures

2) Gas flow rates

3) Liquid levels in supply vessels

4) Valve position status for selected valves

5) Piping and component temperatures

6) Component pressure drops.

59 7) Radioactivity concentrations

c. <u>Controls</u>

The following general control functions are to be provided as required:

1) Liquefied Gas Supply Vessels: level control to automatically switch to full tanks in sequence on low-level signals from another tank or tanks, and high-flow shutdown capability

2) Supply Headers: pressure reduction and regulation

3) Vessel Cover Gases: pressure regulation and over-pressure relief

4) Containment Isolation: remote controls for valves.

9.5.5.2 <u>Argon Distribution Subsystem</u>

The specific instrumentation requirements for the Argon Distribution Subsystem are:

1) Control of the use of the auxiliary in-containment argon bottles to respond to low argon pressure in the normal supply headers

2) Pressure regulation for the fuel handling cell atmosphere

3) Controls for automatic regenerative operation of the FHC atmosphere purification unit

4) Temperature controls on freeze vents, vapor traps, vapor condensers, and heated argon lines

5) Controls to minimize reactor cover gas pressure oscillation during temperature transients

48 6) Control of flow of recycle argon cover gas to the PHTS pumps and of total flow from the reactor and PHTS cover gas spaces to RAPS.

9.5.5.3 Nitrogen Distribution Subsystem

The specific instrumentation requirements for the Nitrogen Distribution Subsystem are:

1) Control of the use of auxiliary nitrogen bottles to respond to low-nitrogen pressure in valve-actuation line headers

2) Pressure regulation for nitrogen-inerted cells

3) Pressure and/or flow regulation for equipment cooling circuits for the control rod drive mechanism and for the RAPS and CAPS cold boxes

4) Control of cell atmosphere purges, by automatic-sequencing cell atmosphere sampling unit, and provision for on-line analyses for oxygen, water vapor, and radiation levels

5) Controls to divert the cell purge gas exhaust to CAPS when radio-activity exceeds the low-level radiation setpoint

6) Purge controls for cell atmosphere on selected signal of high oxygen content or water vapor level

7) Alarm signal when cell atmosphere radiation exceeds the high-level radiation setpoint, with operator re-set only.

9.5.5.4 RAPS and CAPS

The RAPS and CAPS subsystems have specific control requirements for the functions listed below:

a. RAPS

1) Pressure regulation in the vacuum vessel

2) Pressure regulation in the surge vessel

3) Gas flow rate regulation at the inlet to the cryogenic section (control of flow from surge vessel)

59 4) Radiation level measurement and indication in inlet stream to RAPS cold box

5) Radiation level measurement and indication of RAPS effluent stream to recycle argon vessels, with alarm on high signal

6) Manual flow bypass controls for cold box

7) Manual controls for the diversion to CAPS of cold-box effluent and
59 48 maintenance purge.

8) Automatic control of compressors with manual override

9) Automatic cryostill level control

10) Indication of totalized compressor running time

11) Manual controls for drain of noble gases from cryostill to noble gas storage vessel

12) Automatic controls for bleed of noble gases from noble gas storage vessel to CAPS input

13) Automatic controls for recirculation of portion of RAPS process effluent at inlet to the recycle vessel back to RAPS vacuum vessel

14) Alarm on signal of low surge vessel gas pressure and manual controls for fresh argon make-up

48 15) Automatic controls for isolation of RAPS cold-box cell upon signal of high radioactivity concentration in cell atmosphere.

16) Automatic control for isolation of RAPS noble gas storage vessel cell or high radiation in cell atmosphere

59 17) Automatic switch of flow from depleted liquid nitrogen vessel to full vessel

b. CAPS

1) Pressure regulation in the vacuum vessel

2) Pressure regulation in the surge vessel

3) Gas flow rate regulation at the inlet to the cryogenic section (control of flow from surge vessel)

4) Radiation level measurement and indication on CAPS effluent stream

59 5) Temperature control and indication of effluent process gas streams down stream of liquid nitrogen injectors

6) Manual flow bypass controls for the first delay bed in the cold box

7) Automatic control of compressors, with manual override

8) Tritium-water removal unit controls for automatic regenerative operation

9) Automatic temperature control of heater section in oxidizer unit

48 10) Indication of totalized compressor running time

11) Automatic division of cell vent flow from H&V to CAPS on high radiation signal

59 12) Measure and alarm high radiation concentration on CAPS inlet stream

TABLE 9.5-1

INERT GAS RECEIVING AND PROCESSING DESIGN CLASSIFICATION

(Sheet 1 of 2)

	Component	ASME B&PV Code-Class	Seismic Category
	RAPS Components (all)	III-3	I
	CAPS Components (all)	III-3	I
	Storage Vessels - Fresh Argon and Nitrogen	VIII	III
	Vaporizers - Fresh Argon and Nitrogen	Industrial	III
	Gas Filters		
	Argon and Nitrogen Supply	VIII	III
	Argon Auxiliary Supply	III-3	I
	CRDM Argon Supply	VIII	III
49	Intermediate Pumps	VIII	III
	Vacuum Pumps	Industrial	III
	Nitrogen Sampling and Analysis Units	Industrial	III
	Bottle Racks - Auxiliary Argon and Nitrogen	Industrial	III
	Purification Unit - FHC Atmosphere	VIII	III
49	Recycle Argon Storage Vessels	III-2	I
	Freeze Vents		
	FHC and EVS Sodium Systems	III-2	I
	PHTS Sodium Lines	III-1	I
	IHX Intermediate Sodium	III-1	I
	Reactor Makeup Line	III-1	I
	Primary Sodium Makeup Pump Lines	III-1	I
	Primary Sodium Cold Trap Lines	III-3	I
	Intermediate Sodium Cold Trap and Pump Lines	III-3	I
	Overflow Heat Exchanger	III-2	I
	Sodium Vapor Traps		
	Primary Sodium Storage Vessels	III-3	I
48	Reactor Cover Gas (including condensers)	III-2	I

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TABLE 9.5-2

SUMMARY OF PRESSURE VESSELS +

(Sheet 2 of 2)

Vessel	Gas Contained	Pressure (psig) Design	Operating	Maximum**	Operating Temperature (°F)	Volume (acf)	Stored Energy (10^6 ft-lb)
Argon Subsystem							
Recycle Argon Vessels (2 vessels)	Ar*	-14.7, 200	35	35	70 to 120	723 (total)	9.6 (total)
Liquid Argon Storage Vessels (4 vessels)	Ar	250	200	235	-240	200 (each)	5.4 (each)
59 Argon Bottle Racks (16 bottles each for 2 racks)	Ar	2,640	250 to 2,640	2,640	Ambient	23 (each rack)	11.1 (each)
59 Regeneration Gas H_2 Supply (1 rack) 2 bottles	H_2	2,640	250 to 2,640	2,640	Ambient		
Nitrogen Subsystem							
LN_2 Storage Vessels (4)	LN_2	250	235	235	-320	400 (each)	14.4 (each)
LN_2 Storage Vessels (2)	LN_2	250	235	235	-320	800 (each)	30.5 (each)
Nitrogen Bottle Racks (16 bottles each for 2 racks)	N_2	2,500	50 to 2,200	2,200	Ambient	26 (each rack)	16.0 (each)

* The named gases are the major constituents, but these gases also contain radioactive fission gases and tritium in various amounts.

\+ Parameters of individual vessels.

48 ** Maximum operating pressure.

TABLE 9.5-3

SUMMARY OF PROTECTIVE MEASURES TO ENSURE STORED ENERGY IN GAS VESSELS CANNOT DAMAGE REACTOR SHUTDOWN EQUIPMENT

Vessel	Protective Containment Feature
CAPS	
1. Charcoal Beds (2 beds)	Located in CAPS Cold Box in RSB Cell
2. Vacuum Vessel	Located in RSB Cell With Surge Vessel
3. Surge Vessel	Located In RSB Cell with Vacuum Vessel
RAPS	
4. Cryogenic Still	Located in RAPS Cold Box in RSB Cell
5. Vacuum Vessel	Located in RCB Cell with Surge Vessel
6. Surge Vessel	Located in RCB Cell with Vacuum Vessel
7. Noble Gas Storage Vessel	Located in RSB Cell
8. LN_2 Storage Vessels (2 vessels)	Located on RSB Operating Floor (Elevation 816')
Argon Subsystem	
9. Recycle Argon Vessels	Located in RCB Cell
10. Liquid Argon Storage Vessels	
(2 vessels)	Located on RSB Pad Outside the Building
(2 vessels)	Located on SBG Pad Outside the Building
11. Argon Bottle Racks	
(2 racks) of 8 bottles/loop	Located at each intermediate loop in SGB in a Barrier-protected Enclosure
59 (1 rack) 2 bottles	Located in RCB Cell for SCRDM pressure
12. Regeneration Gas H_2 Supply	
(1 rack) 2 bottles	Located on RSB Pad Outside the Building
Nitrogen Subsystem	
13. LN_2 Storage Vessels	
(2 vessels)	Located on RSB Pad Outside the Building
(4 vessels)	Located on SGB Pad Outside the Building
14. Nitrogen Bottle Racks	
(1 rack)	Located in RCB in a Barrier-protected Enclosure
48 (1 rack)	Located in RSB Cell

the basis of the ventilation requirements and pressurization requirements. The outside air flow is constant all-year-round. The supply air is distributed to the various areas by the supply ductwork to satisfy the ventilation requirements. The filters provided in the air conditioning units maintain the cleanliness of the supply air during normal operation. The cooling coils provided in the air conditioning units, and the duct mounted reheat coils, along with their instrumentation and controls maintain the temperature of the Control Room Areas during normal operation. The chilled water supplied to the Control Room air conditioning units is provided by the Emergency Chilled Water System. The humidifiers and cooling coils provided in the air conditioning units, along with their instrumentation and controls, maintain the required humidity for the various areas in the Control Room during normal operation. Two (2) 100% capacity redundant return fans are provided for returning and exhausting the air supplied to the Control Room. Four (4) sound attenuators are provided, two (2) downstream of the supply fans, and two (2) upstream of the return fans, to maintain the required sound level in the Control Room. Two (2) 100% redundant standby filter units are provided to reduce radioactive airborne contamination. Supply, return and exhaust ductwork, isolation valves, dampers, air outlets, instrumentation and controls are provided to make this system complete and operate as required. The 100% redundant air conditioning units which are located in separate cells are connected by ductwork to missile protected outside air intake structures.

Each air conditioning unit consists of a pre- and after filter, cooling coil, a V-belt driven centrifugal fan with automatic inlet vanes and access sections. The length of the access sections is determined by the maintenance requirements of the individual components. The return side of each unit is connected to the outside air ductwork by an opposed blade damper. The discharge side of each fan is connected to the supply ductwork by a flexible connection followed by an automatic isolation damper. The supply ducts join into a common duct which is provided with a branch duct that serves the Control Room HVAC equipment rooms. The two (2) branch supply ducts are provided with sound absorbers before serving the Control Room.

The two (2) 100% redundant filter units are located in separate
59 cells and are connected by a branch duct with an automatic damper to the outside air duct serving the Control Room air conditioning units.

The Control Room 100% redundant filter units consist of a prefilter section, a high efficiency particulate air (HEPA) filter section, a charcoal filter bank for radio-iodine adsorption, a final HEPA filter located downstream of the charcoal filter bank for removal of charcoal fines carried over from the adsorber, access sections between each component for maintenance and filter instrumentation consisting of flow and pressure differential switches, transmitters and indicators to facilitate
49 monitoring and testing of the filter operation.

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

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

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

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

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

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

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

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

the airborne activity levels for each intake location. The intake locations are positioned such that the airborne activity at one intake will be significantly less than the other. By providing activity monitors at these intakes, the cleaner intake for Control Room pressurization is selected to reduce the dose from airborne activity to persons in the Control Room.

The Control Room air intakes will be provided with detectors for
59 toxic chemicals such as hydrogen fluoride, chlorine, and ammonia.

A containment isolation signal or a high radioactivity signal
from the redundant radiation monitors located in the Control Room outside
air intake ducts or a signal of high levels of toxic chemicals such as
hydrogen fluoride chlorine and ammonia or smoke in the Control Room out-
side air intake ducts will initiate closure of the Control Room HVAC system
59 redundant outside air and exhaust isolation valves, open the redundant
filter unit outside air isolation valves, and start one of the 100%
redundant filter unit fans to allow a minimum amount of outside air,
required for pressurization, to pass through the operating filter unit.
Control Room operators are provided with the capability to manually
initiate isolation of the Control Room HVAC System if higher than normal
levels of toxic chemicals or radioactivity are detected in other areas of
the plant by toxic chemical detectors or by the Radiation Monitoring
system. Detailed descriptions of the radiation monitoring capability
provided for the CRBRP are given in Sections 7.3, 11.4, 12.1.2 and
12.2.4. Information on toxic gases is provided in Section 6.3.1.6.2.

9.6.1.2.2 <u>Control and Diesel Generator Buildings Emergency HVAC System</u>

Two (2) 100% capacity, minimum outside air, air conditioning
units provide conditioned supply air to their respective "A" or "B"
areas at El. 765'-0" and the cable spreading rooms of the Control
Building, and El. 794'-0" of the Diesel Generator Building. The con-
stant outside air flow provided by this system is based on the ventilation
requirements and on the Battery Rooms hydrogen dilution requirements.
Supply ductwork distributes the air to the various areas to satisfy the
ventilation requirements. The cleanliness of the supply air is provided
by the air conditioning units filters. The cooling coils provided in
the air conditioning units, along with their instrumentation and controls
maintain the space temperatures. The chilled water supplied to the air
conditioning units is provided by the Emergency Chilled Water System.
The four (4) Battery Rooms are provided with separate exhaust fans for
the dilution of the generated hydrogen. The remaining air supplied to
the spaces is returned by two (2) 100% capacity fans to their respective
air conditioning units where it is mixed with outside air, conditioned
and resupplied. Supply and exhaust ductwork, dampers, air outlets,
controls and instrumentation are provided to make the system complete and
49 operate as required.

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The two (2) 100% air conditioning units are located in separate
Control Building HVAC equipment rooms and are connected by ductwork to the
59 Control Building missile protected air intake structure.

Each air conditioning unit consists of a pre- and after filter,
cooling coil, V-belt driven centrifugal fan and access sections. The
59 discharge of each fan is connected to ductwork by a flexible connection.

The supply ducts from each air conditioning unit supplies air to
their respective "A" or "B" areas of the Control Building cable spreading
rooms, Vital AC/DC Rooms, and the Diesel Generator Building Class IE Switch-
gear Rooms. A transfer duct is provided in each Vital AC/DC Room to permit
59 the infiltration of air to the Battery Rooms.

A separate exhaust duct is provided from each Battery Room and is
connected to its respective exhaust fan. The exhaust fans are direct
driven centrifugal fans. The fans are connected to a common exhaust plenum
59 by a flexible connection followed by a gravity damper. The exhaust plenum
is connected with the Control Building missile protected exhaust structure.

59 The suction and discharge sides of the return fans are connected
to their respective return ducts with flexible connections. Each return
fan is located in separate Control Building HVAC equipment rooms. The
59 return fans are direct driven vaneaxial fans.

9.6.1.2.3 Control and Diesel Generator Buildings Non-Essential HVAC System

A recirculating type air handling unit with two (2) 50% capacity supply fans provides conditioned supply air to the Motor Generator Set Rooms and Switchgear Rooms of the Control and Diesel Generator Building, to the CB CRDM Rooms and to other areas of the Diesel Generator Building.

A supply duct distributes the air to the various areas to
49 satisfy the Ventilation Requirements. The exhaust, return and supply
dampers provide minimum outside air and return air mixture during

summer and winter operation, and 100% outside air during intermediate seasonal operation. The cooling coils, in conjunction with the dampers and their controls and instrumentation, maintain the supply air temperature. The chilled water is provided by the Normal Chilled Water System. The air handling unit's filters maintain the cleanliness of the supply air. Two (2) 50% capacity return fans are provided to return or exhaust the air supplied according to the seasonal operating mode. Supply and exhaust ductwork, dampers, air outlets, instrumentation and controls are provided to make the system complete and operate as required.

A separate unit cooler is provided for each Motor Generator Set located in the Control and Diesel Generator Buildings to remove the internally generated heat. The unit coolers are connected by ductwork to the outlet openings of the motors and generators. The room air is induced through the motors and generators by their internal fans and then discharged through the ductwork and unit coolers back to the room. The cooling water supplied to the unit coolers is provided by the Normal Plant Service Water System.

The recirculating type air handling unit is supplied with two (2) 50% capacity supply fans. The air handling unit is connected by ductwork to the Control Building missile protected air intake structure. Outside
59 air ductwork is connected to the outside air damper of the air handling unit. The air handling unit consists of a pre- and after filter, cooling coils and access sections. Downstream of the cooling coil section a sufficiently long end access section is provided for the connection of the 50% capacity supply fans. The suction side of each fan is connected to the air handling unit by a manual damper (normally locked open), an inlet bell and a flexible connection.

The discharge side of each fan is connected to a "Y" duct connector by a flexible connection followed by an automatic isolation damper. The "Y" duct connector joins the two (2) fans to a common supply duct. The supply fans are direct driven vaneaxial fans.

The supply duct branches off to serve the Motor Generator Set located in the Control Building and Diesel Generator Building, the Control Building CRDM Rooms, and the Diesel Generator Building Non-Class IE Switchgear Rooms.

Exhaust registers located in each Motor Generator Set Room, Control Building CRDM Room, and in the Diesel Generator Building Non-Class IE Switchgear Rooms are connected by ductwork with two (2) 50% capacity return fans. The suction side of each fan is connected to a common plenum by a manual damper (normally locked open) followed by a flexible connection. The fans are direct driven vaneaxial fans. The discharge of each fan is connected to a "Y" duct connector by a flexible connection followed by an automatic isolation damper. The "Y" duct connector joins the two (2) fans to a common exhaust duct. The common duct splits into two (2) branch ducts. One (1) branch duct is provided with an automatic damper and is connected to the Control Building missile protected exhaust structure. The other branch duct is connected to the return air
59 49 damper of the air handling unit.

The unit coolers provided for each Motor Generator Set located in
59 the Control Building and Diesel Building are located close to the respective M/G Set they serve and consist of disposable dry-type filters (for initial start-up only) cooling coils and a V-Belt driven centrifugal fan. The unit coolers are connected by ductwork to the outlet openings of the motors and generators of the MG Sets. An automatic damper is provided in the duct from the MG set to the unit cooler to return room air to the unit cooler during normal operation or to the induced air from the MG Sets if the unit cooler serving it should fail to operate.

9.6.1.3 Safety Evaluation

9.6.1.3.1 Control Room HVAC System

The Control Room HVAC System equipment must maintain functional integrity during emergency conditions. Therefore, the HVAC equipment serving the Control Room is designed to Seismic Category I requirements. All of the Control Room HVAC System equipment is located within the Control Building which is a tornado hardened, Category I building. Electrical power is available for the operation of all Control Room HVAC safety related equipment from the Class IE AC power supply. During failure of the Normal Chilled Water System, chilled water is supplied to the Control Room air conditioning units by the Emergency Chilled Water System. A single failure analysis for the Control Room HVAC System equipment is provided in Table 9.6-2.

The toilet exhaust and kitchen exhaust systems are the only non-essential portions of the Control Room HVAC System. They are separated from the essential portions of the Control Room HVAC System by two (2)
59 automatic isolation dampers in series. The exhaust duct from the exhaust fans connect with the Control Building missile protected exhaust structure.

Airborne radioactivity monitors and area monitors are provided throughout the plant and in the Control Room. A signal of high radioactivity from the Containment Isolation Radiation Monitors (Section 7.3) or from the redundant radiation monitors located in the Control Room air intake ducts, will alarm in the Control Room, and will initiate isolation of the Control Room and divert the outside air supply from one of the two
59 widely separate intakes, through one of the Control Room filter units. The capability to manually initiate isolation of the Control Room and divert
59 the outside air supply through one of the Control Room filter units is provided if high levels of radioactivity are alarmed in the Control Room by the various airborne radioactivity monitors or area monitors located throughout the plant. Monitors will be provided at both intake locations in order to select the cleanest intake.

Toxic chemical detectors and smoke detectors are provided in the Control Room air intake ducts. A signal from these detectors indicating high levels of smoke or toxic chemicals in the Control Room air intake ducts, will be annunciated in the Control Room and initiate isolation of the Control Room and divert the outside air supply through one of the
59 49 Control Room filter units.

9.6.1.3.2 Control and Diesel Generator Buildings Emergency HVAC Systems

The Control and Diesel Generator Building Emergency HVAC System equipment must maintain functional integrity during emergency conditions. Therefore, the Control and Diesel Generator Buildings Emergency HVAC System equipment is designed to Seismic Category I requirements. All the HVAC equipment is located within the Control and Diesel Generator Buildings which are tornado hardened, Category I buildings. Electrical power is available for the operation of all the Control and Diesel Generator Buildings Emergency HVAC System Equipment from the Class IE AC power supply. During failure of the Normal Chilled Water System, chilled water is supplied to the Control and Diesel Generator Buildings Emergency HVAC System air conditioning units, from the Emergency Chilled Water System. (A single failure analysis for the Control and Diesel Generator Buildings Emergency HVAC system and equipment is provided in Table 9.6-3.)

9.6.1.3.3 Control and Diesel Generator Buildings Non-Essential HVAC System

The Control and Diesel Generator Buildings Non-Essential HVAC System is not a safety-related system. All of the Control and Diesel Generator Buildings Non-Essential HVAC System components are located within the Control and Diesel Generator Buildings which are tornado hardened, Category I buildings.

9.6.1.3.4 Operation of the Control Building HVAC System During Off-Normal Conditions

1. Single Failure of the Control Building HVAC System Components

The active components in the Control Building HVAC System which are susceptible to failure are as follows:

CR Supply Fans
CR Automatic Roll-type Filters
CR Filter Unit Supply Fans
CR Isolation Valves
CR Return Fans
Switchgear Supply Fans
Switchgear Automatic Roll-type Filters
Switchgear Return Fans
Battery Room Exhaust Fans
MG Sets & SWGR Supply Fans
MG Sets & SWGR Automatic Roll-type Filter
49 MG Sets & SWGR Return Fans
Automatic and Remotely Operated Dampers
59 MG Set Unit Cooler Supply Fans

The Control Room HVAC System is provided with two (2) 100% redundant Air Conditioning Unit supply fans, two (2) 100% redundant return fans and two (2) 100% capacity filter unit supply fans to maintain positive pressure as well as to maintain the normal design temperature in the CR during off-normal operation. The (2) 100% capa-
49

city Control Room air conditioning unit supply fans, the two (2) 100% capacity Control Room return fans and the two (2) 100% capacity filter unit fans are connected to the on-site emergency Class IE AC power supply. Failure of the operating fan would automatically start the redundant fan.

The Control Building HVAC System Air Conditioning Units and Air Handling Unit are provided with automatic roll-type filters. Failure of the advance mechanism for the filter media would result in increased pressure across the filters. A sensing device is provided with an alarm setpoint to indicate that the differential pressure across any filter is higher than normal. This alarm setpoint is selected on the basis that after initiation of the alarm, 72 hours are available to correct the failure without significantly deviating from the system design parameters.

The Control Building Cable Spreading Rooms, Battery Rooms, Vital AC/DC Rooms, and the Diesel Generator Building Class IE Switchgear Rooms HVAC System are provided with two (2) 100% air conditioning units and two (2) return fans, to maintain the upset design temperature during off-normal operation, for their respective "A" or "B" trains. The Air Conditioning units supply fans and the return fans are connected to their respective "A" or "B" train on-site Class IE AC power supply. Failure of one (1) fan would not affect operation of its redundant train.

The MG Set & Switchgear HVAC Supply System is provided with two (2) 50% supply and exhaust fans. Failure of any one supply or exhaust fan would not increase the room temperature above the upset design
49 temperature.

Each motor-generator set in the Control Building is provided with a Unit Cooler. An automatic damper is installed in the Unit Cooler ductwork to allow the induced air to be returned back to the room in case of failure of the Unit Cooler. Failure of the MG Set Unit Cooler Supply fan will be increase the cell temperature above
59 the upset design temperature.

2. Protection Against the Increase of Internal Airborne Radioactivity in the Control Room During Off-Normal Operation

Two (2) 100% redundant filter units, each with a 100% capacity fan, are provided to prevent the increase of internal airborne radiation into the Control Room above the limits of the CRBRP General Design Criterion 17 and 10CFR20. Upon a signal of high radioactivity content from the radiation monitors located in the outside air intake ducts or from the containment isolation radiation monitors, the redundant isolation valves located in the outside air duct serving the Control Room Air Conditioning units will close, the redundant filter unit out-
59 49 side air duct isolation valves will open, the redundant isolation

valves located in the exhaust duct will close, the main or remote
59 supply duct isolation valves will close (depending upon the amount of
radioactivity detected at the widely separated intakes), and one of
the filter unit fans will start. The outside air will mix with the
recirculated air before being filtered by the Control Room filter
unit. The filtered air will then mix with return air from the Control
49 Room before entering the Control Room air conditioning units.

3. Purging of the Control Room After a Fire

After the fire is extinguished, the Control Rocm return/exhaust fan will be manually started, and fans will pull the air from the Control Room and exhaust it to the outside through the Control Building
59 missile protected exhaust structure.

4. Dilution of Hydrogen Concentration in the Battery Rooms During Off-Normal Operation

Four (4) 100% capacity exhaust fans (one for each battery room) are provided to exhaust the air supplied by the main system. The fans
59 are connected to the on-site Class IE AC power supply.

5. Occurrence of Fire Inside Control Building

Smoke venting capability is provided for areas containing safety related equipment. The stairwell will be pressurized during a fire accident.

A smoke vent fan and a smoke vent chase is provided to vent smoke from
59 areas containing Safety Related equipment except the Control Room. Upon a signal from the fire detection panel, a stair pressurization fan will start and pressurize the stairwell, relative to the surrounding area, to prevent smoke from entering the stairwell.

9.6.1.4 Testing and Inspection Requirements

All components are tested and inspected as separate components and as integrated systems. Velometer readings are taken to ensure that all systems are balanced to deliver and exhaust the required air quantities. All water coils are hydraulically tested for leakage prior to being placed in service. Capacity and performance of the fans are tested according to the Air Moving and Conditioning Association requirements prior to operation of the plant.

The filter trains will be periodically tested in accordance with
49 Regulatory Guide 1.52.

9.6.2 Reactor Containment Building

9.6.2.1 Design Basis

9.6.2.1.1 Reactor Containment Building HVAC System

The Reactor Containment Building HVAC System contains three EI&C cubicle unit coolers which are safety related. Otherwise it is a non-safety related system designed to provide filtered and conditioned air to the operating floor (El. 816'-0") and normal atmospheric areas below the operating floor of the Reactor Containment Building, to permit personnel occupancy and to ensure the operability of all equipment located in the above mentioned areas, during normal plant operation. The Reactor Containment Building (RCB) HVAC System is designed to:

a) Maintain a maximum of $95^{\circ}F$ throughout the building with the exception of $75^{\circ}F$ in the Instrumentation and Control Cubicles, during normal operation.

b) Maintain a maximum of $120^{\circ}F$ within the specified EI&C cubicles under single failure of a HVAC System component or upon loss of cooling from the Normal Chilled Water System.

c) Maintain a slightly negative pressure of approximately 1/8" water gauge during normal plant operation.

d) Maintain the required ventilation rate to dilute the cover gas leakage on the operating floor and to dilute radioactive gas leakage from the inerted cells below the operating floor.

e) Provide normally closed connections for temporary duct-work and equipment to facilitate purging of the inerted cells, to permit personnel entry for inspection and maintenance in an air atmosphere.

f) Provide isolation of the building upon a signal from the plant protection system indicating a release of high radioactivity to the Containment atmosphere.

49 g) Provide connections in the exhaust ducts of the LCCV and Sodium Pump oil seal tank vents.

h) Provide ducted cool air directly to the lube oil cooling panels.

9.6.2.1.2 Containment Cleanup System

The Containment Cleanup System mitigates the consequences of
59 49 events beyond the design base and is described in Reference 10b to PSAR Section 1.6.

9.6.2.1.3 Annulus Air Cooling System

The Annulus Air Cooling System mitigates the consequences of events beyond the design base and is described in Reference 10b to PSAR Section 1.6.

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9.6.2.1.4 Annulus Filtration System

The Annulus Filtration System is a Safety Class 3 system. The Annulus Filtration System design shall satisfy the following criteria:

a) The containment/confinement annulus space shall be maintained under 1/4 inch W.G. negative pressure during normal plant operation and accident conditions.

b) Capability shall be provided to filter the containment/confinement annulus exhaust during normal operation.

c) Capability shall be provided to filter the RCB ventilation exhaust air through the annulus filter system during refueling operations, when the RCB/RSB refueling hatch is open.

d) Capability shall be provided to filter and recirculate the annulus air during accident conditions. For every 1000 CFM filtered exhaust air (required for the maintenance of 1/4 in W.G. negative pressure) not less than 3500 CFM air shall be recirculated through the filters.

e) The Recirculating Duct System shall be designed to accomplish proper mixing in the annulus in accordance with USNRC Standard Review Plan Section 6.5.3.

f) The Annulus Filtration System shall be designed to meet
59 49 the requirement of the USNRC Regulatory Guide 1.52.

9.6.2.2 System Description

9.6.2.2.1 Reactor Containment Building HVAC System

The RCB HVAC System P&IDs are shown on Figures 9.6-4 through 9.6-
59 6 and 9.6-9. The Classification of the RCB HVAC System equipment and their primary parameters are indicated in Table 9.6-4.

9.6.2.2.2.2 Normal Plant Operation

During this operating mode, outside air if drawn through the missile protected outside air intake, conditioned by the air handling unit and directed to the operating floor supply air header and to the below operating floor air conditioning units by one of the redundant supply fans and associated ductwork. The conditioned outside air is mixed with air from the operating floor unit coolers and is then supplied to the various spaces above the operating floor. In the same manner the outside air is mixed with return air to the air conditioning units below the operating floor, supplying conditioned air to the various spaces below the operating floor. The RCB is maintained at 1/8" water gauge negative pressure by one of the redundant exhaust fans. In addition, the annulus pressure maintenance fan exhausts air from the annulus through the annulus filter to maintain a minimum 1/4" water gauge negative pressure with respect to the outside atmosphere.

59 a. Outside Air System

The Reactor Containment Building HVAC System consists of a 100%
59 outside air handling unit with two (2) 100% capacity redundant supply fans which provide conditioned outside air to the Operating Floor and to the normal atmospheric areas below the Operating Floor of the RCB. The outside air supplied by this unit is distributed by the supply ducts to the various areas, to satisfy the Ventilation Requirements. The filters provided in the air handling unit maintain the required cleanliness of the outside air supplied. The cooling and heating coils provided in the air handling unit, along with their instrumentation and controls, maintain the temperature of the supply air downstream of the supply fans.

Two (2) 100% capacity redundant Exhaust Fans are provided for the removal of the air supplied to the RCB. Three differential Pressure Controllers, with probes located at 0, 120, 240° Azimuth outside the RCB, control the Exhaust Fan's adjustable inlet vanes to maintain 1/8" WG slightly negative pressure inside the building to satisfy the Operational Requirements.

A purge exhaust duct, located below the operating floor and connected to the main exhaust duct is used for de-inerting cells using a portable exhaust filter fan unit with flexible ducts. Radiation monitors
59 are provided to stop the portable fan upon sensing a high level of radio-
49 activity.

Two booster fans are provided to exhaust the purged nitrogen to control the oxygen concentration level in the inerted cells.

A branch duct from Nuclear Island General Purpose Maintenance System, connected to the purge exhaust duct, is provided for removal of the effluents released from the decontamination process in the large
59 component cleaning cell 125.

All exhaust air, before leaving the RCB, must pass through a Time Delay duct provided with radiation monitors so that upon detection of a high level of radioactivity, adequate time is available to close the Supply and Exhaust Isolation Valves, to prevent the release of air-borne radioactive materials to the environment.

Since the operation of this system, although mostly not safety related, is essential for the Plant Operation and Availability, 100% redundancy is provided for the supply and exhaust fans. Supply and exhaust ductwork, containment isolation valves, dampers, air outlets, and instrumentation and controls are provided to make this system complete and operate as required.

59

The 100% Outside Air System Air Handling Unit with the redundant supply fans is located at El. 836'-0" in the west side of the SGB Intermediate Bay. The air handling unit is connected to a tornado missile protected air intake structure and consists of an intake damper, pre and after filters, heating and cooling coils and access sections. Downstream of the cooling coils, a sufficiently long end-access section is provided for connection of the redundant supply fans. The supply fans are connected to the end access section by fan inlet bells, manual dampers (normally locked open) and flexible
49 connections. The supply fans are direct driven, vaneaxial type

with a manual damper at the suction side and an automatic damper at the discharge side. The redundant supply fan discharges, through flexible connections, are joined together into a common duct by a "Y" duct connector. The 100% Outside Air System supply and exhaust ducts are provided with three (3) containment isolation valves, two
59 immediately outside and one inside the containment. The three (3) valves and interconnecting piping are designed in accordance with ASME, Section III, Class 2 requirements. A quick acting automatic relief damper is provided in each branch duct between the containment isolation valves and the supply and exhaust fans. The purpose of these dampers is to relieve the excessive pressure or vacuum on the ductwork which would be originated by the activation of the containment isolation valves.

The supply air is distributed to the Operating Floor by the main supply header running along the containment wall above the I&C cubicles. The air outlets serving the Operating Floor are selected to provide long air throw characteristics and each branch duct is provided with a remotely operated automatic damper. A separate branch duct is provided for each I&C room. The three (3) Unit Coolers, located on the top of the I&C Cubicles are connected to the main Operating Floor duct by branch connections. Automatic isolation dampers are provided in these branch duct connections.

The outside air for the normal atmospheric areas below the Operating
59 Floor is connected to the main duct above the operating floor.

Exhaust air radiation monitors are located between the Operating Floor Exhaust Register and the containment penetration. The radiation monitors are located and the exhaust duct velocity is determined according to the closure time of the isolation valves.

If an abnormal radiation level is sensed by the exhaust air radiation monitors, the isolation valves will close before the contaminated exhaust air reaches the penetration point, to prevent the release of radioactivity to the outside environs.

The main exhaust duct connects with the redundant Exhaust Fans which
59 are located at Elev. 861'-6" of the RSB. The exhaust fans are connected to the common exhaust duct on the suction and discharge side by a "Y" connector. Attached to the "Y" connector on the suction side of the fans is an inlet cone followed by a locked open damper and a flexible connection. The discharge side of each fan is attached to a flexible connection followed by a normally open automatic damper and discharge "Y" connector. The exhaust fans are direct driven vaneaxial fans provided with automatic inlet vanes (dampers). The exhaust duct connects with a missile protected exhaust structure located on the
59 49 roof of the RSB Ventilation Equipment Bay at Elevation 884'-0".

b. Above Operating Floor Areas

Three (3) Unit Coolers are provided above the EI&C cubicles to supplement the cooling capacity of the 100% Outside Air System. During normal operation, the unit cooler fans are drawing air from the Operating Floor atmosphere through their cooling coils and then discharging the cooled air through the Supply Duct Header back to the Operating Floor. The cooling coils provided in the unit coolers, along with their instrumentation and controls, maintain the Indoor Design Conditions.

One (1) Unit Cooler is located in the west side of the RCB at El. 857'-11". A second Unit Cooler is located in the north side of the RCB at El. 857'-11". The third Unit Cooler is located in the east side of the RCB at El. 857'-11".

The Unit Coolers consist of disposable filters (for initial start-up only), cooling coils and V-belt driven centrifugal fans. A short duct is attached to the discharge side of each unit cooler, to connect it with the RCB main supply header duct. A normally open automatic isolation damper provided at the end of each short duct permits the flow of air to the main supply header serving the Operating Floor and I&C Cubicles.

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Three (3) Unit Coolers are provided to serve the safety related equipment located in the EI&C Cubicles. The cooling coils provided in the Unit Coolers, along with their instrumentation and controls, will supplement the cooling capacity of the Operating Floor Unit Coolers and maintain the Indoor Design Conditions. 49

The three Unit Coolers are located at the west side of the RCB at Elev. 842'-0", the north side of the RCB at Elev. 824'-3" and the east side of the RCB at Elev. 842'-0". 59

The Unit Coolers consist of disposable filters (for initial start-up only), cooling coils and V-belt driven centrifugal fans.

Three (3) Bearing Oil panels serving the Primary Sodium Pumps are cooled by the conditioned air through branch duct from the main 59 Supply Duct Header. The air temperature leaving the panels is below the space temperature and is discharged to the Operating Floor to provide additional cooling. 49

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The three Primary Sodium Pump wells are cooled by the conditioned air above the operating floor. The cell air is drawn into the well through an air intake enclosure, located directly above the pump well. The air is then induced through openings in the Primary Pump Motor Casing by the shaft mounted fan and then discharged through flanged openings in the motor casing,
54 connected to vertical ducts terminating above the operating floor.

Two (2) Dome Recirculating Fans are provided above the Operating Floor to prevent the stagnation of air at the top of the containment dome. The fans are direct driven vaneaxial type.

59 c. Below Operating Floor Areas

Two (2) Air Conditioning Units are provided for the cooling and ventilation of the normal atmospheric areas below the Operating Floor of the RCB. A branch duct from the 100% Outside Air duct header provides outside air to these Air Conditioning Units. The mixture of outside and return air is drawn through the air conditioning units by their supply fans and then distributed by ductwork to the various atmospheric areas below the Operating Floor. Return ductwork and return fans, one for each air conditioning unit, return air to the units.

The filters provided in the air conditioning units maintain the required cleanliness of the supply air. The cooling coils, along with their instrumentation and controls, maintain the Indoor Design Conditions.

One (1) Air Conditioning Unit and its respective Return Fan are located in the west side of the RCB at El. 752'-8". The other Air Conditioning Unit and its respective Return Fan are located in the east side of the RCB at El. 752'-8". The Air Conditioning Units consist of disposable filters (for initial start-up only), cooling coils, and V-belt driven centrifugal fans.

The return air fans serving the air conditioning units are direct driven vaneaxial fans. The suction and discharge of the fans are provided with flexible connections to isolate the fan from the connecting ductwork.

One (1) Unit Cooler is provided to serve the LCCV Cell. The cooling coil provided in the Unit Cooler, along with its instrumentation and controls, maintains the Indoor Design Conditions.

The Unit Cooler consists of a disposable filter (for initial
49 start-up only), a cooling coil and a V-belt driven centrifugal fan.

One (1) Unit Cooler is provided to serve the Head Access Area (HAA). The cooling coil provided, along with its instrumentation and controls, will maintain the Indoor Design Conditions. The unit cooler, located in the south side of the RCB at El. 800'-9" consists of a disposable filter (for initial start-up only), a cooling coil and a V-belt driven centrifugal fan. The discharge side of the fan is provided with a flexible connection and
59 connected to a ductwork which supplies air to the HAA.

A purge exhaust duct, with flanged connections, located below the Operating Floor, and connected to the main exhaust duct is used
49 for de-inerting the inerted cells.

If purging is required for one of the inerted cells, a portable filter-fan unit will be connected by a flexible duct to a normally blanked-off connection located on the Recirculating Gas Coolant System return line from the cell and to the closest inlet connection of the exhaust duct. After the portable filter-fan unit is started, the blind flange located on the Recirculating Gas Coolant System supply line to the cell will be removed. The portable filter-fan unit will draw air into the cell through the opened connection and will replace the inert gas in the cell with air. When reinerting of the cell becomes necessary, the blind flange will be replaced and inerting gas will be introduced into the cell. After the cell atmosphere is completely replaced by the inert gas, the portable filter-fan unit will be stopped, and disconnected from the Recirculating Gas Coolant System return line, the flexible connection will be removed from the exhaust duct and the duct opening will be recapped. A radiation monitor provided at the discharge side of the portable filter-fan unit will stop the purging operation if high radiation levels are
53 sensed.

The portable filter-fan unit is a factory fabricated, modular, draw-through unit consisting of a disposable pre-filter, a HEPA filter, and a V-belt driven centrifugal fan. The unit is provided with flexible ducts and is mounted on a movable cart.

Two (2) V-belt driven centrifugal Booster Fans, each with a gravity type discharge damper, are connected by ductwork to the purge exhaust duct, and are provided to exhaust the purged nitrogen to control the oxygen concentration level in the inerted cells.

A branch duct from the maintenance system, connected to the purge exhaust duct, is provided for removal of the effluents released
49 from the decontamination process in cell 125.

2. Open Hatch Refueling

During Open Hatch Refueling, the RCB supply and exhaust fans will be stopped. As soon as 1/4" WG negative pressure is achieved in the RCB, the RCB supply fan will be started to provide the necessary ventilation in the RCB. The RCB Exhaust Air will be diverted through the RCB Annulus Filter Units. The Annulus Filter Unit Fans and the Annulus Pressure Maintenance Fans will pull the exhaust air through the Annulus
59 Filter Unit and exhaust it at the top of the RCB.

3. <u>Accident Conditions</u>

Upon a containment isolation signal, the redundant supply and exhaust containment isolation valves will close automatically, the RCB supply and exhaust fans will shut down, and the redundant Annulus Pressure Maintenance Fan and the Annulus Filter Fan will start. The annulus pressure maintenance and filter fans, through the filters, will draw air from the annulus space. Approximately [illegible] of the air will be discharged to the outside environs at the top of the RCB to maintain a minimum 1/4" water gauge negative pressure and the rest will be recirculated back to the annulus space. The recirculation is provided to give good air mixing and dilution from the bottom of the annulus to the exhaust point at the top of the annulus. 59

9.6.2.2.3 <u>Deleted</u>.

9.6.2.2.4 <u>Annulus Filtration System</u>

During normal operation, one of the two (2) annulus pressure maintenance fans will exhaust air from the annulus through its associated filter unit and discharge it to the outside atmosphere to maintain a minimum 1/4" water gauge negative pressure in the annulus with respect to the outside atmosphere.

Upon a containment isolation signal, the redundant supply and exhaust containment isolation valves will close, the RCB supply and exhaust fans will shut down, and the redundant annulus pressure maintenance fan and the two (2) annulus filter fans will start.

When the two (2) annulus pressure maintenance fans and the two (2) annulus filter fans are proven, one (1) annulus pressure maintenance fan and its associated annulus filter fan will shut down. The other pressure maintenance fan and filter fan will continue to
49 operate with its associated filter unit.

During this condition, only a portion of the total air flow is exhausted to the outside atmosphere and the remainder of the total flow is returned back to the annulus space below the 816'-0" elevation where it is relieved to the upper annulus through equally spaced openings
59 at elevation 816'-0".

The RCB Annulus Filter Unit, the RCB Annulus Filter Unit
59 Fan, and the RCB Annulus Pressure Maintenance Fan located at El. 861'-6" of the Reactor Service Building and associated ductwork comprise one (1) of the two (2) 100% redundant Annulus Filtration Systems. Similar com-
59 ponents at El. 840'-6" of the Reactor Service Building and associated
49 ductwork form the second redundant filtration system.

The Annulus Filter Unit consists of a moisture separator, a heating coil, a high efficiency bag type filter, a high efficiency particulate filter (HEPA), an adsorber, and an after HEPA filter. The Annulus Pressure Maintenance Fans and the Annulus Filter Fans are V-belt
49 driven centrifugal fans with automatically adjustable inlet vanes.

9.6.2.2.5 RAPS Exhaust System

The Radioactive Argon Processing System (RAPS) cells are maintained under negative pressure during NORMAL OPERATION. Air is infiltered through the RAPS cells from the normal atmospheric areas of the RCB and exhausted by a separate exhaust duct which is connected to the RCB main exhaust duct. A quick closing valve is provided within the exhaust duct.

If the radiation monitor, provided in the exhaust duct, senses abnormal radioactivity, the quick closing valve closes and isolates the RAPS cells exhaust system. The Inert Gas Receiving and Processing System will determine which cell has a radiation leak such that the RAPS cells exhaust system can manually re-instate the maintenance of the negative pressure to the other RAPS cells. The leaking cell is kept isolated from the RAPS cells exhaust system until that cell is decontaminated.

When maintenance is required in one of the RAPS cells, the cell purging port is opened and air from the RCB will infiltrate through the port opening to purge the cell. After the cell atmosphere is completely replaced by infiltrated air, the purging port is closed and the cell door is opened for maintenance. When maintenance in that cell is completed,
53 the cell door is closed, and negative pressure in this cell is re-instated.

49 ### 9.6.2.3 Safety Evaluation

Single Failure Analysis for the Safety Related RCB HVAC System
59 and presented in Table 9.6-4a and 9.6-4b.

9.6.2.3.1 Reactor Containment Building HVAC System

Heat is removed from the RCB atmospheric areas by the Outside Air Air-Conditioning System, the three Operating Floor unit coolers,
59 three EI&C cubicle unit coolers, and the two below operating floor recirculating Air Conditioning units. This diversity of heat removal capability precludes the possibility of a single active failure that could result in excessively high temperatures. An evaluation of the considered failure modes follows:

1. Single Failure of the RCB HVAC System Components

   Single failure of any HVAC system components will not result in space temperatures exceeding upset design conditions.

   The RCB HVAC System consists of active and passive components. "An active component is one in which mechanical movement must be initiated or electrical power must be provided to accomplish a function of the component". All other components are considered as passive components. The system design has no provisions for failures of passive components.

   The active components in the RCB HVAC system consist of the following:

   Outside Air System Supply Fans
   Outside Air System Exhaust Fans
   Outside Air System Automatic Roll-Type Filter
   Unit Cooler Fans
   Below the Operating Floor Air-Conditioning Unit Supply Fans
   Below the Operating Floor Return Fans
   Containment Isolation Valves
   Dome Recirculating Fans
   49 Automatic Dampers
   59 Inerted Cell Booster Fans

   The 100% Outside Air System Supply and Exhaust Fans have 100% stand-by redundant fans. If any one fan fails, its redundant fan will automatically start to maintain normal system operation.

   The Outside Air System Automatic Roll-Type Filter has an automatic advance mechanism which advances the filter media on the basis of
49 sensed differential pressure across the filter. The differential

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

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

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

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

2. Loss of Normal Chilled Water Supply

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

3. Loss of Normal Power

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

4. Radioactive Contamination Protection of the RCB Areas

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

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

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The source of airborne radioactive contamination originates from the covergas leakage at the reactor head access area and at the primary pump seals. The ventilation air quantities for the above operating floor areas are selected to provide sufficient margin for the covercas leakage dilution. This selection provides a highly conservative design.

The radiation monitoring system, as described in Section 12.2.4 of the PSAR continuously monitors the RCB atmosphere and provides the Plant Protection System with an indication of the radiation level present. Upon detection of high radiation levels in the RCb atmosphere, the containment isolation system acts to close the
59 three containment isolation valves and provides sufficient protection to prevent the release of airborne radioactivity to the outside environs and ensures the maintenance of the site boundary limits during accident conditions, as required by 10CFR100.

9.6.2.3.2 Deleted.

9.6.2.3.3 Deleted.

59 9.6.2.3.4 Annulus Filtration System

The Annulus Filtration System is a 100% redundant system. If any one (1) component fails, its redundant component will start automatically.

The annulus pressure maintenance fans, the filter unit fans,
49 and the filter units are connected to the on-site emergency Class IE AC power supply.

9.6.2.4 Testing and Inspection Requirements

All components are tested and inspected as separate components and as integrated systems. Velometer readings are taken to ensure that all systems are balanced to deliver and exhaust the required air quantities. All water coils are hydraulically tested for leakage prior to being placed in service. Capacity and performance of the fans are tested according to the Air Moving and Conditioning Association requirements prior to operation of the plant.

59 The filter trains for the Annulus Filtration System will be periodically tested in accordance with Regulatory Guide 1.52.

9.6.3 Reactor Service Building HVAC System

9.6.3.1 Design Basis

9.6.3.1.1 Fuel Handling Area HVAC System

59 The Fuel Handling Area HVAC System (Figures 9.6-7, 7a, 8, 8a, 9) provides filtered and conditioned air to the atmospheric areas of the Reactor Service Building Fuel Handling Area, to permit continuous routine access to personnel and to ensure operability of the equipment during normal operation. The HVAC system serving these areas is designed to:

a) Maintain $95^{\circ}F$ to $110^{\circ}F$ within the Fuel Handling Area during normal operation except for the FHC Operating Gallery, the Refueling Communication Center, and the Fuel Failure Monitoring Operating Areas which will be maintained at $75^{\circ}F$.

b) Maintain the ventilation rate for dilution of gas leakage from the Fuel Handling equipment seals and from the inerted cells.

c) Minimize the spread of airborne radioactive materials within the areas below the operating floor, in the event of a leak from an inerted cell.

d) Provide normally closed connections for temporary ductwork and equipment to facilitate purging of the inerted cells to permit personnel entry for inspection and maintenance in an air atmosphere.

e) Maintain a negative pressure and capability for isolation within the Radioactive Argon Processing System (RAPS) and
49 the Cell Atmospheric Processing System (CAPS) Cells.

f) During normal plant operation and accident conditions, the normal atmospheric areas of the RSB shall be maintained at a minimum 1/4" W.G. negative pressure with respect to the outside atmosphere except when the railroad door is open.

g) Reduce and maintain the combined RSB and RCB building pressure in conjunction with the Annulus Filtration System at a minimum 1/4" W.G. negative pressure with respect to the outside atmosphere after the containment refueling hatch is opened.

h) Supply fresh air to the RSB during normal operation to maintain the concentration of airborne radioactivity levels at not higher than one-tenth of MPC occupational limits.

i) Limit the release of airborne radioactive materials to
59 the outside atmosphere by filtering the RSB exhaust air
through the RSB Cleanup System filter-fan units. The
filter system consists of two completely redundant
trains, each train consisting of a moisture separator, an
electric heating coil, a bag type pre-filter, a HEPA
filter, an adsorbent filter and an after HEPA filter, in
addition to associated fans, ducts and valves, and
59 related instrumentation.

j) For design basis events, the filter efficiencies of
Section 6.2.6.2 are applied. Capability shall be pro-
vided to isolate the normal supply and exhaust air flow
to and from the floors where an accident has occurred and
to maintain these areas at a greater negative pressure
than the rest of the building using a separate exhaust
duct system, in order to prevent the spread of airborne
49 radioactivity or sodium combustion products from con-
taminated to clean areas within the building.

k) During fuel handling accident mode, capability of isolating the HVAC System and to operate in recirculating mode to provide cooling when the outside air inlet damper is closed.

l) Capability shall be provided to (1) filter the by-pass
leakage to the RSB from RCB during a Site Suitability
Source Term event (2) mitigate the consequences of an RSB
fuel handling accident margin source term, and (3) pro-
59 vide margin for the TMBDB event.

9.6.3.1.2 Refueling Communications Center HVAC System

The Refueling Communications Center HVAC System (Figure 9.6-8a)
59 is a non-safety related system designed to provide filtered and conditioned air to the Refueling Communications Center, to permit personnel access and to ensure operability of the equipment during normal operation. The HVAC System serving this area is designed to:

a) Maintain 75°F within the Refueling Communications Center during normal operation.

b) Maintain a positive pressure within the Refueling Communications Center.
49

c) Maintain the ventilation rate within the Refueling Communications Center.

9.6.3.1.3 Radioactive Waste Treatment Area HVAC System

The Radioactive Waste Treatment Area HVAC System (Figure 9.6-10) is a non-safety related system that provides filtered and conditioned air to the Radioactive Waste Treatment Areas, to permit continuous routine access to personnel and to ensure operability of the equipment during normal operation. The HVAC System serving these areas is designed to:

a) Maintain 95^{o}-110^{o}F within the Radioactive Waste Treatment Area during normal operation except for the Control Room which will be maintained at 75^{o}F.

b) Maintain the ventilation rate within the Radioactive Waste Treatment Area during normal operation.

c) Maintain a negative pressure within the decontamination areas during normal operation.

d) Limit the spread of airborne radioactive materials within the accessible areas of the Radioactive Waste Treatment
59 Area below the limits set forth in 10CFR20.

9.6.3.2 System Description

The RSB HVAC System P&ID is shown on Figures 9.6-7 through 9.6-10. The classification of the RSB HVAC system components and their primary parameters are indicated in Table 9.6-5.

1. Fuel Handling Area (RSB-FHA)

The RSB Fuel Handling Area (FHA) HVAC System is a minimum outside air system consisting of an air handling unit, two (2) 50% capacity supply fans and two (2) 50% capacity exhaust/return fans with associated duct-work, dampers, valves, and instrumentation and controls. The air handling unit, supply and exhaust/return fans are located in a tornado hardened and missile protected Seismic Category I enclosure situated in the hardened portion of the RSB. The air handling unit supplies conditioned air via supply ductwork to the normal atmospheric areas of the RSB (FHA) in order to satisfy the ventilation requirements. The air handling unit filters maintain the cleanliness of the supply air.

Cooling and heating coils provided in the air handling unit along
49 with their instrumentation and controls, maintain the Indoor Design
Conditions. Two (2) 50% capacity exhaust/return fans are provided

with ductwork for the exhaust or recirculation of air supplied to
the atmospheric areas of the RSB-FHA. The exhaust air is filtered
by one of two 100% capacity RSB Clean Up Filter Units and then dis-
59 charged to the atmosphere. The exhaust/return fans are also pro-
vided with inlet vane dampers to maintain 1/4" W.G. negative pressure
of the RSB as required. Dampers, air outlets, instrumentation and
controls are provided to make the system complete and operate as
59 required.

The RSB HVAC System will maintain a minimum of 1/4" W.G. negative pressure in the RSB. The pressure in various building areas is monitored continuously to verify the maintenance of the specified negative pressure.

The system is automatically capable of functioning in the different operating modes outlined below:

a. <u>Normal plant operation</u> - During this mode, minimum outside air
is drawn through the outside air intake and mixes with the
return air before being conditioned and directed to the various
normal atmospheric areas of the RSB by the air handling unit,
associated supply fan and ductwork. A minimum 1/4" W.G.
negative pressure is maintained in the normal atmospheric
areas of the RSB by the exhaust/return fans. The pressure in
the various building areas is monitored continuously to verify
the maintenance of the specified negative pressure. Failure
of negative pressure maintenance is monitored and alarmed
locally and remotely alarmed in the Control Room. The exhaust
air from the building is discharged to the outside environment
through a missile protected exhaust structure located on the
RSB roof. The exhaust air is filtered through the RSB Clean Up
59 Filter Units before being discharged to the atmosphere.

59 b. Each filter-fan unit consists of a demister, an electric
heating coil, a bag type prefilter, a HEPA bank, an adsorbent
filter bank, and an after HEPA filter bank. Downstream of
each of the filter units are the RSB Clean Up Filter Unit
59 Fans provided with an automatic inlet valve control, and a
discharge damper.

59 b. <u>Open Hatch Refueling</u> - When the containment vessel refueling hatch is opened initially for refueling operations, the outside air dampers to the RSB are closed. The RSB Cleanup System filter-fan unit, in conjunction with the RSB return
59 fans, and the annulus filter system fans pulls down the pressure in the two buildings, in order to provide a minimum 1/4" W.G. negative pressure in the RCB and RSB after the containment vessel refueling hatch is opened. Once the required negative
59 pressure level is achieved, the outside air intake dampers will be opened to provide the required ventilation rate, and the system will operate as during normal operation.

59 A vent line with a manual valve is provided from the equipment and personnel airlock to the intake duct of the Annulus Filtration System to eliminate the airlock's bypass leakage.

The hardened RSB railroad door shall be closed during normal operation except when fuel is being transported or waste products are being removed from the RSB. At these times, the doors on both the RCB/RSB equipment and personnel air locks
59 shall remain closed, and the vent line manual valve from the equipment and personnel air lock to the Annulus Filtration System will be opened, venting the RCB bypass leakage into the annulus. The railroad door will not be open during refueling operation.

c. <u>Accident Conditions</u> - During accident conditions, the RSB HVAC System is switched to a recirculation mode, the outside air damper is closed and the normal supply and exhaust air flow to
59 the areas where the accident has occurred are isolated. The filter-fan unit draws exhaust air directly from the areas where the accident has occurred by the use of a separate exhaust duct system. After the contaminated air is filtered, a portion is exhausted to the outside environs at the roof of the RSB and the remaining quantity of filtered air is returned
59 to the RSB operating floor. The RSB Cleanup System filter-fan unit with the associated ductwork, valves and dampers maintains a greater negative pressure within the affected areas than the rest of the building to limit the spread of airborne radioactivity or sodium combustion products.

The air handling unit with the two (2) 50% capacity supply fans are located at El. 733'-0" of the RSB. The air handling unit is connected to an outside air intake plenum and also
49 connected to the Exhaust/Return Fans by ductwork. The air

handling unit consists of a mixing box with mixing dampers, roll filter, bag filter, heating and cooling coils, and access sections for maintenance. Downstream of the cooling coil section, a sufficiently long end access section (inlet plenum) is provided for the connection of the two 50% capacity supply fans. The length of the inlet plenum section is selected to permit equalization of the air flow through the cooling coils required by the off-center location of the fans. The length of the access sections is determined by the maintenance requirements of the individual components. The inlet plenum is connected to the fan inlets by manual dampers, transition pieces and flexible connections. The discharge sides of the fans are connected together by flexible connections, transition pieces, automatic isolation dampers, and "Y" connection to the main supply duct. From the main supply duct, branch ducts are connected to various cells.

The exhaust ducts from the cells are connected to the inlet plenum of the exhaust fans. Fan inlets are connected to the inlet plenum by manual dampers, transition pieces, and flexible connections. The discharge sides of the fans are connected together by flexible connections, transition pieces, automatic isolation dampers and "Y" connection to a single exhaust duct. The single exhaust duct is split into two, one for the exhaust
49 and one for the return. The return air is ducted back to the air handling unit. The exhaust air is passed through one of two 100% capacity RSB clean-up filter units and RSB filter exhaust fans. Automatic dampers and duct work direct the exhaust air from the fans to a missile protected exhaust
59 structure located on the RSB roof.

2. Refueling Communication Center

The RSB Refueling Communication Center, temperature and pressure requirements are maintained by a separate Air Conditioning Unit. The cleanliness of the air supplied by this unit is maintained by the filter section of the unit. The cooling coils, instrumentation and controls associated with the Air Conditioning Unit, maintain the indoor design conditions. The supply ductwork with dampers and air outlets provides cool air to satisfy the ventilation requirements.

Manual dampers provided on the supply and relief air registers are balanced to maintain a relative positive pressure in the room.

The Air Conditioning Unit is located at El. 865'-0" and is provided with an outside air connection. The A/C Unit consists of a mixing box with dampers, a roll filter, a cooling coil, and a V-belt driven centrifugal fan. Supply and return ducts from the Refueling
49 Communication Center are provided for circulating the air.

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3. RAPS and CAPS Exhaust and Unit Cooler System

The Radioactive Argon Processing System (RAPS) and the Cell Atmosphere Processing System (CAPS) cells are maintained under negative pressure during Normal Operation. 100% redundant Exhaust Fans are provided for the cells. Air infiltrated from the normal atmospheric areas of the RSB-FHA is exhausted by the RAPS and CAPS
53 59 Cells exhaust fans.

If the radiation monitor, provided in the exhaust duct, senses abnormal radioactivity, the quick closing valve closes and isolates the cells from the RAPS and CAPS Cells exhaust system. The Inert
53 Gas Receiving and Processing System (Section 9.5) will determine which cell has a radiation leak such that the RAPS and CAPS Cells exhaust system can manually re-instate the maintenance of the
53 negative pressure to the other RAPS and CAPS cells. The leaking cell is kept isolated from the RAPS and CAPS Cells' exhaust system until that cell is decontaminated.

A separate branch de-inerting exhaust duct, connected to the intake plenum of RAPS and CAPS cells exhaust fans, is provided with capped inlets at each level below El. 816'-0". These capped inlets are used for purging the inerted cells and for ventilating and cooling the RAPS and CAPS Cells when they are open for maintenance.

If purging is required for one of the inerted cells, a portable filter-fan unit will be connected by a flexible duct to a normally blanked-off connection located on the Recirculating Gas Coolant System return line from the cell and to the closest inlet connection of the exhaust duct. After the portable filter-fan unit is started, the blind flange located on the Recirculating Gas Coolant System supply line to the cell will be removed. The portable filter-fan unit will draw air into the cell through the opened connection and will replace the inert gas in the cell with air. When reinerting of the cell becomes necessary, the blind flange will be replaced and inerting gas will be introduced into the cell. After the cell atmosphere is completely replaced by the inert gas, the portable filter-fan unit will be stopped, and disconnected from the Recir-
53 culating Gas Coolant System return line, the flexible connection will be removed from the exhaust duct and the duct opening will be recapped. A radiation monitor provided at the discharge side of the portable filter-fan unit will stop the purging operation if
49 high radiation levels are sensed.

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The exhaust from CAPS and bleed off from the Inert Gas Receiving
59 and Processing System is exhausted by the RAPS and CAPS Cell exhaust fan. Bleed off from this system is monitored by the Inert Gas Receiving and Processing System before it is connected to the Inerted Cell Booster Fan and then exhausted to the atmosphere by
49 the RAPS and CAPS Exhaust Fans.

When maintenance is required in one of the RAPS & CAPS cells, the cell purging port is opened and air from the FHA will infiltrate through the port opening to purge the cell. After the cell atmosphere is completely replaced by infiltrated air, the purging port is closed and the cell door is opened for maintenance. When maintenance in that cell is completed, the cell door is closed and
53 negative pressure in this cell is re-instated.

A Unit Cooler is provided to cool the potentially contaminated
53 cells (365, 366, 371). Air is recirculated through the cells to prevent the spread of airborne radioactive materials to the building. Negative pressure is maintained in the cells by a branch duct connected to the RAPS and CAPS exhaust system. For initial start-up only, the unit cooler filters maintain the cleanliness of the supply air. The cooling coil, along with its instrumentation and controls, maintain the Indoor Design Conditions.

The Unit Cooler serving the RAPS and CAPS Cells is located at
53 El. 755'-0". The unit consists of a disposable filter, cooling coil and V-belt driven centrifugal fan. Supply and return air
49 ducts from the unit to the cells are provided.

The RAPS and CAPS cells redundant exhaust fans are located in the SW side of the RSB at El. 779'-0". The exhaust fans are connected to a "Y" connector on the suction and discharge sides. Manual dampers, transition pieces, and flexible connections are provided in the suction sides. On the discharge sides, flexible connections, transition pieces, and automatic isolation dampers are provided. The exhaust fans are vaneaxial fans.

A main exhaust duct, provided with a radiation monitor and an automatic isolation damper, is connected to the exhaust fans. Branch ducts from RAPS and CAPS cells are connected to this main exhaust duct. The exhaust ducts from each cell are provided with manual isolation valves and check valves.

For purging the inerted cells in the fuel handling areas, a separate deinerting exhaust duct is connected to the main exhaust duct of the RAPS and CAPS cells exhaust fans. Capped purging inlets are provided on this separate deinerting exhaust duct at each floor elevation of the RSB-FHA. An inert gas exhaust line from Inert Gas Receiving and Processing System is also connected to this de-
59 inerting exhaust duct, through the Booster Fan.

A separate exhaust duct from Inert Gas Receiving and Processing System CAPS is also connected to the main exhaust duct.

On the discharge side of the exhaust fans, downstream of the "Y" connector, an exhaust duct is provided with a radiation monitor
59 and runs vertically upward to the missile protected exhaust structure.

4. ABHX Cooling System

Two (2) unit coolers are provided for the two ABHX cells (352A,
59 353A). The unit cooler filters maintain the cleanliness of the supply air (for initial start-up only). The cooling coils of these unit coolers, along with their instrumentation and controls, main-
59 tain the Indoor Design Conditions. The supply air provided to the cells from the Air Handling Unit maintain the ventilation requirements.

The 1st and 2nd loop ABHX cell unit coolers are located above the ABHX cells. Each of these unit coolers consists of a disposable filter, a cooling coil, and a V-belt driven centrifugal fan.
49 Supply ducts to the ABHX cells are provided for distribution of air.

A Unit Heater is provided for the ABHX Cell 332 to maintain the Indoor Design Conditions.

5. a. <u>RSB and Annulus Filter Cells Cooling System</u>

Four (4) unit coolers are provided for cooling the cells containing the RSB and annulus filter units. Each unit cooler consists of a disposable filter, a cooling coil and a V-belt driven centrifugal fan. Each unit cooler is located inside the cell it is serving.

The unit coolers recirculate and cool the air in the cells that house the filter equipment. Supply air from the RSB-FHA HVAC System provides the necessary ventilation requirements. The unit coolers, along with their instrumentation and controls,
49 maintain the Indoor Design Conditions.

b. <u>Electrical Equipment Cells Cooling Systems</u>

Two (2) unit coolers are provided for cooling cells 305E & F. Each unit cooler consists of a disposable filter (for initial start-up only), a cooling coil, and a V-belt driven centrifugal fan. It is located within the cell it serves. Supply ducts are provided to distribute the air.

The unit cooler filters maintain the cleanliness of the supply air (for initial start-up only). The cooling coils of these unit coolers, along with their instrumentation and controls, maintain the Indoor Design Conditions.

The supply air provided to the cells from RSB-FHA HVAC System maintains the ventilation requirements.

c. <u>Containment Clean-up Pump Cells Cooling Systems</u>

Two (2) unit coolers are provided for cooling cells 305I & G. Each unit cooler consists of a disposble filter (for initial start-up only), a cooling coil, and a V-belt driven centrifugal fan. It is located within the cell it serves.

The unit cooler filters maintain the cleanliness of the supply air (for initial start-up only). The cooling coils of these unit coolers, along with their instrumentation and controls, maintain the Indoor Design Conditions. The supply air provided to the cells from the RSB-FHA HVAC system maintains the ventilation
59 requirements.

d. Containment Clean-up Scrubber Cell Cooling System

Two (2) 100% capacity unit coolers are provided for cell 359. Each unit cooler consists of a disposable filter (for initial start-up only), a cooling coil, and a V-belt driven centrifugal fan. They are located within the cell 359.

The unit cooler filters maintain the cleanliness of the supply air (for initial start-up only). The cooling coils of these unit coolers, along with their instrumentation and controls, maintain the Indoor Design Conditions. The supply air provided to the cells from the RSB-FHA HVAC system maintains the ventilation requirements.

e. Containment Clean-up Pipe Chase Valve Areas Cooling Systems

Two (2) unit coolers are provided for cells 348 & 349. Each consists of a disposable filter (for initial start-up only), a cooling coil, and a centrifugal fan. It is located within the cell it serves.

The unit cooler filters maintain the cleanliness of the supply air (for initial start-up only). The cooling coils of these unit coolers, along with their instrumentation and controls, maintain the Indoor Design Conditions. The supply air provided
59 to the cells from the RSB-FHA HVAC system maintains the ventilation requirements.

6. Radioactive Waste Treatment Area (RSB-RWA)

A 100% outside air, Air Handling Unit with two (2) 50% capacity Supply Fans is provided for the RSB Radioactive Waste Treatment Area. The supply ductwork distributes the conditioned supply air to the various areas to satisfy the Ventilation Requirements. Filters are provided for the air handling unit to maintain the cleanliness of the supply air. The air handling units cooling and heating coils, along with their instrumentation and controls, maintain a constant discharge temperature downstream of the supply fans. The branch duct serving the areas at El. 816'-0" is provided with a reheat coil. A separate reheat coil is provided in the branch duct serving the Control Room and Motor Control Center. The reheat coils with their instrumentation and controls, maintain the Indoor Design Conditions all-year-round.

Two (2) Hot Water Unit Heaters are provided for the railroad and truck areas at El. 816'-0", and two (2) Hot Water Unit Heaters are provided for the HVAC Equipment RM at Elev. 867'-0". The unit heaters, along with their instrumentation and controls, maintain
49 the Indoor Design Conditions.

The areas at El. 795'-0" and 775'-0" are not provided with space
temperature control since the cooling load is constant. Two (2)
50% capacity Exhaust Fans are provided. The exhaust fans and
connecting exhaust ductwork remove the air supplied to the RSB-RWA.
An Exhaust Filter Unit with one 100% capacity exhaust fan is pro-
vided to clean the potentially contaminated air before releasing it
to the atmosphere. A radiation monitor is provided in the main
exhaust duct. If abnormal radioactivity is detected, the exhaust
air is directed to the exhaust filter unit and exhausted by an
exhaust fan to the atmosphere. Dampers, air outlets, instrumentation
and controls are also provided to make the system complete and
49 operate as required.

Negative pressure shall be maintained within the decontaminated areas with respect to the other areas of the Radwaste Building. The exhaust fans are provided with automatic inlet vanes to maintain the required negative pressure within the building with
59 respect to atmosphere. The supply fans and supply dampers provide constant air supply to the various areas inside the building to maintain the Indoor Design Conditions.

The 100% Outside Air System Air Handling Unit with two (2) 50% capacity Supply Fans is located at El. 867'-0" of the Radioactive Waste Treatment Area of the RSB. The air handling unit is connected to an inlet louver located in the south wall of the building and consists of an intake damper, roll and bag filters, heating coil, cooling coil and access sections. Downstream of the cooling coil section a sufficiently long end access section is provided for the connection of the supply fans. The inlet plenum is connected to the fan inlets by manual dampers, transition pieces and flexible connections.

The discharge sides of the fans are connected together by flexible connections, transition pieces, automatic isolation dampers and "Y" connections to the main supply duct.

59 The branch supply duct serving the Control Room is provided with a reheat coil and is connected to the main supply duct. Another branch supply duct serving the decontamination area and the operating
59 floor at El. 816'-0", and the motor control center at Elev. 833'-0" of the Radwaste Building is provided with a reheat coil and is connected to the main supply duct. Separate branch supply ducts are provided without reheat coils to serve the other areas at El. 816'-0" and below.

The supply fans are direct driven vaneaxial fans.

Exhaust ductwork is provided for the different areas and connected to the main exhaust duct. The main exhaust duct is provided with a radiation monitor and a "Y" connector. One leg of the "Y" connector is connected to the exhaust filter unit by an automatic damper. The exhaust filter unit consists of bag filters, HEPA filters and access sections. The inlet of the exhaust filter fan is connected to the outlet of the exhaust filter unit by a transition piece and ductwork. The exhaust filter fan is of the centrifugal type and is provided with an automatic variable vortex damper. Exhaust ductwork is provided at the discharge side of the fan and connected to the weather protected exhaust structure located at the roof. The other leg of the "Y" connector is connected to an inlet plenum by an automatic damper and exhaust ductwork. Two (2) 50% capacity exhaust fans are provided and connected to the inlet plenum by manual operated dampers, transition pieces, and flexible connections. The two (2) 50% capacity exhaust fans are of the vaneaxial type
49 and are provided with automatic variable inlet vane dampers. The

discharge sides of these fans are connected to the weather protected exhaust structure by flexible connections, transition pieces, automatic dampers, "Y" connector, and exhaust ductwork.

9.6.3.3 Safety Evaluation

The RSB HVAC System is designed to minimize the spread of airborne contamination within the buildings and to the outside atmosphere.

59 During normal plant and open hatch refueling operation and
accident conditions, the RSB shall be maintained at a minimum 1/4" W.G.
negative pressure with respect to the outside atmosphere. RSB building
pressure is to be monitored continuously, with alarms provided to
59 indicate abnormal conditions. Capability is provided to filter the RSB
exhaust during all operations with the RSB Cleanup System filter fan
units. Safety Class 3 ductwork, valves, dampers, and associated controls are provided for cross connection with the RSB Cleanup System
filter-fan units. This portion of the system is an Engineered Safety
Feature because it is used to mitigate the consequences of the Site
59 Suitability Source Term (SSST) event, and is described further in Section
6.2.6.

The exhaust air from each of the potentially contaminated RAPS and CAPS cells will be continuously monitored. Upon detection of high radioactivity in the cell exhaust, the cell is automatically isolated, and the exhaust from the affected cell is vented to CAPS for processing prior to discharge.

Differential pressures will be maintained in the radioactive waste handling area operating areas and potentially contaminated cells and pipeways, to create air flow in the direction of increasing contamination potential. Dampers and air control devices will be provided to prevent cross contamination between building spaces.

An evaluation of the considered failure mode follows:

1. Single Failure of RSB HVAC Components

Active components in the RSB HVAC System which are susceptible to failure are as follows:

RSB-FHA Supply Fans
RSB-FHA Exhaust Fans
RSB-FHA Automatic Roll Filters
RSB-RAPS, CAPS Cells Exhaust Fans

RSB-RWA Supply Fans
RSB-RWA Exhaust Fans
49 RSB-RWA Automatic Roll Filters

Refueling Communication Center A/C Unit Supply Fan
Refueling Communication Center A/C Unit Automatic Roll Filter
ABHX Cell Unit Coolers Supply Fan
49 RAPS and CAPS Unit Cooler Supply Fans
RSB Clean Up Filter Fan
Annulus Filter Cell Unit Cooler Supply Fan
RSB Clean Up Filter Cell Unit Cooler Supply Fan
Electrical Equipment Cell Unit Cooler Supply Fan
Containment Clean Up Pump Cell Unit Cooler Supply Fan
Containment Clean Up Pipe Chase Unit Cooler Supply Fan
59 Containment Clean Up Scrubber Cell Unit Cooler Supply Fan

During normal operation the two supply and exhaust fans operate simultaneously for the RSB-FHA as well as for the RSB-RWA. Failure of any one supply or exhaust fan would automatically stop its respective exhaust or supply fan. During the loss of any supply/ exhaust fan pair in the RSB, the cooling capacities of the systems will be able to maintain the upset design temperature within the affected area.

59 The RSB FHA and RWA HVAC System Air Handling Units and the Refueling Communication Center A/C Unit, are provided with automatic roll type filters. Failure of the advance mechanism for the filter media would result in increased pressure across the filters. A sensing device is provided with an alarm setpoint to indicate that the differential pressure across any filter is higher than normal. This alarm setpoint is selected on the basis that after initiation of the alarm, 72 hours are available to correct the failure without significantly deviating from the system design parameters.

The RAPS and CAPS cells are provided with two (2) 100% redundant exhaust fans. Failure of the operating exhaust fan would automatically start the redundant exhaust fan.

Each of the ABHX cells is provided with one unit cooler. Failure of one unit cooler will send out an alarm.

49 Each of the Annulus and RSB Cleanup filter cells, Electrical Equip-
ment Cells, Containment Clean Up Pump Cells, and Containment Clean
Up Pipe Chase Cells, is provided with one unit cooler. Failure of
59 the operating unit cooler will automatically start the other redundant
system and consequently start the respective unit cooler serving
49 that cell.

Two (2) 100% capacity RSB Clean Up Filter Fans are provided. Failure
of the operating filter fan will initiate an alarm and start the
59 other redundant Clean Up Filter Fan.

The Containment Clean Up Scrubber Cell is provided with two (2)
59 100% redundant Unit Coolers. Failure of the operating unit cooler
will initiate an alarm and start the other redundant unit cooler.

2. Loss of Normal Electric Power

The HVAC System shall be designed to maintain the upset design
temperature within the Annulus and RSB Clean-Up Filter Cells, the
49 ABHX Cells, Electrical Equipment Cells, Containment Clean Up Pump
Cells, Containment Clean Up Pipe Chase, and the Containment Clean
59 Up Scrubber Cell of the RSB during the loss of normal electric
power.

The following components are connected to the on-site emergency
Class IE AC power supply. During the failure of normal electric
49 power supply, these components operate from the Class IE AC power
supply and maintain the upset design temperature.

Electrical Equipment Cell Unit Cooler
Containment Clean Up Pump Cell Unit Cooler
59 Containment Clean Up Pipe Chase Unit Cooler
Containment Clean Up Scrubber Cell Unit Cooler
ABHX Cells Unit Coolers
49 Annulus and RSB Filter Cell Unit Coolers

The RSB Clean Up Filter Fan is connected to Class 1E Power Supply and will continue to operate even during loss of normal power
59 supply.

3. Loss of Cooling from the Normal Chilled Water System

The unit coolers serving the ABHX Cells, the Annulus and RSB Filter
59 Cells, Electrical Equipment Cell, Containment Clean Up Pipe Chase, and Containment Clean Up Scrubber Cell will be connected to the emergency chilled water system. Upon the loss of normal chilled water, the unit coolers will be switched to the emergency chilled water system.

4. Occurrence of Fire in the Building

In the event of a fire in the building, a signal from the Fire Protection System will start the Stairwell Pressurization Fans to prevent the smoke from entering the stairwells.

The Smoke Vent Fan shall be started manually to provide smoke venting from the areas containing safety related equipment.

9.6.3.4 Testing and Inspection Requirements

All components are tested and inspected as separate components and as integrated systems. Velometer readings are taken to ensure that all systems are balanced to deliver and exhaust the required air quantities. All water coils are hydraulically tested for leakage prior to being placed in service. Capacity and performance of the fans are tested according to the Air Moving and Conditioning Association requirements prior to operation of the plant.

The RSB Clean Up Filter Units will be periodically tested in
52 accordance with Regulatory Guide 1.52.

59 9.6.4 Turbine Generator Building HVAC System

9.6.4.1 Design Basis

The objective of the ventilation system for the Turbine Generator Building (Figure 9.6-16) is to circulate the air in order to control the building air temperature. The system provides the required environment to ensure operability of the equipment and personnel accesses of the building. The system is not designed as a safety related system. The
49 ventilation system for this building is designed to:

59 a. Limit maximum air temperatures to the following: 75°F in the secondary plant sampling room, and 95°F in the remainder of the building.

b. To provide filtered outside air to reduce dust and airborne particles within the building.

c. To provide air movement from the lower elevations of the building to the final exhaust located under the building roof. The exhaust air is not filtered prior to discharge to the atmosphere.

d. To deliver the air to the building roof exhaust fans.

9.6.4.2 System Description

This section describes the major design features and the normal operation of the system, to satisfy the operational requirements.

Two (2) Recirculating Air Handling Units, each with two (2) 50% capacity supply fans, provide conditioned supply air to all levels of the Turbine Generator Building. The minimum fresh air flow provided by this system is established on the basis of the ventilation requirements. The supply air is distributed to the various areas by the supply ductwork to satisfy the temperature requirements. The filters provided in the Air Handling Units maintain the cleanliness of the supply air. Modulating outside air and return air dampers, the heating coil and the cooling coil provided in the air handling units, along with their instrumentation and controls, maintain the discharge temperature downstream of the supply fans.

The cooling coils are supplied with chilled water and the heating coils are supplied with hot water.

The two (2) recirculating air handling units, each with two (2) 50% capacity supply fans, are located in the NW corner at El. 892'-0" and in the NE corner at El. 862'-0" of the TGB respectively. One (1) Air Handling Unit is connected directly to a fixed louver in the north wall of the building. Each air handling unit consists of outside air and return air dampers, a roll type filter, heating and cooling coils and access sections. Downstream of the cooling coil section a sufficiently long front access section is provided for the connection of the 50% capacity supply fans. The length of the front access section is determined to permit equalization of the air flow through the cooling coils required by the off-center location of the fans. The length of the other access sections is determined by the maintenance requirements of the individual components. The flexible sections are connected to the front access section and are followed by manual dampers (normally locked open), fan inlet bell sections, fans, flexible connections and motor operated isolation dampers. The two (2) motorized damper sections are joined
49 together by a "Y" duct section.

Main supply ducts leaving the air handling units run in the Operating Floor El. 862'-0" parallel to column lines TC and TF to serve the west and east zones respectively. Branches from the main ducts are routed to the lower elevations to distribute the supply air to the various areas of the building.

Ten (10) roof exhaust fans are provided, with a total capacity equal to the total air supply, for the building. The three (3) exhaust fans serving the Chemical Storage Area, Lube Oil Storage Area and Operating Floor are sized to handle the minimum outside air and run continuously while the number of remaining exhaust fans that are operating is determined by the percentage of outside air in the total air supply for the building.

The Roof Exhaust Fans are located as follows:

2 Fans at Roof El. 861'-0"
1 Fan at Roof El. 878'-0"
5 Fans at Roof El. 910'-6"
2 Fans at Roof El. 921'-0"

A relief hood, located above the deaerator area, will relieve the air from the building and maintain the air balance during the various steps of exhaust fans operation.

Two (2) unit coolers are provided to serve the condensate pumps and L.P. Feedwater Heaters to supplement the main HVAC system and conserve energy during part load operation.

A separate Heating and Ventilating Unit at El. 816'-0" along with a roof exhaust fan at Roof El. 861'-0" serve the Ammonia Storage Room and operate continuously. The supply and exhaust air quantities are balanced to maintain slightly negative pressure inside the room.

The sampling room is served by a branch duct from one air
handling unit. Constant temperature and humidity are maintained inside
49 the room through a heating coil and a Steam Humidifier.

The caustic and acid storage room is served by a branch duct
from the main air handling system with a reheat coil that maintains the
59 required indoor design conditions.

The Radiation Monitoring System will provide the equipment necessary to sample and analyze tritium in the exhaust air released from the building to meet the requirements of 10CFR20.

9.6.4.3 Safety Evaluation

This section describes the design and operation of the TGB
49 HVAC System during single failure of the TGB HVAC System Components·

The TGB HVAC System consists of active and passive components.

This system design has no provisions for failures of passive components. Active components in the TGB HVAC System which are susceptible to failure are as follows:

Supply Fans
Exhaust Fans
Automatic Roll-Type Filters
Outside Air Dampers
Return Air Dampers
Exhaust Air Dampers
Unit Cooler Supply Fans

The TGB HVAC System is provided with five (5) supply fans and eleven (11) exhaust fans. Failure of any one supply or exhaust fan would not increase the average temperature in the affected area above 120°F.

The TGB Air Handling Units are provided with automatic roll-type filters with an automatic advance mechanism that advances the filter medium on the basis of sensed differential pressure across the filter. The failure of the advance mechanism results in increased pressure across the filters. A sensing device is provided for each filter with an alarm setpoint to indicate higher than normal differential pressure across the filter. The alarm setpoint is selected on the basis that after initiation of the alarm, 72 hours are available to correct the failure without significantly deviating from the system design parameters.

The steam generator feed pump area, the condensate pump area, the L.P. Feedwater Heater Area, are provided with two (2) unit coolers. Each unit cooler has one (1) 100% capacity centrifugal fan. Failure of any one unit cooler fan would not increase the space temperature in the immediate area above 120°F.

9.6.4.4 Testing and Inspection Requirements

All components are tested and inspected as separate components and as integrated systems. Velometer readings are taken to ensure that all systems are balanced to deliver and exhaust the required air quantities. All water coils are hydraulically tested for leakage prior to being placed in service. Capacity and performance of the fans are tested according to the Air Moving and Conditioning Association requirements prior to operation of the plant.

9.6.5 Diesel Generator Building HVAC System

9.6.5.1 Design Basis

9.6.5.1.1 Diesel Generator Rooms HVAC System

49 The Diesel Generator Rooms HVAC System is a safety related system designed to provide ventilation to the Diesel Generator Rooms

9.6.5.4 Testing and Inspection Requirements

All components are tested and inspected as separate components and as integrated systems. Velometer readings are taken to ensure that all systems are balanced to deliver and exhaust the required air quantities. All water coils are hydraulically tested for leakage prior to being placed in service. Capacity and performance of the fans are tested according to the Air Moving and Conditioning Association requirements prior to operation of the plant.

9.6.6 Steam Generator Building HVAC System

9.6.6.1 Design Basis

9.6.6.1.1 Steam Generator and Auxiliary Bay HVAC System

The Steam Generator and Auxiliary Bay HVAC System is a safety-related system designed to provide filtered and conditioned air to the Steam Generator Loop Cells, the Auxiliary Bay Cells and the Intermediate Bay IHTS Cells to permit continuous routine personnel access during normal operation and to ensure operability of the equipment under all conditions. The HVAC System serving these areas is designed to:

a) Maintain 100^{o}F max. within all areas during normal operation.
b) Maintain 120^{o}F max. within all areas under single failure of an HVAC System component.
c) Maintain 120^{o}F max. within all areas under loss of cooling from the Normal Chilled Water System.
d) Maintain the ventilation rate within all areas under all operating conditions.
e) Comply with the single failure criterion of Regulatory Guide 1.53.
f) Operate from the Class IE AC power supply during loss of off-site power.
49 g) Maintain 120^{o}F max. within the Auxiliary Feedwater Pump Cells during off-normal conditions.
h) Provide ducted cool air directly to the lube oil cooling panels.
i) Provide exhaust ductwork for the Intermediate Sodium Pump Drive to exhaust hot discharge air directly outside to atmosphere.
59 j) Provide ducted exhaust from the Intermediate Sodium Cold Trap.

9.6.6.1.2 Steam Generator Building Intermediate Bay HVAC Systems

The Steam Generator Building Intermediate Bay HVAC Systems
49 are safety related systems designed to provide filtered and conditioned

air throughout the Intermediate Bay (except IHTS cells) to permit continuous routine personnel access and to ensure operability of the equipment during normal operation. The HVAC System serving these areas is designed to:

a) Maintain 95^{o}F max. within all areas during normal operation.

49 b) Maintain 120^{o}F max. within all areas under single failure of an HVAC system component.

c) Maintain 120°F max. within the Primary Sodium Storage Tank Cell when the Primary Sodium Storage tank is full.

d) Maintain 105°F max. within the Emergency Chilled Water Equipment Room during off-normal conditions.

e) Maintain the ventilation rate throughout the Intermediate Bay during normal operation.

f) Maintain the temperature in the immediate vicinity of the Remote Shutdown Panel less than or equal to 110°F to insure long-term minimum habitability following loss of
49 off-site power.

g) Maintain 120°F maximum within all areas under loss of
59 cooling from the Normal Chilled Water System.

9.6.6.1.3 Steam Generator Building Maintenance Bay HVAC System

The Steam Generator Building Maintenance Bay HVAC System is a non-safety related system designed to provide conditioned air throughout the Maintenance Bay, to permit personnel access and to ensure operability of the equipment during normal operation. The HVAC system serving the Steam Generator Building Maintenance Bay is designed to:

a) Maintain 105°F max. throughout the Maintenance Bay during normal operation.

b) Maintain the ventilation rate throughout the Maintenance Bay during normal operation.

9.6.6.2 System Description

The SGB HVAC System PI&Ds are shown on Figures 9.6-12 through 9.6-15. The classification of the SGB HVAC system components and their primary parameters are indicated in Table 9.6-8.

1. Steam Generator and Auxiliary Bay (SGB-SB, SGB-AB) System

Three (3) identical recirculating type Air Handling Units each with its two (2) 100% capacity redundant Supply Fan provides conditioned supply air for the three (3) Steam Generator Loop cells, the IHTS cells (SGB-IB, El. 765'), and the Auxiliary Bay Cells (SGB-AB). The supply air is distributed to each loop by an independent ductwork system to satisfy the ventilation requirements. The Air Handling Unit filters maintain the cleanliness of the outside air supply. The cooling coils provided in the air handling units, along with their instrumentation and controls maintain the supply air temperature to satisfy the Indoor Design Conditions. Two (2) 100% capacity redundant Exhaust Fans along with their independent exhaust ductwork are provided for the return and exhaust of air supplied to the areas of each loop. The IHTS pump motor draws the air required for cooling from the cell and discharges the hot air to the atmosphere through the steam vent structure. Dampers, air outlets, instrumentation and controls are provided to make the systems complete and
49 operate as required. The chilled water supplied to the air handling units is provided by the Normal Chilled Water System.

filters maintain the cleanliness of the air supply. The heating and cooling coils provided in the Air Handling Unit, along with their instrumentation and controls, maintain a constant discharge temperature downstream of the supply fans to satisfy the Indoor Design Conditions. Two (2) 100% capacity Exhaust Fans along with exhaust ductwork are provided for the return and exhaust of air supplied to the SGB-IB. Dampers, air outlets, instrumentation and controls are provided to make the system complete and operate as required.

The first SGB-IB Air Handling Unit with two (2) 100% supply fans is located in the east side of the SGB-Intermediate Bay El. 836'-0". The air handling unit is connected to a missile protected intake structure located on the east side of the SGB-Intermediate Bay roof. The air handling unit consists of a mixing plenum with outside and return intake dampers, pre and after filter, heating coil, cooling coil and access sections. Downstream of the cooling coil section a sufficiently long end access section is provided for the connection of the supply fans. The length of the end access section is selected to permit equalization of the air flow through the cooling coils required by the off-center location of the fans. The length of the other access sections is determined by the maintenance requirements of the individual components. The fan inlet bell sections are connected to the end access section and are followed by flexible connections, manual dampers (normally locked open), fans, flexible connections and automatic isolation dampers. The two (2) automatic damper sections are joined together by a "Y" duct section. The "Y" duct section connects to supply ductwork which serves the respective cells.

The return air is either transferred or ducted, to exhaust fans located at El. 836'-0". The discharge from the exhaust fans is either returned to the cell for recirculation or exhausted out to the atmosphere.

The second of the two recirculating type Air Handling Units with
two (2) 100% capacity supply fans provides conditioned supply air
to all areas in the SGB-IB except those served by the SGB Loop Cell
59 System and the Primary Sodium Storage Pump Cell. The supply air is
distributed by supply ductwork to the various areas to satisfy the
Ventilation Requirements.

The Air Handling Unit filter maintains the cleanliness of the air
supply. The heating and cooling coils provided in the air handling
unit, along with their instrumentation and controls, maintains a
constant discharge temperature downstream of the supply fans. The
areas located below El. 836'-0" are not provided with individual
space temperature controls because the cooling requirements are
relatively constant. Two (2) 100% capacity Exhaust Fans along with
exhaust ductwork are provided for the exhaust and return of air
supplied to the SGB-IB areas. Dampers, air outlets, instrumentation
and controls are provided to make the system complete and operational
49 as required.

The second SGB-IB Air Handling Unit with two (2) 100% supply fans is located in the west side of the SGB-Intermediate Bay at El. 816'-0". The air handling unit is connected to a missile protected intake structure located on the west side of the SGB-Intermediate Bay roof. The air handling unit consists of a mixing plenum with outside and return intake dampers, pre and after filters, heating coil, cooling coil and access sections. Downstream of the cooling coil section a sufficiently long end access section is provided for the connection of the supply fans. The length of the end access section is selected to permit equalization of the air flow through the cooling coil required by the off-center location of the fans. The length of the other access sections is determined by the maintenance requirements of the individual components. The fan inlet bell sections are connected to the end access section and are followed by flexible connections, manual dampers (normally locked open), fans, flexible connections, and automatic isolation dampers. The two (2) automatic damper sections are joined togehter by a "Y" duct section. The "Y" duct section connects with supply ductwork which serves the respective cells.

The exhaust plenum is connected to two (2) Exhaust Fans with manual dampers (normally locked open), flexible connection and inlet cone. The discharge side of each fan is connected to a flexible connection followed by ductwork and an automatic isolation damper. The automatic damper section is connected to the common discharge duct by a "Y" duct connector. The discharge is either returned to the cell for recirculation or exhausted out to the atmosphere.

The two exhaust ducts from exhaust fans join into a common exhaust shaft and discharge exhaust air to the atmosphere through a missile protected exhaust structure located at the southeast corner of the SGB-IB at El. 857'-6".

Two (2) Unit Coolers are provided, one for each of the Emergency Chilled Water System Equipment Rooms. Each Unit Cooler is provided with a V-belt driven centrifugal fan. The operation of the Unit Cooler fans is interlocked with the operation of the Emergency Chilled Water equipment.

One Unit Cooler is located in Cell No. 216 of the SGB-IB at El. 733'-0". The other Unit Cooler is located in Cell No. 217 of the SGB-IB at El. 733'-0". Each unit cooler consists of a disposable filter, a cooling coil and a V-belt driven centrifugal fan. The unit coolers are free standing and are not provided with ductwork.

One (1) Unit Cooler is provided for cooling of the Primary Sodium Storage Tank Cell. The cycling of the unit cooler fan provides heat removal capacity control.

49

Amend. 49
April 1979

59 Supply and exhaust ducts with normally closed dampers are provided to ventilate the cell whenever maintenance or personnel accessibility is required. Connections to the Inert Gas Receiving and Processing System nitrogen piping are provided in the supply and exhaust duct to provide capability of inerting the cell. The Unit Cooler is located at El. 733'-0" in Cell No. 211 of the SGB-IB. It consists of a disposable filter, a cooling coil and a V-belt driven centrifugal fan. The fan discharge is connected to a duct which supplies the air to the cell. The return air to the unit Cooler is not provided with ductwork.

Three (3) Unit Coolers are provided, one for each I&C Panel Cell (272A, B and C). The unit coolers will maintain the Upset Design Condition in these cells in the event of fire in the SGB-IB. The three (3) unit coolers are located in the three I&C panel cells located at the SGB-IB El. 836'-0". Each unit cooler consists of a disposable filter, cooling coil, and a V-belt driven centrifugal fan.

3. Maintenance Bay (SGB-MB) System

A recirculating type Air Conditioning Unit provides conditioned outside air and return air mixture to the various areas in the SGB-Maintenance Bay. The supply air is distributed by ductwork to the various areas, to satisfy the Ventilation Requirements. The Air Conditioning Unit filter maintains the cleanliness of the supply air. The heating and cooling coils provided in the air conditioning unit along with their instrumentation and controls maintain a constant discharge temperature downstream of the supply fan to satisfy the Indoor Design Conditions. An exhaust louver with gravity damper is provided for the exhaust of the air supplied to
59 the SGB-M. Dampers, air outlets, instrumentation and controls are provided to make the system complete and operate as required.

The Maintenance Bay Air Conditioning Unit is located in the east side of the SGB-MB at El. 816'-0". The air conditioning unit is connected to an inlet Louver followed by an automatic damper located in the east wall of the Maintenance Bay. The air conditioning unit consists of a roll type filter, a heating coil, a cooling coil, a V-belt driven centrifugal fan and access sections. The lengths of the access sections provided for each component are determined by the maintenance requirements of the individual components. The supply ductwork from the air conditioning unit splits
49 into branch ducts to serve the areas above and below El. 816'-0".

59 9.6.6.3 Safety Evaluation

The SGB HVAC System is provided with sufficient redundancy to satisfy the temperature and ventilation requirements stated in Section 9.6.6.1 of the PSAR. All safety-related systems are provided with proper separation to prevent common mode failures. Safety-related ductwork and equipment is provided with Seismic Category I duct and equipment supports.

Single Failure Analyses for the safety-related SGB HVAC System are presented in Table 9.6-9. The evaluation of failure modes considered are as follows:

1. Single Failure of SGB HVAC System Components

   This system design has no provisions for failure of passive components. The active components in the SGB HVAC System which are susceptible to failure are as follows:

IB	Supply Fans
IB	Exhaust Fans
IB	Automatic Roll-Type Filters
SB-AB	Supply Fans
SB-AB	Exhaust Fans
SB-AB	Automatic Roll-Type Filters
IB	I&C Cell Unit Coolers
IB	Emergency Chilled Water Equipment Room Unit Coolers
59 IB	Primary Na Tank Cell Unit Coolers
SB-AB	Auxiliary Feedwater Pump (Electric Driven) Unit Coolers
SB-AB	Auxiliary Feedwater Pump (Turbine Driven) Unit Coolers
49 SB-AB	Outside Air Supply Fan
59 IB, SB-AB	Automatic and Remote Operated Dampers

Amend. 59
Dec. 1980

The two (2) Air Conditioning Systems serving the SGB-IB except ex-containment sodium storage tanks are each provided with two (2) 100% capacity redundant supply and exhaust fans. During the loss of any supply or exhaust fan pair, its redundant fan will start to maintain normal system operation. The motors for these fans are connected to the on-site Class 1E AC power supply. If normal electrical supply is interrupted, the fans are able to operate from the Class 1E AC power supply.

The SGB-SB/AB HVAC Systems are provided with 100% capacity redundant supply and exhaust fans. If any one fan fails, its redundant fan will start to maintain normal system operation. If the normal electric supply is interrupted, the fans are able to operate from the Class IE AC power supply to which they are connected.

The SGB-IB/SB/AB Air Handling Units automatic roll-type filters are provided with an automatic advance mechanism which advances the filter media on the basis of sensed differential pressure across the filter. The failure of the advance mechanism results in increased pressure across the filters. A sensing device is provided for each filter with an alarm setpoint to indicate higher than normal differential pressure across the filter. The alarm setpoint is selected on the basis that after initiation of the alarm, 72 hours are available to correct the failure without significantly deviating from the system design parameters.

The (100% capacity) emergency chilled water system equipment rooms are served by two (2) unit coolers, one for each room. If any of the unit coolers fail, the operating chiller served by this unit cooler must be shut down and the (100%) capacity redundant chiller will be used.

The (100% capacity) Turbine Driven Auxiliary Feedwater Pump Cell is served by two (2) 100% capacity unit coolers. If the operating unit cooler fails, the redundant unit cooler will automatically start.

The (50% capacity) Electric Driven Auxiliary Feedwater Pump Cells are served by two (2) unit coolers, one for each cell. If any of the unit coolers fail, only one electric driven feedwater pump can operate and the redundant turbine driven auxiliary feedwater pump should be used if required.

The Automatic and Remotely Operated Dampers are provided with remote position indicators, alarms and manual operators. The failure of any damper can be detected, identified and corrected within 2 hours. During this time period the temperature of the space affected by the damper failure will be maintained under
49 120°F.

2. Loss of Normal Electric Power

This system shall be designed to maintain the Upset Design Temperature within the SGB-IB (except ex-containment sodium storage tank), Steam Generator Bay and Auxiliary Bay during the loss of normal electric power supply.

The following components are connected to the on-site emergency Class IE AC power supply. During the failure of normal electric power supply, these components operate from the on-site emergency Class IE AC power supply and maintain the Upset Design Temperature:

SB-AB	Six (6) Supply Fans
SB-AB	Six (6) Exhaust Fans
AB	Four (4) Auxiliary Feedwater Pump Unit Coolers
IB	Four (4) Supply Fans
IB	Four (4) Exhaust Fans
IB	Two (2) Emergency Chiller Room Unit Coolers
IB	Three (3) I&C Panel Unit Coolers

3. Loss of Normal Chilled Water

This system shall be designed to maintain the Upset Design Temperature within the SGB-IB (except ex-containment sodium storage tank), Steam Generator Bay and Auxiliary Bay during the loss of normal chilled water. Both supply and exhaust fans will operate to permit maintenance of the UPSET conditions within the above areas with ventilation provided only if the Normal Chilled Water System fails.

The five (5) air handling units are designed with a bypass section.
During the failure of normal chilled water, an increase in the
49 supply air quantity is required to maintain the Upset Design Tem-
perature. This is achieved by simultaneously starting the other
redundant fan and opening the by-pass section damper to reduce the system
pressure drop by diverting part of the supply air through the by-pass
59 section.

The four (4) auxiliary feedwater pump cell unit coolers, the two (2) emergency chiller room unit coolers, and the three (3) I&C Panel Cell Unit Coolers are supplied with emergency chilled water to maintain the Upset Design Temperature.

4. Occurrence of Fire in the Building

In the event of fire in the building, the Stairwell Pressurization
Fans will be manually started to prevent the smoke from entering
the stairwells. Six (6) fans are provided for pressurization of
stairwells 201, 205, 212, 213, 214 and 233. Fans for stairwells
201, 212, 213 and 214 are wall mounted and are provided with missile
protected intake openings. Fans for stairwells 205 and 233 are
mounted on the roof and are provided with missile protected roof
49 openings.

The Smoke Vent Fan shall be started manually to provide smoke venting from the areas containing safety related equipment. The Smoke Vent Fan is located on the roof on top of the missile protected exhaust opening at El. 857'-6".

The three (3) Unit Coolers provided for each of the I&C Panel Cells will maintain the Upset Design Condition in these cells in the event of fire in the SGB-IB Building.

9.6.6.4 Testing and Inspection Requirements

All components are tested and inspected as separate components and as integrated systems. Velometer readings are taken to ensure that all systems are balanced to deliver and exhaust the required air quantities. All water coils are hydraulically tested for leakage prior to being placed in service. Capacity and performance of the fans are tested according to the Air Moving and Conditioning Association requirements
49 prior to operation of the plant.

