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Arizona Nuclear Power Project
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December 8, 1986
ANPP-39257-JGH/BJA/98.05

Director of Nuclear Reactor Regulation
Attention: Mr. George W. Knighton, Project Director
PWR Project Directorate #7
Division of Pressurized Water Reactor Licensing - B
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 3
Docket No. STN 50-530
PVNGS Unit 3 Reactor Containment Building Integrated
Leak Rate Test Summary Technical Report
File: 86-G-056-026

Dear Mr. Knighton:

In accordance with the requirements of 10 CFR 50, Appendix J, please find attached the PVNGS Unit 3 Reactor Containment Building Integrated Leak Rate Test Summary Technical Report.

If you have any questions on this matter, please contact Mr. W. F. Quinn of my staff.

Very truly yours,

J G Haynes

J. G. Haynes
Vice President
Nuclear Production

JGH/BJA/rw
Attachment

cc: O. M. De Michele (all w/a)
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A. C. Gehr

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Arizona Nuclear Power Project

PALO VERDE NUCLEAR GENERATING STATION

UNIT NO. 3

REACTOR CONTAINMENT BUILDING

INTEGRATED LEAK RATE TEST

SUMMARY TECHNICAL REPORT

DECEMBER 1, 1986

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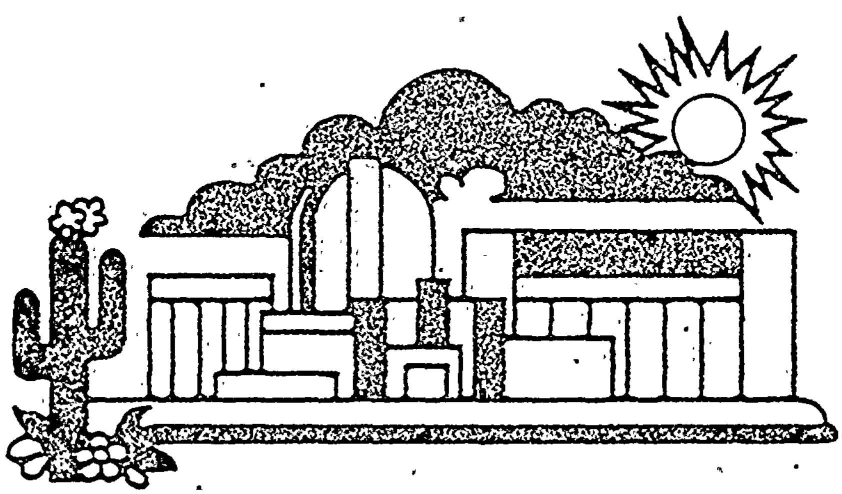
ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
UNIT NO. 3

# REACTOR CONTAINMENT BUILDING INTEGRATED LEAK RATE TEST SUMMARY TECHNICAL REPORT


PREPARED BY
WILLIAM D. ROMAN
ARIZONA PUBLIC SERVICE COMPANY
OPERATIONS ENGINEERING SECTION


December 1, 1986



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# I. INTRODUCTION

A preoperational Type "A" Integrated Leak Rate Test (ILRT) was performed on the containment structure of the Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS) - Unit No. 3 pressurized water reactor in September of 1986. The results of this test were analyzed utilizing the "Absolute Method". This test was performed for a period of eight (8) hours at a pressure equal to or greater than the calculated peak containment internal pressure related to the design bases accident ($P_a$) and specified in the Technical Specifications. This report describes and presents the results of this preoperational Type "A" Leakage Rate Test including the supplemental test method (Controlled Leakage Rate Test or CLRT) utilized for verification.

The test results are reported in accordance with the requirements of 10 CFR 50, Appendix J, Section V.B.2., ANSI N45.4 (1972) and ANSI/ANS 56.8 (1981).

## II. SUMMARY

Prior to performance of the ILRT, Local Leak Rate Tests (LLRTs) were performed to verify containment integrity. These Type "B" and Type "C" tests were performed on containment electrical penetrations, mechanical penetrations, containment isolation valves, fuel transfer tube, equipment hatch and air locks. The acceptance criteria for the LLRTs is that the total leakage from these tests does not exceed 0.60 ($L_a$) where $L_a$ is the maximum allowable leakage rate at the pressure $P_a$ stated as a percent of containment free volume per day (24 hours). The total leakage from these tests was well within these limits and the results are presented in the official copy of preoperational test procedure 91PE-3CL01, Local Leak Rate Test, which is on file at PVNGS.

At the start of the Type "A" test, all valves were in their normal position for accident conditions. Exceptions to this valve lineup were noted and corrected prior to test start and are listed in the official copy of preoperational test procedure 91PE-3CL02, Integrated Leak Rate Test, which is also on file at PVNGS.

The first order least-squares fit analysis of the data utilizing the Total-Time method yielded a leak rate of 0.0234% per day with a 95% upper confidence limit (one sided) of 0.0521% per day. These values are well within the allowable limit of 0.075% per day.

## III. TEST DISCUSSION

### A. Description of Containment

The containment design basis is to limit release of radioactive materials, subsequent to postulated accidents, such that resulting calculated offsite doses are less than the guideline values of 10CFR100. In order to meet this requirement, a design (maximum) containment leakage rate has been defined in conjunction with performance requirements placed on the engineered safety features (ESF) systems.

The capability of the containment structure to maintain design leaktight integrity and to provide a predictable environment for operation of ESF systems is ensured by a comprehensive design analysis and testing program that includes consideration of:

- °The peak containment pressure and temperature associated with the most severe postulated accident coincident with the operating basis earthquake (OBE) or safe shutdown earthquake (SSE).
- °Maximum external pressure loading condition to which the containment may be subjected as a result of inadvertent containment systems operations that potentially reduce containment internal pressure below outside atmospheric pressure.

The bases in determining design are containment peak pressure (and temperature) and external pressure. For the containment structure peak pressure analysis, it is assumed that each postulated accident is concurrent with the most limiting single active failure in systems required to mitigate the consequence of the accident or to shutdown the plant. No two accidents are postulated to occur simultaneously or consecutively.

A. Description of Containment (Cont'd)

The design basis accident (DBA) for each of the categories of: containment peak pressure (and temperature) and containment maximum external pressure is defined as the most severe accident postulated for each case. The difference between the design pressure (60 psig) and the calculated peak pressure of the as-constructed design (49.2 psig) results in a design margin of approximately 20%.

The containment structure is designed to house the reactor coolant system (RCS) and is referred to as the containment. The containment is part of the containment system whose functional requirements are summarized by the following criteria:

- °The containment must withstand the peak pressure and time-varying thermal gradient resulting from a hypothetical failure of the RCS or main steam system.
- °The containment must provide biological shielding during normal operation and following a postulated loss-of-coolant accident (LOCA) to minimize radiation exposure.
- °The containment must be leaktight in order to minimize leakage of airborne radioactive materials.
- °The containment must provide approximately 150 penetrations for piping and electrical cabling, as well as, personnel and equipment access, and provides rigid anchor points for piping entering or leaving.

The containment consists of three basic parts:

- °Flat base slab with a central cavity and an instrumentation tunnel.
- °Right circular cylinder
- °Hemispherical dome

A. <u>Description of Containment</u> (Cont'd)

Principal nominal dimensions of the containment are as follows:

°Interior diameter..............146 ft.

°Interior height (above.........206 ft. - 6 in.
filler slab)

°Cylindrical wall thickness.....4 ft. - 0 in.

°Dome thickness.................3 ft. - 6 in. at dome apex
4 ft. - 0 in. at wall springline

°Base mat thickness.............10 ft. - 6 in.

°Liner plate thickness..........1/4 in.

°Internal free volume...........2,600,000 $ft^3$ net

The containment is constructed of reinforced concrete prestressed by post-tensioned tendons in the cylinder and the dome. The base mat is designed and constructed of conventionally reinforced concrete. Special reinforcing details are provided at discontinuities and at openings in the shell.

A welded steel liner attached to the inside face of the concrete limits the release of radioactivity from the containment. The base liner is installed on the top of the base mat and is covered by a 2 ft. - 9 in. thick concrete slab. The containment building provides biological shielding during normal operation and following a LOCA. It also functions as a leaktight barrier following an accident inside the containment.

The post-tensioning or tendon system consists of high strength wires which are used with button-head anchorage techniques. There are 186 one-quarter inch diameter wires per tendon.

Each tendon assembly consists of wires together with end anchor heads and ring nuts. The tendons transfer load to the structure through shims and a bearing plate.

A. Description of Containment (Cont'd)

Tendons are installed in sheaths that form ducts through the concrete between anchorage points. Trumpets, which are enlarged ducts attached to the bearing plate, allow the wires to spread out at the anchorage to suit button-head spacing requirements. Further, trumpets facilitate field button-heading of wires.