TABLE 9.6-1

CONTROL BUILDING HVAC SYSTEM

EQUIPMENT LIST

	EQUIPMENT TITLE	LOCATION BLDG.	LOCATION ELEV.	SAFETY CLASS	SEISMIC CLASS	PRIMARY PARAMETER (CFM)
	CB Control Room A/C Unit	CB	863'-3"	SC-3	I	51,500
	CB Control Room A/C Unit	CB	847'-3"	SC-3	I	51,500
	CB SWGR A/C Unit "A"	CB	863'-3"	SC-3	I	16,500
	CB SWGR A/C Unit "B"	CB	847'-3"	SC-3	I	16,500
	CB M-G Set Unit Cooler	CB	733'	None	III	41,000
	CB M-G Set Unit Cooler	CB	733'	None	III	41,000
	CB M-G Set Unit Cooler	CB	733'	None	III	41,000
	CB M-G Set & SWGR. Air Hand. Unit	CB	847'-3"	None	III	64,000
	CB M-G Set Unit Cooler	CB	733'	None	III	41,000
	M-G Set & SWGR. Supply Fan	CB	847'-3"	None	III	32,000
	M-G Set & SWGR. Supply Fan	CB	847'-3"	None	III	32,000
59	CB Stairwell Press. Fan	CB	880-'0"	None	III	6,000
	CB Control Room Filter Unit Fan	CB	863'-3"	SC-3	I	8,500
	CB Control Room Filter Unit Fan	CB	847'-3"	SC-3	I	8,500
	M-G Set & SWGR. Return Fan	CB	863'-3"	None	III	32,000
59	M-G Set & SWGR. Return Fan	CB	863'-3"	None	III	32,000
	CB Control Room Return Fan	CB	863'-3"	SC-3	I	49,400
	CB Control Room Return Fan	CB	847'-3"	SC-3	I	49,400
	CB SWGR. Return Fan "A"	CB	863'-3"	SC-3	I	10,700
	CB SWGR. Return Fan "B"	CB	847'-3"	SC-3	I	11,400
49	CB Battery Room Exhaust Fan	CB	847'-3"	SC-3	I	2,850

TABLE 9.6-1 (cont'd)

CONTROL BUILDING HVAC SYSTEM

EQUIPMENT LIST

EQUIPMENT TITLE	LOCATION BLDG.	LOCATION ELEV.	SAFETY CLASS	SEISMIC CLASS	PRIMARY PARAMETER (CFM)
CB Battery Room Exhaust Fan	CB	863-'3"	SC-3	I	3,600
CB Battery Room Exhaust Fan	CB	863'-3"	SC-3	I	2,000
CB Battery Room Exhaust Fan	CB	847'-3"	SC-3	I	2,000
CB Toilet Exhaust Fan	CB	863'-3"	None	III	950
CB Smoke Vent Fan	CB	880-'0"	None	III	10,000
CB Kitchen Exhaust Fan	CB	863'-3"	None	III	650
CB Control Room Filter Unit	CB	863'-3"	SC-3	I	8,500
CB Control Room Filter Unit	CB	847'-3"	SC-3	I	8,500
CB Reheat Coil	CB	847'-3"	None	I	12,850
CB Reheat Coil	CB	816'-0"	None	I	9,850
CB Reheat Coil	CB	816'-0"	None	I	9,850
CB Reheat Coil	CB	816'-0"	None	I	7,750
CB Reheat Coil	CB	816'-0"	None	I	2,300
CB Reheat Coil	CB	831'-0"	None	I	3,550
CB Reheat Coil	CB	831'-0"	None	I	1,850
CB Control Room Supply Attenuator	CB	847'-3"	None	I	18,250
CB Control Room Supply Attenuator	CB	831'-0"	None	I	29,750
CB Control Room Return Attenuator	CB	831'-0"	None	I	25,450
CB Control Room Return Attenuator	CB	831'-0"	None	I	20,650

TABLE 9.6-2

SINGLE FAILURE ANALYSIS

CONTROL ROOM HVAC SYSTEM

Component	Malfunction	Analysis
Air Conditioning Unit Supply Fan	Loss of fan in operating train; all operating modes	No effect. The redundant full capacity fan in the second train is available.
Return Air Fan	Loss of fan in operating train; all operating modes	No effect. The redundant full capacity fan in the second train is available.
Filter Unit Fan	Loss of fan in operating train; emergency mode	No effect. The redundant full capacity fan in the second train is available.
Air Conditioning Unit Chilled Water Coil	Loss of cooling capacity in operating train; all operating modes.	No effect. The redundant full capacity coil in the second train is available.
Air Conditioning Unit Automatic Roll Filter	High differential pressure in operating train; all operating modes.	No effect. The redundant full capacity Air Conditioning unit in the second train is available.
Air Conditioning Unit Bag Filter	High differential pressure in operating train; all operating modes.	No effect. The redundant full capacity Air Conditioning unit in the second train is available.
49 Filter Unit Prefilter	High differential pressure in operating train; emergency mode.	No effect. The redundant full capacity filter unit in the second train is available.

TABLE 9.6-2 (cont'd.)

Component	Malfunction	Analysis
Filter Unit First HEPA filter	High differential pressure in operating train; emergency mode.	No effect. The redundant full capacity filter unit in the second train is available.
Filter Unit Carbon Filter	High differential pressure in operating train; emergency mode or loss of adsorption capability.	No effect. The redundant full capacity filter unit in the second train is available.
Filter Unit second HEPA filter	High differential pressure in operating train; emergency mode.	No effect. The redundant full capacity filter unit in the second train is available.
Air Conditioning Unit Humidifier	Loss of humidifying capability in operating train; normal operating mode.	No effect. The redundant full capacity Air Conditioning Unit in the second train is available.
	Loss of humidifying capability in operating train; emergency mode.	No effect. The service of the humidifier is not required during emergency (recirculating) mode.
Outside Air Intake Isolation Valve	Inadvertent closure in operating mode.	No effect. System continues to operate in recirculating mode. Valve position indicator activates alarm in Control Room.
	Failure to close upon actuation to emergency mode.	No effect. The redundant isolation valve is available for isolation.
59 49 Exhaust Air Isolation Valve	Failure to close upon actuation to emergency mode.	No effect. The redundant isolation valve is available for isolation.

TABLE 9.6-2 (cont'd.)

	Component	Malfunction	Analysis
	Air Conditioning Unit Discharge Damper	Inadvertent closure in operating train; all operating modes.	No effect. The redundant full capacity Air Conditioning Unit in the second train is available.
	Filter Unit Fan Discharge Damper	Failure to open upon actuation to emergency mode.	No effect. The redundant full capacity filter unit in the second train is available.
		Inadvertent closure in operating train; emergency mode.	No effect. The redundant full capacity filter unit in the second train is available.
59	Filter Unit Outside Air Intake Valves	Failure to open upon actuation to emergency mode.	No effect. The redundant Emergency Air Intake Valve is available.
		Inadvertent closure in operating train; emergency mode.	No effect. The redundant Emergency Air Intake Valve is available.
59	Toilet Exhaust Fan	Loss of fan; normal operating mode	No effect. The Toilet Exhaust Fan is not essential for the system operation.
	Kitchen Exhaust Fan	Loss of fan; normal operating mode	No effect. The Kitchen Exhaust Fan is not essential for the system operation.
49	Electrical Power Supply	Loss of power from normal AC distribution system (Plant, preferred off-site AC and reserve off-site AC power supplies)	No effect. Power to both trains are automatically supplied by the standby (on-site) AC power supply (diesel generators)

TABLE 9.6-2 (cont'd.)

Component	Malfunction	Analysis
	Loss of one diesel generator when standby AC power supply provides power.	No effect. Standby train which is connected to the second diesel generator start automatically.
49 Electric Reheat Coil	Loss of Power; normal operating mode.	No effect. The Reheat Coil is not essential for the system operation.

TABLE 9.6-3

SINGLE FAILURE ANALYSIS

CONTROL AND DIESEL GENERATOR BUILDING EMERGENCY HVAC SYSTEM

Component	Malfunction	Analysis
Air Conditioning Unit Supply Fan	Loss of Fan; all operating modes.	No effect. The full capacity fan in the second train is available.
Return Air Fan	Loss of Fan; all operating modes.	No effect. The full capacity fan in the second train is available.
Battery Room Exhaust	Loss of fan; all operating modes.	No effect. The three redundant full capacity fans are available.
Air Conditioning Unit Chilled Water Coil	Loss of cooling capacity; all operating modes.	No effect. The full capacity coil in the second train is available.
Air Conditioning Unit Automatic Roll Filter	High differential pressure; all operating modes.	No effect. The full capacity Air Conditioning unit in the second train is available.
Air Conditioning Unit Bag Filter	High differential pressure; all operating modes.	No effect. The full capacity Air Conditioning unit in the second train is available.

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TABLE 9.6-3 (cont'd.)

Component	Malfunction	Analysis
59 Electrical Power Supply	Loss of pcwer from normal AC distribution system (Plant, preferred off-site AC and reserve off-site AC power supplies)	No effect. Power to both trains are automatically supplied by the standby (on-site) AC power supply (diesel generators)
49	Loss of one diesel generator when standing AC power supply provides power	No effect. Second train is connected to the second diesel generator.

TABLE 9.6-4

REACTOR CONTAINMENT BUILDING HVAC SYSTEM

EQUIPMENT LIST

	EQUIPMENT TITLE	LOCATION BLDG.	LOCATION ELEV.	SAFETY CLASS	SEISMIC CLASS	PRIMARY PARAMETER (CFM)
	RCB Below Oper. Floor A/C Unit	RCB	752'-8"	None	I	17,600
	RCB Below Oper. Floor A/C Unit	RCB	752'-8"	None	I	18,200
	RCB Operating Floor Unit Cooler	RCB	857'-11"	None	I	40,000
	RCB Operating Floor Unit Cooler	RCB	857'-11"	None	I	40,000
	RCB Operating Floor Unit Cooler	RCB	857'-11"	None	I	40,000
	RCB EI&C Cubicle Unit Cooler	RCB	842'-0"	SC-3	I	2,480
	RCB EI&C Cubicle Unit Cooler	RCB	824'-3"	SC-3	I	2,420
	RCB EI&C Cubicle Unit Cooler	RCB	842'-0"	SC-3	I	2,260
	HAA Unit Cooler	RCB	800'-9"	None	I	18,000
	RCB Below Oper. Floor Return Fan	RCB	752'-8"	None	III	15,000
	RCB Below Oper. Floor Return Fan	RCB	752'-8"	None	III	15,400
	RCB Dome Recirculating Fan	RCB	916'-0"	None	I	10,000
	RCB Dome Recirculating Fan	RCB	916'-0"	None	I	10,000
	LCCV Unit Cooler	RCB	733'-0"	None	I	2,700
	RCB Supply Air Handling Unit	SGB	836'-0"	None	III	14,000
	RCB Supply Fan	SGB	836'-0"	None	III	14,000
	RCB Supply Fan	SGB	836'-0"	None	III	14,000
49	RCB Exhaust Fan	RSB	861'-6"	None	III	14,000
59	RCB Exhaust Fan	RSB	861'-6"	None	III	14,000

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	762'-0"	None	III	200
50	RCB Inerted Cells Booster Fan	RCB	733'-0"	None	III	200
59	RCB Portable Filter Fan	RCB	-	None	III	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	SGB	816'-0"	SC-2	I	14,000
	RCB Exhaust Isolation Valve	RSB	816'-0"	SC-2	I	14,000
	RSB Annulus Pressure Maintenance Fan	RSB	840'-0"	SC-3	I	3,000
	RCB Supply Isolation Valve	SGB	816'-0"	SC-2	I	14,000
59	RCB Exhaust Isolation Valve	RSB	816'-0"	SC-2	I	14,000

49

TABLE 9.6-4 (cont'd).

EQUIPMENT TITLE	LOCATION BLDG.	ELEV.	SAFETY CLASS	SEISMIC CLASS	PRIMARY PARAMETER (CFM)
RCB Annulus Pressure Maintenance Fan	RSB	861'-6"	SC-3	I	3,000
RCB Annulus Filter Unit	RSB	840'-0"	SC-3	I	14,000
RCB Annulus Filter Unit	RSB	861'-6"	SC-3	I	14,000
RCB Annulus Filter Fan	RSB	840'-0"	SC-3	I	11,000
RCB Annulus Filter Fan	RSB	861'-6"	SC-3	I	11,000
RCB Purge Supply Isolation Valve	RSB	840'-0"	SC-2	I	21,770
RCB Purge Supply Isolation Valve	RSB	816'-0"	SC-2	I	21,770
RCB Purge Supply Isolation Valve	RSB	840'-0"	SC-2	I	21,770
RCB Purge Supply Isolation Valve	RSB	816'-0"	SC-2	I	21,770
RCB Containment Cleanup Exhaust Isolation Valve	RSB	816'-0"	SC-2	I	21,770
RCB Containment Cleanup Exhaust Isolation Valve	RSB	816'-0"	SC-2	I	21,770

49
59

TABLE 9.6-4 (cont'd).

EQUIPMENT TITLE	LOCATION BLDG.	ELEV.	SAFETY CLASS	SEISMIC CLASS	PRIMARY PARAMETER (CFM)
RCB Containment Cleanup Exhaust Isolation Valve	RSB	840'-0"	SC-2	I	21,770
49 59 RCB Containment Cleanup Exhaust Isolation Valve	RSB	840'-0"	SC-2	I	21,770

TABLE 9.6-4a

SINGLE FAILURE ANALYSIS

RCB HVAC SYSTEM

	COMPONENT	MALFUNCTION	ANALYSIS
59	EI&C Cubicle Unit Cooler	Failure to operate, all operating modes	No effect. Unit Cooler serving the other EI&C cubicles are still in operation.

TABLE 9.6-4b

SINGLE FAILURE ANALYSIS

ANNULUS FILTRATION SYSTEM

COMPONENT	MALFUNCTION	ANALYSIS
Filter Unit Electric Heating Coil	Failure to operate, all operating modes	No effect. The redundant full capacity filter unit is available.
Filter Unit Bag Filter	High differential pressure in operating train, all operating modes	No effect. The redundant full capacity filter unit is available.
Filter Unit First HEPA Filter	High differential pressure in operating train, all operating modes	No effect. The redundant full capacity filter unit is available.
Filter Unit Carbon Filter	High differential pressure or loss of adsorption capability in operating train, all operating modes	No effect. The redundant full capacity filter unit is available.
Filter Unit Second HEPA Filter	High differential pressure in operating train, all operating modes	No effect. The redundant full capacity filter unit is available.
Annulus Filter Fan	Failure to operate, all operating modes	No effect. The redundant annulus filter fan is available.
Annulus Filter Fan	Failure to open, operating mode	No effect. The redundant annulus filter fan is available.
59 Discharge Damper	Inadvertent closing, operating mode	No effect. The redundant annulus filter fan train is available.

Annulus Pressure Maintenance Fan	Failure to operate, all operating modes	No effect. The redundant annulus pressure maintenance fan is available.
Annulus Pressure Maintenance Fan Discharge Damper	Failure to open, operating mode	No effect. The redundant annulus pressure maintenance fan train is available.
	Inadvertent closing, operating mode	No effect. The redundant annulus pressure maintenance fan train is available.
Annulus Filter Fan Common Discharge Duct Exhaust Damper	Inadvertent opening, operating mode	No effect. The redundant exhaust damper in series is in "closed" position.
Annulus Filter Fan Common Discharge Duct Return Damper	Failure to open, operating mode	No effect. Redundant return damper in parallel is available.
59	Inadvertent closing, operating mode	No effect. Redundant return damper in parallel is in "open" position.

TABLE 9.6-5

REACTOR SERVICE BUILDING HVAC SYSTEM

EQUIPMENT LIST

EQUIPMENT TITLE	LOCATION BLDG.	LOCATION ELEV.	SAFETY CLASS	SEISMIC CLASS	PRIMARY PARAMETER (CFM)
RSB Air Hand. Unit	RSB	733'-0"	None	III	82,000
RSB Radwaste Area Air Hand. Unit	RSB	867'-0"	None	III	40,000
RSB Refueling Commun. Center Air Conditioning Unit	RSB	865'-0"	None	III	3,500
ABHX Cell Unit Cooler	RSB	816'-0"	SC-3	I	5,000
ABHX Cell Unit Cooler	RSB	816'-0"	SC-3	I	5,000
RAPS & CAPS Unit Cooler	RSB	764'-0"		III	2,500
Annulus Ftr. Cell Unit Cooler	RSB	840'-0"	SC-3	I	1,000
Annulus Ftr. Cell Unit Cooler	RSB	861'-6"	SC-3	I	1,500
RSB Clean-up Ftr. Cell Unit Cooler	RSB	794'-6"	SC-3	I	1,000
RSB Clean-up Ftr. Cell Unit Cooler	RSB	816'-0"	SC-3	I	2,200
RSB Radwaste Area H.W. Unit Heater	RSB	816'-0"	None	III	4,500
RSB Radwaste Area H.W. Unit Heater	RSB	816'-0"	None	III	2,400
RSB Supply Fan	RSB	733'-0"	None	III	41,000
RSB Supply Fan	RSB	733'-0"	None	III	41,000
RSB Radwaste Area Supply Fan	RSB	867'-0"	None	III	20,000
RSB Radwaste Area Supply Fan	RSB	867'-0"	None	III	20,000
RSB Inerted Cells Booster Fan	RSB	779'-0"	None	III	200

TABLE 9.6-5 (cont'd)

EQUIPMENT TITLE	LOCATION BLDG.	LOCATION ELEV.	SAFETY CLASS	SEISMIC CLASS	PRIMARY PARAMETER (CFM)
RSB Normal Exhaust Fan	RSB	733'-0"	None	III	41,700
RSB Normal Exhaust Fan	RSB	733'-0"	None	III	41,700
RSB Radwaste Area Exhaust Fan	RSB	867 -0"	None	III	23,000
RSB Radwaste Area Exhaust Fan	RSB	867'-0"	None	III	23,000
RSB RAPS and CAPS Exhaust Fan	RSB	779'-0"	None	III	1,000
RSB RAPS and CAPS Exhaust Fan	RSB	798'-0"	None	III	1,000
RSB Radwaste Area Exhaust Filter Unit Fan	RSB	867'-0"	None	III	46,000
RSB Smoke Vent Fan	RSB	884'-0"	None	III	7,000
RSB Stairwell Press. Fan	RSB	816'-0"	None	III	6,300
RSB Stairwell Press. Fan	RSB	884'-0"	None	III	6,300
RSB Stairwell Press. Fan	RSB	884'-0"	None	III	6,300
RSB Stairwell Press. Fan	RSB	840'-0"	None	III	6,300
RSB RW Area Exhaust Filter Unit	RSB	867'-0"	None	III	46,000
RSB Portable Filter/Fan Unit	RSB	--	None	III	1,000
RSB RW Area Control Room H.W. Reheat Coil	RSB	867'-0"	None	III	4,040
49 RSB Radwaste Area Oper. Area H.W. Reheat Coil	RSB	867'-0"	None	III	17,040
59 RSB ABHX Cell H.W. Unit Heater	RSB	840'-0"	None	III	5,530

TABLE 9.6-5 (cont'd)

	RSB Elec. Equip. Cell Unit Cooler	RSB	733'-0"	SC-3	I	2500
	RSB Elec. Equip. Cell Unit Cooler	RSB	733'-0"	SC-3	I	3000
	RSB Clean Up Pump Cell Unit Cooler	RSB	733'-0"	SC-3	I	3000
	RSB Clean Up Pump Cell Unit Cooler	RSB	733'-0"	SC-3	I	3000
	RSB Clean Up Scrubber Unit Cooler	RSB	758'-0"	SC-3	I	5100
	RSB Clean Up Scrubber Unit Cooler	RSB	758'-0"	SC-3	I	5100
	RSB Cont. Clean Up Pipe Chase Unit Cooler	RSB	816'-0"	SC-3	I	1000
	RSB Cont. Clean Up Pipe Chase Unit Cooler	RSB	840'-0"	SC-3	I	1000
	RSB Clean Up Filter Unit	RSB	794'-6"	SC-3	I	18,000
	RSB Clean Up Filter Unit	RSB	816'-0"	SC-3	I	18,000
	RSB Clean Up Filter Fan	RSB	794'-6"	SC-3	I	18,000
	RSB Clean Up Filter Fan	RSB	816'-0"	SC-3	I	18,000
	RSB RWA Unit Heater	RSB	867-'0"	None	III	1100
	RSB RWA Unit Heater	RSB	867'-0"	None	III	1100
59	RSB Roof Exhaust Hood	RSB	884'-0"	None	III	46,000

TABLE 9.6-5a

SINGLE FAILURE ANALYSIS

RSB HVAC SYSTEMS

COMPONENT	MALFUNCTION	ANALYSIS
ABHX Cell Unit Cooler Supply Fan	Failure to operate, all operating modes	No effect. Redundant Unit Cooler for the other redundant train is available.
Annulus Filter Cell Unit Cooler Supply Fan	Failure to operate, all operating modes	No effect. Redundant Unit Cooler for the other redundant train is available.
Clean Up Filter Cell Unit Cooler Supply Fan	Failure to operate, all operating modes	No effect. Redundant Unit Cooler for the other redundant train is available.
Electric Equipment Cell Unit Cooler Supply Fan	Failure to operate, all operating modes	No effect. Redundant Unit Cooler for the other redundant train is available.
Cont. Clean Up Pump Cell Unit Cooler Supply Fan	Failure to operate, all operating modes	No effect. Redundant Unit Cooler for the other redundant train is available.
Cont. Clean Up Pipe Chase Unit Cooler Supply Fan	Failure to operate, all operating modes	No effect. Redundant Unit Cooler for the other redundant train is available.
59 Cont. Clean Up Scrubber Unit Cooler Supply Fan	Failure to operate, all operating modes	No effect. Redundant full capacity unit cooler is available.

TABLE 9.6-5a (cont'd)

RSB Clean Up Filter Unit Electric Heating Coil	Failure to operate, all operating modes	No effect. The redundant full capacity filter unit train is available.
RSB Clean Up Filter Unit Bag Filter	High differential pressure in operating train, all operating modes	No effect. The redundant full capacity filter unit train is available.
RSB Clean Up Filter Unit First HEPA Filter	High differential pressure in operating train, all operating modes	No effect. The redundant full capacity filter unit train is available.
RSB Clean Up Filter Unit Second HEPA Filter	High differential pressure in operating train, all operating modes	No effect. The redundant full capacity filter unit train is available.
RSB Clean Up Filter Unit Carbon Filter	High differential pressure or loss of adsorption capability in operating train, all operating modes	No effect. The redundant full capacity filter unit train is available.
RSB Clean Up Filter Unit Fan	Failure to operate, all operating modes	No effect. The redundant full capacity filter fan is available.
RSB Clean Up Filter Fan Discharge Damper	Failure to open, all operating modes	No effect. The redundant full capacity filter unit train is available.
59 RSB Clean Up Filter Fan Discharge Damper	Inadvertent closing, all operating modes	No effect. The redundant full capacity filter unit train is available.

TABLE 9.6-5a (cont'd)

RSB Return Damper (Fuel Accident Mode)	Failure to open. Fuel Accident Mode	No effect. Redundant damper and full capacity filter train is available.
RSB Return Damper (Fuel Accident Mode)	Inadvertent closing in operating mode, Fuel Accident Mode	No effect. Redundant damper and full capacity filter train is available.
RSB Clean Up Filter Inlet Damper (Fuel Accident Mode)	Failure to open, Fuel Accident Mode	No effect. Redundant damper and full capacity filter train is available.
59 RSB Clean Up Filter Inlet Damper (Fuel Accident Mode)	Inadvertent closing in operating mode, Fuel Accident Mode	No effect. Redundant damper and full capacity filter train is available.

TABLE 9.6-6

DIESEL GENERATOR BUILDING HVAC SYSTEM

EQUIPMENT LIST

	EQUIPMENT TITLE	BLDG.	ELEV.	SAFETY CLASS	SEISMIC CLASS	PRIMARY PARAMETER (CFM)
	DGB M-G Sets Unit Cooler	DGB	733'	None	III	41,000
	DGB M-G Sets Unit Cooler	DGB	733'	None	III	41,000
	DGB Diesel Room "A" HV Unit	DGB	816'	None	III	6,300
59	DGB Diesel Room "B" HV Unit	DGB	816'	None	III	5,600
	DGB Diesel Room "A" Emerg. Supply Fan	DGB	816'	SC-3	I	60,000
	DGB Diesel Room "A" Emerg. Supply Fan	DGB	816'	SC-3	I	60,000
	DGB Diesel Room "B" Emerg. Supply Fan	DGB	816'	SC-3	I	60,000
	DGB Diesel Room "B" Emerg. Supply Fan	DGB	816'	SC-3	I	60,000
	DGB Day Tank Cell Exhaust Fan	DGB	816'	SC-3	I	450
	DGB Day Tank Cell Exhaust Fan	DGB	816'	SC-3	I	450
49	DGB Smoke Vent Fan	DGB	847'-3"		III	6,000

TABLE 9.6-7

SINGLE FAILURE ANALYSIS

DIESEL GENERATOR ROOMS HVAC SYSTEM

	Component	Malfunction	Analysis
59	Day Tank Cell Exhaust Fan	Failure to operate, all operating modes	Supply air from the switchgear A/C Units will ventilate the day tank cell
	Ventilation Supply Fan Intake Damper	Inadvertent closure; emergency operating modes	No effect, Redundant Intake damper will be open.
		Failure to open upon actuation to emergency mode	Same as Above
		Inadvertent opening; normal operation	No effect, Fan Discharge Damper remains closed.
	Ventilation Supply Fan Discharge Damper	Inadvertent closure; emergency operation	No effect; two redundant fans in second diesel Room are available
		Failure to open upon actuation to emergency mode.	Same as above.
		Inadvertent opening; normal operation	No effect; Diesel Room is pressurized by HV unit supply fan.
	Electrical Power Supply	Loss of power from normal AC distribution system (Plant, preferred off-site AC and reserve off-site AC power supplies)	No effect; power to both trains are automatically supplied by the stand-by (on-site) AC power supply (diesel generators)
49		Loss of one diesel generator when standby AC power supply provides power	No effect; standby fans which are serving the second diesel generator are available. The second Diesel Generator is capable to handle the loads required for the safe shutdown of the plant.

TABLE 9.6-8

STEAM GENERATOR BUILDING HVAC SYSTEM

EQUIPMENT LIST

EQUIPMENT TITLE	LOCATION BLDG.	ELEV.	SAFETY CLASS	SEISMIC CLASS	PRIMARY PARAMETER (CFM)
SGB Loop #1 Air Hand. Unit	SGB	852'-6"	SC-3	I	81,000
SGB Loop #2 Air Hand. Unit	SGB	852'-6"	SC-3	I	74,000
SGB Loop #3 Air Hand. Unit	SGB	852'-6"	SC-3	I	91,000
SGB Intermed. Bay Air Hand. Unit	SGB	836'-0"	SC-3	I	27,000
SGB Intermed. Bay Air Hand. Unit	SGB	816'-0"	SC-3	I	37,000
SGB Maintenance Bay A/C Unit	SGB	816'-0"		III	9,000
SGB Pri. Na Tanks Unit Cooler	SGB	733'-0"		III	7,000
SGB Aux. Feed Pump Unit Cooler	SGB	746'-0"	SC-3	I	7,000
SGB Aux. Feed Pump Unit Cooler	SGB	746'-0"	SC-3	I	7,000
SGB Aux. Feed Pump Unit Cooler	SGB	733'-0"	SC-3	I	6,000
SGB Aux. Feed Pump Unit Cooler	SGB	733'-0"	SC-3	I	6,000
SGB I&C Panel Unit Cooler	SGB	836'-0"	SC-3	I	1,500
SGB I&C Panel Unit Cooler	SGB	836'-0"	SC-3	I	1,500
SGB I&C Panel Unit Cooler	SGB	836'-0"	SC-3	I	1,500
SGB Emerg. Chiller Unit Cooler	SGB	733'-0"	SC-3	I	1,000
SGB Emerg. Chiller Unit Cooler	SGB	733'-0"	SC-3	I	1,000
SGB Unit Heater	SGB	806'-0"		III	1,250
SGB Unit Heater	SGB	806'-0"		III	1,250
SGB Unit Heater	SGB	806'-0"		III	1,250
SGB Unit Heater	SGB	806'-0"		III	1,250

49

TABLE 9.6-8 (cont'd)

	EQUIPMENT TITLE	LOCATION BLDG.	LOCATION ELEV.	SAFETY CLASS	SEISMIC CLASS	PRIMARY PARAMETER (CFM)
	SGB Unit Heater	SGB	806'-0"	None	III	1,250
	SGB Unit Heater	SGB	806'-0"	None	III	1,250
	SGB Unit Heater	SGB	846'-0"	None	III	1,150
	SGB Unit Heater	SGB	846'-0"	None	III	1,150
59	SGB Unit Heater	SGB	846'-0"	None	III	2,000
	SGB Loop #1 Supply Fan	SGB	852'-6"	SC-3	I	81,000
	SGB Loop #1 Supply Fan	SGB	852'-6"	SC-3	I	81,000
	SGB Loop #2 Supply Fan	SGB	852'-6"	SC-3	I	74,000
	SGB Loop #2 Supply Fan	SGB	852'-6"	SC-3	I	74,000
	SGB Loop #3 Supply Fan	SGB	852'-6"	SC-3	I	91,000
	SGB Loop #3 Supply Fan	SGB	852'-6"	SC-3	I	91,000
	SGB Intermediate Bay Supply Fan	SGB	836'-0"	SC-3	I	27,000
	SGB Intermediate Bay Supply Fan	SGB	836'-0"	SC-3	I	27,000
	SGB Intermediate Bay Supply Fan	SGB	816'-0"	SC-3	I	37,000
59	SGB Intermediate Bay Supply Fan	SGB	816'-0"	SC-3	I	37,000
	SGB Loop #1 Exhaust Fan	SGB	851'-6"	SC-3	I	60,000
	SGB Loop #1 Exhaust Fan	SGB	851'-6"	SC-3	I	60,000
	SGB Loop #2 Exhaust Fan	SGB	851'-6"	SC-3	I	53,000
	SGB Loop #2 Exhaust Fan	SGB	851'-6"	SC-3	I	53,000
	SGB Loop #3 Exhaust Fan	SGB	851'-6"	SC-3	I	70,000
49	SGB Loop #3 Exhaust Fan	SGB	851'-6"	SC-3	I	70,000

TABLE 9.6-8 (cont'd)

EQUIPMENT TITLE	LOCATION BLDG.	ELEV.	SAFETY CLASS	SEISMIC CLASS	PRIMARY PARAMETER (CFM)
SGB Intermediate Bay Exhaust Fan	SGB	836'-0"	SC-3	I	27,000
SGB Intermediate Bay Exhaust Fan	SGB	836'-0"	SC-3	I	27,000
SGB Intermediate Bay Exhaust Fan	SGB	816'-0"	SC-3	I	37,000
SGB Intermediate Bay Exhaust Fan	SGB	816'-0"	SC-3	I	37,000
SGB OA Filter	SGB	733'-0"		III	450
SGB OA Filter	SGB	746'-0"		III	250
SGB Elec. Duct Heater	SGB	733'-0"		III	250
SGB Elec. Duct Heater	SGB	746'-0"		III	450
SGB Stairwell Press Fan	SGB	880-'0"		III	6,000
SGB Stairwell Press Fan	SGB	866'-0"		III	6,000
SGB Stairwell Press Fan	SGB	843'-0"		III	6,000
SGB Stairwell Press Fan	SGB	866'-0"		III	6,000
SGB Stairwell Press Fan	SGB	892'-0"		III	6,000
SGB Stairwell Press Fan	SGB	863'-0"		III	6,000
SGB OA Supply Fan	SGB	733'-0"		III	250
SGB OA Supply Fan	SGB	746'-0"		III	450
49 SGB Smoke Vent Fan	SGB	857'-0"		III	13,000

Amend. 49
April 1979

TABLE 9.6-9

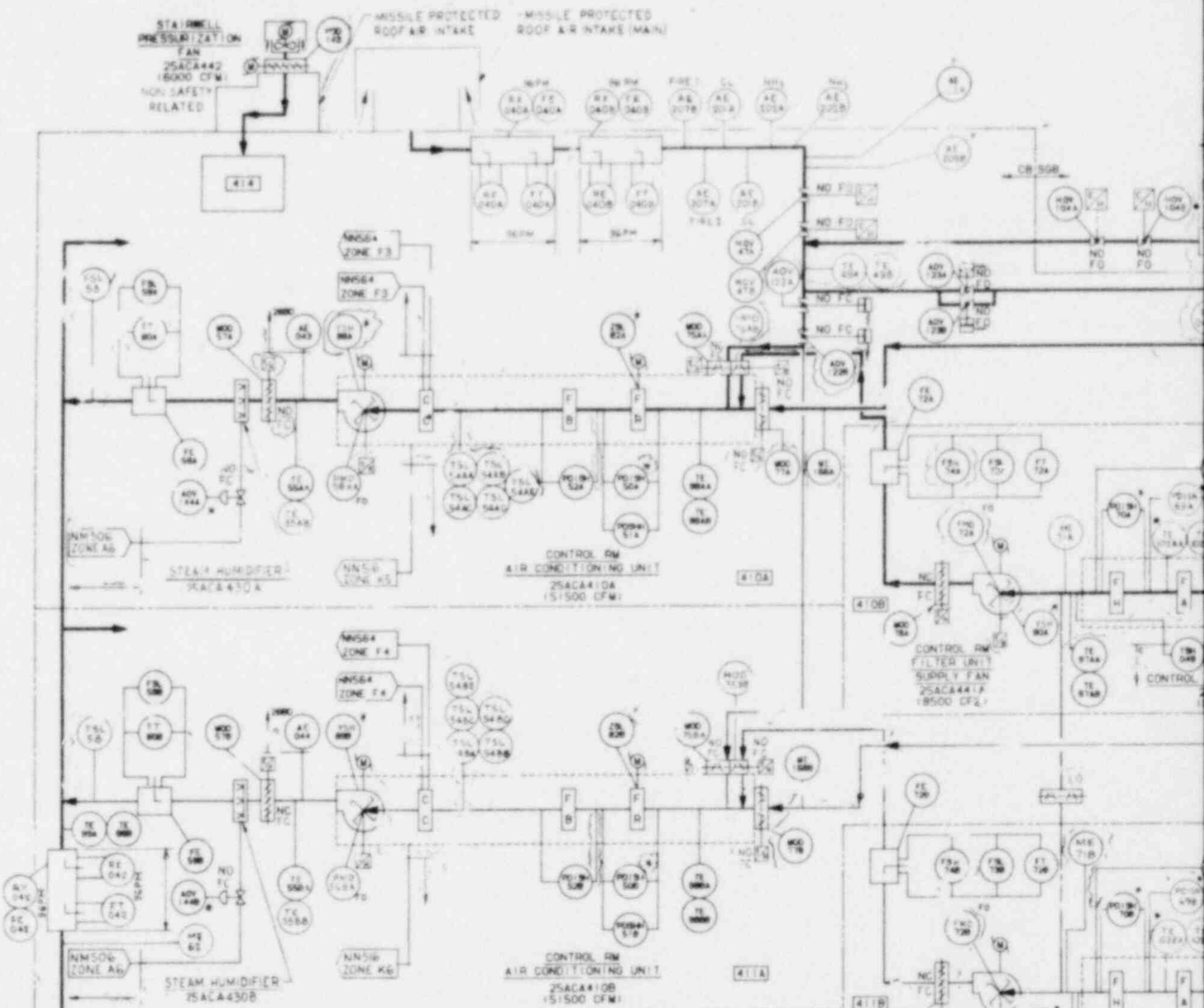
SINGLE FAILURE ANLYSIS

STEAM GENERATOR BUILDING, STEAM GENERATOR CELLS AND SGB INTERMEDIATE BAY HVAC SYSTEM

Component	Malfunction	Analysis
59 Air Handling Unit Supply Fan	Loss of Fan; all operating modes	No effect; the redundant full capacity fan is available.
Exhaust Air Fan	Loss of Fan; all operating modes	No effect; the redundant full capacity fan is available.
59 Air Handling Unit Chilled Water Coil	Loss of cooling capacity in operating train; all operating modes	No effect; ventilation air is capable to maintain 120°F max. temperature.
59 Air Handling Unit Automatic Roll Filter	High differential pressure on operting train; all operating modes.	No effect; pressure differential switch alarms Control Room. Redundant fan if available to overcome increased pressure with malfunction corrected.
59 Air Handling Unit Bag Filter	High differential pressure in operating train; all operating modes.	Same as above.
Supply Fan Discharge Damper	Inadvertent closure in operating train; all operating modes.	No effect; the redundant full capacity fan is available.
Return Fan Discharge Damper	Inadvertent closure in operating train; all operating modes.	No effect; the redundant full capacity fan is available.
Unit Cooler Fan	Loss of fan in operating train; all operating modes.	No effect; the redundant unit cooler is available.
49 Unit Cooler Cooling Coil	Loss of cooling capacity in operating train; emergency mode.	Same as above.

Component	Malfunction	Analysis
Electrical Power Supply	Loss of power from normal AC distribution system (Plant, preferred off-site AC and reserve off-site AC power supplies)	No effect; power to both trains is automatically supplied by the standby (on-site) AC power supply (diesel generators)
	Loss of one diesel generator when standby AC power supply provides power	No effect; standby train, which is connected to the second diesel generator, starts automatically.

59 49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25ACA442
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MISSILE PROTECTED ROOF AIR INTAKE (MAIN)
414
STEAM HUMIDIFIER
CONTROL RM AIR CONDITIONING UNIT
25ACA410A
(51500 CFM)
410A
410B
CONTROL RM FILTER UNIT SUPPLY FAN
25ACA441A
(8500 CFM)
STEAM HUMIDIFIER
25ACA430B
CONTROL RM AIR CONDITIONING UNIT
25ACA410B
(51500 CFM)
411A
411B
CONTROL RM FILTER UNIT SUPPLY FAN
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(8500 CFM)
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417B
416A
416B
ATM
TO CELL 411A
TO CELL 410A
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25ACA485
25ACA491
25ACA490
TO CELLS 425, 426 & 437
FROM CELL 429
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450
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422
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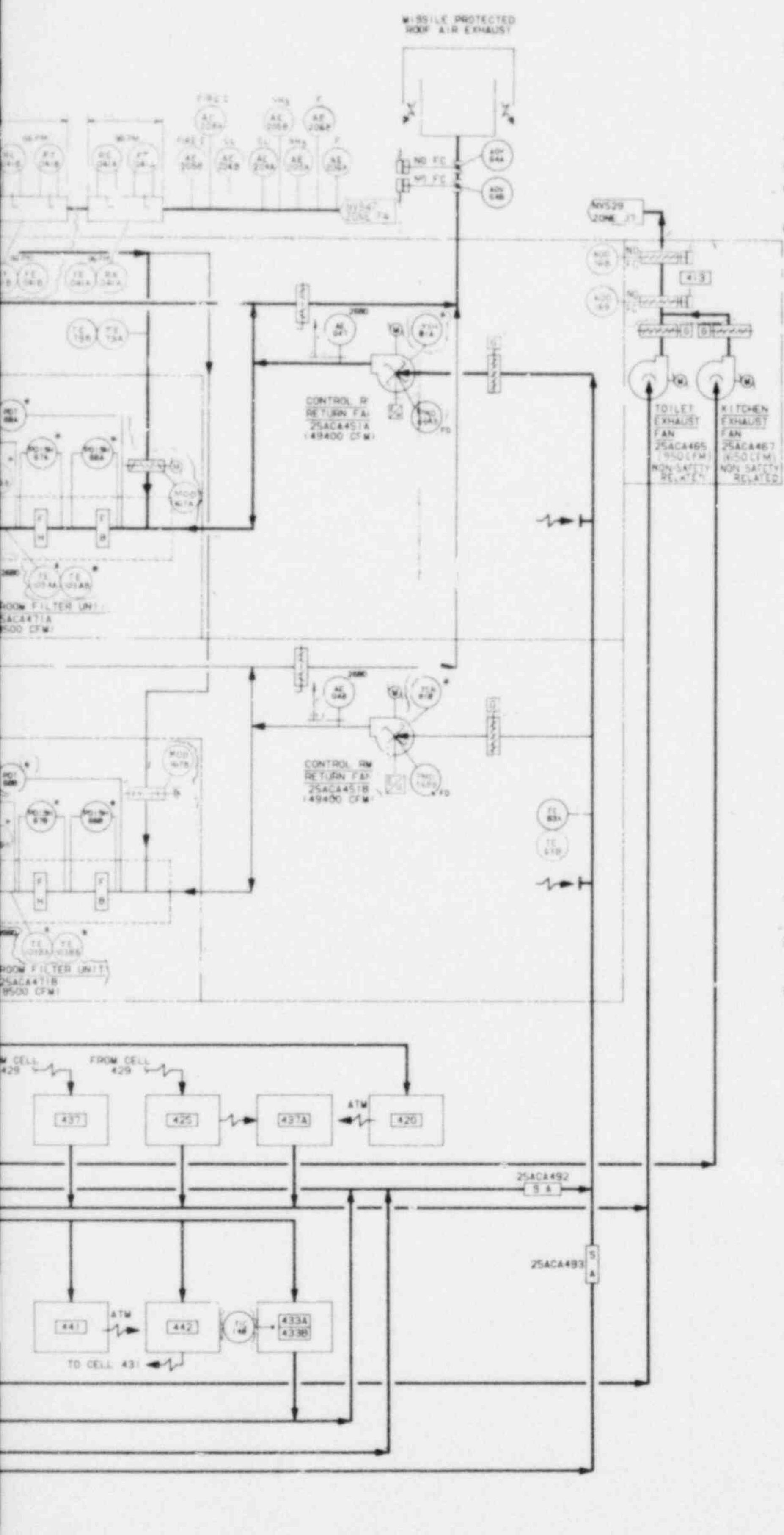
NV530-3



GENERAL NOTES

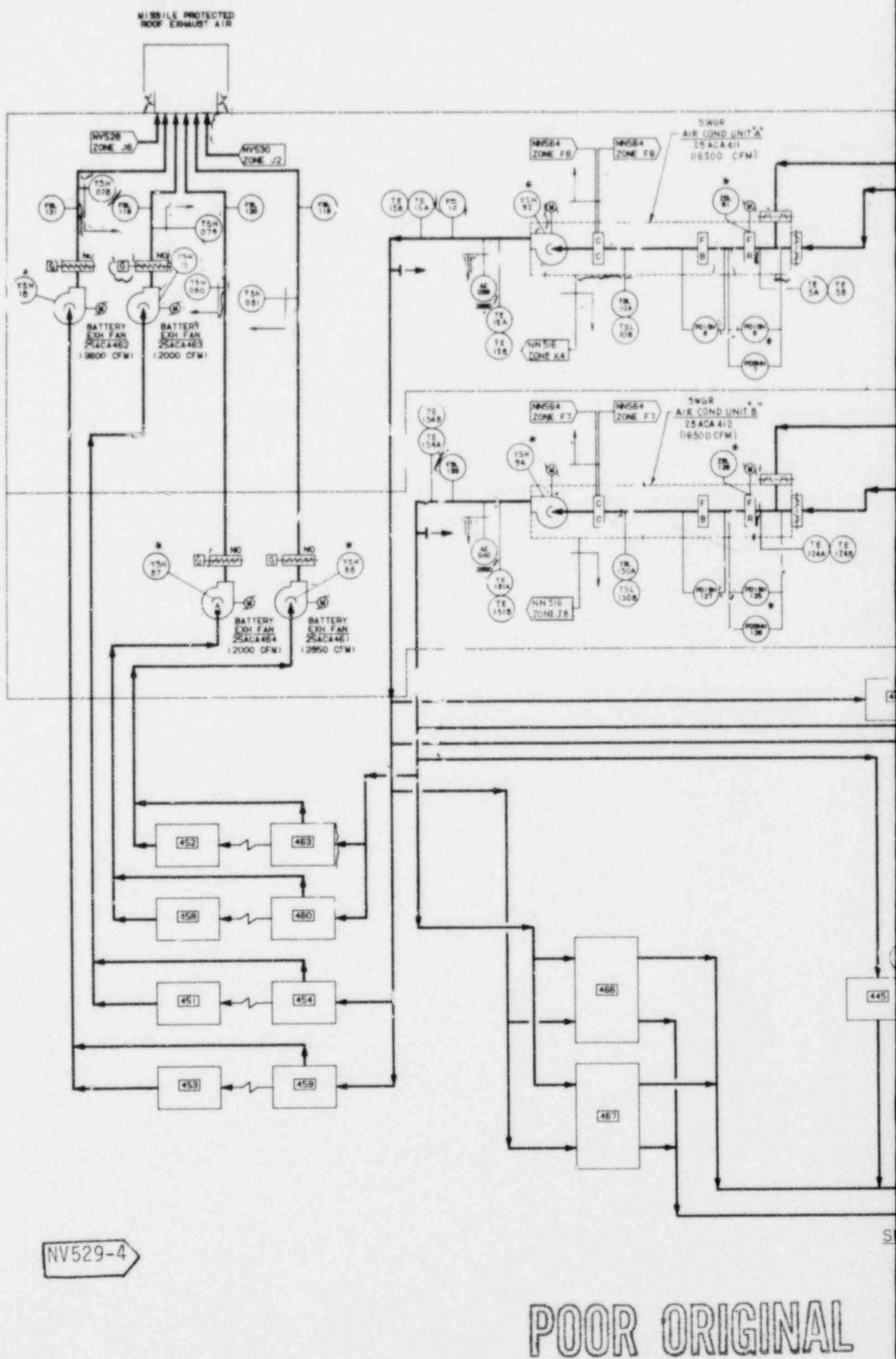
1. SYMBOLS AND ABBREVIATIONS WARD DOC D-0036
2. ALL EQUIPMENT, INSTRUMENT AND CONTROL VALVE NUMBERS ARE PREFIXED BY 25AC UOS
3. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY "K"
4. SYSTEM CLEANLINESS CLASSIFICATION:
5. SEISMIC CATEGORY: I
6. DELETED
7. SAFETY CLASSIFICATION: 3 UOS
8. SEE AIR FLOW BALANCE SHEETS FOR AIR QUANTITIES APPENDIX C-1
9. ALL INSTRUMENT ITEMS SHOWN MARKED WITH AN ASTERISK * SHALL BE SUPPLIED BY THE ASSOCIATED EQUIPMENT MANUFACTURER

POOR ORIGINAL

REFERENCE DRAWINGS

1. P&ID EMERGENCY CHILLED WATER SYSTEM CB & SGB
B&H DWG NN564
2. FLOW DIAGRAM CONTROL BUILDING FLOOR & EQPT DRAINS
BER DWG NN516
3. P&ID HW HEATING SYSTEM PSB
BER DWG NM506

FIGURE 9.6-1 P & I Diagram Control Room HVAC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NV530 ZONE J2
BATTERY EXH FAN 25ACA462 (3800 CFM)
BATTERY EXH FAN 25ACA463 (2000 CFM)
BATTERY EXH FAN 25ACA464 (2000 CFM)
BATTERY EXH FAN 25ACA461 (2850 CFM)
NN564 ZONE F6
SWGR AIR COND UNIT "A" 25ACA411 (16500 CFM)
NN516 ZONE K4
NN564 ZONE F7
SWGR AIR COND UNIT "B" 25ACA412 (16500 CFM)
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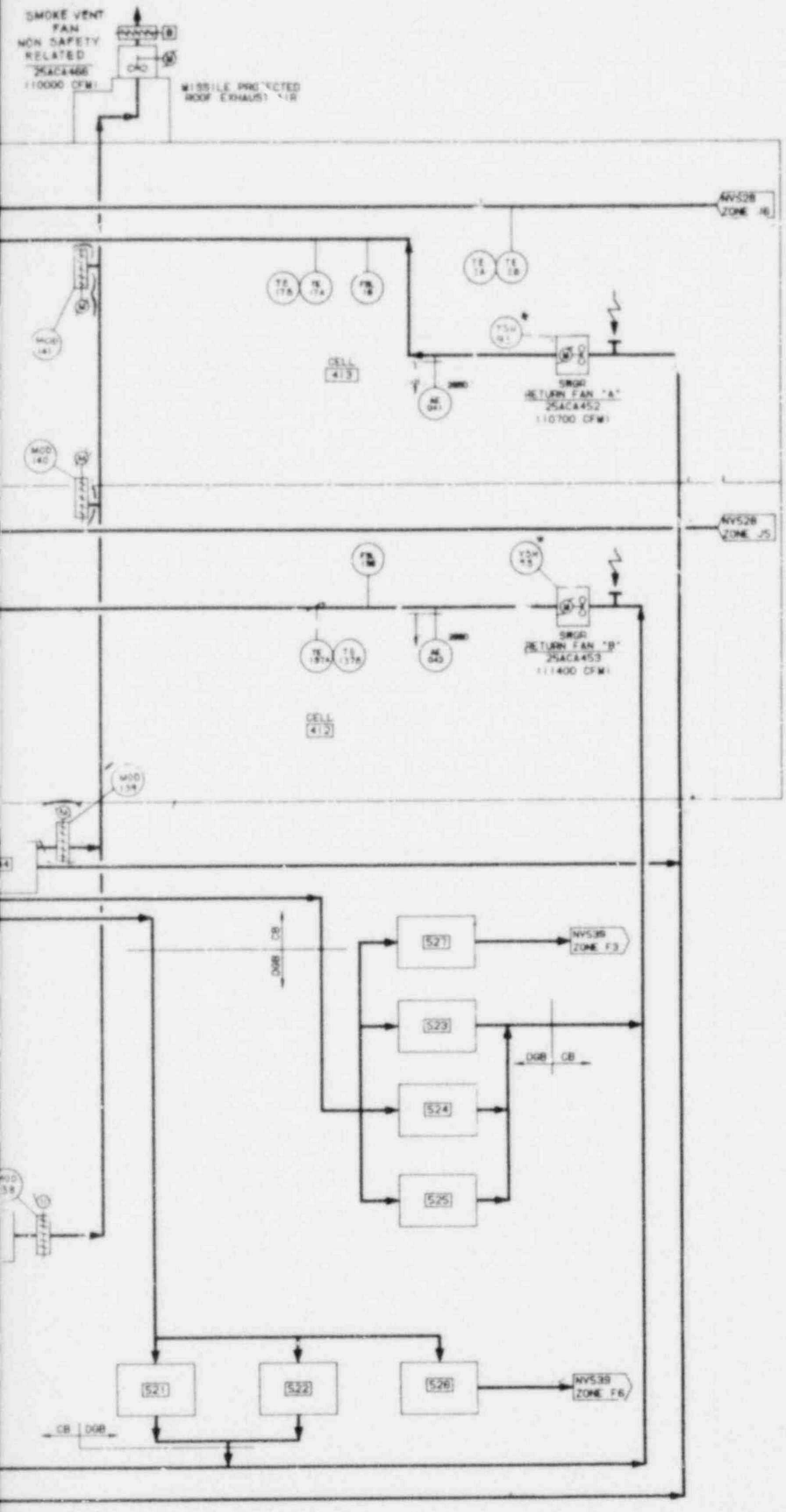
NV529-4
POOR ORIGINAL



GR SYSTEMS "A" & "B" HVAC

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4. SYSTEM CLEANLINESS CLASSIFICATION
5. SEISMIC CATEGORY: I
7. SAFETY CLASSIFICATION: 3 UOS
8. SEE AIR FLOW BALANCE SHEETS FOR AIR QUANTITIES APPENDIX C-1
9. ALL INSTRUMENT ITEMS SHOWN MARKED WITH AN ASTERISK * SHALL BE SUPPLIED BY THE ASSOCIATED EQUIPMENT MFR.

POOR ORIGINAL

REFERENCE DRAWINGS

1. P&ID EMERGENCY CHILLED WATER SYSTEM CB, SGB
   B&R DWG NN564
2. P&ID CONTROL BUILDING FLOOR AND EQUIPMENT DRAINS
   B&R DWG 4N516

FIGURE 9.6-2 P & I Diagram CB SWGR Systems "A" & "B" HVAC

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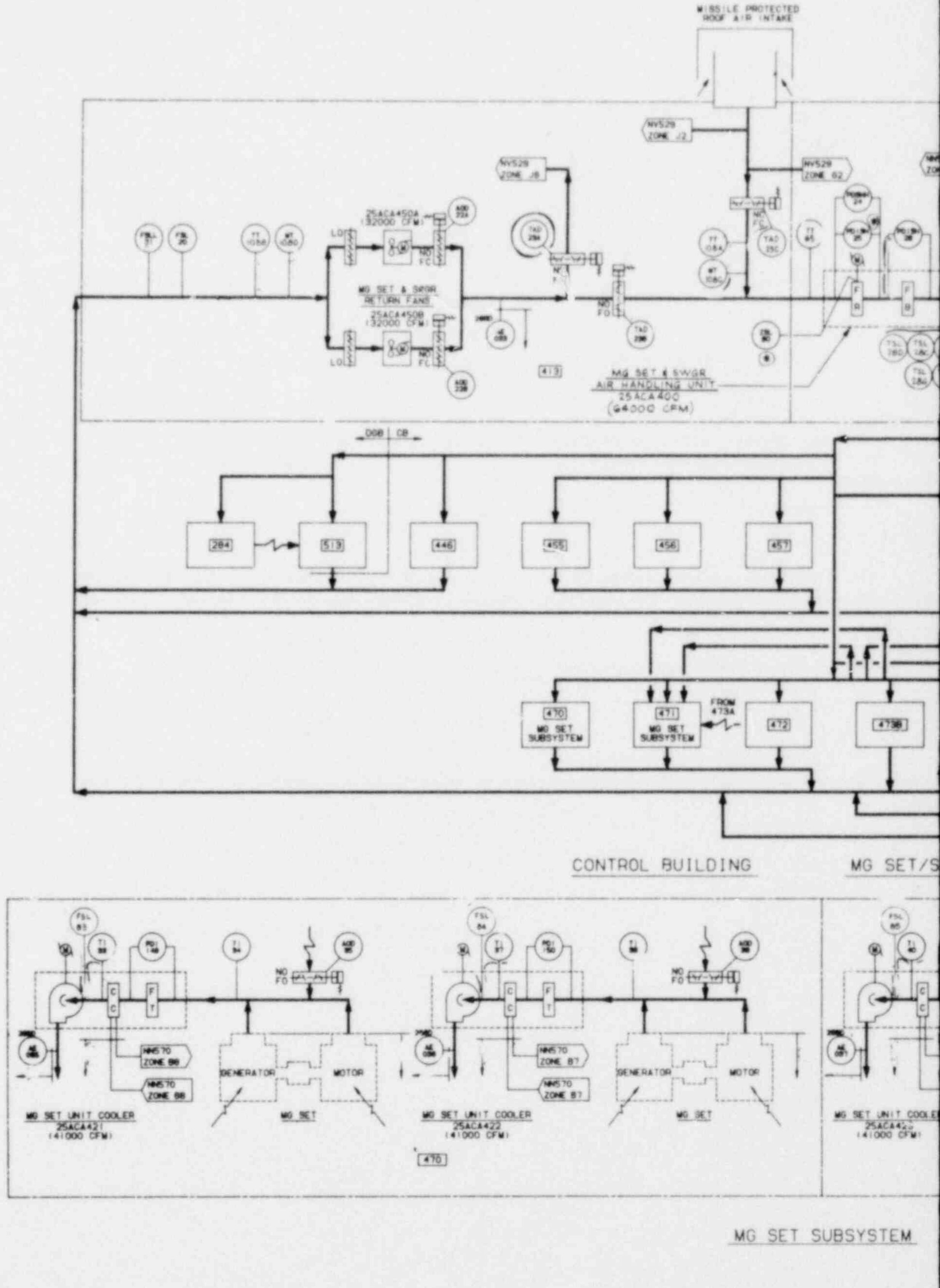
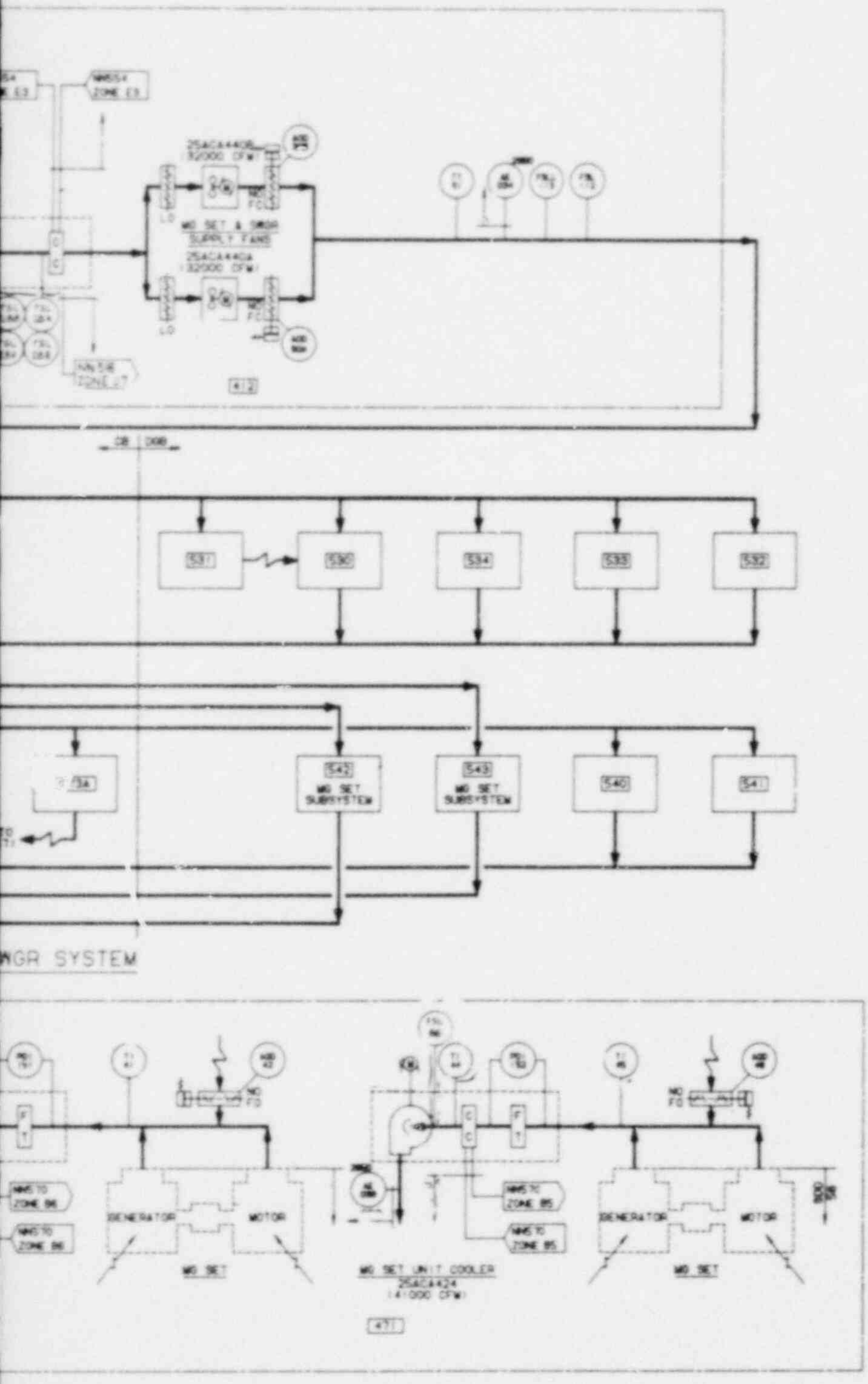


MISSILE PROTECTED ROOF AIR INTAKE
MG SET & SWGR RETURN FANS
25ACA450A (32000 CFM)
25ACA450B (32000 CFM)
MG SET & SWGR AIR HANDLING UNIT
25ACA400
(64000 CFM)
284
513
446
455
456
457
470 MG SET SUBSYSTEM
471 MG SET SUBSYSTEM
FROM 473A
472
473B
CONTROL BUILDING
MG SET/S
MG SET UNIT COOLER
25ACA421
(41000 CFM)
GENERATOR
MOTOR
MG SET
MG SET UNIT COOLER
25ACA422
(41000 CFM)
MG SET SUBSYSTEM
NV528-4
POOR ORIGINAL



GENERAL NOTES

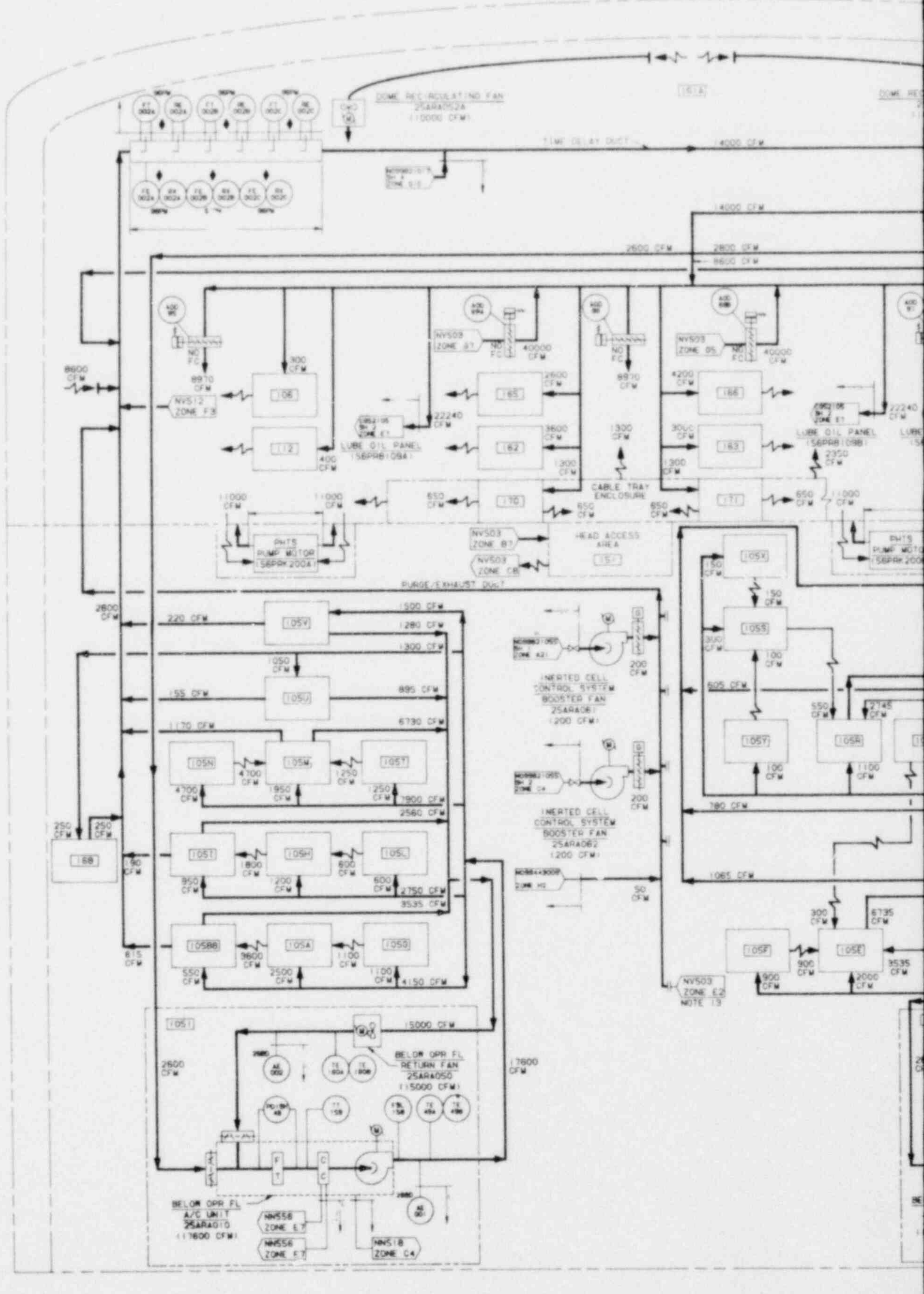
1. SYMBOLS AND ABBREVIATIONS B&R DOC D-0036
2. ALL EQUIPMENT, INSTRUMENT AND CONTROL VALVE NUMBERS ARE PREFIXED BY 25AC UNLESS OTHERWISE NOTED
3. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY "X"
4. SYSTEM CLEANLINESS CLASSIFICATION
5. SEISMIC CATEGORY: III
6. [illegible]
7. [illegible] CLASSIFICATION: NONE
8. SEE AIR FLOW BALANCE SHEETS FOR AIR QUANTITIES APPENDIX C-1
9. ALL INSTRUMENT ITEMS SHOWN MARKED WITH AN ASTERISK* SHALL BE SUPPLIED BY THE ASSOCIATED EQUIPMENT MANUFACTURER

POOR ORIGINAL

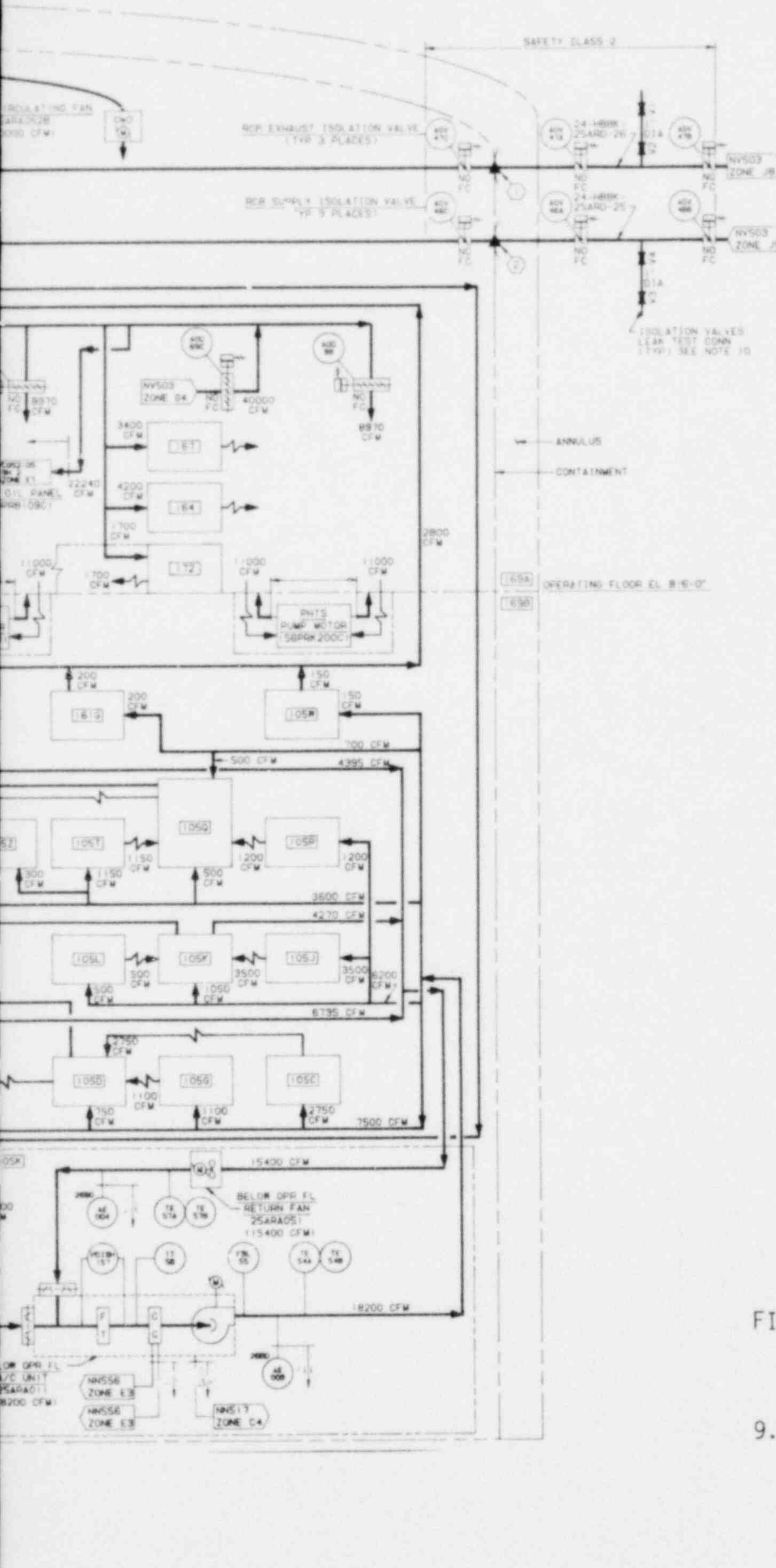
REFERENCE DRAWINGS

1. P&ID NORMAL CHILLED WATER SYSTEM PSB, RCB, CB & SGB — B&R DWG NN554
2. P&ID NORMAL PLANT SERVICE WATER — B&R DWG NN570
3. P&ID CONTROL BUILDING FLOOR & EQPT DRAINS — B&R DWG NN516

FIGURE 9.6-3 P & I Diagram
MG Set & SWGR HVAC


DOME RECIRCULATING FAN
25ARA052A
(10000 CFM)
TIME DELAY DUCT
14000 CFM
2600 CFM
2800 CFM
8600 CFM
8600 CFM
NV512 ZONE F3
LUBE OIL PANEL (56PRB109A)
LUBE OIL PANEL (56PRB109B)
CABLE TRAY ENCLOSURE
HEAD ACCESS AREA
PHTS PUMP MOTOR (56PRK200A)
NV503 ZONE B7
NV503 ZONE C8
PURGE/EXHAUST DUCT
INERTED CELL CONTROL SYSTEM BOOSTER FAN 25ARA061 (200 CFM)
INERTED CELL CONTROL SYSTEM BOOSTER FAN 25ARA062 (200 CFM)
NV503 ZONE E2 NOTE 13
BELOW OPR FL RETURN FAN 25ARA050 (15000 CFM)
BELOW OPR FL A/C UNIT 25ARA010 (17600 CFM)
NN558 ZONE E7
NN556 ZONE E7
NN518 ZONE C4
15000 CFM
17600 CFM

NV501-7



GENERAL NOTES

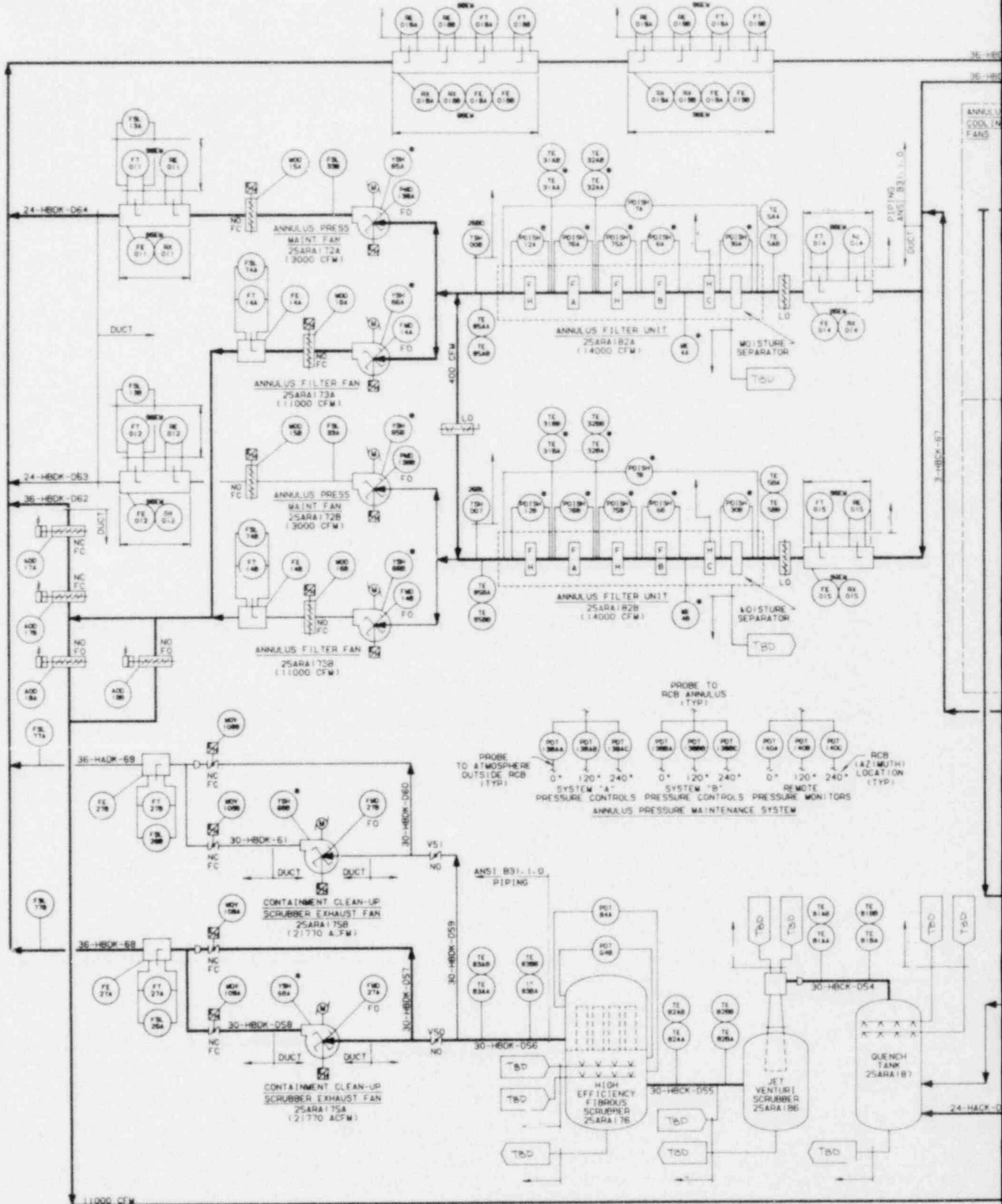
1. SYMBOLS AND ABBREVIATIONS WARD D-0036
2. ALL EQUIPMENT, INSTRUMENT AND CONTROL VALVE NUMBERS ARE PREFIXED BY 25AR U05
3. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY K
4. SYSTEM CLEANLINESS CLASSIFICATION:
5. SEISMIC CATEGORY I
6. DELETED
7. SAFETY CLASSIFICATION: NONE U05
8. SEE AIR FLOW BALANCE SHEETS FOR AIR QUANTITIES APPENDIX C-1
9. PURGE EXHAUST DUCT IS USED TO DE-INERT CELLS USING PORTABLE EXHAUST FAN WITH FLEXIBLE DUCTS SEE DWGS NV575, NV576, NV577
10. ALL TEST NIPPLES WITH REFERENCE TO THIS NOTE SHALL HAVE THREADED ENDS
11. CERTAIN EQUIPMENT SHOWN ON THIS DRAWING IS INCLUDED IN THE PLANT PROTECTION SYSTEM (PPS). THIS EQUIPMENT IS SPECIFICALLY IDENTIFIED ON THE DRAWING AS FOLLOWS: THE SYMBOL ♦ DESIGNATES PPS EQUIPMENT
12. THIS DRAWING INCLUDES EQUIPMENT IDENTIFIED AS PART OF THE PLANT PROTECTION SYSTEM (PPS). BEFORE MODIFYING OR MAINTAINING THE EQUIPMENT SO IDENTIFIED THE APPROVAL OF THE COGNIZANT PERSONNEL FOR THE PPS MUST BE OBTAINED
13. BLIND FLANGED CONNECTION SHALL BE LOCATED WITHIN 20-0" OF EACH INERTED CELL PURGE CONNECTION
14. ALL REACTOR CONTAINMENT PENETRATION NUMBERS ARE PREFIXED BY 25ARHR

POOR ORIGINAL

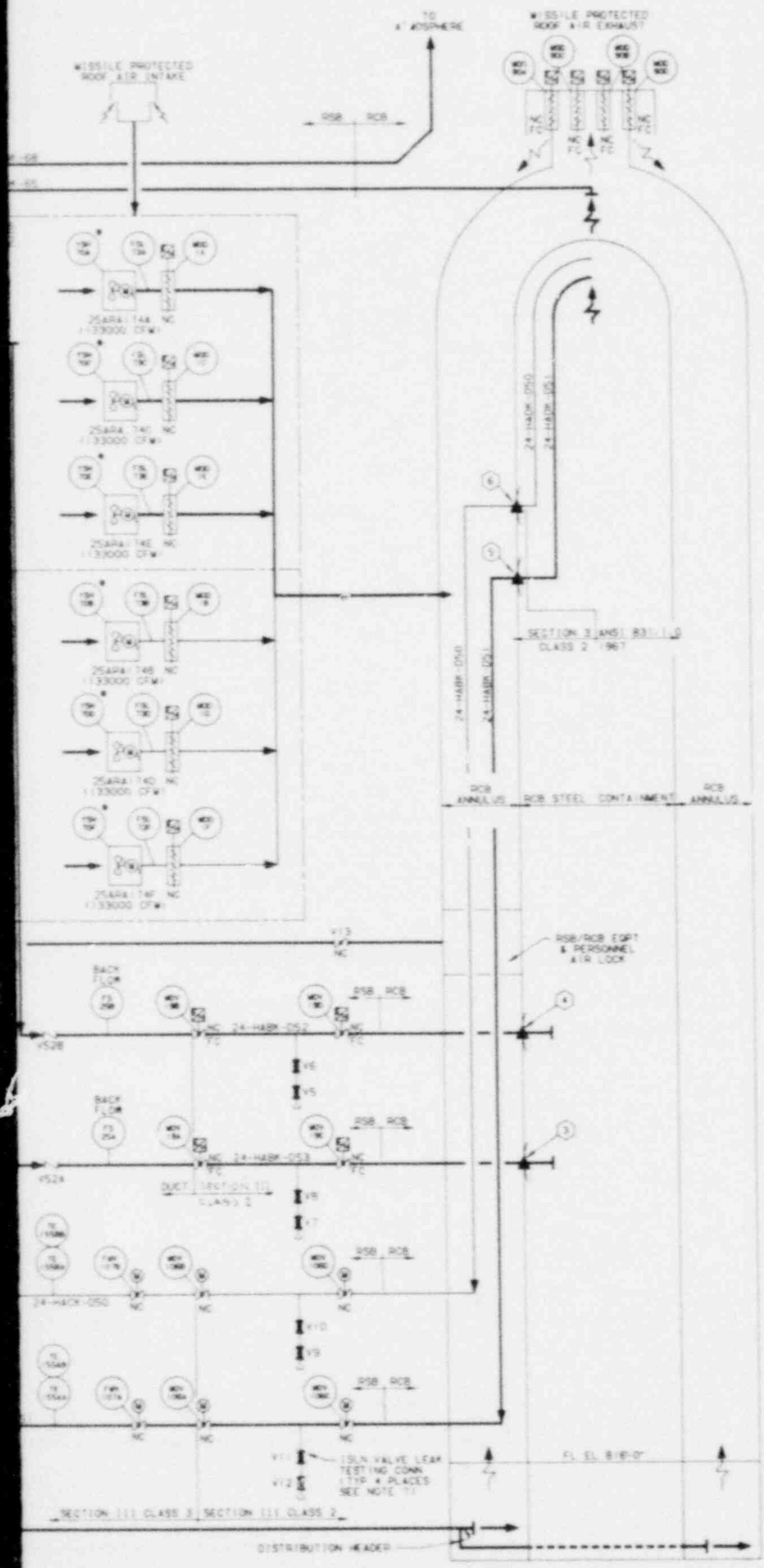
REFERENCE DRAWINGS

1. P&ID RCB ANNULUS & CONTAINMENT CLEAN-UP HVAC — B&R DWG NV502
2. P&ID RCB HVAC — B&R DWG NV503
3. P&ID RCB RGCS — B&R DWG NV575
4. P&ID RCB RGCS — B&R DWG NV576
5. P&ID RCB RGCS — B&R DWG NV577
6. P&ID NORMAL CHW SYS RCB & SGB — B&R DWG NW556
7. RCB-P&ID PRIMARY SODIUM REMOVAL & DECONTAMINATION SYSTEM — AI DWG N099443009
8. P&ID RCB N_2 DISTRIBUTION — AI DWG N09982I055
9. P&ID RCB N_2 DISTRIBUTION — AI DWG N09982I017
10. ICD SODIUM PUMP DRIVE SYSTEM MAIN DRIVE ASSY BEARING OIL PNL — GE DWG C052I05
11. ICD SODIUM PUMP DRIVE SYSTEM MAIN PUMP DRIVE ASSY — GE DWG C052I09 SH2
12. P&ID RCB/RSB RAPS & CAPS HVAC — B&R DWG NV512
13. FLOW DIAGRAM RCB FLOOR & EQUIPMENT DRAINS — B&R DWG NN517
14. FLOW DIAGRAM RCB FLOOR & EQUIPMENT DRAINS — B&R DWG NN518

FIGURE 9.6-4 P & I Diagram
RCB HVAC


24-HBDK-D64
ANNULUS PRESS MAINT FAN
25ARA172A
(3000 CFM)
ANNULUS FILTER FAN
25ARA173A
(11000 CFM)
ANNULUS PRESS MAINT FAN
25ARA172B
(3000 CFM)
ANNULUS FILTER FAN
25ARA173B
(11000 CFM)
ANNULUS FILTER UNIT
25ARA182A
(14000 CFM)
ANNULUS FILTER UNIT
25ARA182B
(14000 CFM)
MOISTURE SEPARATOR
DUCT
400 CFM
24-HBDK-D63
36-HBDK-D62
PIPING ANSI B31.1.0
PROBE TO RCB ANNULUS (TYP)
PROBE TO ATMOSPHERE OUTSIDE RCB (TYP)
SYSTEM "A" PRESSURE CONTROLS
SYSTEM "B" PRESSURE CONTROLS
REMOTE PRESSURE MONITORS
RCB (AZIMUTH) LOCATION (TYP)
ANNULUS PRESSURE MAINTENANCE SYSTEM
36-HADK-68
30-HBDK-61
30-HBDK-D60
CONTAINMENT CLEAN-UP SCRUBBER EXHAUST FAN
25ARA175B
(21770 ACFM)
ANSI B31.1.0 PIPING
36-HBDK-68
30-HBDK-D58
30-HBDK-D57
30-HBDK-D59
30-HBDK-056
CONTAINMENT CLEAN-UP SCRUBBER EXHAUST FAN
25ARA175A
(21770 ACFM)
HIGH EFFICIENCY FIBROUS SCRUBBER
25ARA176
30-HBCK-D55
JET VENTURI SCRUBBER
25ARA186
30-HBCK-D54
QUENCH TANK
25ARA181
TBD
11000 CFM

NV502-2



GENERAL NOTES

1. SYMBOLS AND ABBREVIATIONS MARD DOC D-0036
2. ALL EQUIPMENT, INSTRUMENT AND CONTROL VALVE NUMBERS ARE PREFIXED BY 25AR UOS
3. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY "K"
4. SYSTEM CLEANLINESS CLASSIFICATION
5. SEISMIC CATEGORY
6. SAFETY CLASSIFICATION 3 UOS
7. ALL TEST NIPPLES WITH REF TO THIS NOTE SHALL HAVE THREADED ENDS
8. ALL DIM ARE IN INCHES UOS
9. PIPING CODE CLASSIFICATIONS: ASME SECTION III CLASS 2 & 3 ANSI B31.1.0-1967
10. ALL REACTOR CONTAINMENT PENENTRATION NUMBERS ARE PREFIXED BY 25ARHR
11. ALL LINE NUMBERS ARE ABBREVIATED AS PER THE FOLLOWING EXAMPLE UNLESS OTHERWISE NOTED 24-HACK-25ARD-50 IS WRITTEN AS 24-HACK-050
12. ALL INSTRUMENT ITEMS SHOWN MARKED WITH AN ASTERISK * SHALL BE SUPPLIED BY THE ASSOCIATED EQUIPMENT MANUFACTURER

POOR ORIGINAL

REFERENCE DRAWINGS

1. P&ID RSB HVAC — B&R DWG NY501
2. P&ID RSB HVAC — B&R DWG NY503

FIGURE 9.6-5 P & I Diagram
RCB Annulus &
Cont. Clean-Up
HVAC

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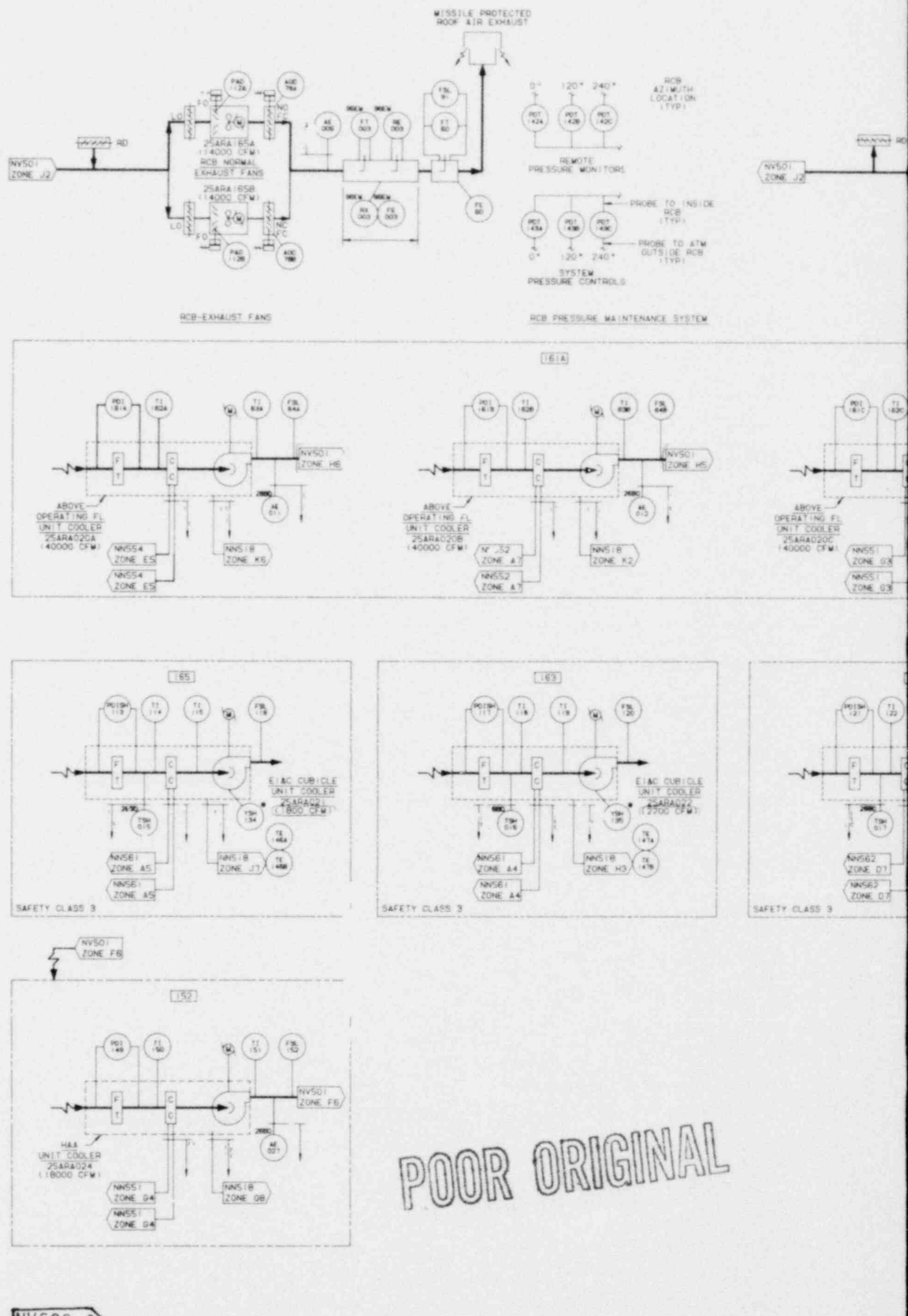
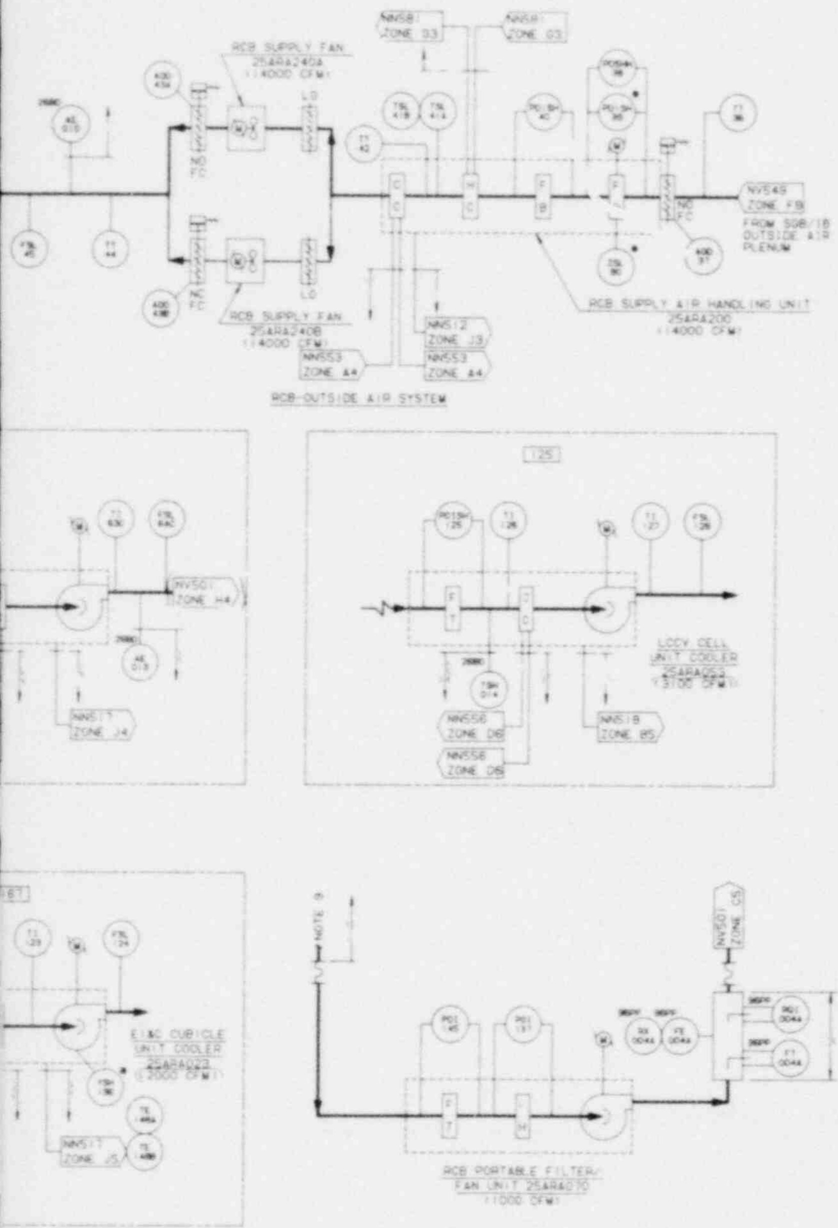


POOR ORIGINAL



GENERAL NOTES

1. SYMBOLS AND ABBREVIATIONS WARD-D-0036
2. ALL EQUIPMENT, INSTRUMENT AND CONTROL VALVE NUMBERS ARE PREFIXED BY 25AR UOS
3. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY "K"
4. SYSTEM CLEANLINESS CLASSIFICATION:
5. SEISMIC CATEGORY:
7. SAFETY CLASSIFICATION: NONE UOS
8. SEE AIR FLOW BALANCE SHEETS FOR AIR QUANTITIES APPENDIX C-1
9. TO BE CONNECTED TO SYSTEM 28 INERTED CELLS PURGE CONNECTIONS FOR DE-INERTING
10. ALL INSTRUMENT ITEMS SHOWN MARKED WITH ASTERISK SHALL BE SUPPLIED BY THE ASSOCIATED EQUIPMENT MANUFACTURER

POOR ORIGINAL

REFERENCE DRAWINGS

1. P&ID RCB HVAC — B&R DWG NV501
2. P&ID RCB ANNULUS & CONTAINMENT CLEAN-UP HVAC — B&R DWG NV502
3. P&ID SGB IB HVAC — B&R DWG NV549
4. P&ID NORMAL CHW SYS RCB, SGB — B&R DWG NN551
5. P&ID NORMAL CHW SYS RCB & SGB — B&R DWG NN552
6. P&ID NORMAL CHW SYS SGB — B&R DWG NN553
7. P&ID NORMAL CHW SYS PSB, RCB, CB & SGB — B&R DWG NN554
8. P&ID NORMAL CHW SYS RCB, SGB — B&R DWG NN556
9. P&ID EMERGENCY CHW SYS SGB, RCB — B&R DWG NN561
10. P&ID EMERGENCY CHW SYS RCB, PSB & SGB — B&R DWG NN562
11. P&ID HIGH TEMP HOT WATER HEATING SYS — B&R DWG NN581
12. FLOW DIAGRAM SGB FLOOR AND EQUIPMENT DRAINS — B&R DWG NN512
13. FLOW DIAGRAM RCB FLOOR AND EQUIPMENT DRAINS WEST SIDE — B&R DWG NN518
14. FLOW DIAGRAM RCB FLOOR & EQUIPMENT DRAINS EAST SIDE — B&R DWG NN517

FIGURE 9.6-6 P & I Diagram RCB HVAC

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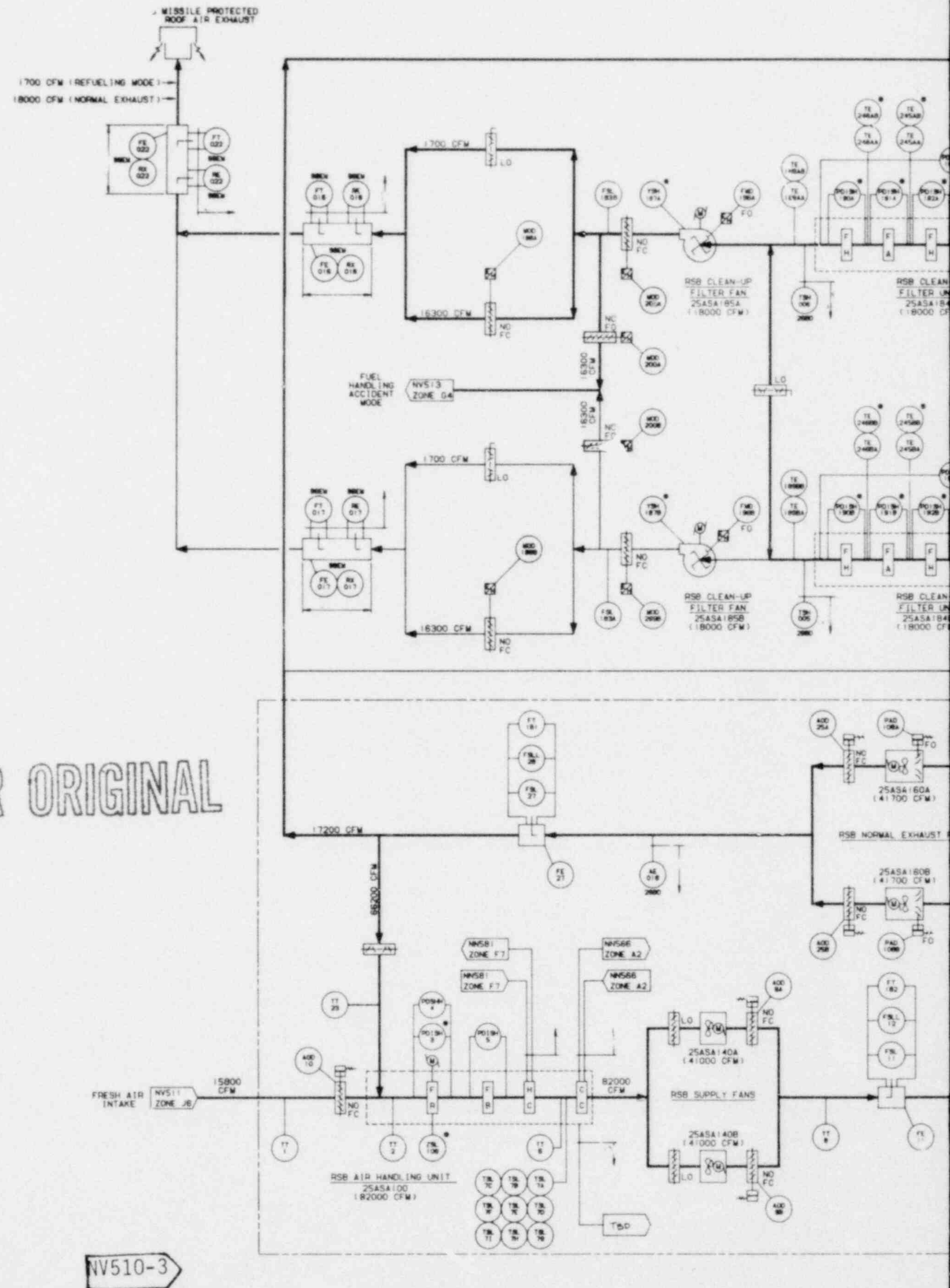
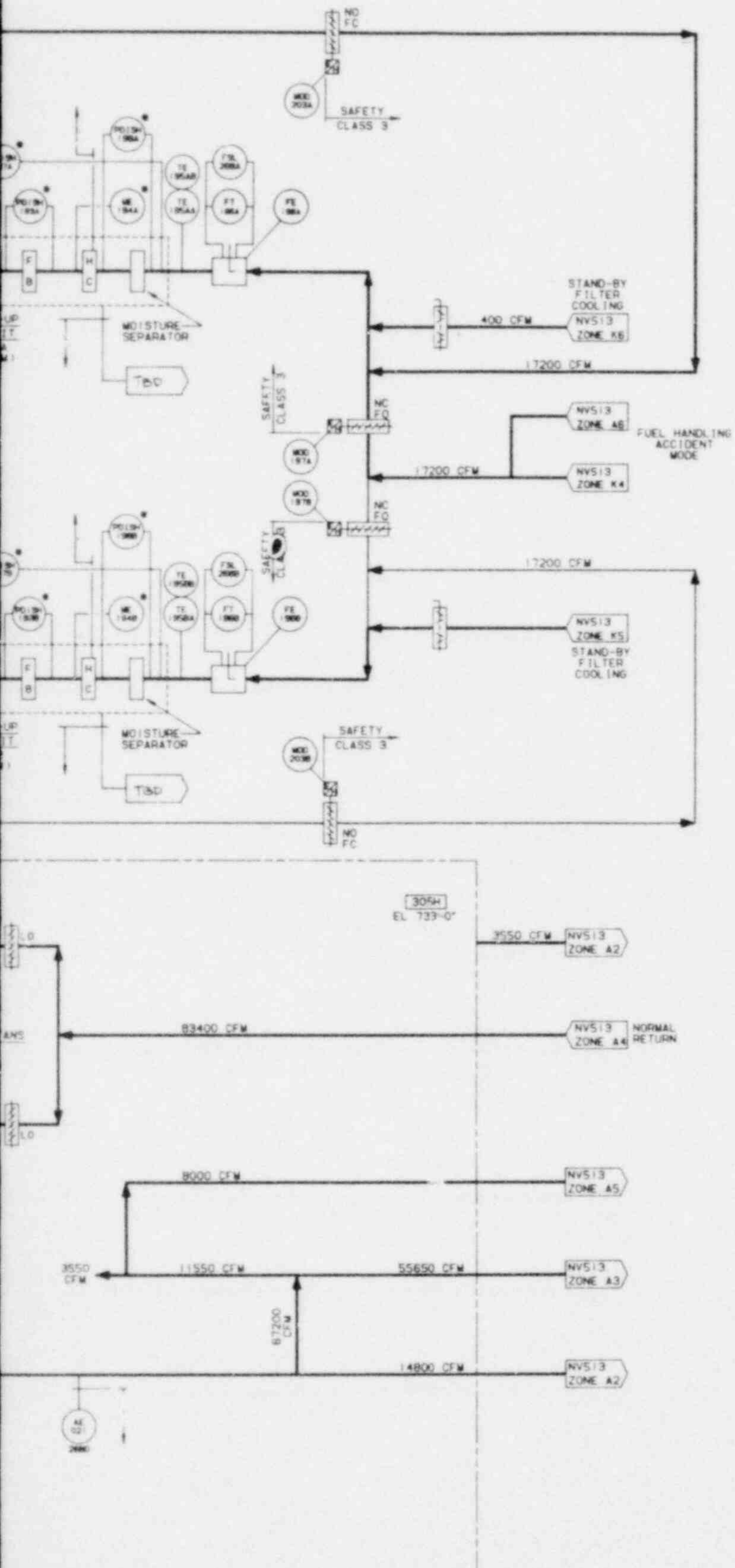

MISSILE PROTECTED
ROOF AIR EXHAUST
1700 CFM (REFUELING MODE)
18000 CFM (NORMAL EXHAUST)
1700 CFM
16300 CFM
FUEL
HANDLING
ACCIDENT
MODE
NV513
ZONE G4
RSB CLEAN-UP
FILTER FAN
25ASA185A
(18000 CFM)
RSB CLEAN-UP
FILTER FAN
25ASA185B
(18000 CFM)
RSB CLEAN
FILTER UN
25ASA184
(18000 CF
17200 CFM
98200 CFM
25ASA160A
(41700 CFM)
RSB NORMAL EXHAUST
25ASA160B
(41700 CFM)
NN581
ZONE F7
NN566
ZONE A2
FRESH AIR
INTAKE
NV511
ZONE J8
15800
CFM
82000
CFM
RSB SUPPLY FANS
25ASA140A
(41000 CFM)
25ASA140B
(41000 CFM)
RSB AIR HANDLING UNIT
25ASA100
(82000 CFM)
TBP
NV510-3