Tendon sheathing provides an enclosed space surrounding each tendon. A valved vent at the highest points of curvature permits release of entrapped air during greasing operations. Drains are provided at the lowest points of curvature to remove accumulated water prior to installing tendons. After the greasing operation, the vents and drains are closed and sealed.

The prestressing tendons are protected against atmospheric corrosion during shipment and installation, and during the life of containment.. Prior to shipment, the tendons are coated with a thin film of petrolatum containing rust inhibitors. The sheathing filler material used for permanent corrosion protection is a modified, refined petroleum-base product. The material is pumped into the sheathing after stressing.

Prestressing of the cylindrical wall is achieved by a post-tensioning system consisting of both vertical inverted U-shaped and circumferential (hoop) tendons. Vertical tendons are anchored at the base slab and extended up and over the dome to form an inverted U shape. Three buttresses are equally spaced at 120° around the cylinder and extend over the dome joining together at the crown. The hoop tendons are anchored at buttresses located at 240° apart. The successive hoop tendons are anchored at alternate buttresses so that two complete horizontal loops are achieved by three consecutive horizontal tendons.

A. Description of Containment (Cont'd)

Prestressing of the hemispherical dome is achieved by a two-way pattern of tendons, which are an extension of the continuous vertical tendons and are anchored at the base slab. They are arranged to produce two families of tendons mutually intersecting each other at 90° on the horizontal projected plane. Hoop tendons extend into the hemispherical region to provide a two-way pattern up to the 90° solid angle of the dome.

A welded steel liner plate covers the entire inside surface of the containment (excluding penetrations) to satisfy the leaktight criteria. The liner is typically 1/4-inch thick and is thickened locally around penetration sleeves, large brackets, and attachments to the basemat and shell wall. The stability of the liner plate, including the thickened plate, is controlled by anchoring it to the concrete structure. The shell wall and dome liner plate system is also used as a form for construction.

A circular equipment hatch and two personnel airlock assemblies (100' and 140' elevations) penetrate the concrete cylinder walls. Penetration assemblies consist of steel sleeves or nozzles, reinforcing plates and anchors. They are anchored to the concrete walls and are welded to the steel liner. Hatch and air lock doors are provided with double-gasketed flanges with provisions for leak testing the flange-gasket combinations.

The 100' elevation personnel air lock is for emergency access. Each personnel air lock has a door at each end and is an ASME Code stamped pressure vessel. A quick-acting equalizing valve connects the personnel air lock with the interior or exterior of the containment to equalize pressure in the two systems.

A. Description of Containment (Cont'd)

During plant operation, the two doors of each personnel air lock are interlocked to prevent both being opened simultaneously. Remote indicating lights and annunciators in the control room indicate the operational status of the doors. Provision is made to bypass the interlock system during plant cold shutdown.

Single barrier piping penetrations are provided for all piping passing through the containment walls. The closure for process piping to the liner plate is accomplished with a special flued head welded into the piping system and to the penetration sleeve which is, in turn, welded to a reinforced section of the liner plate. In the case of piping carrying hot fluid, the pipe is insulated to prevent excessive concrete temperatures and to prevent excessive heat loss from the fluid. Closures to these penetration assemblies are provided by the piping systems that are served by the penetrations.

Electrical penetration assemblies provide means for carrying one or more electric circuits through a single aperture (nozzle) in the containment pressure barrier while maintaining the integrity of the pressure barrier.

Medium voltage power penetrations are configured in the form of tubular canisters slightly shorter than the containment structure nozzle into which it will be installed. The penetration assemblies are installled in 24-inch diameter nozzles. The canister is used as a pressure chamber to monitor penetration leakage rate by pressurizing the interior space with nitrogen and measuring the leak rate with a pressure gauge. The medium voltage power penetration is flange-mounted to the outside containment wall with nuts, bolts, washers, and lock-washers. The aperture seal is formed between the header plate and the flange with two concentric Viton O-rings.

[illegible]

A. Description of Containment (Cont'd)

The low voltage power, control, and instrumentation penetrations are also flange-mounted to the outside containment wall in the manner described for the medium voltage power penetrations. Each penetration in this category has a stainless steel header plate at the outside containment end. Stainless steel feed-through subassemblies, containing electrical conductors, pass through the header plate and are secured and sealed with special stainless steel compression fittings. The interstices between the seals and feed-through subassemblies provide a pressure chamber which is used to monitor the leakage rate.