GENERAL NOTES

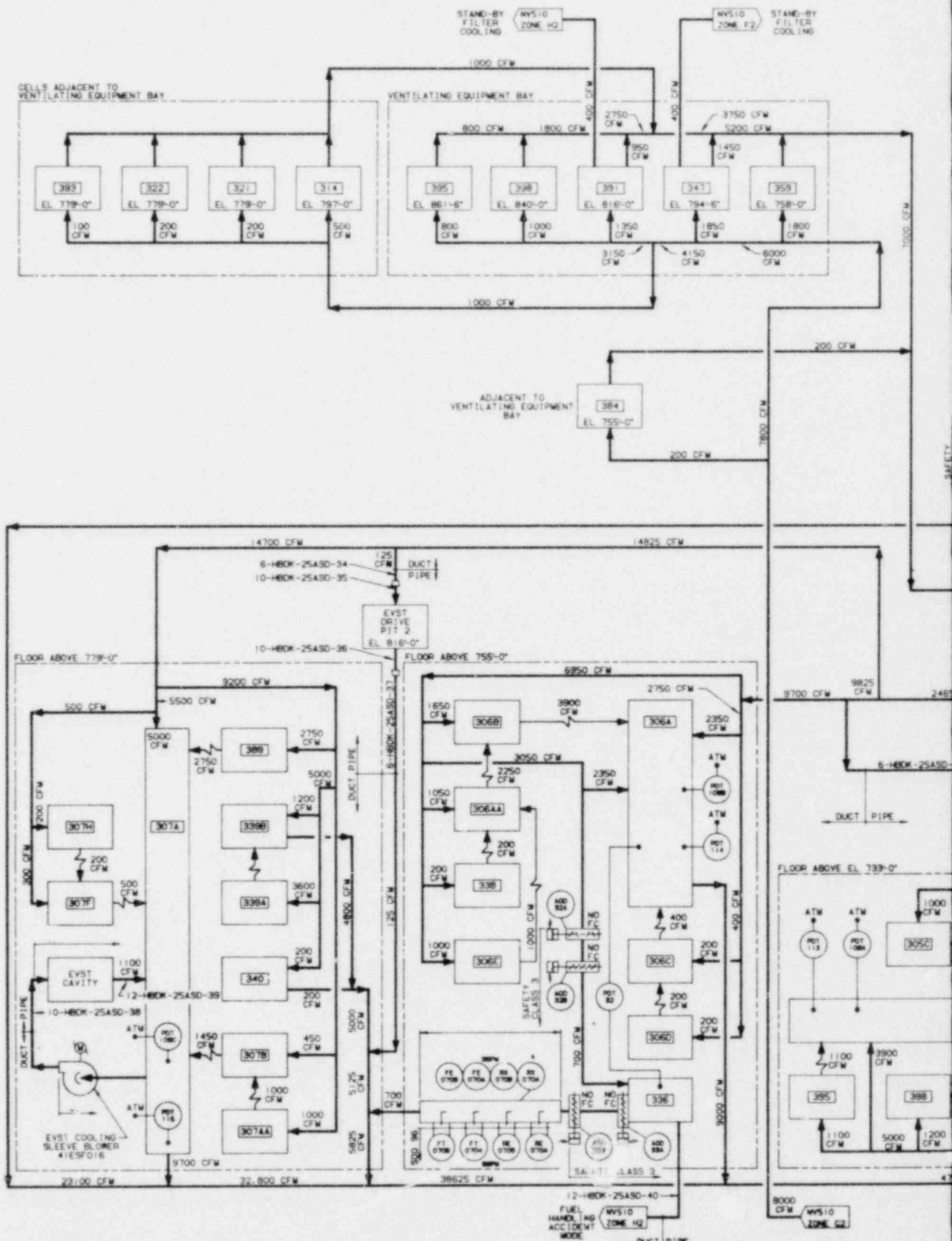
1. SYMBOLS AND ABBREVIATIONS WARD DOC D-0036
2. ALL EQUIPMENT INSTRUMENT AND CONTROL VALVE NUMBERS ARE ARE PREFIXED BY 25AS UOS
3. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY "K"
4. SYSTEM CLEANLINESS CLASSIFICATION:
5. SEISMIC CATEGORY:
6. SAFETY CLASSIFICATION: NONE UOS
7. ALL INSTRUMENT ITEMS SHOWN MARKED WITH AN ASTERESK * SHALL BE SUPPLIED BY THE ASSOCIATED EQUIPMENT MANUFACTURER
8. AIR QUANTITIES ARE LISTED FOR NORMAL OPERATION

POOR ORIGINAL

REFERENCE DRAWINGS

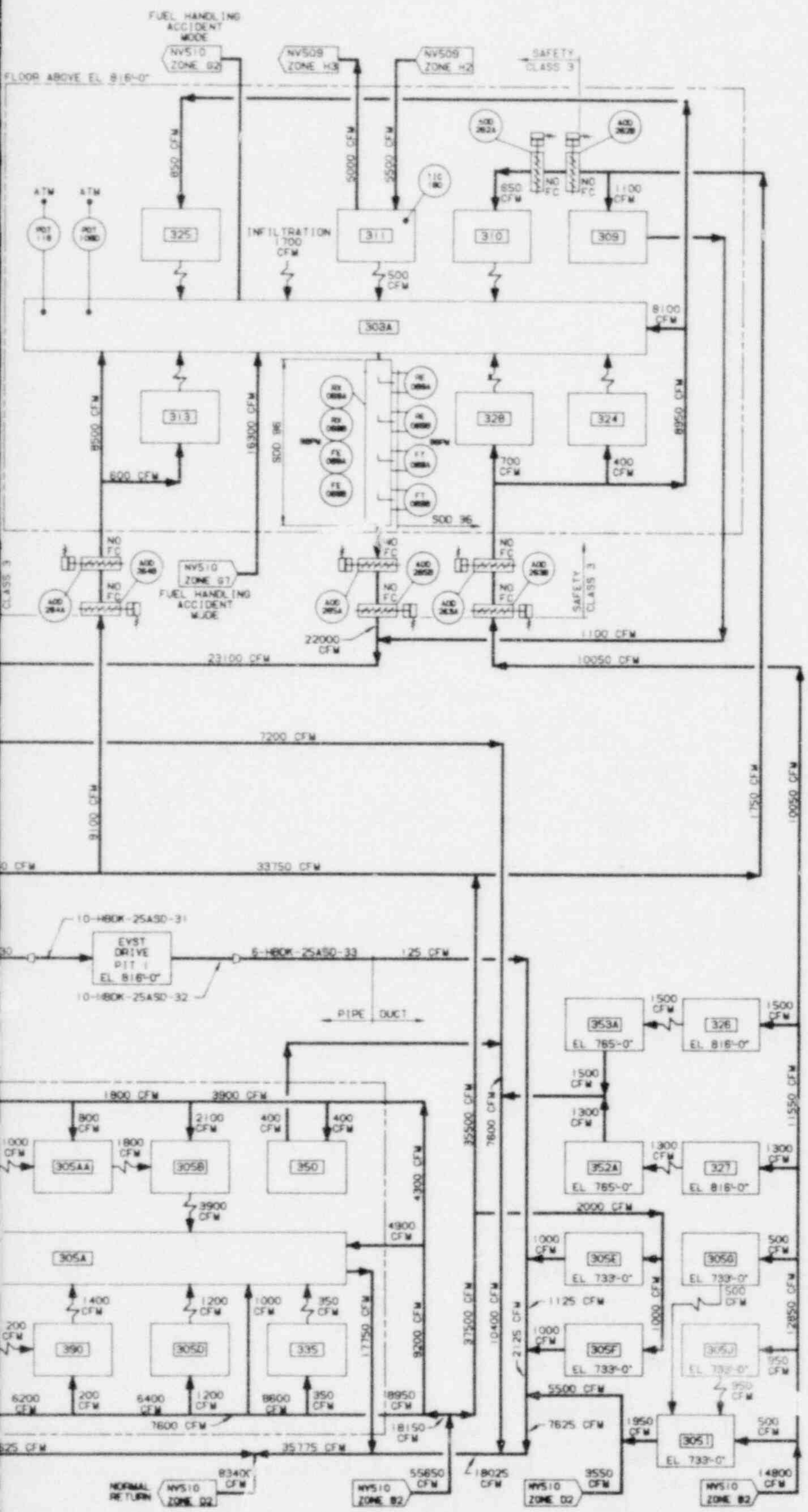
1. P&ID RSB HVAC — B&R DWG NV509
2. P&ID RSB HVAC — B&R DWG NV511
3. P&ID RSB HVAC — B&R DWG NV513
4. P&ID NORMAL CHILLED WATER SYSTEM RSB — B&R DWG NN566
5. P&ID HIGH TEMP HOT WATER HEATING SYSTEM — B&R DWG NN581

FIGURE 9.6-7 P & I Diagram
RSB HVAC



NV513-0

POOR ORIGINAL



GENERAL NOTES

1. SYMBOLS AND ABBREVIATIONS WARD DOC D-0036
2. ALL EQUIPMENT INSTRUMENT AND CONTROL VALVE NUMBERS ARE ARE PREFIXED BY 25AS UOS
3. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY "K"
4. SYSTEM CLEANLINESS CLASSIFICATION:
5. SEISMIC CATEGORY:
6. SAFETY CLASSIFICATION: NONE UOS
7. ALL INSTRUMENT ITEMS SHOWN MARKED WITH AN ASTERESK * SHALL BE SUPPLIED BY THE ASSOCIATED EQUIPMENT MANUFACTURER
8. AIR QUANTITIES ARE LISTED FOR NORMAL OPERATION

POOR ORIGINAL

REFERENCE DRAWINGS

1. P&ID RSB HVAC — B&R DWG NV509
2. P&ID RSB HVAC — B&R DWG NV510
3. P&ID RSB HVAC — B&R DWG NV511

FIGURE 9.6-7a P & I Diagram RSB HVAC

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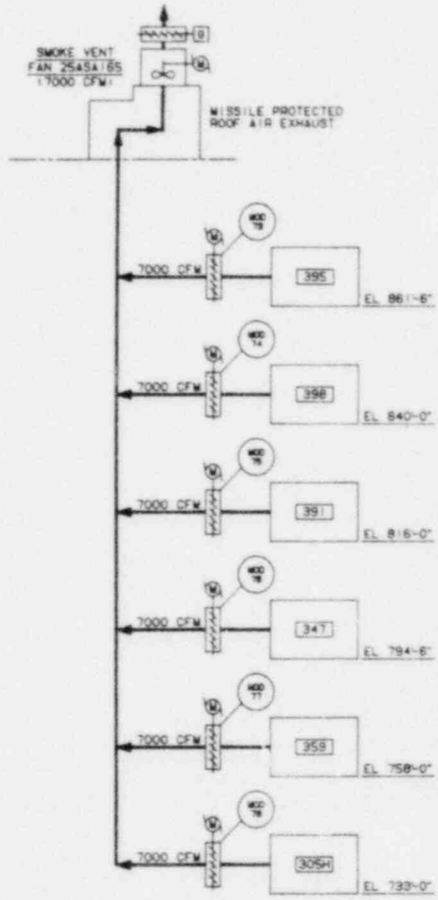
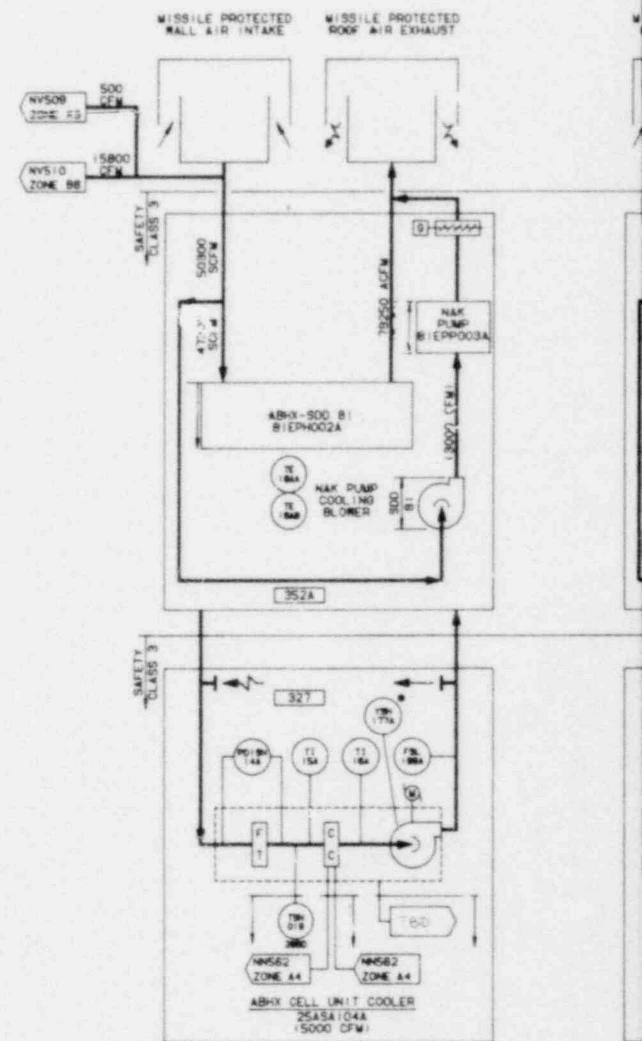
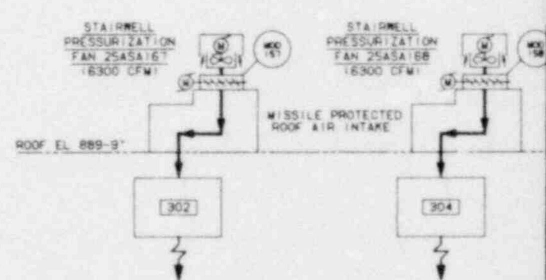
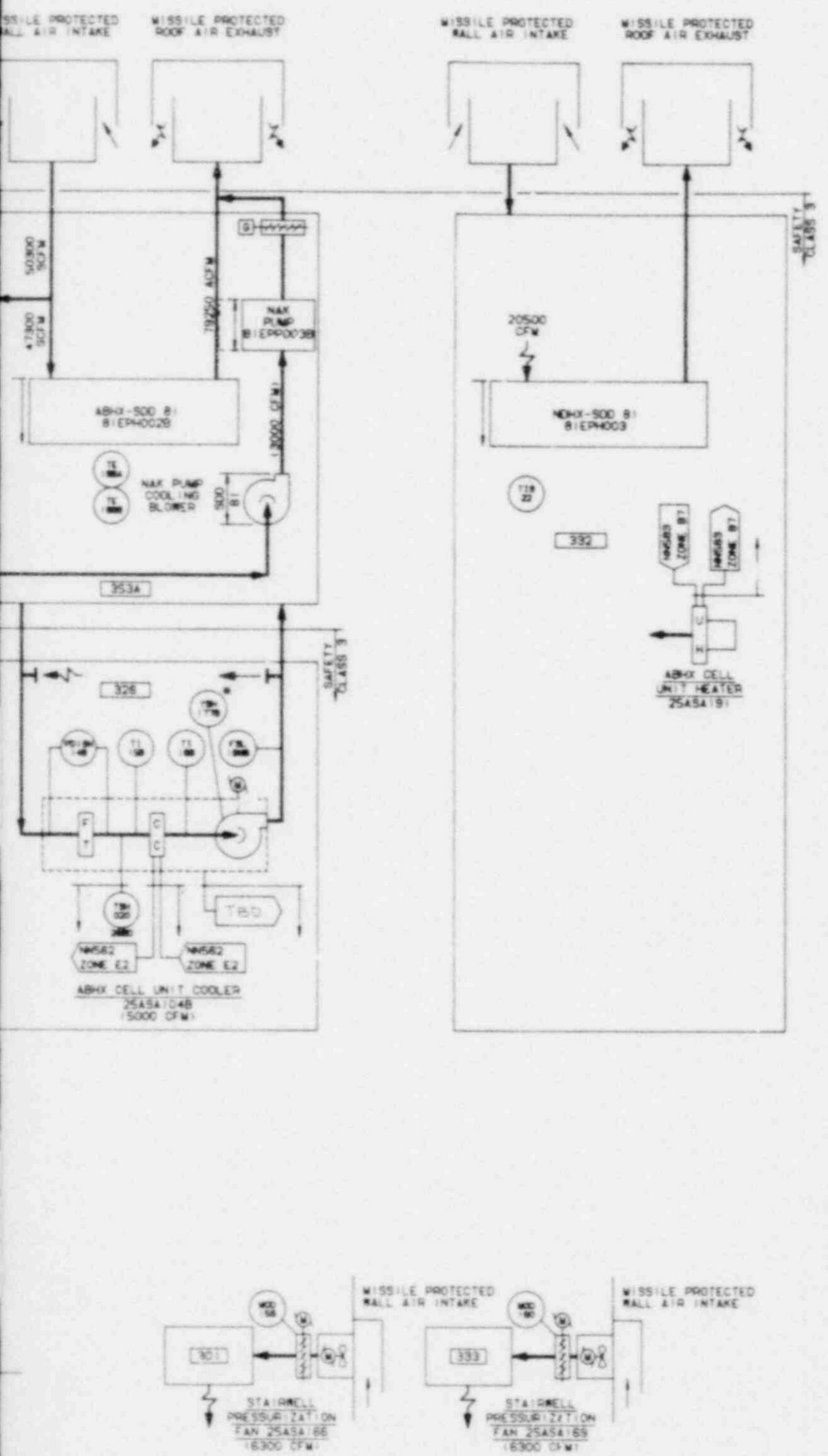






NV511-4

POOR ORIGINAL



GENERAL NOTES

1. SYMBOLS AND ABBREVIATIONS WARD DOC D-0036
2. ALL EQUIPMENT, INSTRUMENT AND CONTROL VALVE NUMBERS ARE PREFIXED BY 25AS UOS
3. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY "K"
4. SYSTEM CLEANLINESS CLASSIFICATION:
5. SEISMIC CATEGORY:
6. SAFETY CLASSIFICATION: NONE UOS

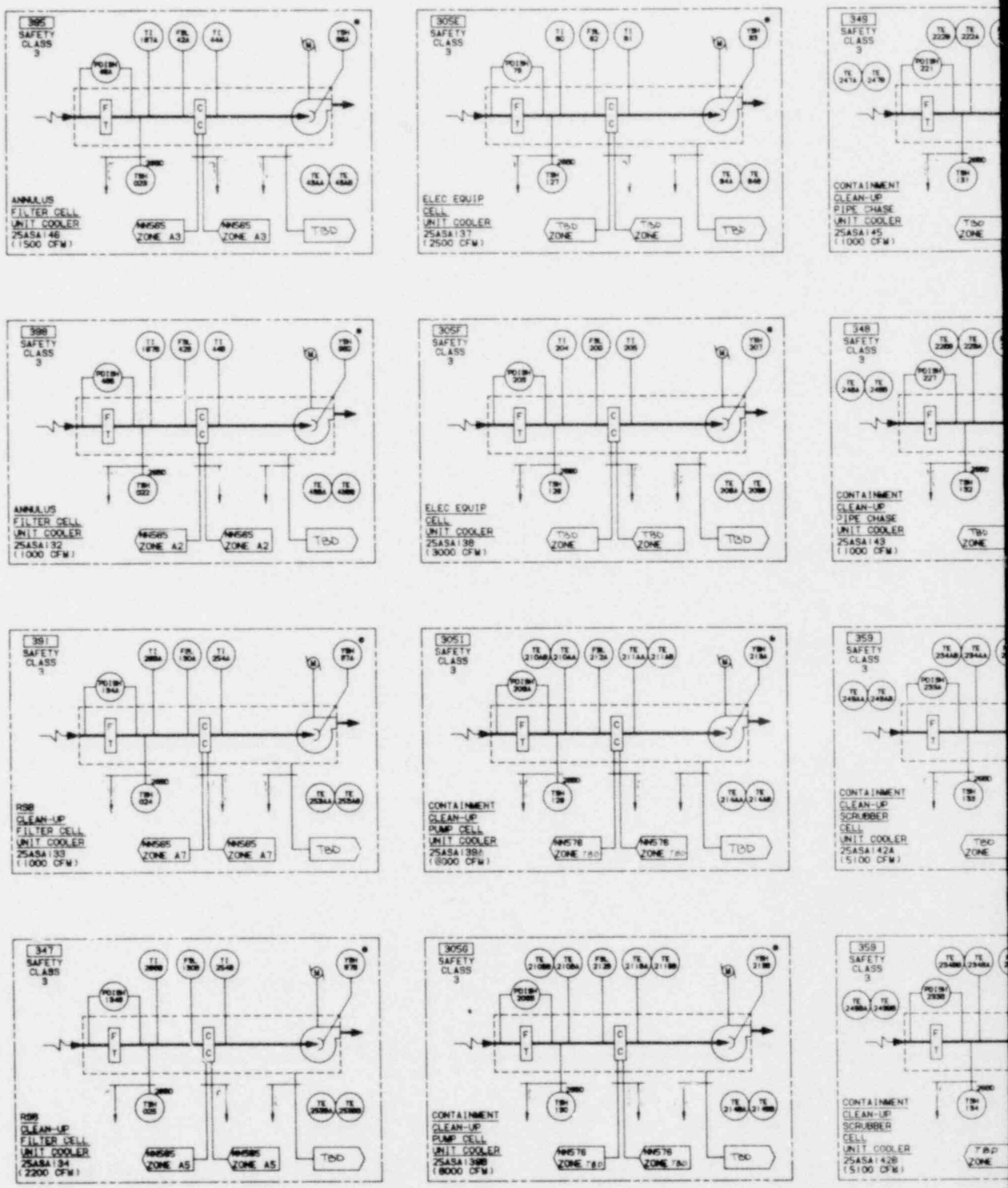
POOR ORIGINAL

REFERENCE DRAWINGS

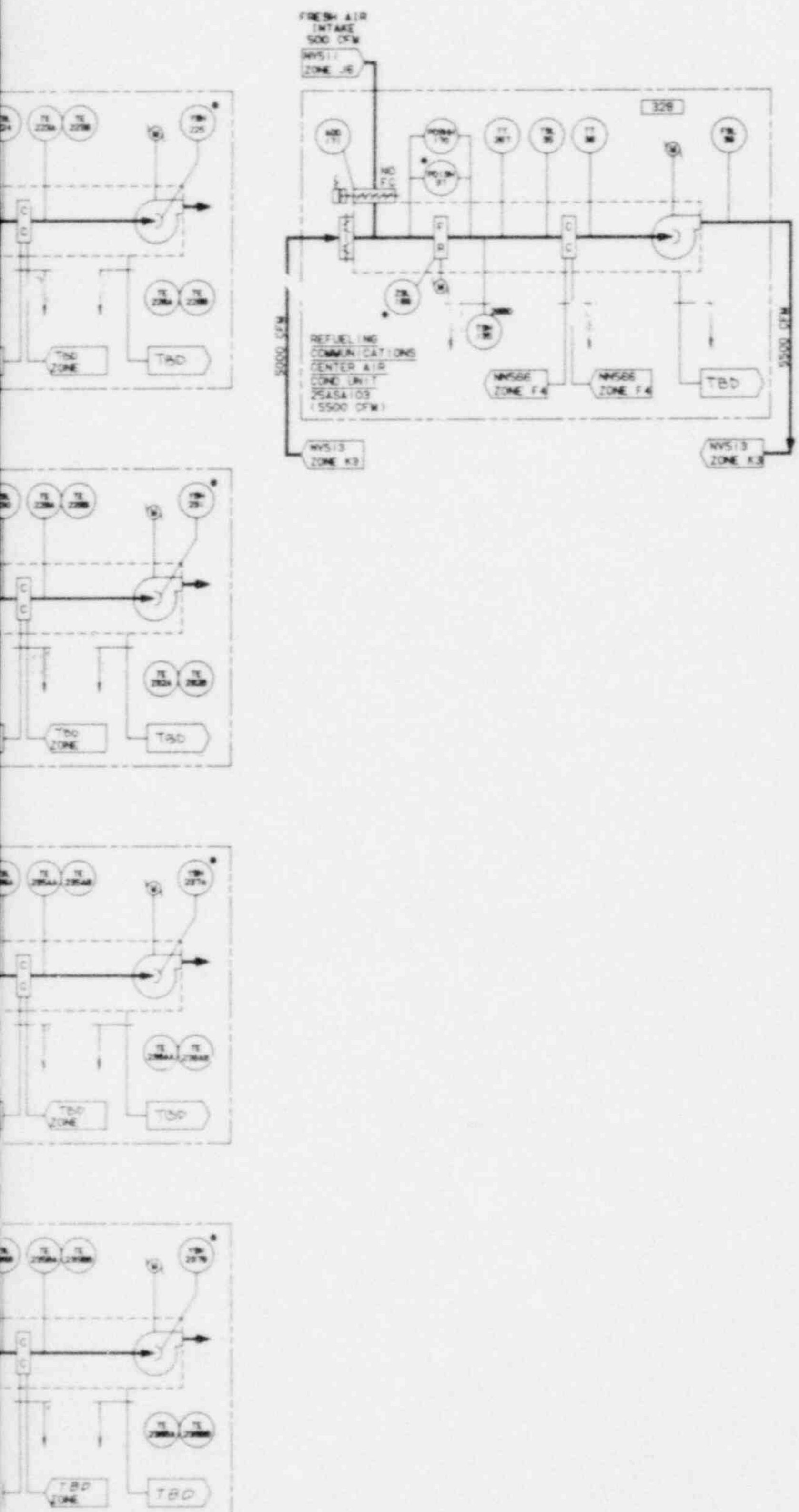
1. P&ID RSB HVAL
   B&R DWG NV508
2. P&ID RSB HVAC
   B&R DWG NV510
3. P&ID RSB HVAC
   B&R DWG NV513
4. P&ID EMERGENCY CHILLED WATER SYSTEM RCB, RSB & SGE
   B&R DWG NM562
5. RSB-EVST AIR BLAST HEAT EXCHANGER INTERFACE
   AI DWG CA53008
6. P&ID HOT WATER HEATING SYSTEM RSB & RSB-RWB
   B&R DWG NM583

FIGURE 9.6-8 P & I Diagram
RSB HVAC

POOR ORIGINAL



NV509-0



GENERAL NOTES

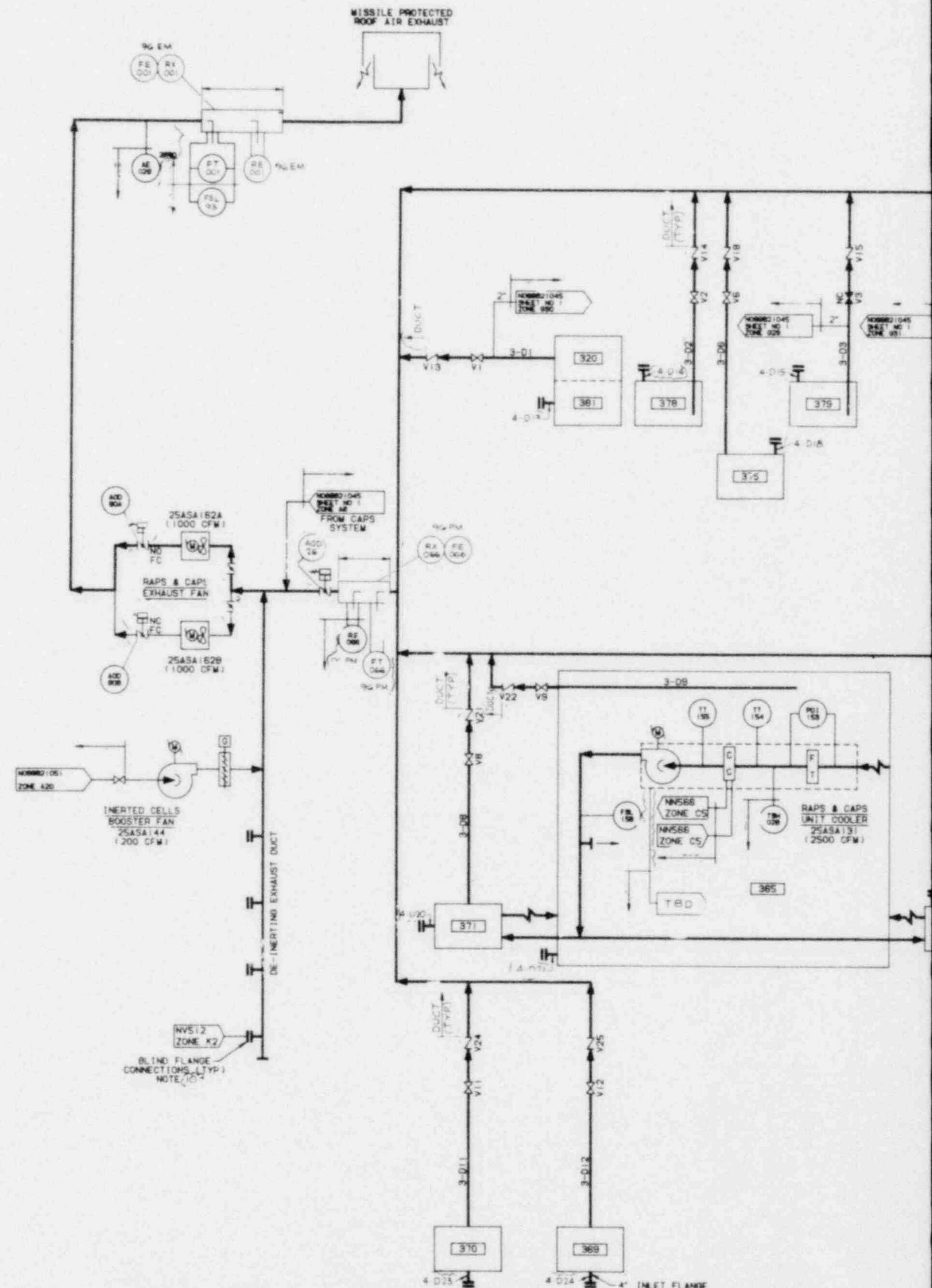
1. SYMBOLS AND ABBREVIATIONS B&R DOC D-0036
2. ALL EQUIPMENT INSTRUMENT AND CONTROL VALVE NUMBERS ARE ARE PREFIXED BY 25AS U05
3. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY "K"
4. SYSTEM CLEANLINESS CLASSIFICATION:
5. SEISMIC CATEGORY:
6. SAFETY CLASSIFICATION: NONE U05
7. ALL INSTRUMENTS SHOWN MARKED WITH AN ASTERISK * SHALL BE SUPPLIED BY THE ASSOCIATED EQUIPMENT MANUFACTURER

POOR ORIGINAL

REFERENCE DRAWINGS

1. P&ID RSB HVAC — B&R DWG NV510
2. P&ID RSB HVAC — B&R DWG NV511
3. P&ID NORMAL CHILLED WATER SYSTEM RSB — B&R DWG NW586
4. P&ID EMERGENCY CHILLED WATER SYSTEM RSB — B&R DWG NW585

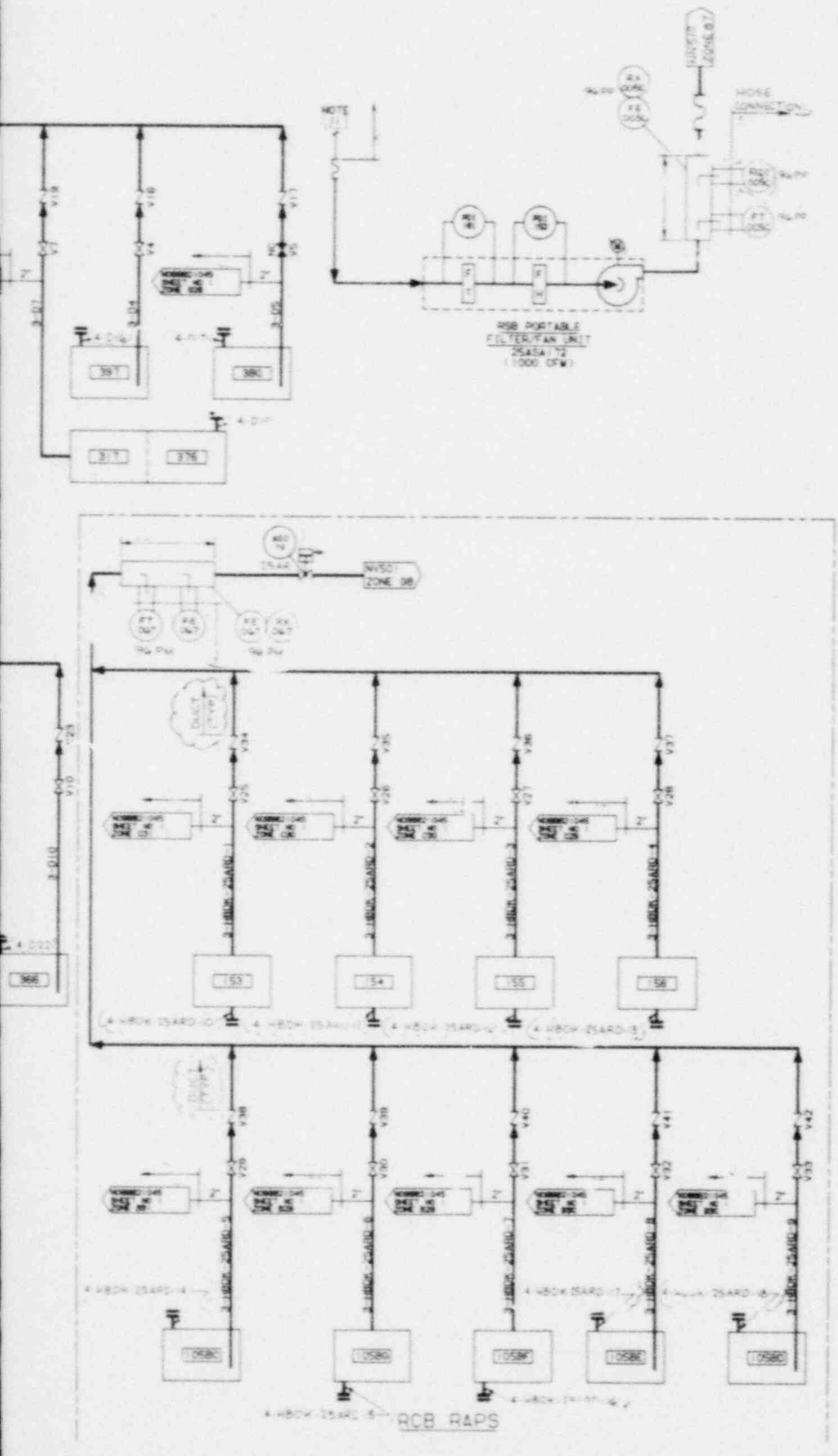
FIGURE 9.6-8a P & I Diagram RSB HVAC



RSB RAPS & CAPS

NV512-4

POOR ORIGINAL



GENERAL NOTES

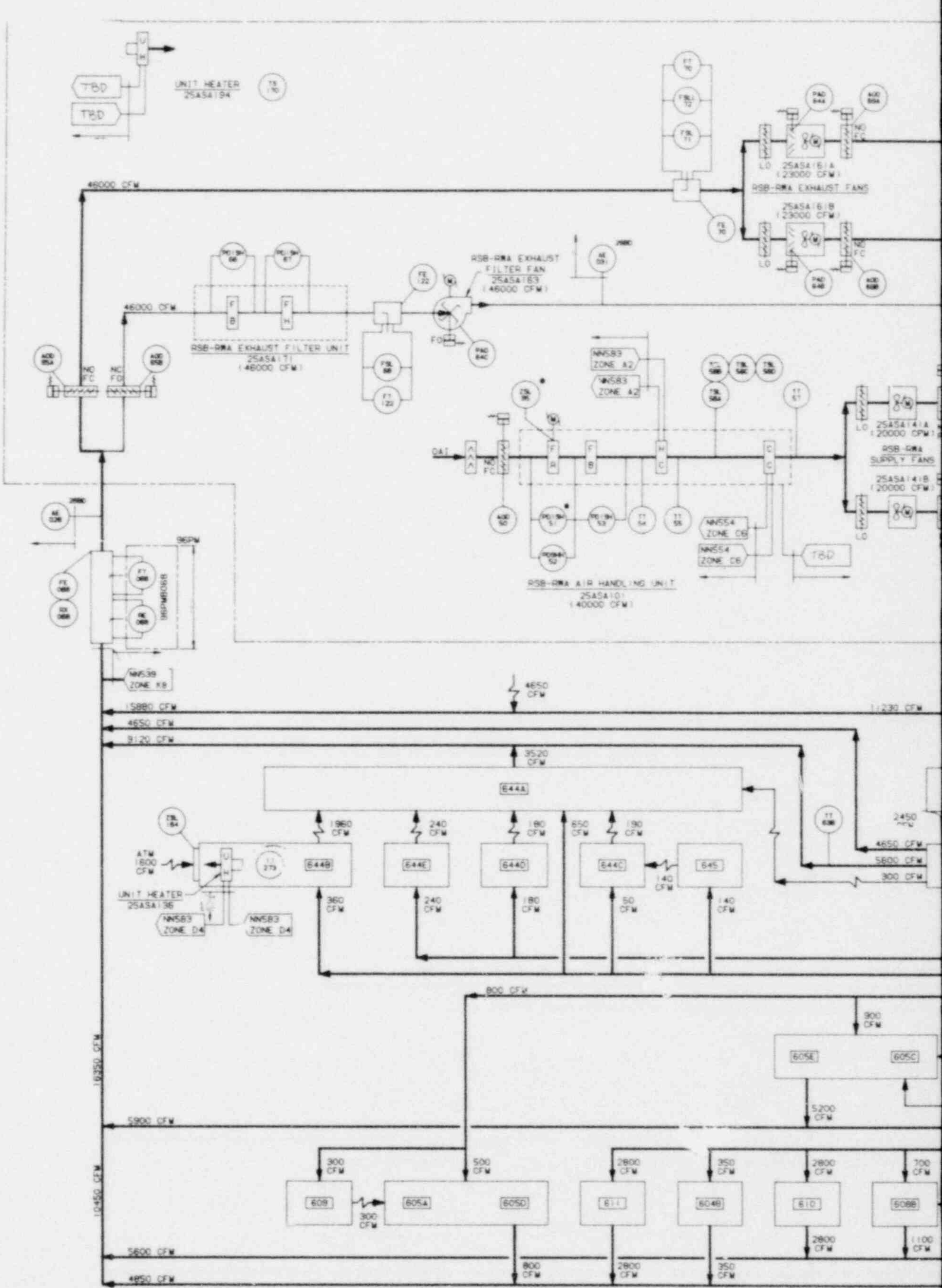
1. SYMBOLS AND ABBREVIATIONS HARD DOC D-0036
2. ALL EQUIPMENT, INSTRUMENT AND CONTROL VALVE NUMBERS ARE PREFIXED BY 25AS
3. LINE NUMBERS ARE ABBREVIATED AS PER THE FOLLOWING EXAMPLE UOS: 3-HBDK-25ASD-1 IS WRITTEN AS
4. SYSTEM CLEANLINESS CLASSIFICATION:
5. SEISMIC CATEGORY: III UOS
6.
7. SAFETY CLASSIFICATION: NONE UOS
8. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY K
9. SEE AIR FLOW BALANCE SHEETS FOR AIR QUANTITIES APPENDIX C-1
10. BLIND FLANGED CONNECTION SHALL BE LOCATED WITHIN 20'-0" OF EACH INERTED CELL PURGE CONNECTION
11. TO BE CONNECTED TO INERTED CELL PURGE CONNECTION (FOR DE-INERTING)

REFERENCE DRAWINGS

1. P&ID CELL ATM PROCESSING SYSTEM AI DWG NO99821045
2. P&ID RSB N_2 DISTRIBUTION AI DWG NO99821051
3. P&ID NORMAL CHILLED WATER SYSTEM RSB, SGB B&R DWG NM586
4. P&ID RCB HVAC B&R DWG NV501

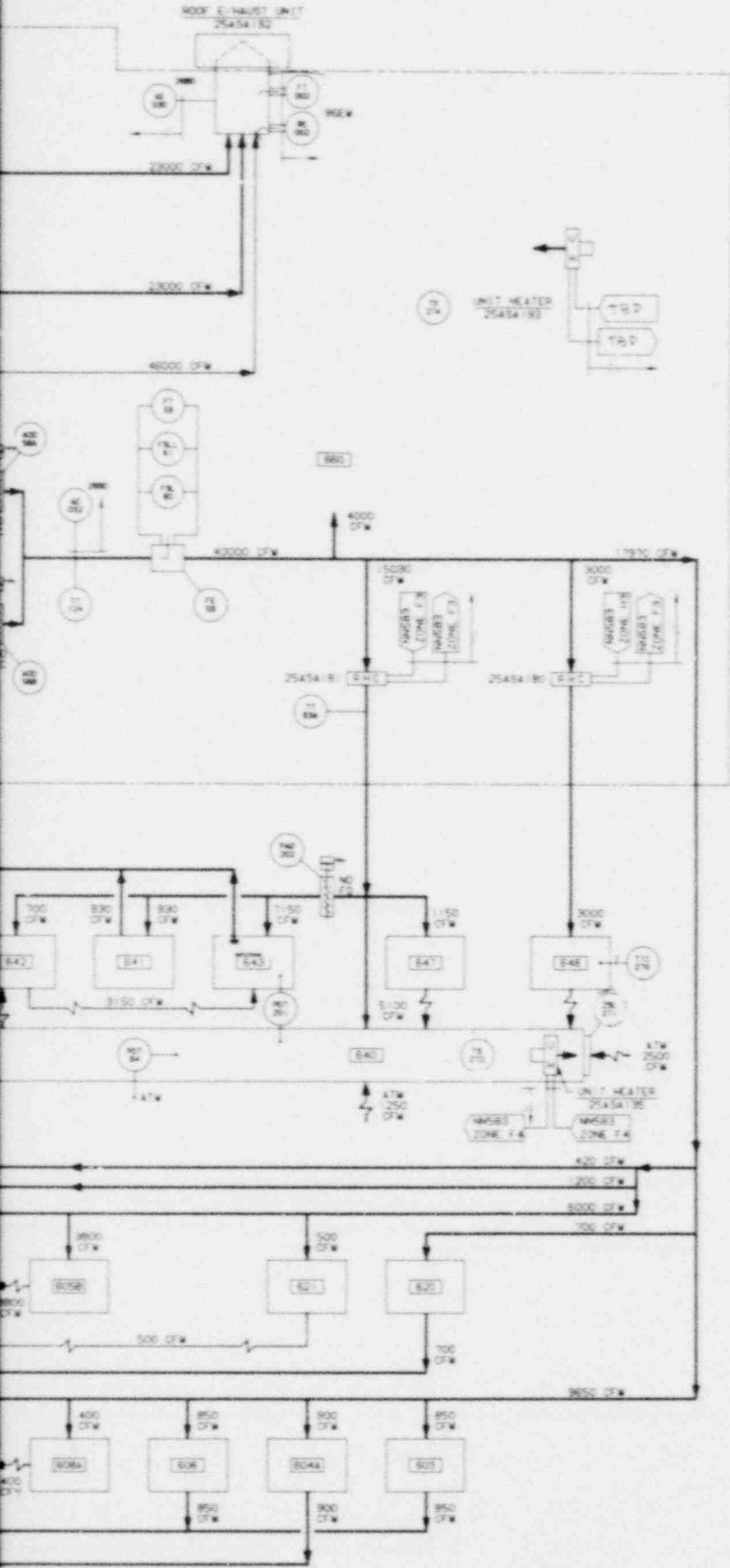
POOR ORIGINAL

FIGURE 9.6-9 P & I Diagram RCB/RSB RAPS & CAPS HVAC



NV520-5

POOR ORIGINAL



GENERAL NOTES

1. SYMBOLS AND ABBREVIATIONS WARD-O-DULL
2. ALL EQUIPMENT, INSTRUMENT AND CONTROL VALVE NUMBERS ARE PREFIXED BY 25A3 UDS
3. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY "K"
4. SYSTEM CLEANLINESS CLASSIFICATION:
5. SEISMIC CATEGORY: III
6. SAFETY CLASSIFICATION: NONE
7. ALL INSTRUMENT ITEMS SHOWN MARKED WITH AN ASTERISK * SHALL BE SUPPLIED BY THE ASSOCIATED EQUIPMENT MANUFACTURER

POOR ORIGINAL

REFERENCE DRAWINGS

1. P&ID NORMAL CHILLED WATER SYSTEM RSB, RCB CB & SGB — B&R DWG NM554
2. P&ID HOT WATER HEATING SYSTEM RSB & RSB-RWB — B&R DWG NM583
3. P&ID LALL COLLECTION & NEUTRALIZATION LIQUID RADIOACTIVE WASTE — B&R DWG NM539

FIGURE 9.6-10 P & I Diagram RSB-RW Area HVAC

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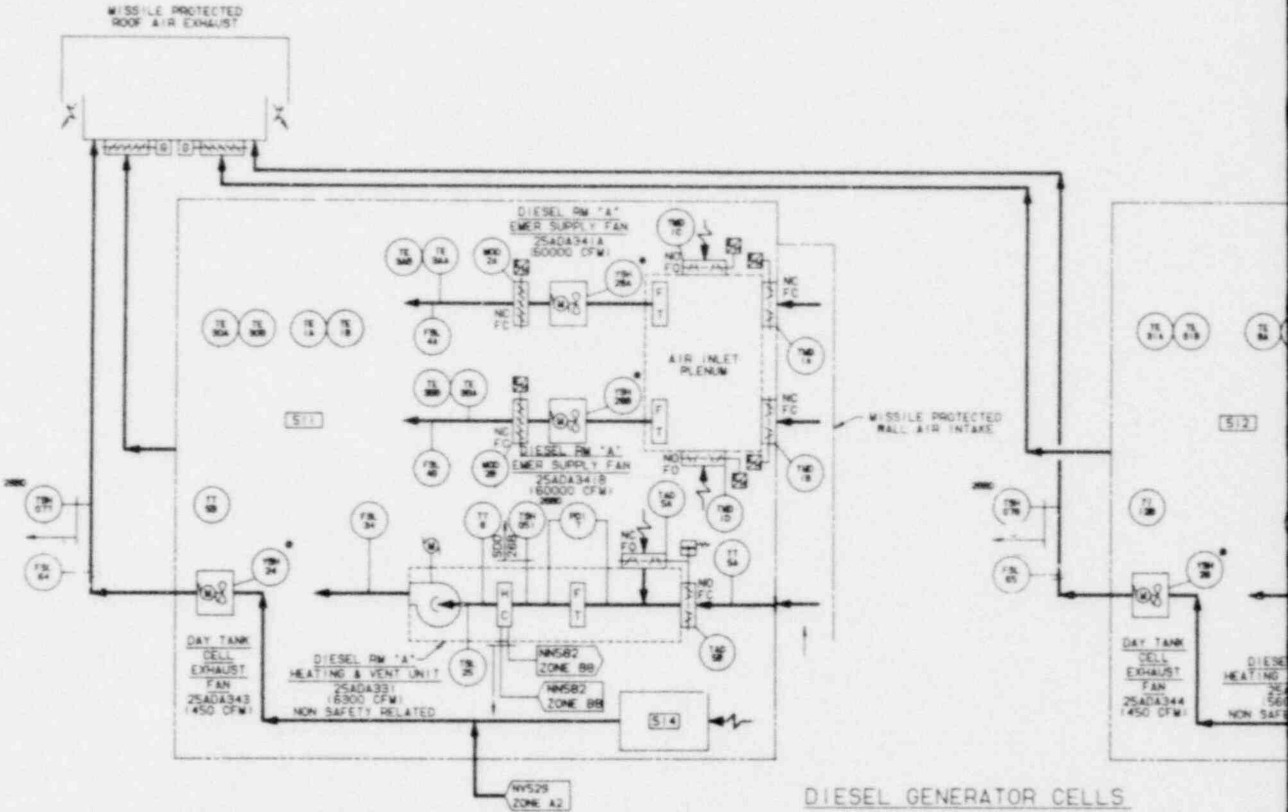
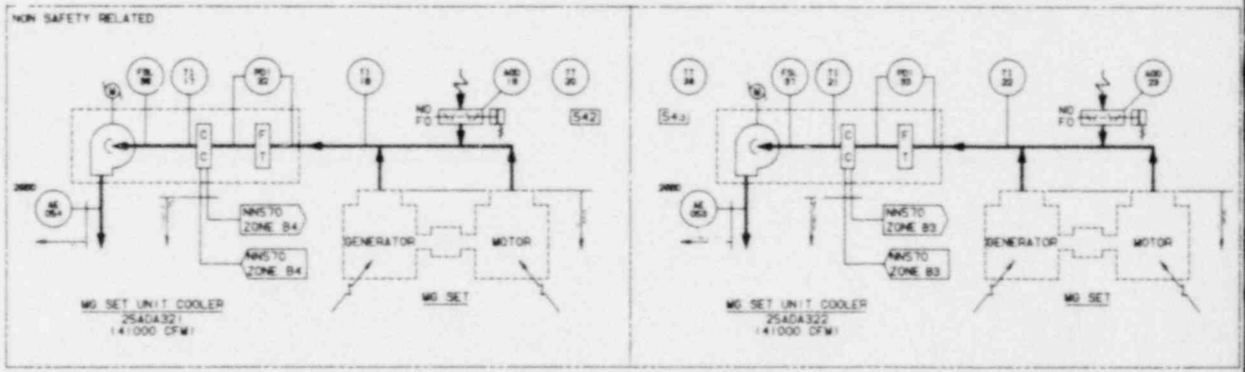


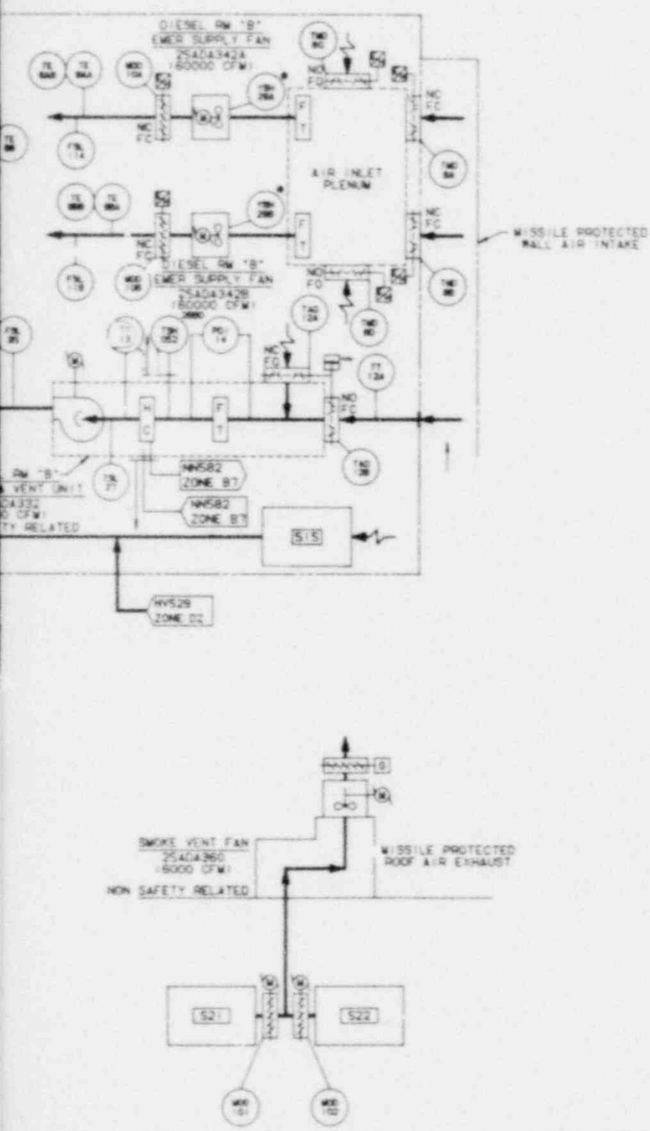
DIESEL GENERATOR CELLS



MG SET SUBSYSTEM

POOR ORIGINAL

NV539-6



GENERAL NOTES

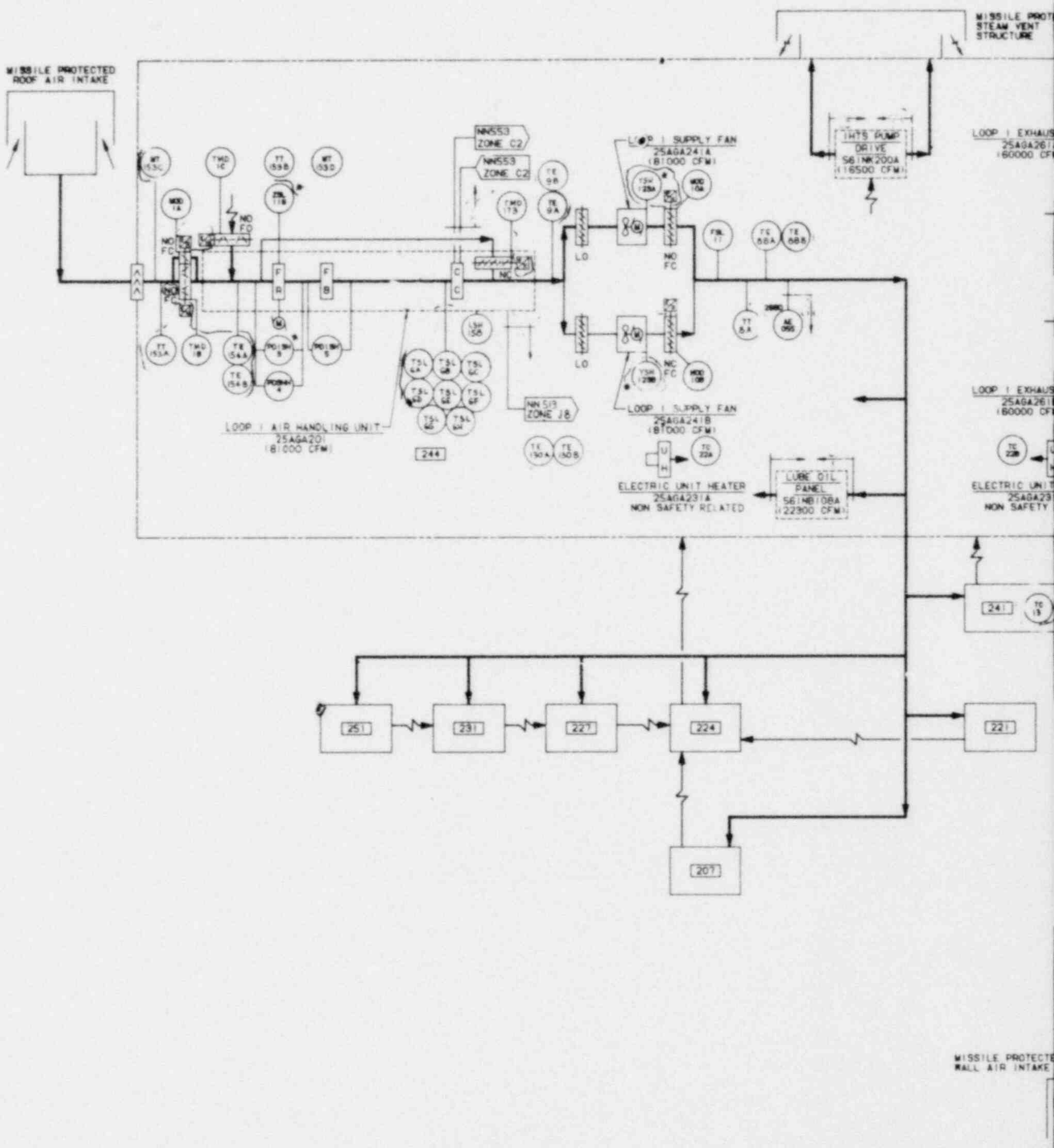
1. SYMBOLS AND ABBREVIATIONS WARD-D-0036
2. ALL EQUIPMENT, INSTRUMENT AND CONTROL VALVE NUMBERS ARE PREFIXED BY 25AD UOS
3. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY "X"
4. SYSTEM CLEANLINESS CLASSIFICATION:
5. SEISMIC CATEGORY:
6. SAFETY CLASSIFICATION: 3 UOS
7. SEE AIR FLOW BALANCE SHEETS FOR AIR QUANTITIES APPENDIX C-1
8. ALL INSTRUMENT ITEMS MARKED WITH AN ASTCRISK * SHALL BE FURNISHED BY THE ASSOCIATED EQUIPMENT MANUFACTURER

POOR ORIGINAL

REFERENCE DRAWINGS

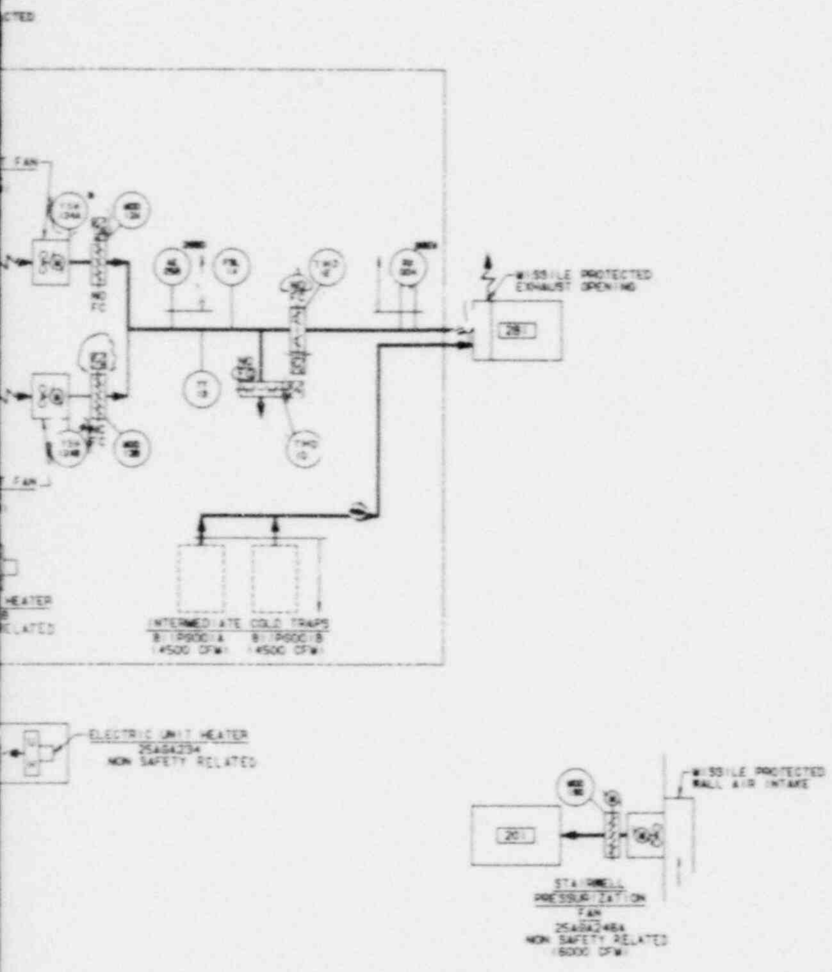
1. P&ID NORMAL PLANT SERVICE WATER B&R DWG NM570
2. P&ID HOT WATER HEATING SYSTEM SGB & DGB B&R DWG NM582

FIGURE 9.6-11 P & I Diagram DGB HVAC



NV545-3

POOR ORIGINAL



GENERAL NOTES

1. SYMBOLS AND ABBREVIATIONS WARD DOC D-0036
2. ALL EQUIPMENT, INSTRUMENT AND CONTROL VALVE NUMBERS ARE PREFIXED BY 25A0 UOS
3. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY "K"
4. SYSTEM CLEANLINESS CLASSIFICATION
5. SEISMIC CATEGORY: I
6. DELETED
7. SAFETY CLASSIFICATION: 3 UOS
8. SEE AIR FLOW BALANCE SHEETS FOR AIR QUANTITIES APPENDIX C-1
9. ALL INSTRUMENT ITEMS SHOWN MARKED WITH AN ASTERISK* SHALL BE SUPPLIED BY THE ASSOCIATED EQUIPMENT MANUFACTURER

REFERENCE DRAWINGS

1. P&ID NORMAL CHILLED WATER SYSTEM SGB B&R DRG NM553
2. P&ID EMER CHILLED WATER SYSTEM SGB, RCF B&R DRG NM561
3. FLOW DIAGRAM SGB INTERMEDIATE BAY FLOOR DRAINS

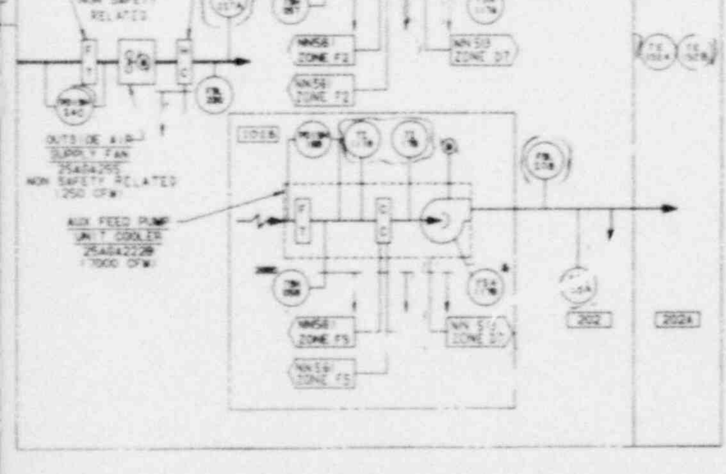


FIGURE 9.6-12 SGB Loop I & Aux. Bay HVAC

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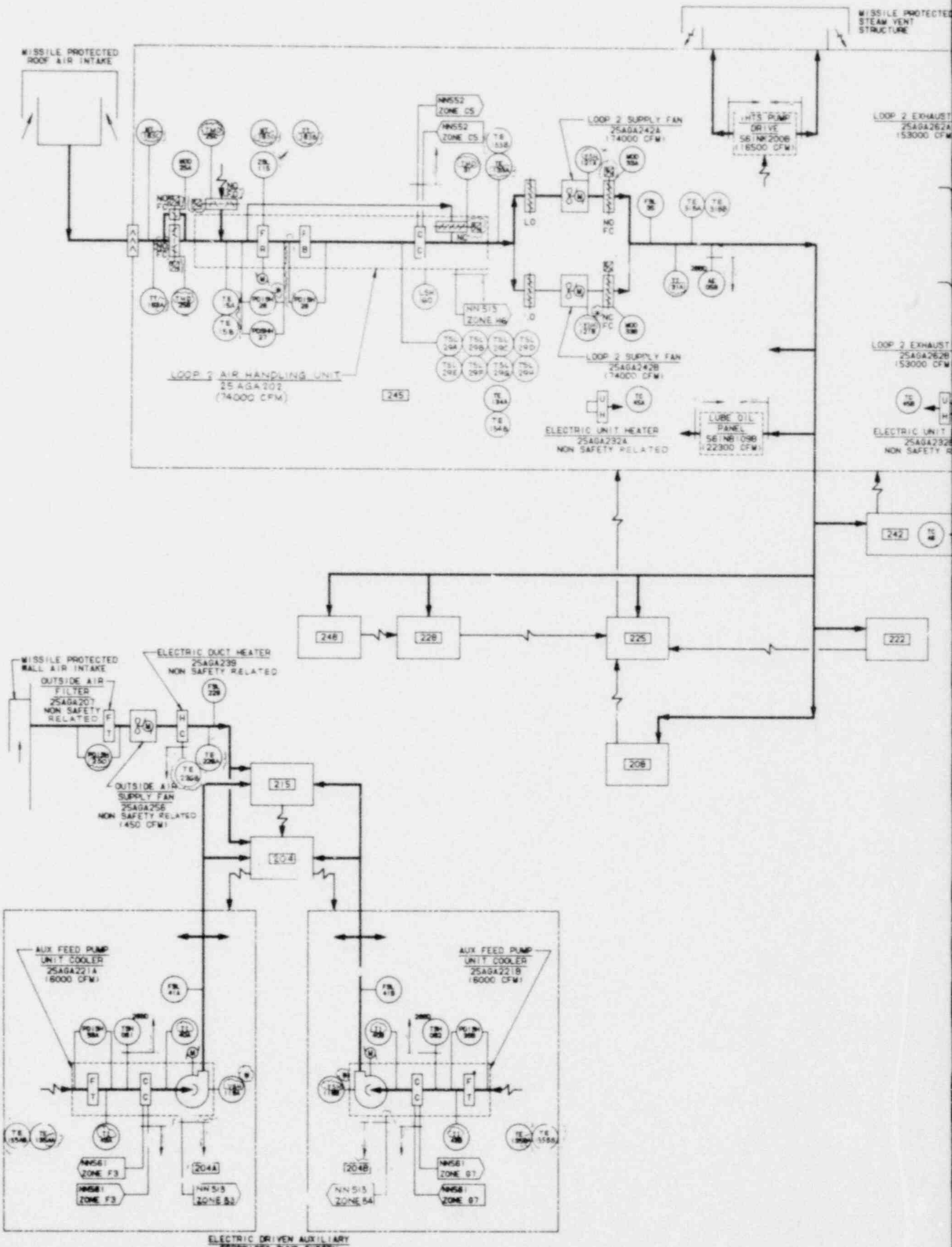
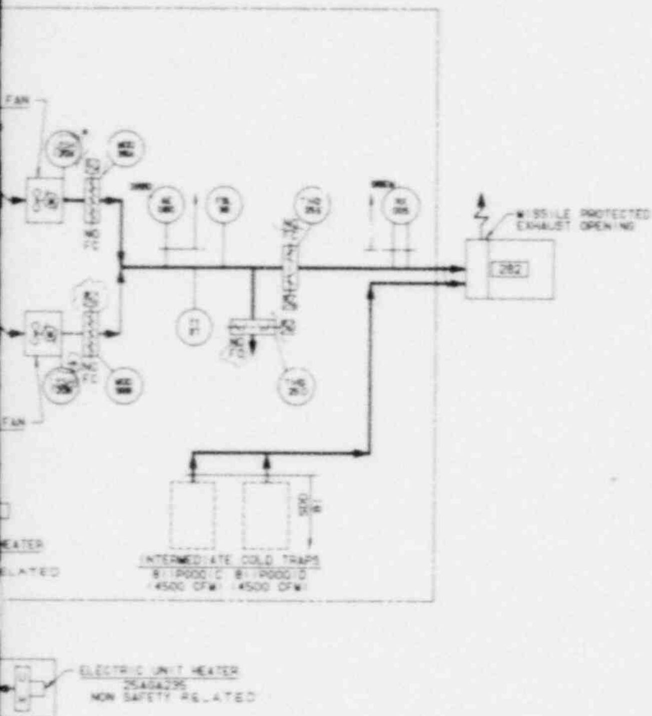


NV546-3

POOR ORIGINAL



GENERAL NOTES

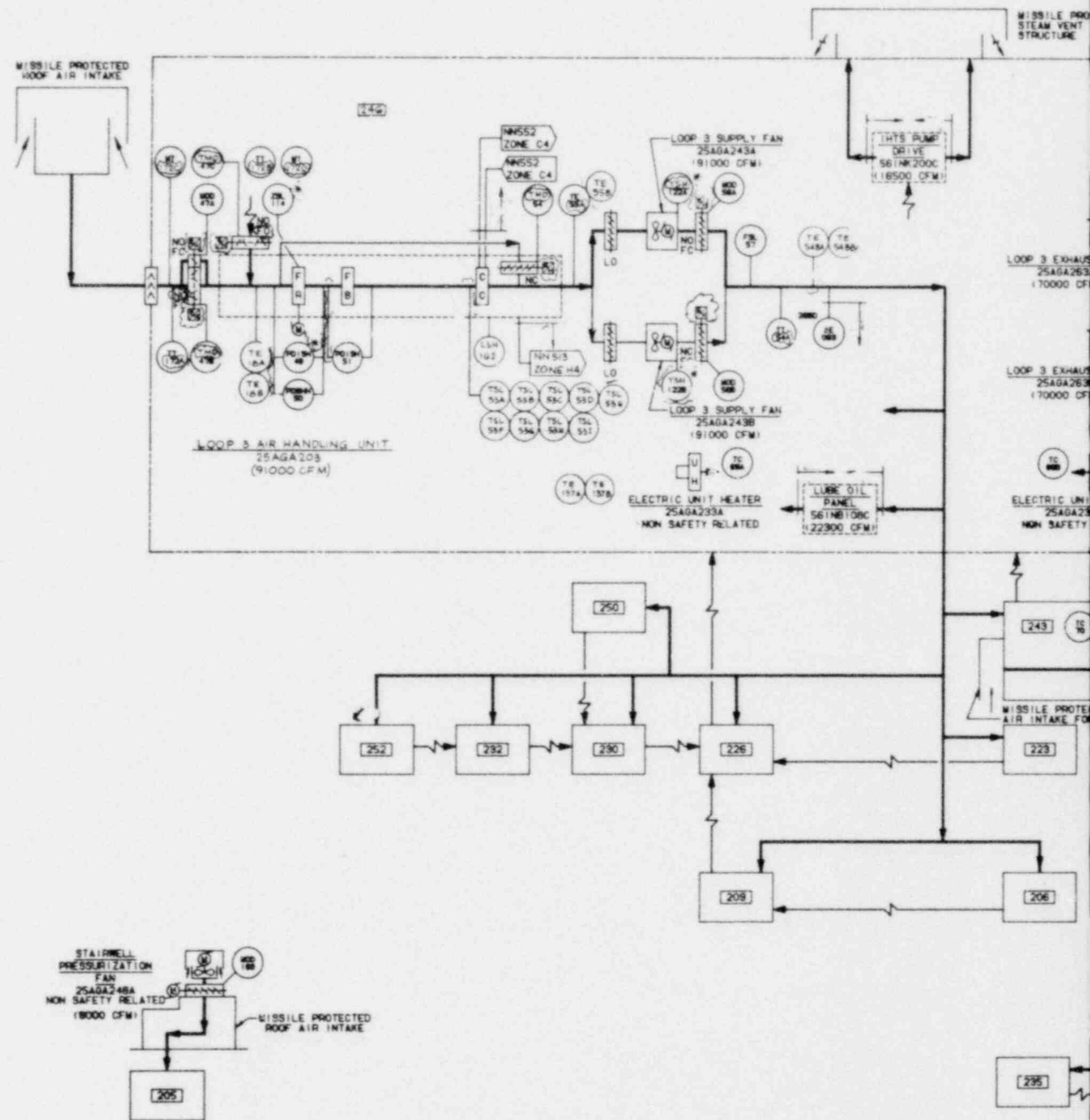
1. SYMBOLS AND ABBREVIATIONS WARD DOC D-0036
2. ALL EQUIPMENT, INSTRUMENT AND CONTROL VALVE NUMBERS ARE PREFIXED BY 25A0 UOS
3. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY "K"
4. SYSTEM CLEANLINESS CLASSIFICATION:
5. SEISMIC CATEGORY: I
6. (DELETED)
7. SAFETY CLASSIFICATION: 3 UOS
8. SEE AIR FLOW BALANCE SHEETS FOR AIR QUANTITIES APPENDIX C-1
9. ALL INSTRUMENT ITEMS SHOWN MARKED WITH AN ASTERISK * SHALL BE SUPPLIED BY THE ASSOCIATED EQUIPMENT MANUFACTURER

POOR ORIGINAL

REFERENCE DRAWINGS

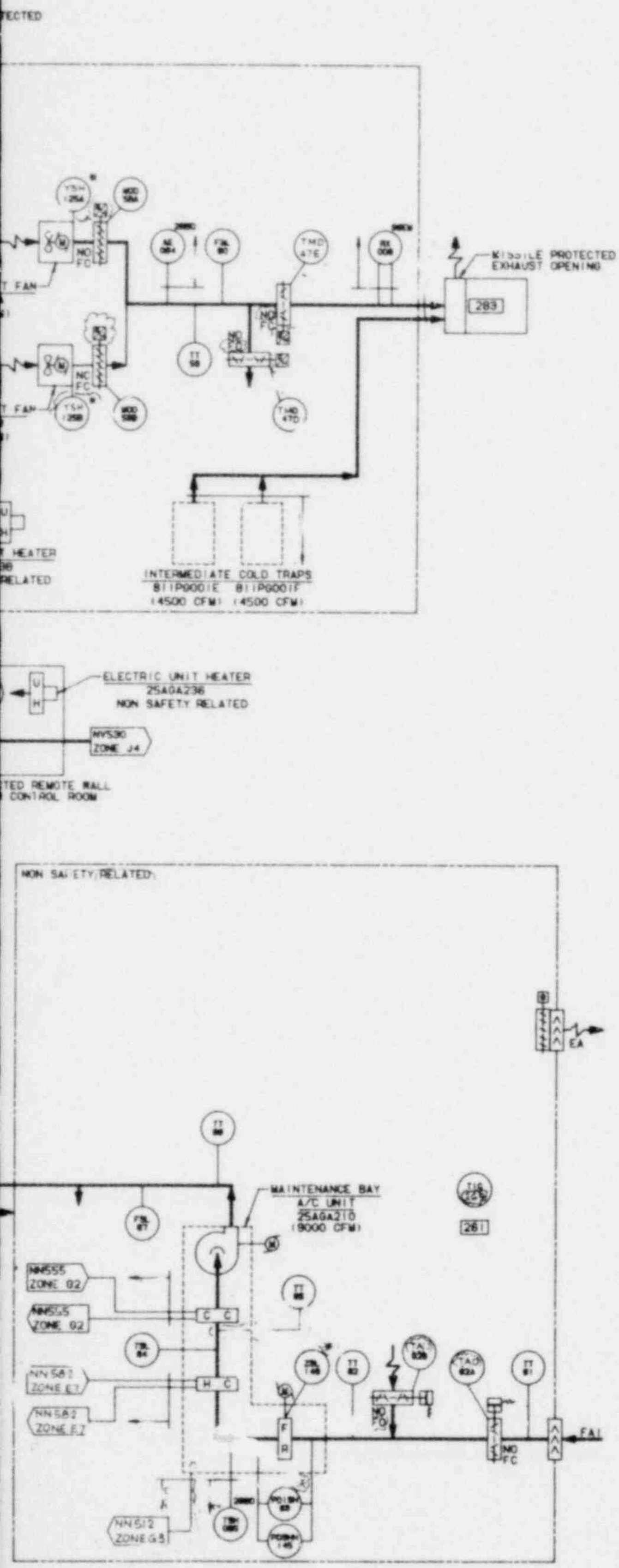
1. P&ID NORMAL CHILLED WATER SYSTEM RCB & SGB
B&R DWG NM552
2. P&ID EMERGENCY CHILLED WATER SYSTEM SGB & RCB
B&R DWG NM[illegible]
3. FLOW DIAGRAM SGB INTERMEDIATE BAY FLOOR DRAINS
B&R DWG NM518

FIGURE 9.6-13 SGB Loop 2 & Aux. Bay HVAC



POOR ORIGINAL

NV547-4



GENERAL NOTES

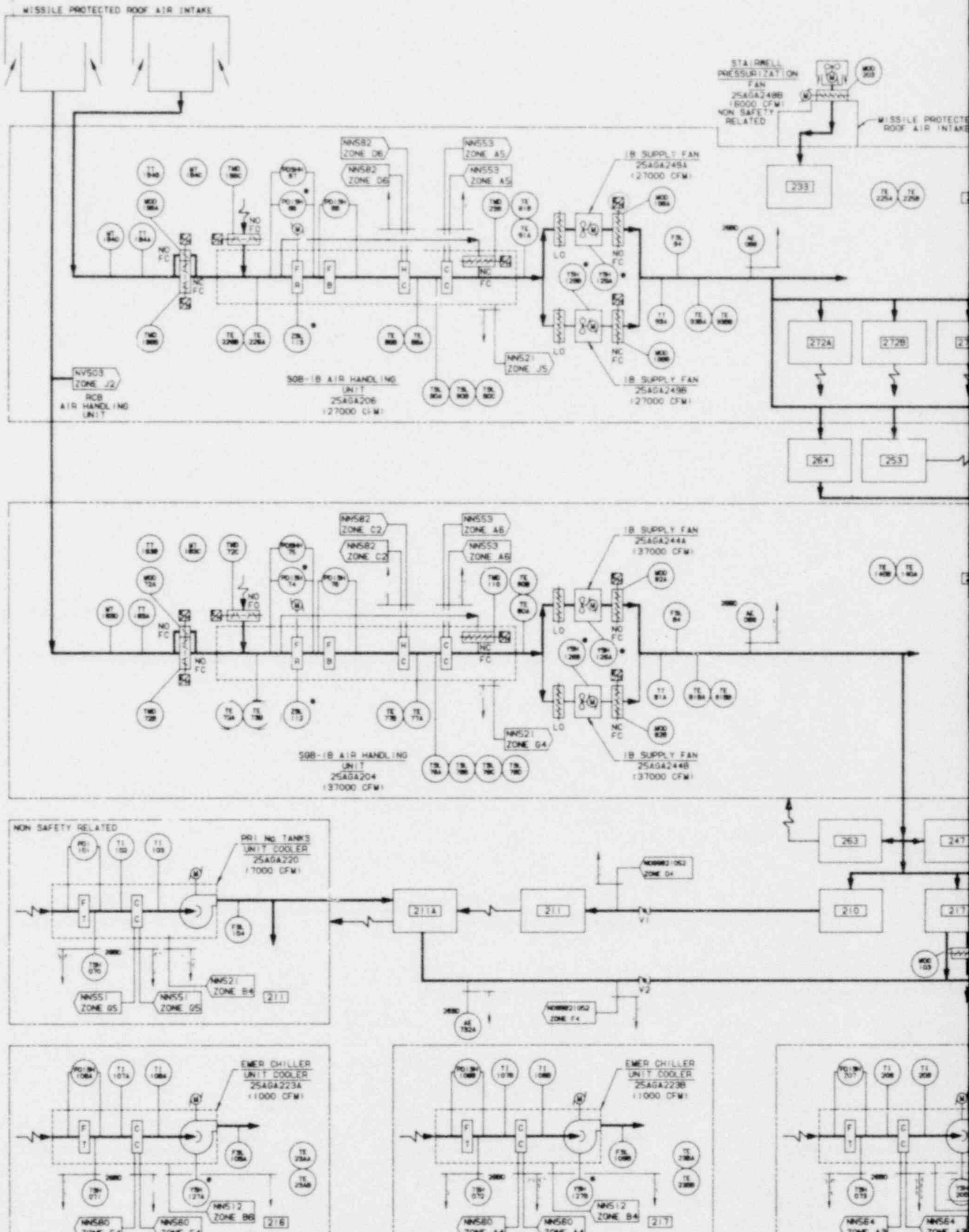
1. SYMBOLS AND ABBREVIATIONS HARD DOC D-0036
2. ALL EQUIPMENT, INSTRUMENT AND CONTROL VALVE NUMBERS ARE PREFIXED BY 25AG UOS
3. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY "K"
4. SYSTEM CLEANLINESS CLASSIFICATION:
5. SEISMIC CATEGORY: I
6. (DELETED)
7. SAFETY CLASSIFICATION: 3 UOS
8. SEE AIR FLOW BALANCE SHEETS FOR AIR QUANTITIES APPENDIX C-1
9. ALL INSTRUMENT ITEMS SHOWN MARKED WITH AN ASTERISK * SHALL BE SUPPLIED BY THE ASSOCIATED EQUIPMENT MANUFACTURER

POOR ORIGINAL

REFERENCE DRAWINGS

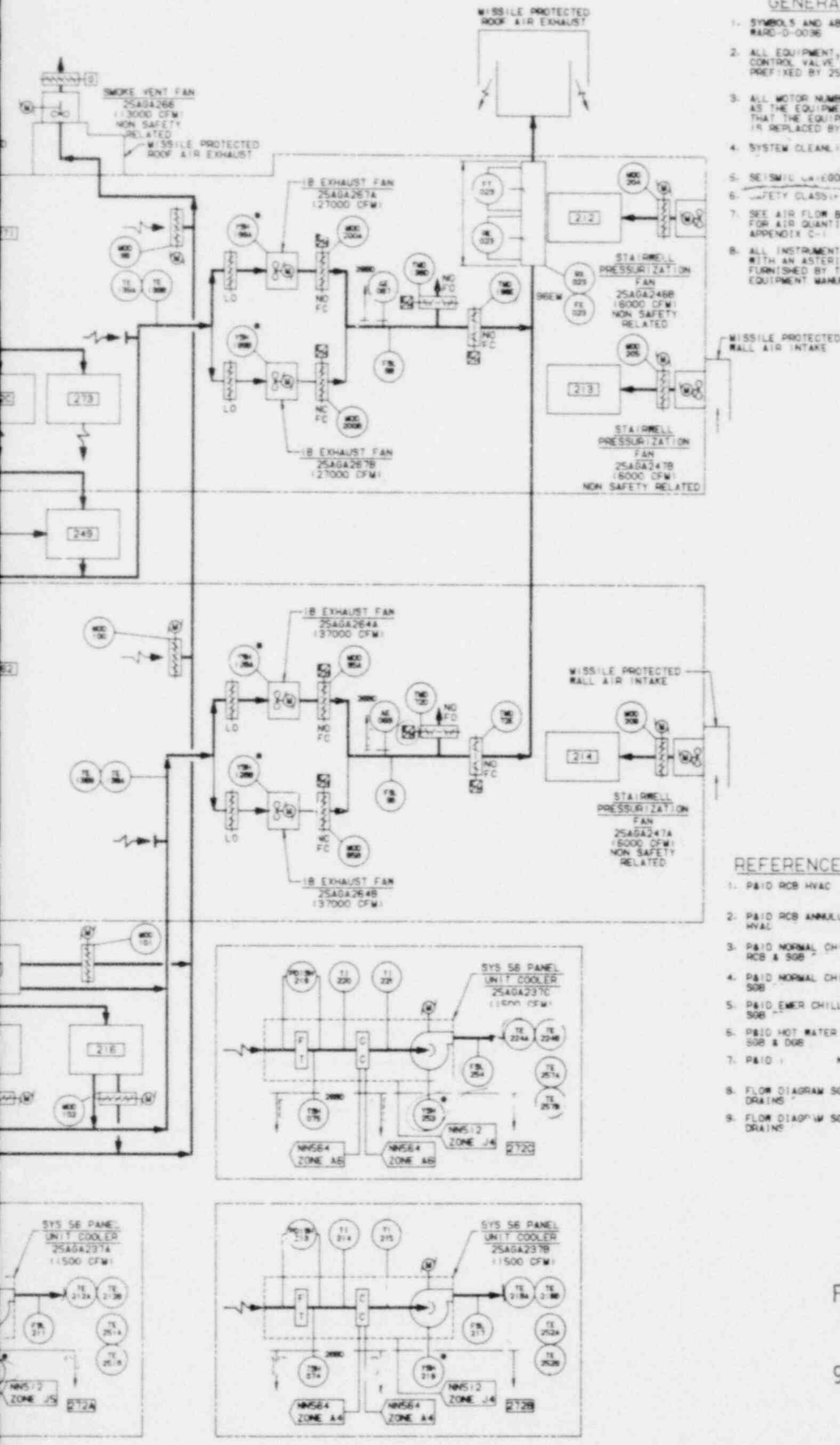
1. P&ID NORMAL CHILLED WATER SYSTEM RCB & SGB
   B&R DWG NN552
2. P&ID NORMAL CHILLED WATER SYSTEM SGB, MSW & TGB
   B&R DWG NN555
3. FLOW DIAGRAM HIGH TEMP HOT WATER HEATING
   B&R DWG NN582
4. FLOW DIAGRAM SGB INTERMEDIATE BAY FLOOR DRAINS
   B&R DWG NN513
5. FLOW DIAGRAM SGB FLOOR AND EQPT. DRAINS
   B&R DWG NN512

FIGURE 9.6-14 SGB Loop 3, Aux. Bay & MB HVAC



NV549-5

POOR ORIGINAL



GENERAL NOTES

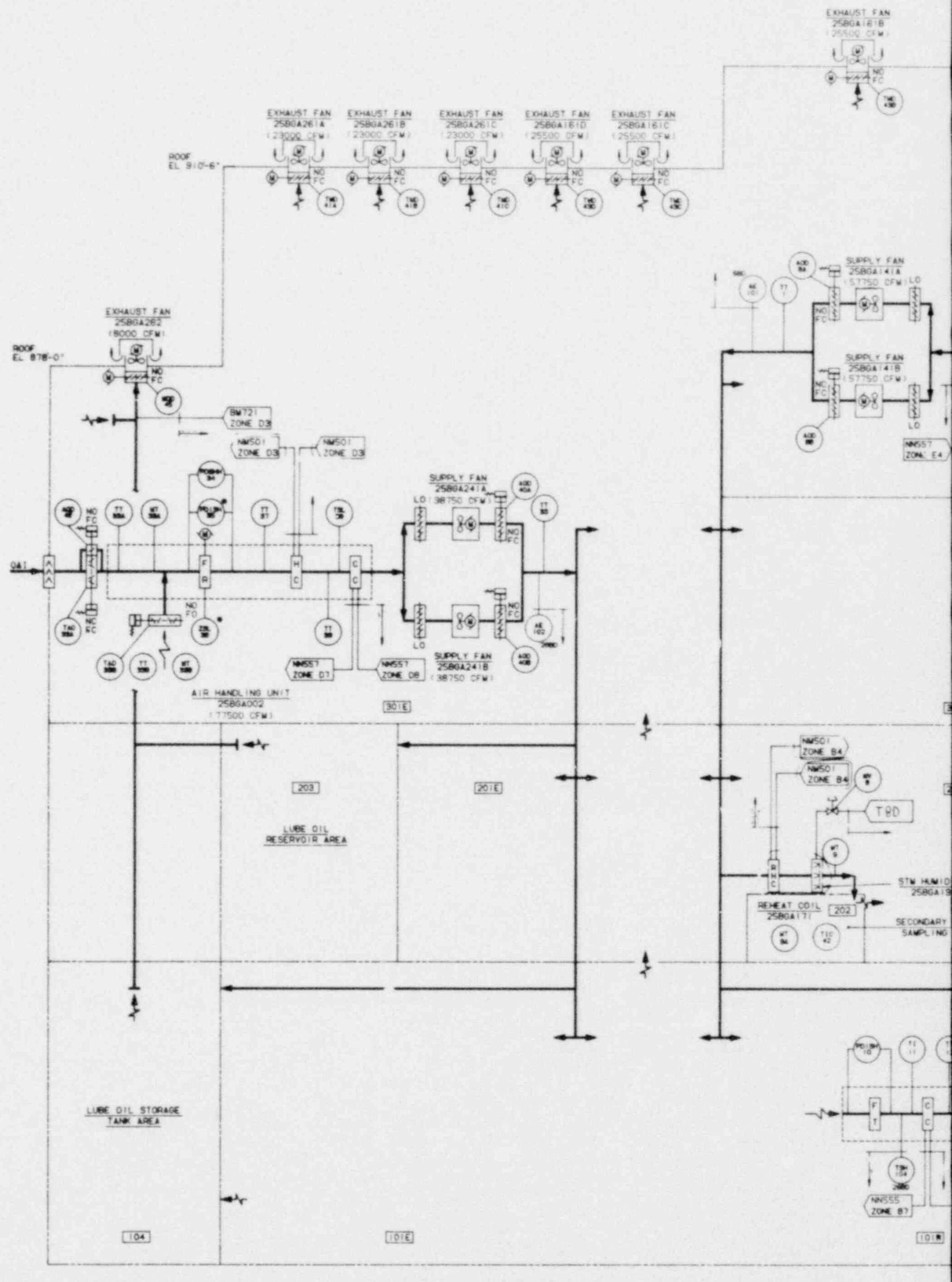
1. SYMBOLS AND ABBREVIATIONS WARD-D-0036
2. ALL EQUIPMENT, INSTRUMENT AND CONTROL VALVE NUMBERS ARE PREFIXED BY 25AG UOS
3. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY "K"
4. SYSTEM CLEANLINESS CLASSIFICATION
5. SEISMIC CATEGORY
6. SAFETY CLASSIFICATION: 3 UOS
7. SEE AIR FLOW BALANCE SHEETS FOR AIR QUANTITIES APPENDIX C-1
8. ALL INSTRUMENT ITEMS SHOWN MARKED WITH AN ASTERISK * SHALL BE FURNISHED BY THE ASSOCIATED EQUIPMENT MANUFACTURER

POOR ORIGINAL

REFERENCE DRAWINGS

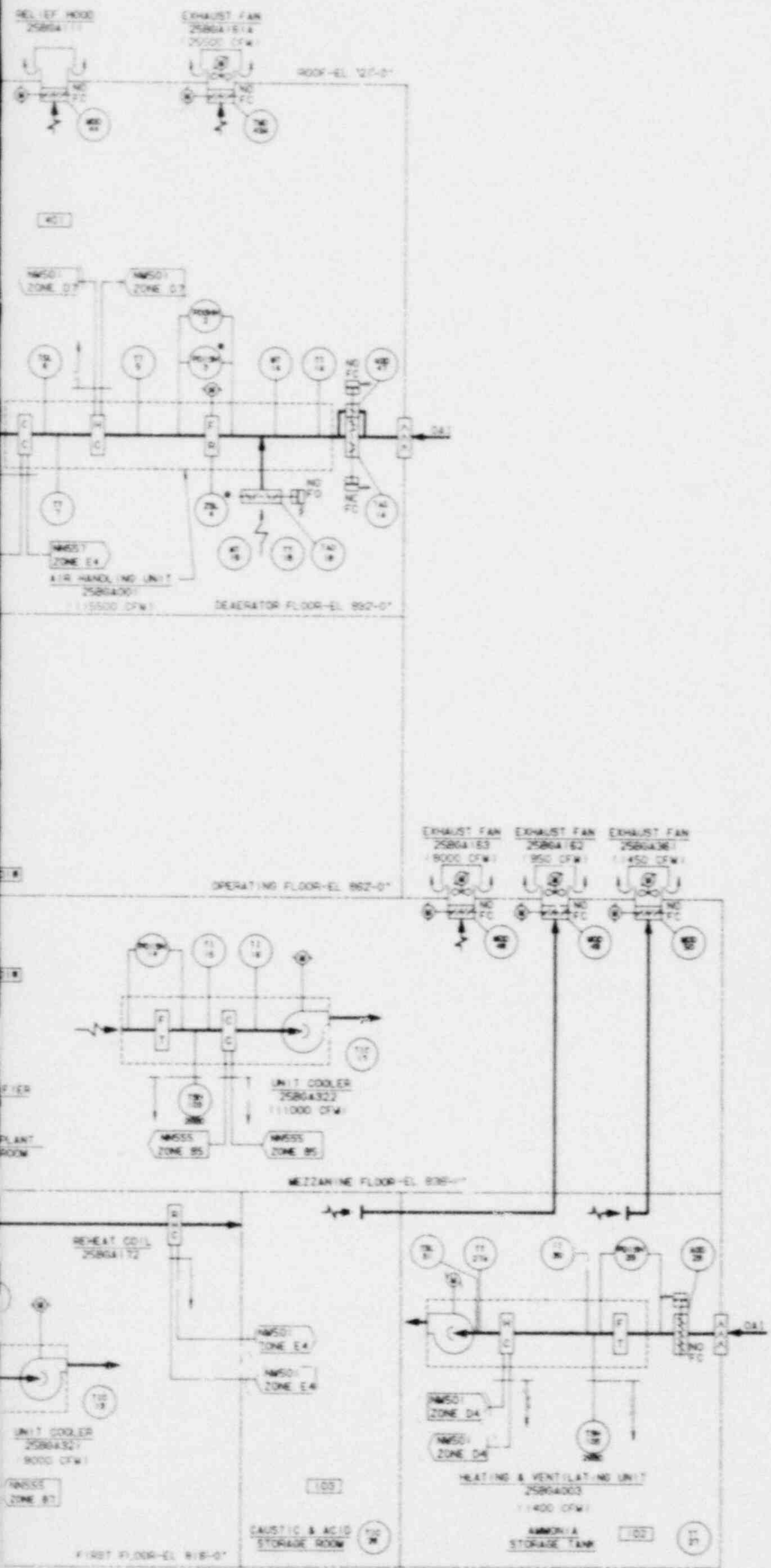
1. P&ID RCB HVAC B&R DWG NY501
2. P&ID RCB ANNULUS & CONT CLEAN-UP HVAC B&R DWG NY502
3. P&ID NORMAL CHILLED WATER SYSTEM RCB & SGB B&R DWG NN551
4. P&ID NORMAL CHILLED WATER SYSTEM SGB B&R DWG NN553
5. P&ID EMER CHILLED WATER SYSTEM SGB B&R DWG NN560
6. P&ID HOT WATER HEATING SYSTEM SGB & DGB B&R DWG NN582
7. P&ID N2 SUPPLY AI DWG N09982I052
8. FLOW DIAGRAM SGB FLOOR & EQUIPMENT DRAINS B&R DWG NN512
9. FLOW DIAGRAM SGB FLOOR & EQUIPMENT DRAINS B&R DWG NN521

FIGURE 9.6-15 SGB-IB HVAC



BV511-7

POOR ORIGINAL



GENERAL NOTES

1. SYMBOLS AND ABBREVIATIONS WARD-D-0036
2. ALL EQUIPMENT, INSTRUMENT AND CONTROL VALVE NUMBERS ARE PREFIXED BY 2580 UOS
3. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY "X"
4. SYSTEM CLEANLINESS CLASSIFICATION: "C" COILS/"D" DUCTWORK AND FANS
5. SEISMIC CATEGORY: III
6. SAFETY CLASSIFICATION: NONE
7. SEE AIR FLOW BALANCE SHEETS FOR AIR QUANTITIES SDD 258. APPENDIX C-1
8. RADIATION MONITORS ARE TBD
9. ALL INSTRUMENT ITEMS SHOWN MARKED WITH AN ASTERISK * SHALL BE FURNISHED BY THE ASSOCIATED EQUIPMENT MANUFACTURER.

POOR ORIGINAL

REFERENCE DRAWINGS

1. P&ID NORMAL CHILLED WATER SYSTEM B & R DWG NW555
2. P&ID NORMAL CHILLED WATER SYSTEM B & R DWG NW557
3. P&ID HOT WATER HTG SYSTEM B & R DWG NW501
4. P&ID CONDENSER AIR EXTRACTION B&R DWG BW52

FIGURE 9.6-16 Turbine Generator Building HVAC

9.7 CHILLED WATER SYSTEMS

9.7.1 Normal Chilled Water System

9.7.1.1 Design Basis

The Normal Chilled Water System is designed to provide chilled water for all nonsafety related air conditioning and recirculating gas cooling system cooling coils and primary cold trap heat exchangers located 44
59 throughout the plant. Chilled water satisfies area temperature requirements and removes heat from confined areas and equipment during normal plant operation.

The Normal Chilled Water System supplying chilled water to the nuclear safety-related air conditioning and recirculating gas cooling unit coolers and cooler units during normal plant operation is isolated 44
by two sets of ASME Section III, Class 3 isolation valves at the Emergency Chilled Water System headers (Section 9.7.2).

The Normal Chilled Water System is not required to cool any safety related equipment during accident conditions. It is a non-safety 44
class system. The system is designed to ASME Section VIII/ANSI B31.1.0 requirements outside of the Reactor Containment Building (RCB). All piping and piping components in the RCB are ASME Section III, Class 3. 44

9.7.1.2 System Description

59 The Normal Chilled Water System which is shown in Figure 9.7-1 thru 9.7-9 has six 20 percent capacity electric motor driven mechanical refrigera-
47 tion water chillers, six 20 percent capacity chilled water circulation pumps. an air separator, an expansion tank, and associated piping, valves and instrumentation. 44
The Normal Chilled Water System has a chilled water operating temperature of less than 60°F and an operating pressure of less 44
than 150 psig. The Normal Plant Service Water System provides cooling water for the chiller condensers as described in Section 9.9.1. Piping distributes chilled water to the chilled water coils located in unit coolers, coolers and the air handling and air conditioning units. 44
These water coils are located in the major plant areas, such as the Turbine Generator Building, Control Room Building, Steam Generator Building, Reactor Service Building, Reactor Containment Building, Plant Service Building, and Shop and Warehouse Building. Return piping directs the water back to 44
the water chilling equipment located in the Steam Generator Building - Intermediate Bay. In addition, the Normal Chilled Water System has one intermediate Dowtherm J loop to provide cooling to the Primary Cold Trap NaK cooler (located in the RCB). The Dowtherm J is cooled via a heat exchanger by Normal Chilled Water (See Section 9.7.4 for secondary coolant loop discussion). 44

Generally, the Normal Chilled Water System is Non-Seismic Category and is non-safety related. However, all piping, piping components, and chilled water coils located in the Reactor Containment Building (RCB) are Seismic Category I and are built to the requirements of ASME Boiler and Pressure Vessel Code, Section III, Division 1. Subsection ND for Class 3 Components, "Rules for Construction of Nuclear Power Plant Components". Containment isolation valves are built to the requirements of Section III, Class 2 of the same code. |44

All Normal Chilled Water piping and piping components located outside of the RCB are built to the requirements of ANSI B31.1, "Power
59 Piping", whereas heat exchangers and pressure vessels outside the RCB are 44
built to the requirements of ASME Boiler and Pressure Vessel Code, Section VIII.

The Normal Chilled Water System is terminated by two sets of ASME, Section III, Class 3 isolation valves, where cross-connections are
made to the Emergency Chilled Water System. Upon loss of Normal Chilled 44
Water Supply to the Emergency Chilled Water System headers, the isolation valves are closed automatically, and the Emergency Chilled Water System starts. Where the Normal Chilled Water System penetrates the RCB, one remote manually-actuated ASME Section III, Class 2, isolation valve is provided on each line. The piping on the RCB side of this valve, up
to the next manual isolation valve, is ASME Section III, Class 2. 44

The components served by the Normal Chilled Water System are listed in Table 9.7-1. The major system component design data are listed in Table 9.7-2.

9.7.1.3 Safety Evaluation

One 20 percent capacity standby chiller unit is provided 44
to ensure continuous cooling capability in case of a malfunction of a
47 chiller unit. One 20 percent capacity standby chilled water circula- 44
tion pump is also provided for the same purpose. The diversity of the
cooling loads provides additional refrigeration margin for the system. 44

In addition to these considerations, Section 9.7.3 lists system design features intended to prevent water/sodium interactions.

9.7.1.4 Tests and Inspections

The system is tested and inspected as separate components at the manufacturer's facilities and as an integrated system prior to plant operation. All water flow rates are balanced and set to the design flow
conditions. Periodic inspection of the equipment is scheduled to ensure 44
proper operation of the system.

All chilled water lines penetrating the containment shall be
59 provided with vents and drains to permit drainage. Normal 44
chilled water supply and return headers immediately upstream and downstream of the containment isolation valves shall be drainable.

44

Vents and drains will be opened to permit drainage and to per-
59 mit transmission of containment test pressure to the closed isolation valves.

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9.7.1.5 Instrumentation Application

Chilled water system control panels are located in the area of the water chillers. These panels include control switches, monitors, and system alarms. Local alarms are provided for the following conditions:

a. Expansion tank high water level

b. Expansion tank low water level

c. Leak detection and isolation (described in Section 9.7.3)

d. High chilled water discharge temperature

e. Water chiller trip alarm (includes following chiller malfunctions):

1. low chilled water temperature

2. high condensing pressure

3. low refrigerant temperature or pressure

4. low chilled water flow

5. low condenser water flow

6. low oil pressure

7. high shaft vibration

8. high bearing temperature

9. high motor temperature

15 A common system annunciator for "a" through "e" above is provi-
59 ded in the control room to indicate trouble in the Normal Chilled Water
System. In addition, an annunciator alarm is provided for condition
"e" in the control room, with first out indication locally for condi-
59 tions "e.1" through "e.9" above. 44

9.7.2 Emergency Chilled Water System

9.7.2.1 Design Basis

The function of the Emergency Chilled Water System is to provide
59 chilled water for systems listed in Table 9.7-3. The Emergency
Chilled Water System has a chilled water operating temperature of less
15 than 60°F and an operating pressure of less than 150 psig. The system 44
is designed to meet the following design criteria:

a. The system equipment, piping and valves meet the ASME Code, Section III/Class 3 requirements, except the compressor casing material of the chillers, which shall be contractor's standard with the quality assurance requirements of ASME Section III Class 3. In addition, a volumetric inspection of radiography shall be made to verify structural integrity of the component.

The equipment, piping, and valves of the Fuel Handling Cell Dowtherm J coolant loop shall be designed and fabricated in accordance with requirements of ANSI B31.1, ASME Section VIII and Seismic Category 1, as applicable. This is not a safety related sub-system. 44

59 b. A redundancy of 100% is provided in all active and passive system components, as per the intent of the single failure criteria set forth by Regulatory Guide 1.53.

c. The components, piping, valves, instrumentation, and controls of each redundant loop are installed with proper spatial separation or protected by a barrier to conform to common mode failure criteria.

d. The system is connected to the on-site Class IE AC Electrical power supply.

e. All the system components are housed in tornado hardened, external flood protected, Seismic Category I structures and are internal missile protected.

f. Each Emergency Chilled Water Loop provides chilled water at temperatures and flow rates required by the systems it serves, as shown on Table 9.7-3.

g. Means for in-service inspection of the equipment piping and valves are provided in accordance with the requirements of ASME Code, Section XI, Division 3. 44

h. Instrumentation and controls are tested periodically according to Regulatory Guide 1.22.

i. Minimum service life desired of all major components of the system is 30 years. Wherever 30 year life expectancy is not reasonably assured, the equipment is designed with redundancy and it is installed to permit easy replacement with minimum effect on plant availability.

j. The system equipment is in standby during normal plant operation. 44

k. Accumulators, provided to actuate air operated valves, will be sized with sufficient margin for the valves to perform their safety related function for the required duration. 59

9.7.2.2 System Description

The Emergency Chilled Water System is shown in Figure 9.7-10 thru 44
59 9.7-15. Emergency Chilled Water is supplied by two independent loops, each loop capable of meeting the total chilled water demand during an accident. 44
Each loop contains one electric motor-driven, mechanical refrigeration water chiller, circulation pump, air separator, expansion tank, piping, 44
valves, and instrumentation.