A fuel transfer tube penetration is provided for refueling. An inner pipe acts as the refueling tube with an outer pipe as the housing. The tube is fitted with a double-gasketed blind flange in the refueling canal and a standard gate valve in the spent fuel pool.. This arrangement prevents leakage through the refueling tube. Outer sleeves permit the transfer tube to penetrate the refueling canal wall, the containment shell, and the exterior wall of the fuel handling building, while maintaining a pressure-tight boundary at each wall. The sleeves are anchored into each wall respectively and welded to each wall's liner plate. The housing is supported by the sleeves in the vertical and horizontal directions. Bellows at both the interior and exterior faces of the containment shell and of the fuel handling building permit thermal expansion of the transfer tube and of the housing. The same expansion bellows permit differential movement between structures.

The structural acceptance criteria complies with ASME Section III, Division 2, Article CC-3000. The fundamental acceptance criteria for the complete containment is successful completion of the structural integrity test with measured responses within the limits predicted by analyses.

[illegible]

[illegible]

A. <u>Description of Containment</u> (Cont'd)

Prediction of limits are based on test load combinations and code values for stress, strain, or gross deformation for the range of material properties and construction tolerances specified.

The structural integrity test is planned to yield information on both the overall response of the containment and the response of localized areas, such as major penetrations or buttresses, which are important to its design functions.

The design and analysis methods, as well as the type of construction and construction materials, are chosen to allow assessment of the structure's capability throughout its service life. Additionally, surveillance testing provides further assurances of the structure's continuing ability to meet its design functions.

[illegible]

## B. Description of Instrumentation

A "state-of-the-art" ILRT instrumentation package was utilized to allow leak rate determination by the "Absolute Method". The primary measurement variables include containment pressure, dewpoint temperature and drybulb temperature as a function of time. Ancillary measurements include outside ambient temperature and barometric pressure. During the supplemental CLRT, containment verification (fixed-orifice) flow is also measured. Instrument readings were output at 15 minute intervals via a data acquisition system and line printer. The measurement system is shown in Figure 8. The mass of air (Q) is calculated by the Perfect Gas Law as follows:

$$Q = \frac{P_a V}{RT} = \frac{(P_t - P_{wv}) V}{RT}$$

where:
- $P_a$ = air partial pressure
- $V$ = free volume
- $R$ = gas constant
- $T$ = temperature
- $P_t$ = total pressure, psia
- $P_{wv}$ = water vapor pressure, psia

### 1. Temperature Instrumentation

Twenty-four (24) precision platinum Resistance Temperature Detectors (RTD's) were located throughout containment to allow measurement of the volumetrically weighted average drybulb temperature. The specified accuracy of the RTD's is ± 0.1°F (40°F to 120°F range). The specified repeatability for each sensor is 0.025% of temperature or 0.05°C, whichever is greater.

[illegible]

B. Description of Instrumentation (Cont'd)

2. Dewpoint Instrumentation

Six (6) chilled-mirror Dewcells were located throughout the containment to allow measurement of the volumetrically weighted dewpoint temperature. The specified accuracy of each of the sensors is ± 0.3°C (± 0.54°F), nominal over a range of -50°C to +100°C (-58°F to 212°F). The specified repeatability for each sensor is ± 0.11°F.

3. Pressure Instrumentation

Two (2) precision fused quartz bourdon tube pressure indicators (0-100 psia) were provided for the determination of containment absolute pressure. One pressure indicator was utilized as a primary while the second indicator was available as a backup. The specified accuracy of the indicators is ± 0.015% of reading. The repeatability of the indicator is ± 0.0005% full scale.

4. Flow Instrumentation

Two (2) thermal mass flowmeters with a range of 0 to 10scfm were utilized during the supplemental CLRT for verification flow. The specified accuracy of the instrument is ± 1.0% full scale. The specified repeatability of the instrument is ± 0.2% full scale.

5. Ancillary Instrumentation

The outside ambient temperature and barometric pressure as well as wind speed and wind direction were obtained from Luke Air Force base and local weather station meteorological instrumentation via telecon.