Two separate supply loops provide chilled water from the chillers to the nuclear safety-related air conditioning and recirculating system cooling coils in the plant. The Emergency Chilled Water is returned to 44
the chillers through their respective circulating pumps by independent return loops. Each emergency loop has an intermediate Dowtherm J loop to provide cooling to the FHC Recirculating Gas Cooler Units. Each Dowtherm J loop is cooled via a heat exchanger by Emergency Chilled Water (see Section 9.7.4 for secondary coolant loop discussion). Under normal 44
operating conditions, the above equipment is supplied with chilled water from the Normal Chilled Water System as described in Section 9.7.1.

The components served by the Emergency Chilled Water System are 44
listed in Table 9.7-3. The major system components design data are listed in
59 Table 9.7-4. Table 9.7-7 lists active safety related air operated valves.

9.7.2.3 Safety Evaluation

The Emergency Chilled Water System provides a reliable source of chilled water for the chilled water coils of the air handlers, cooler units and unit coolers located in the Control Building, Reactor Service 44
Building, Reactor Containment Building and Steam Generator Building. The system is designed to operate during accident conditions without loss of function. The Emergency Chilled Water System is designed to Seismic Category I, ASME Section III, Class 3 requirements and is connected to the 44
Class IE AC power supply. All equipment is located in the Steam Generator Building - Intermediate Bay, which is a tornado hardened, flood protected, Seismic Category I structure. 44

Each loop consists of a 100 percent capacity water chiller, a chilled water circulating pump, an air separator, an expansion tank, piping, valves and instrumentation. Instrumentation and control devices associated with the operation of the water chillers are located within the local control panel and will activate a local and Control Room alarm upon system malfunction. The water chiller starting circuit is interlocked with the chiller's condenser cooling water and chilled water flow switches to prevent starting of the chillers without a sufficient supply of condenser cooling water or chilled water ciruclation. Failure of any component in the operating chilled water
59 loop activates an alarm in the control room. A single failure analysis of the Emergency Chilled Water System is given in Table 9.7-5.

If during normal operation normal chilled water supply is interrupted, 44
33 flow switches in the emergency chilled water supply header will close the ASME III Class 3 isolation valves between the two systems and automatically start the Emergency Plant Service Water System and then the Emergency
33 Chilled Water System.

In addition to these considerations, Section 9.7.3 lists system
59 design features which are provided to prevent a water/sodium reaction.

9.7.2.4 Tests and Inspections

After testing each individual component of the system, the entire system is tested prior to plant operation. Instruments and controls are provided for periodically testing the performance of the system during normal plant operation or scheduled shutdown. All water flow rates are balanced and set to design flow conditions. Periodic inspections of equipment are scheduled to ensure the proper operation of the system. 44
In-service inspections will be conducted according to ASME Code Section
15 XI, as described in Section 9.7.2.1.g.

All chilled water lines penetrating containment shall be pro-
59 vided with vents and drains to permit drainage. Emergency chilled water supply and return lines immediately upstream and downstream of the containment isolation valves shall be drainable. Vents and drains will be opened to permit drainage and to permit communication of containment test pressure to the closed isolation valves. 44

9.7.2.5 Instrumentation Application

Chilled water system control panels are located in the area of the water chillers. The panels include the control switches, monitors, and system alarms. Local alarms are provided for the following conditions:

a. High chilled water temperature

b. Low chilled water flow

c. Normal chilled to emergency chilled changeover valve malfunction

d. Expansion tank high water level

e. Expansion tank low water level

f. Leak detection and isolation (described in Section 9.7.3)

g. Water chiller trip alarm (includes the following chiller malfunctions): 44

1. low chilled water temperature
2. high condensing pressure
3. low refrigerant temperature or pressure
4. low chilled water flow
5. low condenser water flow
6. low oil pressure
7. high shaft vibration
8. high bearing temperature
9. high motor temperature

Individual annunciator alarms are provided for conditions "a"
59 through "c" above in the control room main control board and locally for
both loops A and B. A common system annunciator alarm for conditions "d"
through "g" is provided on the control room main control board and
locally for both loops A and B. In addition, an individual annunciator
alarm is provided for condition "g" or back panel with first out in- 44
59 dication locally for conditions "g.1" through "g.9" listed above.

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9.7.3 Prevention of Sodium or NaK/Water Interactions 44

The design of both the Normal and Emergency Chilled Water Systems incorporates the required features to maintain three barriers (three passive or two passive and one active) between water and sodium or NaK. 44

To prevent water contact with sodium or NaK equipment through leaks in the piping, the following measures have been incorporated into the system design: 44

1) Chilled water piping is not routed through areas and cells containing radioactive sodium or NaK. The 44
three barriers between the chilled water and radioactive sodium or NaK are (1) the chilled water pipe 44
wall; (2) the cell boundary; and (3) the sodium or NaK pipe wall.

2) Where chilled water is routed through areas containing nonradioactive sodium or NaK, the barriers between the chilled water and sodium or NaK are: (1) 44
the chilled water pipe wall; (2) guard piping or drain pans and spray shields; and (3) the sodium or NaK 44
pipe walls.

3) All areas and cells containing chilled water piping are provided with a floor drain system, which is designed to accommodate the maximum water flow rate resulting from failure of the largest chilled water pipe in the area.

59 4) All floor drain systems for the chilled water system, 44
which are located in areas where sodium or NaK is present or in areas adjacent to cells containing sodium or NaK, 44
are provided with water leak detectors. These leak
59 detectors will detect and identify the 44
approximate location of any chilled water leak.

5) The chilled water system piping is provided with strategically located isolation valves. Upon receiving a leak detection signal, the isolation 44
valves serving the area where the leak occurs will be closed to isolate the leaking pipe. 44

6) Provisions are incorporated to prevent the seepage of water through floors or walls during any leak, and the proper pitching of floors toward the floor drains ensures the flow of water to the drainage system.

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45 7) Where liquid metal is the fluid to be cooled, a secondary cooling loop of Dowtherm J is provided to separate the sodium or NaK and the chilled water. The working fluid in the secondary loop is not reactive with either water or liquid metal. The pressure in the secondary loop is higher than in both the water loop and the liquid metal containing loop to prevent contact between water and liquid metal during a heat exchanger tube leak. P&I Diagrams are shown in
59 Figures 9.7-7 and 9.7-13. 44

8) Units above the RCB operating floor are serviced with chilled water by piping equipped with guard cases. The guard cases run from the containment penetrations, which are close to the unit served and terminate at 44
the unit.

To prevent water contact with sodium or NaK equipment by the 44
transport of moisture or water through ducting, the following measures have been incorporated into the system design:

1) To prevent moisture or water carryover from the HVAC system cooling coils due to water condensation or coil tube leakage, every HVAC unit used to cool sodium or NaK containing areas and cells is provided 44
with a moisture eliminator.

2) The cooling coils and moisture eliminators are located above drain pans in the HVAC system casings. Double drain connections are provided for each drain pan. The primary drains are provided and sized for draining water due to condensation, which is a normal system event during the summer cooling cycle. The secondary drains are provided and sized for normal drainage back-up and for handling chilled water coil tube leaks.

3) The secondary drains are provided with water leak detectors. The activation of these detectors indicates failure of the primary drains or higher than normal leakage rate due to chilled water coil failure. The activation of these detectors initiates an alarm, stops the affected HVAC system fan, and closes the chilled water coil isolation valves.

4) The most probable reason for chilled water coil leakage is the rupture of the coil tubes during the winter months due to freeze-up. To prevent chilled water coil tube ruptures where outside air is circulated, an 44

override to the control valve is provided which auto-
matically opens the control valve when the outside
temperature drops below 34°F This permits the
required amount of chilled water to flow through the coil 44
to prevent coil freeze-up.

5) In addition to the above system features, the duct routing and design provides inherent protection from water carryover. The usually tortuous duct routing with its turns, valves, dampers and other system components, inhibits the transport of water and prevents
water contact with sodium or NaK containing components. 44

6) The Recirculating Gas Cooling System (RGCS) coil circuits
are fabricated from finned copper tubing to ASME Section
59 III Class 3 requirements. The high quality of the coil 44
fabrication required by the ASME Code minimizes the
probability of leakage due to tube wall failure.

Each coil nozzle is provided with an automatic butterfly 44
gas isolation valve (fail closed). The supply and return water lines to each cooler unit are provided with an automatic isolation valve (fail closed). One dew point detector and one water detector are provided for each cooler. Redundant drain valves are provided in
the bottom of each cooler. The actuation of any one of 44
the detectors will alarm and annunciate in the Control Room, close the gas and water isolation valves, and stop the fans. The location of the leak will be displayed
52 on the Recirculating Gas Cooling System back panel in the Control Room and on the local water leak detection panels. The water which has leaked prior to closure of the water isolation valves is drained away through the redundant drain valves.

52

In the event that all isolation valves and drains fail,
the orientation of the RGCS components are such that a 44
leak rate postulated in accordance with USNRC-SRP 3.6.1 and 3.6.2 would require approximately one hour to fill the RGCS components to a point where water would start
59 to spill into the cell being serviced. This amount 44
of time is sufficient to take corrective action.

9.7.4 Secondary Coolant Loops (SCL)

Dowtherm J physical properties are described in Table 9.7-6.
Figures 9.7-7 and 9.7-13 show the piping, valves, and equipment schema-
59 tics of the Secondary Coolant Loops.

1) Secondary Coolant Loops, containing Dowtherm J as the heat transfer fluid, are provided to isolate water from sodium in those areas where the measures described in Section 9.7.3 cannot be met. Secondary Coolant Loops are used as follows:

   - o Fuel handling cell secondary coolant loop (two redundant loops). Equipment located in RSB cells 321 and 322 at elev. 779'-0". Each loop contains approximately 500 gals. of Dowtherm J.

   - o Primary Na cold trap NaK cooler secondary coolant loop. Equipment located RCB cell 168 at elev. 752'-8". The loop contains approximately 600 gals. of Dowtherm J.

Each secondary coolant loop consists of a Dowtherm J to
Water heat exchanger, a circulating pump, an air separator,
59 an expansion tank, and associated piping, valves and in-
strumentation. Each coolant loop is fabricated to high
quality standards from schedule 40 pipe. The Primary Na
Cold Trap NaK cooler piping loop is designed and construc-
59 ted to ASME Section III, Class 3 requirements. The Fuel
Handling Cell cooler unit piping loop is designed and con-
structed to ANSI B31.1 requirements. All secondary coolant
loop piping is designed to the requirements of Seismic
Category I. The maximum system pressure will be 150 psig
59 and the maximum operating temperature will be 175°F accor-
dingly.

The design pressure of system components is 200 psig. The
59 design temperature is 200°F. The allowable working
pressure (-20°F to +650°F) of schedule 40 pipe varies from
906 psig for 2" pipe to 839 psig for 14" pipe. The quality
and conservative design makes the possibility of any
Dowtherm J leak extremely unlikely. If a leak should
occur, leak detectors will actuate valves which isolate the
water side of the Dowtherm J/Water heat exchanger from the
chilled water system and isolate the NaK cooler or Fuel
Handling Cell cooler unit from the secondary coolant
loop to prevent Dowtherm J leakage into the chilled water
system and NaK loop of Fuel Handling Cell. In addition, 44
the leak detectors are provided to shutdown the secondary
59 coolant loop circulating pump upon Dowtherm leakage.

2) The design basis for the Dowtherm J/primary sodium system separation is to provide at least two passive barriers between the two coolants to prevent the introduction of Dowtherm J into the sodium. In the case of the Fuel Handling Cell unit coolers the two barriers are the Dowtherm J pipe and cooler coils and the sodium primary boundary.

The first passive barrier separating Dowtherm J from the air-conditioning gases are the coils built in the coolers. In the coolers, the Dowtherm J operates at a higher pressure than the gas being cooled. If a coil leaks,
59 the leak is detected on the gas side by vapor detectors or liquid level probes. Leaks will also be detected by the low level indicator in the Dowtherm J expansion tank. A leak detection signal closes the Dowtherm J isolation valves to the cooling coil and shuts down the circulating pump.

The second passive barrier is the primary sodium container (pipe). Additionally, the fuel handling components are housed in nitrogen filled cells, the walls of which provide a third passive barrier for the liquid Dowtherm J
59 in the FHC unit coolers. This arrangement also provides
45 a lengthy transport path from the FHC unit coolers to the FHC, so that liquid Dowtherm J is not expected to enter the Fuel Handling Cell.

In the case of the Primary Na Cold Trap NaK cooler secondary coolant loop, a different leakage path is possible than that for the FHC. The potential path consists of leakage of Dowtherm J through the NaK cooler, through the NaK system, and through the Primary Cold Trap to the primary sodium. In this instance, double barrier protection is provided by the tubes of the NaK cooler and the Primary Cold Trap. Detection of leakage in both the secondary coolant loop and the NaK loop is provided. Double failure of heat exchanger tube sheets is not considered credible. Leakage of Dowtherm J into the NaK of the primary cold trap cooling loop poses no safety problems (See Section 9.7.4 for possible effects).

3) Dowtherm J is thermally stable up to 575°F when isolated from other compounds. Above this temperature, the thermal degradation rate will be significant and volatile low boiling and viscous high boiling materials are formed. The high boilers could cause poor circulation, decreased heat transfer efficiency, and carbon formation on
59 heat transfer surfaces. The low

44

Amend. 59
Dec. 1980

boilers would increase the vapor pressure and lower the flash point of the fluid. Thermal breakdown may be precipitated and/or accelerated by the presence of less stable contaminants such as corrosion products.

Dowtherm J (~95% diethyl benzene) is relatively stable
59 and inert in sodium as well as in water and NaK. The absence of oxygen in the compound is also favorable from the stability standpoint.

In the presence of sodium, the decomposition temperature of Dowtherm J is expected to be less than 575°F. The decomposition rate will increase rapidly above this temperature. The heat of decomposition is expected to be small. Decomposition or polymerization releases free carbon and hydrogen and longer chain molecules. At reactor operating temperature, nearly 100% decomposition is expected in a few days.

If the leakage is small, the decomposition products can be dissolved in the sodium. Large amounts of decomposition products, however, could cause carbonaceous residue deposition on heat transfer surfaces with loss of heat transfer efficiency.

At intermediate temperatures, it is probable that Na-benzene compounds will form. Some of these compounds could be pyrophoric, which would require special precautions during maintenance.

Radiation decomposition of Dowtherm J has not been tested, but most organic fluids decompose at about 10^9 rads. This dose requires about a year of exposure to Sodium 24 activity; however, this would be reduced to a few hours by circulation through an operating core.

Decomposition results in the breaking and reforming of molecules and the creation of some long chain molecules with high carbon to hydrogen ratios. Small quantities of decomposition products can be dissolved in the Na; large quantities may form coke deposits on surfaces and areas where corrosion products exist.

Carbonaceous decomposition products have been removed from sodium systems through the cold trapping operation. An example of this was the Sodium Reactor Experiment (SRE), which had its original startup in April 1957 and experienced a (~10 gal.) tetralin leak from a pump shaft freeze seal into the primary sodium coolant in June 1959. (1) 44

After about six weeks of trapping during shutdown and low power operation, failure of several fuel elements occurred due to blockage of fuel channel flow. A large amount of fission product was released to the coolant, causing extensive contamination of the primary heat transfer system. In addition, carbonaceous decomposition products of tetralin blocked or partially blocked approximately 10% of the tubes in the SRE Main Intermediate Heat Exchanger.

Carbon was removed by mechanical cleaning of the drained reactor and by hot and cold trapping after fill. Low power operation was initiated and trapping continued with monitoring for carbon in sodium to minimize risk of carbonaceous deposits. Several cold traps were filled as the fission product activity in the system was reduced. The continued operation at high temperatures with extensive hot trapping and cold trapping gradually dissolved and removed this carbonaceous material and other impurities from the systems. Post operation examination of one of the latter cold traps, which had been in service for ~3,000 hours, revealed the existence of carbon in a maximum concentration of 1500 ppm in sodium, or about 100 times as high as in the sodium. Despite the presence of this impurity, and others, the cold trap was able to restore and maintain a high level of sodium purity in the system.

As pointed out above, most organic fluids decompose at about 10^9 rads. Radiation sources released to containment for the events discussed in Chapter 15 "Accident Analysis" would result in exposure levels orders of magnitude below 10^9 rads and will result in negligible decomposition of the Dowtherm J.

59 4) Secondary coolant loop Dowtherm J equipment is housed in individual cells isolated from other plant equipment to minimize the hazards to the plant from a Dowtherm J fire. The secondary coolant loop equipment cells, RSB-321 & 322, RCB-168 (See RCB and RSB General arrangement drawings in Section 1.2 for cell locations) and the fuel handling unit cooler cell, RSB-342 & 343, are provided with fire detection and protection services by SDD-26B, Non-Sodium Fire Protection System. The Primary Na Cold Trap NaK Cooler cell (RCB cell 131) is an inerted cell, which due to lack of oxygen, precludes any fire. Dowtherm J piping not contained in these cells is run within a guard casing to contain and drain any potential leakage, thus avoiding a fire hazard.

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In addition, tests by the manufacturer show that due to the low vapor pressure of Dowtherm J, there is insufficient vapor present to support combustion at any temperature below 145°F which is well above the normal operating temperature of the containment. The temperature in the Reactor Containment Building during normal operation is 90°F. During loss of normal power, the temperature will rise to 120°F.

At temperatures above 145°F, a source of ignition must also be present to initiate the deflagration. Dowtherm J piping will be routed away from any hot surfaces, thus eliminating the possibility of a local area which might be capable of supporting combustion. Quick leak detection and isolation would limit the spill of Dowtherm J to a minor amount. Therefore a deflagration or a detonation, due to a vapor cloud forming as a result of Dowtherm J, leak is not credible. 44

TABLE 9.7-1
COMPONENTS SERVED BY
THE NORMAL CHILLED WATER SYSTEM

EQUIPMENT TITLE	LOCATION BLDG.	CELL	ELEVATION
MG Sets A/H Unit	CB	412	847'-3"
Loop #1 A/H Unit	SGB	244	852'-6"
Loop #2 A/H Unit	SGB	245	852'-6"
Loop #3 A/H Unit	SGB	246	852'-6"
SGB-IB A/H Unit	SGB	262	816'-0"
Maintenance Bay A/C Unit	SGB	261	816'-0"
Primary Na Tank Unit Cooler	SGB	211	733'-0"
Below Operating Floor A/C Unit	RCB	105I	752'-8"
Below Operating Floor A/C Unit	RCB	105K	752'-8"
Operating Floor Unit Cooler	RCB	161A	857'-11"
Operating Floor Unit Cooler	RCB	161A	857'-11"
Operating Floor Unit Cooler	RCB	161A	857'-11"
LCCV Unit Cooler	RCB	125	733'-0"
RCB A/H Unit	SGB	271	836'-0"
RSB A/H Unit	RSB	305H	733'-0"
RWA A/H Unit	RSB	660	867'-0"
Communication Center A/C Unit	RSB	328	865'-0"
Air Handling Unit	TGB	-	892'-0"
Air Handling Unit	TGB	-	862'-0"
Unit Cooler	TGB	-	816'-0"
Unit Cooler	TGB	-	838'-0"
Air Conditioning Unit	PSB	105	816'-0"
Air Conditioning Unit	PSB	151	816'-0"
Air Conditioning Unit	PSB	151	816'-0"
Air Conditioning Unit	PSB	151	816'-0"
Air Conditioning Unit	WB	212	828'-0"
Air Conditioning Unit	WB	210	828'-0"
Air Conditioning Unit	WB	210	828'-0"
CRDM	RCB	152	794'-0"
CRDM	RCB	152	794'-0"
Cold Trap, NaK Cooling, etc.	RCB	105V	794'-0"
EVST Cavity	RSB	306B	755'-0"

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47
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TABLE 9.7-1 (continued)

EQUIPMENT TITLE	LOCATION BLDG.	CELL	ELEVATION
PHTS, #1	RCB	150I	752'-8"
PHTS, #2	RCB	105J	752'-8"
PHTS, #3	RCB	105K	752'-8"
Reactor Cavity	RCB	105E	733'-0"
Cooler	RCB	125	746'-0"
Intermediate System Cooler & Condenser	SGB	235	TBD
Vapor Condenser	RCB	125	746'-9"
Intermediate System Condenser	SGB	235	TBD
Autoclave Sparge Gas Condenser	RSB	640	816'-0"
Primary Cold Trap NaK Cooler	RCB	131	790'-4 7/8"
CAPS Compressor Cooler	RSB	365	755'-0"
59 CAPS Compressor Cooler	RSB	366	755'-0"
RAPS Compressor Cooler	RCB	105BD	733'-0"
RAPS Compressor Cooler	RCB	105BE	733'-0"
SGB/IB Air Handling Unit	SGB	271	836'-0"
Constant Temperature Bath	TGB		838'-0"
47 Third Loop	RSB	324	816'-0"
HAA Unit Cooler	RCB	152	800'-9"
59 RAPS & CAPS Unit Cooler	RSB	365	755'-0"

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TABLE 9.7-6

PROPERTIES OF DOWTHERM-J*

Color	Clear, colorless solution
Boiling Point	358^{o}F
Flash Point	145^{o}F
Fire Point	155^{o}F
Auto Ignition Temperature	806^{o}F
Density at 60^{o}F	54.13 pounds/ft^{3}
Viscosity	1 centipoise
Specific Heat at 60^{o}F	0.434 BTU/lb. ^{o}F
Thermal Conductivity	0.077 BTU/hr. ^{o}F-ft-ft^{2}
Vapor Pressure at 60^{o}F	0.01 psia
Temperature Use Range	-100^{o}F to 575^{o}F

* Fluid properties from "Dowtherm J heat transfer fluid ... excellent stability from -100^{o}F to +575^{o}F", form number 176-1240-72, by Dow Chemical U.S.A., Functional Products and Systems Department, Midland. Michigan 48640

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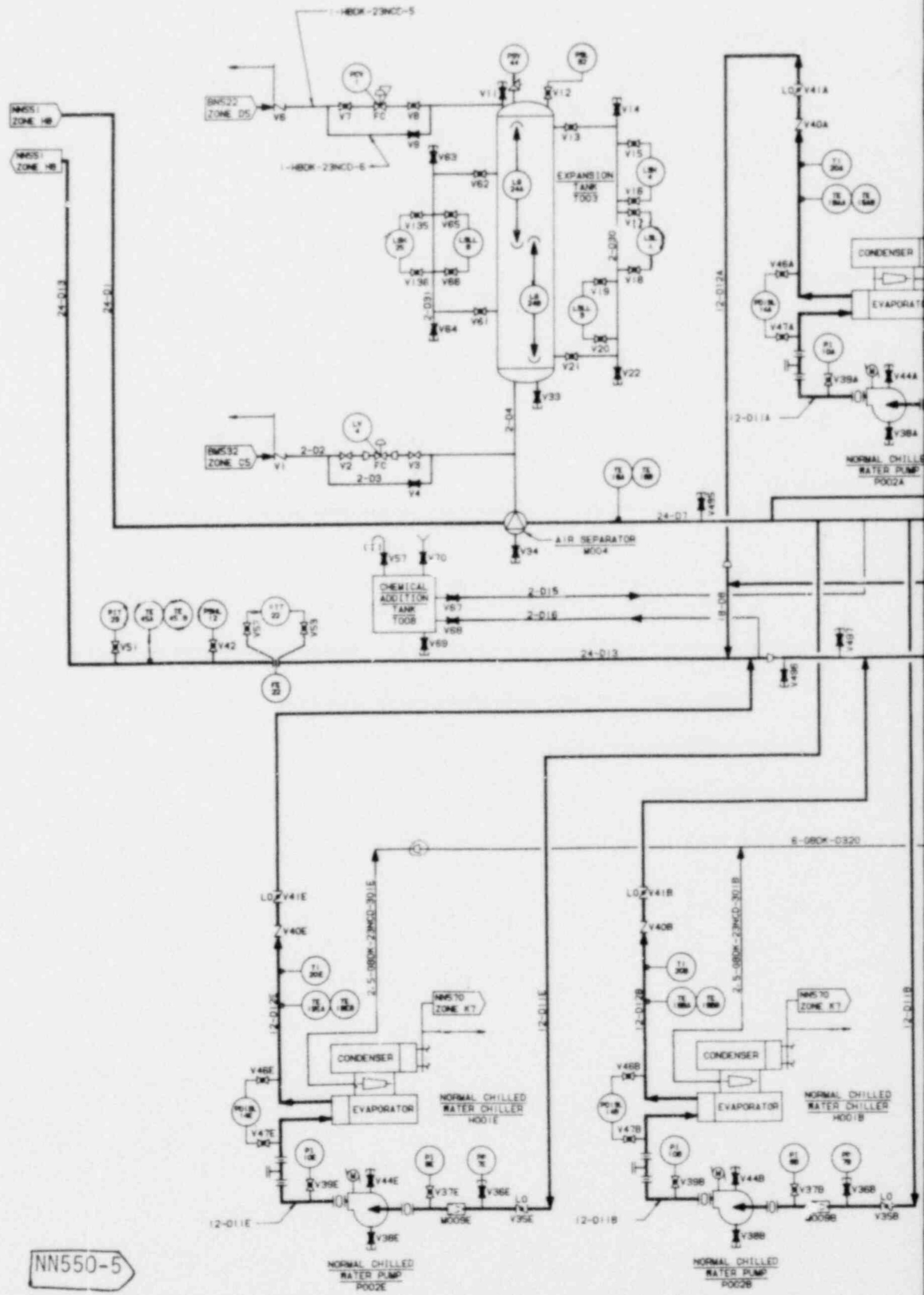
TABLE 9.7-7

ACTIVE SAFETY RELATED AIR OPERATED VALVES

Valve No.	Figure No.	Function	Normal Position	Fail Position
NV353	9.7-11	System Isolation	Open	Close
NV354	9.7-11	System Isolation	Open	Close
AOV165	9.7-11	Containment Isolation	Open	Close
AOV166	9.7-11	Containment Isolation	Open	Close
AOV167	9.7-11	Containment Isolation	Open	Close
AOV168	9.7-11	Containment Isolation	Open	Close
AOV211	9.7-11	Containment Isolation	Open	Close
AOV212	9.7-11	Containment Isolation	Open	Close
NV400	9.7-12	System Isolation	Open	Close
NV401	9.7-12	System Isolation	Open	Close
NV403	9.7-12	System Isolation	Open	Close
NV404	9.7-12	System Isolation	Open	Close
NV409	9.7-12	System Isolation	Open	Close
NV410	9.7-12	System Isolation	Open	Close
AOV79	9.7-12	Containment Isolation	Open	Close
AOV80	9.7-12	Containment Isolation	Open	Close
AOV415	9.7-12	Containment Isolation	Open	Close
AOV418	9.7-12	Containment Isolation	Open	Close
NV141AC	9.7-13	System Isolation	Open	Close
NV141AD	9.7-13	System Isolation	Open	Close
NV141BC	9.7-13	System Isolation	Open	Close
NV141BD	9.7-13	System Isolation	Open	Close
TV302	9.7-14	System Isolation	Open	Open
TV303	9.7-14	System Isolation	Open	Open

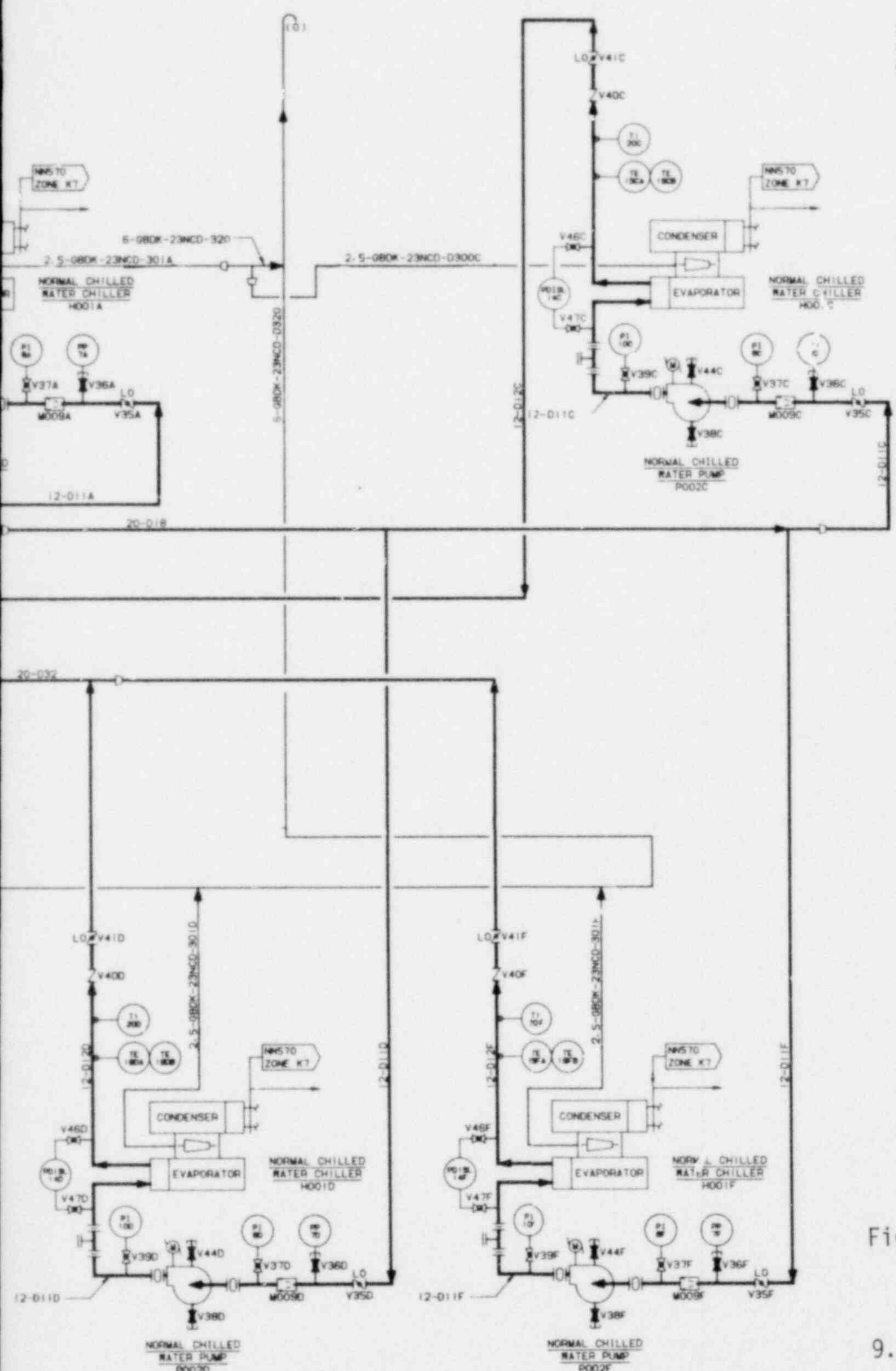
59


NN551 ZONE H8
BN522 ZONE D5
EXPANSION TANK T003
BM532 ZONE C5
AIR SEPARATOR M004
CHEMICAL ADDITION TANK T008
CONDENSER
EVAPORATOR
NORMAL CHILLED WATER CHILLER H001E
NORMAL CHILLED WATER CHILLER H001B
NORMAL CHILLED WATER PUMP P002E
NORMAL CHILLED WATER PUMP P002B
NORMAL CHILLED WATER PUMP P002A
NN570 ZONE K7
6-GBDK-D320
24-D13
2-D15
2-D16
NN550-5

NOTES

1. FOR GENERAL NOTES SEE DRAWING NN551
2. SYSTEM CLEANLINESS CLASSIFICATION: VALVES & PIPING CLASS C; CHILLERS CLASS B
3. SEISMIC CATEGORY: III UOS
4. RADIATION ZONE: UNRESTRICTED UOS
5. CODE CLASSIFICATION: ANSI B31.1

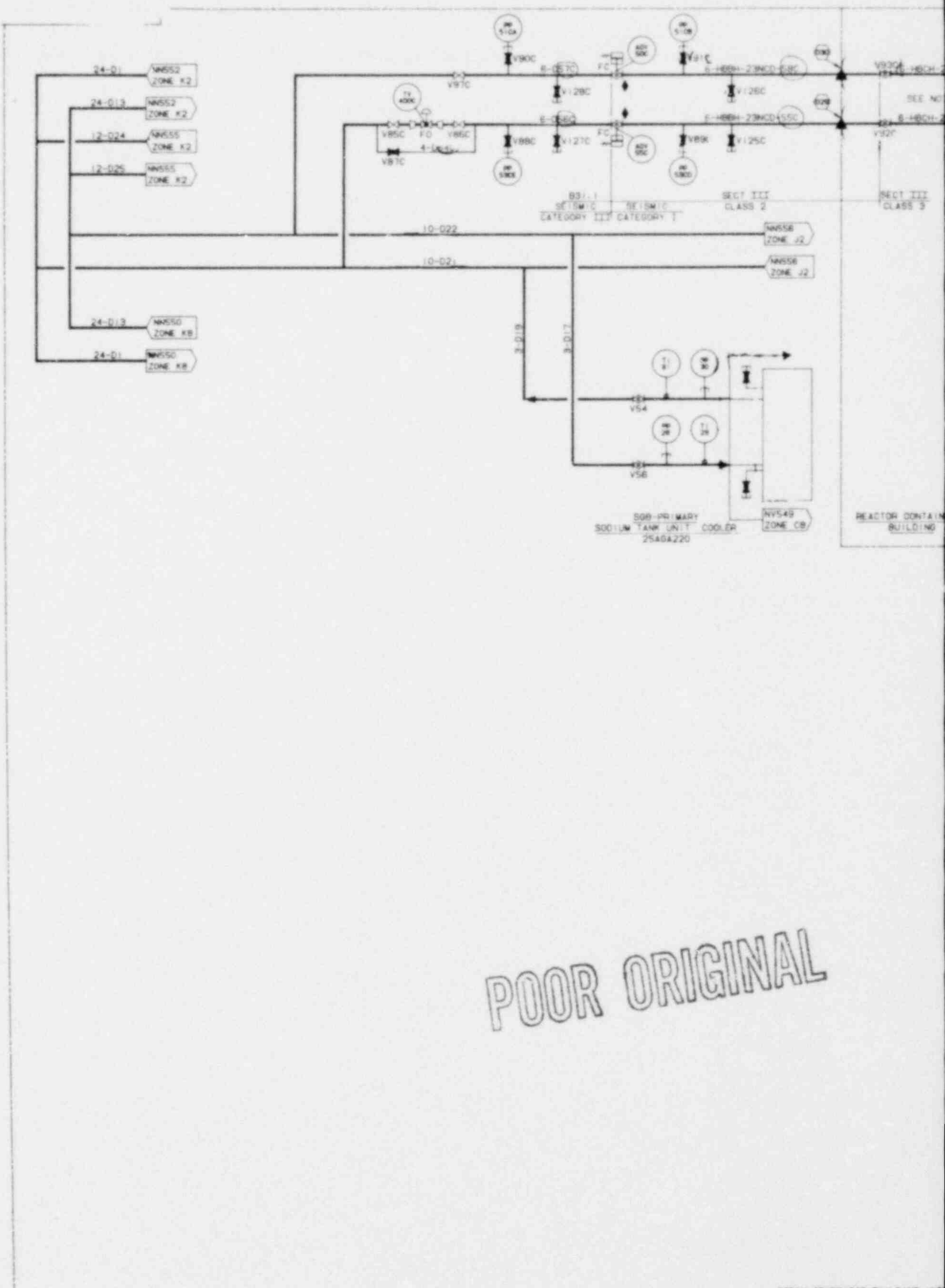
POOR ORIGINAL



REFERENCE DRAWINGS

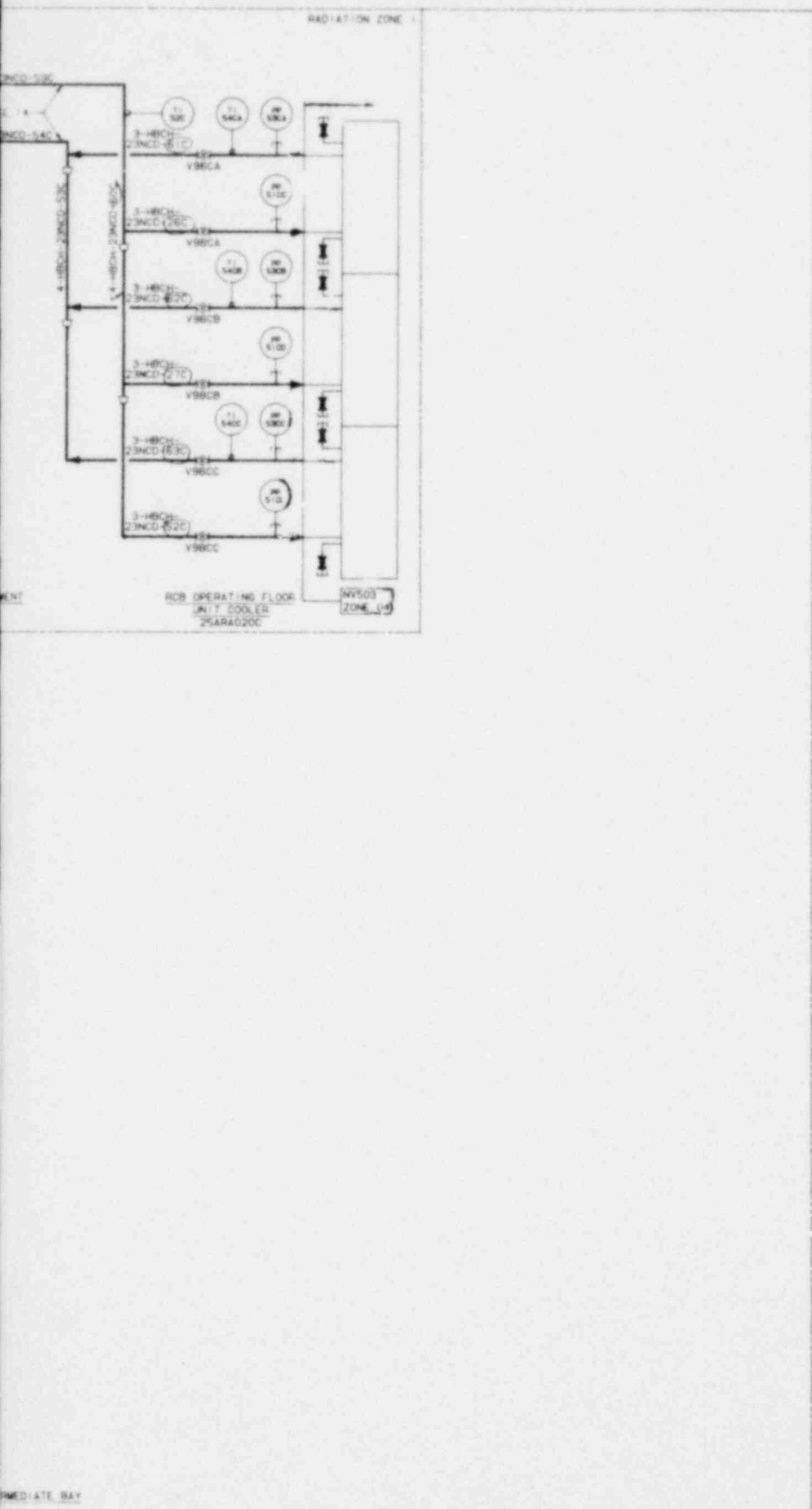
SEE DRAWINGS NN551, NN552 & NN553

Figure 9.7-1 NORMAL CHILLED WATER SYSTEM SGB



POOR ORIGINAL

NN551-5



GENERAL NOTES

1. SYMBOLS AND ABBREVIATIONS WARD-D-0036
2. ALL EQUIPMENT NUMBERS AND MANUAL VALVE NUMBERS ARE PREFIXED BY 23NC UOS
3. ALL INSTRUMENT NUMBERS AND CONTROL VALVE NUMBERS ARE PREFIXED BY 23NC UOS
4. LINE NUMBERS ARE ABBREVIATED AS PER THE FOLLOWING EXAMPLE UOS: 8-HBDH-23NCD-1 IS WRITTEN AS 8-D1
5. ALL PRESSURE AND FLOW CONNECTIONS SHALL BE 0.75" UOS
6. ALL VENT & DRAIN CONNECTIONS SHALL BE 0.75" UOS
7. SEISMIC CATEGORY: I & III
8. RADN ZONE: UNRESTRICTED & ZONE I
9. CLEANLINESS CLASSIFICATION: VALVES & PIPING CLASS C; CHILLERS CLASS B
10. CODE CLASSIFICATION: ANSI B31.1 & ASME SECTION III/CLASS 2&3
11. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY K
12. CERTAIN EQUIPMENT SHOWN ON THIS DWG IS INCLUDED IN THE PLANT PROTECTION SYSTEM (PPS). THIS EQUIPMENT IS SPECIFICALLY IDENTIFIED ON THE DWG AS FOLLOWS: THE SYMBOL ♦ DESIGNATES PPS EQUIPMENT
13. THIS DWG INCLUDES EQUIPMENT IDENTIFIED AS PART OF THE PLANT PROTECTION SYSTEM (PPS). BEFORE MODIFYING OR MAINTAINING THE EQUIPMENT SO IDENTIFIED THE APPROVAL OF THE COGNIZANT PERSONNEL FOR THE PPS MUST BE OBTAINED
14. ALL NORMAL CHILLED WATER SYSTEM PIPING INSIDE THE RCB AND ABOVE THE OPERATING FLOOR SHALL BE ENCLOSED WITHIN GUARD PIPING OR SPRAY SHIELD ENCLOSURES
15. ALL REACTOR CONTAINMENT PENETRATION NUMBERS ARE PREFIXED BY R23W UOS

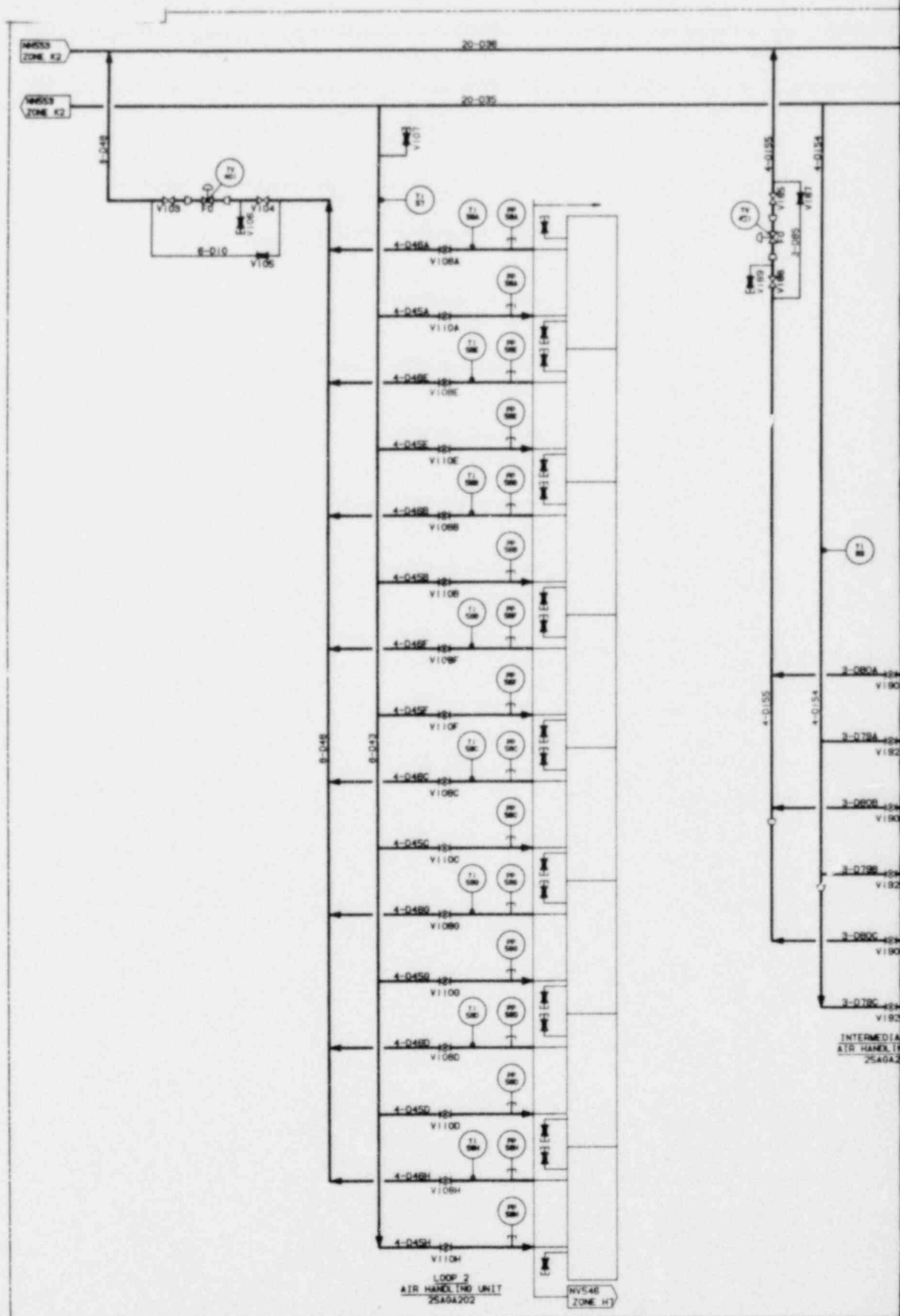
POOR ORIGINAL

REFERENCE DRAWINGS

1. P&ID NORMAL CHILLED WATER SYS RCB & SGB — B&R DWG NW552
2. P&ID NORMAL CHILLED WATER SYS SGB — B&R DWG NW553, B&R DWG NW550
3. P&ID NORMAL CHILLED WATER SYS PSB, RCB, SGB & CB — B&R DWG NW554
4. P&ID NORMAL CHILLED WATER SYS SGB, WSW & TGB — B&R DWG NW555
5. P&ID NORMAL CHILLED WATER SYS RCB & SGB — B&R DWG NW556
6. P&ID NORMAL CHILLED WATER SYS RCB — B&R DWG NW557
7. P&ID NORMAL CHILLED WATER SYS RSB — B&R DWG NW586
8. P&ID NORMAL PLANT SERVICE WATER — B&R DWG NW570
9. P&ID RCB HVAC — B&R DWG NV501
10. P&ID RSB HVAC — B&R DWG NV510
11. P&ID RSB HVAC — B&R DWG NV511
12. P&ID RSB/RW AREA HVAC — B&R DWG NV520

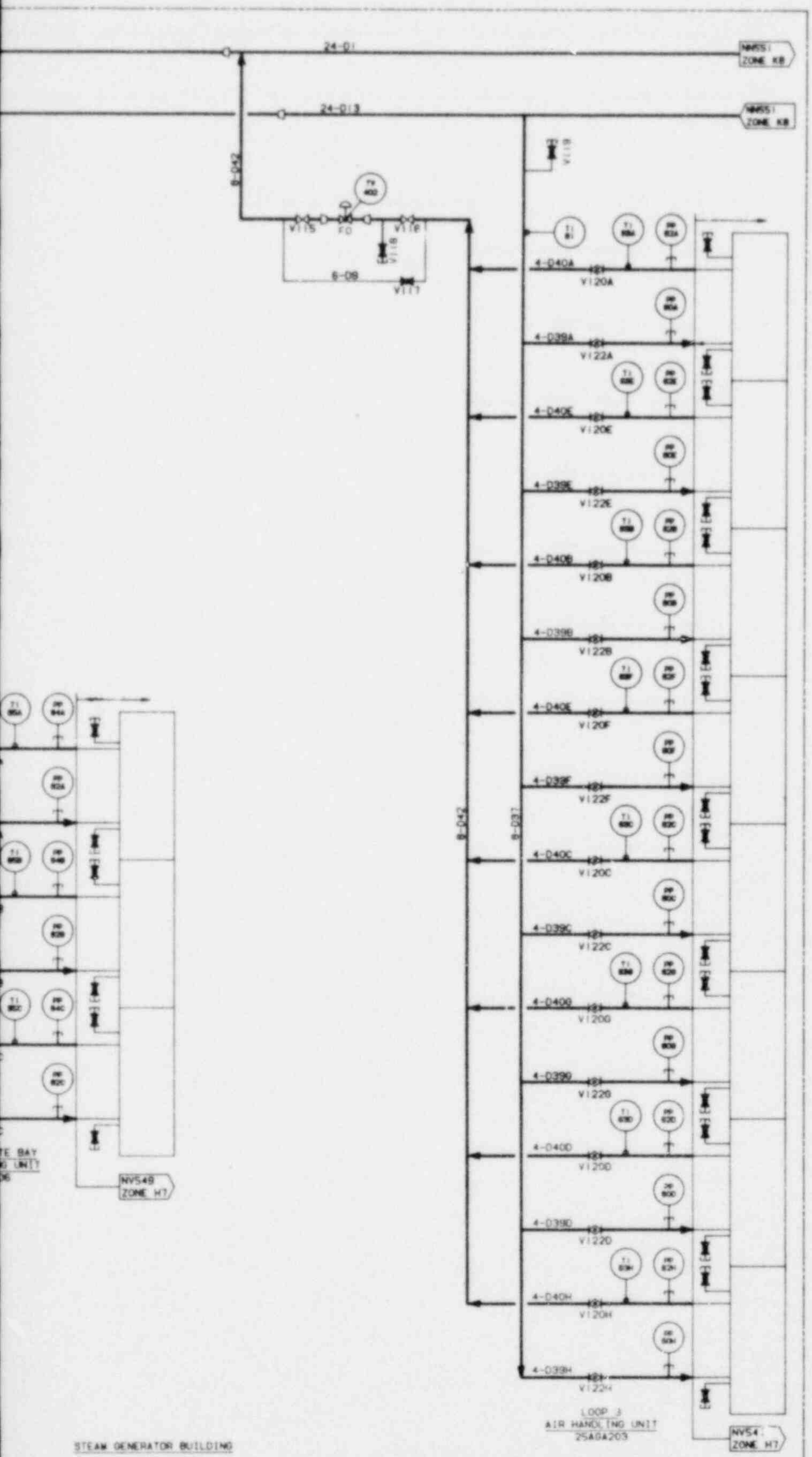
CONTINUED ON NW552

Figure 9.7-2 NORMAL CHILLED WATER SYSTEM RCB, SGB



NN552-3

POOR ORIGINAL



NOTES

1. FOR GENERAL NOTES SEE DWG NN551
2. SEISMIC CATEGORY: III
3. RADIATION ZONE: UNRESTRICTED
4. SYSTEM CLEANLINESS CLASSIFICATION: CLASS C
5. CODE CLASSIFICATION: ANSI B31.1

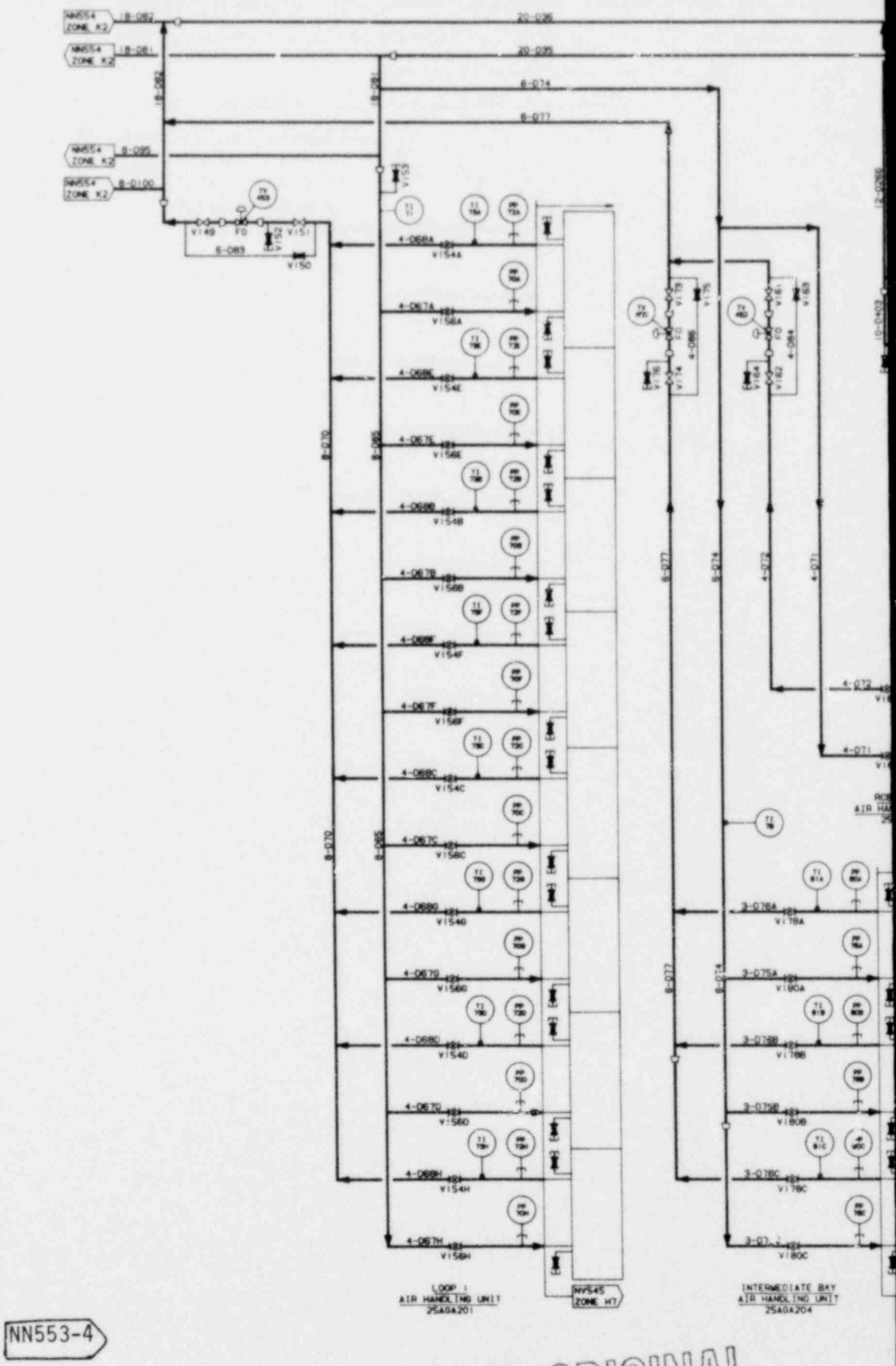
REFERENCE DRAWINGS

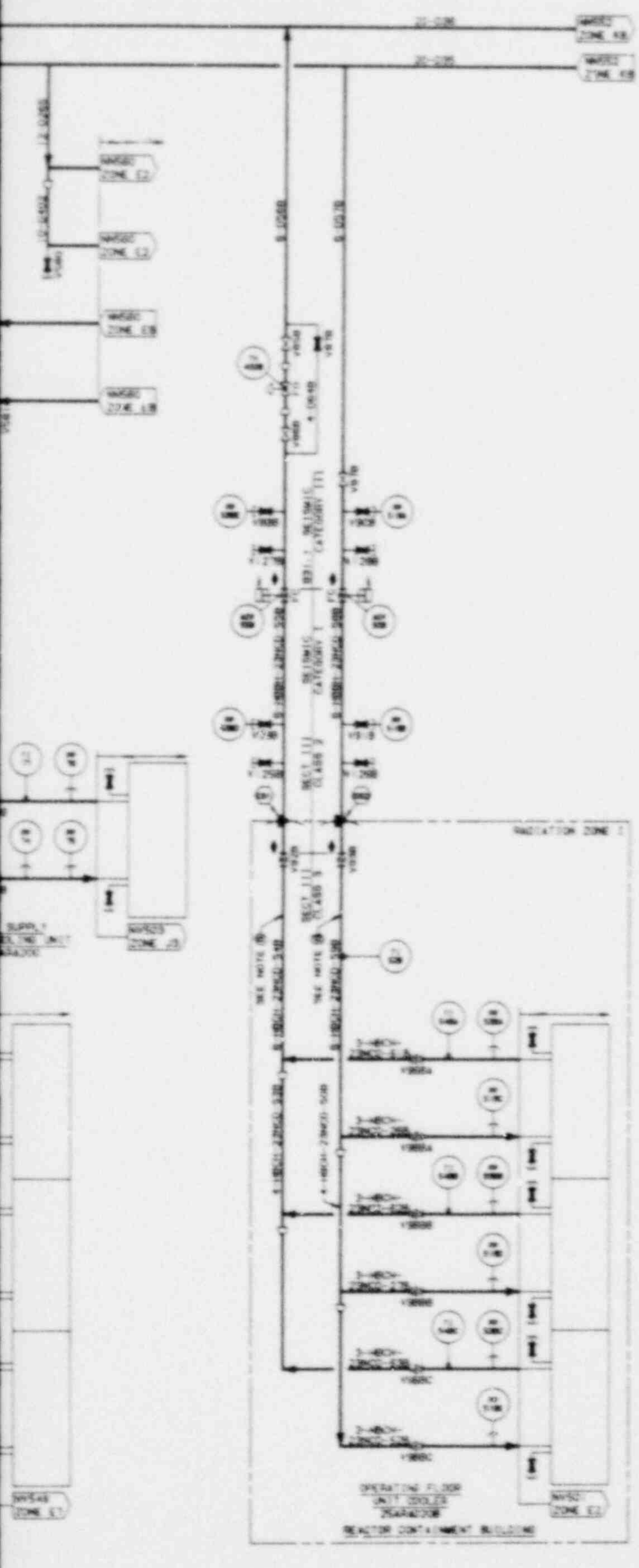
CONTINUED FROM NN551

13. P&ID MG SET & SWGR HVAC — B&R DWG NV528
14. P&ID CB CRDM & SWGR HVAC — B&R DWG NV529
15. P&ID CONTROL ROOM HVAC — B&R DWG NV530
16. P&ID SGB LOOP 1 & AUX BAY HVAC — B&R DWG NV545
17. P&ID SGB LOOP 2 & AUX BAY HVAC — B&R DWG NV546
18. P&ID SGB LOOP 3, AUX BAY & MB HVAC — B&R DWG NV547
19. P&ID SGB/IB HVAC — B&R DWG NV549
20. P&ID DEMINERALIZED WATER SYSTEM — B&R DWG BW532
21. P&ID TURBINE GENERATOR BUILDING HVAC — B&R DWG BV511
22. P&ID PLANT SERVICE BUILDING HVAC — B&R DWG BV521
23. P&ID PLANT SERVICE BUILDING HVAC — B&R DWG BV522
24. FLOW DIAGRAM MAINTENANCE SHOP & WAREHOUSE HVAC — B&R DWG BV531

CONTINUED ON DWG NN553

Figure 9.7-3 NORMAL CHILLED WATER SYSTEM RCB & SGB


NN554 ZONE K2
18-082
NN554 ZONE K2
18-081
20-036
20-035
8-074
8-077
NN554 ZONE K2
8-095
NN554 ZONE K2
8-0100
V153
V149
FO
V151
6-089
V150
8-070
8-065
4-068A
V154A
4-067A
V156A
4-068E
V154E
4-067E
V156E
4-068B
V154B
4-067B
V156B
4-068F
V154F
4-067F
V156F
4-068C
V154C
4-067C
V156C
4-068G
V154G
4-067G
V156G
4-068D
V154D
4-067D
V156D
4-068H
V154H
4-067H
V156H
V173
V175
V174
V176
4-086
V161
V163
V162
V164
4-084
8-077
8-074
4-072
4-071
3-078A
V178A
3-075A
V180A
3-078B
V178B
3-075B
V180B
3-078C
V178C
V180C
LOOP 1
AIR HANDLING UNIT
25A0A201
NV545
ZONE H7
INTERMEDIATE BAY
AIR HANDLING UNIT
25A0A204
NN553-4



NOTES

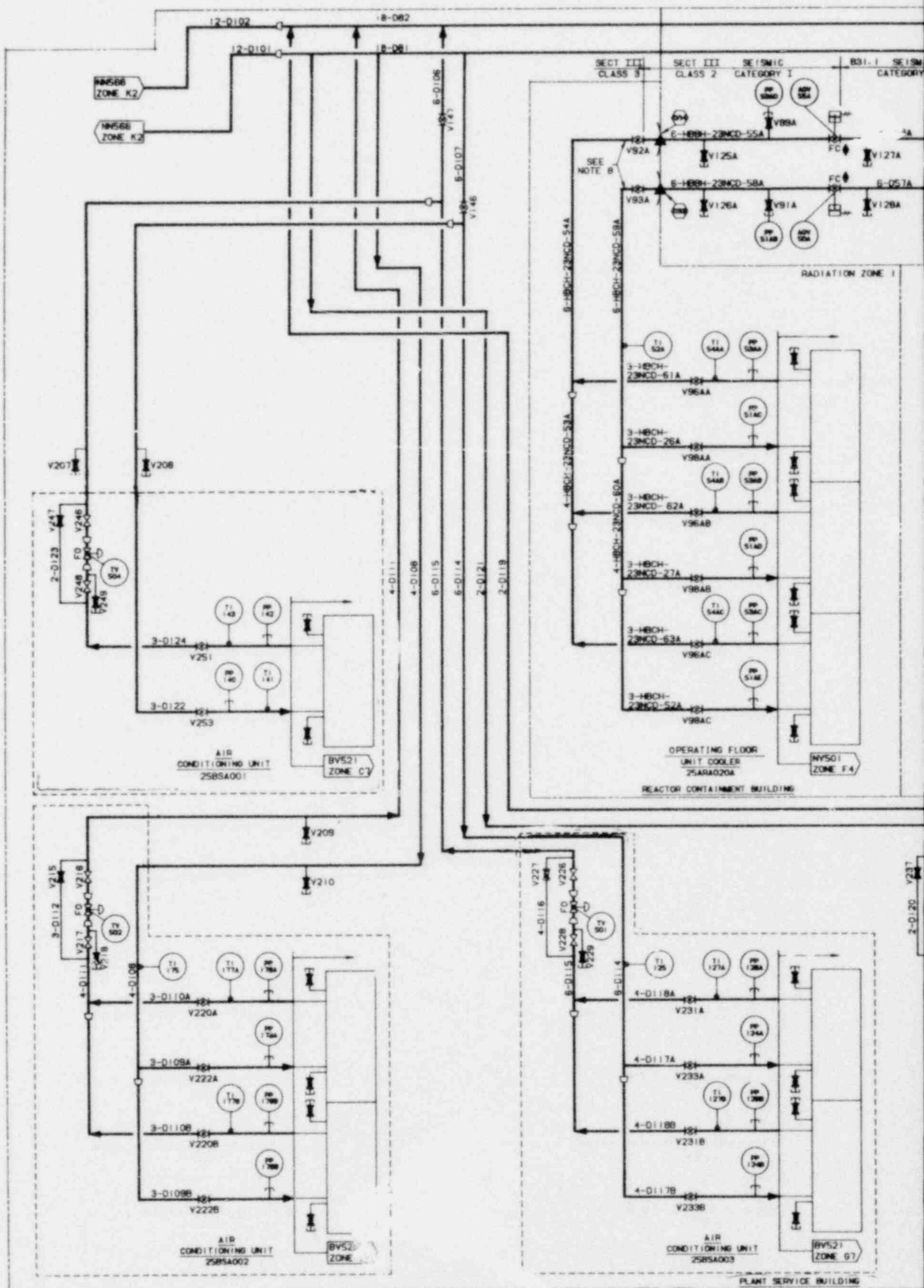
1. FOR GENERAL NOTES AND REFERENCE DRAWINGS SEE DWG NA551
2. SEISMIC CATEGORY I & III
3. RADIATION ZONE: UNRESTRICTED AND ZONE I
4. SYSTEM CLEANLINESS CLASSIFICATION CLASS [illegible]
5. CODE CLASSIFICATION: ANSI B31.1 ASME SECTION III/CLASS 2 & 3
6. CERTAIN EQUIPMENT SHOWN ON THIS DRAWING IS INCLUDED IN THE PLANT PROTECTION SYSTEM (PPS). THIS EQUIPMENT IS SPECIFICALLY IDENTIFIED ON THE DRAWING AS FOLLOWS: THE SYMBOL ◆ DESIGNATES PPS EQUIPMENT
7. THIS DRAWING INCLUDES EQUIPMENT IDENTIFIED AS PART OF THE PLANT PROTECTION SYSTEM (PPS). BEFORE MODIFYING OR MAINTAINING THE EQUIPMENT SO IDENTIFIED THE APPROVAL OF THE COGNIZANT PERSONNEL FOR THE PPS MUST BE OBTAINED
8. ALL NORMAL CHILLED WATER SYSTEM PIPING INSIDE THE RCB AND ABOVE THE OPERATING FLOOR SHALL BE ENCLOSED WITHIN GUARD PIPING OR SPRAY SHIELD ENCLOSURES
9. ALL REACTOR CONTAINMENT PENETRATION NUMBERS ARE PREFIXED BY [illegible]

REFERENCE DRAWINGS

CONTINUED FROM DWG NA552

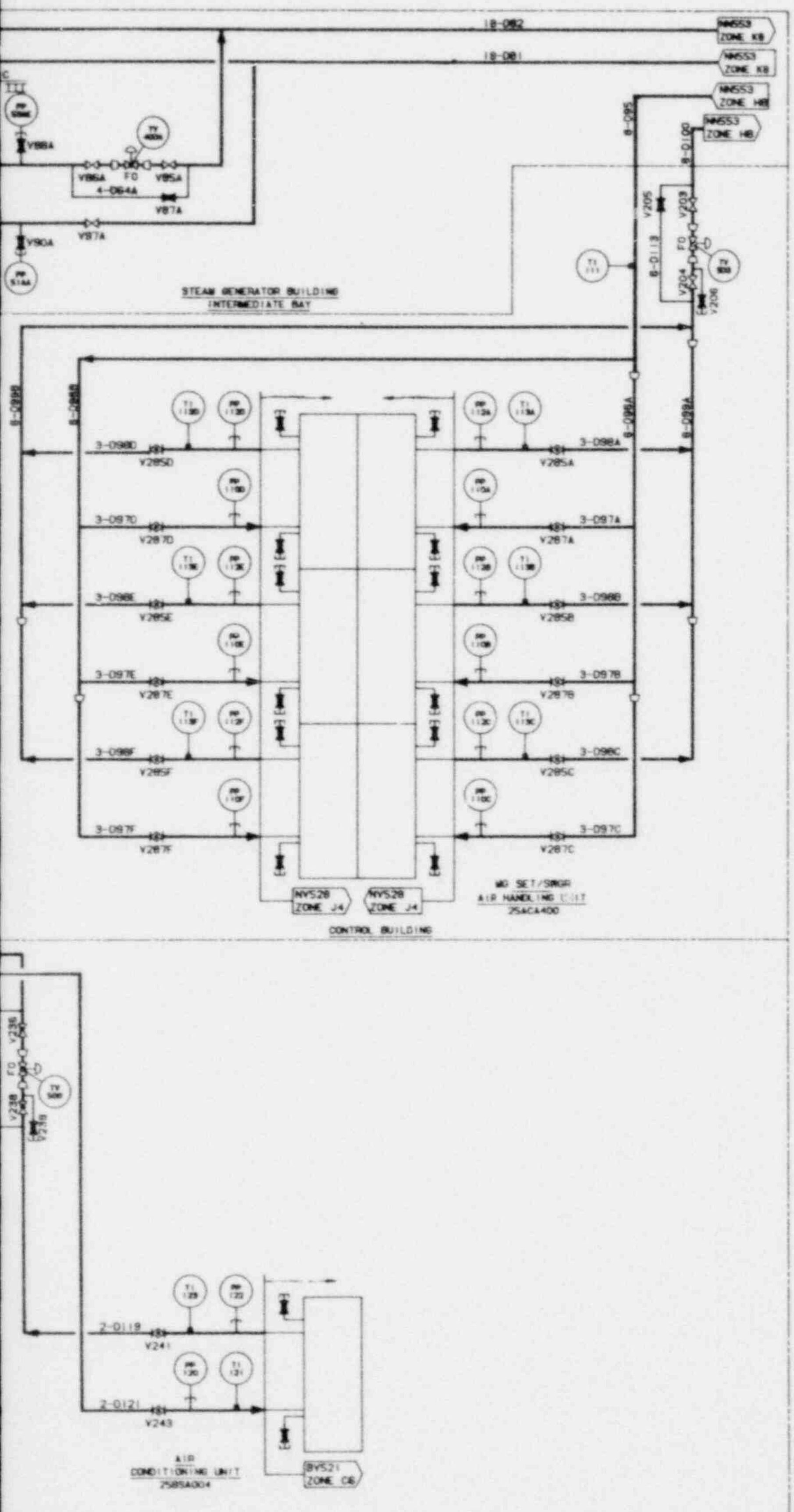
25. FLOW DIAGRAM MAINTENANCE SHOP & WAREHOUSE HVAC [illegible] DWG [illegible]
26. P&ID CELL ATM PROCESSING AI DWG [illegible]
27. P&ID PRIMARY SODIUM REMOVAL & DECONTAMINATION SYS AI DWG [illegible]
28. FLOW DIAGRAM SGB PROCESS INTERMEDIATE SODIUM REMOVAL SYS AI DWG [illegible]
29. P&ID SMALL COMPONENT AUTOCLAVE SODIUM REMOVAL SYS AI DWG [illegible]
30. P&ID RADIOACTIVE ARGON PROCESSING SYS AI DWG [illegible]
31. P&ID RADIOACTIVE ARGON PROCESSING SYS AI DWG [illegible]
32. P&ID RCB HI DIST HIGH PRESSURE AI DWG [illegible]

Figure 9.7-4 NORMAL CHILLED WATER SYSTEM SGB



NN554-7

POOR ORIGINAL



NOTES

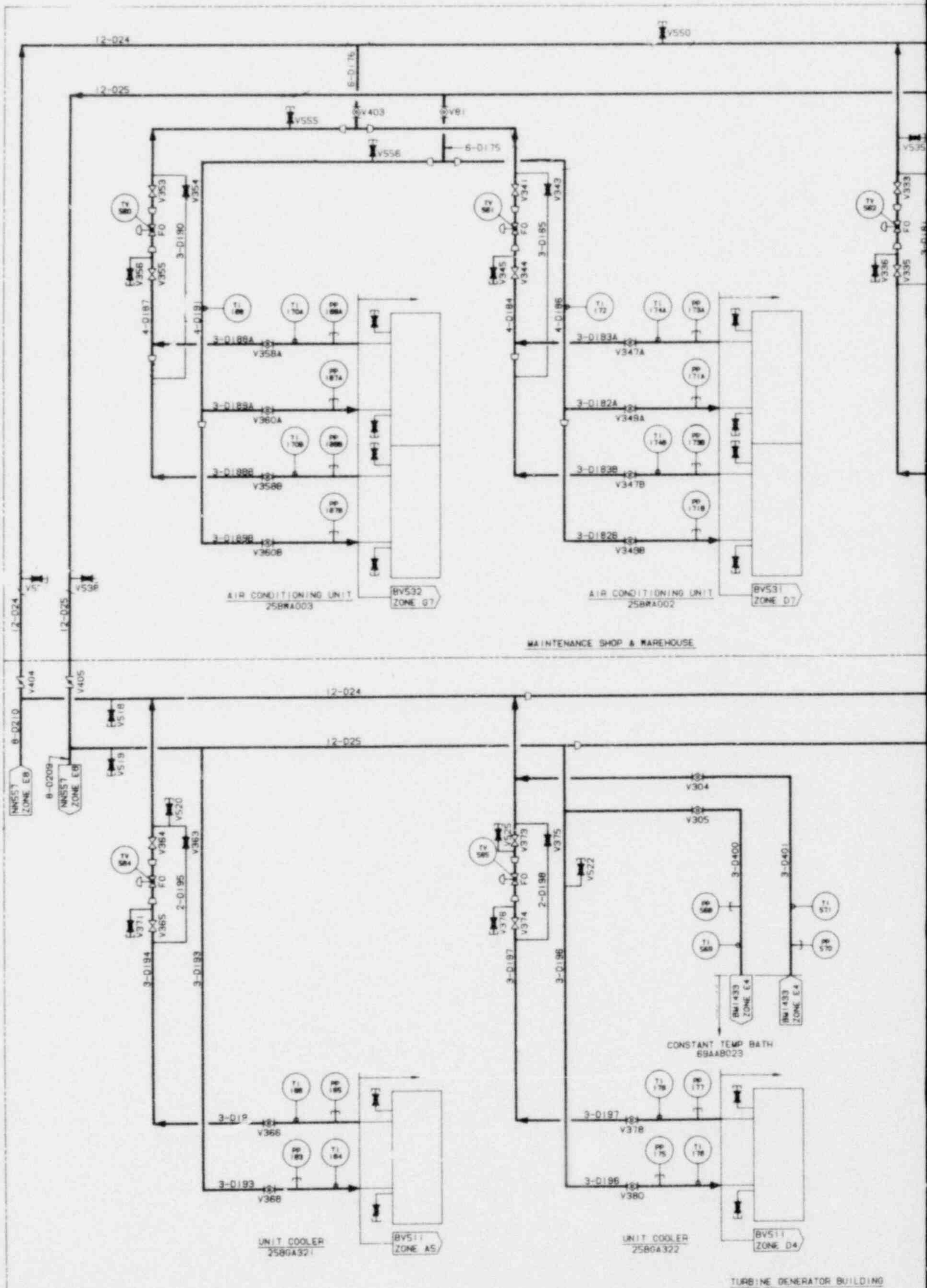
1. FOR GENERAL NOTES SEE DWG NN551
2. SEISMIC CATEGORY: I & III
3. RADN ZONE: UNRESTRICTED & ZONE I
4. CLEANLINESS CLASSIFICATION: CLASS C
5. CODE CLASSIFICATION: ANSI B31.1 & ASME SECTION III/CLASS 2&3
6. CERTAIN EQUIPMENT SHOWN ON THIS DWG IS INCLUDED IN THE PLANT PROTECTION SYSTEM (PPS). THIS IDENTIFIED ON THE DWG AS FOLLOWS: THE SYMBOL ◆ DESIGNATES PPS EQUIPMENT
7. THIS DWG INCLUDES EQUIPMENT IDENTIFIED AS PART OF THE PLANT PROTECTION SYSTEM (PPS). BEFORE MODIFYING OR MAINTAINING THE EQUIPMENT SO IDENTIFIED THE APPROVAL OF THE COGNIZANT PERSONNEL FOR THE PPS MUST BE OBTAINED
8. ALL NORMAL CHILLED WATER SYSTEM PIPING INSIDE THE RCB AND ABOVE THE OPERATING FLOOR SHALL BE ENCLOSED WITHIN GUARD PIPING OR SPRAY SHIELD ENCLOSURES
9. ALL REACTOR CONTAINMENT PENETRATION NUMBERS ARE PREFIXED BY R23W UOS
10. ITEMS SHOWN WITHIN DASHED LINES ARE ROOFTOP AIR CONDITIONING UNIT VENDOR PACKAGE ITEMS

POOR ORIGINAL

REFERENCE DRAWINGS

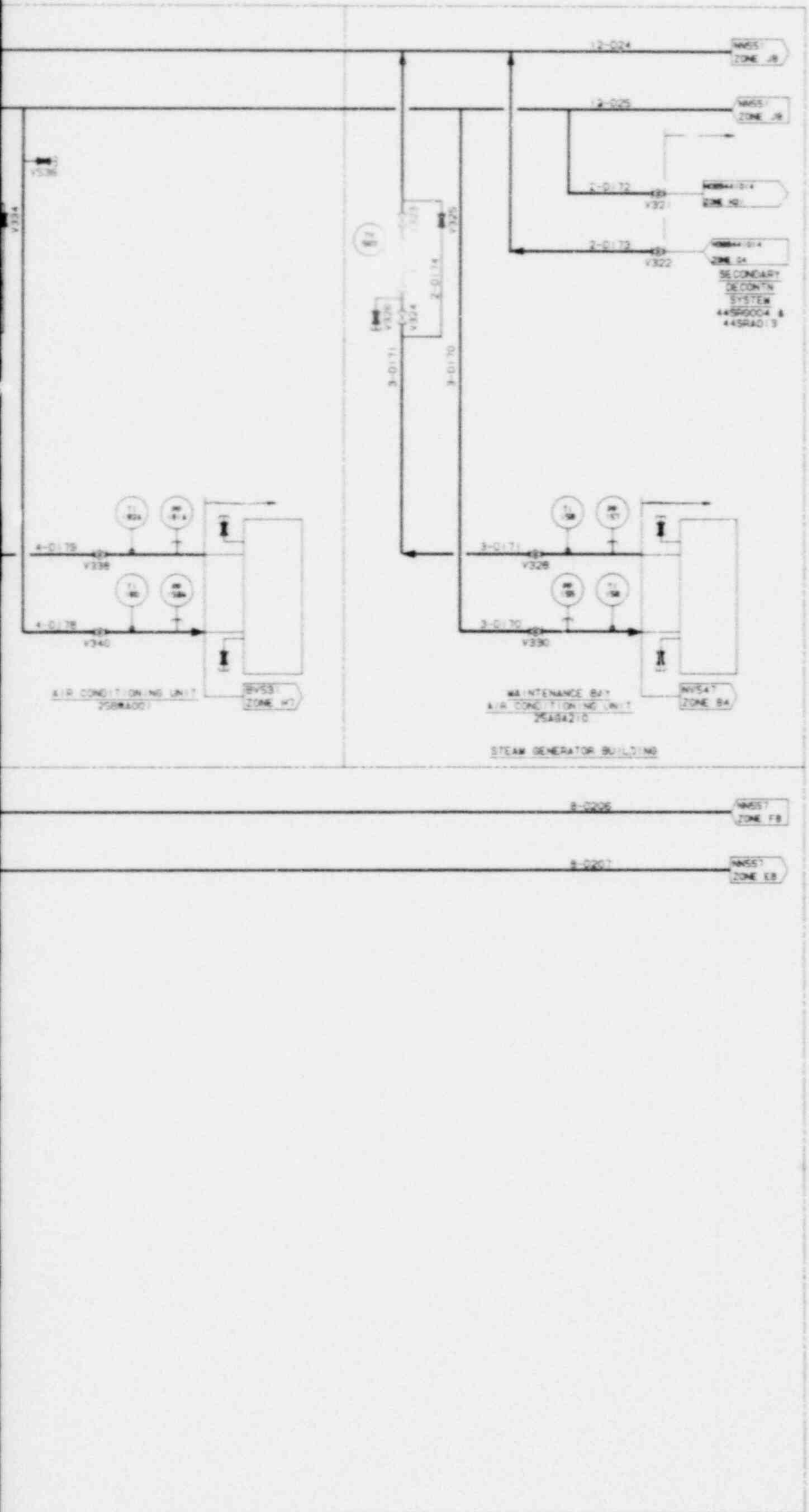
FOR REFERENCE DWGS SEE NN551, NN552 & NN553

Figure 9.7-5 NORMAL CHILLED WATER SYSTEM PSB, RCB, CB & SGB



NN555-6

POOR ORIGINAL



NOTES
1. FOR GENERAL NOTES SEE DWG NN551
2. SEISMIC CATEGORY: III
3. RADIATION ZONE: UNRESTRICTED
4. CLEANLINESS CLASSIFICATION: CLASS C
5. CODE CLASSIFICATION: ANSI B31.1

POOR ORIGINAL

REFERENCE DRAWINGS

FOR REFERENCE DWGS SEE NN551, NN552 & NN553

Figure 9.7-6 NORMAL CHILLED WATER SYSTEM SGB, MSW, TGB

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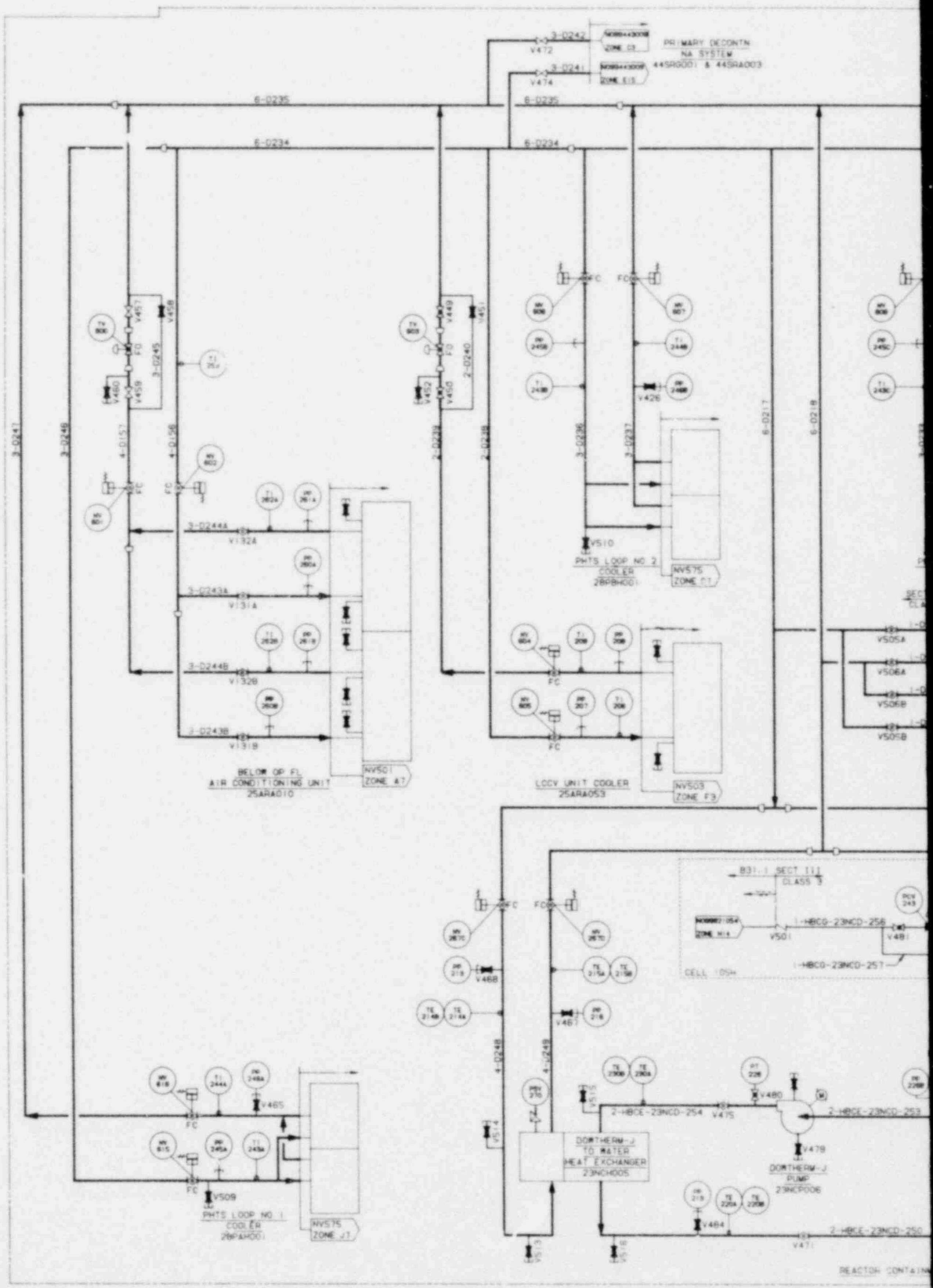
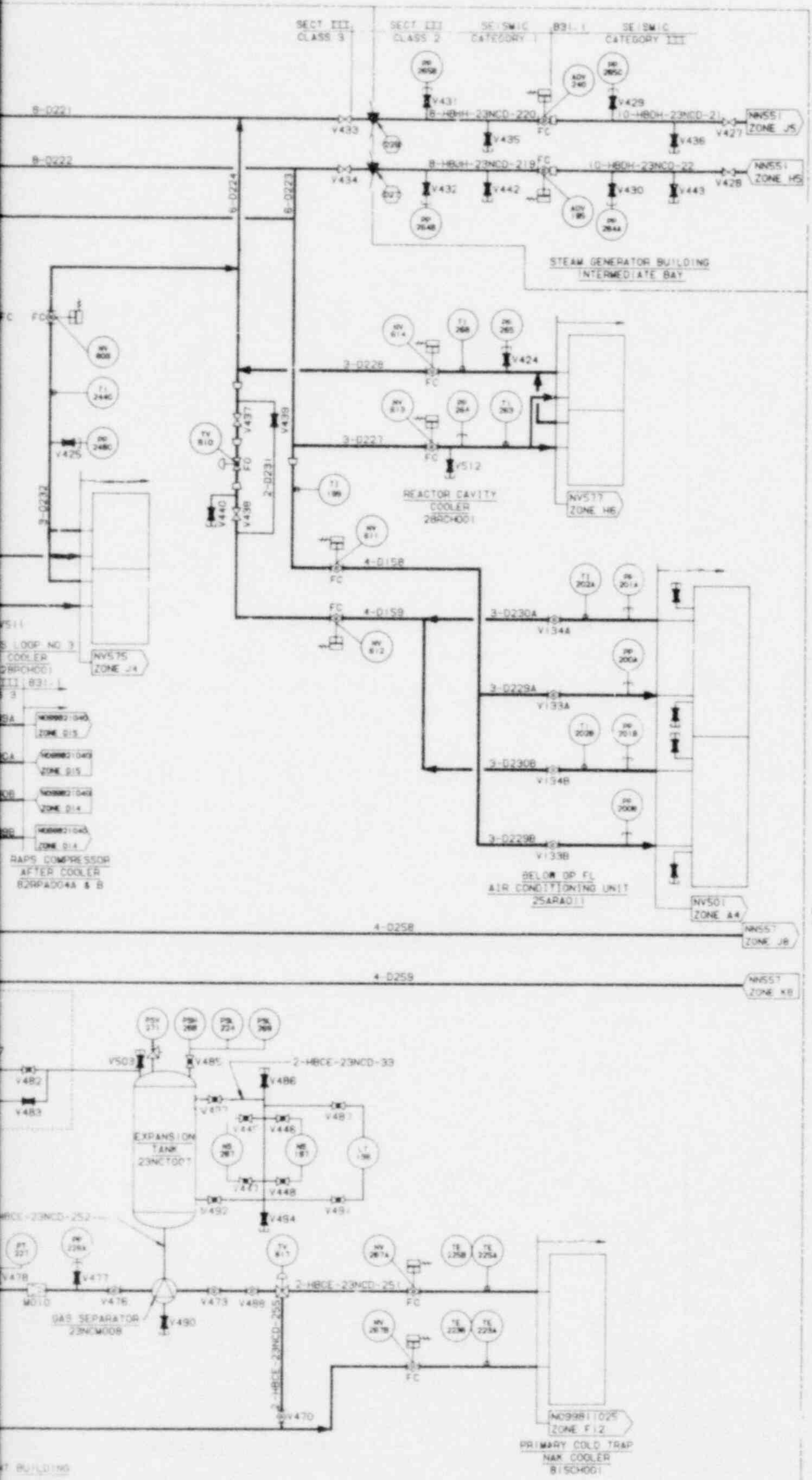


NN556-9

POOR ORIGINAL



NOTES

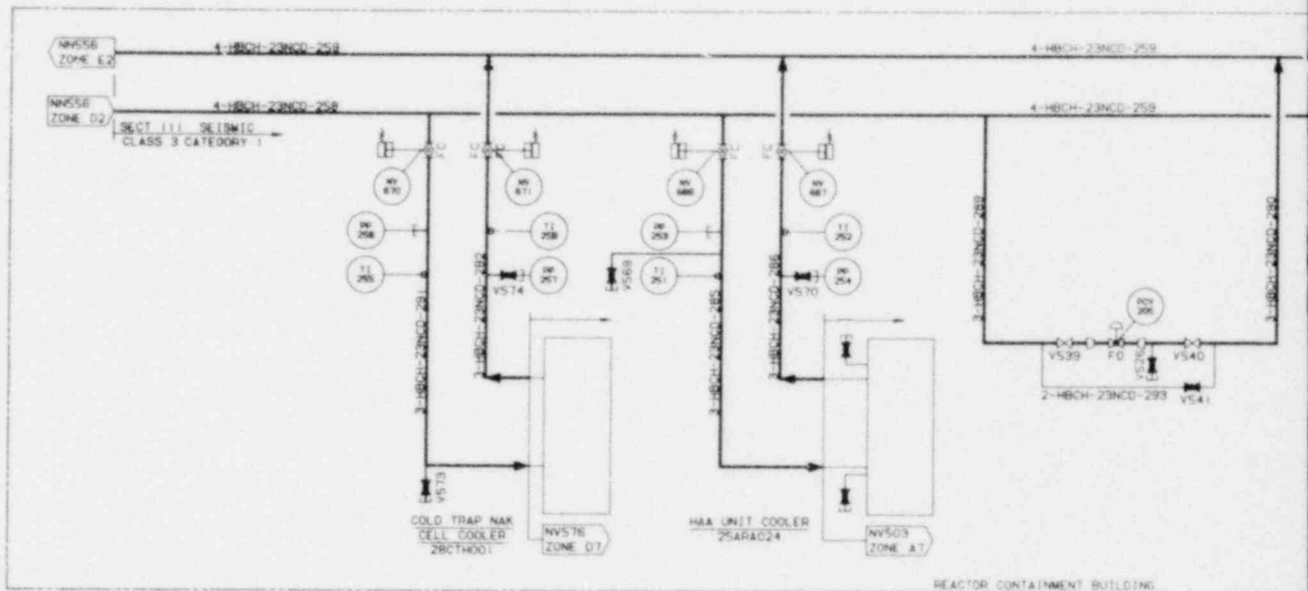
1. FOR GENERAL NOTES SEE DWG NN551
2. SEISMIC CATEGORY: I & III
3. RADIATION ZONE
4. CLEANLINESS CLASSIFICATION: CLASS C
5. CODE CLASSIFICATION: ANSI B31.1 & ASME SECT III/CL 2 & CL 3
6. CERTAIN EQUIPMENT SHOWN ON THIS DWG IS INCLUDED IN THE PLANT PROTECTION SYSTEM (PPS). THIS EQUIPMENT IS SPECIFICALLY IDENTIFIED ON THE DWG AS FOLLOWS: THE SYMBOL ♦ DESIGNATES PPS EQUIPMENT
7. THIS DWG INCLUDES EQUIPMENT IDENTIFIED AS PART OF THE PLANT PROTECTION SYSTEM (PPS). BEFORE MODIFYING OR MAINTAINING THE EQUIPMENT SO IDENTIFIED THE APPROVAL OF THE COGNIZANT PERSONNEL FOR THE PPS MUST BE OBTAINED
8. LINE NUMBERS ARE ABBREVIATED AS PER THE FOLLOWING EXAMPLE UOS 8-HBCH-23NCD-221 IS WRITTEN AS 8-D221
9. ALL REACTOR CONTAINMENT PENETRATION NUMBERS ARE PREFIXED BY R23W

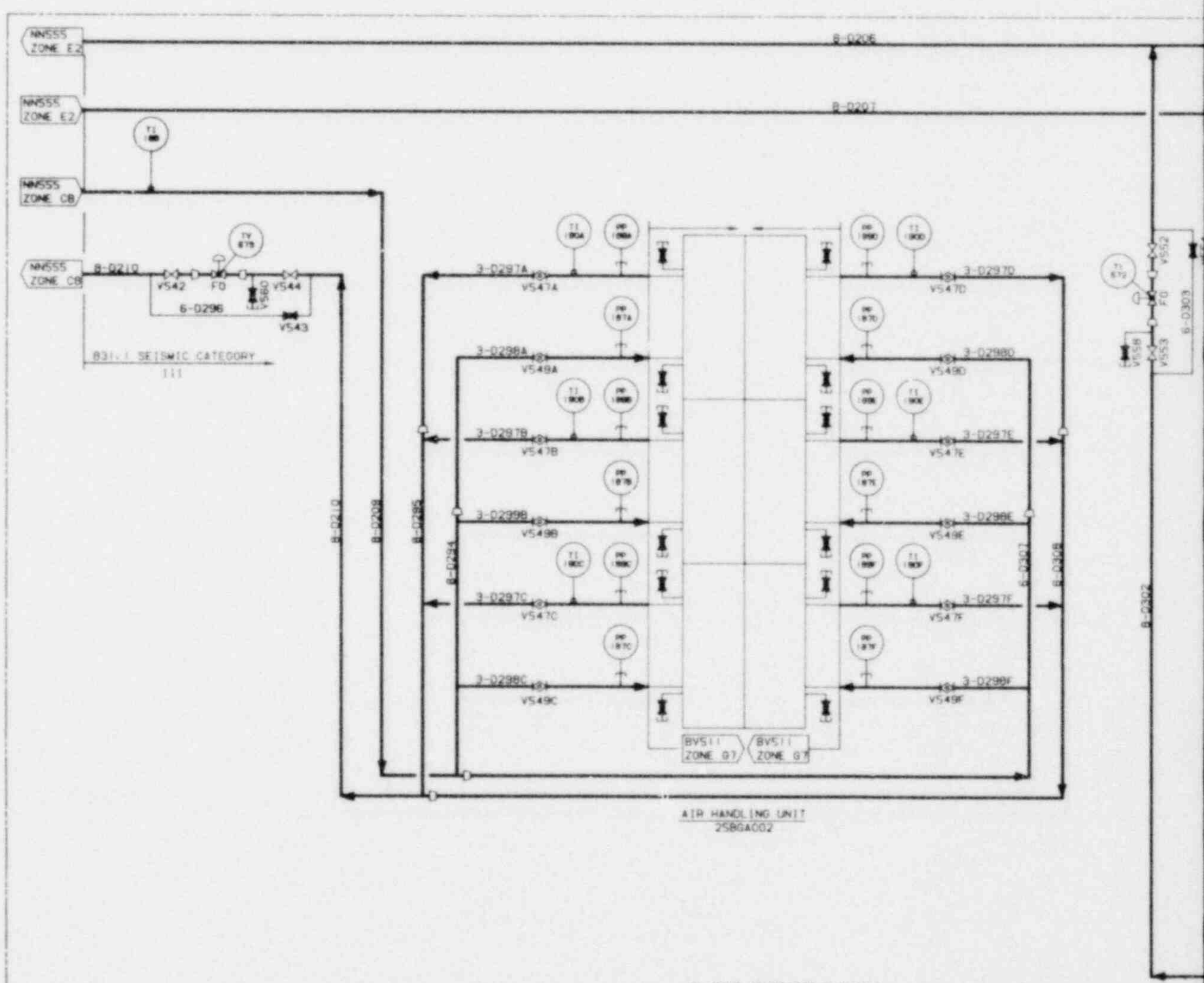
POOR ORIGINAL

REFERENCE DRAWINGS

FOR REFERENCE DWGS SEE NN551, NN552 & NN553

Figure 9.7-7 NORMAL CHILLED WATER SYSTEM RCB, SGB



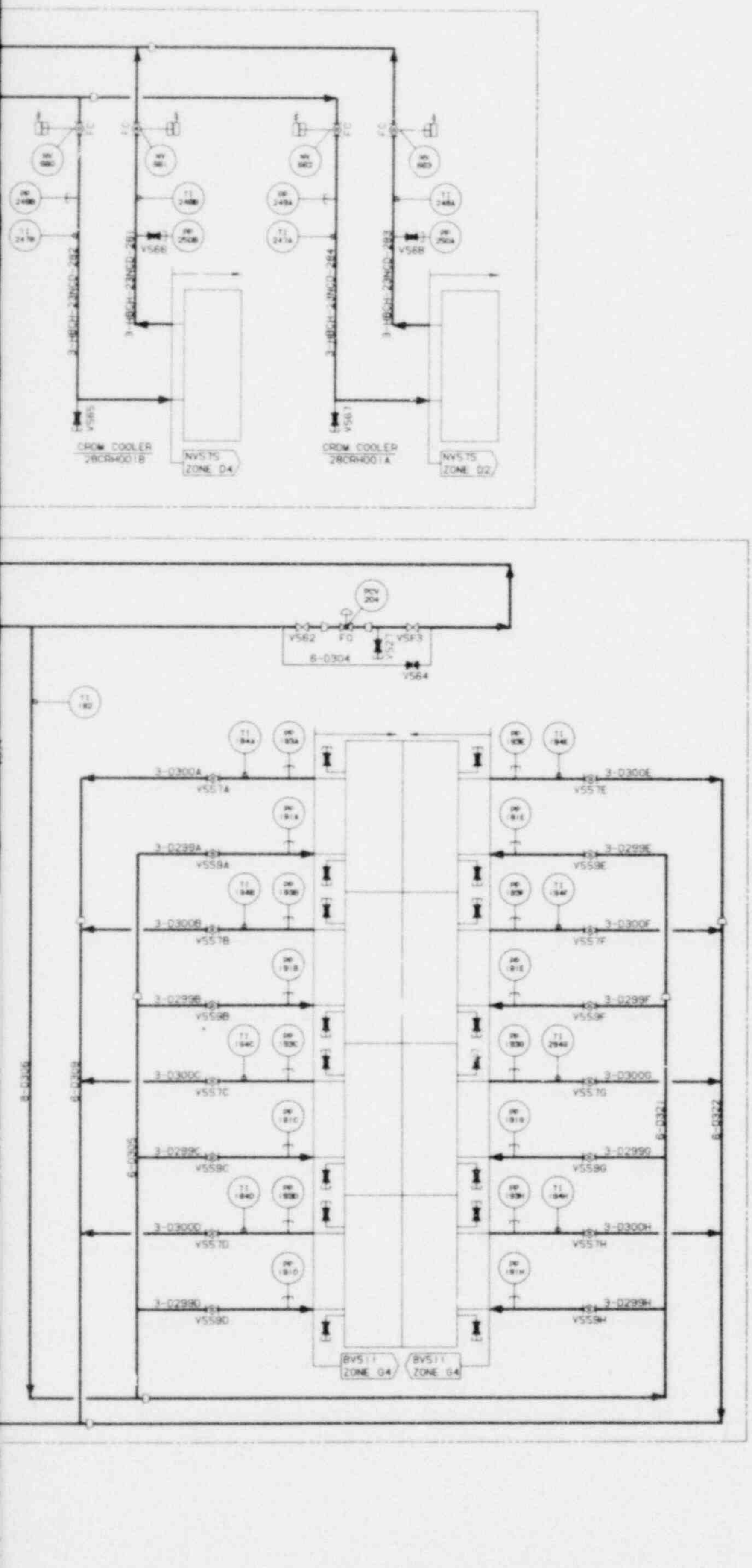


NN557-6

POOR ORIGINAL

NOTES

1. FOR GENERAL NOTES SE DWG NN551
2. SEISMIC CATEGORY: I & III
3. RADIATION ZONE:
4. CLEANLINESS CLASSIFICATION: CLASS C
5. CODE CLASSIFICATION: ANSI B31.1 & ASME SECT III/CLASS 3



POOR ORIGINAL

REFERENCE DRAWINGS

FOR REFERENCE DWGS SEE NN551, NN552 & NN553

Figure 9.7-8 NORMAL CHILLED WATER SYSTEM RCB & TGB

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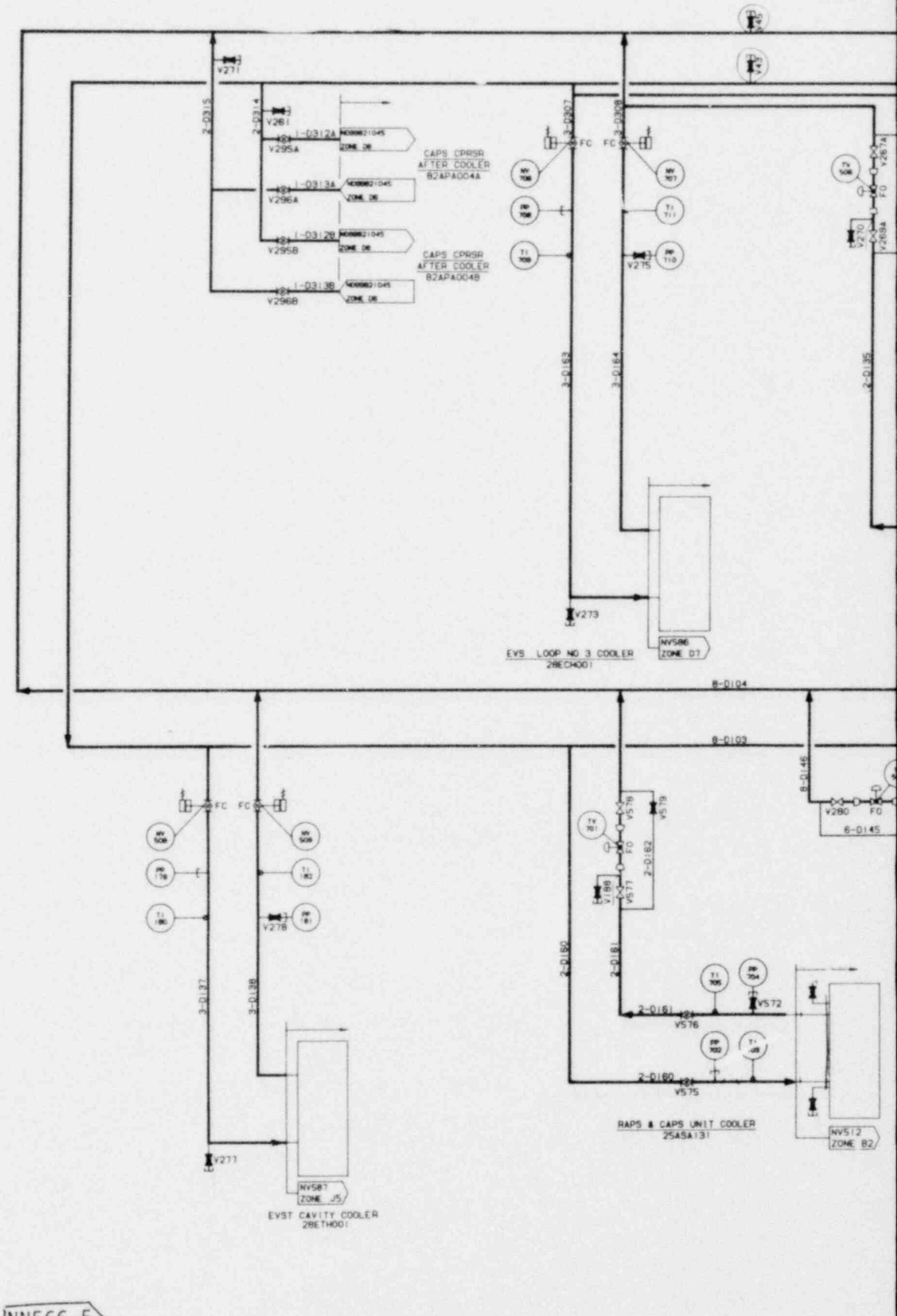
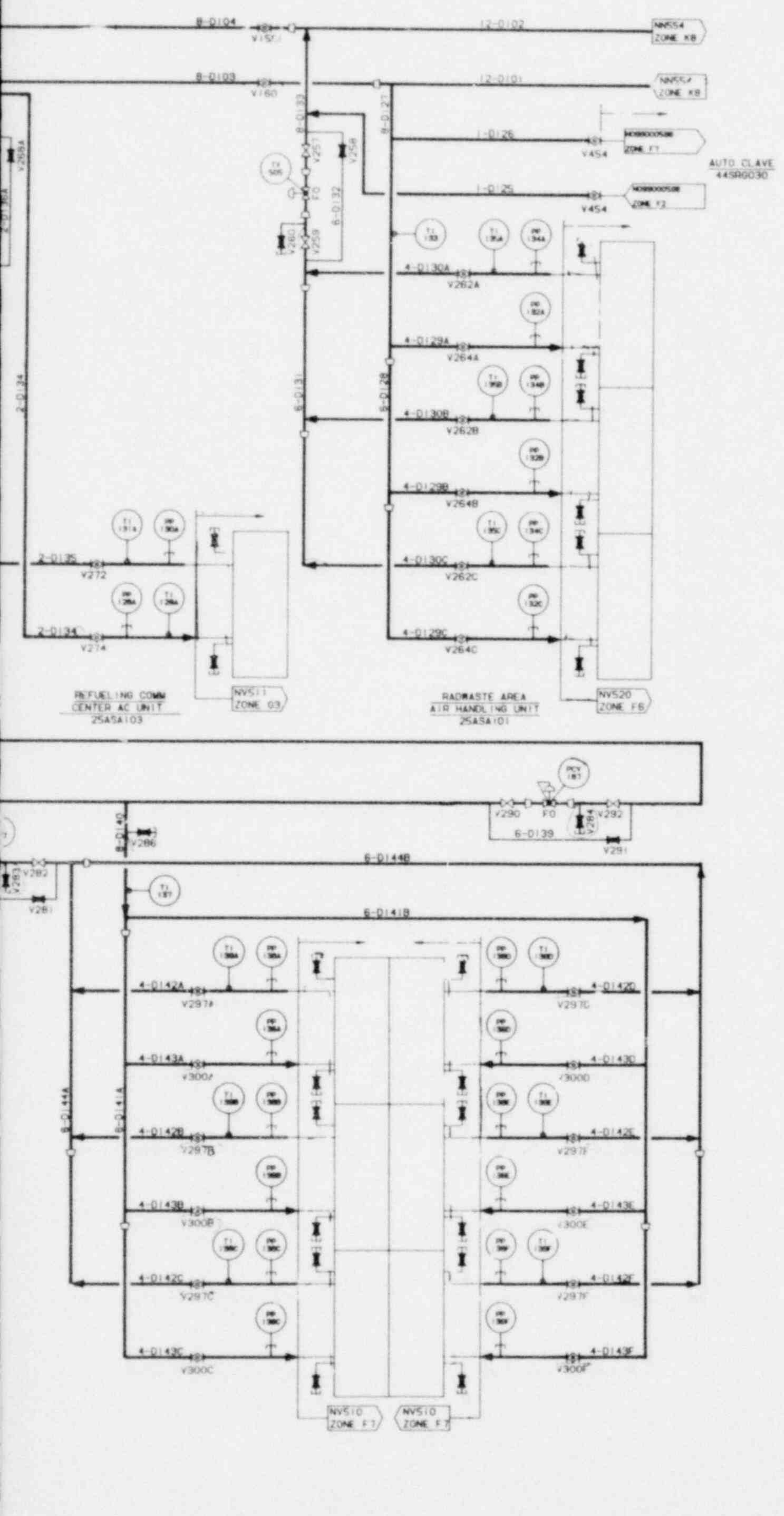


NN566-5

POOR ORIGINAL



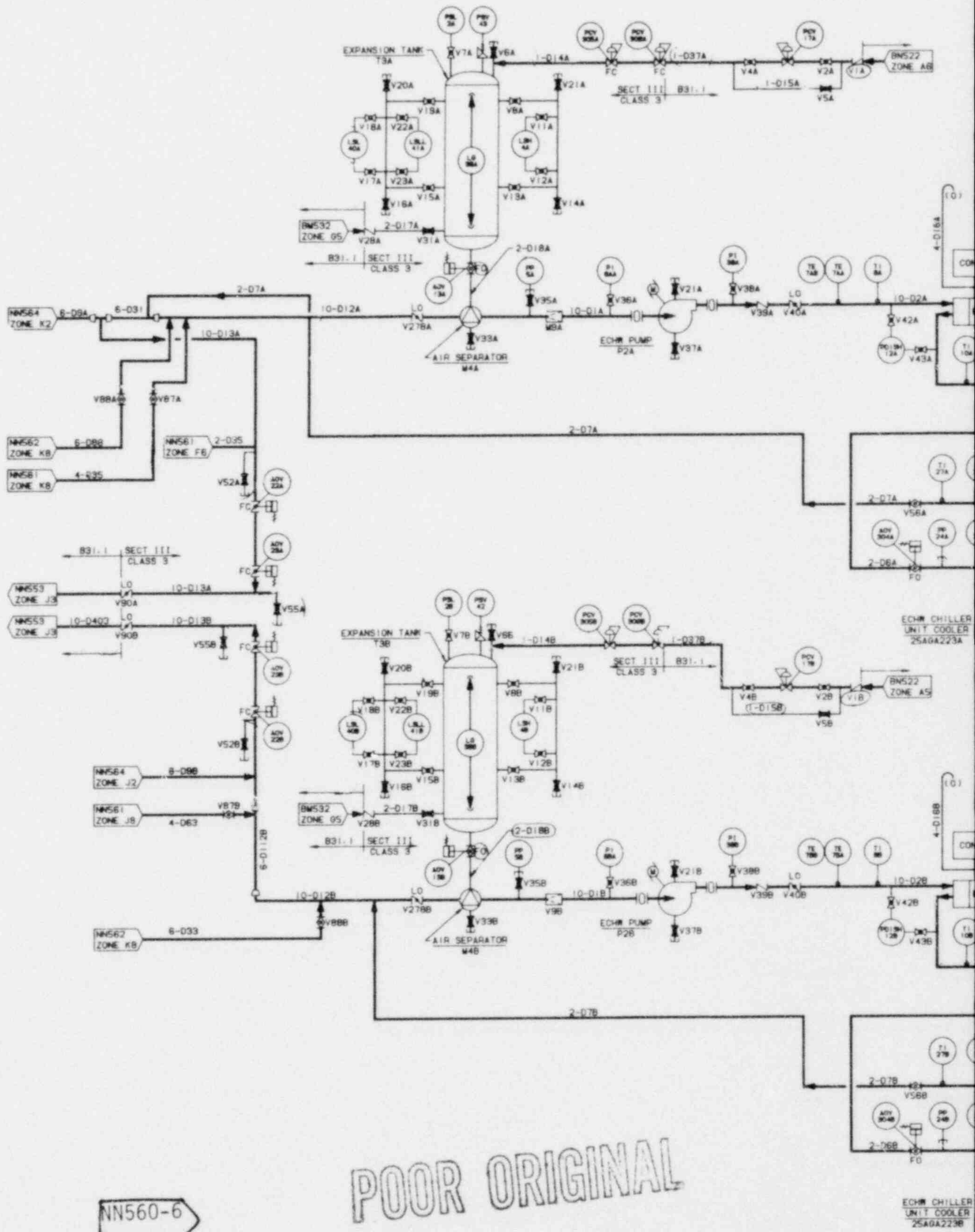
NOTES

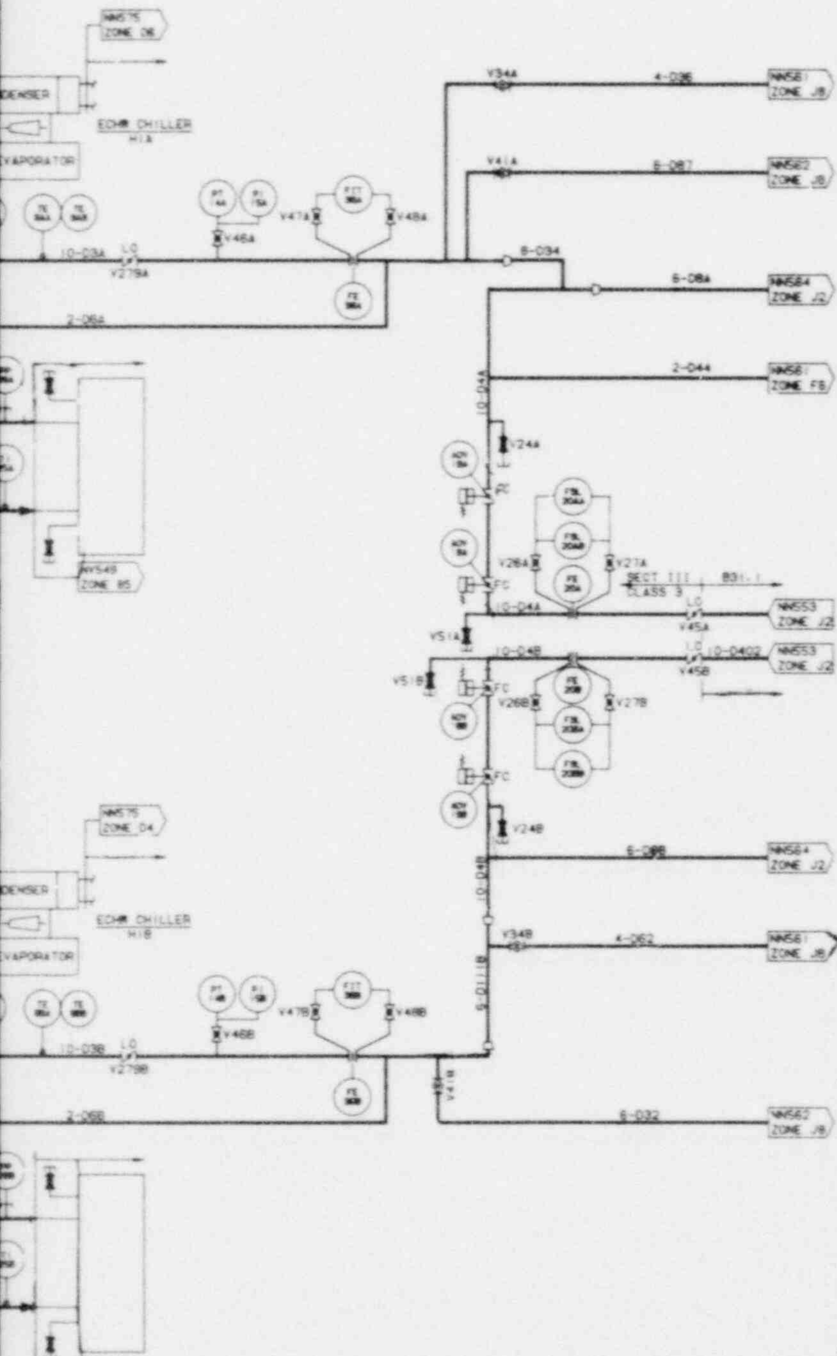
1. FOR GENERAL NOTES SEE DWG NN551
2. SEISMIC CATEGORY: III
3. RADIATION ZONE:
4. SYSTEM CLEANLINESS CLASSIFICATION: CLASS C
5. CODE CLASSIFICATION: ANSI B31.1

REFERENCE DRAWINGS

FOR REFERENCE DWGS SEE NN551, NN552 & NN553

Figure 9.7-9 NORMAL CHILLED WATER SYSTEM RSB


EXPANSION TANK
T3A
EXPANSION TANK
T3B
AIR SEPARATOR
M4A
AIR SEPARATOR
M4B
ECHW PUMP
P2A
ECHW PUMP
P2B
SECT III
CLASS 3
B31.1
BN522
ZONE A6
BN522
ZONE A5
BN532
ZONE G5
NN564
ZONE K2
NN562
ZONE K8
NN561
ZONE F6
NN561
ZONE K8
NN553
ZONE J3
NN564
ZONE J2
NN561
ZONE J8
NN562
ZONE K8
1-D14A
1-D37A
1-D15A
1-D14B
1-D37B
1-D15B
2-D17A
2-D18A
2-D17B
2-D18B
2-D7A
2-D7B
10-D12A
10-D12B
10-D1A
10-D1B
10-D2A
10-D2B
10-D13A
10-D13B
6-D9A
6-D31
6-D88
4-D35
2-D35
6-D98
4-D63
6-D33
6-D112B
4-D16A
4-D16B
ECHW CHILLER
UNIT COOLER
25AGA223A
ECHW CHILLER
UNIT COOLER
25AGA223B
NN560-6



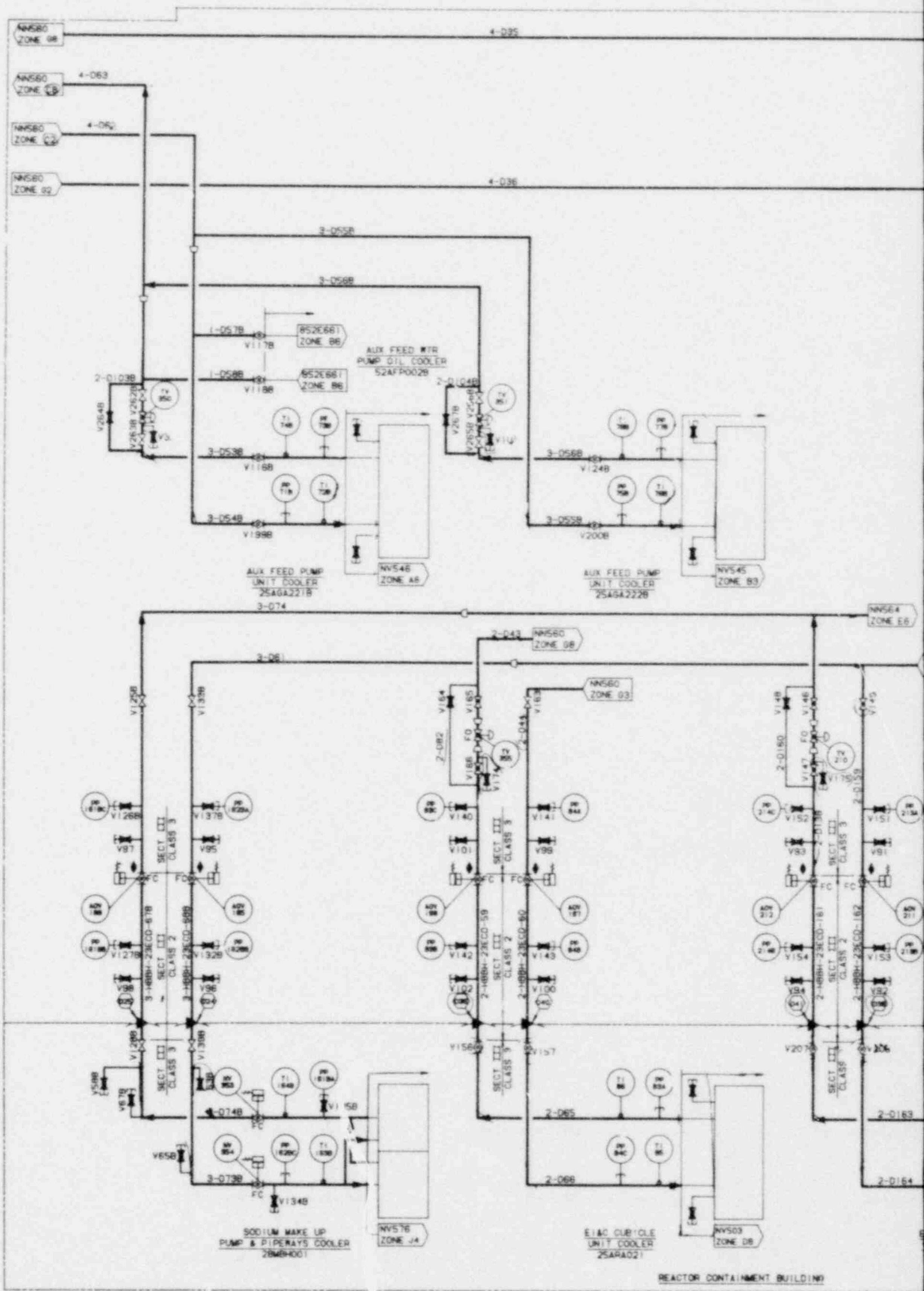
GENERAL NOTES

1. SYMBOLS AND ABBREVIATIONS WARD-D-0038
2. ALL EQUIPMENT NUMBERS AND MANUAL VALVE NUMBERS ARE PREFIXED BY 23EC UOS
3. ALL INSTRUMENT NUMBERS AND CONTROL VALVE NUMBERS ARE PREFIXED BY 23EC UOS
4. LINE NUMBERS ARE ABBREVIATED AS PER THE FOLLOWING EXAMPLE UOS: 8-HBCH-23ECD-1 IS WRITTEN AS 8-D1
5. SYSTEM CLEANLINESS CLASSIFICATION: CLASS C
6. SEISMIC CATEGORY: I UOS
7. RADIATION ZONE: UNRESTRICTED UOS
8. CODE CLASSIFICATION: ASME SECTION III/CLASS 3
9. ALL PRESSURE AND FLOW CONNECTIONS SHALL BE 0.75" UOS

REFERENCE DRAWINGS

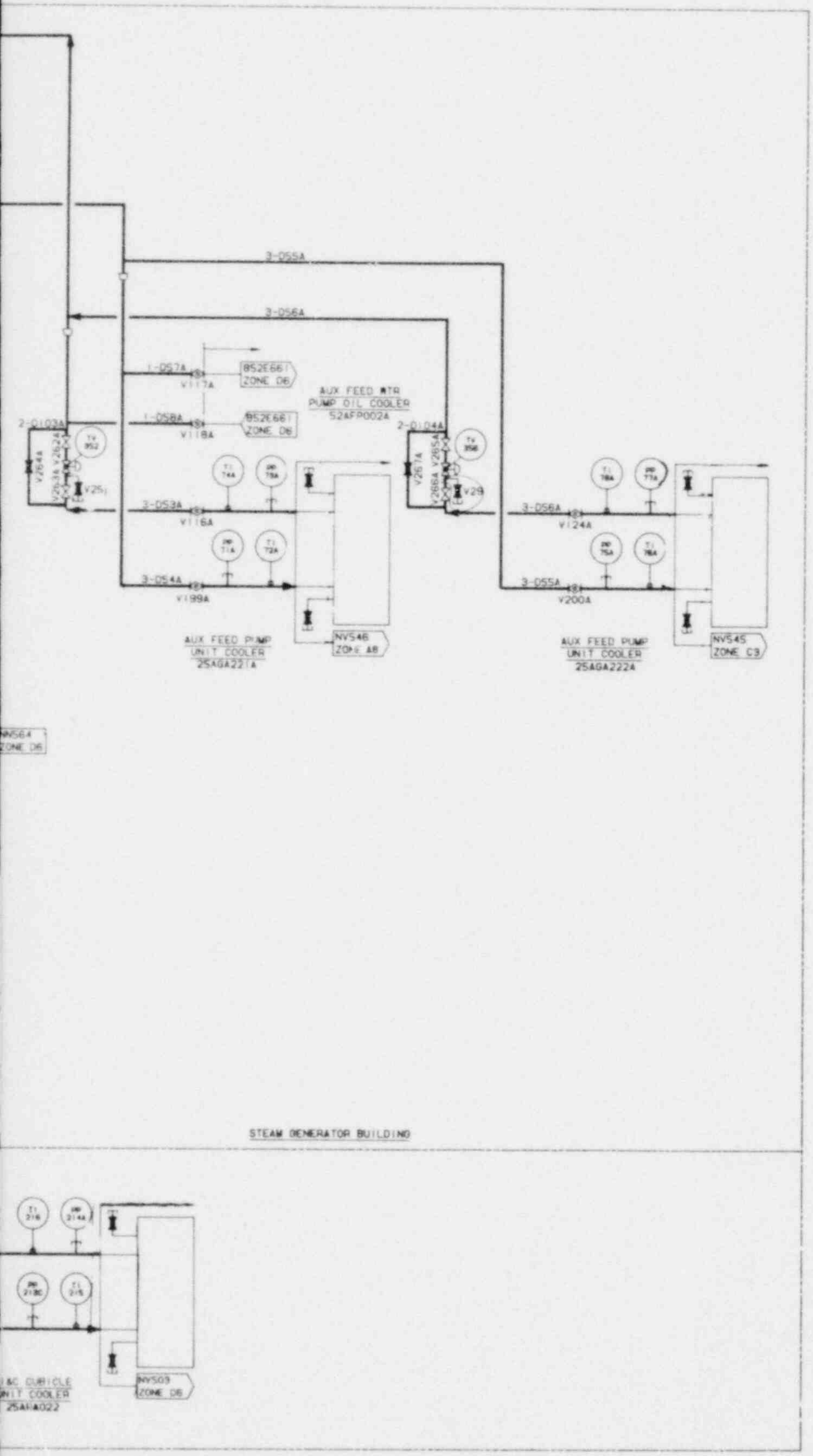
1. P&ID EMERGENCY CHILLED WATER SYS SGB & RCB — B&R DWG NW561
2. P&ID EMERGENCY CHILLED WATER SYS RCB, RSB & SGB — B&R DWG NW562
3. P&ID EMERGENCY CHILLED WATER SYS SEC COOLING LOOP RSB — B&R [illegible] NW563
4. P&ID EMERGENCY CHILLE[illegible]ER SYS CB & SGB — B[illegible]G NW564
5. P&ID EMERGENCY CHILLED WATER SYS RSB — B&R DWG NW565
6. P&ID NORMAL CHILLED WATER SYS SGB — B&R DWG NW553
7. P&ID STEAM GENERATOR AUXILIARY HEAT REMOVAL SYSTEM — GE DWG #52E861
8. P&ID RSB & RCB N_2 SUPPLY — AI DWG ND9982105I
9. P&ID DEMINERALIZED WATER SYS — B&R DWG BW532
10. P&ID RSB HVAC — B&R DWG NV510
11. P&ID RSB HVAC — B&R DWG NV511
12. P&ID CB CRDM & SMDR HVAC — B&R DWG NV529
13. P&ID CONTROL ROOM HVAC — B&R DWG NV530
14. P&ID SGB LOOP 1 & AUX BAY HVAC — B&R DWG NV545
15. P&ID SGB LOOP 2 & AUX BAY HVAC — B&R DWG NV546
16. P&ID SGB/IB HVAC — B&R DWG NV549
17. P&ID RCB RGCS — B&R DWG NV576
18. P&ID RSB RGCS — B&R DWG NV586
19. P&ID EMERGENCY PLANT SERVICE WATER SYS — B&R DWG NW575
20. P&ID INSTRUMENT AIR SYS SGB — B&R DWG BW522

Figure 9.7-10 EMERGENCY CHILLED WATER SGB



NN-561-4

POOR ORIGINAL



GENERAL NOTES

1. FOR GENERAL NOTES SEE DWG NN580
2. SEISMIC CATEGORY: I
3. RADIATION ZONE:
4. SYSTEM CLEANLINESS CLASSIFICATION: CLASS C
5. CODE CLASSIFICATION: ASME SECT III/CLASS 3 AND CLASS 2
6. CERTAIN EQUIPMENT SHOWN ON THIS DWG IS INCLUDED IN THE PLANT PROTECTION SYSTEM (PPS). THIS EQUIPMENT IS SPECIFICALLY IDENTIFIED ON THE DWG AS FOLLOWS: THE SYMBOL ♦ DESIGNATES PPS EQUIPMENT
7. THIS DWG INCLUDES EQUIPMENT IDENTIFIED AS PART OF THE PLANT PROTECTION SYSTEM (PPS). BEFORE MODIFYING OR MAINTAINING THE EQUIPMENT SO IDENTIFIED THE APPROVAL OF THE COGNIZANT PERSONNEL FOR THE PPS MUST BE OBTAINED
8. ALL EMERGENCY CHILLED WATER SYSTEM PIPING INSIDE THE RCB AND ABOVE THE OPERATING FLOOR SHALL BE ENCLOSED WITHIN GUARD PIPING OR SPRAY SHIELD ENCLOSURES
9. ALL REACTOR CONTAINMENT PENETRATION NUMBERS ARE PREFIXED BY R23M

POOR ORIGINAL

REFERENCE DRAWINGS

FOR REFERENCE DRAWINGS SEE NN580

Figure 9.7-11 EMERGENCY CHILLED WATER SYSTEM SGB, RCB

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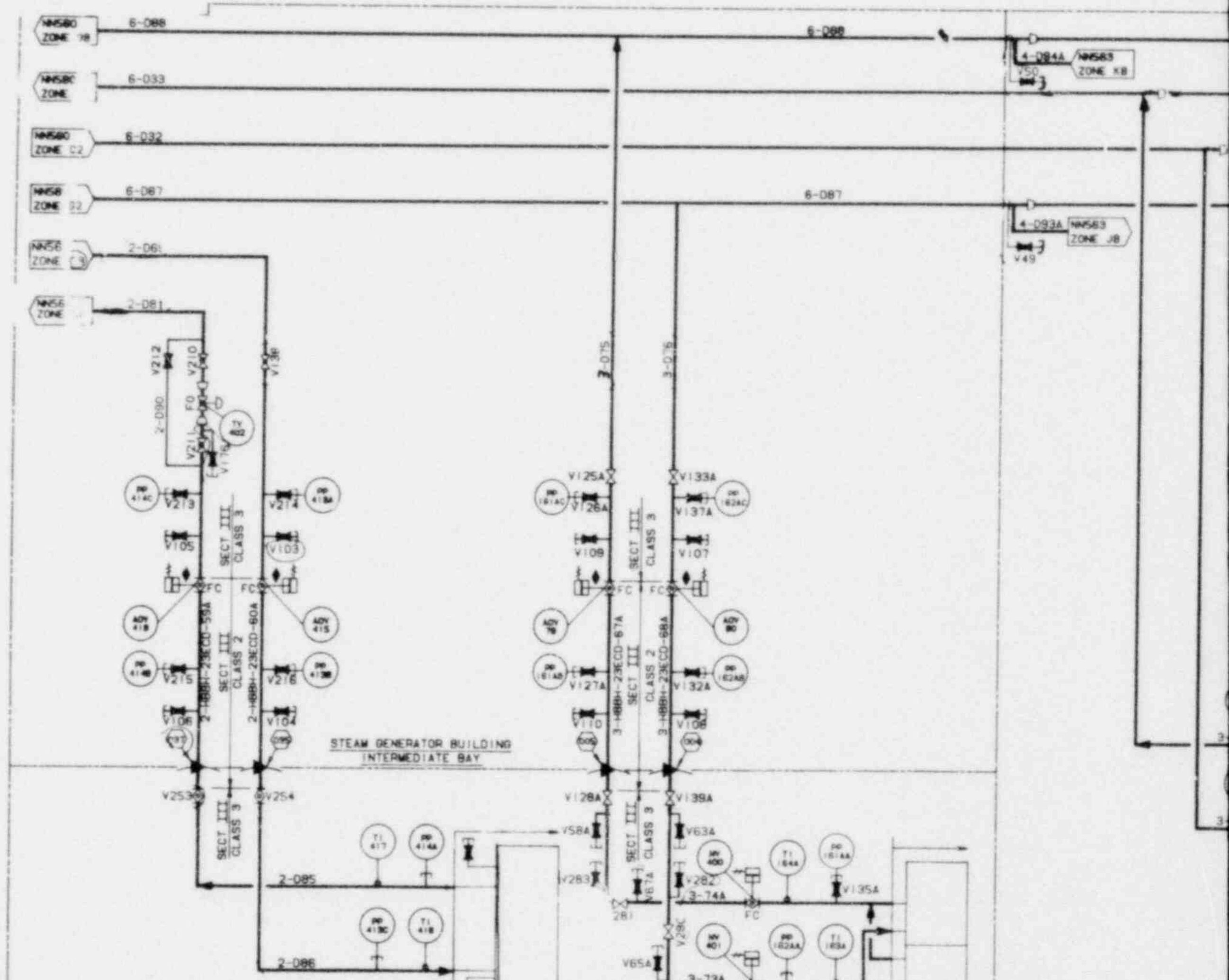
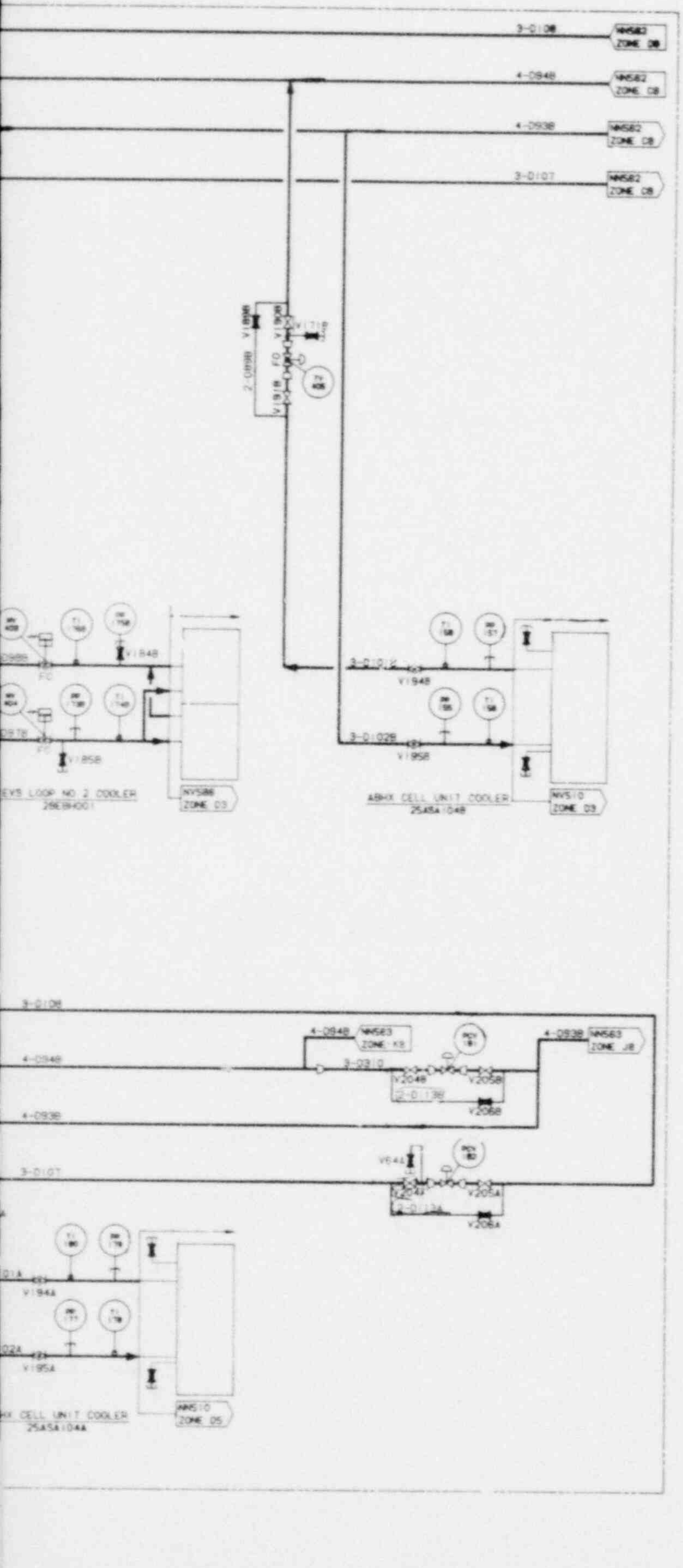

6-D88
6-D33
6-D32
6-D87
2-D61
2-D81
4-D94A
4-D93A
V49
3-D75
3-D76
V125A
V133A
V210
V212
V213
V214
V105
V103
V109
V107
V215
V216
V106
V104
V110
V108
V127A
V132A
V137A
V126A
STEAM GENERATOR BUILDING
INTERMEDIATE BAY
V253
V254
V128A
V139A
V58A
V63A
V283
V282
V65A
V135A
V134A
3-74A
3-73A
2-D85
2-D86
EI&C CUBICLE
UNIT COOLER
25ARA023
NV503
ZONE D4
SODIUM MAKE UP
PUMP & PIPEWAYS COOLER
28MAH001
NV576
ZONE J7
REACTOR CONTAINMENT BUILDING
NN582
ZONE K2
3-D108
4-D94B
4-D93B
3-D107
ZONE J2
V59A
V184A
V185A
3-D98A
3-D97A
V190A
V191A
V189A
2-D89A
EVS LOOP NO 1 COOLER
28EAH001
NV586
ZONE D5
REACTOR SERVICE BUILDING

NN562-6



GENERAL NOTES

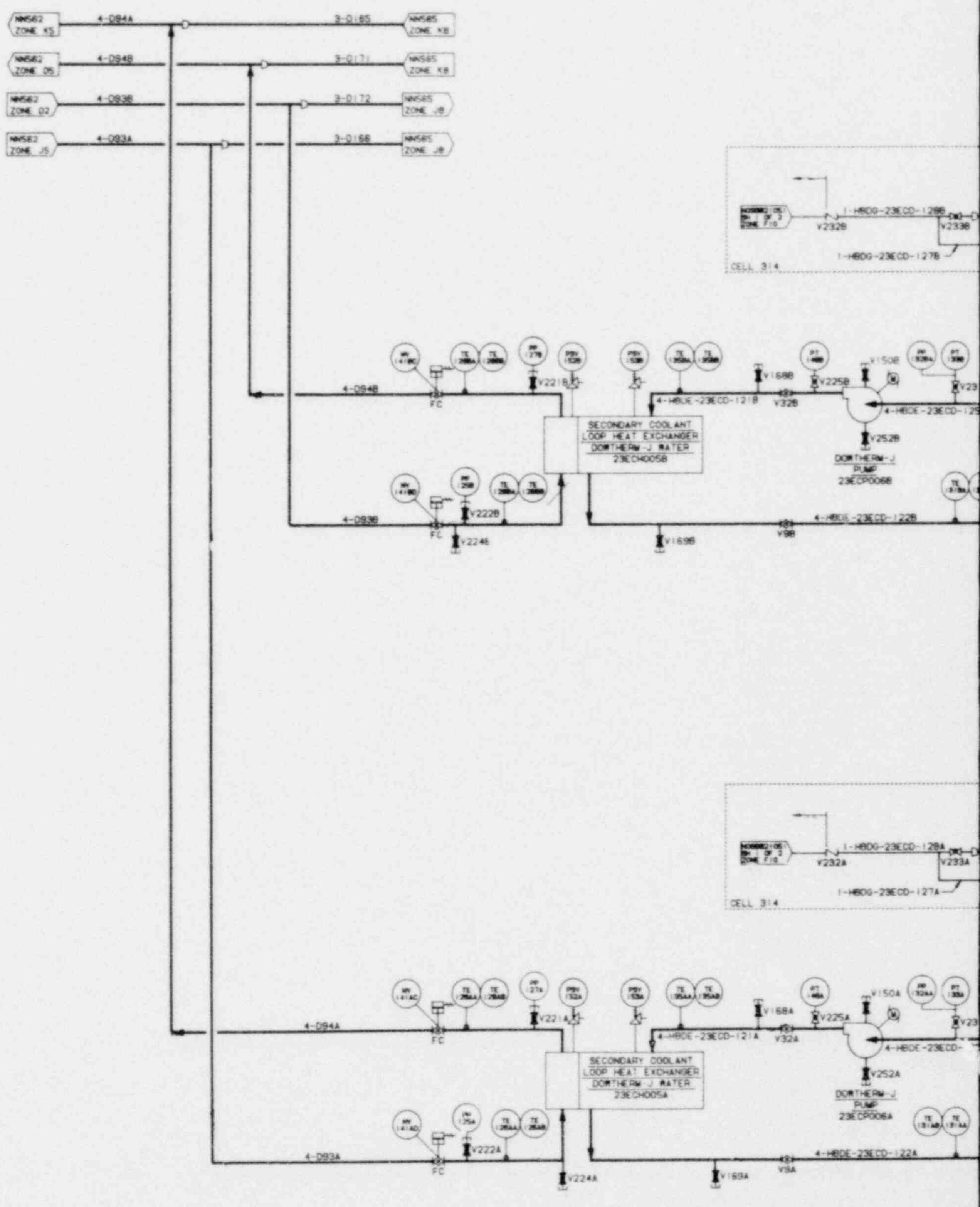
1. FOR GENERAL NOTES SEE DWG NN580
2. SEISMIC CATEGORY: I
3. RADIATION ZONE
4. SYSTEM CLEANLINESS CLASSIFICATION: CLASS C
5. CODE CLASSIFICATION: ASME SECT III/CLASS 3 AND CLASS 2
6. CERTAIN EQUIPMENT SHOWN ON THIS DWG IS INCLUDED IN THE PLANT PROTECTION SYSTEM (PPS). THIS EQUIPMENT IS SPECIFICALLY IDENTIFIED ON THE DWG AS FOLLOWS: THE SYMBOL ♦ DESIGNATES PPS EQUIPMENT
7. THIS DWG INCLUDES EQUIPMENT IDENTIFIED AS PART OF THE PLANT PROTECTION SYSTEM (PPS). BEFORE MODIFYING OR MAINTAINING THE EQUIPMENT SO IDENTIFIED THE APPROVAL OF THE COGNIZANT PERSONNEL FOR THE PPS MUST BE OBTAINED
8. ALL EMERGENCY CHILLED WATER SYSTEM PIPING INSIDE THE RCB AND ABOVE THE OPERATING FLOOR SHALL BE ENCLOSED WITHIN GUARD PIPING OR SPRAY SHIELD ENCLOSURES
9. ALL REACTOR CONTAINMENT PENETRATION NUMBERS ARE PREFIXED BY R23W

POOR ORIGINAL

REFERENCE DRAWINGS

FOR REFERENCE DRAWINGS SEE NN580

Figure 9.7-12 EMERGENCY CHILLED WATER SYSTEM RCB, RSB & SGB

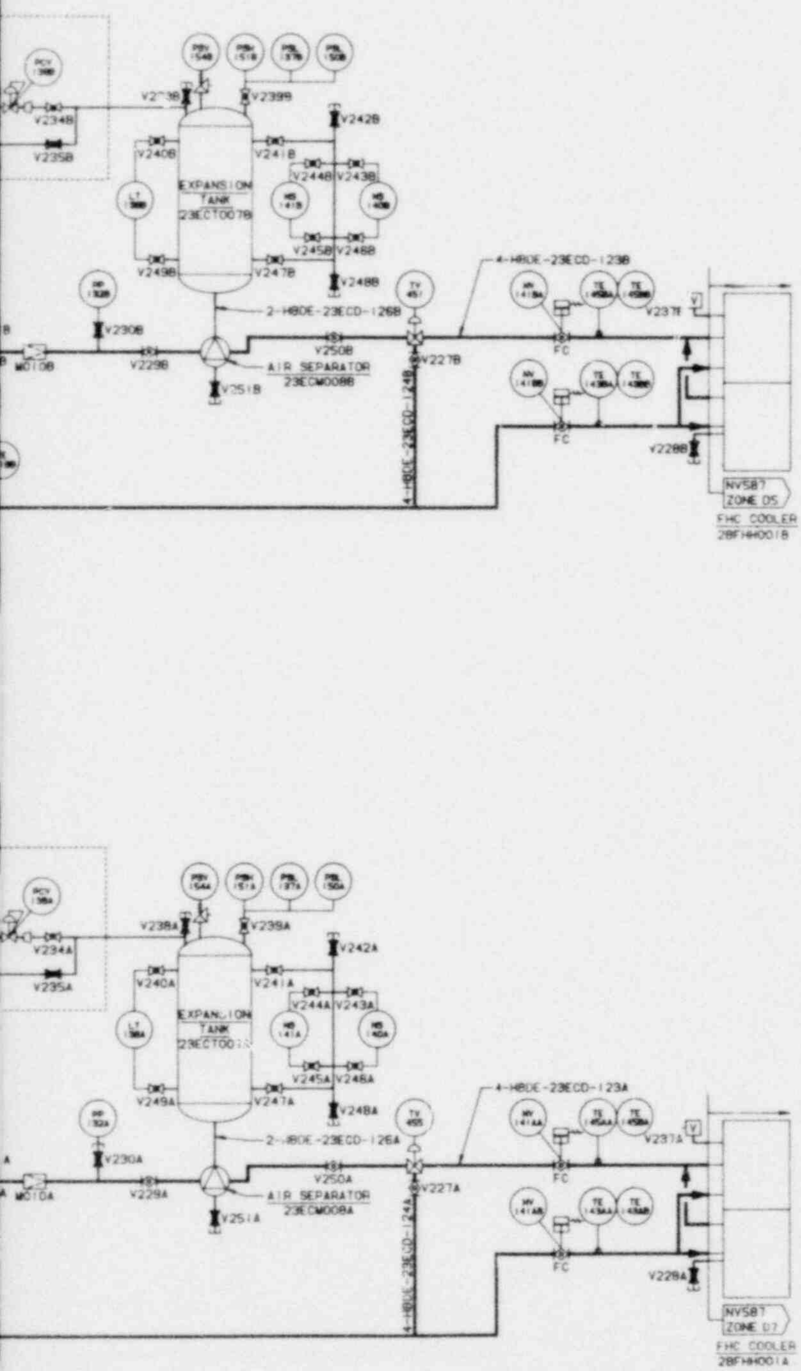


NN563-4

POOR ORIGINAL

GENERAL NOTES

1. FOR GENERAL NOTES SEE DWG NN560
2. SEISMIC CATEGORY: I
3. RADIATION ZONE:
4. SYSTEM CLEANLINESS CLASSIFICATION: CLASS C
5. CODE CLASSIFICATION: ASME SECT III/CLASS 3 & B31.1



REFERENCE DRAWINGS

FOR REFERENCE DRAWINGS SEE NN560

Figure 9.7-13 EMERGENCY CHILLED WATER SYSTEM SEC COOL LOOP RSB

Amend. 59
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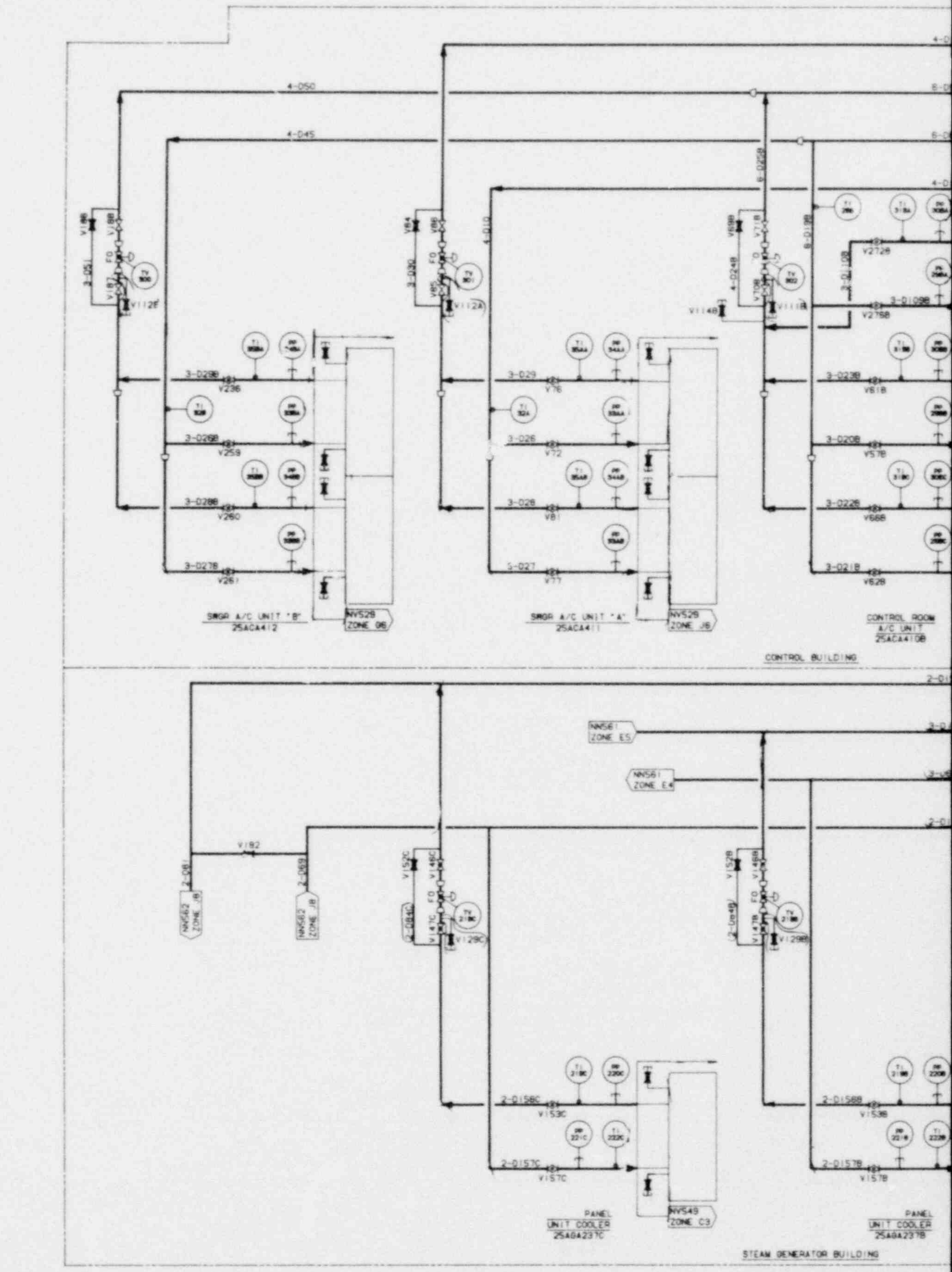
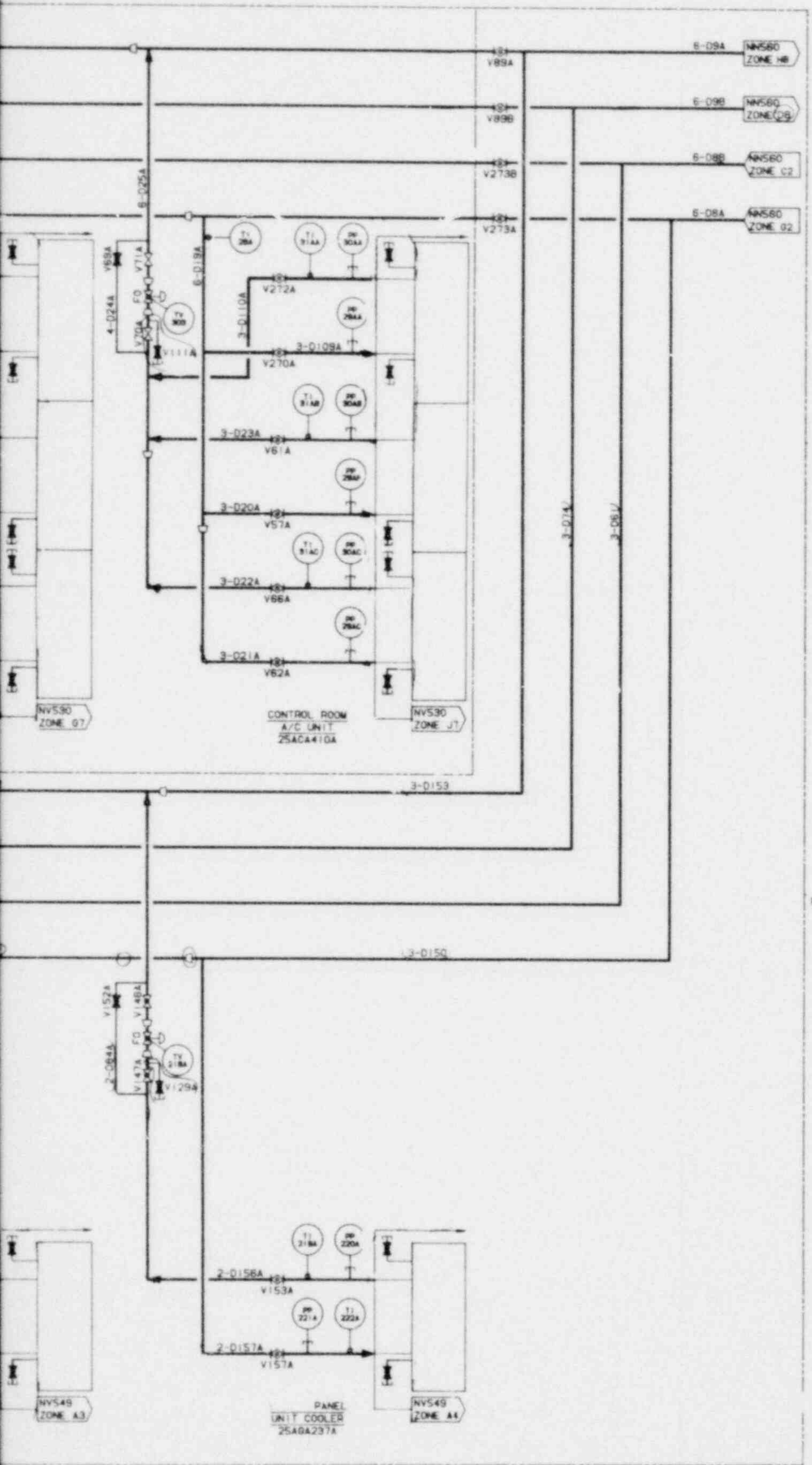


POOR ORIGINAL



NOTES

1. FOR GENERAL NOTES SEE DWG NN560
2. SEISMIC CATEGORY: I
3. RADIATION ZONE: UNRESTRICTED
4. SYSTEM CLEANLINESS CLASSIFICATION: CLASS C
5. CODE CLASSIFICATION: ASME SECT III/CLASS 3

POOR ORIGINAL

REFERENCE DRAWINGS

FOR REFERENCE DRAWINGS SEE NN560

Figure 9.7-14 EMERGENCY CHILLED WATER SYSTEM

Amend. 59
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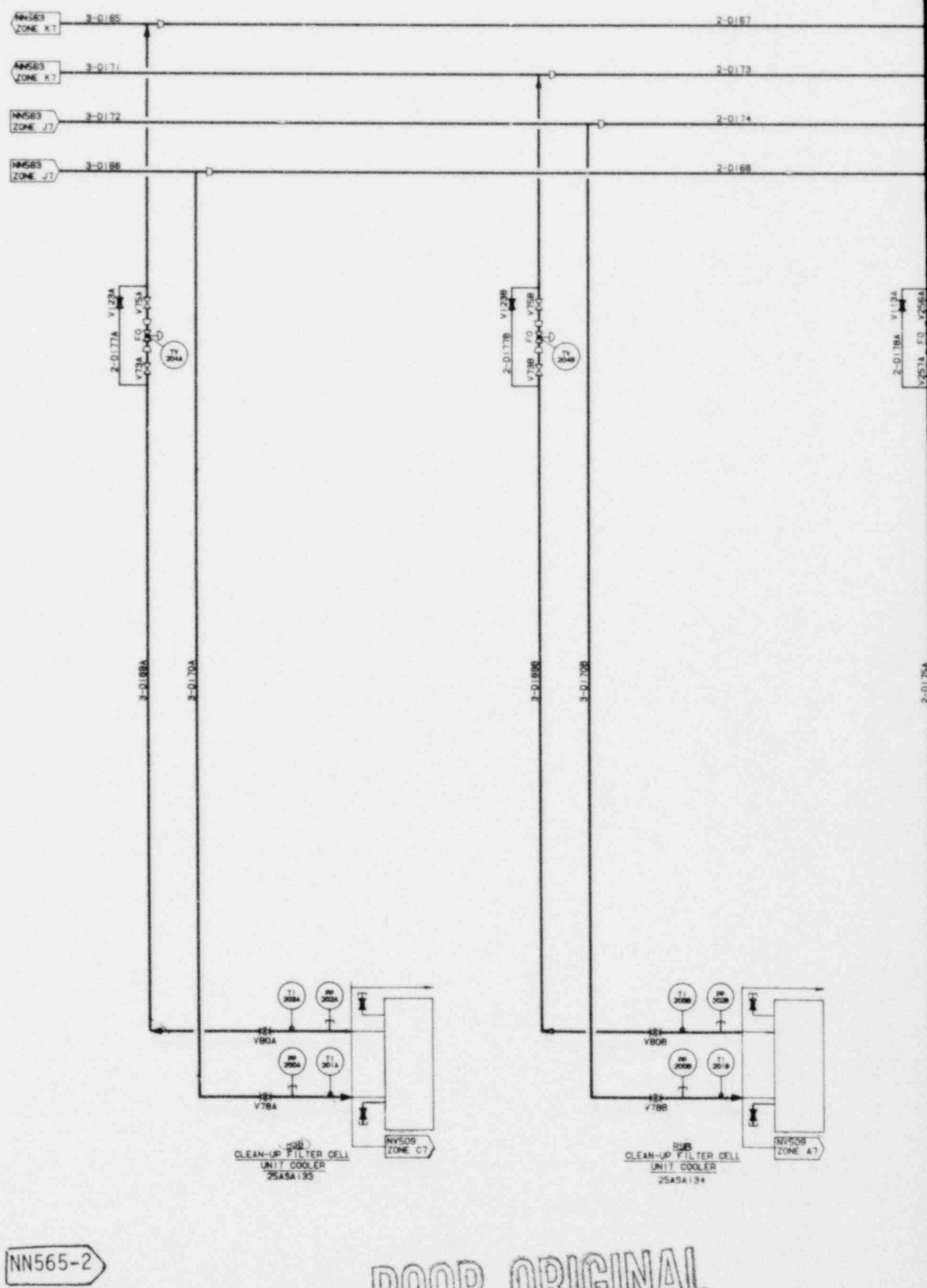
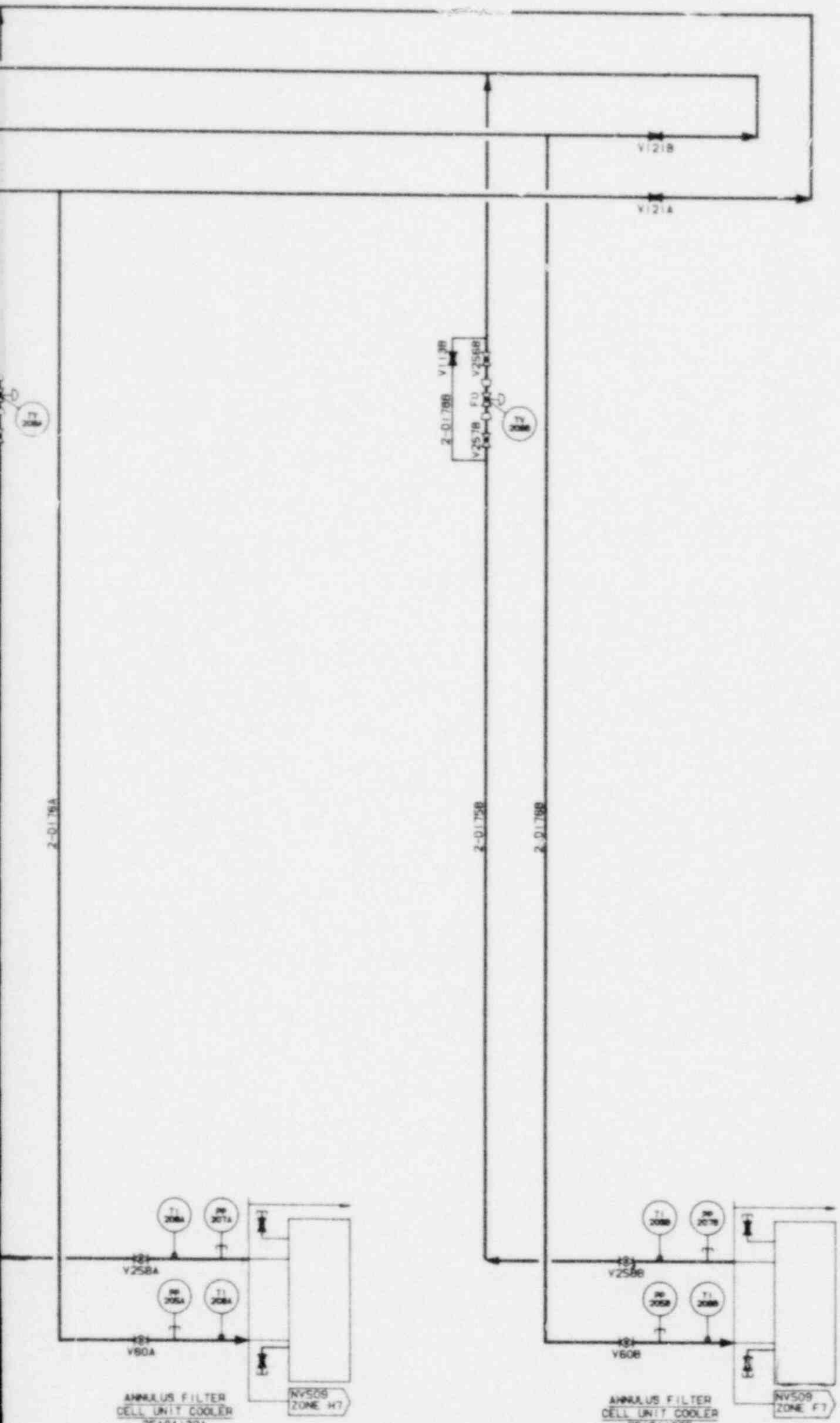


POOR ORIGINAL

GENERAL NOTES

1. FOR GENERAL NOTES SEE DWG NN580
2. SEISMIC CATEGORY: I
3. RADIATION ZONE
4. SYSTEM CLEANLINESS CLASSIFICATION: CLASS C
5. CODE CLASSIFICATION: ASME SECT III/CLASS 3



POOR ORIGINAL

REFERENCE DRAWINGS

FOR REFERENCE DRAWINGS SEE NN580

Figure 9.7-15 EMERGENCY CHILLED WATER SYSTEM RSB

Amend. 59
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9.9 SERVICE 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
59 43 15 9.9-1. The system consists of two (approximately 26,600 GPM) 100 percent
capacity electric motor driven vertical, wet-pit, circulating water
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
43 the cooling tower return header. The pumps are located in the Circulating
Water Pumphouse. Normally, one pump is operating with the second pump
33 in an 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

15 #### 9.9.1.3 Safety Evaluation

The Normal Plant Service Water System is a nonseismic, non-safety class system.

59 15

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
59 is provided in the Control Room. Normal Plant Service Water low 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.

9.9.2 Emergency Plant Service Water System

9.9.2.1 Design Basis

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

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

9.9.2.2 System Description

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

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

Upon loss of Normal Chilled Water or upon start of the Standby Diesel Generators, the EPSW pumps, EPSW makeup pumps, and Cooling Tower
59 Fans will automatically start and provide cooling water at 90^{o}F maximum

to the Emergency Chiller Condensers in the SGB and the Standby Diesel Generators in the DGB. The EPSW pumps take suction from the Emergency Cooling Tower operating basins which are located adjacent to the Emergency Cooling Tower. During system operation the EPSW makeup pumps
59 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^{o}F, the minimum required for chiller operation.

In addition to cooling the Emergency Chilled Water chillers and the standby Diesel Generators, each loop of the EPSW System provides a
59 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
59 pumps are provided with instrumentation that will automatically terminate operation when a prescribed amount of water has been used (see Section 9.13). This ensures that the guaranteed 30 day supply of water for EPSW system operation will not be compromised. In addition, this system is connected to the EPSW loops in such a manner as to preclude a single failure from compromising the capability of the EPSW system to perform its required function.

9.9.2.3 Safety Evaluation

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

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

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

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

15 9.9.2.4 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 on the supply and return headers in the pumphouses and the DGB permit
50 inservice inspection of the buried piping by hydrostatic testing.

9.9.2.5 Instrumentation Application

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

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

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

15 9.9.3 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
designed to provide adequate cooling water supply for power generation
equipment and auxiliaries during startup, normal operation, and normal
59 plant shutdown.

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

15 9.9.3.2 System Description

59 The SSCCW System is shown in Figure 9.9-3 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 which 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, accommo-
dates 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 demineralized water system to the surge tank outlet line.
59 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 demineralized water line to maintain the
surge tank at the desired level. The surge tank is readily accessible
59 during operation for manual level adjustment if desired.

A butterfly valve is installed in a bypass line around the SSCCW
heat exchangers to regulate the bypass flow thereby providing a tempering
59 effect to maintain a constant 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
high and low pressure and alarmed in the Control Room. Pressure indicators
are provided at each pump discharge. Pressure test connections are provided
43 at each heat exchanger outlet. Temperature indication is located on
59 each heat exchanger outlet and a temperature test connection is located
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

59 The Emergency Cooling Towers (Figure 9.9-2) 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 the Emergency Cooling Tower
Storage Basin are designed according to the applicable requirements of
Regulatory Guide 1.27. The integral piping associated with the cooling
33 towers is 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.

The storage capacity of the Emergency Cooling Tower Storage
50 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 to be the first day of the worst 30 day period.

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.
59 50 One loop is provided with electrical power from System Class 1E Division 1
and the other from System Class 1E Division 2.

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

The Emergency Cooling Tower Structure consists of two 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
59 30,000 gallons of water storage for the non-sodium Fire Protection System
plus adequate allowance for drift and evaporation losses. Each cooling
tower is designed to achieve the required heat dissipation rate at any
time. This is approximately 2.36×10^7 BTU/HR at the maximum Emergency
50 Plant 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 compen-
sation 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 purchased
59 under agreements with the Department of Energy, Oak Ridge Operations.

The top elevation of the Emergency Cooling Tower Basin is
818 ft. which is 9 ft. above the probable maximum flood level. The
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, pro-
vided 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 dis-
charge 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 a counter-flow, wet-type,
mechanically induced draft design. The internal distribution piping distributes
the intake water evenly over the fill area so that sufficient water area
is exposed to the counter air flow to provide evaporation for the
59 required heat removal. The counter air flow is provided by the induced
draft fans.

Drift eliminators are located above the internal water distri-
bution piping and before 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
59 50 concrete storage basin. The top of the cooling towers is approximately
44 ft. above the maximum water level of the storage basin.

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

50 The Emergency Cooling Tower structure consists of two 100
percent capacity cooling towers pumphouses, and operating basins and
one 100 percent capacity below grade cooling water storage basin. The
entire structure is Seismic Category I, tornado, and flood protected.
Piping, associated with the Emergency Cooling Tower is designed to ASME
Section III, Class 3 requirements. The structure can withstand the most
severe natural phenomena expected, and other site related events, such
that the Emergency Cooling Tower cooling capability is assured under
required conditions. The method of analysis is similar to that used
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.

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

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

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

All materials used for the Emergency Cooling Tower Structure
43 are designed to be non-flammable in order to negate the possibility of loss of the cooling function due to fire.
50

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

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

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

    a. Meteorological Condition for Minimum Heat Removal Rate.

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

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

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 a design basis accident. For the following 29 days, the worst month of record is assumed.

(2) $90^{0}F$ 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.

59 50 (5) 30,000 gallons of water reserved for fire protection use is not considered available for cooling (see Section 9.13).

3. The maximum water usage based on the above assumptions will be calculated by a computer program that models 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 <u>Test and Inspection</u>

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 diesel generators, emergency chilled water chiller, cooling water supply temperature readouts.

59

43 33 c. 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
59 43 the Main Cooling Tower 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

59 The RWS consists of an intake structure 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.

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

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

15 9.9.5.3 <u>Safety Evaluation</u>

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

15 9.9.5.4 <u>Tests and Inspections</u>

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 <u>Instrumentation Application</u>

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.

15

TABLE 9.9-1

COMPONENTS SERVED BY NORMAL PLANT SERVICE WATER SYSTEM

	COMPONENT	NUMBER OF COMPONENTS	COMPONENT LOCATION BUILDING
15	NCHW Chillers	6	Steam Generator Building - IB
	SSCCW Heat Exchangers	2	Turbine Generator Building
43	IALL & LALL Evaporators	4	Reactor Service Building - Radwaste Area
	Primary & Intermediate Na Pump Drive Motor Generator Set Lube Oil Coolers	4	Control Building
	NI HVAC Motor Generator Set Unit Coolers	4	Control Building
	Primary & Intermediate Na Pump Drive Motor Generator Set Lube Oil Coolers	2	Diesel Generator Building
33	NI HVAC Motor Generator Set Unit Coolers	2	Diesel Generator Building
59	Auxiliary Steam System Blowdown Tank	1	Turbine Generator Building

15 NOTE: NCHW - Normal Chilled Water (9.7-1)
SSCCW - Secondary Service Closed Cooling Water (Figure 9.9-5)
33 IALL - Intermediate Activity Level Liquid
LALL - Low Activity Level Liquid

15 TABLE 9.9-2

NORMAL PLANT SERVICE WATER SYSTEM MAJOR COMPONENTS

	DESCRIPTION	QUANTITY	APPROX. NPSW FLOW FOR EACH COMPONENT
59 15	Normal Plant Service Water Pump	2	26,600 GPM
43			
50			

59 15 NOTE: NPSW - Normal Plant Service Water (Figure 9.9-1)

TABLE 9.9-3

COMPONENTS SERVED BY EMERGENCY PLANT SERVICE WATER SYSTEM

Component	Component Location			Component Service Requirements		
	Bldg.	Cell	Elev.	Flow GPM	*EWT °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 Max.	10.5
Emergency Chilled Water System Chiller B	SGB	217	733'-0"	2100	90^0 Max.	10.5

*Entering Water Temp.

15 TABLE 9.9-4

EMERGENCY PLANT SERVICE WATER SYSTEM MAJOR COMPONENTS

	Description	Quantity	Design Data For Each Component
59	Emergency Plant Service Water System circulating pump	2	3800 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

43 33

TABLE 9.9-5

EQUIPMENT COOLED BY THE SECONDARY SERVICE CLOSED COOLING WATER SYSTEM

Equipment
Generator Hydrogen Coolers
Generator Stator Coolers
Alternator Exciter Coolers
Isophase Bus Coolers
EHC Hydraulic Fluid Coolers
Turbine Lube Oil Coolers
Steam Generator Feed Pump Coolers
Service and Instrument Air Compressors
Condensate Pump Bearing Oil Coolers
Secondary Sampling Panel
Secondary Plant Sampling Roughing Coolers
43 59 Auxiliary Steam Secondary Plant Sampling Roughing Cooler

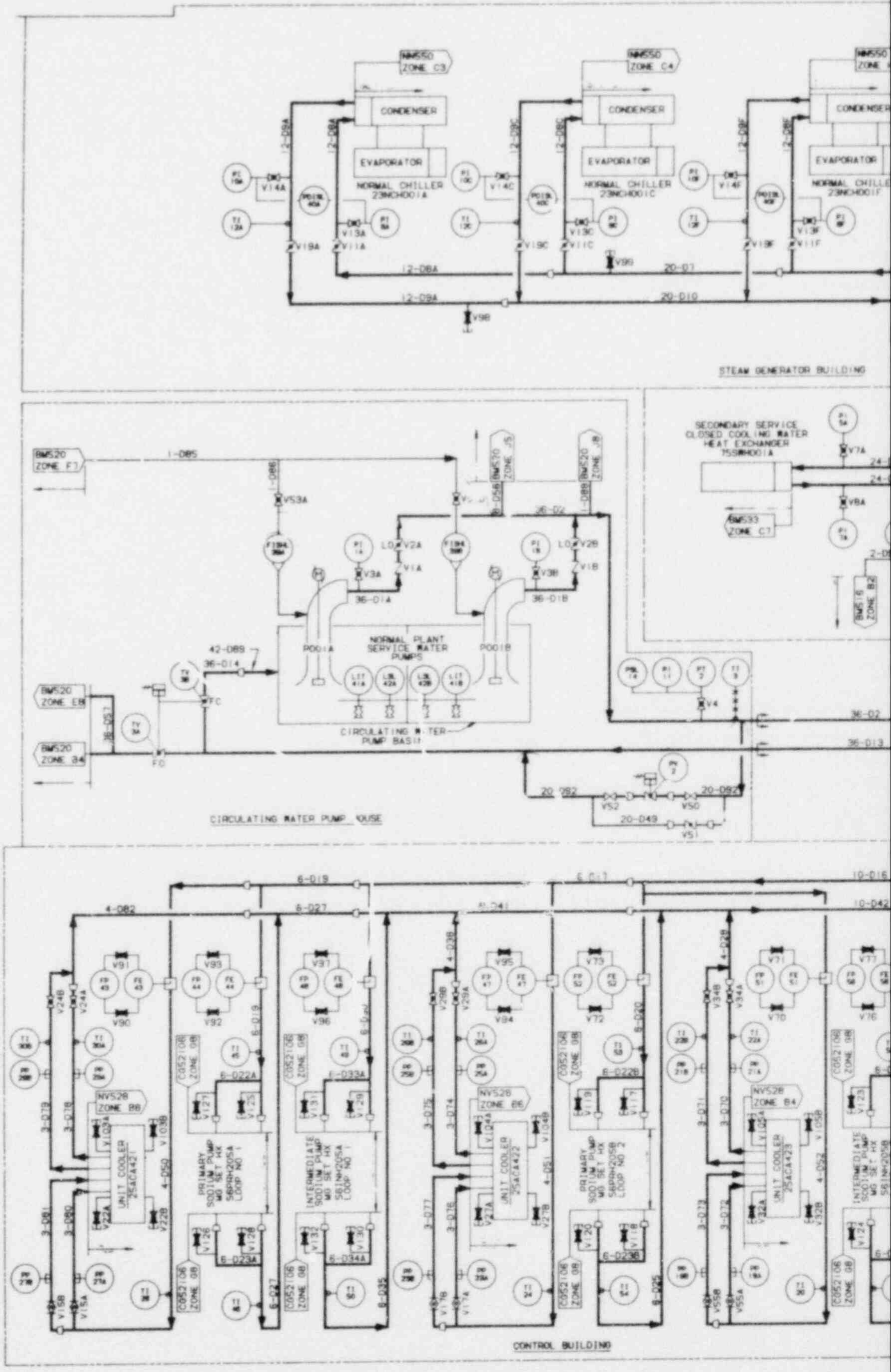
TABLE 9.9-6

SINGLE FAILURE ANALYSIS - EMERGENCY

PLANT SERVICE WATER SYSTEM

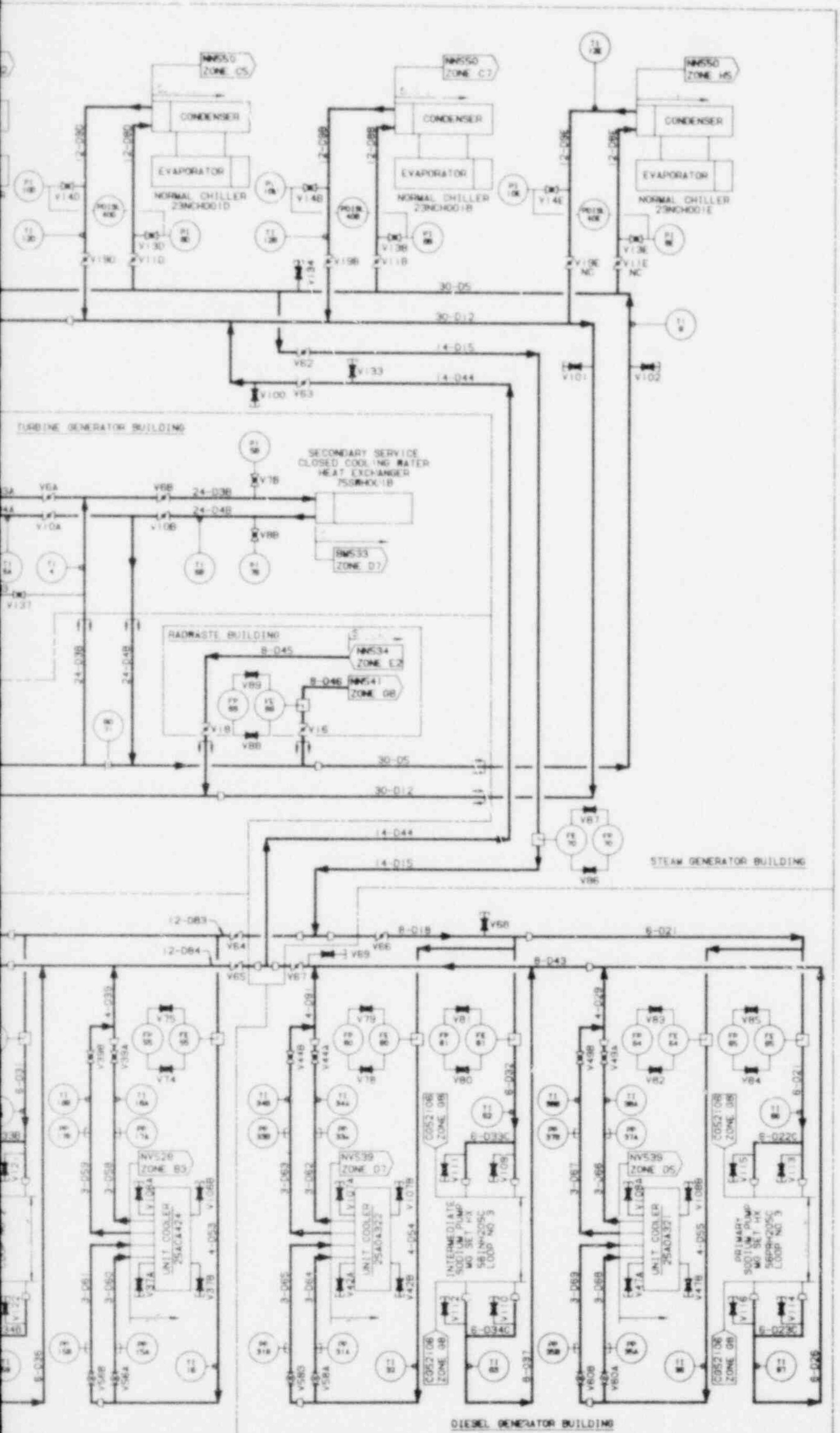
COMPONENT	MALFUNCTION	ANALYSIS
Cooling Towers	One cooling tower fails to start upon activation to emergency mode.	No effect on system availability. The cooling tower malfunction is alarmed to the control room. Each of the redundant trains has sufficient capacity to handle the load.
	Loss of cooling tower in operating train, emergency mode.	No effect on system availability. The cooling tower malfunction is alarmed in the control room. Each of the redundant trains has sufficient capacity to handle the load.
Pumps	Failure to start upon activation to emergency mode.	No effect on system availability. Low flow is alarmed in the control room. Each of the redundant trains has sufficient capacity to handle the load.
	Loss of pump in operating train, emergency mode.	No effect on system availability. Low flow is alarmed in the control room. Each of the redundant trains has sufficient capacity to handle the load.
Electrical Power Supplies	Loss of power from normal AC distribution system (Plant, preferred offsite AC and reserve offsite AC power supplies).	No effect on system availability. Power to both the trains is automatically supplied by the onsite AC power supply (diesel generators).
	Loss of one onsite diesel generator, emergency mode.	No effect on system availability. Each of the redundant trains has sufficient capacity to handle the load.

59



NN570-9

POOR ORIGINAL



GENERAL NOTES

1. SYMBOLS AND ABBREVIATIONS KARD-D-0036
2. ALL EQUIPMENT NUMBERS AND MANUAL VALVE NUMBERS ARE PREFIXED BY 75NP UOS
3. MOTOR NUMBERS ARE IDENTICAL TO THE DRIVEN EQUIPMENT NUMBER EXCEPT THE EQUIPMENT TYPE CODE IS REPLACED BY "K"
4. ALL INSTRUMENT NUMBERS AND CONTROL VALVE NUMBERS ARE PREFIXED BY 75NP UOS
5. LINE NUMBERS ARE ABBREVIATED AS PER THE FOLLOWING EXAMPLE UOS: 36-HBDJ-75NP-D1 IS WRITTEN AS 36-D1
6. SYSTEM CLEANLINESS CLASSIFICATION: MANUFACTURERS STANDARD
7. SEISMIC CATEGORY: III
8. RADIATION ZONE: RWB, II, III & V ALL OTHER AREAS, UNRESTRICTED
9. SAFETY CLASSIFICATION: NONE
10. APPLICABLE CODES ARE ANSI B31.1 FOR PIPING SHOWN ON THIS DRAWING
11. ALL PRESSURE, FLOW AND SECONDARY SAMPLING CONNECTIONS SHALL BE 0.75" UOS
12. ALL VENTS AND DRAINS SHALL BE 0.75" UOS
13. PERFORMANCE TEST CONNECTIONS IN ACCORDANCE WITH REQUIREMENTS OF ASME POWER TEST CODE PTC-6 SHALL BE ADDED LATER

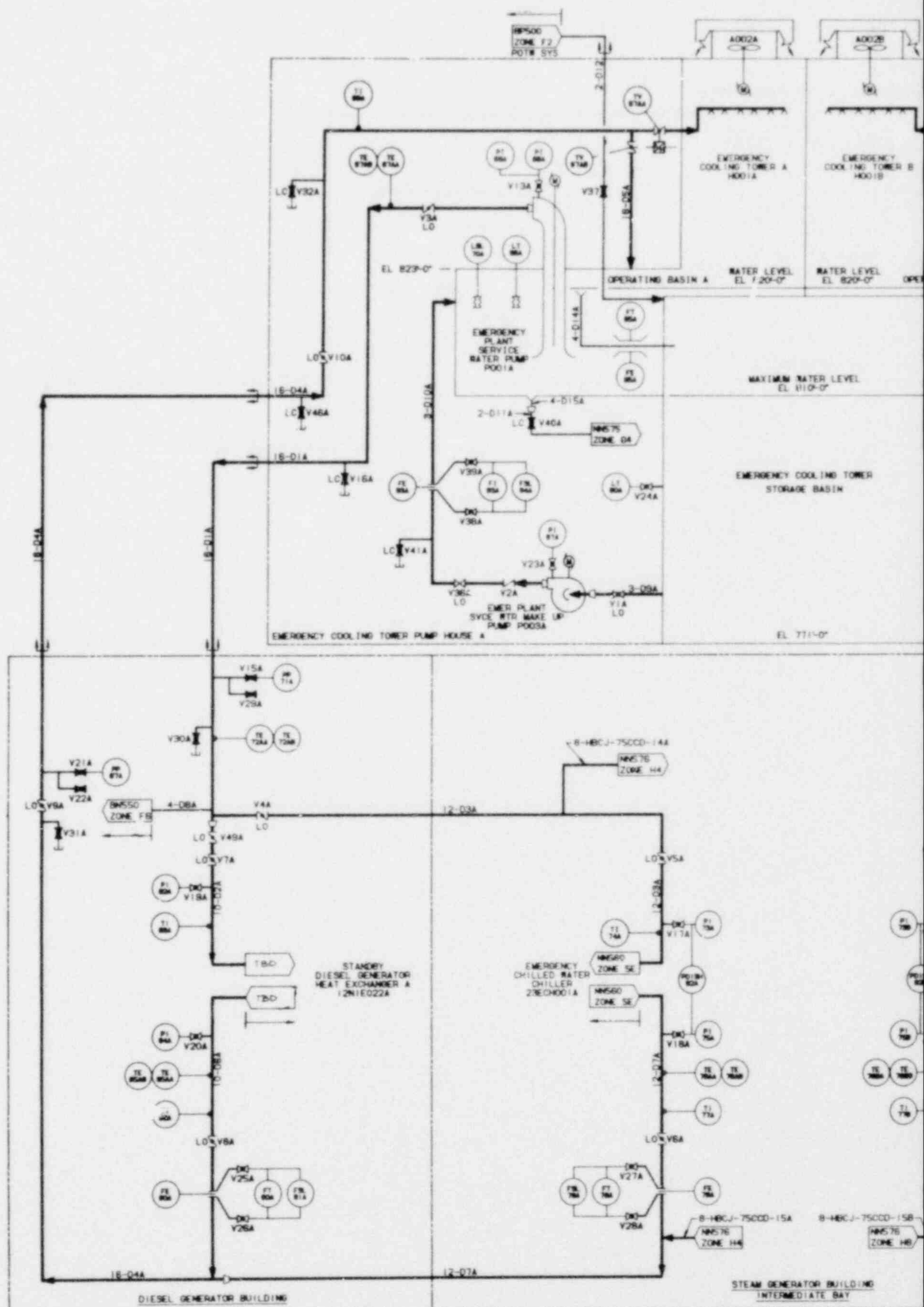
POOR ORIGINAL

REFERENCE DRAWINGS

1. P&ID CIRCULATING WATER SYSTEM — B&R DWG BW520
2. P&ID LALL EVAPORATION LIQUID RADIOACTIVE WASTE — B&R DWG NW534
3. P&ID LALL DEMINERALIZATION & DISCHARGE LIQUID RADIOACTIVE WASTE — B&R DWG NW541
4. P&ID SECONDARY SERVICE CLOSED COOLING WATER SYSTEM — B&R DWG BW533
5. P&ID MG SET & SGB HVAC — B&R DWG NV528
6. P&ID DGB HVAC — B&R DWG NV539
7. SODIUM PUMP DRIVE SYSTEM LUBE OIL SYSTEMS — GE DWG C052106
8. INSTRUMENTATION LOOP DIAGRAM NORMAL PLANT SERVICE WATER — B&R DWG BE4211
9. P&ID NORMAL CHILLED WATER SYSTEM — B&R DWG NW550
10. P&ID AUX BOILER FEEDWATER SYSTEM — B&R DWG BW516

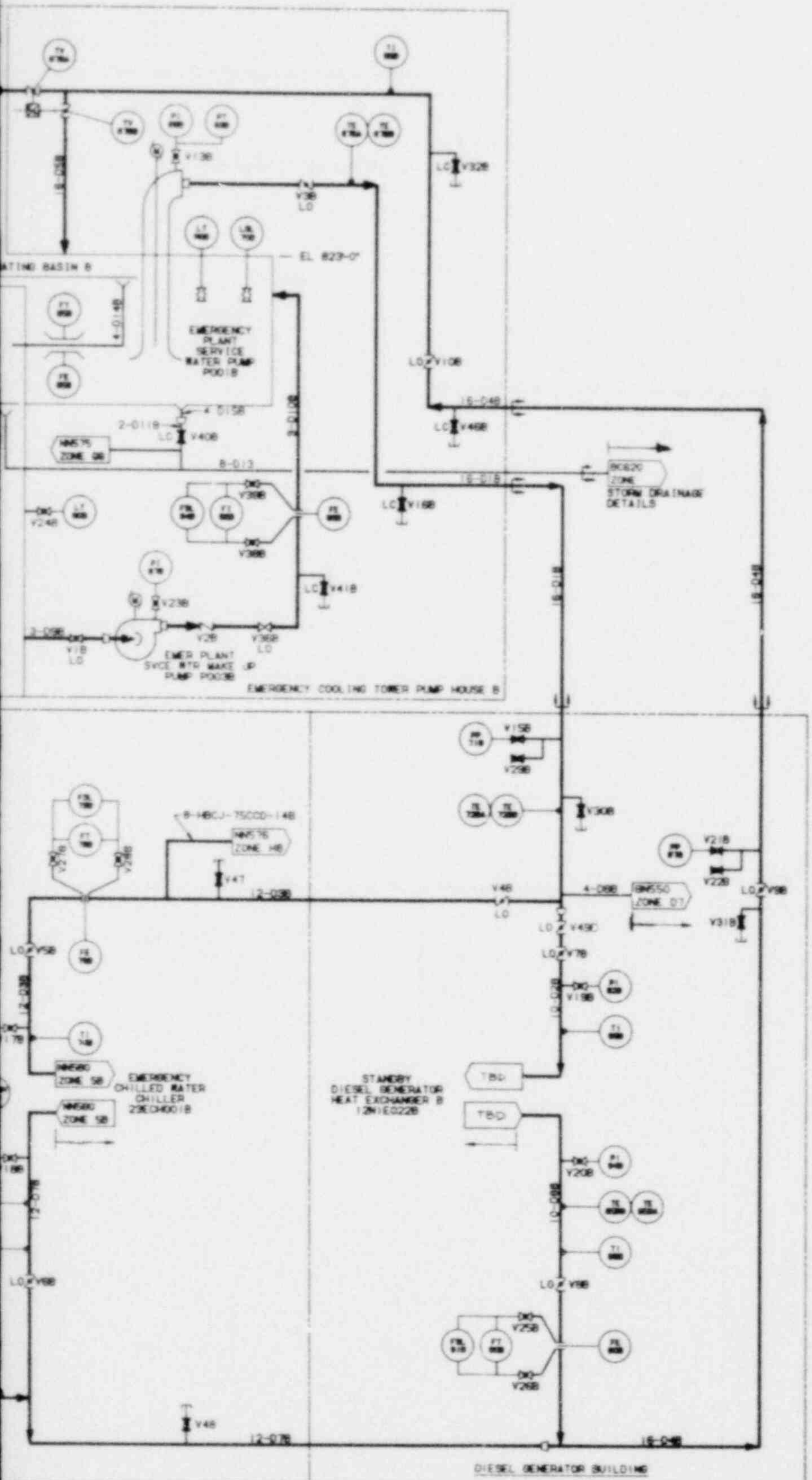
Figure 9.9-1 NORMAL PLANT SERVICE WATER

Amend. 59
Dec. 1980



NN575-6

POOR ORIGINAL



GENERAL NOTES

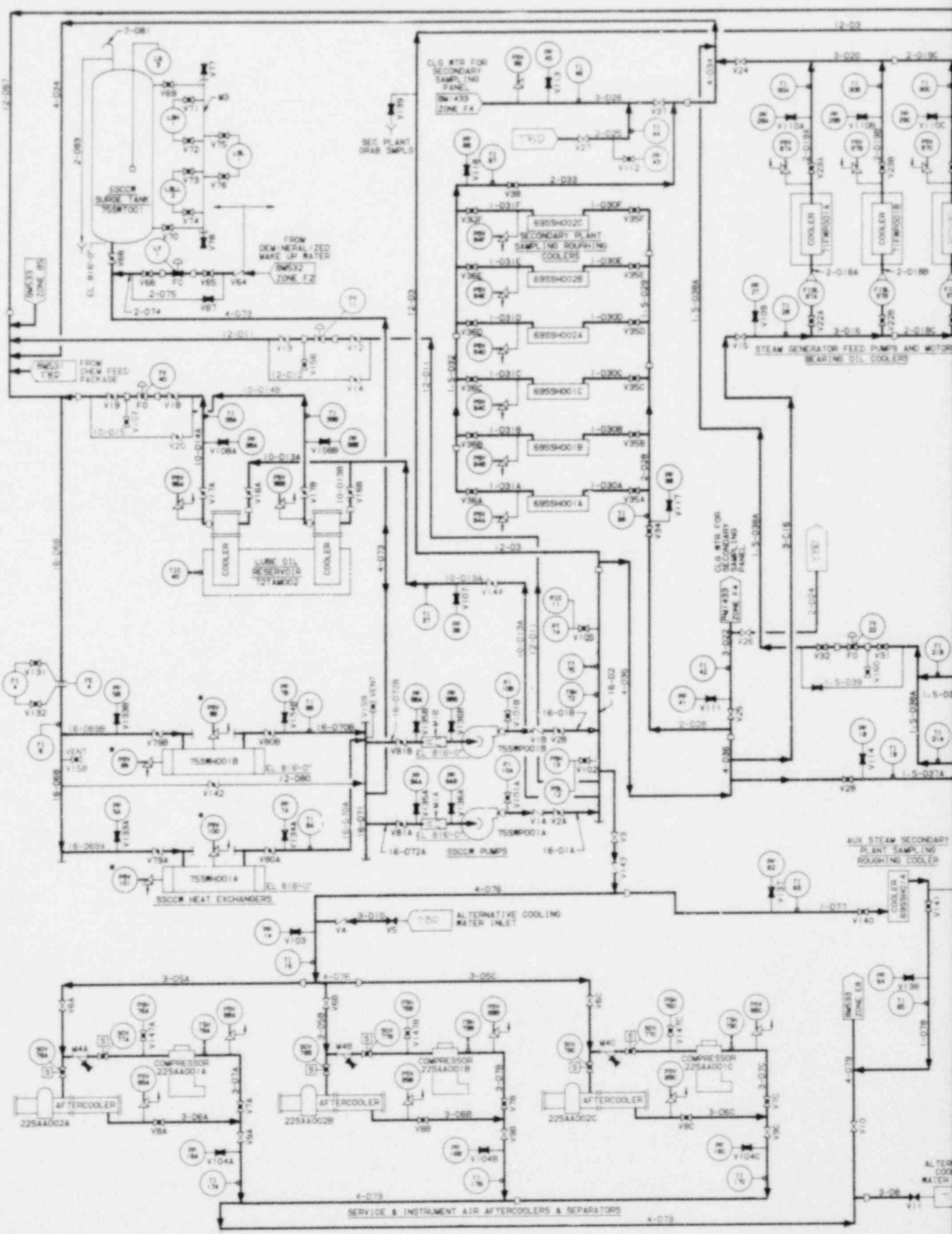
1. SYMBOLS AND ABBREVIATIONS WARD-D-0036
2. ALL EQUIPMENT NUMBERS AND MANUAL VALVE NUMBERS ARE PREFIXED BY 7SEP UNLESS OTHERWISE NOTED
3. ALL INSTRUMENT NUMBERS AND CONTROL VALVE NUMBERS ARE PREFIXED BY 7SEP UNLESS OTHERWISE NOTED
4. LINE NUMBERS ARE ABBREVIATED AS PER THE FOLLOWING EXAMPLE UNLESS OTHERWISE NOTED: 18-HBCJ-7SEPD-1 IS WRITTEN AS 18-01
5. SYSTEM CLEANLINESS CLASSIFICATION PUMPS - LEVEL C ALL OTHER COMPONENTS - MANUFACTURERS STANDARD (M)
6. SEISMIC CATEGORY: I UOS
7. RADIATION ZONE: UNRESTRICTED UOS
8. ALL PRESSURE AND FLOW CONNECTIONS SHALL BE 0.75" IPS UNLESS OTHERWISE NOTED
9. SECONDARY SAMPLING CONNECTIONS SHALL BE 0.75" IPS UNLESS OTHERWISE NOTED
10. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY K
11. APPLICABLE CODES ARE ASME SECTION III CLASS 3 FOR PIPING UOS
12. PERFORMANCE TEST CONNECTIONS IN ACCORDANCE WITH REQUIREMENTS OF ASME POWER TEST CODE PTC-6 SHALL BE ADDED LATER
13. SAFETY CLASSIFICATION: 3 UOS
14. ALL VENT AND DRAIN CONNECTIONS ARE 0.75" UNLESS OTHERWISE NOTED

POOR ORIGINAL

REFERENCE DRAWINGS

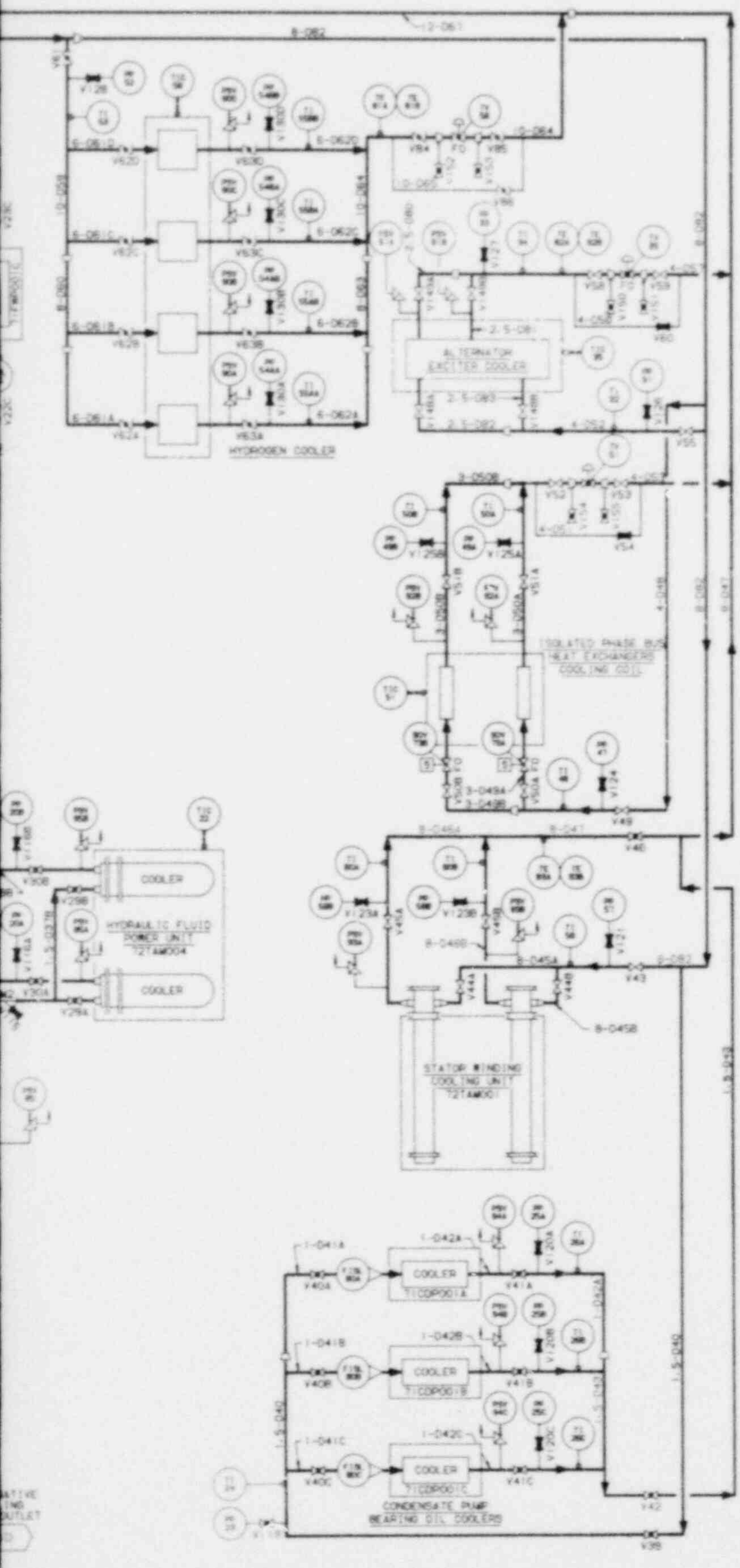
1. P&ID EMERGENCY CHILLED WATER SYSTEM B&R DWG HM680
2. P&ID RIVERWATER SERVICE B&R DWG BM540
3. P&ID POTABLE WATER SYSTEM B&R DWG BP500
4. STORM DRAINAGE PLANT AREA B&R DWG BC620
5. P&ID STANDPIPE SYS DGB SP & SSE FIRE PUMPS B&R DWG BM550
6. INSTRUMENT LOOP DIAGRAM EMERGENCY PLANT SERVICE WATER B&R DWG BE4218
7. INSTRUMENT LOOP DIAGRAM EMERGENCY PLANT SERVICE WATER B&R DWG BE4218

Figure 9.9-2 EMERGENCY PLANT SERVICE WATER



BM533-3

POOR ORIGINAL



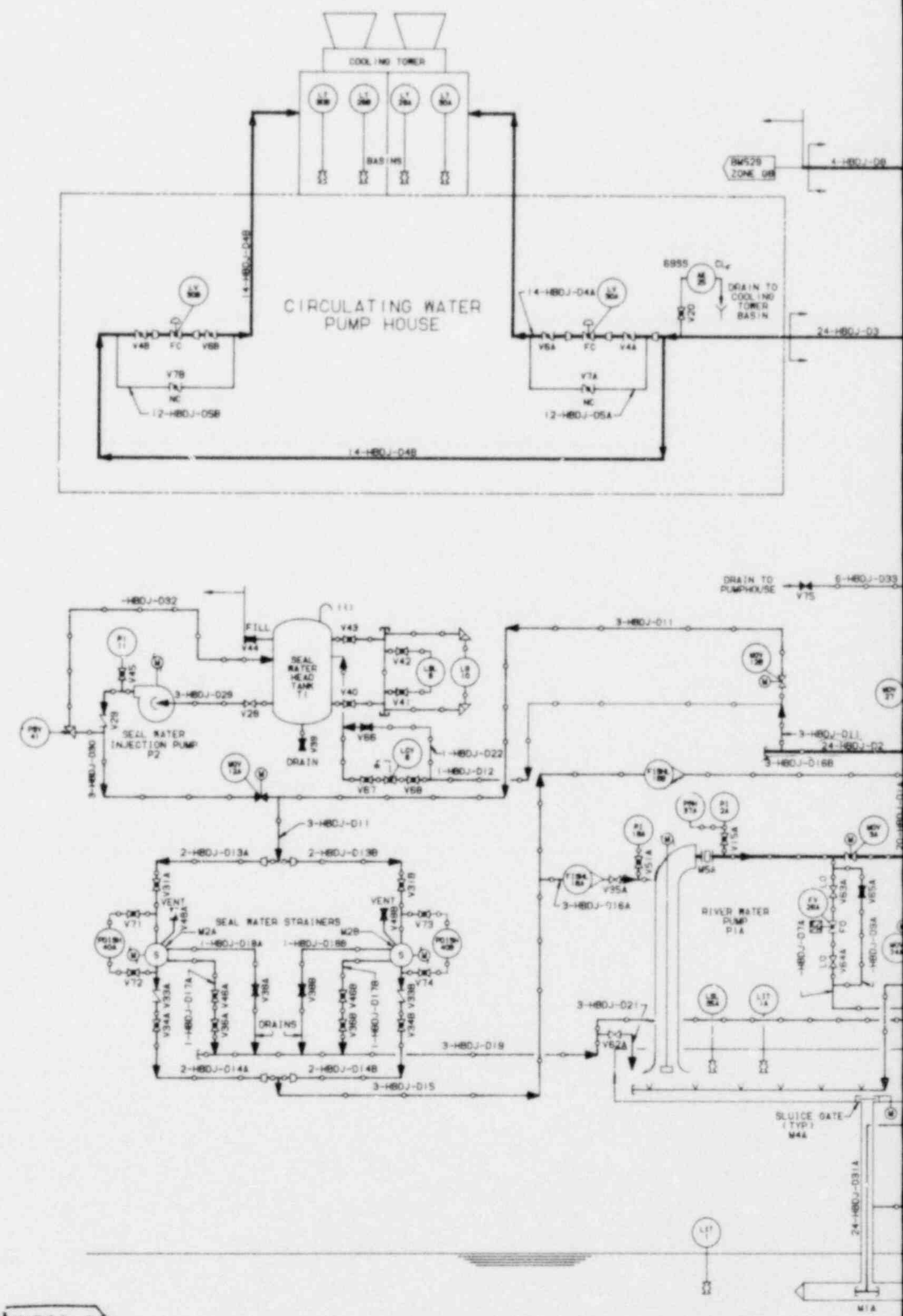
GENERAL NOTES

1. SYMBOLS AND ABBREVIATIONS MARD-O-0036
2. ALL EQUIPMENT NUMBERS AND MANUAL VALVE NUMBERS ARE PREFIXED BY 75SM UOS
3. ALL INSTRUMENT NUMBERS AND CONTROL VALVE NUMBERS ARE PREFIXED BY 75SM UOS
4. LINE NUMBERS ARE ABBREVIATED AS PER THE FOLLOWING EXAMPLE UOS: 3-HBCJ-75SM-052 IS WRITTEN AS 3-052
5. SYSTEM CLEANLINESS CLASSIFICATION: MANUFACTURERS STANDARD (M)
6. SEISMIC CATEGORY: III
7. RADIATION ZONE: UNRESTRICTED
8. ALL PRESSURE AND FLOW CONNECTIONS SHALL BE 0.75 UOS
9. SECONDARY SAMPLING CONNECTIONS SHALL BE 0.75 UOS
10. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY K
11. APPLICABLE CODES ARE ANSI B31.1 FOR PIPING AND ASME CODE SECTION VIII FOR HEAD TANK
12. SAFETY CLASSIFICATION: NONE
13. * VENDOR SUPPLIED

REFERENCE DRAWINGS

1. P&ID DEMINERALIZED WATER SYSTEM B&R DWG 8M532
2. P&ID CHEMICAL FEED SYSTEM B&R DWG 8M531
3. P&ID SECONDARY SAMPLING SYSTEM B&R DWG 8M439
4. LOGIC DIAGRAM SECONDARY SERVICE CLOSED COOLING WATER PUMP P001A "START/STOP" B&R DWG 8E4831
5. LOGIC DIAGRAM SECONDARY SERVICE CLOSED COOLING WATER PUMP P001B "START/STOP" & SURGE TANK LEVEL CONTROL B&R DWG 8E4832
6. LOGIC DIAGRAM SSCCW COMMON TROUBLE ALARM B&R DWG 8E4833
7. INSTRUMENT LOOP DIAGRAM SECONDARY SERVICE CLOSED COOLING WATER SYSTEM B&R DWG 8E4176
8. INSTRUMENT LOOP DIAGRAM SECONDARY SERVICE CLOSED COOLING WATER SYSTEM B&R DWG 8E4177

Figure 9.9-3 FLOW DIAGRAM SECONDARY SERVICE CLOSED COOLING WATER



NN540-12

POOR ORIGINAL

GENERAL NOTES

1. SYMBOLS AND ABBREVIATIONS WARD-D-0036
2. ALL EQUIPMENT NUMBERS AND MANUAL VALVE NUMBERS ARE PREFIXED BY 74RW UOS
3. ALL INSTRUMENT NUMBERS AND CONTROL VALVE NUMBERS ARE PREFIXED BY 74RW UOS
4. LINE NUMBERS ARE ABBREVIATED AS PER THE FOLLOWING EXAMPLE UOS: 24-HBDJ-74RW-03 IS WRITTEN AS 24-HBDJ-03
5. SAFETY CLASSIFICATION: NONE
6. SEISMIC CATEGORY: III
7. RADIATION ZONE: UNRESTRICTED UOS
8. SYSTEM CLEANLINESS CLASSIFICATION: (M) MANUFACTURERS STANDARD
9. ALL PRESSURE AND FLOW CONNECTIONS SHALL BE 0.75 UOS
10. ALL MOTOR NUMBERS ARE THE SAME AS THE EQUIPMENT NUMBERS EXCEPT THAT THE EQUIPMENT TYPE CODE IS REPLACED BY K
11. CL_2 INDICATES FREE CHLORINE
12. BOTH RWS PUMPS ARE EQUIPPED WITH SIPHON VALVES

REFERENCE DRAWINGS

1. P&ID MAKE UP WATER TREATMENT B&R DWG BW529
2. INSTRUMENT LOOP DIAGRAM RIVER WATER SERVICE B&R DWG BE4170
3. INSTRUMENT LOOP DIAGRAM RIVER WATER SERVICE B&R DWG BE4171
4. P&ID CHEMICAL FEED SYSTEM CPH B&R DWG BW528

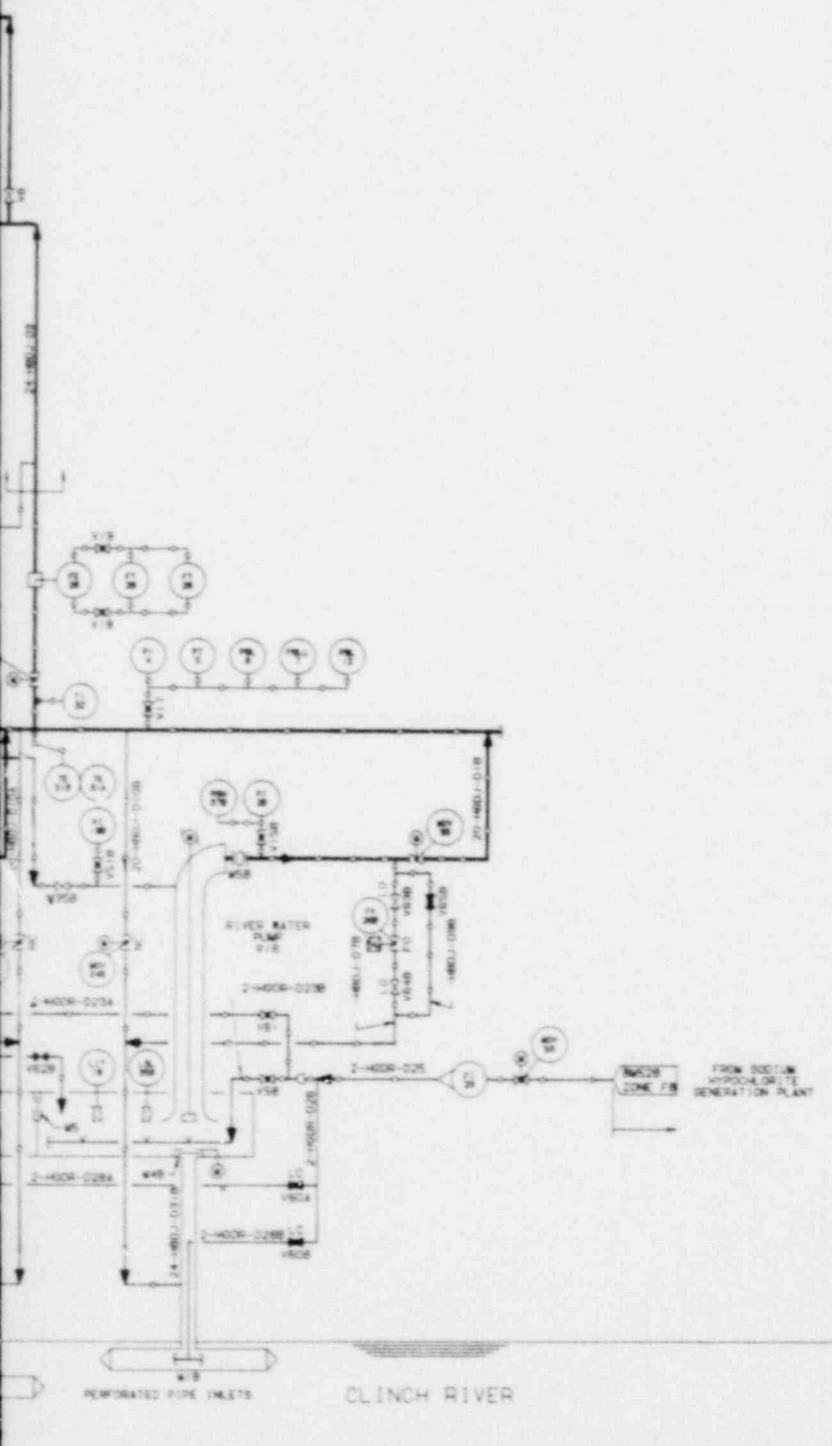


Figure 9.9-4 RIVER WATER SERVICE

31 9.11 Communications System

The Clinch River Breeder Reactor Plant is provided with the following communication systems providing effective and diversified means of communication between plant personnel in all vital areas during the full spectrum of accident or incident conditions under maximum potential
58 noise levels:

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

b. Private Automatic Exchange (PAX)

c. Microwave Communications

d. Powerline Carrier Communications (PLC)

e. Maintenance Communications Jacking (MCJ)

f. VHF Radio Station

g. Portable Radio

h. Manual Telephone Switchboard 29

i. Offsite Law Enforcement Radio

j. Security Intercom

47 k. Security Portable Radio System

9.11.1 Design Bases

a. The Public Address Intra-Plant Communications System (PA-IC) shall:

1. Provide primary communications throughout the plant.

2. Insure availability by being powered from the instrument AC bus which is capable of receiving power from offsite and normal AC power supply or the Non-Class IE station batteries. The PA-IC system equipment will be mounted on seismically qualified supports when located in Seismic Category I structures. The power and sound circuits are arranged so that any disruption of the system in the non-Seismic Category I areas does not affect the operation of the system in the Seismic Category I areas.

3. Provide fire, high radiation, and evacuation alarms throughout the plant to the plant personnel by means of manually actuated multitone generator signals broadcast through the page channel. 29

b. The Private Automatic Exchange (PAX) shall provide intra-plant communications in all plant areas and inter-plant communications between the plant and key locations within the TVA system. Office areas shall be supplemented with a key-telephone system.

c. The Microwave Communications System shall be the primary inter-plant communications system between CRBRP and other TVA facilities, through the existing TVA microwave communications network.

d. The Powerline Carrier Communications System shall be an alternate inter-plant communications system to the Microwave Communications System. 29

e. The Maintenance Communications Jacking System (MCJ) shall:

   1. Provide sound-powered communications between the Control Room, local instrument panels in all plant areas for supporting maintenance and instrument calibration activities, and pre-selected stations for fixed emergency communications. 48

   2. Provide sound-powered communications between the Plant Control System Steam Generator Building remote shutdown panel and all
47 local panels required for the support of remote plant shutdown.

f. The VHF Radio Station shall provide direct communication with the TVA Emergency Staff Operations Office in Chattanooga, Tenn.

g. The Portable Radio Communications System shall provide a "Total Area Coverage (TAC)" portable two-way radio communication system in all areas of the plant, both inside and outside the buildings for traffic control, maintenance operations, fire control and general communications. 48

h. The Manual Telephone Switchboard shall provide an inter-plant communication terminal in the Control Room. Call director type telephones shall be provided for the shift engineer and unit operator as an extension of the switchboard. 29

i. The Offsite Law Enforcement Radio System shall provide an independent voice channel to the local offsite law enforcement agency.

j. The Security Intercom System shall provide communications between remote security stations and shall provide tone paging as required
47 for use by plant security personnel.

Electrical power for the Fire Protection System is provided from
the normal plant AC power distribution system. If normal AC power
is unavailable, the water supply system pressure will be maintained
by two diesel-driven fire pumps, and the fire detection system will
be energized by a non-class 1-E 4-hour DC battery/inverter system
which has the capability of being connected to an emergency diesel
generator through qualified isolation devices. 48

The control of smoke, heat, flame and combustible and explosive
gases which might be produced in the different plant areas is
discussed in Table 9.13-3. In areas with forced air ventilation
30 systems the operation of the ventilation systems is automatically
controlled before or simultaneously with the start of the extin-
guishment. Exhaust system controls for these areas are designed to
include manual override.

The Control Room Fire Protection Panel is located in the Main
30 Control Room. A general warning signal "fire" annunciated at the
Main Control Board supplemented by an audible alarm device alerts
the plant operators. Detailed information is displayed on a Control
Room Fire Protection Panel so that the operator can make a complete
assessment of the situation in case of an outbreak of fire.

The following information is shown on the Control Room Fire Protection Panel:

1. Signals indicating the automatic and/or manual actuation of fire fighting systems

2. Presence of smoke or heat signalled by automatic detectors

3. Signals indicating presence of smoke or heat in auxiliary systems
58 (e.g. ventilation, etc.) which actuate fire alarms

4. Repetitions of local fire alarms 48

5. Confirmation of water reserves, availability and operation of pumps and pressure of the water supply system

6. Controls and indication for manual actuation of the Fire
58 Protection System Pumps

7. Capability for testing the integrity of all fire protection panels 13

58 8. Annunciation and alarm of electrically supervised water supply isolation valves

9. Annunciation and alarm of fire doors, as required

59 58 48 10. Signals indicating trouble in the detection or supervision system 43

d. The fire detection and protection systems used to contain, control, and extinguish electrical cable fires are identified in Table 9.13-3. The design description of these fire protection systems is provided in Table 9.13-4.

The electrical design criteria for circuit integrity and protection are described in Section 8.3 of the PSAR.

Fire suppressing or extinguishing agents which result in corrosive products upon pyrolysis are not employed to protect essential circuitry.

9.13.1.3 <u>System Evaluation</u>

The following evaluation refers to the design basis fires identified in Section 9.13.1.1a.

<u>Lubricating Oil Fire</u>

The maximum fire involving lubricating oil (in the Turbine Generator Building) involves approximately 26,000 gallons of turbine lubricating oil. This fire does not involve any safety-related equipment and is controlled by a wet type sprinkler system.

This area is also served by standpipe system and portable fire extinguishers are provided in the area for the purpose of manual fire fighting. A fire in this area would produce severe flame development, high heat output and high smoke development. The time involved from the initial detection to activation of the protection system shall be maintained at a minimum. Since this event does not involve any safety related equipment, safe shutdown of the plant can be accomplished.

<u>Diesel Generator Fuel Oil Fire</u>

The maximum fire involving fuel oil (in diesel generator fuel oil tanks located below grade outside of DGB) involves approximately 50,000 gallons of No. 2-D diesel fuel oil. This fire does not involve any safety related structures as the tanks are located 48
outside of NI buildings. Since the fuel is stored below grade, the 13

9.16 Recirculating Gas Cooling System

The Recirculating Gas Cooling System (RGCS) provides heat removal capability for primary CRDM, primary Na makeup pumps, EVS Na pumps and cold trap and for inerted cells in the RCB and RSB and maintains the cell temperature below a level which would be deleterious to concrete, electrical wiring, instrumentation, components or equipment. The RGCS is comprised of 13 independent subsystems, 8 of which are located in the RCB and 5 in the RSB. Table 9.16-1 lists these subsystems and their seismic and safety classification, Table 9.16-2 lists major system parameters.

9.16.1 Design Basis

The RGCS is designed to provide the following capability:

1) Provide heat removal capability and maintain the following required temperature in the inerted cells of the RCB and RSB.

   a) 120^{O}F nominal cell gas temperature under normal operating conditions.

   b) TBD ^{O}F cell temperature under off-normal operating conditions.

   c) 150^{O}F concrete temperature, except local hot spot which shall not exceed 200^{O}F during operating condition.

   d) less than 350^{O}F concrete temperature for a period of 24 hours.

2) Provide cooling gas directly to the primary Na makeup pumps, EVS Na pumps, EVS Na Cold Trap and the primary Control Rod Drive Mechanism.

3) Prevent leaked water from the cooling coil entering the cells containing Na or Nak components.

4) Isolate the RGCS components in the event of an Na or NaK spill or leak.

5) Maintain the independence of the served redundant system.

47 6) Isolate individual cells for maintenance.

59 7) Provide accumulators or air operated valves sized with sufficient margin, to perform their safety related function for the duration required.

9.16.2 System Description

All RGCS subsystems except for the Primary Control Rod Drive Mechanism cooling (CR) subsystem operate at approximately atmospheric pressure. The subsystem (CR) operates at 100 psig. A typical low pressure RGCS subsystem is shown in Figure 9.16-1. All the operating equipment such as fan, cooler and valves are located outside the inerted cells in normally
47 accessible areas. The return gas from the cell is drawn through piping embedded

in shielding concrete, an isolation valve and a cooler by a fan located downstream of the cooler. The cooled gas is supplied to the cell through an isolation valve and piping embedded in the shielding concrete. Inside the cell gas is distributed by the ducts. The isolation valve is located close to the cell and the inerting and deinerting connection are provided on the component side of the valve to facilitate inerting and deinerting of the cooler and fan without deinerting the cell served. Figures 9.16-3 through 9.16-7 show the P&ID's for the various RGCS subsystems including the identification of their safety, seismic and code classifications. Table 9.16-2 lists major parameters of these subsystems.

The low pressure fans are direct-driven vaneaxial fans with manually adjustable blades. The motors are totally enclosed, nominal 460V, 3 phase induction motors with NEMA class 'H' insulation.

A typical cooler contains commercially available cooling coils in a factory fabricated ASME Code rated casing and is shown in Figure 9.16-2.

9.16.2.1 Primary Heat Transport Systems (PHTS) Subsystem PA, PB, PC

Each of the PHTS cells group is served by a separate RGCS subsystem to maintain the independence of the PHTS loops. Each subsystems consists of one 100% cooler and one 100% fan, as shown in Figure 9.16-3 and cools (1) the PHTS cells, (2) the associated hot leg pipe chases and (3) the associated check valve cells, up to and including the reactor cavity bellow seals. The supply duct distributes cooled gas to each of these cells. The return from the hot leg pipe chase and check valve cell is through the clearance around the pipes as they penetrate the neutron shield wall. The return gas from the PHTS cell is drawn from a high point and passes through the shielding block located within the cell. The return gas is drawn by a fan through the cooler and supplied back to the cell.

9.16.2.2 Control Rod Drive Mechanism (CRDM) - Subsystem CR

The subsystem CR consists of two 100% coolers and two 100% blowers, is shown in Figure 9.16-3 and cools only the primary control rod drive mechanisms. A separate subsystem is used for this purpose because of the high system pressure (100 psig) and high system pressure
59 drop (8 psi). The pressure relief valve is set at 150 psig.

The return gas passes through a manual butterfly valve, the operating blower, the cooler and an automatic butterfly valve before returning to the PCRDMs. Unlike the low pressure subsystems, in this subsystem the blowers are located upstream of the coolers. This is necessitated by a large energy input in the blower resulting in approximately 45°F rise in gas temperature. The blower operates in
59 47 a gas environment at a maximum temperature of 175°F.

In order to maintain the integrity of the cell liner, the piping from the cell liner up to and including isolation valves for non-safety related subsystems which serve cells containing Na or Nak is designed to the requirements of Seismic Category I.

All the safety related subsystems are designed in accordance with the requirements of ASME Section III Class 3, Seismic Category I, supplied Class IE electrical power and emergency chilled water.

Safety related subsystems MA and MB serve the two primary Na makeup pumps. Since the primary Na makeup pumps are redundant to one another, no further redundancy in components is provided for subsystems MA and MB.

Safety related subsystems EA and EB serve the two EVS Na pumps in the two active EVS Na cooling loops. No redundancy in components is provided for subsystems EA and EB since there is redundancy in the EVS Na cooling loops themselves.

All the subsystems using water as the coolant and serving areas containing Na or Nak are provided redundant water leakage detection sensors. On detection of water leakage in a subsystem, the operation of the subsystem is stopped automatically, the automatic isolation valves in the gas lines and chilled water system lines are closed and the redundant drain valves on the cooler are opened automatically.

In case of a sodium or Nak spill or leak in a cell, the operation of the subsystem serving the cell is stopped and the automatic isolation valves in the gas stream are closed.

9.16.4 Tests and Inspection

Each individual component of the system is tested at the factory and, before the plant startup, entire system is tested and the gas flow rate is balanced and set at design flow conditions. Periodic inspection of the components is scheduled to ensure proper system operation. In-service inspection will be conducted according to ASME Section XI for safety related subsystems as described in detail in Section 3.1.

9.16.5 Instrumentation and Control

The RGCS instrumentation is designed to provide for measurements, controls and alarms of system parameters. Each subsystem is provided with a control panel located near the fan. The panels include the control switches, monitors and system alarms.

All non-safety related subsystems are provided with local monitoring and alarm and a remote alarm in the main control room.

All safety related subsystems are provided with local monitoring
47 and alarm and remote monitoring and alarms in the main control room.

A list of compressed air operated safety related valve which are provided with safety class 3 air accumulators is contained in
59 Table 9.16-3.

TABLE 9.16-1

SAFETY AND SEISMIC CLASSIFICATION

Subsystem	Area Served	Seismic Category*	Safety Class
PA	Primary Heat Transport System (PHTS) Loop #1	III	**
PB	PHTS Loop #2	III	**
PC	PHTS Loop #3	III	**
CR	Control Rod Drive Mechanisms (CRDM)	III	**
MA	Sodium Makeup Pump and Vessels	I	SC-3
MB	Sodium Makeup Pump and Pipeways	I	SC-3
CT	Cold Trap, Nak Cells	III	**
RC	Reactor Cavity	I	**
EA	Ex-Vessel Storage (EVS) Loop #1	I	SC-3
EB	EVS Loop #2	I	SC-3
EC	EVS Loop #3	III	**
ET	Ex-Vessel Storage Tank (EVST) Cavity	III	**
FH	Fuel Handling Cell	I	**

54

* All Subsystems' (Except Subsystem CR) isolation valves and piping between the automatic isolation valves and the liner of served cells shall be Seismic Category I.

47

**These subsystems have no designated Safety Classification

TABLE 9.16 2

SYSTEM PARAMETERS

Subsystem Designation	Title	Gas	Operating Cooling Capacity BTU/HR	Operating Gas Flow SCFM	Design Pressure PSIG	Design Temperature °F
PA	PHTS Loop #1	N_2	1.17×10^6	16,300	TBD	TBD
PB	PHTS Loop #2	N_2	1.09×10^6	15,200	TBD	TBD
PC	PHTS Loop #3	N_2	1.17×10^6	16,400	TBD	TBD
CR	Control Rod Drive Mechanism	N_2	0.28×10^6	3,200	150	TBD
MA	Sodium Makeup Pump and Pipeways	N_2	1.24×10^6	18,300	TBD	TBD
MB	Sodium Makeup Pump and Vessels	N_2	0.60×10^6	9,400	TBD	TBD
CT	Cold Trap, Nak Cell	N_2	0.33×10^6	5,010	TBD	TBD
RC	Reactor Cavity	N_2	1.49×10^6	20,950	TBD	TBD
EA	EVS Loop #1	N_2	0.52×10^6	8,730	TBD	TBD
EB	EVS Loop #2	N_2	0.94×10^6	12,480	TBD	TBD
EC	EVS Loop #3	N_2	0.19×10^6	2,610	TBD	TBD
ET	Ex-Vessel Storage Tank Cavity	N_2	0.36×10^6	5,000	TBD	TBD
FH	Fuel Handling Cell	Ar	0.55×10^6	8,000	TBD	TBD

TABLE 9.16-3

LIST OF SAFETY-RELATED VALVES REQUIRING COMPRESSED AIR

TO PERFORM THEIR SAFETY-RELATED FUNCTION

Valve No.	Figure no.	Subsystem	Normal Position	Fail Position
28MANV001A	9.16-4	MA	Open	Closed
28MANV001B	9.16-4	MA	Open	Closed
28MBNV001A	9.16-4	MB	Open	Closed
28MBNV001B	9.16-4	MB	Open	Closed
28EANV001A	9.16-6	EA	Open	Closed
28EANV001B	9.16-6	EA	Open	Closed
28EBNV001A	9.16-6	EB	Open	Closed
59 28EBNV001B	9.16-6	EB	Open	Closed

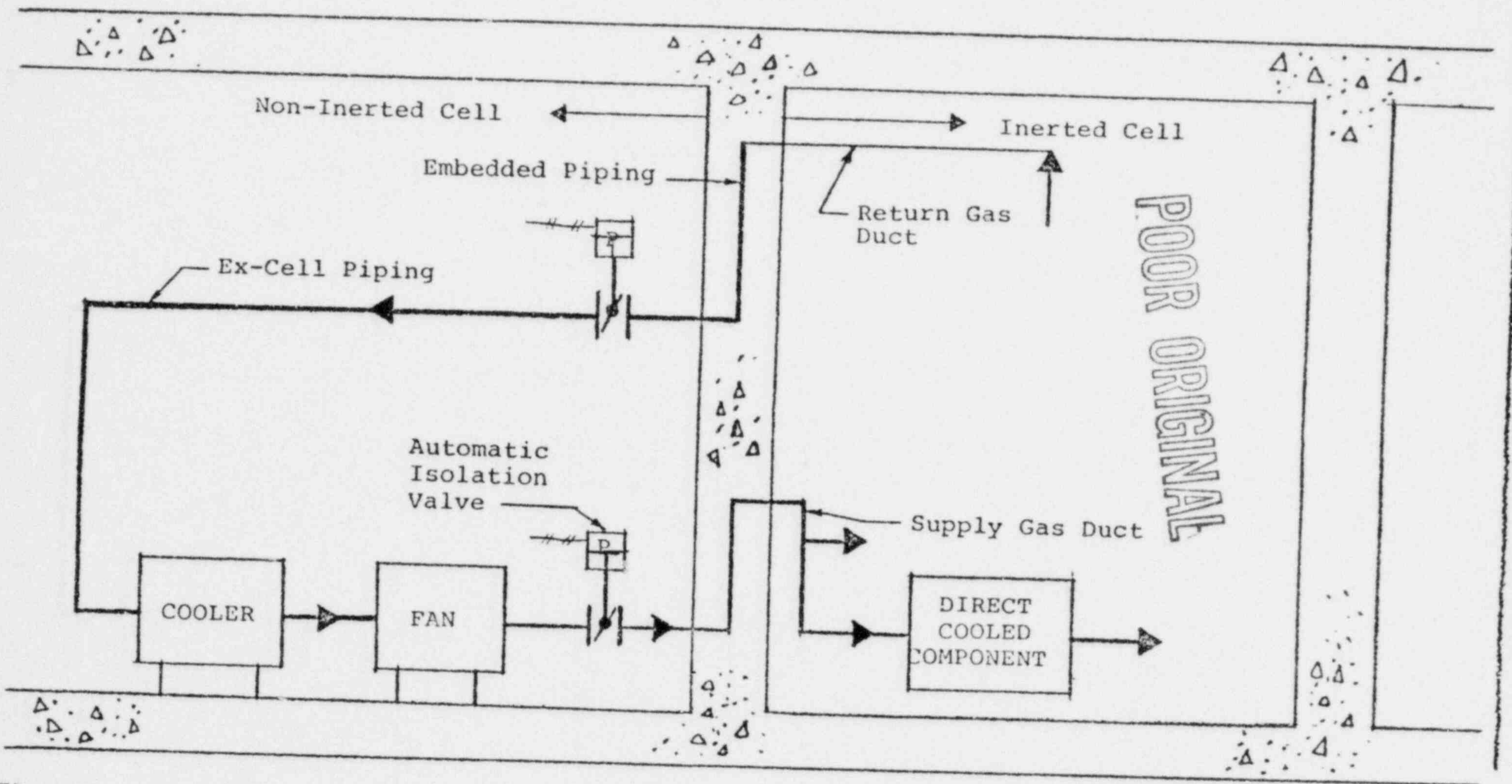
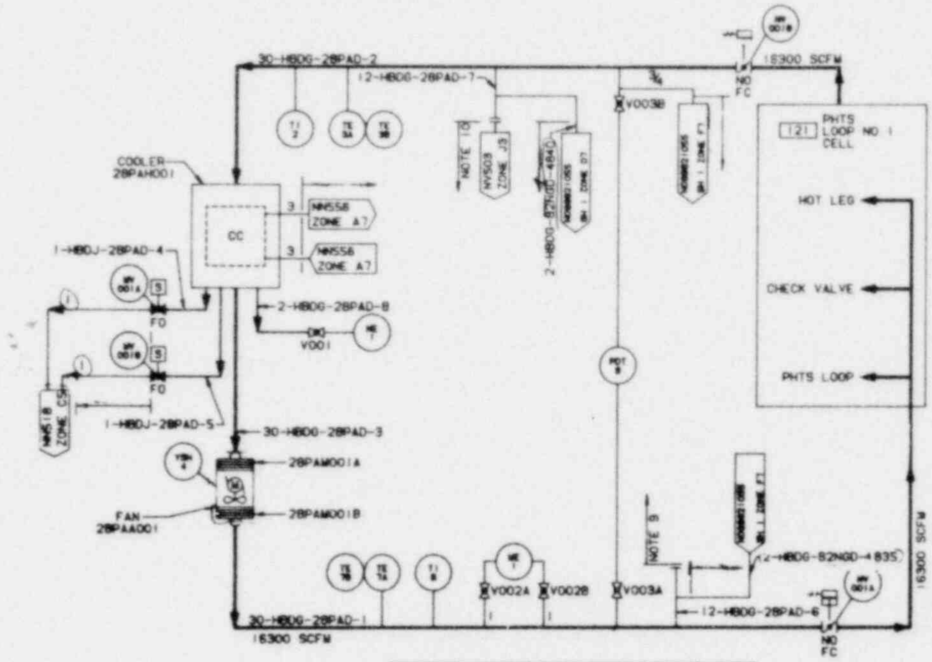
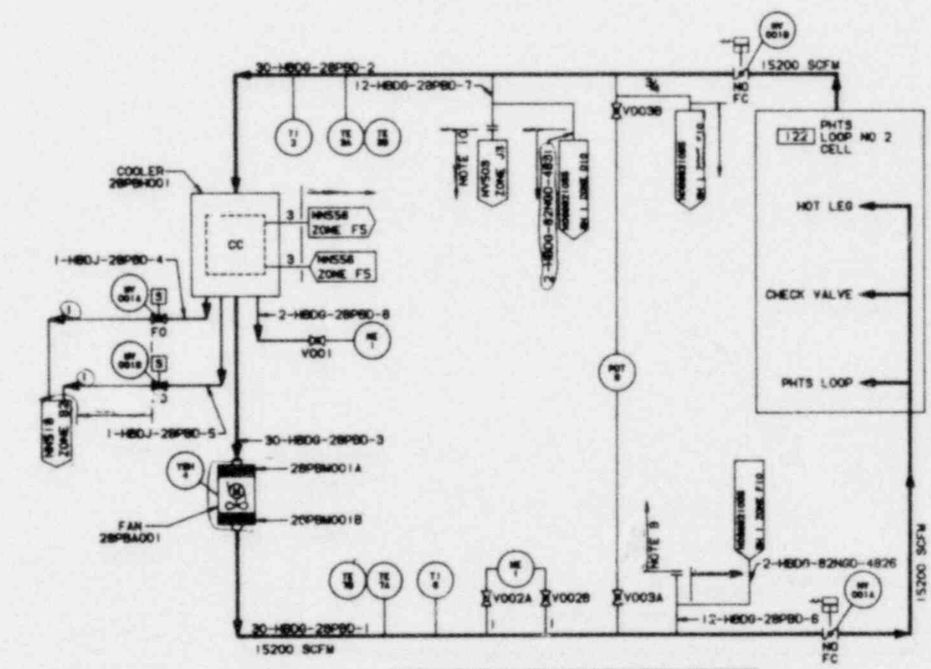


Figure 9.16-1 Typical Subsystem of Recirculating Gas Cooling System

Amend. 47
Nov. 1978

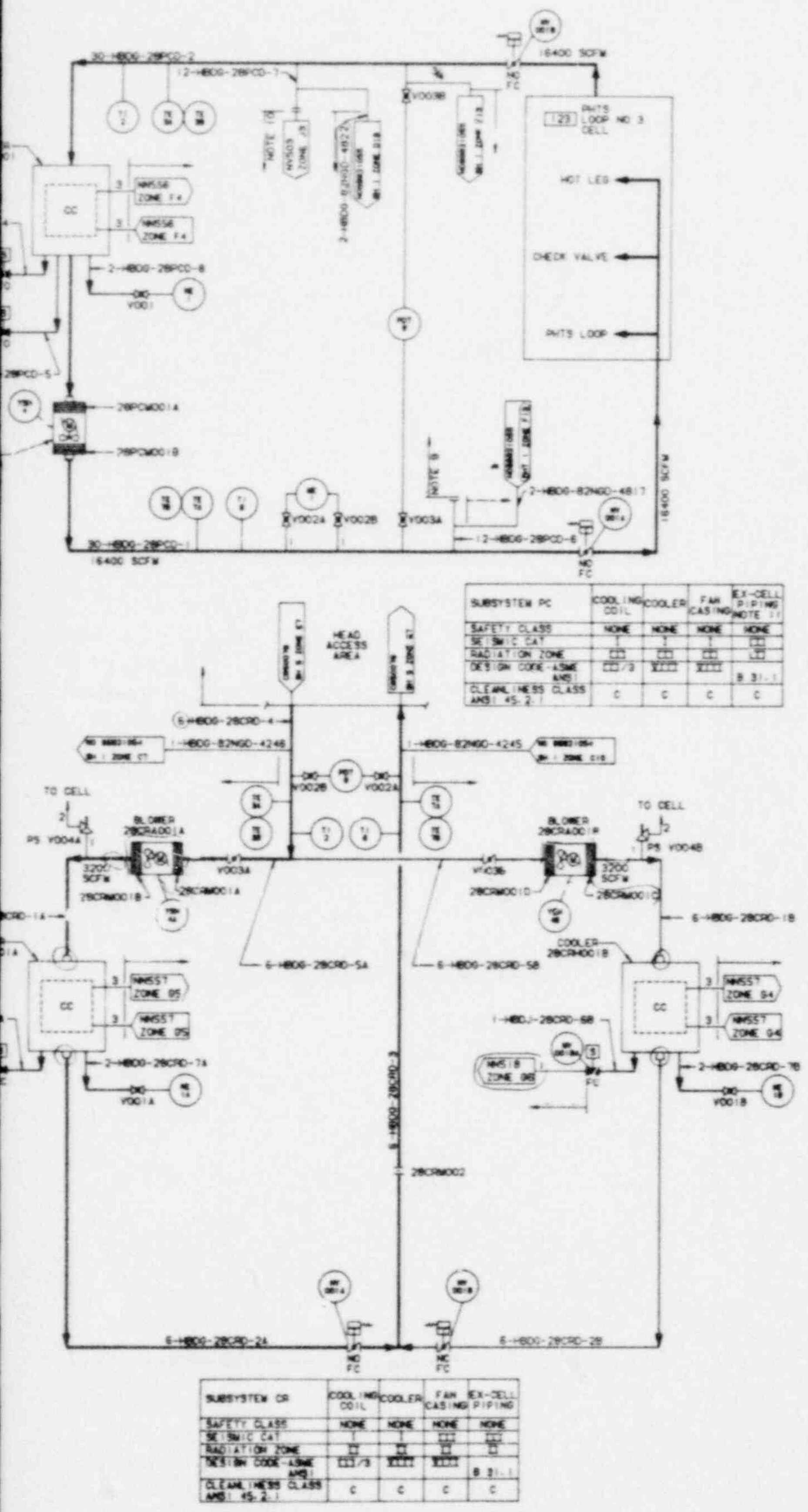


SUBSYSTEM PA	COOLING COIL	COOLER	FAN CASING	EX-CELL PIPING NOTE 11
SAFETY CLASS	NONE	NONE	NONE	NONE
SEISMIC CAT	I	I	I	[illegible]
RADIATION ZONE	[illegible]	[illegible]	[illegible]	[illegible]
DESIGN CODE-ASME ANSI	[illegible]/3	VIII	VIII	B 31.1
CLEANLINESS CLASS ANSI 45.2.1	C	C	C	C



SUBSYSTEM PB	COOLING COIL	COOLER	FAN CASING	EX-CELL PIPING NOTE 11
SAFETY CLASS	NONE	NONE	NONE	NONE
SEISMIC CAT	I	I	I	[illegible]
RADIATION ZONE	[illegible]	[illegible]	[illegible]	[illegible]
DESIGN CODE-ASME ANSI	[illegible]/3	VIII	VIII	B 31.1
CLEANLINESS CLASS ANSI 45.2.1	C	C	C	C

NN575-7



SUBSYSTEM PC	COOLING COIL	COOLER	FAN CASING	EX-CELL PIPING NOTE 11
SAFETY CLASS	NONE	NONE	NONE	NONE
SEISMIC CAT	I	I	I	[illegible]
RADIATION ZONE	[illegible]	[illegible]	[illegible]	[illegible]
DESIGN CODE-ASME ANSI	[illegible]/3	[illegible]	[illegible]	B 31.1
CLEANLINESS CLASS ANSI 45.2.1	C	C	C	C

SUBSYSTEM CR	COOLING COIL	COOLER	FAN CASING	EX-CELL PIPING
SAFETY CLASS	NONE	NONE	NONE	NONE
SEISMIC CAT	I	I	[illegible]	[illegible]
RADIATION ZONE	[illegible]	[illegible]	[illegible]	[illegible]
DESIGN CODE-ASME ANSI	[illegible]/3	[illegible]	[illegible]	B 31.1
CLEANLINESS CLASS ANSI 45.2.1	C	C	C	C

GENERAL NOTES

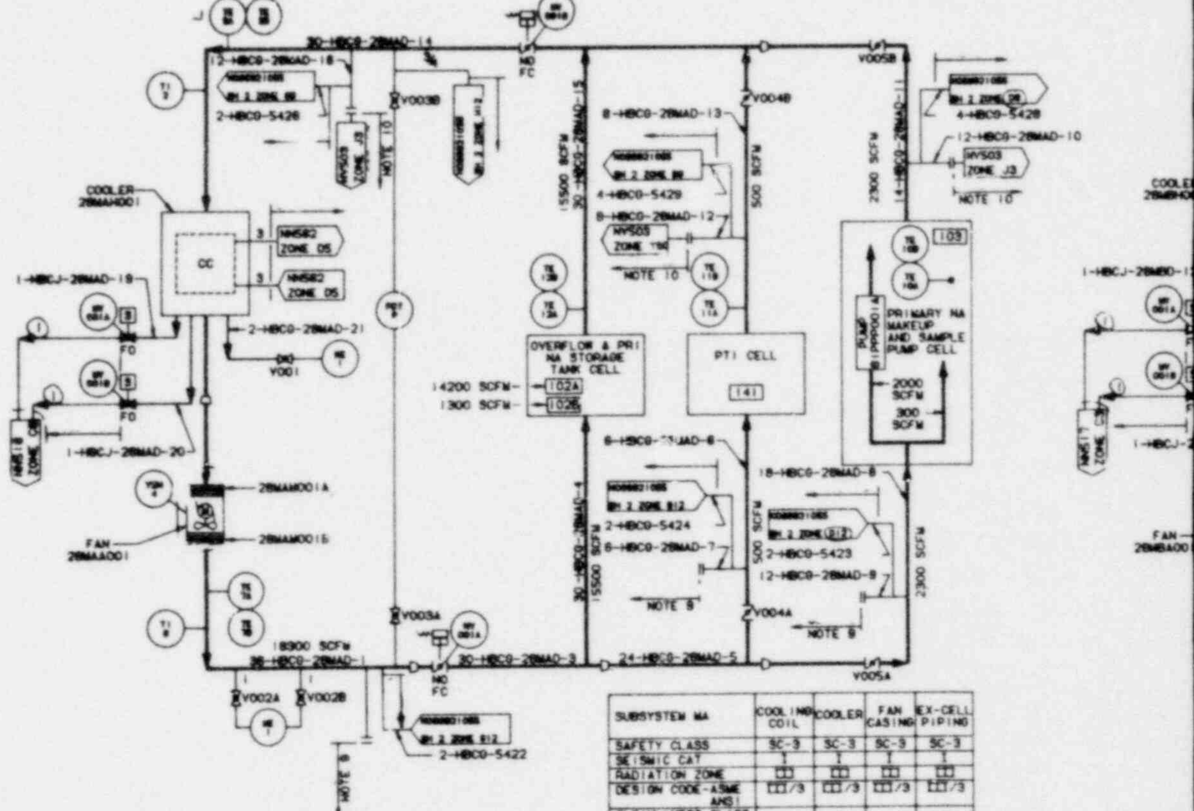
1. ALL PRESSURE CONNECTIONS SHALL BE 3/4" UNLESS OTHERWISE NOTED
2. ALL EQUIPMENT NUMBERS AND MANUAL VALVE NUMBERS ARE PREFIXED BY 28 FOLLOWED BY SUBSYSTEMS AS DESIGNATED ON THIS DRAWING
3. ALL INSTRUMENT NUMBERS AND CONTROL VALVE NUMBERS ARE PREFIXED BY 28 FOLLOWED BY SUBSYSTEMS AS DESIGNATED ON THIS DRAWING
4. SEE GAS FLOW BALANCE SHEETS, APPENDIX C FOR QUANTITIES
5. ALL VALVES ARE NORMALLY OPEN UNLESS OTHERWISE NOTED
6. ALL TEMPERATURE CONNECTIONS SHALL BE 1" UNLESS OTHERWISE NOTED
7. ALL PIPE SIZES ARE IN INCHES
8. SYMBOLS AND ABBREVIATIONS WARD-D-0038
9. BLIND FLANGE TO BE REMOVED DURING DE-INERTING OF INDIVIDUAL CELL. AIR INTRODUCED FROM RCB NORMAL ATMOSPHERIC AREAS
10. BLIND FLANGE TO BE REMOVED DURING DE-INERTING OF INDIVIDUAL CELL AND FLEXIBLE DUCT CONNECTED TO SYSTEM 25 PORTABLE PURGE FAN AND DISCHARGED TO SYSTEM 25 PURGE EXHAUST DUCT
11. PIPING BETWEEN THE AUTOMATIC ISOLATION VALVES AND THE LINER OF CELLS SERVED INCLUDING THE ISOLATION VALVES SHALL BE SEISMIC CATEGORY I

POOR ORIGINAL

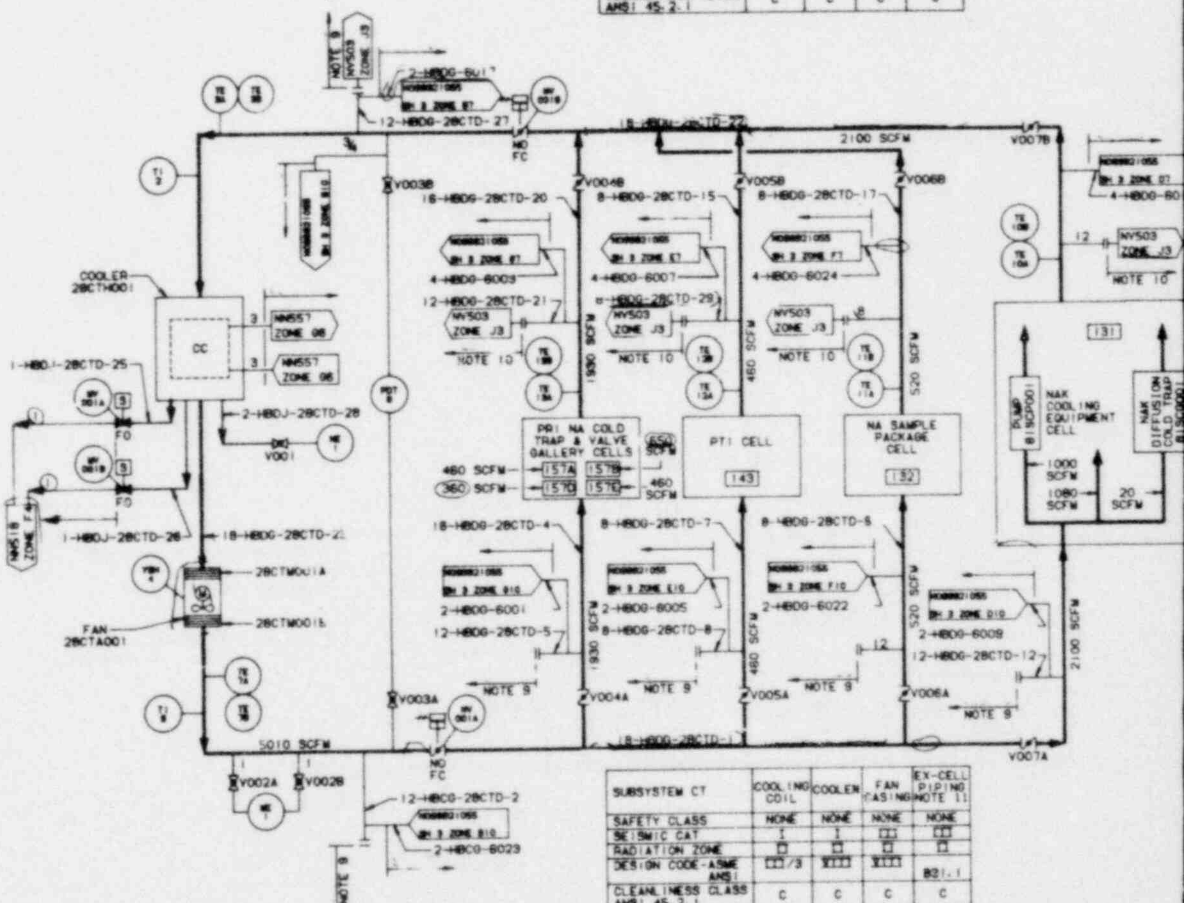
REFERENCE DRAWINGS

1. P&ID RCB HVAC — B&R DWG HV503
2. P&ID NORMAL CHW SYSTEM RCB & SGB — B&R DWG NM556
3. P&ID NORMAL CHW SYSTEM RCB & TGB — B&R DWG NM557
4. RCB N_2 DISTRIBUTION HIGH PRESSURE P&ID — AI DWG NO8882I054
5. RCB N_2 DISTRIBUTION P&ID — AI DWG NO8882I055
6. HEAD ACCESS AREA GAS SERVICE INTERFACE CONTROL — WARD DWG CM50078
7. P&ID RCB EQUIPMENT & FLOOR DRAINS, EAST SIDE — B&R DWG NM517
8. P&ID RCB EQUIPMENT & FLOOR DRAINS, WEST SIDE — B&R DWG NM518

Figure 9.16-3 RCB RGCS



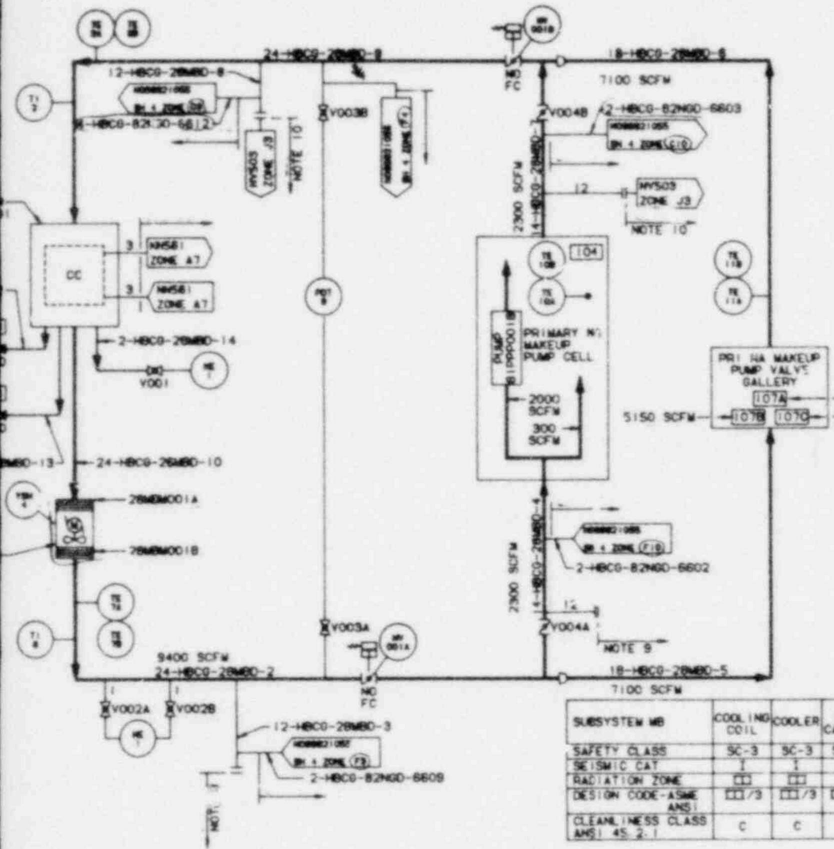
SUBSYSTEM MA	COOLING COIL	COOLER	FAN CASING	EX-CELL PIPING
SAFETY CLASS	SC-3	SC-3	SC-3	SC-3
SEISMIC CAT	I	I	I	I
RADIATION ZONE	II	II	II	II
DESIGN CODE-ASME ANSI	III/3	III/3	III/3	III/3
CLEANLINESS CLASS ANSI 45.2.1	C	C	C	C



SUBSYSTEM CT	COOLING COIL	COOLER	FAN CASING	EX-CELL PIPING NOTE 11
SAFETY CLASS	NONE	NONE	NONE	NONE
SEISMIC CAT	I	I	II	II
RADIATION ZONE	I	I	I	I
DESIGN CODE-ASME ANSI	III/3	VIII	VIII	B31.1
CLEANLINESS CLASS ANSI 45.2.1	C	C	C	C

NV576-8

POOR ORIGINAL



SUBSYSTEM WB	COOLING COIL	COOLER	FAN CASING	EX-CELL PIPING
SAFETY CLASS	SC-3	SC-3	SC-3	SC-3
SEISMIC CAT	I	I	I	I
RADIATION ZONE	□□	□□	□□	□□
DESIGN CODE-ASME ANSI	□□/3	□□/3	□□/3	□□/3
CLEANLINESS CLASS ANSI 45.2.1	C	C	C	C

GENERAL NOTES

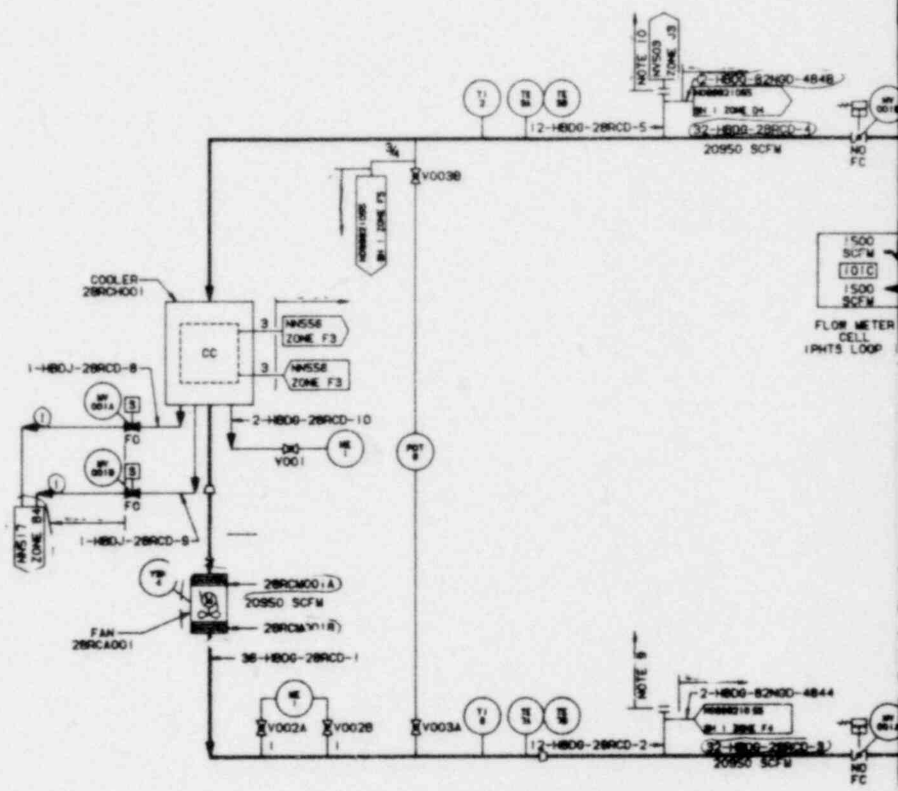
1. ALL PRESSURE CONNECTIONS SHALL BE ¾" UOS
2. ALL EQUIPMENT NUMBERS AND MANUAL VALVE NUMBERS ARE PREFIXED BY 28 FOLLOWED BY SUBSYTEMS AS DESIGNATED ON THIS DRAWING
3. ALL INSTRUMENT NUMBERS AND CONTROL VALVE NUMBERS ARE PREFIXED BY 28 FOLLOWED BY SUBSYSTEMS AS DESIGNATED ON THIS DRAWING
4. SEE GAS FLOW BALANCE SHEETS. APPENDIX C FOR GAS QUANTITIES
5. ALL VALVES ARE NORMALLY OPEN UOS
6. ALL TEMPERATURE CONNECTIONS SHALL BE 1" UOS
7. ALL PIPE SIZES ARE IN INCHES
8. SYMBOLS AND ABBREVIATIONS AARD-D-0036
9. BLIND FLANGE TO BE REMOVED DURING DE-INERTING OF INDIVIDUAL CELL. AIR INTRODUCED FROM RCB NORMAL ATMOSPHERIC AREAS
10. BLIND FLANGE TO BE REMOVED DURING DE-INERTING OF INDIVIDUAL CELL AND FLEXIBLE DUCT CONNECTED TO SYSTEM 25 PORTABLE PURGE FAN AND DISCHARGED TO SYSTEM 25 PURGE EXHAUST DUCT
11. PIPING BETWEEN THE AUTOMATIC ISOLATION VALVES AND THE LINER OF CELLS SERVED INCLUDING THE ISOLATION VALVES SHALL BE SEISMIC CATEGORY I

POOR ORIGINAL

REFERENCE DRAWINGS

1. P&ID NORMAL CHM SYSTEM B&R DWG NN557
2. P&ID EMERGENCY CHM SYSTEM SGB, RCB B&R DWG NN561
3. P&ID EMERGENCY CHM SYSTEM RCB, RSB & SGB B&R DWG NN562
4. RSB, RCB & SGB NA & NAK PUMP INTERFACE (SYS 82) AI DWG CA53010
5. RCB N_2 DISTRIBUTION P&ID AI DWG 99821055
6. P&ID RCB EQUIPMENT & FLOOR DRAINS, EAST SIDE B&R DWG NN517
7. P&ID RCB EQUIPMENT & FLOOR DRAINS, WEST SIDE B&R DWG NN518

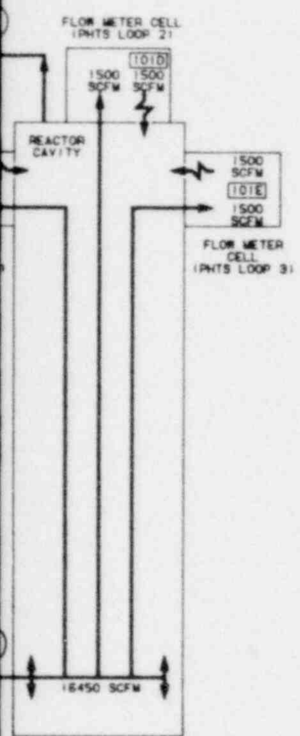
Figure 9.16-4 RCB RGCS



SUBSYSTEM RC	COOLING COIL	COOLER	FAN CASING	EX-CELL PIPING
SAFETY CLASS	NONE	NONE	NONE	NONE
SEISMIC CAT	I	I	I	I
RADIATION ZONE	[illegible]	[illegible]	[illegible]	[illegible]
DESIGN CODE-ASME ANSI	[illegible]/3	[illegible]	[illegible]	B31.1
CLEANLINESS CLASS ANSI 45.2.1	C	C	C	C

NN577-6

POOR ORIGINAL



GENERAL NOTES

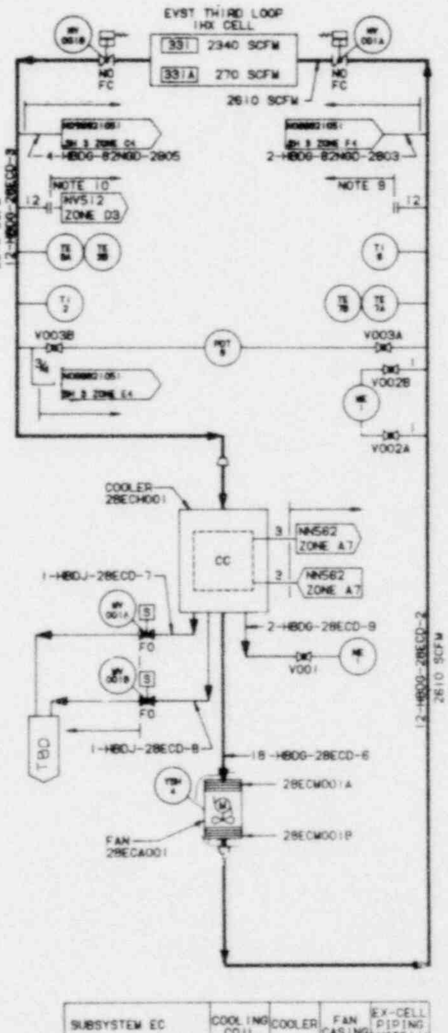
1. ALL PRESSURE CONNECTIONS SHALL BE 3/4" UOS
2. ALL EQUIPMENT NUMBERS AND MANUAL VALVE NUMBERS ARE PREFIXED BY 28 FOLLOWED BY SUBSYTEMS AS DESIGNATED ON THIS DRAWING
3. ALL INSTRUMENT NUMBERS AND CONTROL VALVE NUMBERS ARE PREFIXED BY 28 FOLLOWED BY SUBSYSTEMS AS DESIGNATED ON THIS DRAWING
4. SEE GAS FLOW BALANCE SHEETS, APPENDIX C FOR GAS QUANTITIES
5. ALL VALVES ARE NORMALLY OPEN UOS
6. ALL TEMPERATURE CONNECTIONS SHALL BE 1" UOS
7. ALL PIPE SIZES ARE IN INCHES
8. SYMBOLS AND ABBREVIATIONS RARD-D-0036
9. BLIND FLANGE TO BE REMOVED DURING DE-INERTING OF INDIVIDUAL CELL. AIR INTRODUCED FROM RCB NORMAL ATMOSPHERIC AREAS
10. BLIND FLANGE TO BE REMOVED DURING DE-INERTING OF INDIVIDUAL CELL AND FLEXIBLE DUCT CONNECTED TO PORTABLE PURGE FAN AND DISCHARGED TO PURGE EXHAUST DUCT

POOR ORIGINAL

REFERENCE DRAWINGS

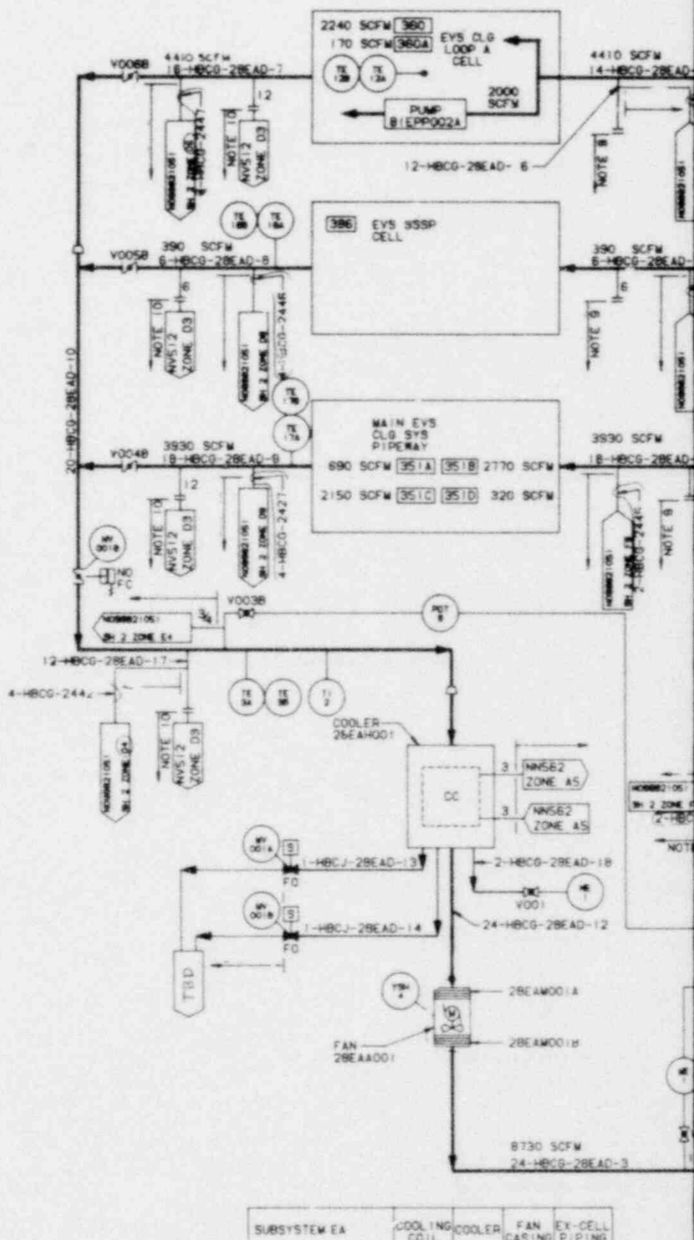
1. P&ID NORMAL CHW SYSTEM RCB & SGB B&R DWG NM558
2. RCB N_2 DISTRIBUTION P&ID AI DWG NOR08621055
3. P&ID RCB EQUIPMENT & FLOOR DRAINS, EAST SIDE B&R DWG NM517

Figure 9.16-5 RCB RGCS



SUBSYSTEM EC	COOLING COIL	COOLER	FAN CASING	EX-CELL PIPING (NOTE 11)
SAFETY CLASS	NONE	NONE	NONE	NONE
SEISMIC CAT	I	I	III	III
RADIATION ZONE	I	I	I	I
DESIGN CODE-ASME ANSI	III/3	VIII	VIII	B31.1
CLEANLINESS CLASS ANSI 45.2.1	C	C	C	C

NN586-8

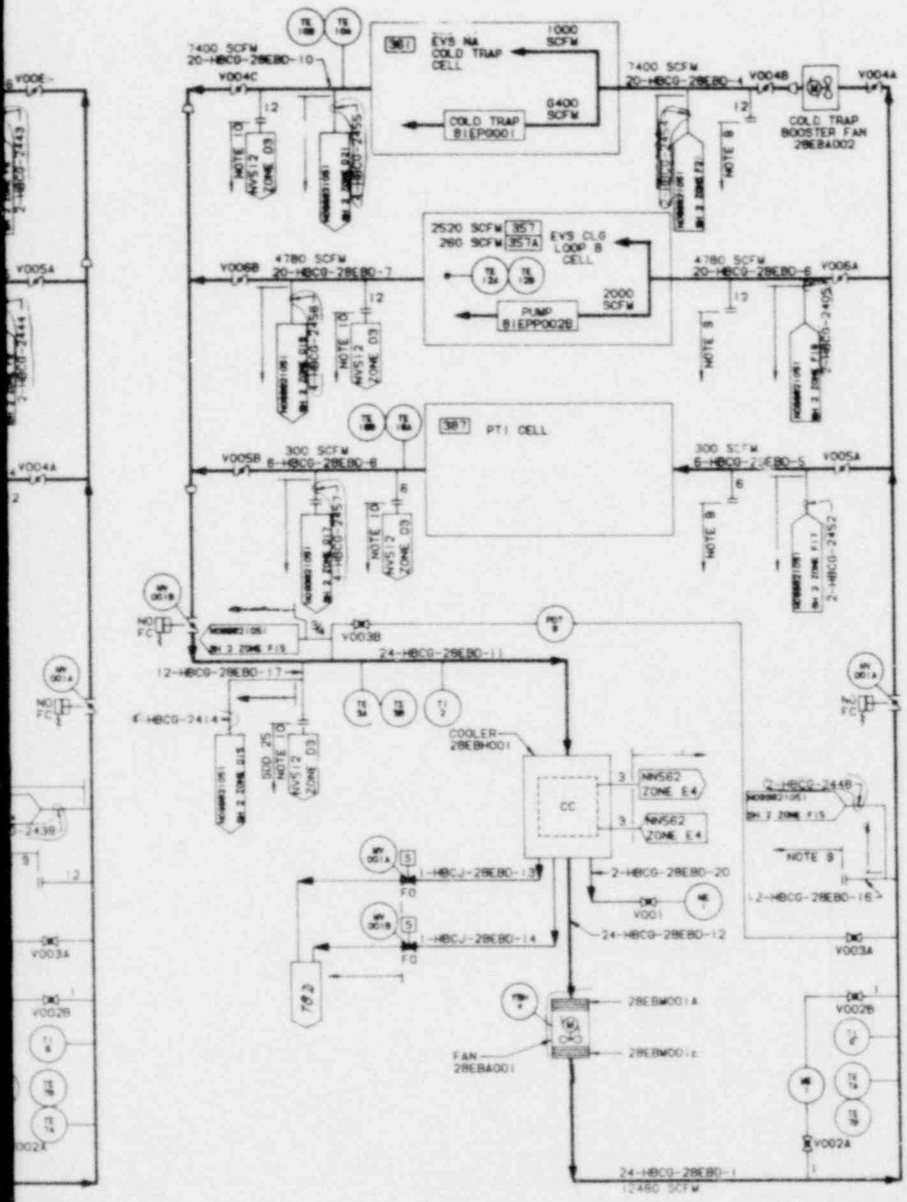


SUBSYSTEM EA	COOLING COIL	COOLER	FAN CASING	EX-CELL PIPING
SAFETY CLASS	SC-3	SC-3	SC-3	SC-3
SEISMIC CAT	I	I	I	I
RADIATION ZONE	I	I	I	I
DESIGN CODE-ASME ANSI	III/3	III/3	III/3	III/3
CLEANLINESS CLASS ANSI 45.2.1	C	C	C	C

POOR ORIGINAL

GENERAL NOTES

1. ALL PRESSURE CONNECTIONS SHALL BE ¾" UNLESS OTHERWISE NOTED
2. ALL EQUIPMENT NUMBERS AND MANUAL VALVE NUMBERS ARE PREFIXED BY 28 FOLLOWED BY SUBSYSTEMS AS DESIGNATED ON THIS DRAWING
3. ALL INSTRUMENT NUMBERS AND CONTROL VALVE NUMBERS ARE PREFIXED BY 28 FOLLOWED BY SUBSYSTEMS AS DESIGNATED ON THIS DRAWING
4. SEE GAS FLOW BALANCE SHEETS. APPENDIX C FOR GAS QUANTITIES
5. ALL VALVES ARE NORMALLY OPEN UNLESS OTHERWISE NOTED
6. ALL TEMPERATURE CONNECTIONS SHALL BE 1" UNLESS OTHERWISE NOTED
7. ALL PIPE SIZES ARE IN INCHES
8. SYMBOLS AND ABBREVIATIONS AARD-D-CD36
9. BLIND FLANGE TO BE REMOVED DURING DE-INERTING OF INDIVIDUAL CELL. AIR INTRODUCED FROM RSB NORMAL ATMOSPHERIC AREAS
10. BLIND FLANGE TO BE REMOVED DURING DE-INERTING OF INDIVIDUAL CELL AND FLEXIBLE DUCT CONNECTED TO [illegible] PORTAB[illegible] FAN AND DISCHARGED TO [illegible] PURGE EXHAUST DUCT
11. PIPING BETWEEN THE AUTO[illegible]IC ISOLATION VALVES AND THE [illegible]INER OF CELLS SERVED INCLUDING THE ISOLATION VALVES SHALL BE SEISMIC CATEGORY I

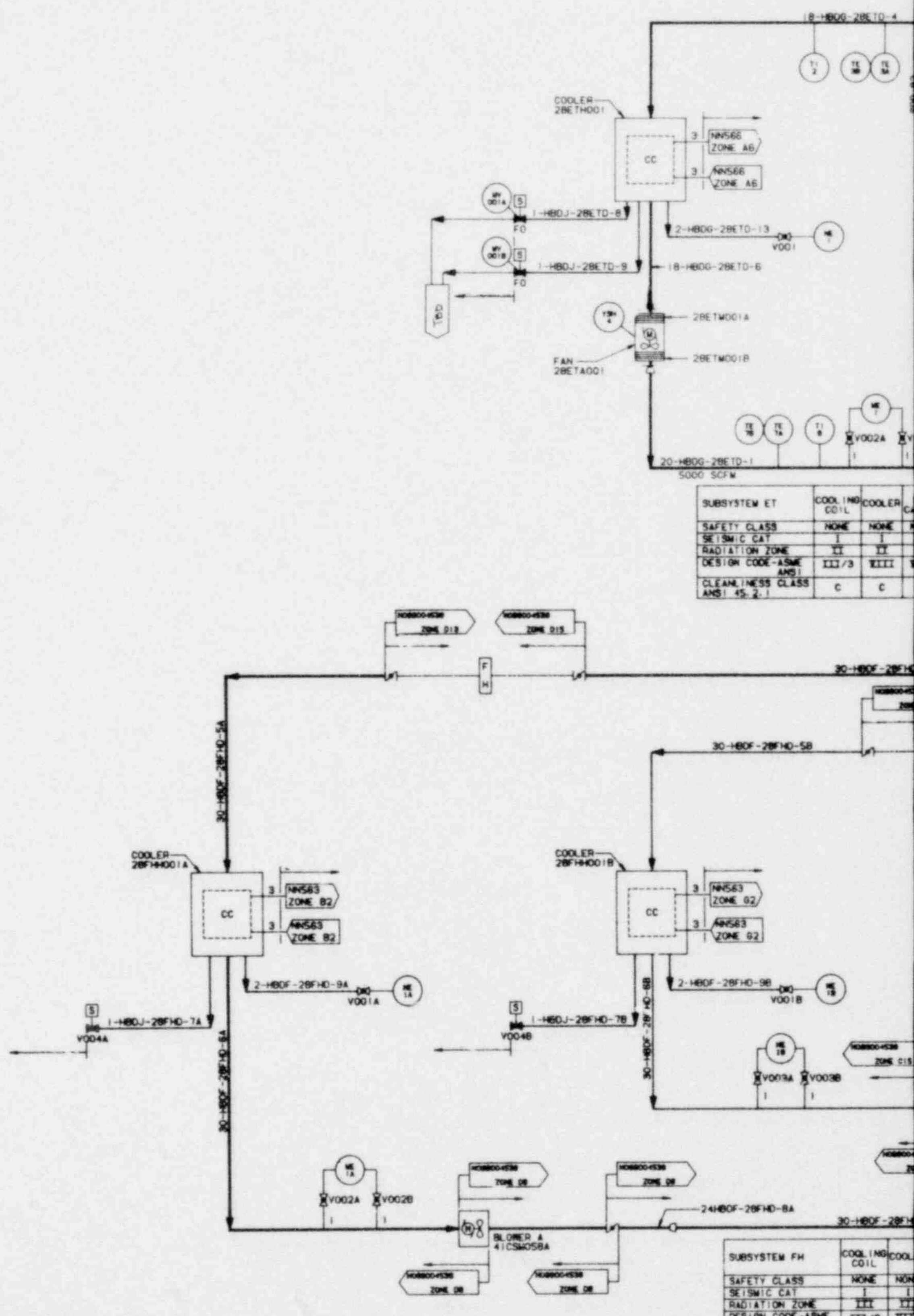


SUBSYSTEM EB	COOLING COIL	COOLER	FAN CASING	EX-CELL PIPING
SAFETY CLASS	SC-3	SC-3	SC-3	SC-3
SEISMIC CAT	I	I	I	I
RADIATION ZONE	☐	☐	☐	☐
DESIGN CODE-ASME ANSI	III/3	III/3	III/3	III/3
CLEANLINESS CLASS ANSI 45.2.1	C	C	C	C

REFERENCE DRAWINGS

1. RSB N_2 DISTRIBUTION P&ID
   AI DWG NO99821051
2. RSB, RCB & SGB NA AND NAK PUMP INTERFACE
   AI DWG CAS3010
3. RSB-EVS NA COLD TRAP INTERFACE
   AI DWG CAS3013
4. P&ID EMERGENCY CHW SYSTEM RCB, RSB & SGB
   B&R DWG NN562

Figure 9.16-6 RSB RGCS

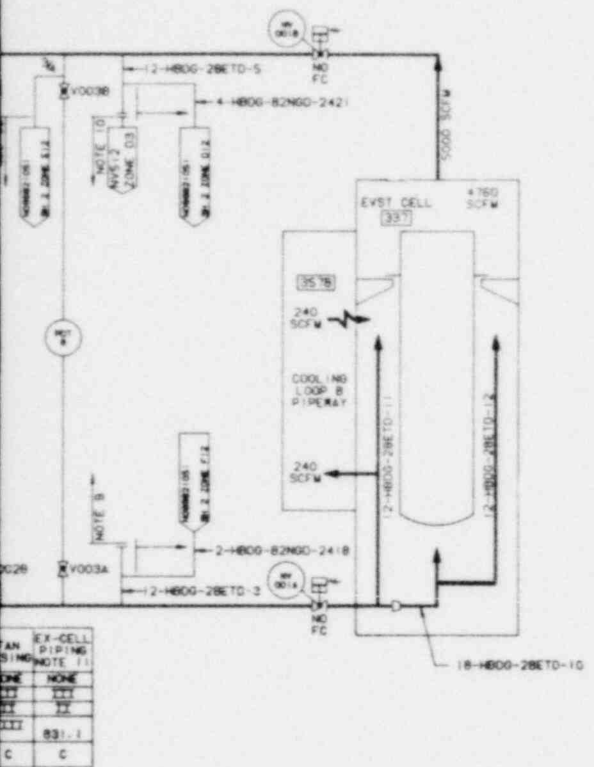


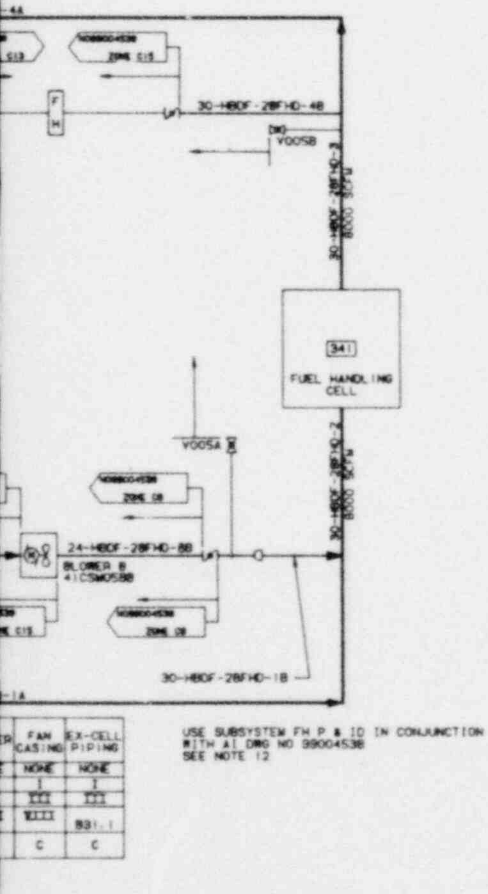
SUBSYSTEM ET	COOLING COIL	COOLER
SAFETY CLASS	NONE	NONE
SEISMIC CAT	I	I
RADIATION ZONE	II	II
DESIGN CODE-ASME ANSI	III/3	VIII
CLEANLINESS CLASS ANSI 45.2.1	C	C

SUBSYSTEM FH	COOLING COIL	COOL
SAFETY CLASS	NONE	NON
SEISMIC CAT	I	I
RADIATION ZONE	III	III
DESIGN CODE-ASME ANSI	III/3	VIII
CLEANLINESS CLASS ANSI 45.2.1	C	C

NN587-8

POOR ORIGINAL





GENERAL NOTES

1. ALL PRESSURE CONNECTIONS SHALL BE 3/4" UNLESS OTHERWISE NOTED
2. ALL EQUIPMENT NUMBERS AND MANUAL VALVE NUMBERS ARE PREFIXED BY 28 FOLLOWED BY SUBSYSTEMS AS DESIGNATED ON THIS DRAWING
3. ALL INSTRUMENT NUMBERS AND CONTROL VALVE NUMBERS ARE PREFIXED BY 28 FOLLOWED BY SUBSYSTEMS AS DESIGNATED ON THIS DRAWING
4. SEE GAS FLOW BALANCE SHEETS, APPENDIX C FOR GAS QUANTITIES
5. ALL VALVES ARE NORMALLY OPEN UNLESS OTHERWISE NOTED
6. ALL TEMPERATURE CONNECTIONS SHALL BE 1" UNLESS OTHERWISE NOTED
7. ALL PIPE SIZES ARE IN INCHES
8. SYMBOLS AND ABBREVIATIONS WARD-D-0098
9. BLIND FLANGE TO BE REMOVED DURING DE-INERTING OF INDIVIDUAL CELL. AIR INTRODUCED FROM RSB NORMAL ATMOSPHERIC AREAS
10. BLIND FLANGE TO BE REMOVED DURING DE-INERTING OF INDIVIDUAL CELL AND FLEXIBLE DUCT CONNECTED TO [illegible] PORTABLE PURGE FAN AND DISCHARGED TO [illegible] PURGE EXHAUST DUCT
11. PIPING BETWEEN THE AUTOMATIC ISOLATION VALVES AND THE LINER OF CELLS SERVED INCLUDING THE ISOLATION VALVES SHALL BE SEISMIC CATEGORY I
12. SUBSYSTEM FH P&ID MUST BE USED IN CONJUNCTION WITH AI DWG N099004538 WHICH PROVIDES DETAILED INSTRUMENTATION & INTERFACE WITH [illegible] INSTRUMENTATION SHOWN ON THIS P&ID IS INSTRUMENTION ONLY

POOR ORIGINAL

REFERENCE DRAWINGS

1. P&ID EMERGENCY CHR SYSTEM SEC COOL LOOP RSB
   B&R DWG NM583
2. P&ID NORMAL CHR SYSTEM RSB
   B&R DWG NM586
3. RSB N_2 DISTRIBUTION P&ID
   AI DWG N09982105I
4. ARGON CIRCULATION SYSTEM P & ID
   AI DWG N099004538

Figure 9.16-7 RSB RGCS

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The bulk storage area is completely enclosed by protective fencing. Administrative and safety procedures are enforced to minimize the possibility of fires or explosions. Access by unauthorized personnel is controlled by permanent warning placards and a locked gate.

Hydrogen feed to the generator is manually initiated on an "as required" basis to maintain design pressure. A normally closed automatic shut-off valve is provided at the bulk storage facility and at the local hydrogen control panel. Each shutoff valve is opened by means of a palm operated spring return valve located near the local control panel. The operator must depress both palm operated valves simultaneously to obtain hydrogen feed to the turbine generator from storage. Release of palm operated valves automatically shuts off hydrogen feed.

Provisions are included for purging of the system. Flammable gas safety precautions are utilized. Initial purging of the bulk Hydrogen Storage System is achieved with carbon dioxide following evacuation of the cylinder and dead end piping legs. The system is charged with hydrogen only after purging and when the oxygen concentration has been reduced to within the permissible 0.5 percent by volume before hydrogen filling.

The stator liquid cooling system includes a storage tank, pumps, filters, heat exchangers, deionizers, and associated valves and control equipment. Two full capacity pumps provide 100% back-up protection with automatic start-up of the emergency pump, if needed. Two coolers provide flexibility of operation and maintenance since one cooler can be removed from service while the other carries the full load.

10.2.3 Turbine Missiles

The turbine generator for the CRBRP is of a new design for nuclear units, i.e., a TC6F-23 machine operating at 3,600 rpm. The overall arrangement of the Turbine-Generator Building is such that the axis of the turbine-generator unit is oriented so that its rotation is perpendicular to the reactor containment and support buildings. The possibility of high energy, low angle turbine generated missiles reaching those critical areas is minimal.

Since the turbine-generator is a new design, previous turbine missile analyses for nuclear plants are not applicable. Missile data for the CRBRP has been developed, however, to provide the basis for analysis of the probability of high energy, low angle turbine-generated missiles penetrating or perforating vital plant areas.

The probability of significant damage to the critical systems and vital structures of CRBRP due to missiles from its single-unit turbogenerator can be assessed by determining the combined probabilities of turbine failure and ejection of an external missiles (P_1), a missile from the turbine striking a critical component (P_2), and significant damage occurring to the target component (P_3). Thus the overall probability (P_4) = (P_1) x (P_2) x (P_3). 12

Ejection Probability (P_1)

The probabilities of missile generation due to turbine failure at; (a) design overspeed (below 120% speed), and (b) destructive overspeed, are as
58 follows (PSAR Section 1.6, Ref. 9):

Low Speed Failure (Below 120% Speed)	Not Statistically Significant
Runaway Failure (at Higher Speed)	5.7×10^{-8}
58 Total Lifetime Failure Probability	5.7×10^{-8}

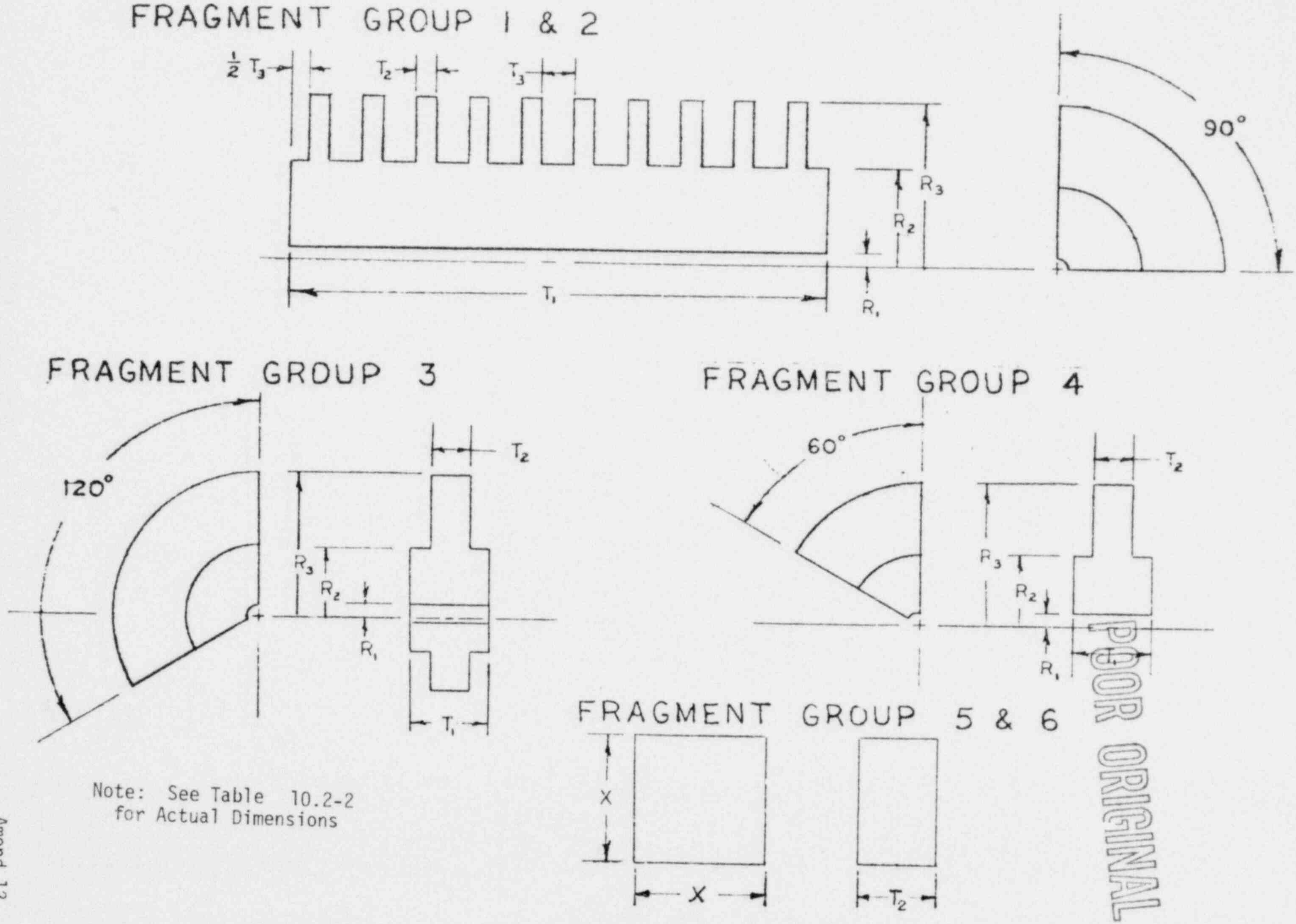
The corresponding average annual probability of rotor failure is found to be
58 1.9×10^{-9}.

59 58 Table 10.2-1 (PSAR Section 1.6, Ref. 9) summarizes the results of the analysis and lists the major assumptions on which the combination of probabilities is based.

The hypothetical missile data, listing the weight, velocity and energy ranges of the postulated missile fragments, are summarized in Table 10.2-2
58 (PSAR Section 1.6, Ref. 9).

The fragments, ejected from the L-P turbine last-state rotor failure, are grouped by the turbine manufacturer into potential missiles of six size classifications according to their geometry. The configurations of these
58 postulated missile fragments are given in Figure 10.2-1 (PSAR Section 1.6,
Ref. 9). 12

Figure 10.2-1 Turbine Missile Dimensions



Note: See Table 10.2-2 for Actual Dimensions

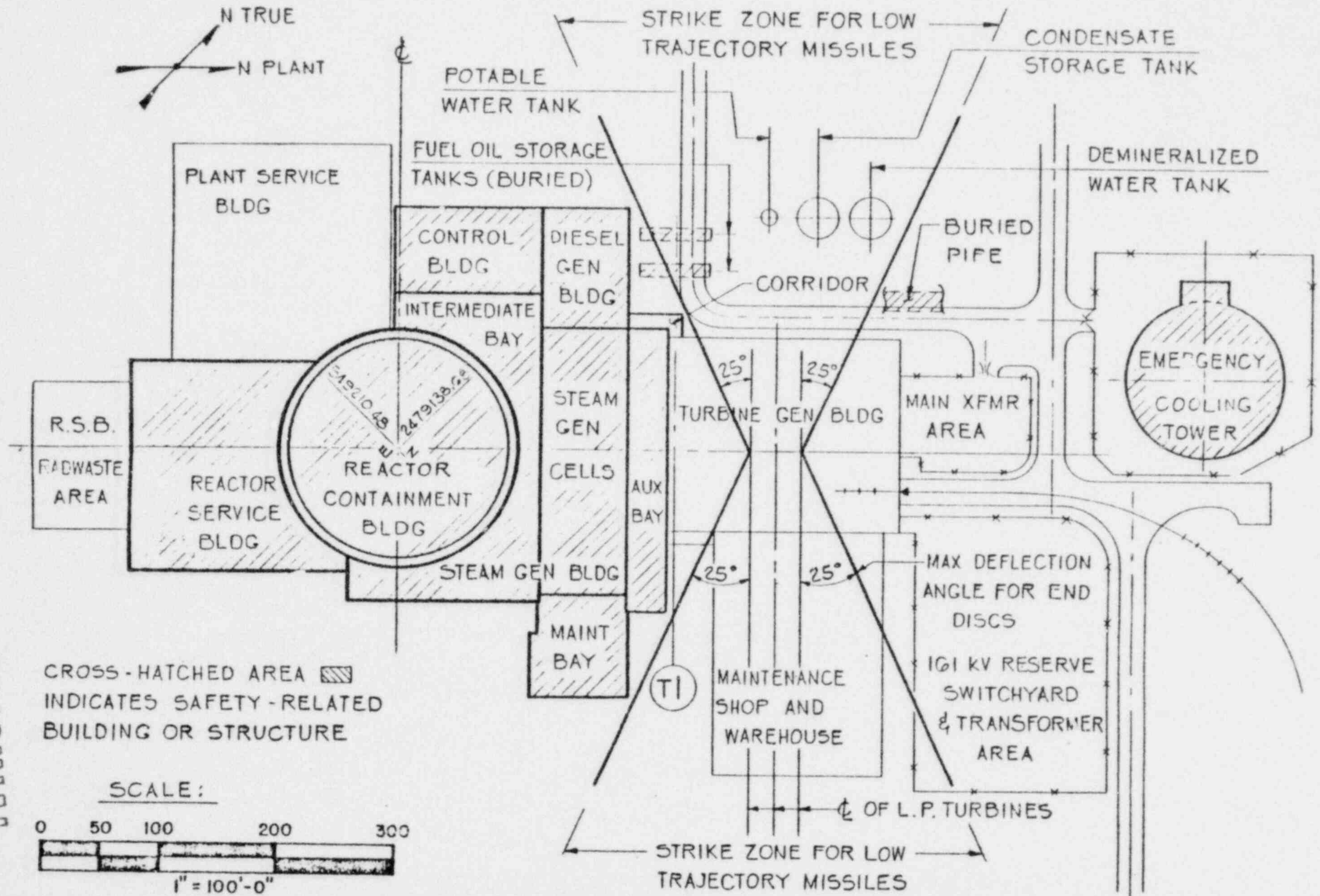


FIGURE 10.2-2 PLAN VIEW OF CRBRP NUCLEAR ISLAND STRUCTURES

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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.

Path 6. Other plant systems, specifically Refueling, Maintenance and
50 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
59 spaces by distillation 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 flow. The 5.15
59 scfm output of RAPS is delivered to the recycle argon vessels. The RAPS cryogenic distillation column operates with liquid argon as the still bottom. The krypton and xenon isotopes accumulate in the bottoms, which permits their removal by draining, evaporating, and transferring
59 them periodically 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
50 gases through the piping and components. This source of 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 gas will be leaked
50 into the cells; further, they will be exhausted to CAPS without delay, except that Ne23 is assigned a delay of 8 minute . The RSB inerted cells atmosphere will divert from HVAC to CAPS, for processing when activity concentrations exceed a predetermined setpoint value. For RCB cells,
59 normal cell-atmosphere nitrogen passes directly to 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 operations, 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
59 50 maintenance, RAPS 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
50 a repair or correction. 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 distilla-
tion-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 in
RAPS permits maintaining a steady throughout under conditions of changing
59 49 output demand requirements.

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
59 of 1% failed fuel is the design base point for RAPS.

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

values listed under "design" correspond to the design base condition of 1% failed fuel. The values listed under "expected" correspond to operation with 0.1% failed fuel. The effective decontamination factors of RAPS are given in Table 11.3-5.

c. CAPS Process Stream - Tables 11.3-6 and 11.3-7 list, respectively, the isotope inventories and concentrations of activity at selected points in the CAPS process stream. The effective decontamination factors of CAPS are given in Table 11.3-8.

For all tables involving the CAPS inputs, CAPS processes, and radioactive releases through CAPS, the terms "Design" and "Expected" are further defined and limited as follows:

"Design": The CAPS input activities correspond to the design base condition of reactor operation with 1% failed fuel after one year; in addition, the Failed Fuel Monitoring System
59 input corresponds to operation with 60 reactor failed fuel pins and, eight samples per day being processed.

"Expected": The CAPS input activities correspond to the expected condition of reactor operation with 0.1% failed fuel after one year; the Failed Fuel Monitoring input corresponds to 0.1% failed fuel and two samples per day being processed.

11.3.2.4 Release Path Calculations

The various release paths listed in Section 11.3.2.1, Paths 1 through 10, and shown in Figure 11.3-1, were evaluated as follows to obtain the activity release rates.

55 Path 1a. Reactor Cover Gas Leakage - The leakage of 0.0044 scc/min was multiplied by the data in Table 11.3-2 ("Concentration" column) and was converted to give daily leakage. Decay during transit through seals was not considered, except for Ne^{23} (5 minute delay).

Path 1b. Buffered Head Seals Leakage - The leakage of 7 scc/min from the seals was multiplied by the data in Table 11.3-4 ("Recycle Argon Vessel Effluent" column) and was converted to give daily leakage.

Path 2. Primary Piping Leakage - The leakage of 1 scc/min of reactor cover gas from the equalization line and hot primary gas piping was multiplied by the data in Table 11.3-2 ("Concentration"
49 column) and was divided by the CAPS decontamination factors (Table 11.3-8). Diffusion of tritium is also included.

Path 3. RAPS Cold Box Component Leakage - The leakage of 1 scc/min was multipled by the data in Table 11.3-4 ("RAPS Cold Box Influent" column) and was divided by the CAPS decontamination factors (Table 11.3-8).

Path 4. RAPS Noble Gas Bleed to CAPS - Based on a bleed period of one year, an average daily radioactivity release rate to CAPS was calculated for the Kr^{85} and Ar^{39}. This was then assumed to pass through CAPS with no attenuation.

Path 5. Impurity and Failed Fuel Monitoring - These contributions to the CAPS input were divided by the CAPS decontamination factors (Table 11.3-8).

Path 6. Gas Services Exhaust - For each of the three interfacing plant systems, the respective radioactive isotopic inputs to CAPS were divided by the CAPS decontamination factors (Table 11.3-8).

Path 7. Intermediate Bay Cells Leakage - The diffusion of tritium is calculated from an analysis, as indicated in Section 11.3.2.1, involving the evaluation of both tritium and hydrogen fluxes in the Primary and Intermediate Sodium Systems, and using appropriate assumptions of cold-trapping efficiency, and the hydrogen-tritium permeability of components and piping.

Path 8. RCB H&V Exhaust - The sum of Paths 1a and 1b.

Path 9. CAPS H&V Exhaust - The sum of Paths 2 through 6.

Path 10. T-G Building H&V Exhaust - BOP tritium release is discussed in Section 11.3.6.2.

11.3.2.5 Activity Release Tabulations

Tables 11.3-9 and 11.3-10 tabulate for each radionuclide the design and expected daily releases, respectively; the first six data columns represent Release Paths 1a, 1b, and 2 through 5 in the order given in Section11.3.2.4; the next three columns represent Path 6.

Table 11.3-11 gives annual releases for the design service condition (1% failed fuel). The first three data columns correspond to Release Paths 8, 9, and 7, in that order; the last column gives the total plant release through the H&V system. Table 11.3-12 gives the
49 annual releases for the expected service condition of 0.1% failed fuel.

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.

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
50 ventilating air, resulting in an expected concentration of 3.9×10^{-9} $\mu Ci/cm^3$, 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
5L (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

59 The RAPS and CAPS System designs emphasize all-welded construction, wherever practicable, and bellows-sealed process 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
59 normal operating conditions.

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.

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). RAPS and CAPS process and instrumentation diagrams are shown in Figures 11.3-10 through 11.3-13. The normal pressures and temperatures for the vessels
50 are listed in Table 11.3-17. 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
59 50 individual looP cell, to prevent release of radioactive gases.

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 μCi/cm) 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 $\mu Ci/cm^3$), 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
59 | 49 is not held below 12.7 psia. The compressors stop in sequence when the vacuum

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
59 inadequate cooling, a high alarm will be triggered to alert the opera-
50 tor to the abnormal condition.

RAPS and CAPS are independently operated, with process control
being automatic and with local provisions for overriding automatic
controls if conditions so dictate. Both subsystems have control and
59 alarm instrumentation. Also instant data retrieval is 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
59 operates at about 40 psig but can be operated up 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
59 that increases with surge-vessel pressures 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, between
40°F and 70°F, it is drained to the CAPS rad-water holding vessel. From
this vessel, it is periodically transferred to the Radioactive Waste
59 49 System by manual actuation of the transfer valve.

The dried, essentially tritium-free gas from the on-line unit is passed through two charcoal delay beds at cryogenic temperatures; this adsorbs the radioactive xenon and krypton and permits their decay on the beds. The effluent gas from the second bed is normally released to the H&V, but if it contains radioactivity greater than the preset limit, the gas stream is automatically diverted back to the CAPS vacuum vessel and an alarm is sounded. The operator may allow the system to continue to operate with the effluent diverted, or he may manually reset the flow controller to reduce the radioactive gas input to the tritium-water removal unit and the delay beds. When the radioactivity concentration in the effluent gas is reduced to a value below the preset value, as observed in the effluent gas radiation monitor, the three-way valve is returned to its normal position to divert the gases to H&V. However, the alarm signal must be manually reset; this ensures that the operator will be aware of the high reading and can take further action in the event of a prolonged signal or of a series of signals.

RAPS is designed to process highly radioactive cover gas from the Primary Sodium System. This system contains essentially the entire inventory of radioactive gas in the plant (excluding that within the fuel assemblies). Under normal operation, RAPS does not discharge gas to the environment. Any unacceptable leakages from RAPS are collected in cells whose atmospheres are processed in CAPS before being released
59 to the environment. RAPS is also divided into an inlet complex and a processing section. The RAPS inlet complex, consisting of a vacuum vessel, filters, and compressors, operates in the same manner and serves the same function as that in CAPS. A cryogenic still is used to remove stable xenons and kryptons, and Kr^{85}, as well as the short-lived noble gas isotopes.

RAPS automatically processes argon reactor cover gas, which, when purified, is normally delivered to the recycle vessels. If the radioactivity level of the RAPS effluent gas exceeds a preset level, an alarm is sounded. This alerts the operator, who then manually closes the inlet valve to the cold box, and opens the cold-box bypass line.
59 The time limit on use of the bypass is 3 weeks at 0.3% failed fuel. After corrective action has been completed, normal flow will be resumed and the effluent will again be directed to the recycle argon storage vessels.

The RAPS cryogenic still removes xenon and krypton isotopes by solution in the liquid argon that collects at the bottom of the column. This liquid is periodically drained, evaporated, and transferred as gas to the noble gas storage vessel, from which the gas is bled into CAPS at a controlled rate.

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Off-normal operation of the cryogenic still will be detected and an alarm sounded on the basis of off-normal temperature, pressure, or flow. The first operator action would be to attempt to correct the condition by adjusting the out-of-limit parameter. If this could not be accomplished, the RAPS cold box would be bypassed as described above.

Although under normal conditions RAPS is a closed system, the following are two conditions under which gas may be bled from the circuit (1) in the event the RAPS system pressure rises above a preset limit, the operator can manually discharge some of the RAPS gas inventory to CAPS as a corrective measure; and (2) prior to required maintenance, the gas inventory in RAPS components can be discharged to CAPS. In all cases, the diverted RAPS gas is taken to CAPS for additional treatment. If the radioactivity concentration in the CAPS effluent stream at any
59 time exceeds the allowable amount, the gas is automatically recycled back to the CAPS vacuum vessel, and an alarm will sound.

Performance tests, to be conducted prior to plant operation, will verify the operability of the system. Selected system performance data are to be recorded during plant operation; these are to be continually reviewed to ascertain that critical components are performing within specifications. In addition, the process gas streams are to be continuously monitored for radioactivity level, in order to provide surveillance of the process system's decontamination effectiveness. The calibrations of all instrumentation sensors, and checkout of alarm circuits and controls, are to be performed during reactor downtime, in order to assure that they are functioning properly. Selected tests are to be repeated during scheduled plant maintenance periods in order to
59 ensure that critical components are performing within specifications.

11.3.5 Estimated Releases

The Cell Atmosphere Processing Subsystem removes fission product gases from nitrogen cell atmospheres, and those air cell atmospheres that normally or potentially contain radioactivity, prior to the discharge of these gases to the environment. The direct discharge of unprocessed radioactive gases to the environment is limited to the radioactive cover gas and buffer-gas leakages through various reactor closure head seals, and to the tritium that diffuses through the piping
49 and components in the Intermediate Bay cells which vent directly into the SGB-IB H&V exhaust system.

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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.

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
49 the Reactor Confinement Building.

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 6.3×10^{-5} Ci/day.

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

Thirteen other release points associated with the balance of plant could contain some radioactivity. These points are:

1) Plant Service Building (PSB) exhausts from the hot laboratory and decontamination area, identified as Point 19 on Figure 11.3-9.
59 Levels or radioactivity in this area will make no significant contribution to off-site dose rates.

2) Turbine Generator Building exhausts receiving ventilation exclusively from the Turbine Generator Buidling atmosphere are identified as Points 7 thru 18 on Figure 11.3-9. Levels of radioactivity in these areas are expected to make no significant contribution to off-site dose rates. However, as per Section 11.4.2.2.3, samples of the TGB atmosphere will periodically be analyzed.

The location, height, discharge flow rate, discharge velocity, discharge air temperature, and size and shape of the discharge orifice, for each BOP release point, are presented in Table 11.3-20.

11.3.7 Dilution Factors

The maximum dose at the site boundary due to normal releases from the gaseous waste system will occur at a point on the boundary that has the highest average annual x/Q as determined from meteorological data. For the CRBRP site, the average annual x/Q for this point is 5.10×10^{-5} s/m^3.

11.3.8 Dose Estimates

The release of radioactive noble gases in the gaseous effluent from the CRBRP during normal operation will create a slightly radioactive plume downwind of the site; this will expose the public located in the downwind direction to small doses of gamma and beta radiation. It
59 should be noted that these doses are calculated assuming the public is completely exposed to the environment, whereas, in reality, most persons spend a significant portion of the lives within structures that reduce the exposure of these types of radiation. The reduction in external dose could range from a factor of two to 1,000 depending on the type of structure and the location of the person within (Ref. 1).

Exposure to tritium in the form of tritiated water vapor (HTO) can occur through several pathways including:

1) Inhalation and skin absorption

49 2) Ingestion of milk contaminated by the fallout of HTO to the cow's forage and by inhalation by the cow

3) Ingestion of vegetables contaminated by HTO fallout

4) Ingestion of meat (beef) through fallout to forage and inhalation of grazing cattle

5) Ingestion of drinking water contaminated by HTO fallout.

11.3.8.1 Dose Rate Estimates

Doses received from exposure to gaseous effluents from the CRBRP were evaluated, using equations 1 through 12 presented in the Appendix to Section 11.3 and released as described in Table 11.3-11 (design conditions). Maximum external gamma and beta doses are expected, assuming continuous exposure at the site boundary location associated with the largest value of average annual x/Q. The associated distance is 2500 feet in the NW direction, and the average annual $x/Q = 5.1 \times 10^{-5}$ s/m^3. Although activity is released from rooftop vents, analyses
assume ground releases. External gamma dose is 0.55 mrem/yr and total
55 skin dose is 4.2 mrem/yr at this point, assuming no protection from
clothing. This annual dose is well below the requirements of 10 CFR 50,
59 Appendix I. Release point contributions to the dose are
tabulated in Table 11.3-18.

Baased on the population distribution for the year 2010 within 50 miles of the site, as presented in Table 2.1-12, the annual population dose associated with the external gamma dose is 0.9 man-rem/yr. Assuming a conservative value of 100 mrem as the average annual dose from naturally occurring external sources of radiation (Section 11.6), the associated population dose due to naturally occurring radioactivity is estimated to be 98,700 man-rem/yr. The calculated contribution from the CRBRP is less than 0.001 percent of the population dose from naturally occurring radioactivity.

Internal doses via the various exposure pathways to gaseous effluents (inhalation and ingestion of milk, water, vegetables, and meat) will be due almost exclusively to the presence of tritium. The noble gases are relatively inert and result in practically no internal exposure. Internal doses are reported in Table 11.3-19 on a release point basis. All dose calculations have included the BOP tritium contribution.

The growing season for leafy vegetables in the Eastern Tennessee Region is assumed to be 90 days. All other variables used in the calculation of dose from ingestion of leafy vegetables, such as total daily intake of leafy vegetables and yield per unit area of cultivated
49 land, are provided in Table 11.3A-6 of the Appendix to Section 11.3.

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
59 52 H^3	6.6E-5	6.6E-5	1.6E-3	1.6E-3	5.4E-4	5.4E-4	9.5E-4	9.5E-4
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

TABLE 11.3-16

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

	MPC* (μCi/cc)	Concentration (μCi/cc)	Concentration ÷ MPC
59 49	2E-7	1.5E-12	7.5E-6

TABLE 11.3-16a

TRITIUM CONCENTRATION AT SITE BOUNDARY
FROM IB EXHAUST

	MPC* (μCi/cc)	Concentration (μCi/cc)	Concentration ÷ MPC
59 49	2E-7	9.4E-14	4.7E-7

*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

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
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

*Does not apply because of adsorption variable 9
**ASME Section III, Class 3 - III-3
+Design Code listed may be higher for reasons other than safety
++Saturation Temperature at normal operating pressure

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
6 Room analysis of effluent samples obtained at each location listed above.

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). A monitor
59 will be located at the CAPS inlet 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 10 CFR 50, App. I and ALARA limits for
59 those gases exhausted to the H&V system by CAPS.

e&f. Analysis of samples of air, particulate matter, soil, vegetation, food crops, and milk will be used to estimate the dose to the surrounding popu-
57 lation through the consumption of food or dairy products.

The environmental monitoring program to be conducted throughout operation of the plant provides the necessary means of evaluating the dose to man through critical exposure pathways.

Environmental concentrations of radioactivity due to plant releases to unrestricted areas may be so low as to be unmeasurable with present techniques. Therefore, methods to calculate the potential exposure to man have been derived for both gaseous and liquid releases.

11.6.2.1 Doses from Gaseous Effluents

The following doses to humans living in the vicinity of the CRBRP will be calculated for the releases of radioactive gases:

a. External beta- and gamma-air doses from airborne radioactivity

57 b. Total body and skin doses from direct radiation due to ground contamination

c. Internal doses from inhalation

d. Internal doses from ingestion

The basic assumptions and calculational methods that will be used in computing these doses are similar to that described in the appendix to Section 11.3.

57 Review of the data resulting from the offiste monitoring program and reevaluations of the adequacy of the dose models will verify that the actual doses received by individuals and the population as a whole remain within the applicable Federal Regulations and as low as reasonably achievable.

11.6.2.2 Internal Doses from Liquid Effluents

The following doses will be calculated for exposures to radionuclides routinely released in liquid effluents:

a. Internal doses from the ingestion of water

b. Internal doses from the consumption of fish

57 c. External and internal doses from water sports

d. External doses from shoreline activities

The basic assumptions and calculational methods that will be used in computing these doses are similar to that described in the appendix to Section 11.2.

The dose models that are employed will be reevaluated in light of the data resulting from the offsite monitoring program to ensure that all signi-

57 ficant pathways are included in the calculations and to verify that the actual
doses received by individuals and the population as a whole remain within the
38 applicable Federal Regulations and as low as reasonably achievable.

11.6.3 Sampling Media, Locations, and Frequencies

The sampling media, the locations from which the samples are
collected, and the frequency with which the samples are collected are presented
in Table 11.6-1. Tentative sampling locations are shown in Figures 11.6-1 and
57 11.6-2. The final selection of sampling locations will be made approximately one
year prior to implementation of the program. The media selected were chosen on
two bases: First, those vectors which would readily indicate significant
increases in radioactivity levels, and secondly, those vectors which would indicate long-term buildup of radioactivity. Consideration was also given to the pathways which would result in exposure to man, such as milk and food crops. Locations for sampling stations were chosen after considering meteorological factors and population density around the site. Frequencies for sampling the various vectors were established so that seasonal variations in radioactivity levels might be determined. In addition, samples are collected during the season in which the major growth occurs to ascertain radioactivity uptake by the vectors during their most susceptible period of growth.

11.6.4 Analytical Sensitivity

Samples will be collected routinely following established procedures so that uniformity in sampling methods will always be assured. The samples
57 will be transported to a laboratory facility for preparation and processing. All
the radioanalytical and radiochemical analyses will be conducted in that laboratory. The following types of equipment will be utilized in performing the
analyses: Pulse height analyzers with solid and well NaI detectors and Ge(Li)
detectors; low background beta counters; liquid scintillation counters; GM detec-
57 tors; and internal proportional counters. Data will be coded and stored in computerized data base.

The detection capabilities for environmental sample analyses will
59 be presented in the FSAR. The nominal lower limit of detection (LLD) for the
various analytical techniques will be based on the method discussed in HASL-300
57 (ref. 1). The nominal LLD values are expected to approximate the values recommended in Regulatory Guide 4.2: However, the LLDs will vary depending on the
38 activities of the various components in the samples.

11.6.5 Data Analysis and Presentation

A quality control program has been established with the Tennessee
Department of Public Health Radiological Laboratory. Samples of air particu-
57 lates, water, and milk will be collected and forwarded to this laboratory for
analysis. The results will be exchanged for comparison to aid the laboratories
in evaluating their analytical systems and minimizing errors in data production.

Data collection around the operating plant will be compared to data from control stations and from the preoperational program to identify the earliest possible indications of the accumulation or buildup of radionuclides in
the environment. During the life of the plant, this accumulation should exist in
38 no more than trace amounts, with only minor impact on the environment.

corridors exterior to the reactor cavity wall. Each of these areas are
45 accessible during full power operation and are specified as Zone III.

The configuration and thicknesses of the shielding internal to the reactor vessel have been principally defined by mechanical and structural requirements of the in-vessel components in order to maintain a safe and operable configuration under design steady state, transient, and accident conditions. To achieve these design objectives, in-vessel shielding is provided to insure that reactor core support structures and the reactor vessel have adequate structural properties following neutron irradiation over the design lifetime of the component. Shielding thickness requirements for the closure head assembly, reactor vessel support ledge, and reactor cavity wall are determined based on mechanical, structural, thermal and shielding requirements. Additional shielding and special design configurations are provided for penetrations through the closure head assembly to reduce radiation streaming. Gaps through the reactor vessel support area and streaming paths formed by the primary piping and recirculating gas cooler duct penetrations through the reactor cavity wall also receive special treatment.

36 Thirty-three penetrations of the reactor closure head assembly require shielding to attenuate radiation due to sodium activation, argon cover gas and neutron sources. These penetrations are shielded as required by the following methods:

a. In-vessel shielding is located below the reactor head and provides both thermal and radiation shielding for the reactor vessel head and sufficient depth to achieve required offset gap geometries.

b. The penetration gaps are minimized in width and the gaps are stepped wherever possible.

c. The component penetrating the head is designed to have a solid structure equivalent to the closure head assembly.

d. Local shielding is provided in the head access compartment as required.

12.1.2.4 Specific Shield Design Parameters by Building

General biological radiation protection design objectives for the plant shielding and the criteria for shielding design to meet those objectives are given in the previous subsections. In the following subsections, descriptions of the shielding provided by the various plant buildings and by their internal structures are provided. Occupancy and access restrictions are given.

Reactor Containment Building

The shielding in the reactor containment building (RCB) has two general classifications: (1) the enclosure system shielding consisting of in-vessel, reactor cavity and head access area shielding and (2) the primary heat transfer system and auxiliary system shielding. Plan views of the cell structure and its associated shielding can be seen on Figures 12.1-1 through 12.1-7. Additional shielding outside of the Containment Building is provided by approximately 4 feet of reinforced concrete which make up the Confinement
:C Building surrounding the Containment.

The Head Access Area (HAA) is a 44' square by 14' high area providing access to the reactor closure head (elevation 802') from the RCB operating floor (OF) at the 816' elevation. The HAA is a Zone III radiation area with design criteria of ≤25 mrem/hr (below the OF elevation) and ≤10 mrem/hr at the operating floor elevation. The reactor vessel, its closure assembly, in-vessel shielding
40 and support structure provide shielding for the head access area. Although these components provide a conservative bulk shield design, their design criteria are also based on mechanical, thermal, and structural design considerations (see Chapter 4.0). The reactor vessel closure assembly (closure head and steel/laminar shield array) has a total material thickness of 53 inches. Therefore, no additional shielding is planned between the HAA and operating floor, as the HAA shield
59 design is adequate to meet its design criteria.

40

The reactor support ledge design provides an annulus between the support ledge and reactor vessel. This annulus is closed at the top by the reactor vessel support structure (support ring and vessel flange) which has a
59 steel thickness of ~30 inches. To reduce general area radiation levels in the HAA, a B_4C shield ring is installed at the entrance to the annulus along the bottom of the support ledge. A steel ring is installed at the bottom of the reactor vessel flange to reduce annular gap streaming. The B_4C shield is ~13½ inches thick, with the steel ring providing about 9 inches of carbon steel as
40 complementary shielding. The shield design parameters associated with the HAA
49 are summarized on Table 12.1-2.

Penetration shielding is an important aspect of the overall shield design for the HAA. These shields are necessary to control radiation levels from radioactive cover gas sources and primary sodium, gamma, and neutron streaming. Two of the major penetrations are associated with the Control Rod Drive Mechanisms (CRDM's) and the three rotating plugs in the reactor closure head. In addition, the flow of radioactive cover gas from above the sodium pool to components located within the HAA and above the HAA shielding must be prevented. This potential source is controlled by sealing cover gas flow annuli and purging the annuli formed by components penetrating the closure head with purified recycled cover gas.

Amend. 59
Dec. 1980

Ex-Vessel Fuel Storage Tank Sources

The ex-vessel fuel storage tank (EVST) is designed to hold a maximum of 650 new and expended fuel assemblies (Section 9.1.2.1.2). The shield is designed to accommodate the radiation from a full loading of the EVST. The gamma source terms shown in Table 12.1-21 are based on the following design parameters:

a. A complete unloading of the fuel assemblies in an equilibrium core is placed in either the upper or lower tier in contiguous positions nearest the part of the shield being considered. The source term is for 198 assemblies, which is a more conservative value than the 162 fuel assemblies in the core. Fuel assemblies added to other locations in the EVST would have lower activity and would be shielded by the 197 assemblies. Therefore, the source terms in Table 12.1-21
59 are conservative values for a full EVST loading.

b. For conservatism it is assumed that the equilibrium core is in place in the EVST four days after reactor shutdown.

The EVST inherent neutron source is shown in Table 12.1-22. This source term is based on the same design parameters as those given for the gamma sources in a and b above. The neutron energy spectrum is approximately equivalent to the Pu fast fission spectrum. The neutron source by isotope is in the same proportion as shown in Table 12.1-28.

The EVST fission product and transuranium elements inventories are developed from the nuclear design data and reference equilibrium core and blanket management schemes. The fission product inventory and its corresponding gamma energy spectrum, shown in Table 12.1-21 were derived using the fission yields contained in the RIBD code library and consider the cyclic operation and core/blanket management of the equilibrium cycle.

Ex-Vessel Storage Tank (EVST) Sodium Sources

The EVST sodium serves as the coolant for the expended fuel. It is also used as a transfer medium between the reactor vessel and EVST for new and spent fuel. In serving this function, it becomes contaminated with primary coolant sodium and its radioactivity. Table 12.1-23 shows the EVST sodium specific activities based on the following design parameters:

a. The sodium in the reactor vessel is characterized by the specific activities shown in Table 12.1-6.

b. Fuel is transferred to the EVST once per year. 171 assemblies are transferred from the reactor to EVST, and vice versa for each refueling.

c. Each transfer of a core component pot places 2-1/2 ft^3 of primary Na in EVST Na and vice versa.

d. The first transfer takes place 44 hours after reactor shutdown on the average the minimum time will be 26 hours.

e. The time between successive transfers is 1.5 hours.

49 f. The volume of EVST sodium is 9000 ft^3.

g. The effect of the EVST cold trap is not considered.

EVST Cold Trap Sources

The EVST cold trap sources result from the deposition of EVST sodium activity on the cold trap. The activity inventory on the EVST cold trap is shown in Table 12.1-24. The following design parameters have been used to develop these activities:

a. The EVST cold trap is as efficient as the primary cold traps.

b. The activity deposited on the cold trap is based on the EVST sodium activities shown in Table 12.1-23.

c. The cold trap sodium volume is estimated at 140 ft^3.

EVST Plugging Temperature Indicator (EPTI) and EVST Sodium Sample Package (ESSP)

The EPTI and ESSP components contain EVST sodium. The inventory of isotopes in the EPTI and ESSP, based on the source term for EVST sodium shown in Table 12.1-23, are given in Table 12.1-25. The inventory is based on 30 years of reactor operation with failed fuel as described in b, c and d of "Primary Sodium Sources" (Section 12.1.3.1).

49 New Core Component Sources

The primary radiation source in the New Fuel Unloading Station and the New Fuel Shipping Container will be the inherent neutron source in new fuel assemblies. Table 12.1-26 shows the inherent neutron source strength for a single new fuel assembly. The design parameters used to obtain this source term are:

a. The fuel assembly is loaded with LWR recycle plutonium which contains 2 percent Pu^{238}.

b. The active volume of an assembly is 1.17×10^4 cm^3 over a 91.44 cm height.

Subcritical multiplication due to grouping of fuel assemblies shall be
44 considered by the system designer.

44

Fuel Handling Cell

The fuel handling cell (FHC) is designed to handle and inspect
44 spent fuel in preparation for shipment off-site. The gamma ray source term as a function of energy is shown in Table 12.1-27 for a single expended assembly. The inherent neutron source for a single expensed assembly is shown in Table 12.1-28. These source strengths are based on the following design parameters:

a. The design assembly produces power at 120 percent of the average assembly.

b. The fuel is being handled in the FHC at four days after reactor shutdown.

c. The inherent neutron source is based on the outer core loading of Pu exposed to a burnup of 150,000 MWD/MT.

49 d. Each assembly has a volume of 1.173×10^4 cm^3 and an active height of 91.44 cm.

e. The energy spectrum is derived from the RIBD code library as discussed previously.

49 44 The FHC shield design should accommodate one design assembly source given in Tables 12.1-27 and 12.1-28. The energy distribution of the inherent neutron source shown in Table 12.1-28 was treated as a Pu fast fission spectrum.

Fuel Handling Cell Argon Circulation System Sources

The FHC service cells have the following design basis:

A. FHC Argon Filter Cell

(1) Complete release into the FHC of all noble gas, halogen, and volatile fission products from the gaps and fission plenums of two failed fuel pins, is assumed. No credit is taken for iodine attenuation by sodium, since the pin failures are assumed to occur during handling of bare fuel assemblies.

(2) For conservatism in the filter shielding design, it is assumed that 100% of the released Cs and I are collected in the filters. For conservative cell shielding design it is assumed that only 60% of all Cs and I released is collected on the filters. The remaining 40% is equally distributed in the three support cells, and in the FHC itself. Since the filters are frequently removed, there will be no buildup of long-lived Cs-137.

(3) Activities are based on a decay time of 80 days . Normally fuel assemblies with a decay time of at least 80 days (6 Kw decay heat) are handled in the FHC. Special procedures are in effect for the unusual event when a single fuel assembly with 4-day decay time (15 Kw) is handled in the FHC, minimizing the probability of fission gas release
49 from fuel pins.

(4) A FHC volume of 10,700 scf is assumed. Equipment (piping, filter banks, transitions) have void volume of 224 scf/cell.

(5) Cell specific activities are based on the radioactive releases (item 1) divided by the volume of the FHC.

Fuel handling cell, filter cell specific activity and FHC filter cell equipment activity appear in Table 12.1-29.

B. FHC Argon Blower Cell

Ten percent of the Cs and I activities released from the gaps and fission gas plenum of two failed pins are deposited on the inner surface of components. The long-lived Cs-137 activity has been multiplied by a factor of five to account for subsequent pin failures during plant life at a rate of one pin every three years.

FHC argon blower particulate activity is defined in Table 12.1-29.

Gas activity in the FHC Argon Blower Cell is the same as defined above and Table 12.1-29. Equipment in the FHC Argon Blower Cell (storage tank, grapple blowers, valves, piping) have a gas volume of 46 scf/cell.

C. FHC Fan/Cooler Unit

Sources in the FHC Fan/Cooler Unit are defined on the same bases as described for the Argon Blower Cell.

Particulate activity in the FHC Fan/Cooler Unit are the same as defined for the Argon Blower Cell.

Gas activity in the FHC Fan/Cooler Unit has the same specific activity as defined for the Argon Filter Cell.

Equipment in the FHC Argon Blower Unit cell cooler fan assembly has a gas volume of 200 scf.

D. FHC Argon Purification Unit

Sources in the FHC Argon Purification Unit are on the same bases as defined for the Argon Blower Cell.

Particulate activity in the FHC Argon Purification Unit is the same activity as defined for the Argon Blower Cell.

Gas activity in the FHC Argon Purification Unit is the same specific activity as defined for the Argon Blower Cell.

49 Equipment (Argon Purification Unit) in the cell has a gas volume of 100 scf.

The peak dose rates at the site boundary and visitors center from plant radiation are based on the estimated dose rate from a large fuel shipping cask located at the plant rail siding which when loaded meets the maximum dose rate allowed in 49 CFR, Part 173. The annual dose is due to normal plant operations because of the relatively short time period the peak dose rate would be maintained. The normal operating condition is based on a dose rate of 0.2 mrem/hr at the edge of buildings utilizing restricted access areas.

The dose rate in the control room from plant operations has been conservatively estimated at 2×10^{-6} mrem/hr. The real exposure of personnel, due to normal operations, in this heavily shielded space would be controlled by the quantity of naturally occurring radioactive materials; K^{40}, the natural U and Th decay chains, found in the concrete shield.

The peak dose rate and annual dose at the edge of the restricted buildings are anticipated to occur in the intermediate building at the intermediate sodium coolant penetration cells leading to the reactor containment building. The peak dose rate at the penetration will be no greater
55 than 0.2 mrem/hr as shown in Part III of Table 12.1-49. The remaining areas bounding the restricted area will be at or near natural background levels because of the plant shielding, the structural concrete required for hardening the plant buildings, and the below grade placement of the radioactive portions of the plant.

Dose Rates and Annual Doses at Restricted Locations of the Plant

The peak dose rate and annual doses for eight reactor containment and three reactor service building locations are shown in Parts III and V of
55 Table 12.1-49. These selected dose points are considered representative of restricted area accessible locations. In addition to the peak operating dose rates, the shutdown dose rate is also shown at each location.

The shutdown dose rates are based on the plant having been shutdown for 8 days and the sodium coolant having been used for 30 years with 1% failed fuel as described in Section 11.1.3. The annual doses shown are for continuous (24 hours/day) occupancy of the space and do not reflect the administrative controls imposed by the plant zoning criteria shown in Table 12.1-1.

Estimated Yearly Man-Rem Exposure for the Plant

The estimate of yearly man-rem exposure is based on the following parameters:

a. The occupancy of accessible areas within the restricted areas by plant personnel will be consistent with LWR experience.

b. The preliminary estimates of the plant manning by number of individuals and likely character of their work assignments. Three general classifications of individual workers are included; plant operators including shift engineers and unit operators, other non-maintenance personnel such as health physics, chemical engineers/analysts and instrumentation personnel, and finally, maintenance personnel.

c. No individual will receive exposure greater than levels specified by 10 CFR, Part 20.101.

LWR experience indicates that plant operations and non-maintenance support personnel fractional time spent in each of the radiation zones shown in Table 12.1-1 are as follows:

	Plant Operations Personnel	Other Non-Maintenance Support Personnel
Zone I	70%	45%
Zone II	25%	50%
Zone III	5%	5%

The above information is based on data provided in Reference 1. Maintenance personnel annual exposure is controlled solely by 10 CFR, Part 20 criteria and is included in the value below. Personnel assumed is consistent with Figure 13.1-1.

An evaluation of the yearly exposure using the above parameters indicates an exposure estimate for CRBRP plant staff personnel of approximately 195 man-rem per plant year, and for utility/contract personnel of approximately 205 man-rem per plant year. The estimated total radiation exposure to personnel
49 at the CRBRP site, thus, is approximately 400 man-rem per plant year.

Because of the unique nature of this plant, no directly relevant operating experience is available. The above estimate for the radiation
59 exposure to personnel at CRBRP are consistent with LWR experience.

Estimated Exposure of Specific Operations

The estimated annual man-rem radiation dose for CRBRP due to operations, maintenance, and radwaste handling, are as follows:

Operations - The radiation exposure due to operation and surveillance activities within nuclear island cells will be less than 35 man-rem/yr.
59 for contract/utility and staff personnel.

Maintenance - The radiation exposure due to maintenance activities within nuclear island cells will be less than 162 man-rem/yr for contract/utility and staff personnel.

59

Radwaste Handling - Radiation exposure due to radwaste handling has been minimized by the use of automated liquid and solid waste systems. These systems permit personnel to operate the radwaste processes remotely from a control room area in the Radwaste Building. The radiation exposures due to operation and maintenance of radwaste handling will be less than 8 man-rem/yr.

40 | 59

The total manhours, man-rem dose, and approximate dose rates due to all fuel handling and transfer operations are estimated as follows:

59	Total Man Hours	32,500 hours/yr.
	Total Man Rem	12 man-rem/yr.
	Dose Rate Range	<0.2 mrem/hr to 200 mrem/hr

The remote viewing systems are currently in the early design phase and quantitative estimates of the radiation exposure and requisite man-hours cannot be made at this time.

The radiation exposure due to visual in-service inspection will be
59 less than 50 man-rem/yr. This is based on the following information.

1. The optical and cell insertion equipment for visual inspection will be designed for service in an operating PHTS cell. The radiation protection afforded by the system will be consistent with the requirements of 10CFR20. Local dose rates will be controlled to 200 mrem/hr or less on equipment surfaces outboard of the cell shielding.

2. The in-service inspections will normally be scheduled to coincide with refueling periods to allow for decay of the Na^{24} activity. With a four day delay between the time of reactor shutdown and inspection, the dose due to the visual inspections will be 1 percent of that for inspection at operating conditions. 25

Best estimate corrosion product dose rates have been calculated for several locations adjacent to the primary heat transfer system piping. Entry to these locations would only be required for inspection supplemental to the planned activities discussed in the preceeding paragraph. The dose rates are shown for 5, 10, and 30 years of operation at 10 days after shutdown. These levels will permit controlled access to the primary equipment following plant shutdown.

BEST ESTIMATE CORROSION PRODUCT DOSE RATES

	Dose Rate (mrem/hr)		
Dose Point Description	5 Years	10 Years	30 Years
Adjacent to 36" Pipe Near Bend Descending to Pump	230	250	275
Adjacent to 36" Pipe Elevation of Primary Pump Inlet	155	175	190
Adjacent to 24" Pipe Crossover Leg Between Primary Pump and IHX (Maximum Expected)	290	320	350

25

REFERENCES TO SECTION 12.1

1. Bellefonte Preliminary Safety Analysis Report (Docket No. 50438 and 50439)

2. Deleted

59 3. Deleted

4. GEAP-13925, "Capsule B9D Irradiation and Post Irradiation Examination" dated December, 1972.

TABLE 12.1-11

RADIOACTIVE ARGON COVER GAS ACTIVATION AND FISSION PRODUCTS
RELEASE RATES AND RESULTING ISOTOPIC RELEASE RATE

	Isotope	Argon Cover Gas: Release Rate Into atoms/sec	Argon Cover Gas: Concentration Input μ ci/scc*
59			
	Xe-131m	7.16×10^{13}	0.74
	-133m	4.54×10^{14}	24
	-133	1.84×10^{16}	4.3×10^{2}
	-135m	5.56×10^{13}	1.1×10^{2}
55	-135	6.82×10^{15}	1.9×10^{3}
	-138	8.94×10^{13}	1.8×10^{2}
	Kr-83m	6.74×10^{13}	64
	-85m	2.94×10^{14}	1.5×10^{2}
	-85	4.28×10^{14}	1.4×10^{-2}
	-87	1.47×10^{14}	1.7×10^{2}
	-88	4.00×10^{14}	2.9×10^{2}
	Ar-39	6.81×10^{14}	0.783
	-41	1.28×10^{12}	1.2
49	Ne-23	3.34×10^{16}	7.7×10^{4}
	H-3	8.13×10^{8}	1.5×10^{-5}

*μ ci/scc = micro curie per cubic centimeter at standard conditions.

TABLE 12.1-12

RADIOISOTOPE INVENTORY IN RAPS VACUUM VESSEL
DURING REACTOR OPERATION

	Isotope	RAPS Vacuum Vessel+ Ci
	Xe-131m	1.22
	-133m	37.9
	-133	688
59	-135m	24
	-135	2450
	-138	35
	Kr-83m	50
59	-85m	1.7×10^2
	-85	.022
	-87	105
59	-88	2.7×10^2
	Ar-39	3.50
	-41	1.1
	Ne-23	17
	H-3	6.6×10^{-5}
59 49	TOTAL	3.9×10^3

+Daughter isotopes of the rare gas activities are included in the shield calculations. Predominant daughter products are Rb^{88} and Cs^{138} which are present at the same activity levels as Kr^{88} and Xe^{138}.

Table 12.1-12a Radioisotope Inventory in CAPS Vacuum Vessel During Reactor Operation

Isotope	CAPS Vacuum Vessel* Ci
Xe^{131m}	3.9×10^{-3}
Xe^{133m}	5.0×10^{-2}
Xe^{133}	0.18
Xe^{135m}	2.8×10^{-2}
Xe^{135}	0.23
Xe^{138}	1.1×10^{-3}
Kr^{83m}	4.1×10^{-3}
Kr^{85m}	1.5×10^{-2}
Kr^{85}	9.1×10^{-3}
Kr^{87}	8.2×10^{-3}
Kr^{88}	2.4×10^{-2}
Ar^{39}	1.0×10^{-3}
Ar^{41}	6.1×10^{-4}
Ne^{23}	9.1×10^{-6}
59 H^{3}	2.9×10^{-5}

*Daughter isotopes of the rare gas activities should be included in shield calculations. Predominant daughter products are
49 Rb^{88} and Cs^{138} which are present at the same activity levels as Kr^{88} and Xe^{138}.
All inputs are for reactor operation with 1% failed fuel.

Table 12.1-13 Radioisotope Inventory in RAPS and CAPS Compressors During Reactor Operation +

Isotope	RAPS Compressor Ci	CAPS Compressor Ci
Xe131m	2.3 x 10-2	1.7×10^{-4}
Xe133m	0.73	9.3×10^{-4}
Xe133	13	.020
Xe135m	0.43	1.2×10^{-3}
Xe135	47	6.0×10^{-2}
Xe138	0.64	1.5×10^{-4}
Kr83m	0.89	1.05×10^{-3}
Kr85m	3.0	3.9×10^{-3}
Kr85	4.2 x 10-4	3.5×10^{-4}
Kr87	2.0	2.1×10^{-3}
Kr88	4.8	6.1×10^{-3}
Ar39+	6.7×10^{-2}	4.0×10^{-5}
Ar41	1.9×10^{-2}	2.3×10^{-5}
Ne23	0.21	3.5×10^{-7}
H3	2.2 x 10-5	2.0×10^{-5}
49 Total	73	.096

*Daughter isotopes of the rare gas activities should be included in shield calculations. Predominant daughter products are Rb88 and Cs138 which are present at the same activity levels as Kr88 and Xe138.

49 +For 30 years of reactor operation.
All inputs are for reactor operation with 1% failed fuel.

TABLE 12.1-14

RADIOISOTOPE INVENTORY IN RAPS SURGE TANK
DURING REACTOR OPERATION

	Isotope	RAPS Surge Vessel+ Ci
	XE-131m	28.1
	-133m	819
	-133	1.55×10^4
59	-135m	32
	-135	3.92×10^4
	-138	44
	Kr-83m	3.6×10^2
	-85m	2.0×10^3
	-85	0.52
	-87	6.0×10^2
	-88	2.5×10^3
	Ar-39	81
	-41	7.9
	Ne-23	0.97
	H-3	2.4×10^{-3}
59 49	TOTAL	6.2×10^4

+Daughter isotopes of the rare gas activities are included in the shield calculations. Predominant daughter products are Rb^{88} and Cs^{138} which are present at the same activity levels as Kr^{88} and Xe^{138}.

TABLE 12.1-14a RADIOISOTOPE INVENTORY IN CAPS SURGE TANK DURING REACTOR OPERATION

Isotope	CAPS Surge Vessel Ci
Xe^{131m}	5.8×10^{-2}
Xe^{133m}	7.5×10^{-2}
Xe^{133}	2.7
Xe^{135m}	0.11
Xe^{135}	3.2
Xe^{138}	4.1×10^{-3}
Kr^{83m}	4.5×10^{-2}
Kr^{85m}	0.19
Kr^{85}	0.14
Kr^{87}	8.0×10^{-2}
Kr^{88}	0.28
Ar^{39}	1.6×10^{-2}
Ar^{41}	6.6×10^{-3}
Ne^{23}	2.0×10^{-6}
59 H^3	4.4×10^{-4}

*Daughter isotopes of the rare gas activities should be included in shield calculations. Predominant daughter products are Rb^{88} and Cs^{138} which are present at the same activity levels
49 as Kr^{88} and Xe^{138}.
All inputs are for reactor operations with 1% failed fuel.

TABLE 12.1-15

RADIOISOTOPE INVENTORY IN RAPS COLD BOX DURING REACTOR OPERATION

Isotope	RAPS Cold Box+ Ci
Xe-131m	1910
-133m	1.06×10^4
-133	4.67×10^5
-135m	1.92
-135	8.8×10^4
-138	2.5
Kr-83m	1.6×10^2
-85m	2.1×10^3
-85	7.2×10^2
-87	1.8×10^2
-88	1.7×10^3
Ar-39	28
-41	1.5
Ne-23	3.4×10^{-4}
H-3	3.0×10^{-2}
TOTAL	5.8×10^5

59 49

+Daughter isotopes of the rare gas activities are included in the shield calculations. Predominant daughter products are Rb^{88} and Cs^{138} which are present at the same activity levels as Kr^{88} and Xe^{138}.

TABLE 12.1-15a RADIOISOTOPE INVENTORY IN CAPS COLD BOX DURING REACTOR OPERATION

	Isotope	CAPS Cold Box Ci
	Xe^{131m}	23
	Xe^{133m}	5.8
59	Xe^{133}	4.7 x 10^{2}
	Xe^{135m}	.3
	Xe^{135}	45
	Xe^{138}	0.01
	Kr^{83m}	0.12
	Kr^{85m}	1.2
	Kr^{85}	5.1
	Kr^{87}	0.14
	Kr^{88}	1.1
	Ar^{39}	2.5 x 10^{-2}
59	Ar^{41}	4.6 x [illegible]$^{-3}$
	Ne^{23}	3.5 x 10^{-8}
59	H^{3}	6.8 x 10^{-3}

*Daughter isotopes of the rare gas activities should be included in shield calculations. Predominant daughter products are
49 Rb^{88} and Cs^{138} which are present at the same activity levels as Kr^{88} and Xe^{138}.
All inputs are for reactor operations with 1% failed fuel.

TABLE 12.1-16

RADIOISOTOPE INVENTORY IN RAPS RECYCLE ARGON VESSEL
DURING REACTOR OPERATION

	Isotope	Recycle Argon Vessel Ci
	Xe-131m	1.12×10^{-3}
	-133m	3.1×10^{-2}
	-133	0.61
	-135m	6.6×10^{-5}
	-135	1.1
	-138	8.2×10^{-5}
	Kr-83m	3.9×10^{-3}
59	-85m	3.8×10^{-2}
	-85	2.06×10^{-5}
	-87	5.0×10^{-3}
	-88	3.7×10^{-2}
	Ar-39	49
	-41	1.3
	Ne-23	1.3×10^{-3}
49	H-3	1.6×10^{-3}
59	TOTAL	52

TABLE 12.1-17

RADIOISOTOPE INVENTORY IN RAPS NOBLE GAS STORAGE VESSEL

49

	Isotope	RAPS Noble Gas Storage Vessel Activity Ci	
	Xe^{131m}	1910	
	Xe^{133m}	1.06×10^{4}	
	Xe^{133}	4.67×10^{5}	
	Xe^{135m}	1.92	
	Xe^{135}	8.8×10^{4}	25
	Xe^{138}	2.5	
	Kr^{83m}	1.6×10^{2}	
	Kr^{85m}	2.1×10^{3}	
	Kr^{85}	7.2×10^{2}	
	Kr^{87}	1.8×10^{2}	
	Kr^{88}	1.7×10^{3}	
	Ar^{39+}	28	
49	Ar^{41}	1.5	
	Ne^{23}	3.4×10^{-4}	
52	H^{3}	3.0×10^{-2}	
59	Total	5.8×10^{5}	

*Daughter isotopes of the rate gas activities should be included in shield calculations. Predominant daughter products in Rb^{88} and Cs^{138} which are present at the same activity levels as Kr^{88} and Xe^{138}.

+for 30 years of reactor operation

TABLE 12.1-18

49 TRITIUM REMOVAL UNIT RADIOISOTOPE CONCENTRATIONS

	Isotope	Tritium Removal Unit (μCi/scc)*
	Xe^{131m}	8.7×10^{-4}
	Xe^{133m}	1.1×10^{-3}
	Xe^{133}	4.0×10^{-2}
	Xe^{135m}	1.7×10^{-3}
	Xe^{135}	4.7×10^{-2}
	Xe^{138}	6.1×10^{-5}
	Kr^{83m}	6.7×10^{-4}
59	Kr^{85m}	2.9×10^{-3}
	Kr^{85}	2.1×10^{-3}
	Kr^{87}	1.2×10^{-3}
59	Kr^{88}	4.2×10^{-3}
	Ar^{39}	2.4×10^{-4}
	Ar^{41}	9.9×10^{-5}
	Ne^{23}	3.0×10^{-8}
59	H^{3}	6.8×10^{-6}

* All inputs are for reactor operations with 1% failed fuel.

59

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TABLE 12.1-23 EVST SODIUM SPECIFIC ACTIVITY AFTER 30 YEARS OF OPERATION

	Isotope	Specific Activity (μCi/gm)
	Na^{24}*	14.7
	Na^{22}	5.80×10^{-1}
	Cs^{137}	7.10
	Cs^{136}	4.39×10^{-1}
	Cs^{134}	1.48×10^{-1}
	Sb^{125}	8.04×10^{-3}
	Sb^{127}	1.99×10^{-2}
	I^{131}	8.90×10^{-1}
	Te^{132}	1.66×10^{-3}
	I^{132}	1.50×10^{-1}
59	Te^{127m}	1.02×10^{-3}
	Te^{127}	1.02×10^{-3}
	Te^{129m}	2.65×10^{-3}
	Te^{129}	2.65×10^{-3}
	Sr^{89}	4.31×10^{-4}
	Sr^{90}	2.87×10^{-3}
	Y^{90}	2.87×10^{-3}
	Y^{91}	1.23×10^{-4}
	Zr^{95}	2.51×10^{4}
	Nb^{95}	2.51×10^{-4}
	Ru^{103}	3.10×10^{-4}
	Ru^{106}	4.98×10^{-4}
49	Rh^{106}	4.98×10^{-4}

TABLE 12.1-23

(Continued)

Isotope	Specific Activity ($\mu Ci/gm$)
Ba^{140}	1.75×10^{-4}
La^{140}	1.75×10^{-4}
Ce^{141}	3.35×10^{-4}
Ce^{144}	3.44×10^{-4}
Pr^{144}	3.44×10^{-4}
Pr^{143}	1.46×10^{-4}
Nd^{147}	6.25×10^{-5}
Pm^{147}	4.27×10^{-4}
Pu^{238}	6.9×10^{-3}
Pu^{239}	1.86×10^{-3}
Pu^{240}	2.42×10^{-3}
Pu^{241}	1.63×10^{-1}
Pu^{242}	5.18×10^{-6}
Np^{238}	6.78×10^{-9}
Np^{239}	3.41×10^{-5}
Am^{241}	6.39×10^{-4}
Am^{242m}	2.60×10^{-5}
Am^{242}	2.60×10^{-5}
Am^{243}	1.07×10^{-5}
Cm^{242}	6.71×10^{-5}
Cm^{243}	6.35×10^{-6}
Cm^{244}	1.22×10^{-4}
H^{3}	1.40×10^{-2}

49 *Peak Na-24 occurs approximately 24 hours after transfer begins.

TABLE 12.1-27

FUEL HANDLING CELL GAMMA SOURCE TERM
AS A FUNCTION OF ENERGY

Energy Group	Mean Energy (Mev)	Mev/cm^3-sec*
1	2.8	2.27×10^9
2	2.4	5.02×10^{10}
3	2.0	2.84×10^{10}
4	1.575	9.87×10^{11}
5	1.125	1.75×10^{11}
6	0.65	2.36×10^{12}
7	0.20	3.71×10^{11}

*Source strength given per cm^3 of assembly at four days after shutdown.
59 Up to three assemblies can be stored in FHC at one time.

TABLE 12.1-28

49 INHERENT NEUTRON SOURCE TERM

EXPENDED FUEL ASSEMBLIES

	Nuclide	Neutron per Second
49	Pu^{238}	1.2×10^7
	Pu^{239}	8.7×10^5
	Pu^{240}	4.7×10^6
	Pu^{241}	0
	Pu^{242}	9.2×10^5
	U^{235}	0.1
	U^{238}	5.4×10^2
	Cm^{242}	9.7×10^8
	Cm^{243}	1.7×10^5
	Am^{244}	1.1×10^7
	Am^{241}	2.3×10^6
	Am^{243}	1.2×10^4
49	Total	1.0×10^9

*Values include a 1.5 design contingency factor to account for uncertainties in transuranium element inventory.

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
59 d. Fuel Handling Cell Operating Gallery (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

Appendix 12A

Information Related To ALARA for Occupational Radiation Exposures

12A.1 CRBRP ALARA Commitment

The CRBRP management commitment for the plant design and operation is such that every reasonable effort shall be made to keep radiation exposures to plant personnel "as low as reasonably achievable" (ALARA), within the regulations of 10CFR20 and the guidelines in Regulatory Guides 8.8, 8.10, and 8.19.

12A.2 10CFR20 Requirements

10CFR20, "Code of Federal Regulations, Energy Section, Standards for Protection Against Radiation", applies to the CRBRP. The following specific criteria are excerpts from 10CFR20 which are applicable to the CRBRP ALARA Program.

12A.2.1 ALARA

Paragraph 20.1 (c) of 10CFR20 states in part that the licensee should, in addition to complying with the limits specified in 10CFR20, make every reasonable effort to maintain radiation exposures and release of radioactive materials in effluents to unrestricted areas as low as is reasonably achievable. "As low as is reasonably achievable" means as low as is reasonably achievable, taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations, and in relation to the utilization of atomic energy in the public interest.

12A.2.2 Exposure Limits for Restricted Areas

Paragraphs 20.101 and 20.103 of 10CFR20 specify the permissible dose levels and airborne concentrations for restricted areas where the term "restricted area" means any area access to which is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive material. "Restricted area" shall not include any areas used as residential quarters, although a separate room or rooms in a residential building may be set apart as a restricted area.

12A.2.3 Exposure Limits for Unrestricted Areas

Paragraphs 20.105 and 20.106 of 10 CFR20 specify the permissible doses, levels, and concentrations for unrestricted areas where the term "unrestricted area" means any area access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, and any area used for residential
49 quarters.

12A.3 CRBRP ALARA Program

The CRBRP ALARA Program will be managed in two distinct phases through the lifetime of the plant. The first phase consists of the ALARA activities associated with the design, fabrication, construction and preoperational testing activities. For this phase, the Project has established a series of management review and controls designed to incorporate and evaluate specific ALARA features. The second phase will consist of the standard TVA ALARA policies and procedures which will apply during plant operations through the decommissioning of CRBRP. In addition, TVA will participate in the first stage of the ALARA program to assure a smooth transition of the ALARA responsibility to TVA. The two phases of the CRBRP ALARA Program are discussed below.

12A.3.1 CRBRP ALARA Program for the Design Through Preoperational Activities

The first stage of the ALARA Program involves the interaction of multiple engineering disciplines, i.e., radiation analysts, shielding designers, shielding analysts, system designers and component designers. The elements in this stage are as follows:

59 A. Establishment and control of estimated radiation exposure levels.

B. Design of the components and systems to achieve the
59 estimated radiation exposure and shielding objectives.

C. Reviews by the Project ALARA Committee to evaluate and manage the achievement of the objectives for radiation exposure.

D. Reviews by experienced health physicists to obtain applicable current LWR information.

12A.3.1.1 Plant Radiation Exposure Allocations

The Project has developed plant radiation exposure objectives
59 for specific functions and/or systems. The development of these estimated
radiation exposure allocations was based on the consideration of the
total staff required to operate and maintain the plant, and the radiation
exposure objectives for individuals as well as the collective group.
Objectives have also been developed for radiation exposures of contract
59 and utility personnel.

12A.3.1.2 Design of Components and Systems

After radiation exposure objectives, plant shielding criteria, radiation source terms, and time-access requirements are identified, the system designers proceed with the system and components design with the objective to reduce the total annual radiation exposure associated with their system to a level as low as reasonably achievable.

59 Management control of those features of the individual system designs which influence radiation exposure will be assisted by the use of estimated radiation exposure information compiled in a format consistent with Regulatory Guide 8.19. The program to achieve this is being developed
59 to provide estimated radiation exposure information of the following types:

1. Dose for each nuclear island cell

2. Dose for specific categories of cells (primary heat transfer cells, Reactor Containment Building, Reactor Service Building, etc.)

3. Dose by skill classification (operators, mechanical maintenance, electrical maintenance, etc.)

4. Dose by system (auxiliary sodium, RAPS, etc.)

5. Dose by individual piece of nuclear island equipment.

These compilations are based on the following input information provided by the appropriate systems:

1. Component

2. Manhours of operation and maintenance required for each significant system component.

3. Frequency of Activity

4. Cell/Bldg. number

This, together with the predicted cell dose rate will form the basis for the radiation exposure study.

The radiation exposure information is periodically reviewed and updated as the system/component design and analyses is developed. By utilizing this system, the significant contributors to the plant radiation exposure can be identified and the appropriate ALARA action can be taken.

Section 12.1 provides the specific radiation protection and shielding criteria applicable to the CRBRP design. The system/component designers are responsible for meeting these criteria.

Formal component design reviews are peridocially performed which are consistent with the Project QA requirements. The radiation protection/shielding designers at each Reactor Manufacturer and the Architect Engineer participate in the appropriate reviews and must approve the shielding design of the component design before release. All changes to the plant design are reviewed and the impact on ALARA
49 determined.

The appropriate corrective measures are taken as a result of this evaluation.

12A.3.1.3 Project ALARA Committee Reviews

A CRBRP Project ALARA Committee (PAC) has been established for the purpose of periodically reviewing the CRBRP system/component designs to assure that the ALARA goal will be met and to update plant radiation exposure information. The PAC is a multi-discipline group consisting of expertise in radiation analysis, shielding design, safety and licensing, and plant maintenance. These reviews result in guidance on reducing radiation exposure on a system/component basis.

The principal personnel involved in these reviews by position title including their experience are listed below:

59

Manager Shielding Analysis Westinghouse	18 years experience in radiation analysis and shielding design.
Principal Engineer Shielding Analysis Westinghouse	(a) 12 years experience in applied and technical aspects of health physics. (b) 10 years experience in radiation analysis and shielding design.
Senior Engineer Operations and Maintenance Westinghouse	10 years experience in reactor and plant operations and maintenance
Lead Engineer Radiation Safety and Shielding AI	25 years in health physics, and shielding design and analysis
Licensing Engineer, Licensing Group Burns & Roe	(a) 5 years experience in health physics (b) 16 years experience as health physicist in AEC/NRC I&C (c) 3 years experience as licensing engineer

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12A.3.1.4 Health Physicists ALARA Reviews

The other level of review is performed by health physicists from TVA and Commonwealth Edison. There are three health physicists involved in these reviews, two from the TVA ALARA committee and one from Commonwealth Edison. The two TVA health physicists on the CRBRP ALARA committee satisfy the TVA commitments in PSAR Section 12A.3.2. The health physicist's ALARA review meetings are conducted twice a year.
59 The health physicists review system/component design, maintenance outline procedures, and the radiation exposure data and provide recommendations to further reduce radiation exposure based on their ALARA experience at operating nuclear power plants. The specific personnel involved in these reviews by position title, including their health physics training and experience, are listed below:

Title	Training/Experience
Staff Environmental Engineer Plant Engineering Branch Division of Power Production Tennessee Valley Authority	(a) Certified Health Physicist (b) 20 years of technical and management experience in health physics.
Health Physicist Radiological Hygiene Branch Tennessee Valley Authority	20 years of experience in applied and technical aspects of health physics.
Equipment Specialist for Demineralizers, Radiation Monitors and ALARA Commonwealth Edison	15 years experience in health physics

12A.3.2 CRBRP Operations Stage ALARA Program

The purpose of TVA policies and procedures is to guide the official actions expected of TVA employees. A policy or a required procedure will not serve that purpose unless it is known to all those it affects and is understood, interpreted, and applied consistently. Continuing guides of this nature in TVA are published and distributed in such a way as to be available to all employees concerned. They are known as "administrative releases".

The TVA Administrative Release System is composed of Organization Bulletins, TVA Codes, TVA Instructions, and TVA Announcements.

With regard to information that occupational radiation
49 exposures are low as is reasonably achievable, the following quotation is excerpted from TVA's Administrative Release Manual:

This instruction supplements the TVA Codes under VIII HAZARD CONTROL and VIII HEALTH SERVICES. It describes general responsibilities and administrative arrangements of ionizing radiation arising in connection with TVA's work. The detailed administrative arrangements in the instruction apply to all activities involving ionizing radiation.

TVA management is committed to maintaining radiation exposures to its employees and the general public, and the release of radioactive materials to unrestricted areas as low as is reasonably achievable (ALARA), as defined in 10 CFR Part 20. For the protection of its employees, TVA also subscribes to the ALARA philosophy set forth in the Nuclear Regulatory Commission Regulatory Guides 8.8 and 8.10 in the design and operation of all facilities utilizing radioactive materials or radiation sources.

ALARA Program - In view of the commitment in the TVA Administrative Release Manual, TVA has established a formal program to ensure that occupational radiation exposures to employees are kept as low as reasonably achievable (ALARA) and will apply this program to the CRBRP. The program consists of: (1) full management commitment to the overall objectives of ALARA; (2) issuance of specific administrative documents and procedures to the TVA design and operating groups that emphasize the importance of ALARA throughout the design, testing, startup, operation, and maintenance phases of TVA nuclear plants; (3) continued appraisal of inplant radiation and contamination conditions by the onsite radiation protection staff; and (4) a 4-member corporate ALARA committee consisting of management representatives from the TVA design, operations and radiation protection groups, whose purpose is to review and appraise the effectiveness of the ALARA program on a plant-by-plant basis, including the CRBRP. In developing its ALARA program, TVA has closely followed the recommendations of NRC Regulatory Guides 8.8 and 8.10.

The responsibility for implementing the ALARA philosophy in the operation of TVA nuclear power plants is assigned to two divisions. The Division of Power Production has the responsibility of implementing the operational procedures described in Section C.4 of Regulatory Guide 8.8. Further in the implementation of Section C.4, the Division of Environmental Planning provides the radiation protection staff for TVA nuclear facilities and has the ultimate responsibility for determining that TVA
52 maintains radiation exposures as low as reasonably achievable (ALARA) as defined in 10CFR Part 20. The radiation protection program management and staff in the Division of Environmental Planning will, as a minimum, meet the qualification and training guidelines set forth in Regulatory
49 Guides 8.8 and 8.10.

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14.1.4 TEST OBJECTIVES OF FIRST-OF-A-KIND PRINCIPAL DESIGN FEATURES

The following Test Abstracts are provided per US-NRC NUREG-75/087 - Section 14.1, Review Responsibilities Item 2 for special, unique or First-of-a-Kind principal design features included in the CRBRP.

14.1.4.1 IN-VESSEL TRANSFER MACHINE

The only equipment of the reactor refueling system, which is considered first-of-a-kind and unique to the CRBRP is the in-vessel transfer machine (IVTM). The IVTM is installed in the reactor head during reactor refueling and is discussed in detail in Section 9.1.4.4.

In order to minimize preoperational testing of reactor refueling
system equipment at the CRBRP, the IVTM will be tested and checked out exten-
59 44 sively at the ETEC.

The off-site tests are scheduled early in the program to ensure corrective actions can be taken to qualify the IVTM for CRBRP service without jeopardizing the overall plant construction schedule should any IVTM deficiencies be uncovered.

The IVTM prototype will be tested extensively to demonstrate that
the IVTM meets its specification performance and design requirements. The
complete and integrated IVTM assembly will be tested, including the control
44 console with the minicomputer.

After the IVTM has been assembled at the test site, and the assembly
has been checked out, the IVTM will first be subjected to individual and inte-
44 grated checkout tests. Following this, the IVTM will be performance tested
simulating core assembly transfers.

The tests will be performed in special test facilities containing
44 a cluster of at least seven simulated core assemblies. The cluster will be
capable of relative vertical and horizontal displacements and side loads.

A. INDIVIDUAL CHECKOUT TESTS

The purpose of the individual checkout tests is to verify that the following IVTM functions can be performed:

44 1) Grapple and release of a Simulated Core Assembly (SCA).

44 2) Raise and lower a SCA to positions corresponding to those encountered in the reactor vessel. 38

44 3) Identify and orient a SCA

44 4) Provide adjacent SCA holddown when removing a SCA from the SCA cluster.

5) Provide cover gas containment and seal leakage detection capability.

Specific tests will include the following:

1) Calibration and checkout of all IVTM interlocks, load cells, and the entire load control system.

2) Verification of all functions of the core assembly identification system.

44 3) Checkout of the grapple and holddown sleeve drive systems including removal of an artificially jammed core special assembly.

4) Calibration and checkout of the grapple and holddown sleeve position indication systems.

5) Verification of the seal leakage monitoring and the seal pressurization control systems.

B. INTEGRATED CHECKOUT TESTS

The purpose of these tests is to prove that the IVTM meets the following objectives:

1) The IVTM can perform the sequence of functions listed in Section A which are required to transfer a core assembly in accordance with given operating profiles when using computer and manual controls.

2) Insertion and removal of core assemblies into and from the core can be accomplished under maximum misalignment in combination with maximum core assembly push and pull loads.

3) Release of a core assembly in an incorrect core position is prevented.

4) Release of a core assembly into a transfer position in the absence of a core component pot cannot be accomplished. 38

44 5) Premature release of a core assembly during operation over the core is prevented when the core assembly is at a vertical position higher than a small tolerance above the fully seated position.

PERFORMANCE TESTS

These tests are designed to simulate reactor refueling operations
59 equivalent to five refueling periods.

The tests will be performed with a cluster of seven simulated core
44 assemblies. The simulated core assembly cluster will be offset in relation to the IVTM to simulate core assembly insertion and removal under misaligned conditions. Integrated operations of the IVTM, control console, and computer will be performed simulating actual refueling operations. The major test objective is to demonstrate that all IVTM components, especially dynamic
59 seals will perform for a minimum of one refueling period. The test goal for all mechanical components of the IVTM (excluding elastomeric seals) is to demonstrate operation without failure. Post-test inspection of the mechanical components will establish the acceptability of component wear.

The following results will be obtained from these tests:

1) Wear data of dynamic seals.

2) Wear data of mechanical components.

3) Establish transfer cycle speeds for automatic and manual operation.

4) Wear data of core assembly identification pawl.

5) Any operational limitations.

6) Any deficiencies in the operating components and/or in the design.

7) Verify the computer control of the fuel transfer cycle.

8) Verification of the core assembly identification system with respect to wear data obtained in item 4 above.

9) Verification of checkout and operational procedures.

D. PREOPERATIONAL IVTM TESTS AT CRBRP

Those IVTM operations which are not simulated in the special test facilities will be performed after IVTM installation, adjustments, and checkout at the CRBRP reactor small rotating plug prior to fuel loading. These tests will include: 38

15.3.3.2 Loss of Normal Shutdown Cooling System

15.3.3.2.1 Identification of Causes and Accident Description

Loss of normal shutdown cooling will occur following loss of the main condenser, since the heat sink for the normal shutdown cooling mode is provided by the main condenser. Other conditions that affect the ability of the main condenser to provide shutdown cooling include loss of normal feedwater (Section 15.3.1.6), failure of the steam bypass system (Section 15.3.2.4), and main steam line pipe break (Section 15.3.3.1). In the event of a loss of the condenser, the reactor will be scrammed. Since the steam bypass system is prevented from operating in the event of loss of condenser, a loss of condenser would result in a sequence of events similar to that for failure of the Steam Bypass System (Section 15.3.2.4).

15.3.3.2.2 Analysis of Effects and Consequences

The consequences of a loss of condenser are slightly less severe than that for the failure of the steam bypass system (Section 15.3.2.4) since a reactor trip occurs somewhat earlier in the transient. Core temperatures are similar to those for a normal scram.

15.3.3.2.3 Conclusions

Core temperatures following loss of normal shutdown cooling are similar to a normal trip.

15.3.3.3 Large Sodium-Water Reaction

15.3.3.3.1 Identification of Causes and Accident Description

A large leak in a steam generator tube will result in injection of high pressure steam and/or water into the IHTS sodium. The resulting sodium-water reaction (SWR) will generate higher than normal pressures and temperatures in the IHTS. As discussed in Section 15.3.2.3, Steam Generator Tube Leak, the probability of a leak in a tube in the steam generators is expected to be quite small as a result of careful design supported by development and testing of the steam generators. However, a leak detection system, described in Section 7.5.5, has been provided to allow operator action to limit the consequences of a leak in a steam generator tube. The leak detection system will alert the operator to the existence of a leak rate as low as 2×10^{-5} lb. water/sec. For initial leak sizes which can be realistically expected (up to about 10^{-2} lb. water/sec.) there will be sufficient time for operator action to limit damage to the steam generator and to prevent a significant increase of the leak rate. Should a leak occur of such magnitude that operator action as described above is not effective, the Sodium-Water Reaction Pressure Relief Subsystem (SWRPRS) will provide sodium side pressure relief by operation of the rupture discs in the IHTS so that integrity of the IHTS piping and components, e.g., pump and the Intermediate Heat Exchanger (IHX) will be maintained. No operator action is required for the SWRPRS to perform its design function. A description of the SWRPRS is given in Section 5.5.

Large leaks might occur due to sudden rapid propagation of a large flaw in a tube. In this event, the leak could develop in a very short time and in the limit could approach the instantaneous double ended guillotine failure assumed. A second mechanism for developing a large leak is through wastage from a small leak. The latter mechanism is believed to have the higher probability of occurrence. An estimate of the time required for the development of a significant leak due to wastage can be obtained from the results of small SWR leak development and wastage data. Leaks in the range of 10^{-6} to 10^{-3} lb/sec have been observed to self-enlarge as indicated in
59 Figure 15.3.3.3-1. A leak of the order of 10^{-5} lb/sec could over the period of several hours suddenly increase in size to the order of 10^{-3} to 10^{-2} lb/sec. A leak of this magnitude, directed through a drilled hole (an idealized, conservative leak geometry) onto an adjacent target (representing an adjacent steam generator tube) has been observed to cause wastage rates on the target of 1 to 5 mils per second (Ref. 1). At these wastage rates, failure of a steam generator tube adjacent to the leaking tube could occur within about twenty seconds. Definitive data on the ultimate leak size resulting from wastage failure does not exist, however, wasted areas exhibit configurations ranging from cone shaped craters to irregular and diffuse wastage regions. The area of the failure in the adjacent tube wall, if the wasted area is cone shaped, would be small relative to a double ended failure area. If the jet emanating from the original leak is diffuse (as opposed to a concentrated jet) the resultant leak area on the adjacent tube could be larger but would not be expected to approach that of a double ended guillotine.

Based on the foregoing discussion, the largest expected steam generator failure is the double ended guillotine failure of a single tube. However, as explained in detail in Section 5.5.3.6, a more severe event has been postulated to ensure adequate design margin. This DBL is defined as an Equivalent Double Ended Guillotine (EDEG) failure of a steam generator tube which is followed by two additional single DEG failures, spaced at 1.0 second intervals, for a total of 3 DEG. This sequence is superimposed on a system which has been pressurized by an undetected moderate sized leak to just below the rupture disk burst pressure. 22 30

59 The injection of water into sodium results in high IHTS pressure pulses from the sodium-water reaction. This pressure is relieved by the rupture discs in the SWRPRS. Sodium reaction products and hydrogen are expelled from the IHTS into the SWRPRS where hydrogen is separated from the particulate and liquid matter. The hydrogen is vented to the atmosphere through a flare stack and liquid and particulate are contained in the reaction
59 products separator tanks under an inert atmosphere. Operation of the rupture discs automatically isolates and depressurizes the water side of the steam generators to limit damage to the system. The remaining sodium in the affected IHTS loop would be drained by operator action.

The action of the Plant Protection System (PPS) in this event is the following:

a. Primary Shutdown System - Trip on steam flow - feed flow mismatch.

b. Secondary Shutdown System - Trip on sodium water reaction.

Either of the above trips will cause a reactor shutdown well before the temperature transient resulting from the water/steam isolation and dump can be transported back to the reactor inlet. Consequently, no reactor clad or fuel damage is involved with this event.

Details of the resultant pressure pulses and their impact on the adjacent steam generators, IHX and pump can be found in Section 5.5.3.6 of this PSAR. This includes evaluation of various sizes of failures, including discussions of the probable development sequences of various leaks, up to and including the DBL in the evaporator and the superheater modules. 30

15.3.3.3.2 Analysis of Effects and Consequences

The analysis of the effects of the DBL on the steam generators and associated components in the IHTS and SWRPRS will be carried out using the
59 Transwrap computer code, as discussed in Section 5.5.3.6.

The impact of this event on the reactor core is similar to the event analyzed in Section 15.3.1.7 (Inadvertant Actuation of the Sodium-Water Reaction Pressure Relief System) by the DEMO Code. The reactor, due to the long transport delay associated with this event, does not immediately see the temperature changes, so that when the reactor trip occurs on steam-flow - feed flow mismatch (less than 4.0 seconds), the transient at the reactor is the same initially as a conventional trip. The core hot spot temperature will decrease quite rapidly and remain below normal operational temperatures throughout the course of the accident event.

If it is assumed that this event occurs following operation with the maximum undetected intermediate-to-primary sodium leak rate, the possibility of a radiological release resulting from venting of the sodium water reaction products must be considered. Leakage of primary sodium into the IHTS is prevented normally by pressurizing the IHTS such that a pressure differential across the IHX (intermediate to primary) of at least 10 psi exists during plant operation. This pressure differential could be lost following bursting of the SWRPRS rupture discs and it is possible that primary sodium could enter the IHTS. During normal operation
59 (Section 7.5.5) peak rates in excess of approximately 6 gph will be detected. and therefore only small amounts of primary sodium could be introduced into the IHTS during the depressurization transient.

Section 15.6.1.5 looks at a more severe incident in which the 24 inch IHTS pipe is severed between the IHTS pump and the IHX. This results in immediate IHTS depressurization and all IHTS sodium spilled onto the cell floor along with 1.4 pounds of primary sodium leaked across the IHX. This 1.4 pounds of primary sodium represents a conservative envelope of the amount that can be leaked across the IHX during the SWRPRS actuation event. Regardless of the location (superheater or evaporator) of the sodium-water reaction, the
59 type of initiating leak, and the number of secondary tube failures in the sodium-water reaction incident will result in less primary sodium entering the IHTS. This is true because for the sodium-water reaction, there is no sudden depressurization of the IHTS as occurred when the 24-inch pipe was severed.

Primary sodium that leaks across the IHX may be transported to the SWRPRS tank if there is sufficient flow available to move sodium from the IHX to the superheater inlet. The maximum IHTS sodium available to transport primary sodium is calculated by summing: (1) the integral of pump flow as a function of time for pump coastdown and (2) expansion tank and pump tank cover gas expansion down to 28 psia (minimum pressure to elevate IHTS sodium up to IHX inlet). This is very conservative since during pump coastdown, some of the sodium flowing from the expansion tank and pump tank will probably reverse at the pump inlet and will flow towards the evaporator exit rupture disc. The sodium that flows in this direction will not be available to transport the primary sodium.

The primary sodium that reaches the superheater inlet is assumed to react with the water/steam and be swept into the SWRPRS tank. All primary sodium particles are assumed to be airborne and are swept up and stack with the hydrogen gas. The separator will remove approximately 95% of these particles but 5% will escape to the atmosphere as the hydrogen gas is burned. As a result of the burning of the hydrogen gas, the sodium particles will be carried to heights much higher than the actual stack height. From references 3 and 5 the effective release height may exceed 1000 meters depending on wind velocity and quantity of hydrogen gas burned. For conservatism the effective release height is assumed to be 300 meters (Reference 3). Reference 4 contains procedures for calculating centerline doses at various distances from the point source (puff release) and for various effective release heights. The X/Q values calculated here are consistent with site meteorology and an effective release height of 300 meters. 22

15.5.2.5 The Heaviest Crane Load Impacts the Reactor Closure Head

15.5.2.5.1 Identification of Causes and Accident Description

The CRBRP polar crane will service the head access area with refueling and maintenance equipment. The heaviest load identified to be handled over the reactor vessel head is the AHM. The polar crane is a double reeved design with velocity limiting drum brakes which limit the lowering speed to 5 fpm. If the AHM load of ~100 tons is accidentally lowered onto the head at the crane velocity limit, an impact of ~100 tons on the reactor enclosure head assembly is imposed. Normally only the AHM extender weight rests on the head
44 assembly.

Collision of the crane-handled AHM with the head and/or head-mounted equipment such as CRDM's has been classified as unlikely due to restrictions on crane travel, and as a result of operating personnel knowing and carrying out the written and approved procedures. The polar crane lowering speed restrictions mitigate possible damage to the head and head-mounted equipment.

15.5.2.5.2 Analysis of Effects and Consequences

The AHM being lowered at the crane velocity limit of 5 fpm onto the AHM floor valve and port adapter, would result in two overload considerations for the reactor head assembly: (1) supporting the AHM static load of ~100 tons, and (2) absorbing the impact energy developed by a weight of ~100 tons at 5 fpm.

An analysis was perfo[illegible] for this case and results showed that the head assembly can and will be [illegible]d to support the static load (~100 tons) of the AHM. In addition, the [illegible]gn will also be capable of absorbing, without any detrimental structural affect, the ~100-ton load at an impact velocity of
44 5 fpm.

At the extreme limit of damage to the head, the leakage of fission gas from the reactor in this event will not exceed that which was analyzed in Section 15.5.2.4.

15.5.2.5.3 Conclusions

Based on the data currently available, it appears that the head assembly can withstand without any detrimental structural effects, the lowering of
44 the ~100-ton Auxiliary Handling Machine at a velocity of 5 fpm. However, if damage to the CRDM's or other head-mounted equipment should occur, analysis of the consequences shows that the release of reactor cover gas is within the design limits discussed in Section 15.5.2.4.

15.5.3 Extremely Unlikely Events

15.5.3.1 Collision of EVTM with Control Rod Drive Mechanisms

15.5.3.1.1 Identification of Causes and Accident Descriptions

The EVTM is a massive, railway gantry-mounted, shielded cask type fuel transfer machine. It moves, during refueling, on its gantry to within ~1 ft of the CRDM's. Note that these operations only occur when the reactor is shut down. At that time, the absorber assemblies are fully inserted and disconnected from the CRDM's.

During refueling, the EVTM is moved on its gantry between EVST, fuel handling cell, and reactor on rail tracks. These rail tracks are mounted in a floor trench. The gantry wheel track structure incorporates anti-liftoff and over-turning restraints. At both rail ends, fixed mechanical rail stops are mounted to limit the EVTM gantry travel in the event of a travel limit switch failure, and in addition, an operator error or braking failure. These features, combined with the written and approved operating procedures for the operators, reduce the likelihood of an EVTM collision with the CRDM's to the extremely unlikely level.

15.5.3.1.2 Analysis of Effects and Consequences

Because the reactor is shut down at the time, and the absorber assemblies are fully inserted into the core and disconnected from the drive lines, the collision cannot cause an increase in reactivity. Therefore, no reactivity event can occur as a result of this incident. The collision can cause damage to the CRDM's which would result in delaying startup because of repair time. The collision can also cause failure of the CRDM cover gas seals, which would result in the release of cover gas to the Reactor Containment Building and the Reactor Service Building. This event is not as severe as the reactor cover gas release event, because the seal failure would result in a more gradual leakage of cover gas. At the extreme limit, the leakage of fission gas from the reactor in this event will not exceed that which was analyzed in Section 15.5.2.4.

15.5.3.1.3 Conclusions

Because of the inherent design features and operating procedures for the Ex-Vessel Transfer Machine, the likelihood of a collision of this machine with the CRDM is extremely unlikely. However, analysis of the consequences, should this event occur, show that release of reactor cover gas is within the design limits discussed in Section 15.5.2.4.

TABLE 15.7-1
OTHER EVENTS

Section No.	Events	Potential Limiting Parameters	Comments
15.7	Other Events		
15.7.1	Anticipated Events		
15.7.1.1	Loss of One D.C. System	None	No adverse operating conditions have been identified with this event.
15.7.1.2	Loss of instrument or valve air system	None	Detailed description of failure effects or safety-related instrument air supplies, if any will be provided in the FSAR.
15.7.1.3	IHX Leak	None	Core sees normal shutdown.
15.7.1.4	Off-normal cover gas pressure in the reactor primary coolant boundary	None	No adverse operating conditions associated with this event.
15.7.1.5	Off-normal cover gas pressure in IHTS	None	No adverse operating conditions associated with this event.
15.7.2	Unlikely events		
15.7.2.1	Inadvertent release of oil through the pump seal (PHTS)	None	No adverse consequence identified at this time.
15.7.2.2	Inadvertent release of oil through the pump seal (IHTS)	None	No adverse consequence identified at this time.
15.7.2.3	Generator breaker failure to open at turbine trip	None	Core sees only normal shutdown.
49 15.7.2.4	Rupture of RAPS Cryostill	<2.5 REM (integrated 2-hr dose at the site boundary)	Consequences will be within suggested guideline doses.
15.7.2.5	Liquid rad-waste system failure	3.7×10^{-6} REM @ site boundary 3.05×10^{-7} REM @ LPZ	Consequences are well within the suggested guideline doses.
15.7.2.6	**Failure in the EVST NaK System**	**None**	**No adverse consequences associated with these events.**
15.7.2.7	**Leakage from sodium cold traps**	**None**	**No adverse consequences associated with these events.** 8
49 15.7.2.8	Rupture in RAPS Noble Gas Storage Vessel Cell	<2.5 rem (integrated 2-hr dose of the site boundary	Consequences will be within suggested guideline doses.

TABLE 15.7-1

OTHER EVENTS (Cont'd.)

Section No.	Events	Potential Limiting Parameters	Comments
15.7.3	Extremely unlikely events		
15.7.3.1	Leak in a core component pot.	~3200°F Center Fuel Pin	Only slight cladding melting. Fission gas release within umbrella of Section 15.5.2.3. 12
59 15.7.3.2	Spent fuel shipping cask dropped from maximum possible height	8.89 x 10^{-7} REM Whole Body @ SB (2-hr) 1.13 x 10^{-6} REM Whole Body @ LPZ (30-day)	Doses are well within the suggested guidelines.
20 59 15.7.3.3	Maximum possible conventional fires, flood, and storms	None	None
15.7.3.4	Failure of plug seals and annuli	None	No adverse consequences associated with this event
15.7.3.5	Fuel rod leakage combined with IHX and steam generator leakage	None	No adverse consequences associated with this event
15.7.3.6	Sodium Interaction with Chilled Water	None	None 15

15.7.2 Unlikely Events

15.7.2.1 Inadvertent Release of Oil Through Pump Seal (PHTS)

15.7.2.1.1 Identification of Causes

The primary sodium pump has oil-lubricated bearings and/or seals above the pump tank which contains sodium. The seals will be designed to prevent oil leaking into the pump tank for all modes of operation.

The primary pump concept incorporates a seal lubrication system with a fixed total oil inventory (see Figure 5.3-14a). Oil that leaks through the lower seal will be collected in a lower seal leakage tank and pumped to waste during servicing. Abnormal leakage must be made up by deliberate manual action to open the system and add oil. The lower seal leakage tank has the capacity to hold the total seal oil inventory and thereby precludes any seal leakage from entering the pump tank in the event of an abnormal leak rate. An additional and last barrier preventing seal leakage from entering the pump tank sodium is provided in the pump design by a shaft oil slinger and reservoir located below the normal seal
59 rubbing faces.

Any oil overflowing the lower seal leakage collection tank or running down the pump shaft is collected in a reservoir which has a capacity in excess of the total oil inventory. The primary pump concept, therefore, would require a combination of independent failures to occur coupled with a deliberate manual addition of oil to the system before oil could enter the pump tank.

Although the release of oil from the primary pump seal to the primary sodium is considered an extremely low probability event, the results of such an event have been evaluated. Two potential effects have been identified:

1. Plugging Effects
2. Reactivity Effects

19

15.7.2.1.2 Analysis of Effects and Consequences

If it is postulated that the oil were to be released to react with the primary sodium, the following analysis is presented. 19

The oil above the seal would flow down the pump shaft and vaporize, or react with the sodium in the pump tank. The reaction of oil and sodium will result in the release of hydrogen and carbon. The hydrogen gas may be detected in the inert gas monitoring system if the leak is large, and can be used as a means of detecting a sodium-oil reaction. The carbon compounds will either float on the sodium, dissolve (on the order of one ppb), or sink to the bottom of the pump tank. These are small particles which are easily fractured.

The release of these particles from the pump tank to the primary loop will depend upon the manner in which the pump is operating. If the pump is shutdown, the solids will stay in the pump tank.

If the pump continues to operate after a seal failure, the reaction products would eventually go into the primary loop. In the present pump concept the pump tank will contain approximately 800 gal. of sodium, and will be changing at 700 gpm due to flow from the IHX return (200 gpm) and bearing return flow (500 gpm).

Plugging Effect

Three different conditions were evaluated as follows:

A. To calculate the maximum plugging temperature in the pump discharge, the following conservative assumptions were made:

1. Pump tank temperature is 1000°F.

2. The pump tank vents to cover gas system through the pump standpipe bubbler. Maximum gas pressure is 12 in. W.G. plus equivalent static head of sodium @1000°F for elevation between normal RV sodium level and normal level in the pump tank. This assumes no pump draw down. Pressure is 97 in. W.G.

3. Oil leaks into the pump tank at a rate just sufficient to saturate the pump tank sodium volume of 800 gallons with H_2 at the temperature and pressure above. This results in a concentration of 121 ppm of H_2 in the pump tank sodium.

4. The pump tank mixture is drawn into the pump and mixed with primary sodium at the ratio of 700 gpm/34000 gpm (IHX and bearing return flow vs pump discharge flow).

5. The resultant pump discharge contains 2.5 ppm of disolved H_2 and the plugging temperature is 460°F.

B. To calculate the maximum plugging temperature in the core and the remainder of the system the following conservative assumptions were made:

1. Assume that leakage continues as in the previous condition until the entire 6 gallon inventory of oil in the seal system has leaked into the tank which is at 1000°F. 19

plugging temperature was found to be on the order of 440°F well below the minimum operating temperature of approximately 640°F, (2) during Refueling or Hot Standby the plugging temperature was found to be well below 377°F which is below the 400°F temperature for Refueling conditions, and therefore presenting no safety problems, and (3) the potential reactivity effect associated with this event is of such a small nature that the consequences to the core are considered insignificant. 19

15.7.2.2 <u>Inadvertent Release of Oil Through the Pump Seal into Sodium (IHTS)</u>

The release of oil in the PHTS has been discussed in Section 15.7.2.1. The release of oil to the IHTS from the pump oil bearing requires the failure of multiple barriers designed to prevent such a release. If oil contamination of the IHTS sodium did occur, it could be detected by monitoring the seal oil inventories or from a chemical analysis of sodium samples. An undetected
59 loss of the entire seal oil supply to the IHTS sodium would have consequences for the IHTS heat transport capability no more severe than those evaluated in Section 15.3.2.2 (Single Intermediate Loop Pump Seizure), Section 15.3.3.5 (Intermediate Loop Pipe Break), or Section 15.7.2.1 (Inadvertent Release of
59 Oil through Pump Seal (PHTS)).

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.

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 argon 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 argon is condensed to a liquid as it passes through the coiled tubing in the cryostill condenser, which is surrounded by liquid nitrogen. The process gas and liquid nitrogen lines penetrate the cryostill wall at four locations sealed by welding. A major rupture of the cryostill could vent the radioactive process gas, liquid argon in the cryostill,
59 and the liquid nitrogen to the cold box cell atmosphere. Although such a major rupture is not expected to occur, it would result in both a significant activity release and a significant increase in cell-atmosphere pressure if the cell were closed. 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 1 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 process gas 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 1.5 cf of liquid nitrogen released from the cryostill reboiler, plus 1 minute of liquid
59 nitrogen in-flow; after this time, the nitrogen in-flow is automatically valved off by a cell pressure or radiation signal. Seventeen hundred 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 cubic ft., which corresponds to 1191 scf of argon.

Redundant radiation monitors, located in the RAPS cold box cell, will 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 requiring the cell to have a tighter leakage specification.

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 a response time of 30 minutes 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. Another action is to close one of two maintenance valves
59 located on either side of the flow control valves (see Figure 11.3-4). During the maximum operator response period of 30 minutes, 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.

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 6985 scf) is the total of the above 3191 scf. The resultant calculated initial pressure is 6.7 psig. The initial 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 ($\beta+\gamma$) 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 6.7 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 leakage is within the specification. Additional testing of the cell will be required if the cell is accessed.

TABLE 15.7.2.7-1

OFF-SITE DOSE RESULTING FROM A POSTULATED COLD TRAP FIRE

Organ	2 Hour Dose (Rem) At Site Boundary (0.42 Mile)	30 Day Dose (Rem) LPZ (5.0 Miles)
Bone	1.02×10^{-3}	3.03×10^{-4}
Lung	7.51×10^{-4}	2.22×10^{-4}
Thyroid	4.17×10^{-5}	1.23×10^{-5}
Whole Body	7.81×10^{-5}	2.31×10^{-5}
Skin	5.13×10^{-7}	1.51×10^{-7}

15.7.2.8 Rupture in RAPS Noble Gas Storage Vessel Cell

15.7.2.8.1 Identification of Causes and Accident Description

The RAPS noble gas storage vessel normally contains radioactive gas which is off-loaded annually from the RAPS cryostill. It contains mainly argon (including argon-39), but also krypton and xenon isotopes, both stable and radioactive. The gas is bled slowly from the vessel into CAPS, so that its pressure normally decreases over the annual period. A rupture of this vessel or of associated piping and components could release radioactive gas at above-ambient pressure into the noble gas storage vessel cell. Although such a rupture is not expected to occur, a failure based on the scenario described below would require a leak-tightness specification for the cell. This accident is presented in order to report the maximum credible resultant doses to unrestricted areas and the indicated cell leakage specification.

15.7.2.8.2 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, and that one year's accumulation of noble gas isotopes, under that condition, by the cryostill has been off-loaded to the noble gas storage vessel. Furthermore, it is assumed that some unspecified maintenance operation has required that the new fresh cryostill charge also be off-loaded to the storage vessel/this in quick sequence, so that the storage vessel contains two charges and is approximately at maximum pressure. Each cryostill charge of 1.5 cu ft of liquid argon corresponds to 1191 scf of gas, therefore, 2382 scf of gas will be released
59 into the cell which has a net volume of 6466 acf.

Assuming also instant temperature equilibration to ambient, the
59 resultant initial pressure in the cell is calculated to be 19.9 psia, or 7.3 psig. The initial radioactivity inventory is shown in Table 15.7.2.8-1.

The design basis for the leakage specification is that the integrated 2-h site boundary dose ($\beta+\gamma$) be limited to a value below 10% of the 10 CFR 100 value following the rupture incident in the cold box.

15.7.2.8.3 Conclusions

The postulated RAPS noble gas storage vessel rupture incident requires that the vessel and associated components 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.
49 The analysis of the scenario described in Section 15.7.2.8.2 shows that a

15.7.3.1.2 Analysis of Effects and Consequences

Thermal Consequence Analysis

59 A. Thermal Model

The thermal calculations for this accident were performed using the computer codes TAP-4F (Thermal Analyzer Program) and DEAP (Differential Equation Analyzer Program) which are listed in Appendix A of the PSAR.

The thermal analysis network modeling a spent fuel assembly in a core component pot surrounded by the EVTM cold wall is shown in Figure 15.7.3.1-3.

The analysis used the following assumptions as input:

	Fuel assembly decay power	20 kW
44	Heat generation within fuel assembly	86%
44	Heat generation outside fuel assembly (by gamma heating)	14%
	Air flow for coldwall cooling	4,600 lb/hr
	Emissivity for fuel and CCP	0.4
	Emissivity for EVTM coldwall	0.2*

59 B. Thermal Analysis - CCP Submersed After Normal Transfer Time

The analysis results are shown in Figures 15.7.3.1-4 and 15.7.3.1-5. Figure 15.7.3.1-4 is a plot of the maximum transient temperatures of the center fuel rod cladding (hottest rod), cladding of a fuel rod in the outer row, the fuel assembly duct, the core component pot, and the coldwall. The low decay heat flux in spent fuel, as compared to the much larger heat flux during reactor operation, produces temperature differentials of less than 20°F between fuel rod center and cladding. As soon as the sodium has drained below the level of the fueled region, the temperatures of the fuel assembly and CCP rise, and reach steady-state values after about one hour. Clad melting in the center fuel rod starts after about 17 minutes. The clad melting zone progresses to fuel rods in the outer row in about 30 minutes. After about one hour, 90% of the clad in the fueled region has melted, and the fuel duct reaches the melting point in a localized circular zone.

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*Coldwall emissivities are normally expected to be 0.7 or larger. A degradation of coldwall emissivity to 0.2 was postulated since the coldwall was assumed to be covered with a film of recondensed sodium due to the accident. This is a conservative assumption resulting in higher fuel and CCP temperatures. 29

It should be noted that not the fuel assembly but the core component pot is attached to the EVTM grapple. The fuel assembly, standing unrestrained in the pot, experiences only stresses due to its own weight, and due to thermal gradients. The radial clearance between fuel assembly hexagonal duct corners and CCP is about 0.8 in. The CCP temperature after one-hour is about 2050°F, well below the melting point of stainless steel, and has almost reached the steady-state value.

Figure 15.7.3.1-5 gives the axial and radial steady-state temperature distribution along a vertical cut through a part of the EVTM. All maximum temperatures appear at the midplane of the fueled region.

The analysis indicates the steady-state temperatures (see Summary Table 15.7.3.1-2) for the fuel material in a spent fuel assembly in a CCP without sodium in the EVTM are well below the melting point for the mixed oxide fuel.

The isothermal lines near the coldwall show that the nearest seals (lower coldwall seals) will stay below 100°F. All volatile fission products released from the fuel rods into the EVTM will therefore be contained in the EVTM. The maximum pressure in the EVTM due to argon gas heating by the dry fuel assembly, and due to release of fission gas and helium from the fuel
44 rods, could amount to about 26 psia, well within the design pressure of 30 psia.

The results of this analysis indicate the following consequences for a "dry" CCP in the EVTM, if the CCP is not submerged under sodium within the normal CCP transfer time (56 minutes):

(1) release of volatile fission products from all fuel rods to the EVTM containment

(2) extensive fuel rod cladding melting

(3) localized fuel assembly duct melting

(4) no fuel melting

(5) no CCP melting; the CCP can support the fuel assembly

(6) no seal overheating; the EVTM can contain the fission products with only limited diffusion of activity resulting.

59 C. Thermal Analysis - CCP Submersion Delayed

From these consequences it was concluded that the event sequence could be safely terminated and no public safety hazard would ensue, even if all lines of defense preventing this event (see Figure 15.7.3.1-1) were rendered ineffective.

An additional investigation was carried out to examine the consequences of this event if the normally expected CCP transfer time from reactor sodium to EVST sodium (56 minutes) were to be prolonged. Two potentially worse cases were postulated and analyzed, again to explore the worst potential consequences of this event. The two cases were based on the following assumptions: 29

(1) fuel break-up and collapse in a packed-bed configuration

(2) relocation of fragmented fuel outside of the fuel assembly in the CCP bottom.

A scenario has been postulated in which fuel pellets, stripped of their cladding, could break-up in smaller pieces, collapse, and form a packed-bed type structure. The restructured fuel in its new configuration could have a higher energy density than in its original geometry as rods, depending on the size and packing of the fuel particles. This, in turn, could cause the temperature of the fuel assembly duct to exceed the melting point in a localized zone, and could result in a loss of structural integrity of the fuel assembly.

59 1. Improbability of Fuel Collapse

Results of in-pile and out-of-pile experiments with LMFBR fuel assemblies subjected to high temperatures support the above-described scenario as being conservative. Fuel behavior tests performed in the transient reactor test (TREAT) facility at ANL, in support of the analyses for the hypothetical loss-of-core-coolant accident, indicated that fuel pellets did not fall apart once the cladding had melted and gave no further support. These tests (Reference 1) were performed with pre-irradiated fuel and showed that the fuel pellets sintered together with a strong, dense column formed by the equiaxed region. The fuel rods retained their identify as columns and bowed, rather than crumbled as individual pellets or pieces. Similar test
59 results were obtained when new fuel was subjected to loss-of-coolant experiments in the TREAT reactor. Although fuel cladding had melted off in these experiments, the fuel rod pellets remained stacked at termination of the transient (Reference 2). Intact fuel columns were also observed in several loss-of-coolant experiments performed as in-pile transients on new fuel in the Reactor Centrum Nederland (RCN) (References 3, 4, and 5).

Out-of-pile tests ("dry capsule" experiments), reported in Reference 6, also showed that the fuel column of an irradiated pin heated to its solidus held together after the cladding melted, and remained essentially intact even after considerable bowing and buckling.

During reactor operation, a break-up of the solid fuel occurs due to high thermal gradients in the fuel during reactor power transients, and due to changes in the grain structure of mixed oxide fuel. It is well established that initial fuel break-up is followed or accompanied by a crack healing process whose effectiveness is a function of fuel temperature and reactor operating time. Research at ANL has shown that uranium oxide, for example, exhibits crack healing when exposed to temperatures above 2900°F for a period of 48 hours, and recovers its as-fabricated strength. This crack healing does not occur as a consequence of solidification of molten fuel, but proceeds by a mass-transport mechanism involving grain growth and diffusion (Reference 7).

The maximum rate of temperature increase in a fuel rod during the postulated accident was calculated to be about 2°F/sec. This is less than 1/30 of the temperature rates representative of in-core, loss-of-flow accidents and their experimental simulation (References 8 and 9). The lower heating rate of the fuel rods has the following consequences: 29

1. The fuel pellets experience less severe temperature gradients, reducing the potential for thermal shock induced cracks.

2. The less rapid heating rate allows for redistribution of fission gas trapped in grain boundaries and for gas pressure equalization within the entire fuel rod. (Reference 10)

From the heat transfer analysis of the postulated accident and from the above considerations, the following observations are derived:

1. The maximum steady-state fuel temperature in the center fuel rod and in all other fuel rods is well below the fuel melting point.

2. Clad melting could occur over approximately 36 in. length of the fuel rods. The molten cladding will solidify near the lower axial blanket where the temperatures are below the melting point of stainless steel, as evident in Figure 15.7.3.1-5.

59 3. A collapse of the fueled region with resulting dispersal and relocation of fuel fragments will not occur after the cladding of the fueled region has melted.

Despite these considerations, the accident sequence has been extended and fuel fragmentation, followed by fuel collapse into a packed-bed structure, has been hypothesized. The packed bed was postulated to be supported by the lower axial blanket, since cladding temperatures in the lower axial blanket are substantially below the melting point, see Figure 15.7.3.1-5. This region will therefore retain its structural integrity and is the most likely place for fuel to collect if the pellets do fragment and collapse.

The fragments were postulated to be all of equal size, with a representative diameter of 0.1 in. This implies the break-up of each fuel pellet into 12 spherical pieces with equivalent mass. The formerly 36 in. long fueled region consisting of stacked fuel pellets (encased in a cladding tube) could thus be compacted to a length of 28 in., consisting of a "pebble bed" of fuel particles. The packed-bed fuel configuration could lead to a temperature increase due to the higher energy density and reduced effective conductivity. The latter would cause the fuel to retain more heat and transmit less to the fuel assembly duct.

The transient temperature distribution for this hypothetical fuel configuration is shown in Figure 15.7.3.1-6. It can be noted that, due to the reduced effective fuel conductivity, the fuel duct temperature near the midplane of the fueled region is actually lower than in the case when fuel pellets remain stacked. After one hour, this effect is counter-balanced by the higher temperature of the compacted fuel fragments. If the event were not terminated at the normally expected time (56 minutes) by submersion of the CCP under EVST sodium (see Table 15.7.3.1-1) the fuel assembly duct would start to melt in a circumferential zone after about 1.05 hours (63 min.). The transient axial temperature distribution plotted in Figure 15.7.3.1-7, shows that the high temperatures are axially confined to the fueled region and extend only partially into the axial blankets. 29

59 3. Thermal Analysis - Fuel Redistribution Outside Duct

The accident sequence has been further extended to investigate the consequences of a loss of fuel assembly integrity. It was hypothesized that fuel particles might leave the fuel assembly duct, fall down in the annular space between hexagonal housing and circular CCP, and accumulate at the CCP bottom. Due to considerable geometrical distortion of the fuel assembly near the fueled region (from overtemperature during this event) and the presence of solidified, previously molten, material from the fuel assembly duct and cladding near the (colder) CCP wall, only a restricted passage for fuel particles will exist. Only a small amount of fuel material would therefore be expected to fall to the bottom of the CCP. 25% of the fuel material was judged to be the upper limit of this amount. However, the value was varied up to 100%, to show the effect of this parameter.

The calculated peak transient temperatures in the fuel, CCP, and nearest seal are plotted in Figure 15.7.3.1-8 for these amounts of fuel present at the CCP bottom.

After an initial drop of the fuel temperature due to the fuel relocation in a cold area, the fuel temperature rises slowly. The peak CCP temperature at the CCP side and bottom, and the temperature of the nearest seal (lower cold wall) also rise slowly. The calculations show that at about 1.5 hours later initiation of the event, i.e., after loss of sodium from the CCP, the transient temperatures in the fuel and CCP reach steady-state conditions if 25% of the fuel fragments are accumulated at the CCP bottom. The steady-state temperatures are as follows:

Center of Fuel	$3140^{o}F$
CCP, Bottom	$1890^{o}F$
CCP, Side	$1865^{o}F$
Lower Cold Wall Seal	$260^{o}F$

The movement of fuel particles from the original fuel region within the fuel assembly to the CCP bottom has the beneficial effect of lowering the energy density of the heat source and thereby lowering the temperatures of the fuel and its surrounding. This explains the lower temperatures when 25% of the fuel has accumulated in the CCP bottom. A stress analysis indicated that the stresses in the CCP, due to support of its own weight and that of the fuel assembly, are very low. The tensile stress in the tubular part of the CCP is 230 psi, the compressive stress at the CCP bottom is 700 psi. This compares to an ultimate strength of about 6000 psi for the CCP material (SS 304) at $1900^{o}F$.

Fission Product Release Analysis

The analysis and the supporting temperature data presented above show that the postulated accident will not lead to any fuel melting, but 29 could lead to extensive clad melting. It can be conservatively estimated that most of the fission products which are volatile in the temperature range of about 2800° to 3500°F are released into the EVTM. This temperature range corresponds to the maximum axial steady-state temperature which the fuel rods in the assemblies reach, dependent on their radial location (see Figure 15.7.3.1-5).

Table 15.7.3.1-3 lists those fission product elements contained in a fuel assembly of the equilibrium core at the end of cycle which are in the molten or vapor phase below 3500°F. The entire isotopic content of fission products is given in Table 12.1-35. The fission products of Table 15.7.3.1-3 are assumed to be released into the EVTM either partially or completely, depending on their melting points and partial pressures.

The maximum cold wall temperature of the EVTM was calculated to be 435°F (see Figure 15.2.3.1-5). This "hot spot" is at an axial location coresponding to the midplane of the fueled region in the fuel assembly. The nearest seals are 6.3 ft downwards at the lower end of the cold wall near the air inlet module. These seals will not reach temperatures higher than 260°F during this accident. The elastomer seals will contain the radioactive fission products in the EVTM. Permeabilities of elastomeric seals have been experimentally determined up to 300°F (see Reference 1 of Section 15.5.2.3).

It was therefore concluded that all fission products which are in the liquid or gaseous phase above 260°F are plated out on the cold surfaces in the EVTM, specifically at the cold wall and/or near the seals. Only fission products which are in the liquid or gaseous phase at or below 260°F were considered to leave the double seals by diffusion.

The diffusion rates of fission products from the EVTM to the RSB/RCB are given in Table 15.5.2.3-3. In determining these diffusion rates, it was assumed that about 15% of all EVTM seals are at a temperature of 300°F and 85% at 150°F. This assumption is conservative with respect to the postulated accident, since only one set of seals, representing about 1% of all EVTM seals, could exceed a temperature of 150°F. The diffusion rates of Table 15.5.2.3-3 are therefore higher than those which would be expected as a result of the accident discussed here. Fission products other than those listed in Table 15.5.2.3-3, but which are volatile at EVTM seal temperatures, were discussed in the response to Question 001.212. As shown there, only Cs and Rb need to be considered, yet the radioactivity contribution of all Cs and Rb isotopes combined, passing through the hottest EVTM seals, is smaller than that of all other volatile fission product isotopes (i.e. mainly of Xe 133, I131, and I132) by a factor of approximately 10^5 at 36 hr. after reactor shutdown, and by a factor of more
59 than 10^3 at 20 days after reactor shutdown.

Based on the above considerations, the radioactivity leakage from the EVTM to the RSB/RCB due to the postulated accident will be less 29
than, or is enveloped by the leakage presented in Section 15.5.2.3.

15.7.3.1.3 Conclusions

Based on the analysis shown by the steady-state temperatures in Table 15.7.3.1-2 for a postulated spent fuel assembly in a CCP without sodium in the EVTM, no fuel melting, but extensive clad melting of the fueled zone, is expected. Though no rearrangement of the fuel pellets is anticipated, a hypothetical redistribution of fuel fragments was found not to raise the temperature of the nearest EVTM seals beyond 260°F.

This accident could lead to release of fission products which are volatile at temperatures up to 3500°F into the EVTM, but only fission products which are volatile at 260°F could diffuse through the double EVTM seals. This fission product release from the EVTM is discussed in Section 15.5.2.3, and represents the limiting release case. The off-site
59 exposures reported in Section 15.5.2.3 are well within the dose limits.
29

References to Section 15.7.3.1:

1. "Fuel Dynamics Experiments Supporting FTR Loss-of-Flow Analyses," L. W. Deitrich, et al., Proc. Fast Reactor Safety Meeting, CONF-740401-P1, Page 239, April 1974

2. "Fuel Movement in R3, R5, and R6 Loss-of-Coolant Simulations in TREAT," A. De Volpi, et al., Trans. Am. Nucl. Soc., Vol. 21, Page 288, June 1975

3. "Loss-of-Cooling Experiments," Fast Reactor Program Combined Second and Third Quarters 1971 Progress Report, E. K. Hoekstra (Comp.), RCN-164, Page 95, January 1972

4. "Loss-of-Cooling Experiments," Fast Reactor Program Second Quarter 1973 Progress Report, E. K. Hoekstra (Comp.) RCN-190, Page 39, August 1973

5. "Loss-of-Cooling Experiments," Fast Reactor Program First Quarter 1975 Progress Report, E. K. Hoekstra (Comp.), RCN-228, Page 17, July 1975

6. "Studies of Fast Reactor Fuel Element Behavior under Transient Heating to Failure," R. R. Steward, et al., Argonne National Laboratory Report ANL-7552, 1969

7. "Crack Healing in UO_2," J. T. A. Roberts and B. J. Wrona, Journal of the Am. Ceramic Soc., Vol. 56, No. 6, Page 297, June 1973

8. "Laboratory Studies on Melting and Gas Release Behavior of Irradiated Fuel," E. T. Weber, et al., Proc. Fast Reactor Safety Meeting, CONF-740401-P2, Page 641, April 1974

9. "Thermal-Shock Cracking in UO_2 During Power Transients," B. J. Wrona, et al., Trans. Am. Nucl. Soc., Vol. 22, Page 419, November 1975

10. "Internal Pressurization in Solid Fuel Due to Transient Fission-Gas Release," C. C. Meek, et al., Trans. Am. Nucl. Soc., Vo 22, Page 418, November 1975

29

15.7.3.2 Spent Fuel Shipping Cask Drop from Maximum Possible Height

15.7.3.2.1 Identification of Causes and Accident Description

The maximum height for a potential Spent Fuel Shipping Cask (SFSC) drop in the CRBRP is the 72-ft. vertical distance from the operating floor of the RSB to the bottom of the SFSC handling shaft. 12

Section 9.1.4.8 discusses the design features preventing an SFSC drop in the CRBRP. These consist mainly of handling the SFSC above the operating floor of the RSB and within the cask handling shaft only with the double reeved RSB bridge crane (125 ton capacity) using rigging specially designed and tested
59 for the SFSC. The operational requirements of RDT Standard F8-6T applying to critical items will cover all moves of the SFSC when handled by the RSB bridge crane. Due to these design features and operational precautions, dropping of an SFSC within the RSB is considered a hypothetical event. 12

As identified in Section 9.1.2, the SFSC will be licensed separately. The cask is designed to withstand a hypothetical accident condition of a 30-ft. free drop as specified in 10CFR71. Under these conditions, the cask is designed to maintain its structural integrity with zero leakage of its radioactive content. 12 This design condition satisfies the requirements of 10CFR71 which specify radioactivity release limits for a cask under hypothetical accident conditions. 12

15.7.3.2.2 Analysis of Effects and Consequences

The free fall impact energy of an SFSC dropped to the bottom of the handling shaft is smaller than that for which the cask is designed, as discussed in Section 9.1.4.8.

59

Though a 72-ft. drop to the bottom of the cask handling shaft is not expected to occur and would not result in a break of the SFSC containment, a
59 break of the outer cask containment and release of radioactivity through the seals of the inner cask containment has been postulated and analyzed. The purpose of the analysis is to demonstrate the inherent safety margins available, even under the following conservative assumptions:

1) The SFSC is loaded with core assemblies of the highest fission gas inventory and a short decay time. This assumption contains two design margins with respect to radioactivity:

59 a. The spent fuel assemblies are assumed to be of the highest power and shipped at 80 days after reactor shutdown. This exceeds the design requirements that spent fuel shipment commence no sooner than 100 days after reactor shutdown. Administrative procedures will actually require that the highest powered fuel assemblies will be the last ones of one refueling batch to be shipped i.e., at a decay time substantially greater than 100 days. 12

b. The fission gas inventory of six highest powered spent fuel assem-
blies and three blanket assemblies is considered. This inventory
59 corresponds to the maximum total core assembly decay heat load of the
SFSC (26 kw).

2) All fuel rods in the six fuel assemblies are assumed to fail, releasing
the entire fission gas inventory instantaneously into a helium gas
59 space in the SFSC canister. The canister forms the inner containment of
the SFSC. The long-lived, volatile radionuclides of this inventory with
59 significant activities at the time of shipping are shown in Table 15.7.3.2-1.

3) The fission gas is assumed to leak through the inner and outer containments of the SFSC at the maximum allowable (see SFSC SAR) inner containment seal leak rate for helium of 6 x 10^{-5} Std cm^3/sec, adjusted for the higher canister pressure after the drop. This assumption does not take credit for the outer containment seals which have a maximum allowable leak rate of 4.3 x10^{-6} Std cm^3/sec at 10 psi differential. It also does not take account of the fact that the outer containment is at a pressure lower than ambient (nominal 10 psia), which only allows for leakage of gas out of the outer cask containment during the early days of the SFSC shipping. All SFSC seals consist of stainless steel O-rings. Low leakage rates will be assured by appropriate leak testing.

4) The maximum steady temperature near the canister seals was calculated to be 350°F. All fission products which are volatile at this temperature were considered to leak through the seals.

59 12

The potential off-site doses calculated and presented in Table 15.7.3.2-2 assumed that the gases that leaked from the SFSC were exhausted directly to the atmosphere via the RCB/RSB ventilation system. No credit for holdup in the
59 RCB or RSB was taken. The LPZ dose was calculated using the time integrated radioactivity release assuming continuous leakage for 30 days.

15.7.3.2.3 Conclusions 12

This accident would not present any hazard to the public, the doses 12
being well below the 10CFR100 guideline values.

TABLE 15.7.3.2-1

FUEL ASSEMBLY INVENTORY AND RELEASE RATES OF LONG-LIVED, VOLATILE FISSION-GAS ISOTOPES WITH SIGNIFICANT ACTIVITIES FOR SFSC DROP FROM MAXIMUM POSSIBLE HEIGHT

Isotope	Total Activity in One F/A at 80-Day Decay Time (Ci)	Specific Activity in Cask Gas at 80-Day Decay Time (Ci/scc)	Leak Rate from Dropped Cask (Ci/sec)
Kr^{85}	616	1.10×10^{-3}	1.24×10^{-7}
Xe^{131m}	34.0	6.11×10^{-5}	6.97×10^{-9}
Xe^{133}	11.4	2.05×10^{-5}	2.34×10^{-9}
I^{131}	185	3.32×10^{-4}	3.78×10^{-8}
Cs^{134}	3600	1.9×10^{-7}*	2.2×10^{-11}
Cs^{136}	219	0.9×10^{-8}*	1.0×10^{-12}
Cs^{137}	9930	5.2×10^{-7}*	5.9×10^{-11}
59 Rb^{86}	41.5	2.1×10^{-8}*	2.4×10^{-12}

* Based on vapor pressure of Cs and Rb at the maximum SFSC seal temperature of 350°F.

Amend. 59
Dec. 1980

Table 15.7.3.2-2

Off-Site Doses (REM) Due to Fuel Failure and SFSC Leakage

	ORGAN	10CFR100 GUIDELINE	2 HOURS SB (0.42 MILES)	30 DAYS LPZ (5.0 MILES)	
	Cloud				
	D (Whole Body)	25	9.64-7*	1.19-6	
	Inhalation				
	Lung	75	1.29-8	1.59-8	
	Thyroid	300	4.39-4	5.41-4	
59	Whole Body Inhalation	25	8.89-7	1.13-6	12

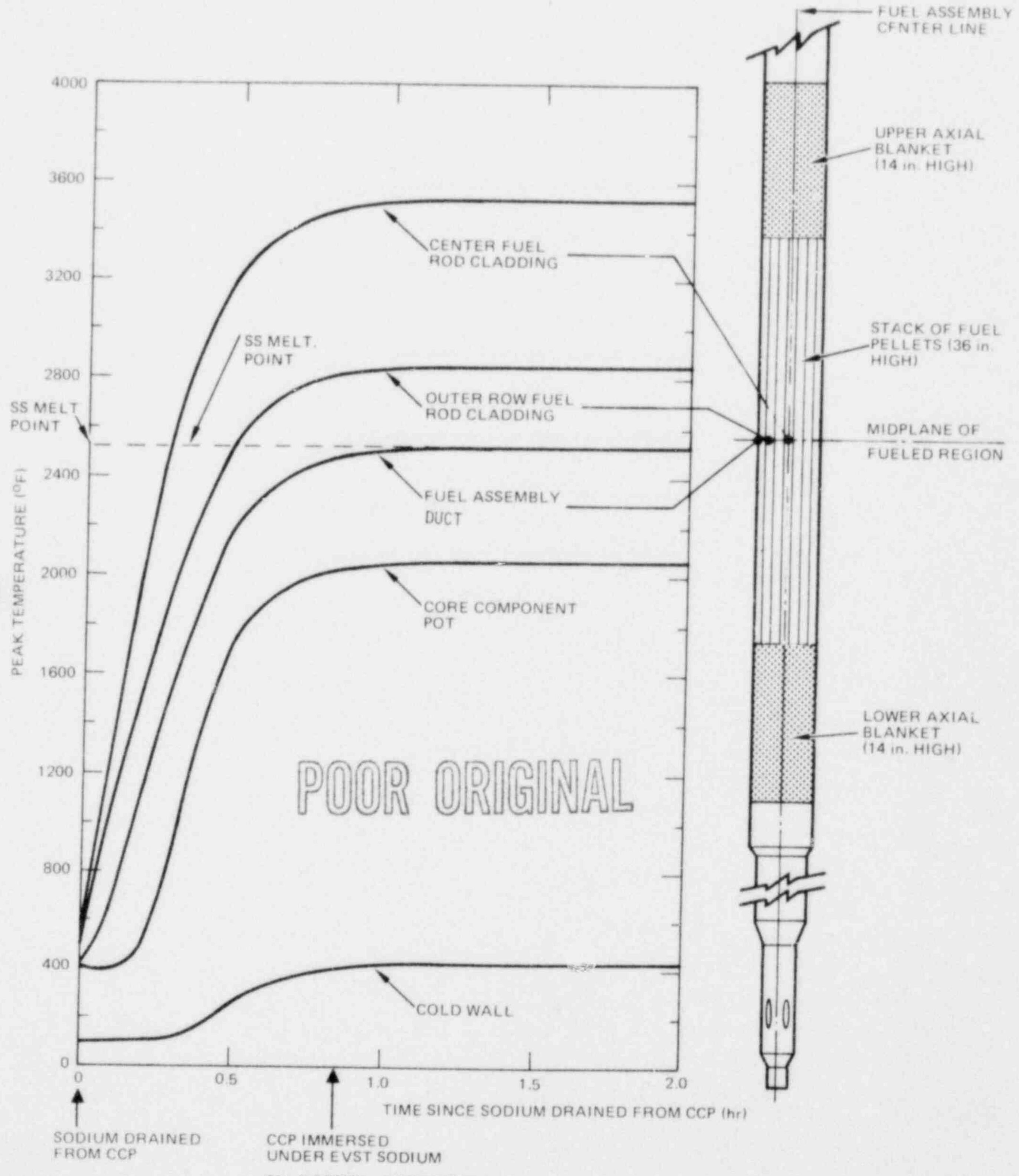

FIGURE 15.7.3.1-4 Peak Transient Temperatures in EVTM with Dry 20-kW Spent Fuel Assembly in CCP. - Stacked Fuel Pellets CCP Not Reimmersed in Sodium After Draining

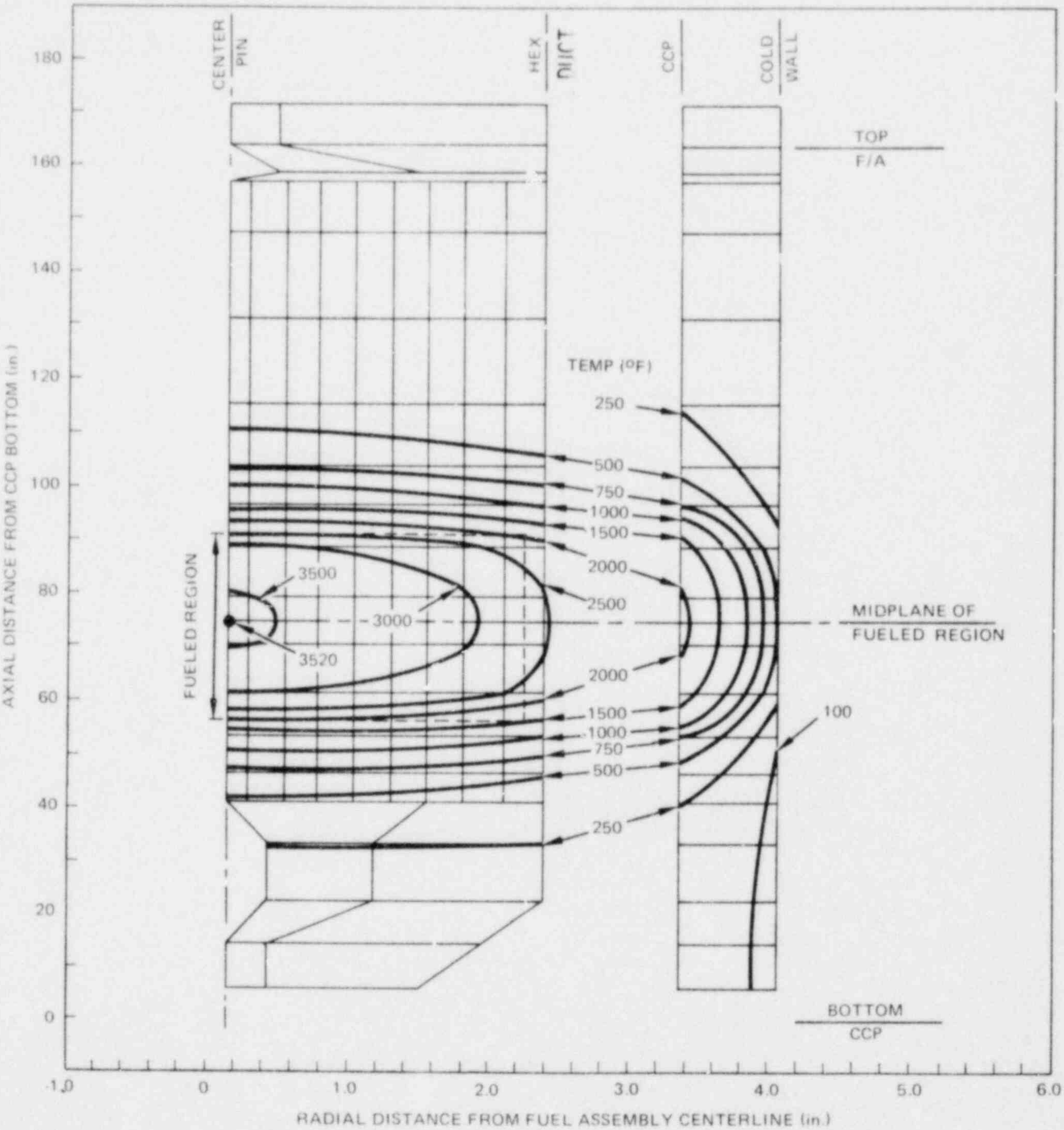

FIGURE 15.7.3.1-5. STEADY STATE TEMPERATURE DISTRIBUTION IN EVTM WITH DRY 20-kW SPENT FUEL ASSEMBLY IN CCP. – STACKED FUEL PELLETS

9

Amend. 29
Oct. 1976

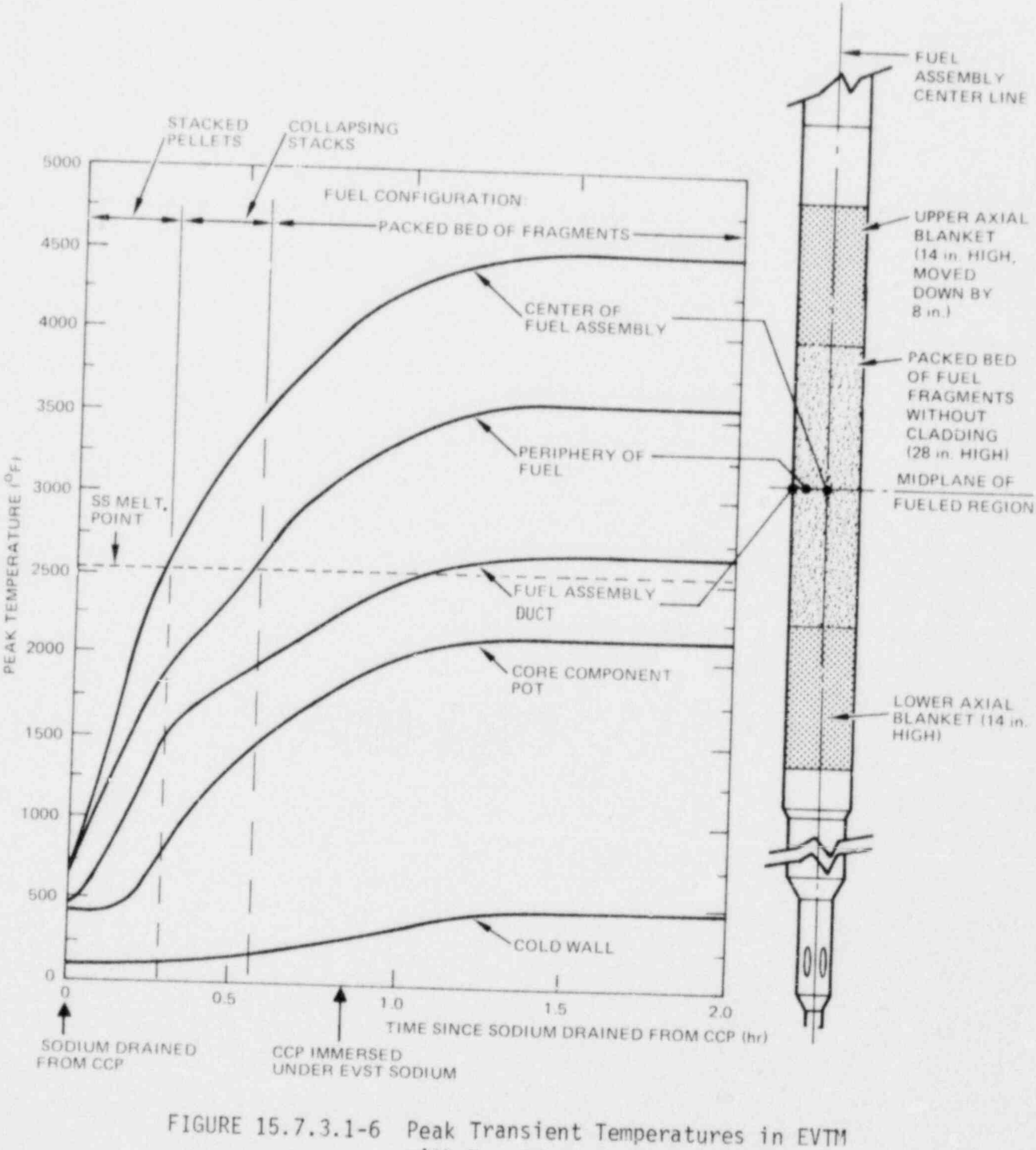

FIGURE 15.7.3.1-6 Peak Transient Temperatures in EVTM with Dry 20-kW Spent Fuel Assembly in CCP. - Packed Bed of Fuel Fragments

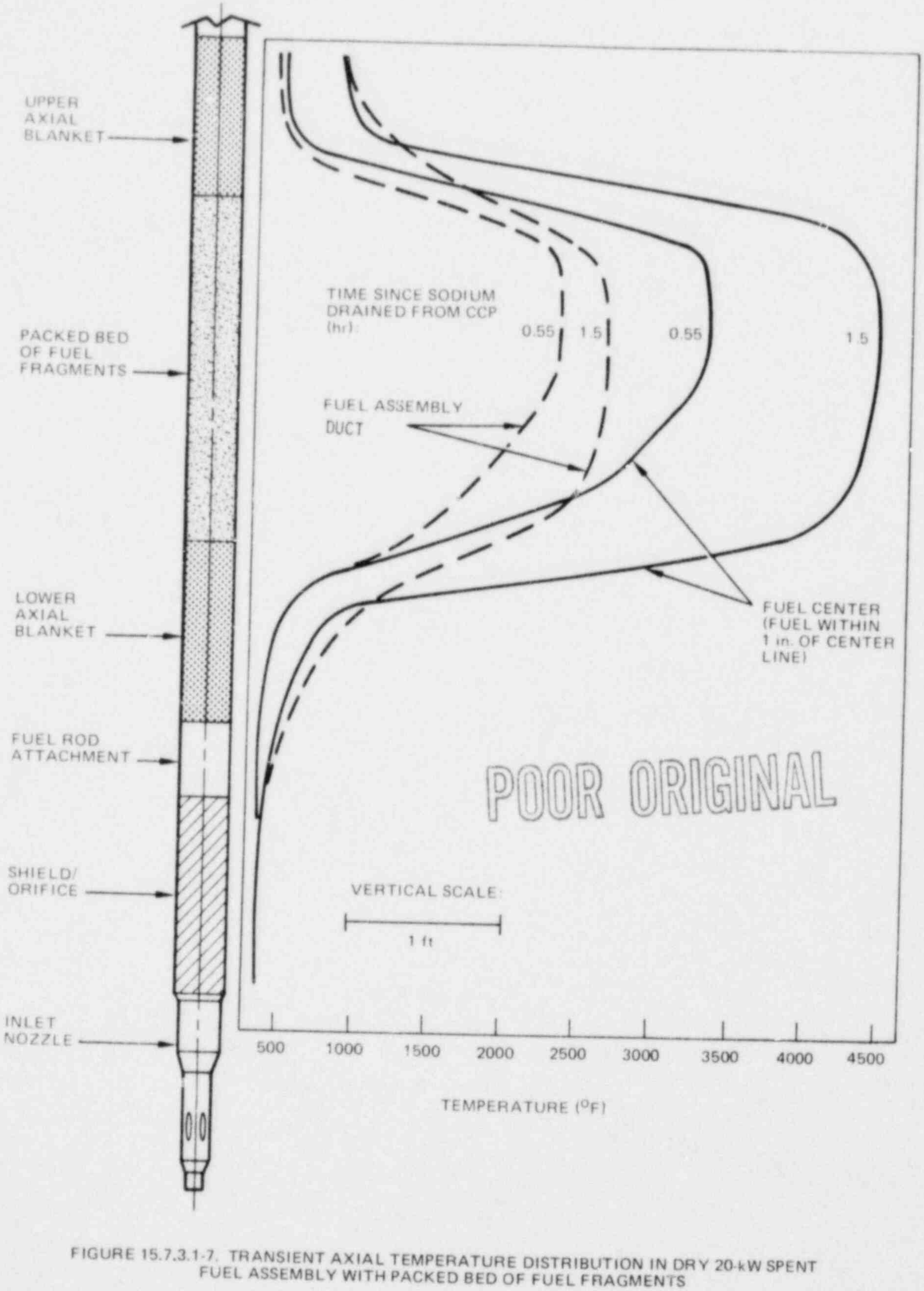

FIGURE 15.7.3.1-7. TRANSIENT AXIAL TEMPERATURE DISTRIBUTION IN DRY 20-kW SPENT FUEL ASSEMBLY WITH PACKED BED OF FUEL FRAGMENTS

CHAPTER 16 TECHNICAL SPECIFICATIONS

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CHAPTER 16 - TECHNICAL SPECIFICATIONS

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16.1 DEFINITIONS

16.1.1 Reactor Operating Condition

16.1.1.1 Rated Power

Rated power is defined as a steady state thermal power output of 975 MWt.

16.1.1.2 Thermal Power

Thermal power is the total rate of thermal energy input to the primary coolant from components inside the reactor vessel.

16.1.1.3 Normal Reactor Power Operation

The reactor is operating between and including the state points of
59 40% rated power and rated power.

16.1.1.4 Two Loop Reactor Power Operation

The reactor is critical, two loops are in operation, and the neutron flux power range instrumentation indicates not more than TBD reactor power.

16.1.1.5 Transitory Operation

The reactor is operating between the state points of refueling, hot shutdown, hot standby, and 40% rated power, exclusive of these.

16.1.1.6 Hot Standby

See Table 16.1-1.

16.1.1.7 Hot Shutdown

See Table 16.1-1.

16.1.1.8 Refueling

59 See Table 16.1-1.

16.1.1.9 Reactor Startup

Reactor Startup is the sequence of operations in which the reactor is
59 brought from hot shutdown to Normal Reactor Power Operation of 2 loop Reactor Power.

16.1.1.10 Operating Cycle

The interval between the end of one refueling outage to the end of the next subsequent refueling outage is one operating cycle.

16.1.1.11 Refueling Outage

Refueling Outage is that period of time between the shutdown of the unit prior to a refueling and the startup of the unit after that refueling. When refueling outage is used to designate a surveillance interval, the surveillance will be performed during the refueling outage or up to six months before the refueling outage. When a refueling outage occurs within eight months of the previous refueling outage, the surveillance testing need not be performed. The maximum interval between surveillance tests is 20 months.

16.1.1.12 Changes in Core Geometry

The addition, removal, relocation, or other movement of any material above the core support plate, below the upper internals or within the core barrel except for functions normally performed during reactor operation in accordance with intended design of equipment such as control rod movement shall constitute a change in core geometry.

16.1.1.13 Reactor Critical

The neutron chain reaction is self-sustaining and keff = 1.0.

16.1.1.14 Reactivity Units

Reactivity units expressed as dollars, multiplied by the effective de-
51 layed neutron fraction of 0.0034 gives reactivity units expressed as $\Delta k/k$.

16.1.2 Reactor Core

16.1.2.1 Fuel Assembly

A Fuel Assembly is an arrangement of 217 fuel rods, containing pellets of (Pu,U) O_2 and axial blanket pellets of UO_2, held in a triangular array by a spiral wire wrap spacing inside a hexagonal duct.

16.1.2.2 Blanket Assembly

51 A Blanket Assembly is an arrangement of 61 rods containing only UO_2 pellets in a triangular array.

16.1.2.3 Control Assembly

A Control Assembly is an assembly of clad boron carbide pins in a hexagonal lower guide assembly which has the same outside geometry as the fuel assembly.

Amend. 51
Sept. 1979

16.1.3.13 Protective Function

A Protective Function is the monitoring of one or more plant variables associated with a particular plant condition, and the initiation and completion of a particular Protective Action, at values of the variables established in the Design Basis. Protective Action is considered complete when the condition initiating the action is brought to a status at which the consequences of terminating the Protective Action are considered to be acceptable.

16.1.3.14 Engineered Safety Features

Engineered Safety Features are all Protective Subsystems which function:

to mitigate the consequences of an incident, and to provide for decay heat removal, for example:

Containment Systems
Reactor Guard Vessel
PHTS Major Components Guard Vessel
Residual Heat Removal System
Habitability Systems

The Reactor Shutdown System is excluded.

16.1.3.15 Class 1E System (Electrical)

The systems that provide electric power used to shutdown the reactor and limit the release of radioactive material following a design basis event constitute a Class 1E System.

16.1.4 Safety Limit

Safety limits for nuclear reactors are limits upon important process variables which are found to be necessary to reasonably protect the integrity of certain of physical barriers which guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor shall be shut down. The licensee shall notify the Commission, review the matter and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude reoccurrence. Operation shall not be resumed until authorized by the Commission.

16.1.5 Limiting Safety System Setting (LSSS)

Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting shall be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor. He shall notify the Commission, review the matter and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude
59 reoccurrence.

16.1.6 Limiting Conditions for Operation (LCO)

Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken
59 to preclude reoccurrence.

16.1.7 Surveillance Requirements

Surveillance Requirements are requirements relating to tests, calibrations, or inspections to assure that the necessary quality of a system and its components is maintained; that the facility operations will be within the safety limits; or that the limiting conditions for operation will be met.

16.1.8 Containment Integrity

Conformance with all the following conditions:

1. All automatic containment isolation valves are operable, or secured in the closed position or isolated by closed manual valves or flanges.

2. All nonautomatic containment isolation valves which are not required to be open during accident conditions are closed and blind flanges are installed where required.

3. Refueling Hatch is closed.

4. At least one door in each air lock is closed and sealed.

16.1.9 Abnormal Occurrence

An abnormal occurrence means the occurrence of a plant condition that results in any of the following conditions:

1. A safety system setting less conservative than the limiting setting established in the Technical Specifications.

2. Violation of limiting condition for operation established in the Technical Specifications.

3. An uncontrolled or unplanned release of radioactive material from any plant system designed to act as a boundary for such material in an amount in excess of the limits prescribed in Technical Specifications.

4. Failure of a component of a Plant Protection System that causes the feature or system to be incapable of performing its intended function as defined in these Technical Specifications or in the Safety Analysis Report.

5. Abnormal degradation of one of the several boundaries designed to contain the radioactive materials resulting from the fission process.

6. Uncontrolled or unanticipated changes in reactivity greater than 1% $\Delta k/k$.

7. Observed inadequacies in the implementation of administrative or procedural controls such that the inadequacy causes the existence or development of an unsafe condition in connection with the operation of the plant.

8. Conditions arising from natural or manmade events that, as a direct result of the event, require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

TABLE 16.1-1. OPERATIONAL STATE POINTS (1)

	Refueling Conditions	Hot Shutdown	Hot Standby	40% Rated Power	Rated Power
Reactor	Subcritical K_{eff} <.95	Subcritical	Critical with the power level not to exceed 5% of rated power	Critical @40% Power	Critical @100% Power
Control Rods	Primary & Secondary Rods fully inserted and disconnected	Primary & Secondary Rods fully inserted and unlatched	Secondary-Parked Primary - Critical elevation in banked configuration with Row 4 fully withdrawn	Secondary-Parked Primary - Critical elevation in banked configuration with Row 4 fully withdrawn	Secondary-Parked Primary - Critical elevation in banked configuration with Row 4 fully withdrawn
Scram Breakers	Open	Open	Closed	Closed	Closed
Turbine Generator Output Breakers	Open	Open	Open	Closed	Closed
PHTS	Pony Motor Flow, Na Temp @ 400°F ± 25°F	TBD Na Temp @ 600°F + 50°F - 10°F	Main Motor Flow (40% nominal) Na Temp @ 600°F + 50°F - 10°F	Main Motor Flow (40% nominal)	Main Motor Flow Power/Flow = 1
IHTS	Pony Motor Flow	Pony Motor Flow	Main Motor Flow (40% nominal) IHTS temp consistent with PHTS	Main Motor Flow (40% nominal)	Main Motor Flow Power/Flow. ≈ 1
SGS	Recirc Pump & Motor cooling & seal cooling/injection systems operating	Recirc Pump & Motor cooling & seal cooling/injection systems operating	Recirc Pump & Motor cooling & seal cooling/injection systems operating	In Operation	In Operation
SGAHRS	PACCs operating as required(see 1.3.8)	PACCs operating as required	PACCs operating as required	On Standby	On Standby
BOP	TBD	Supply feedwater to SGS to support main steam system heating	Steam dump operating as necessary. Feedwater system operating and maintaining proper feedwater temp.	Operating @ 40% Power Conditions	Operating @ Rated Power Conditions. Main steam flow 3.32×10^6 lbs/hr

(1) All plant operations involve operating at, or about, a set of steady state conditions or making a transition from one set of conditions to another. These sets of conditions defined as state points are characterized in this table.

A. Deinerted cell

1. Total plutonium activity - TBD curies

2. Total gross β-γ activity - TBD curies

16.3.5.4 Bases

The bases for these specifications are the accidents analyzed and reported in Chapter 15.6. In all cases, the limits assure compliance with 10CFR100.

16.3.6 Inert Gas System Cover Gas Purification System

16.3.6.1 Purity of Gas

16.3.6.1.1 Applicability

Applies to the concentration of gaseous impurities in the recycle argon (argon after processing by the RAPS).

16.3.6.1.2 Objective

To define the maximum allowable concentration of impurities in the argon to be supplied to the reactor and PHTS pump cover gas spaces.

16.3.6.1.3 Specification

(To be specified in the FSAR.)

16.3.6.2 Limiting Activity in the Radioactive Argon Processing System (RAPS)

16.3.6.2.1 Applicability

Applies to the inventory of the Radioactive Argon Processing System.

16.3.6.2.2 Objective

To define the limiting activity in RAPS.

16.3.6.2.3 Specification

59 |49| 1. The radioactive inventory in the RAPS cryostill shall not exceed TBD Ci.

2. If the above limit is exceeded an orderly shutdown of the plant shall be initiated within TBD hours after this has been determined.
59

16.3.6.2.4 Basis

The specification is designed to limit the site boundary dose to con-
59 |49| form to 10CFR100, in the event of a RAPS cryostill rupture as described in Chapter 15.7.2.

16.3.6.3 Cell Atmosphere-Oxygen Control

16.3.6.3.1 Applicability

This specification applies to the primary heat transfer cells and the EVS cooling system cells during normal operation.

16.3.6.3.2 Objective

To assure that accident design limits in inerted cells are not exceeded in the event of a large sodium spill because of a high oxygen concentration in the cell atmosphere.

16.3.6.3.3 Specifications

1. If the oxygen level in the inerted cell atmosphere is greater than 2% or less than 0.5%, corrective action shall be implemented to bring the level to within the specifications.

2. If, after TBD hours of corrective action, the oxygen level in the inerted cells is not within specification, an orderly isolation, drain,
59 or cooldown of alkali metal inventory in the cell shall be initiated.

16.3.6.3.4 Basis

The upper limit of 2% oxygen is based on the allowable level developed in the accidents analyzed in Chapter 15.6.1.1 and 15.6.1.5. The lower level of 0.5% is established to prevent nitriding.

16.3.7 Auxiliary Cooling System

16.3.7.1 Fuel Storage Heat Removal

16.3.7.1.1 Applicability

Applies to the limiting conditions for operation of the spent fuel storage facilities.

16.3.7.1.2 Objective

To ensure that no incident could occur during spent fuel storage that would adversely affect the public health and safety.

16.3.7.1.3 Specifications

Items a and c through f shall be continuously satisfied.

a. Two independent power supplies shall be available for spent fuel storage facilities and their cooling systems when spent fuel decay heat removal is required.

44 59 b. The EVST shall have at least two heat removal systems operable. Each of the two systems shall be capable of handling the maximum design heat load of 1800 kW. Prior to scheduled inspection or routine maintenance of any heat removal system the two remaining heat removal systems shall be in an operable condition. If during the inspection or maintenance period one of the two remaining heat removal systems fails, the heat removal system undergoing scheduled inspection or routine maintenance shall be returned to service within the time which would be required for the EVST sodium to reach 775°F with no cooling.

44 20 c. The two forced convection, normal, independent EVST sodium cooling loops shall not be operated simultaneously, except when switching trains. When the loop is in operation, the other loop shall be kept on standby with its outlet valve closed.

44 d. Except when required for EVST cooling, the isolation valve in the lower EVST outlet line of loop 2 shall be locked closed.

46 44 e. Before an inerted cell containing one of the EVST sodium cooling loops is to be exposed to the RSB atmosphere, the enclosed sodium cooling loop shall be isolated from the EVST and, if the sodium radioactivity concentration exceeds TBD$_{\mu}$ Ci/CC, the loop will be drained.

59 20 f. Coolant levels in the EVST shall not be less than 31 in. above the upper edge of the lower axial blanket section of stored fuel assemblies.

If any of the above limiting conditions are not met, corrective action must be initiated to resolve the deficiency.

16.3.7.1.4 Bases

44 The first four specifications in Section 16.3.7.1.3 ensure equipment redundancy for cooling spent fuel so that a single failure or an initiating event following a single failure cannot cause overheating of fuel.

Specification e. is required on the basis that a potential sodium spill in the EVST sodium cooling loop cell might result in radioactivity release with a site boundary dose exceeding one tenth of the 10CFR100 limits.

Specification f. ensures that coolant levels are adequate to maintain the minimum safe level even in the event of a tank rupture and loss of sodium to the guard vessel.

16.3.7.2 Fuel Handling Heat Removal

16.3.7.2.1 Applicability

Applies to the limiting conditions for operation of the ex-vessel transfer machine (EVTM).

16.3.7.2.2 Objective

To ensure that no incident could occur during spent fuel handling operations that would adversely affect the public health and safety.

16.3.7.2.3 Specifications

20 The following conditions shall be continuously satisfied while the EVTM is transferring spent fuel or blanket assemblies.

a. Two independent power supplies shall be available for the EVTM.

b. The EVTM shall have two heat removal systems operable, each capable of removing 20 Kw of heat, before being used to handle spent fuel.

59 c. The EVTM shall not be used to handle fuel assemblies from the reactor until their calculated decay heat is less than 20 kw.

If any of the above limiting conditions are not met, corrective action must be initiated to resolve the deficiency. No spent fuel or blanket assemblies shall be handled by the EVTM before the above conditions are restored. However any fuel or blanket assembly in the EVTM may be trans-
20 ferred to the EVST or the reactor fuel transfer position, whichever is closer.

16.3.7.2.4 Bases

The first two specifications in Section 16.3.7.2.3 establish equipment redundancy for cooling spent fuel so that a single failure cannot cause overheating of fuel. The last specification ensures that no fuel assembly is handled with a decay heat exceeding the cooling capacity of the EVTM.

26 16.3.7.3 Direct Heat Removal Service (DHRS)

16.3.7.3.1 Applicability

Applies to the Auxiliary Liquid Metal Subsystem as related to the Direct Heat Removal Service function.

26

d. Operation of the plant may be permitted for up to seven days with one Station battery out of service provided the battery chargers and the other batteries remain operable with the battery charger, which is associated with the failed battery, carrying the DC load in its subsystem. However, if the loss also results in the loss of DC power for controlling the
37| Class 1E 4.16-KV and 480-V buses or for diesel generator field, the requirement of (c) above shall apply.

e. In the event two diesel generators are inoperable, a plant shutdown shall be initiated within two hours.

16.3.9.4 Basis

The electrical system is designed so that no single failure can impair the ability of the system to supply sufficient power to the Engineered Safety Features equipment required for plant safety under all conditions of operation or postulated accidents. The Engineered Safety Feature equipment is divided into redundant load groups, either of which is capable of safely shutting down the plant.

The offsite power system provides a reliable source of AC power to the plant. The system consists of the preferred AC power supply and the reserve AC power supply. The preferred AC power supply provides two connections to the TVA 161-kV grid. The reserve AC power supply provides two physically separate connections to the TVA 161-kV grid. All four of these grid sources are continuously energized and any one of them can supply the plant auxiliary distribution system to facilitate and maintain a safe plant shutdown.

37| Power for each of the Class 1E load groups is distributed by a 4.16-KV switchgear, 480-V load centers and 480-V motor control centers.

Two diesel generators are provided as standby power supplies for the
37| two 4.16-KV Class 1E buses. They are automatically started by a bus undervoltage condition as described in Section 8.3. Each generator is capable of supplying all the loads of one Class 1E load group. Both diesel generators have sufficient onsite fuel supply for seven days continuous full load operation. Sufficient maintenance and test procedures ensure that power for the Class 1E loads is always available during and after any design basis event.

Control power for each of the redundant Class 1E load groups and associated standby power supplies is fed from separate Class 1E DC power supplies.

One redundant Class 1E, DC power supply or standby AC power supply may be taken out of service for TBD hours to permit maintenance, repair and testing.

16.3.10 Refueling

16.3.10.1 Applicability

Applies to the limiting conditions for operation of the Reactor Refueling System (RRS) equipment and facilities, and to refueling operations.

16.3.10.2 Objective

To ensure that during refueling operations, core reactivity is within controlled limits and to ensure that the release of radioactivity from the containment or RSB in the event of a fuel handling accident is within the limits of 10CFR20 and 10CFR100.

16.3.10.3 Specification

16.3.10.3.1 The following conditions shall be continually satisfied while the Reactor Refueling System (RRS) equipment and facilities are operating.

44 a. The EVST and FHC gas activities shall be less than TBD μCi/cc, and TBD μCi/cc, respectively.

59 b. The railroad doors into the hardened portion of the RSB shall be closed and shall remain closed during the following conditions:

(1) When the EVTM is transferring irradiated core fuel assemblies;

44 (2) When irradiated core fuel assemblies are handled in the FHC or are being inserted into the spent fuel shipping cask.

If the above limiting conditions are not met, corrective action shall be taken to resolve the deficiency. No EVTM mating operation shall be initiated to the EVST or FHC if the respective gas activity is higher than specified.

16.3.10.3.2 The following conditions shall be met before initiating refueling operations involving the reactor.

a. The reactor shall be maintained in the Refueling Shutdown Condition as defined in Section 16.1.1.

b. The primary pump main circuit breakers shall be racked out and tagged.

c. During any movement of fuel within the core, a licensed operator shall be present in the Refueling Communication Center or the IVTM mezzanine.

59 d. All refueling system equipment required for the refueling operations shall be checked out and verified to be operational.

59 e. The primary and secondary control rod drive mechanisms shall be
51 disconnected from the control assemblies and the UIS raised and pinned. Prior to movement of the large rotating plug, a verification shall be made that all control rods are disconnected from their drive line assemblies.

59 f. The reactor cover gas activity shall be less than 2.2 μCi/cc.

44 59 g. The IVTM limit switch which precludes premature release of fuel and blanket assemblies shall be set less than 1.38 inches above the fully seated position as indicated in Figure 9.1-16B. 1 16

If any of the above specified limiting conditions are not met, the refueling shall not be initiated.

16.3.10.3.3 The following conditions shall be met during refueling operations involving the reactor.

59 a. Direct communications among personnel in the plant control room at the IVTM control console, and in the refueling communications center shall exist whenever changes in core geometry or fuel transfers are taking place.

b. All three source range flux monitor (SRFM) channels shall be operating with any fuel assemblies in the core. If any one of the channels fails, operations in progress to transfer fuel into
51 or out of the reactor core shall be stopped or reversed to place the reactor in a safe hold point configuration until the defective channel is restored to operation.

59 The Source Range Flux Monitoring System (SRFM) trip
51 points will be set at signal levels equivalent to a subcriticality of TBD for the first core and TBD for the equilibrium core. If the trip points are exceeded, the refueling operation must be stopped immediately and a determination made as to the cause of the reactivity anomaly.

c. During refueling operations, not more than two vacant positions in the core may exist at any one time. These vacant positions may not be adjacent to each other.

If any of the specified limiting conditions for refueling are not met, refueling shall cease until the specified limits are met, and no operations will be initiated which may increase the reactivity of the core beyond the reactivity resulting from normal temperature fluctuations within the refueling temperature dead band.

16.3.10.3.4 Following refueling operations involving the reactor, the following conditions shall be met prior to reactor startup.

a. The reactor rotating plugs shall be secured and their drive power sources physically disconnected.

b. The refueling hatch between the RSB and the RCB shall be closed and leak tested.

16.3.10.3.5 The following conditic s shall be met before initiating fuel handling or shipping operations in the F .

Both FHC cooling grapple blowers, both argon cooling system trains, the dynamic seals, and the FHC radiation monitors, shall be checked and verified to be operational.

If the above specified limiting condition is not met, FHC fuel handling or shipping operations shall not be initiated.

16.3.10.3.6 The following leak rate tests shall be performed at periodic intervals.

59 a. The EVTM shall be leak rate tested at 11 psig. The leak rate shall not exceed 1 vol. % per day.

59 b. The FHC shall be leak rate tested at -3 inches water gauge. The leak rate shall not exceed 0.14 vol. % per day.

If the above limiting conditions are not met, correction action shall be taken to resolve the deficiency. No EVTM or FHC operations involving irradiated core assemblies shall be initiated if the respective leak rates are higher
44 than specified.

16.3.10.4 Bases

The respective limits in Section 16.3.10.3.1 are established on the basis that if either amount of activity was all released instantaneously into the RSB operating area, the radiation dose at the site boundary would be less than the limits of 10CFR20 (Annual).

Immediately prior to refueling, Section 16.3.10.3.2 lists the conditions which must be satisfied. Item a is based on permissible core shutdown levels. Item b is written to prevent the operation of the primary pumps during refueling and Item c is intended to assure that proper
59 supervision will exist during movement of fuel within the core. Items d and e are written to prevent unexpected movement of core components during refueling which could affect core reactivity. Item f is intended to control the release
59 44 of radioactivity to the atmosphere. The level specified in Item f is based on the premise that if this amount of activity was all released instantaneously into the RCB operating area, the radiation dose at the site boundary would not exceed the limits of 10CFR20 (Annual) and the airborne radiation dose in the
44 RCB would be below the quarterly 10CFR20 limits for restricted areas. Item g
is intended to prevent dropping of a core assembly or insertion of a core 1
59 assembly into an incorrect position.

Amend. 59
Dec. 1980

59 The specifications of Section 16.3.10.3.3 during refueling establish
control of the operation. During any subcritical operation other than the
intentional approach to critical, the SRFM must provide a warning to the
operator and thereby assure that the reactor does not approach criticality
any closer than that level from which criticality could be attained by the
worst single refueling error with adequate margin for the associated un-
certainties. The minimum shutdown reactivity requirement during refueling
5' is based on this criterion. An alarm will sound in the control room if the
minimum shutdown requirement, described above, is violated.

Shuffling of blanket assemblies cannot be done without temporarily
leaving open two core positions. If two adjacent core assemblies are removed,
the resulting misalignment could exceed the design value, so that a new core
assembly or an assembly to be reinserted could either not be inserted or be in-
serted in the wrong position. Item c of Section 16.3.10.3.3 is written to prevent
this event. Note, however, that shuffling is not part of the current fuel
management scheme, but is only a capability provided for any future fuel
59 management scheme.

59 The specifications in Section 16.3.10.3.4 are written to assure that modi-
fications made to accommodate the refueling are corrected before reactor startup.

The specification of Section 16.3.10.3.5 is mainly intended to ensure spent fuel cooling capability of the FHC to prevent potential fission gas activity release resulting from overheating of fuel pins. In addition, proper performance of inflatable and dynamic seals will be checked as a further backup of 16.3.10.3.6 b for maintaining a low leakage cell. Operational checkout of FHC radiation monitors is required to ensure that the limits of 16.3.10.1.a will not be exceeded.

The specifications of Section 16.3.10.3.6 are intended to control the release of radioactivity to the atmosphere as a consequence of the respective design basis accidents.

The maximum leakages specified in Section 16.3.10.3.6 Items a and b
are determined by the activities of the highest power fuel assemblies handled
by the EVTM and in the FHC which, if released to the RSB operating area and
subsequently to the site boundary, at the specified leak rate, would be less
47 44 than the limits of 10CFR20 (Annual).

Amend. 59
Dec. 1980

16.3.11 Effluent Release

16.3.11.1 Liquid Waste

16.3.11.1.1 Applicability

Applies to the liquid radioactive effluents from the radioactive waste system to the environment.

16.3.11.1.2 Objective

To assure that liquid radioactive material released to the environment is kept as low as practicable and, in any event, is within the limits of 10CFR20.

16.3.11.1.3 Specification

1. If the experienced release of radioactive materials in the liquid wastes, within a calendar quarter period, is such that these quantities, if continued for a year, would exceed twice the design objectives, the following actions will be taken:

   a) An investigation shall be made to identify the causes for such releases.

   b) A program shall be defined and initiated to reduce such releases to within the design values.

2. The release rate of radioactive materials in liquid waste from the plant shall be controlled, by in-line monitoring, such that the concentration in the cooling tower blowdown will not exceed the concentrations specified in 10CFR20.106.

3. All radioactivity liquid effluents released from the plant shall be reported in accordance with 16.6.7.B.

16.3.11.1.4 Basis

Liquid effluent release rate will be controlled in terms of the concentration in the discharge tunnel containing cooling tower blowdown. This basis assures that even if a person obtained all of his daily water from such a source, the resultant dose would not exceed that specified in 10CFR20. Since no such use of the discharge tunnel is made and considerable natural dilution occurs prior to any location where such water usage could occur, this assures that offsite doses from this source will be far less than the limits specified in 10CFR20.

In addition to the sampling and analysis of each batch prior to discharge, a radiation monitor on the radioactive waste discharge line and a sampler in the discharge tunnel give further assurance that annual average discharge concentration is kept within the specified limits.

16.3.11.2 Gaseous Waste

16.3.11.2.1 Applicability

Applies to the release of radioactive gaseous effluents from design release points.

16.3.11.2.2 Objective

59 To assure that the amount of radioactivity released as low as is reasonably achievable and will result in site boundary doses which are below 10CFR50, Appendix I limits.

16.3.11.2.3 Specification

59 1. Radioactive gases released from design release points shall be continuously monitored and/or sampled such that the total release can be quantified.

59 2. The effluent monitor for CAPS shall be operable and capable of alarming when radioactivity is detected at a maximum pre-set concentration of TBD μCi/cc.

3. The effluent monitor for undefined mixtures from the exhaust of radwaste area of the Reactor Service Building shall be operable and capable of alarming when radioactivity is detected at a level corresponding to (TBD) percent of the maximum permissible radionuclide concentrations given in 10CFR20.

4. The effluent monitor for undefined mixtures from the reactor service area (RSB) exhaust shall be operable and capable of alarming when radioactivity is detected at a level corresponding to (TBD) percent of the maximum permissible radionuclide concentrations given in 10CFR20 for unrestricted areas.

5. The effluent monitor for undefined mixtures from the Intermediate Bay exhuast shall be operable and capable of alarming when radioactivity is detected at a level corresponding to (TBD) percent of the maximum permissible radionuclide concentrations given in 10CFR20 for unrestricted areas.

6. The effluent monitor for undefined mixtures from the Turbine Generator Building exhaust shall be operable and capable of alarming when tritium activity is detected at a level corresponding to (TBD) percent of the maximum permissible concentration given in 10CFR20 for unrestricted areas.

7. In the event of an alarm due to high radioactivity in the effluent of a design discharge point, appropriate action will be taken as defined (to be supplied in FSAR).

8. If an effluent monitor is inoperable, appropriate action will be initiated and be in effect until the monitor is restored to operational status (action to be defined in FSAR).

9. If the quantities of radioactive material released during any semi-annual period are significantly above design objectives, the CRBRP shall:

1) Make an investigation to identify the causes of such releases.
59 2) Define and initiate a program of corrective action.

16.3.11.2.4 Basis

Dose rate estimates have been made for the CRBRP design release points for off-normal occurences. Based on these calculations, release of activity at the alarm limits will result in an off-site annual dose rate which will not exceed (TBD) mr/yr, well below 10CFR20 limits. Estimates of the activity inventory assume failed fuel conditions described in Section 11.3.

16.3.11.3 HVAC and Radioactive Effluents

16.3.11.3.1 Applicability

Applies to the release of radioactive effluents through the HVAC exhausts.

16.3.11.3.2 Objective

To assure that radioactivity released to the environment is kept as low as practicable and, in any event, is within the limits of 10CFR20 guidelines.

To assure that the release of radioactivity to unrestricted areas meet the "as low as practicable" concept, the following design objective applies:

a) The release rate of radioactive isotopes, averaged over a yearly interval except for halogens and particulate radioisotopes with half-lifes greater than 8 days, discharged from the plant, should not exceed:

$$\sum_i \frac{Q_i}{(MPC)_i} \leq 800 \text{ m}^3/\text{sec}$$

where Q_i is the annual average release rate (Ci/sec) of radioisotope i and $(MPC)_i$ in Ci/cc is defined for isotope i in column 1, Table II of Appendix B to 10CFR20.

16.3.11.3.3 Specification

1) The instantaneous release rate of radioactive isotopes, discharged from the plant, shall not exceed:

$$\sum_i \frac{Q_i}{(MPC)_i} \leq 40{,}000 \text{ m}^3/\text{sec}$$

where Q_i and $(MPC)_i$ are as defined above.

2a) The gaseous and particulate activity of the potentially contaminated HVAC discharge paths shall be monitored and recorded along with the corresponding effluent flow rates.

2b) Radiation monitors as required in 16.3.11.3.3-2a above shall be operable and capable of detecting a composite radioactivity release rate less than the design objective rate.

2c) Whenever any of the radiation monitors are inoperable, grab samples shall be taken in the affected discharge path and analyzed.

3) When the annual projected release rate of radioactivity, averaged over a calendar quarter, exceeds the annual objective, corrective action shall be taken to reduce such release rates to below the objective rate and/or orderly shutdown of the reactor shall be initiated.

4) When the instantaneous release rate or radioactivity exceeds twice the design objective rate, the licensee shall identify the cause of such release rates, initiate action to reduce such release rates to below the objective rate.

16.3.11.3.4 Basis

The specifications provide reasonable assurance that the resulting annual exposure rate from noble gases at any location at the site boundary will not exceed 10 millirems per year. At the same time, these specifications permit the flexibility of operation, under unusual operating conditions, which may temporarily result in releases higher than the design levels but well below the concentration limits of 10CFR20.

The release rate stated in the objective sets the concentration of radioisotopes, except for halogens and particulate radioisotopes with half-lives greater than eight days, at less than 2% of 10CFR, Part 20 requirements at the site boundary (<10 mrem per year).

Specification (1) requires the licensee to limit the release of all radioisotopes such that concentrations at the site boundary are less than the levels specified in 10CFR20.

Specification (2) requires that suitable equipment to monitor radioactive releases are operating during any period these releases are taking place.

Specification (3) establishes an upper limit for the quarterly average release rate for noble gases equal to the annual design rate. The intent of this specification is to permit the licensee the flexibility of operation under unusual operating conditions which may result in short-term release higher than the annual objective rate.

12. The HTS shall be designed such that decay heat removal can be effected by utilizing the normal heat removal train. This capability must be assured for both three and two loop operation for all upset, emergency and faulted events. For these events sufficient coolant flow shall be provided to ensure that corresponding fuel design limits defined in Chapter 4 are not exceeded. The relative elevations of the reactor core, IHX tube bundle and the steam generator modules are arranged to promote natural circulation of sodium in the PHTS and IHTS loops in the event of loss of all electrical power to the pumps.

16.5.5 Fuel Storage

16.5.5.1 Applicability

Applies to the storage of new and spent fuel assemblies.

16.5.5.2 Objective

To define those system features which are essential in providing for safe fuel storage.

16.5.5.3 Specification

The fuel storage facilities consist of new fuel storage and spent fuel storage.

A. New Fuel Storage

New fuel is stored in the RSB in the EVST (see Item B). In addition to the EVST, new fuel is temporarily retained in shipping containers after a truck with the Safe Secure Trailer arrives in the hardened part of the RSB and in two new fuel unloading stations below the RSB operating floor. Each fuel unloading station consists of a pit which can contain one shipping container with a single new fuel assembly. New fuel assemblies are unloaded from
44 20 the shipping containers in the two unloading stations using the EVTM.

New fuel is also stored under sodium in the EVST, described below.

B. Spent Fuel Storage

Spent fuel is stored in the RSB in two locations: in the EVST and, on a temporary basis, in the spent fuel transfer station of the fuel handling
44 cell (FHC).

The EVST is a single vessel, sodium-filled storage facility with a two-tier rotatable turntable. It is located between the EVTM gantry rails in
20 the RSB. It can store approximately 650 new and/or spent fuel assemblies, each in a core component pot (CCP). The primary vessel is surrounded by a

guard tank as a safety measure against any sodium leaks. The guard tank is
situated in a nitrogen gas-filled concrete vault. The space between primary
vessel and guard tank is sized to maintain a minimum safe sodium level above
59 20 the fuel assemblies (i.e., 31 inches above the upper edge of the lower axial blanket)
in the extremely unlikely event of a gross primary vessel failure. The primary ves-
sel is supported from its upper flange, suspended into the guard tank. The
turntable is supported through a bearing and seal configuration above the
20 primary vessel flange. The guard tank is bottom supported from the vault floor.
The fuel assembly storage positions are cylindrical tubes, arranged in con-
centric rows, restrained and supported by a stainless steel gridwork in the
rotatable storage rack. Each storage tube holds two CCP's one above the other.
Sodium coolant flow enters each tube at the bottom and leaves at the top, as
well as circulating around the outside of the tube. Heat is removed by two
independent, redundant sodium cooling loops. The primary vessel is sealed
and shielded from the RSB operating floor by a heavy closure head. The closure
44 20 head and a striker plate above it also prevent internal EVST damage from ex-
ternal drop loads.

The FHC spent fuel transfer station is located directly below the
FHC fuel transfer port. It is a temporary storage facility cooled by natural
argon convection with a rotatable basket holding up to 3 fuel assemblies in
20 core component pots in a triangular array. The transfer station is supported
at its upper flange. The rotatable basket consists of a stainless steel web
structure and cylindrical sockets for support of the CCP's. Each storage
location holds one fuel assembly. The method of heat removal is by natural
convection to the FHC argon atmosphere. Sealing and shielding at the RSB
operating floor is provided by the heavy FHC steel roof plug structure. It
44 also provides protection of the FHC interior against external drop loads.

The safety features provided in the EVST and FHC spent fuel trans-
44 fer station design include the following:

59 a. Physical separation of fuel assemblies with structural support to prevent changes in separation distance or displacement due to combined normal and SSE or other abnormal loads.

b. A heavy roof structure and steel-lined concrete vault walls
20 protect the RSB operating floor, FHC operating gallery and neighboring cells from radiation.

c. Double seals around the fuel transfer port plugs, FHC viewing windows and manipulator penetrations, between the EVST primary vessel and head, and between EVST cover plate and vault lining prevent radioactivity release from the EVST and FHC.

d. The location of sodium inlet and outlet pipes, provisions of antisyphon devices, and the presence of a guard vessel prevent any loss of sodium coolant from the EVST that could prevent
44 cooling of spent fuel.

APPENDIX B - GENERAL PLANT TRANSIENT DATA

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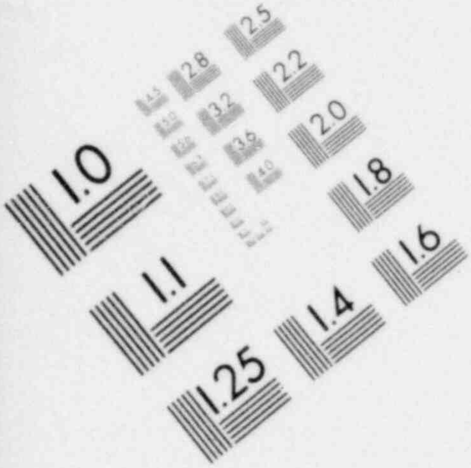

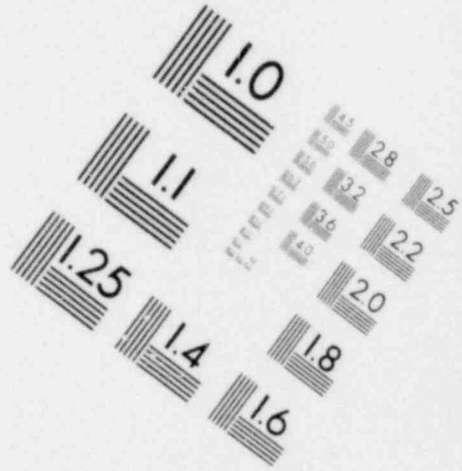

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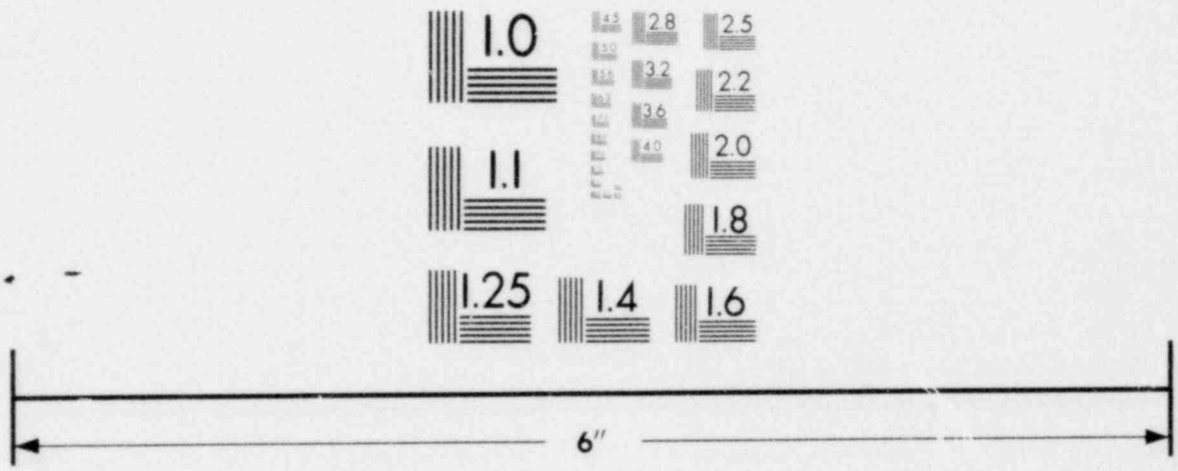

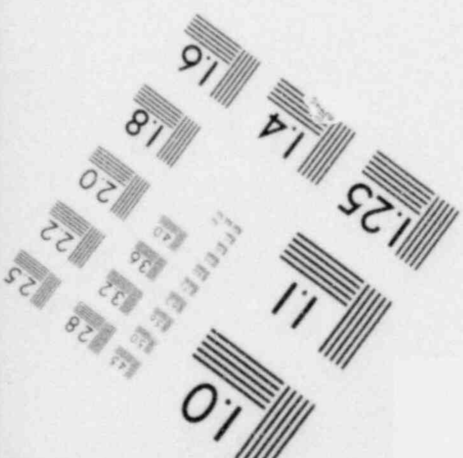

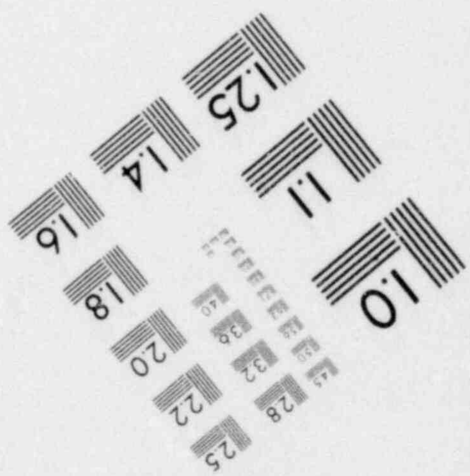

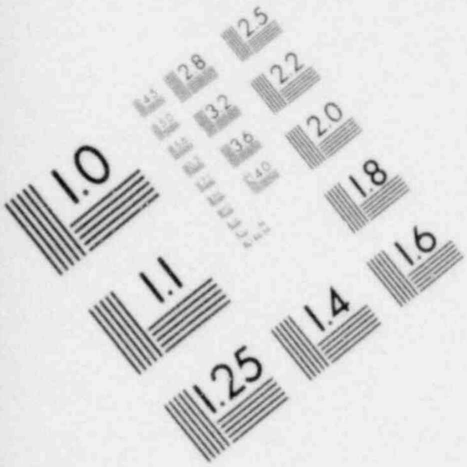

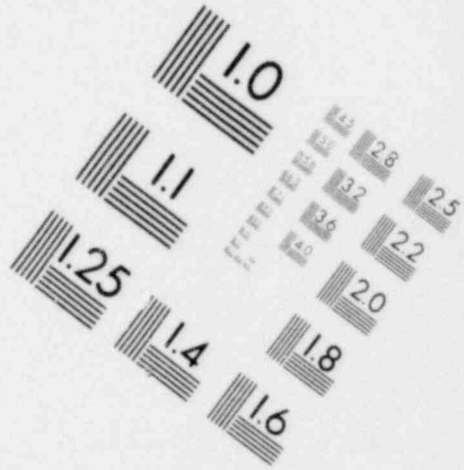

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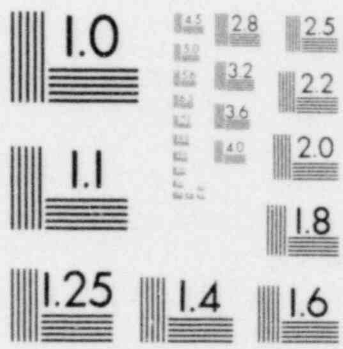

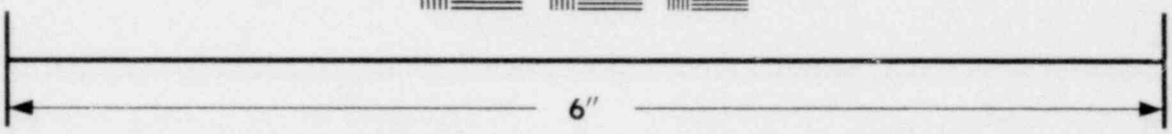

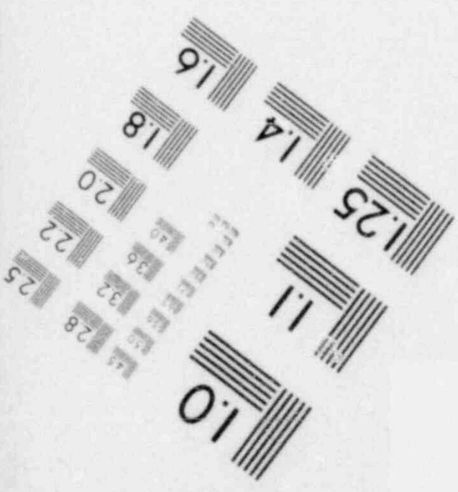

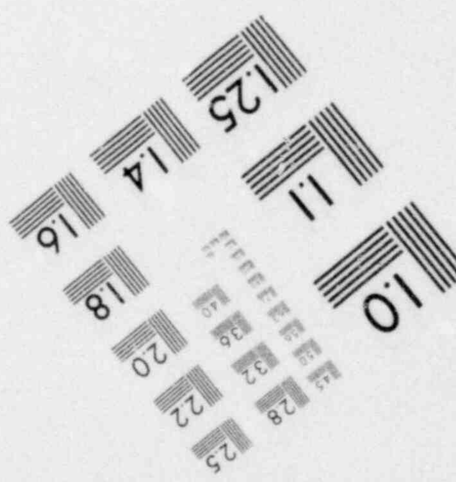

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APPENDIX B GENERAL PLANT TRANSIENT DATA

B.1 CRBRP PLANT DESIGN DUTY CYCLE

This Appendix is a compilation of the events which comprise the CRBRP design duty cycle, together with an explanation of the method of selection of 'umbrella' transients. It should be noted that the inclusion of an item in this list signifies that it has been utilized for design purposes, but not that the event itself is necessarily regarded either as being credible, or to be expected as frequently as is indicated. Table B-1 presents the duty cycle frequency list. Table B-2 presents a preliminary listing of 'umbrella' events to be used as a basis for structural evaluation of the major heat transport system components.

B.1.1 Normal Events

B.1.1.1 N-1 Dry System Heatup and Cooldown, Sodium Fill and Drain

For design purposes, the heatup of the entire sodium system, exclusive of the steam generators, or of individual primary or intermediate loops will be treated as a temperature increase of the outer surface of the sodium containment from ambient (70°F) to 450°F at a constant rate of nominally
58 3°F/hr (desired rate is 25°F/hr), (10°F/hr for the reactor vessel). After a soak at 450°F surface temperature to preheat the internals to a nominal 400°F, the surface will be allowed to cool to 400°F. Similarly, cooldown will be considered as a decrease from 400°F to 70°F at a constant rate of 25°F/hr. (10°F/hr. for the reactor vessel). Plant systems will be filled with argon at one atmosphere. Each heatup cycle will be preceded by three cycles of pressure reduction to as close to full vacuum as practical and back filling with argon to one atmosphere. Each heat up cycle will be followed by one pressure cycle from ambient to maximum attainable vacuum with back filling to one atmosphere using argon. It is assumed that all sodium containing piping and components will be heated by electrical heaters mounted external to the piping, component, or guard vessel, as applicable. The steam generator modules and the steam-water system will be heated from the water side using an auxiliary heat source. Following the heatup, the primary and intermediate systems are filled with 400°F sodium. The systems are drained and backfilled with argon at one atmosphere prior to cooldown below 400°F.

B.1.1.2 N-2 Normal Startup

B.1.1.2.1 N-2a Startup from Refueling

59 The plant startup event from refueling is a heatup transient between the normal refueling temperature of 400°F and the temperature conditions that exist at a minimum operating power level of 40% thermal power. For design purposes, the primary system sodium temperature will increase essentially isothermally at an average rate of 50°F per hour between 400°F and 600°F. This heatup rate will be achieved by utilizing the sodium

47 pumps at 100% flow and may include a minimal amount of reactor power. Between $600^{\circ}F$ and the system conditions existing at 40% thermal power, the hot leg temperature will change at an average rate of $150^{\circ}F$ per hour. This change in temperature will be accomplished by making discrete steps in power level which will result in temperature changes at a rate of $1^{\circ}F$ per second for 25 seconds every 10 minutes. The primary cold leg and intermediate system hot and cold leg temperatures will vary between $400^{\circ}F$ and their appropriate temperatures for the 40% thermal power level. Primary sodium flow rate during this portion of the heatup is taken to be constant at 40% of full flow with appropriate intermediate flowrates. The water/steam side of the steam generators will follow the nuclear island temperatures from $400^{\circ}F$ to the operating temperature level of $600^{\circ}F$ with water/steam circulation through the evaporator module and steam through the superheater. The water/steam pressure will be varied from about 425 psig to ~1500 psig as required to meet the operation conditions at 40% thermal load. Steam flow will vary as required to heat the turbine and reject the excess heat generated by the reactor.

B.1.1.2.2 N-2b Startup from Hot Standby

59 The plant startup event from hot standby is a heatup transient between the hot standby temperature of $600^{\circ}F$ and the temperature conditions which exist at a minimum operating power level of 40% thermal
47 power. This event is the same as the second part of N-2a (starting at $600^{\circ}F$).

B.1.1.3 N-3 Normal Shutdown

B.1.1.3.1 N-3a Shutdown to Refueling

59 The plant shutdown event to refueling is a cooldown transient between the temperature conditions which exist at a minimum operating power level of 40% thermal power to the normal refueling temperature of $400^{\circ}F$. This event is assumed to be essentially N-2a reversed in time with the exception that the reactor will be taken subcritical when the primary hot leg temperature has been reduced to slightly above $600^{\circ}F$ and the sodium pumps will be run at pony motor speed during the cooldown between $600^{\circ}F$ and $400^{\circ}F$. The cooldown between $600^{\circ}F$ and $400^{\circ}F$ is handled by the Protected Air Cooled Condensers (PACC) with assistance from the main feedwater and turbine bypass system to accelerate the cooldown. When refueling conditions are reached the PACC's handle the entire heat load and the main steam stop and feedwater isolation valves are shut. The main feedwater system is used intermittently after this to provide makeup as required for long term
47 leakage.

B.1.1.3.2 N-3b Shutdown to Hot Standby

59 The plant shutdown event to hot standby is a cooldown transient between the temperature conditions which exist at a minimum operating power level of 40% thermal power to the hot standby temperature of $600^{\circ}F$. This event is assumed to be N-2B reversed in time, with the reactor taken subcritical when the primary hot leg temperature has been reduced to slightly above $600^{\circ}F$. When the protected air cooled condenser can handle the decay
47 heat load, the main steam stop and feedwater isolation valves are shut.

B.1.1.4 N-4 Load Following

B.1.1.4.1 N-4a Loading and Unloading

The plant design loading and unloading events are conservatively
58 represented by a continuous and uniform ramp load change through the range of
47 40% to 100% of full load. This load range is the maximum permissible consistent with the reactor control system, which is designed to accommodate automatic load following capability while maintaining rated steam conditions.
47 Rate of load change is up to 3.0% per minute. Load changes in this region are accomplished by linearly varying primary and intermediate sodium flows with power while holding turbine inlet pressure constant.

B.1.1.4.2 N-4b Load Fluctuations

In addition to normal plant loading and unloading (N-4a), there will be load fluctuations resulting from changing electrical network demands. For design purposes, these events are conservatively represented by continuous
47 and uniform ramp load changes through the range of 80% to 100% of full load. This load swing is used since it results in the largest temperature variation in the system for the given 20% load variation. Rate of load change is up to 3.0% per minute. As in loading and unloading, load changes are accomplished
47 by linearly varying primary and intermediate sodium flows with power while holding turbine inlet pressure constant. For calculation of the system conditions during a load fluctuation, it should be assumed that equilibrium conditions are reached between the ramp power changes.

B.1.1.5 N-5 Step Load Changes of ±10% of Full Load

This event involves step changes in generator load equivalent to $\pm$10% of full generator load within the load range of 40% to 100% of full
47 load. These events are assumed to be occasioned by normal disturbances in the electrical network into which the plant output is tied. The nuclear island is assumed to respond to the load change by changing flow and power at the rate of 3% of rated power per minute.

B.1.1.6 N-6 Steady State Temperature Fluctuations

This event consists of the sodium temperature variations produced by power and flow fluctuations within the control system deadband. This fluctuation is taken to be $\pm 6^{\circ}$F peak to peak for 30 x 10^6 cycles and is based on expected deadband fluctuations. The fluctuations may arise from the deadband of the power loop of the control system ($\pm$2%) which would result in a temperature fluctuation period of about 24 seconds at full flow. Since
58 the system is not expected to limit cycle, the frequency is considered to be conservative.

B.1.1.7 N-7 Steady State Flow Induced Vibrations

This event consists of the vibrations in the sodium heat transport system produced by the fluctuations in sodium pressure due to the interaction between the vanes in the impellers and the pump volute. Steam generator sys-
58 tem water side fluctuations caused by other system vibrations and frequency
47 are determined on a system basis.

B.1.2 Upset Events

58 59 All upset events are terminated at hot shutdown unless otherwise specified.

B.1.2.1 U-1 Reactor Trip

This transient includes real scrams due to malfunctions (including rapid reactivity transients) which cause a PPS trip level to be exceeded and spurious scrams covering those situations in which a PPS trip level is not actually exceeded; but a scram occurs due to a fault in the PPS, control system or plant instrumentation.

B.1.2.1.1 U-1a Reactor Trip From Full Power with Normal Decay Heat

47 This transient involves a trip of the reactor (release of primary and secondary control rods) followed in 300 msec by the tripping of the main sodium pumps. The sodium pumps coast to pony motor speed, which results in a
58 47 sodium flow of about 10% of full flow in both the primary and intermediate loops. Feedwater and recirculation pumps remain energized and feedwater flow is controlled by the normal control system that responds to drum level and feedwater/steam flow signals. The turbine is tripped on a low throttle pressure signal.
58 47 The turbine bypass valve starts opening coincident with the turbine trip. The initial decay heat level for this transient is normal 100% decay heat. The Protected Air Cooled Condenser for each loop is turned on within 240
47 seconds of the reactor trip.

B.1.2.1.2 U-1b Reactor Trip from Full Power with Minimum Decay Heat

The same operational sequence of event U-1a is assumed for this transient. The minimum decay heat level is assumed for an initial condition.

B.1.2.1.3 U-1c Reactor Trip from Partial Power with Minimum Decay Heat

The same operational sequence is assumed for this transient as for reactor trip from full power. Initial power level for this transient is
58 40% and the initial decay heat is minimum decay heat. Since larger initial thermal differences can occur in the steam system at part power than full power, this transient is included as a separate case.

B.1.2.2 U-2 Uncontrolled Control Rod Movement

This is a general category of events, which result from control system malfunctions. The U-2 category includes six events: an uncontrolled rod insertion from full ΔT initial conditions, an excessive startup step power change, and four rod withdrawal cases. These events are identified in the duty cycle for the purpose of providing assurance of their consideration in the overall transient analysis task and as a basis for the determination of plant protection system requirements.

B.1.2.2.1 U-2a Uncontrolled Rod Insertion

A single rod is inserted at a rate which causes a 1/2%/second reduction in thermal power due to an assumed malfunction of the controller on that rod. (This event is not to be confused with a rod drop, which is an unlatching of the rod resulting in a free fall of the control rod.) The sodium, feedwater, steam and recirculation flows are held constant, as is the turbine admission valve inlet pressure. It is assumed that this event occurs when full system ΔT's are present. The power level at the beginning of the transient is 100%. An operator manually trips the plant four minutes after event initiation.

B.1.2.2.2 U-2b Uncontrolled Rod Withdrawal From 100% Power

An uncontrolled withdrawal of one control rod is assumed to cause
47 the reactor power to increase at 0.5% nuclear power per second from 100% to
115% (just below the high flux trip point). A manual reactor trip is assumed
after 5 minutes. Sodium flows are maintained at initial values until the
trip occurs and the feedwater and steam flows are increased appropriately for
the increase in reactor power. For the core, it provides the worst case
47 sustained overpower design event.

B.1.2.2.3 U-2c Uncontrolled Rod Withdrawal from Startup with Automatic Trip

The initial conditions for this event are hot standby with minimum
decay heat. Primary and intermediate main pump motors are started and sodium
flows are increased to 40 percent. Uncontrolled withdrawal of one control
47 rod at 0.5% nuclear power/second then occurs. During the withdrawal, all
sodium flows remain at initial values. A reactor trip is initiated by a flux-delayed flux subsystem.

B.1.2.2.4 U-2d Uncontrolled Rod Withdrawal from Startup to Trip Point With Delayed Manual Trip

The initial conditions for this event are hot standby with minimum
decay heat. Primary and intermediate main pump motors are started and sodium
flows are increased to 40 percent. Uncontrolled withdrawal of one control
47 rod at 0.5% nuclear power/second then occurs. The power ramp is terminated
just before the flux-delayed flux trip point is reached. After 10 minutes,
the event is terminated by a manual scram. Flows are maintained constant
at initial value.

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B.1.2.2.5 U-2e Plant Loading at Maximum Rod Withdrawal Rate

From initial plant conditions of 40% reactor thermal power, 40% sodium flow, and ~35% electrical output, the station supervisory control causes the plant to load at the nominal rod withdrawal speed. Sodium flows and reactor power increase from 40% to 100% at 1%/second. The turbine increases output at the same rate. The drum level control operates normally to approximately match feedwater flow to steam flow. The feedwater pump control increases feedwater pump speed at 1% per second. No scram occurs.

B.1.2.2.6 U-2f Reactor Startup with an Excessive Step Power Change

The reactor startup with an excessive step power change is a normal startup as defined in event N-2a. In addition it is assumed that during the course of the startup, there is a power change resulting in temperature changes at a rate of 1°F/second for 50 seconds followed by a constant outlet temperature for 1150 seconds.

B.1.2.3 U-3 Complete or Partial Loss of One Primary Pump

There are two events in this category: partial loss of prim[illegible] flow in one loop and the loss of power to one primary pump main motor.

B.1.2.3.1 U-3a Partial Loss of Primary Pump

The primary flow in one loop is assumed to decrease from 100% to
47 a level immediately above the flow ratio trip setting (at approximately 70% flow) due to a ramp down in pump speed at minimum coastdown rate. The primary sodium flows in the two unaffected loops as well as the intermediate flows in all three loops remain at their initial values. No action is taken to terminate the event for 10 minutes. A manual trip terminates the event at that point. This transient provides an envelope to encompass control malfunction and operator errors causing mismatches in the primary to intermediate flow ratio at design values. The transient will result in an increased reactor outlet temperature and a redistribution of temperatures within the IHX in the affected loop.

B.1.2.3.2 U-3b Loss of Power to One Primary Pump

The primary pump in one loop is assumed to coast down to pony motor speed. The other primary pumps are assumed to be under speed control. The intermediate pump speeds in all loops remain at initial values until reactor/pump trip. A reactor trip is initiated when the ratio of normalized primary to intermediate pump speed is less than 0.7. Following
47 the reactor trip, the remainder of the pumps and the steam/water side are treated as for the normal scram.

47

B.1.2.21.2 U-21b Inadvertent Opening of Superheater Outlet Safety/Power Relief Valves

For initial conditions of 390 MWt and a minimum decay heat, a full steam relief at a superheater exit is assumed. Loop steam flow will increase by 60% of full flow and turbine steam flow will decrease by 13%. The affected loop drum will initially decrease but feedwater flow will increase to maintain normal drum level. A reactor trip is assumed to occur from feed-steam flow mismatch and the transient is terminated by an operator shutting the superheater inlet valve 5 minutes after the trip.

B.1.2.22 U-22 Inadvertent Opening of SGAHRS Steam Drum Vent Valve

58 Inadvertent opening of a SGAHRS vent valve at the steam drum is assumed. The plant will trip on low steam/feedwater flow ratio, and the drum will depressurize. Depressurization may cause cavitation at the recirculation pump suction. The event is characterized by a decrease in recirculating water temperature and evaporator Na outlet temperature, and is terminated by operator action in isolating the SGAHRS steam drum vent line flow ten minutes after the plant trip.

47

B.1.2.23 U-23 Inadvertent Opening of Evaporator Inlet Dump Valve

Both series valves are assumed to open instantaneously and this is accompanied by automatic closing of the inlet isolation valve of the affected evaporator. Pressure from the steam drum closes the check valve in the evaporator outlet line. This valve serves to isolate the evaporator at the outlet line and a oids blowing down the steam drum. The plant is tripped on high evaporator outlet temperature.

B.1.2.24 U-24 Reactor Trip with Failure of One PACC to Perform

The same operational sequence of event U-1a is assumed except the PACC does not start in one loop due to an assumed control logic failure.

47

B.1.3 Emergency Events

All emergency events, which result in a reactor trip, shall be considered to result in a transient followed by a cooldown to refueling.

59

47

B.1.3.1 E-1 Primary Pump Mechanical Failure

The event involves an instantaneous stoppage of the impeller of one primary pump while the system is operating at 100% power. The failure may be a seizure or breakage of the shaft or impeller. Primary system sodium flow in the affected loop decreases rapidly to zero as the pumps in the unaffected loops seat the check valve (thereby causing a check valve slam). A reactor trip will be initiated by the primary-intermediate flow ratio subsystem. Sodium flow in the intermediate circuit of the affected loop decays as in a reactor trip from full power (U-1b), modified by changes in natural circulation head. The event is characterized by a down transient in the hot leg of the intermediate circuit of the affected loop and by a check valve slam in the primary circuit of the affected loop.

B.1.3.2 E-2 Intermediate Pump Mechanical Failure

The impeller of one of the intermediate system pumps is assumed to stop instantaneously causing the flow in that circuit to decrease rapidly. A reactor trip is initiated by the primary to intermediate sodium flow ratio subsystem and the normal trip transient sequence is followed thereafter. The event is characterized by an up transient in the primary cold leg of the affected loop and by down transients in the steam generator modules of the affected loop since intermediate flow is limited to that produced by natural circulation.

43

TABLE B-1

PRELIMINARY DESIGN DUTY CYCLE EVENT FREQUENCIES

	Event		Frequency
	1. Normal Events		
6 58	N-1	Dry system heatup and cooldown, sodium fill and drain	5 total system + 8 per loop + 17 additional for entire intermediate loop exclusive of IHX
	N-2a	Startup from refueling	140
45 58 47	N-2b	Startup from hot standby	700
	N-3a	Shutdown to refueling	60
59	N-3b	Shutdown to hot standby	210
	N-4a	Loading and unloading	9300 (each)
	N-4b	Load fluctuations	46500 (each, up and down)
	N-5	Step load changes of $\pm$ 10% of full load	750 (each)
	N-6	Steady state temperature fluctuations	30×10^6
47	N-7	Steady state flow induced vibrations	10^{10} (sodium)
	2. Upset Events		
47	U-1a	Reactor trip from full power with normal decay heat	180 (1)
	U-1b	Reactor trip from full power with minimum decay heat	0
	U-1c	Reactor trip from partial power with minimum decay heat	0
	U-2a	Uncontrolled rod insertion	10
	U-2b	Uncontrolled rod withdrawal from 100% power	10

47 (1) - The total frequency for U-1 is associated with normal decay heat so as to balance the trips associated with partial decay heat for events U-2 through U-23.

TABLE B-1 (Continued)

	Event		Frequency
	U-2c	Uncontrolled rod withdrawal from startup with automatic trip	17
47	U-2d	Uncontrolled rod withdrawal from startup to trip point with delayed manual trip	3
	U-2e	Plant loading at max. rod withdrawal rate	10
48 47	U-2f	Reactor startup with excessive step power change	50$^{(2)}$
	U-3a	Partial loss of primary pump	2 per loop
	U-3b	Loss of power to one primary pump	5 per loop
	U-4a	Partial loss of one intermediate pump	2 per loop
	U-4b	Loss of power to one intermediate pump	5 per loop
	U-5a	Loss of AC power to one feedwater pump motor	10
	U-5b	Loss of feedwater flow to all steam generators	5
48			
	U-7a	Primary pump speed increase	5
	U-7b	Intermediate pump speed increase	5
	U-8	Primary pump pony motor failure	5 per pump
	U-9	Intermediate pump pony motor failure	5 per pump
	U-10a	Evaporator module inlet isolation valve closure	4 per loop
	U-10b	Superheater module inlet isolation valve closure	2 per loop
	U-10d	Superheater module outlet isolation valve closure	2 per loop

47 (2) - These events are part of the startups specified for event N-2b and should not be added as separate startups.

AMENDMENT 59

LIST OF RESPONSES TO NRC QUESTIONS

There are no new NRC Questions in Amendment 59.

Question 001.245 (15.7.1.2.1)

Identify all safety related valves or instruments which require a compressed air supply.

Response:

The following tabulation of safety related valves and their fail positions summarizes all valves so designated.

1. Primary Sodium Removal and Decontamination System

SAFETY-RELATED VALVES OR INSTRUMENTS	FAIL POSITION
HV 001 A*	Closed
044 A*	Closed
044 B*	Closed
085 A*	Closed
085 B*	Closed
086 B*	Closed

* Isolation Valve - Containment.

2. Impurity Monitoring and Analysis System

Safety Related Valve	Fail Position
85GMB HV600	Closed

3. Emergency Chilled Water System

Air operated safety related valves are identified on Figures 9.7-10 through 9.7-15.

Table 9.7-7 lists Active Safety related air operated valves with their "Preferred Direction".

59

59

4. Inert Gas Receiving and Processing

New Valve No.	Valve Name or Function	Fail Position
82RPHV001 (1)	Diversion of RAPS input gas to CAPS on signal (3-way valve)	to RAPS
82RPHV002 (1)	Containment isolation valve, RAPS inlet	Close
82RPUV015A (1)	Flow control - surge tank effluent - lo range	Close
82RPUV015B (1)	Flow control - surge tank effluent - hi range	Close
82RPHV018 (1)	Selection of fill of drain of noble gas storage vessel (3-way)	Fail to isolate vessel
82RPHV019 (1)	Selection of fill or drain of noble gas storage vessel (3-way)	Fail to isolate vessel

(1) See Figure 11.3-4.

4. Inert Gas Receiving and Processing (continued)

Valve No.	Valve Name or Function	Fail Position
82APHV001 (2)	Containment isolation valve, CAPS inlet	Close
82APHV002 (2)	Containment isolation valve, CAPS inlet	Close
82NGHV351A (3)	Containment isolation valve, N_2 supply	Close
82NGHV351B (3)	Containment isolation valve, N_2 supply	Close
82CGHV501 (4)	Containment isolation valve, Ar supply	Close
82CGHV301 (4)	Containment isolation valve, recycle Ar supply	Close

(2) See Figure 11.3-6.
(3) See Figure 9.5-8.
(4) See Figure 9.5-2.

5. Steam Generator/Steam Generator Auxiliary Heat Removal Systems

Valve	Normal Operating Position	Final Fail Position After Loss of Compressed Air*	Function
Auxiliary Feedwater Pump Inlet	Open	Open	Isolation
Alternate Auxiliary Feedwater Pump Inlet	Closed	Closed	Isolation
Auxiliary Feedwater Pump Discharge	Open	In Place	Isolation
Auxiliary Feedwater Pump Recirculation	Closed	Open	Isolation
Auxiliary Feedwater Supply	Closed**	Open	Isolation
Drive Turbine Steam Supply	Closed**	Open	Isolation
Superheater Outlet	Open	Closed	Isolation
Superheater Inlet	Open	Closed	Isolation
Evaporator Inlet	Open	Open	Isolation
Feedwater Inlet	Open	Closed	Isolation
Evaporator Water Dump	Closed	Closed	Isolation
Superheater Outlet	Closed	Closed	Relief (Power Operation)
Evaporator Outlet	Closed	Closed	Relief (Power Operation)
Steam Drum Outlet	Closed	Closed	Relief (Power Operation)

* Air stored in an accumulator for emergency operation of the valve. 8

**Open during SGAHRS Heat Removal Operation.

6. Recirculating Gas Cooling System

Air operated safety related valves are shown on Figures 9.16-3 through 9.16-7.

Table 9.16-3 list active air operated safety related valves with their "Preferred Direction".

7. Heating, Ventilating and Air Conditioning

Air operated safety related valves are shown on Figure 9.6-1, 9.6-4, 9.6-5, 9.6-7a.

There are no active air operated safety related valves in the
59 HVAC system.

Question 001.405 (9.6)

Justify the open containment concept for both the RSB and the RCB as being consistent with the low as practicable objective and providing defense in depth for the design basis accident yet to be defined for CRBR. It would seem that the H & V system provided has the prime objective of providing a dilution mechanism for the escaped radioactivity from the reactor and defeats the concept of confining radioactivity for treatment, decay, and later disposal. Secondly, the open containment concept becomes an active system rather than a passive system depending upon the closure of dampers and valves to become effective rather than the reverse situation in which no active action has to occur to provide containment. Justify your selection of this containment concept.

Response:

The CRBRP containment is designed and the plant operating philosophy is developed on the basis of an open containment. The design of the containment and the containment ventilation system is provided with sufficient safeguards to prevent accidental release of radioactive materials. In order to achieve additional defense in depth for the design basis accidents currently identified in the PSAR, the containment system is modified as discussed below.

The presently identified steady state release of radioactivity during plant operation, identified in PSAR Section 11.3, results in effluents and associated doses orders of magnitude lower than the levels of 10CFR20. Section 11.3 lists the total annual gaseous effluent release for CRBRP as 40.15 Ci/yr, compared to the minimum total gaseous effluent release of LWR's studied in WASH-1258 of 3600 ci/yr.

PSAR Section 11.3 lists the integrated dose to the population within 50 miles of the CRBRP site in the year 2010 as 9.6×10^{-2} man-rem/yr. The added cost of a closed containment over the reference design open containment cannot be justified from a cost benefit analysis because of the low operational releases from CRBRP. An integral part of the CRBRP design is to allow personnel access during normal plant operation, in order to ensure equipment operability and to perform routine operations of equipment located within the containment. This equipment is located in containment so that it is closer to the primary system equipment it serves. (Examples are sodium sampling, inservice inspection, access to I&C cubicles, and Large Sodium Component Cleaning Vessel use.) The airborne activity would be too high to allow continuous occupancy during operation if the containment were not purged (open containment).

RCB

In order to provide a very low leakage barrier at the primary containment boundary, a seismic Category I, tornado hardened concrete confinement structure is added around the outside of the inner steel containment vessel with an annular space separating the two structures. The nuclear island and balance of plant structures adjacent to containment are relocated outward to accomodate the space required for the added confinement structure and annular space.

The annular space between the inner and outer containments will be maintained at a negative pressure relative to atmospheric pressure during normal operation and accident condition and is exhausted through high efficiency filters. The
59 filtered exhaust point is chosen to obtain maximum dispersal of the radioactive material prior to reaching the Control Room intake. In addition, the recirculation system for the Control Room atmosphere is increased in capacity to 8500 cfm.

59 The containment atmosphere ventilation system is reduced in capacity from 50,000 cfm to about 14,000 cfm in order to minimize the potential release of activity from containment during valve closure time. The containment supply and exhause penetrations will be reduced from 48" to 24". Containment isolation of the HVAC system exhausts will be designed to meet item 4 of the CRBR Design Criteria 47. The containment ventilation/purge system is provided with a time delay duct to prevent the release of radioactive materials during accident conditions. The time delay duct is sized for such velocity, that the containment isolation valves will close before the contaminated air reaches the valve zone. Radiation monitors which provide signals for initiating closure of the containment isolation valves are provided at the inlet of the time delay duct and in the HAA.

During normal plant operation and all accident conditions, the containment/confinement annulus space is maintained at a minimum 1/4" water gauge negative pressure with respect to the outside atmosphere. During normal plant operation, the RCB Operating Floor is maintained under slight negative pressure (<1/8" water gauge). Capability is provided to filter the containment/confinement annulus exhaust through the annulus filter units during
59 normal plant operation and all accident conditions. The filter system will consist of two 100% redundant filter-fan units consisting of prefilter, demister, heating coil, HEPA filter bank, absorbent filter bank, after HEPA filter bank and fan components (approximately 14,000 CFM capacity).

59 A tornado missile protected, Seismic Category I enclosure is provided for the RCB annulus filter-fan units, the RCB normal exhaust fans and the annulus pressure maintenance fans. Shielded wall partitions are provided
59 in the HVAC equipment room between the redundant annulus filter-fan-units, RCB exhaust fans, and the annulus pressure maintenance fans. Tornado missile protected, Seismic Category I air intake and discharge openings
59 are provided.

RSB

The design for the RSB is described in PSAR Section 3.4 and analyzed in Section 15.6. The resulting doses are significantly below appropriate 10CFR100 guidelines values and meet or exceed all of the Design Criteria specified in PSAR Section 3.1. However, modifications to the RSB HVAC system were made to limit air infiltration and to provide recirculation
59 and filtration capabilities during all operating conditions. A discussion of these modifications to provide additional defense in depth are provided in response to NRC question 310.46. 40

Question 130.83

Indicate how the liner anchor allowable force capacity, yield force capacity, ultimate force capacity and ultimate displacement capacity are determined for Nelson stud anchors. State if the strength of the insulating concrete in which a portion of the stud anchor is embedded affects the force or the displacement capacities of such anchors.

Response:

59 The response to this question is included in revised PSAR Section 3A.8.3.3.

Question 130.87

The first paragraph on top of page III-4 indicates that the integrity of the liner system is maintained even in the event of deterioration of the concrete under accident conditions. Indicate the proportion of concrete that can deteriorate under accident conditions, without affecting the integrity of the liner system.

Response:

59 As stated in the new PSAR Section 3A.8.3.3, the integrity of the liner
system is maintained even in the event of the deterioration of the insu-
59 lating concrete under accident conditions. Section 3A.8.3.3 of the PSAR
discusses the effect on the structural concrete of a sodium spill in a
lined cell and the use of the insulating concrete layer. Preliminary
analysis of the integrity of the cell liner system have also been in-
59 cluded in Section 3A.8.3.3 of the PSAR which indicates that the integrity
of the liner system is unaffected under accident conditions.

Question 130.88

In Figure III-2, the space between the floor slab and the floor liner plate is filled with aggregate. If the floor liner plate is so designed that the gravel will support any load applied to the floor, indicate the method of construction used to assure an adequate support for the liner by gravel during construction and in service, and the criteria of acceptance for the layer of gravel. Also indicate how the floor liner plate is analyzed.

Response:

The response to this question is included in the revised PSAR Section 3A.8.2 and paragraph 3.1.1.6 of section 3.8-B. The method of analysis of the wall and floor liner system is included in the revised PSAR
59 Section 3A.8.3.3.

Question 130.93

In the 3rd paragraph on page IV-10, it is indicated that analyses have been made for bi-planar and tri-planar corners. Indicate the mathematical models and assumptions made for such analyses.

Response:

The response to Question 130.91 provided a description of the straight corner detail which has been adopted to replace the curved corner detail presented in the June 1976 report.

An elasto-plastic finite element analysis using the computer program ANSYS was used in the preliminary analysis to find the stresses and strains in the bi-planar straight corner. The mathematical model is shown in PSAR Figure 3A.8-3. Since the tri-planar corner is restrained in the same fashion as the bi-planar corner, the analysis and results of the bi-planar corner and the tri-planar corner will be similar. Consequently, no tri-planar analysis will be done.

59 Section 3A.8.3.3 presents the mathematical model used and the assumptions made in the analysis.

Question 130.94

In the last paragraph on page IV-10, it is indicated that as a result of the assumption of symmetry conditions, the analysis shows no load imposed on anchors. Indicate how the anchors are designed and how the spacing of 15 inches for the anchors is determined.

Response:

59 This information is discussed in Section 3A.8.3.3.

Question 130.95

In the final liner analysis, the following topics should be taken into consideration. Address how each of these points will be considered:

(a) Indicate the method used to evaluate membrane and bending strains/ stresses beyond the yield point of the material.

(b) Evaluate the effect of geometric restraints which may limit the ductility of the material.

(c) Consider the possibility of thermal shock.

(d) Evaluate the influence of boundary and local conditions which are not symmetrical, such as failure of some anchors, effects of penetrations, transition zones between hot and cold portions of liners and at free edges, effect of local hot spots and effect of fixed equipment supports.

(e) Evaluate the effect of welding carbon steels to steels with different thermal expansion coefficients, such as stainless steel, etc., if any.

(f) Evaluate the influence of local fabrication imperfections.

(g) Evaluate the effect of pre-existing cracks and of cracks generated during the life of the plant; discuss possible propagation of these cracks in the liner.

(h) Describe acceptance and in-service surveillance tests of the liner. Indicate if the welding is to be x-rayed.

(i) Describe the effects on the liner of sliding or rolling equipment supports.

(j) Discuss the need for the liner to be electrically grounded.

(k) State whether liner and penetration subassemblies will be stress-relieved, or heat-treated. Explain whether pre-heating during welding will be done and in what measure its uniformity will be achieved.

(l) Indicate with precision by what method the corrosion allowance has been determined.

(m) Indicate the arrangement where two back-strips cross each other and how a gap at this point is avoided. This gap may generate stress-concentrations in the liner.

(n) Evaluate the effect of all possible local stress risers such as tack welds, plug welds, gaps, etc. (see page A-3).

(o) Indicate how the stresses or strains due to loads such as dead, live, seismic, etc., other than those resulting from primary sodium spill, are obtained and combined with those due to sodium spill.

(p) Indicate the effect on concrete of heat transmitted from the cell liner to the cell liner anchors.

Response:

(a) The liner analysis will be conducted with the finite element computer program ANSYS. Beyond the yield point of the material, an incremental technique is used in "ANSYS". The loading is applied in increments and at each loading level an elastic solution is done, with a correction applied to the next loading step to account for the plasticity occuring during this loading step. The von Mises yield surface is used, along with the Prandle-Reuss flow relations. Since the strains/stresses are claculated at the top, middle, and the bottom surfaces of the plate, the membrane and bending components can be easily separated from the total effects. The membrane strain/stress will be calculated as the average through the section; the bending strain/stress will be calculated as the total strain/stress minus the membrane.

(b) In the areas with geometric restraints or with stress/strain concentrations appropriate mathematical models will be constructed to calculate the effects on the liner; detailed models with regular fine mesh will be used as required.

(c) Those portions of the liner surface subject to thermal shock effects will be reinvestigated. A thermal analysis will be conducted to
59 calculate the temperature distributions through the liner as a function of time, until the thermal soak condition is reached, i.e., the liner temperature through its thickness is uniform and equal to the sodium temperature. Incremental elastic-plastic structural analysis will be conducted for the same time span as the thermal transient analysis.

59 (d) The influence of boundary and local conditions which are not symmetrical will be considered by using mathematical models simulating conditions such as effect of penetrations, transition zones between the heated and unheated portion of liners and at free edges, effect of hot spot and effect of fixed equipment supports.

Limited local failure of the stud anchors will be examined and its effect on the overall integrity of the liner will be investigated.

The problem of hot spots was considered in the preliminary liner analyses by heating a local area of the floor liner in the finite element model of a bi-planar liner corner. It is observed from the results of the analysis that this effect does not govern the design of the liner system.

59 (e) The welding of the austenitic stainless steel to the ferritic steel
material, such as in the Fuel Handling liner will not pose a problem
because of the different thermal expansion coefficients of the
materials. The cell liner and hence the dissimiliar weld point
59 will be exposed to a normal operating temperature of 150^{o}F maximum.
The dissimiliar weld joint will not be subject, under normal operating
temperature, to any thermal cycling. As a result, the weld joint
between stainless steel and the carbon steel will not develop
differential expansion strains. Failure is not expected to occur due
to the absence of long-term cyclic temperature service.

(f) The influence of the following local imperfections will be considered
59 in the final analysis:

(1) <u>Initial bow in the liner plate</u>
In the mathematical model an initial bow will be assumed on
a liner panel with adjacent panels assumed straight. This
will induce unbalanced forces on the anchors and begin strains
59 on the bent panel.

(2) <u>Liner thicker than nominal due to rolling tolerances</u>
If the bowed panel is thinner than adjacent panels, the membrane
force imposed on the buckled panel and the shear forces on
the anchors will be larger than for uniform thickness. The
CRBRP analysis accounts for variation in thickness by utilizing
the actual maximum strength of the liner plate in determining
59 the thermally generated in plane load. This combined with the
corrosion allowance assessment bracket the response.

(3) <u>Yield stress higher than normal</u>
Higher yield stress in the straight panels will have an effect
similar to an increased thickness on the bowed panel. The
analysis described in section 3A.8.3.3 have utilized the
maximum actual yield strength of the liner plant and not the
59 code minimum values, in order to maximize liner strains.

(4) <u>Variation of anchor spacing</u>
59 Small variations in spacing as can be expected during construction
are not considered to have appreciable effect on the anchor-liner
system.

(5) <u>Local concrete crushing on the anchor zone</u>
59 The analysis considered two cases: (a) that the insulating
concrete and insulating gravel on the floor provides full
lateral support to the anchors; (b) that the insulating concrete
59 provides no lateral support to the anchors and are free to bend.
These two cases are considered as envelopes to intermediate
situations in which there is partial yield in the supporting
material.

(g) The fabrication requirements for the liners includes an extensive quality assurance program which will detect pre-service cracks. The liner welds will be liquid penetrant inspected or magnetic particle inspected. In addition, all liner welds will be vacuum box tested. The welds are designed to provide a full strength joint. Because of the quality assurance procedures adopted, initial through-wall cracks will be precluded.

In order to propagate a pre-existing crack, a significant cyclic stress must be experienced by the liner. The thermal cycling of the liner due to normal temperature fluctuations within the cell, has been calculated and the cyclic stresses have been determined. Based upon the number of heat-up and cool-down events estimated for the plant, the fatigue usage factor for the liner is very small, and appears to be well below the endurance limit for the liner material.

The liner has been analyzed for sodium spill conditions which greatly exceed the design basis spill event. These analyses indicate that the liner will not develop cracks or tears.

(h) See Paragraph 3 of the response to Question 130.92.

(i) Refer to Section 3A.8.2 of the PSAR and, also, the response to Question 130.88.

(j) The cell liner system will be electrically grounded to ensure personnel safety in the event of electrical equipment ground faults within the cell cavities. The grounding of the steel liner is deemed necessary to preclude any shock hazard that might otherwise be present in cell cavities consisting of steel-lined floors and walls, and containing electrical equipment. Grounding shall be in accordance with Article 250-44 of the National Electrical Code (1975 Edition), and the method for grounding shall conform to the requirements of Article 250-51 of the National Electrical Code.

(k) Pre-heating during welding of liner and penetration assemblies will be in accordance with the applicable ASME Code, Section III, Division 2, sub-sub article CC-4551 requirements.

The nominal material thickness of the liner and penetration assemblies will be controlled to the requirements of ASME Code Section III, Division
59 2, CC-4552.2.7 so that mandatory postweld heat treatment is not required.

(l) Owing to the presence of inert atmosphere in the cells, corrosion of the carbon steel or stainless steel liners, under normal operating conditions, is entirely absent.

The back side of the carbon steel of stainless steel wall liner material is separted by a ¼-inch air gap followed by a 4-inch insulating precast concrete panel. The back side of the carbon steel or stainless steel floor liner is also separated by a 1/8-inch air gap and a 4-inch thick precast panel of insulating concrete followed by structural concrete. The surface of the insulating/structural concrete interface and the joints between adjacent panels will be sealed to minimize the migration of water toward the liner plate during construction and normal operating conditions. The amount of moisture that might be present in these materials will cause negligible corrosion of the
59 liner material. Accordingly, a 1/16 inch corrosion allowance has been used as referenced in PSAR Appendix 3.8-B.

(m) Continuous backup strips will be provided by welding abutting ends prior to covering with the liner plates.

(n) Tack welds will be used to secure alignment of the liner plates. Their stopping and starting ends will be properly prepared by grinding or other suitable means so that they may be satisfactorily incorporated into the final weld. Tack welds will be visually examined and any defective tack welds will be removed and repaired.

59 No plug welds are anticipated in the fabrication of cell liners.

(o) Since the liner stiffness is negligible compared to the stiffness of floors and walls on which the liner is supported, the liner will deflect with the supporting system. The displacements of supporting points of the liner due to loads such as dead, live, seismic, etc., will be determined considering the concrete floors and walls without liner. Using these displacements at the supporting points of the liner and the loads directly imposed on the liner, the complete liner analysis will conducted. The liner stiffness is such that under seismic conditions it will not vibrate indepentently of the supporting structures.

(p) A three-dimensional transient thermal analysis of the liner with stud anchors has been performed and reported in ORNL-TM-5145 (January, 1976). The analysis indicates that, despite rapid heat-up of the liner plate and direct thermal coupling of the liner and stud anchor, the anchor is ineffectual to transmit heat into the structural concrete. This is particularly true over the short term, when the liner is imposing its maximum loads on the stud anchors. Over the long term, the concrete temperature will rise. The analytical results indicate that heat conduction into the stud anchors will neither accelerate nor localize the process. Consequently, no degradation of concrete capabilities directly attributable to heat conduction into the liner anchors is expected to occur.

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 wall at approximately
59 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.
59 (b) 8,500 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 22

(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

Question 310.32 (6.4 Yellow)

In reference to the question 310.10 response regarding the evaluation of sodium fire effects on control room habitability, the expressed intent of providing the requested information for the PSAR is not acceptable for the purpose of evaluating the PSAR. Review question 310.10 and provide the requested information.

Response:

Information concerning the release of sodium or NaK caused by sodium/NaK tank or pipe rupture by external missiles, or by transportation accidents and the subsequent operation of the control room HVAC system upon detection of the release, is discussed in the response to Question 310.10.

The release of sodium and subsequent sodium fires resulting in the release of sodium combustion products to the atmosphere is described in Section 7.2.1.1 of the Environmental Report. A preliminary analysis was conducted to determine the effect of the release of these products to the atmosphere on the habitability of the Control Room in accordance with Regulatory Guide 1.78. The analysis was based on the following assumptions:

a. At the time of the sodium release and start of the fire, any one of the following sensors, which are located in non-inerted cells containing sodium, will initiate an alarm in the Control Room.

59
- o sodium leak detectors
- o cell temperature detectors
- o cell pressure detectors

In addition, sodium combustion product detectors are also provided in the Control Room air intakes and will activate the emergency operating mode of the Control Room HVAC System.

59 b. Upon detection of sodium smoke, the control room HVAC system emergency operating mode functions as follows:

- 59 The outside air intake where smoke is detected is isolated
- The control room is pressurized to 1/4" positive pressure
- 59 The air required for the control room pressurization is supplied from the redundant air intake
- 59 A portion of the control room atmosphere is continuously recirculated through one of the two redundant Safety Class 3, 99.97% efficient HEPA filter units at an 8500 CFM rate
- 59 The air required for the Control Room pressurization is also filtered by one of two redundant, Safety Class 3, 99.97% efficient HEPA filter units
- 59 49 The control room atmosphere is also recirculated through the 85% efficient HVAC system filters contained in the air condidioning units at a 51500 CFM rate.

c. The Control Room is provided with double doors. The unfiltered infiltration is estimated to be 3 CFM.

The findings of the preliminary analysis are as follows:

a. The time required to isolate the normal control room air intake and initiate operation of the emergency control room HVAC system is significantly less than the time between the initiating signal at the source and the time required for the sodium plume to reach the control room intakes.

b. During the first two minutes after the initiation of the sodium release alarm in the control room the toxicity level in the control room will not exceed the 2mg/m^3 toxicity limit, which is identified in Regulatory Guide 1.78. However, this limit may be exceeded during the course of the accident.

Since the sodium or NaK combustion product concentration may exceed the permissible concentration during the early phase of the accident, but not within the first two minutes, the operability of the control room will be assured by requiring the Control Room operators to use breathing apparatus and protective clothing upon the initiation of the sodium or NaK combustion product alarm in the Control Room. Each operator will be taught to use the breathing apparatus and protective clothing. Practice drills will be conducted to ensure that personnel can don breathing apparatus within two (2) minutes. The time period during which the toxicity limit would be exceeded is anticipated to be relatively short, so that long term operation of the Control Room following the accident can be performed without masks.

The analysis of the sodium fire effect on the control room habitability is continuing as part of the design process and the information presented above will be updated if future results will require a change.

Question 323.2 (2.5.1.1)

Provide a regional geologic cross section which passes through the site area and includes the Valley and Ridge and Blue Ridge provinces. This cross section should show the relationship of surface structures in this region to the regional geology, including "basement" geology.

Response:

The response to this question has been incorporated into revised section
59 2.5.1.1.2. 27

Question 323.4 (2.5.1.1)

Provide information on the extent and nature of the Rome Formation "sole thrust" in this area. This should include evidence such as seismic profiles or drill hole data if available.

Response:

The response to this question has been incorporated into revised Section
59 2.5.1.1.2. 27

Amend. 59
Dec. 1980

Question 323.26 (2.5.1.2.4.1)

Provide the location of drag folds, tight folds, and shears which occur in the site area. Provide specific information on the character of the shears and the amount of displacement along these shears.

Response:

59 The information requested is provided in revised Section 2.5.1.2.4.3. 27

Question 323.35 (Figure 2.5-10)

Explain the existence of several 10 foot clay seams noted in several basehole logs. Show all shear zones and large cavities on geologic cross sections.

Response:

59 The information requested is provided in revised Section 2.5.1.2.4.4.

27

Question 323.37 (2.5.4.2.1)

Page 2.5-35 1st paragraph and Figure 2.4-33 give summary information on the Q D evaluation. Please indicate Q D and core recovery percentiles on the boring logs.

Response:

The percent recovery and rock quality designation (RQD) for borings 25
59| through 105 and 127 through 149 are provided in Appendix 2-A.

Amend. 59
Dec. 1980