[illegible]

C. Description of the Computer Program

The ILRT computer program is an APS-specified vendor-supplied program which performs the leak rate calculation utilizing the Generator Temperature Monitor (GTM) mini-computer (LSI 11/23). The computer is connected via a data link to the Data Acquisition System (DAS). The drybulb temperature, dewpoint temperature and absolute pressure data that are scanned by the DAS are fed to the computer for storage and printing. The ILRT computer system consists of:

°Volumetrics A-100 DAS
°DEC VT55 FE graphics terminal with hard copy unit
°LSI 11/23 ILRT computer system with dual double-density disk drives.
°Parallel line printer, Data Royal 5000

After every scan by the DAS, the computer will print a "Raw Data Summary Report" (RDSR). The computer stores the data and, on demand, prints the "ILRT Program Report" (PR). From this report, temperature stabilization can be calculated from average temperature. The ILRT computer uses the Total-Time or Mass-Plot analysis technique to calculate the measured leak rate, calculated leak rate, and 95% upper confidence limit leak rate. The 95% upper confidence limit leak rate is used to determine if the test has met the acceptance criteria. During the verification test or CLRT, the computer will calculate the composite leak rate ($L_c$). To aid the Test Director in data analysis, plots of the data are made. The RDSR, PR and plots are contained in Appendix A.

The computer contains the following:
°DEC LSI 11/23 processor with KEV 11 option
°128K bytes of memory
°two double-density disk drives (RX02 format)
°two serial line interfaces
  °one for console device
  °one for serial link to DAS

C. Description of the Computer Program (Cont'd)

°DEC VT55-FE graphics terminal with hard copy unit
°TCU-50D timing control unit
°Parallel line printer

The system software consists of an operating system and an applications package. The operating system is supplied by DEC as the RT-11 version 4.0 Foreground/Background monitor with the appropriate RT-11 version 4.0 device handlers.

The applications package consists of the following programs (not including special maintenance and editing programs):

°LOOK
°SCAN
°EXAM
°CONWEI
°CALPRE
°CALC
°RELHUM
°INERR
°PLOT
°INLEAK

Program LOOK will read data from the A-100 DAS. These data are displayed on the console device. The data output from the DAS are in the same form as is output during the ILRT (i.e., 24 RTD temperatures, 6 dewpoint temperatures, 2 pressure readings, time and date). This program is used during the initial phases of the equipment set-up. The program is a never-ending loop and requires operator intervention to exit.

C. <u>Description of the Computer Program</u> (Cont'd)

Program SCAN is designed to read data from the A-100 DAS and re-format the data into a form more digestable to the other programs in the application package. The program will run continuously until a total of 257 data scans have been received or halted by operator intervention. This program will also run concurrently with the other programs in the application package; it has priority in execution if a conflict arises. The operation of the program is transparent to the user.

Program EXAM is designed to display the contents of the raw data files acquired by the program SCAN. This program will inspect the data files to determine if the raw data file needs editing before being utilized in the calculation sequence.

Program CONWEI is used to create or modify the containment weighting factors of the sensors used in the calculation program. The containment is divided into various sub-volumes. The sub-volume is represented by RTD's and Dewcells. Their readings are proportionally applied to the total volume.

Program CALPRE is designed to compute the calibration constants for pressure gauges. The program requests the true pressure and gauge readings for both pressure gauges, then derives the multiplication factor and correction constant for each gauge.

Program CALC is the main application module in the applications package. This program takes the pre-formatted data from the raw data files and performs various calculations with it to produce the various parameters required in the final report of leak rate. The results of these calculations are stored in two data files for use in the plot routines.

[illegible]

C. Description of Computer Program (Cont'd)

Upon execution, the program CALC reads the scan data files, containment weighting factors and the pressure gauge calibration constants (see Appendix A-General). The RTD and Dewcell temperatures are then multiplied by their corresponding weighting factors and summed. The program checks each sensor reading to insure that it is within the allowable deviation for that set of readings. The elapsed time from "time zero" is calculated and a true pressure is determined from the gauge readings and calibration constants. The pressure is then corrected for the effects of the water vapor pressure. The weighted average containment temperature, average weighted Dewcell temperature and containment pressure are used to compute the measured and calculated leak rate for the Point-to-Point, Total-Time and Mass-Plot methods.

From these values, a regression line is calculated by the least-squares fit method to compute a calculated leak rate for each of the methods. The upper confidence limit is calculated with the "Students T" analysis of n-2 degrees of freedom where n is the number of data samples utilized at each time n.

Program RELHUM is designed to read the average containment dry-bulb and dewpoint temperatures and compute a value of the relative humidity in the containment.

Program INERR is designed to compute the instrument error as a function of average containment temperature, number of RTD sensors, average corrected containment pressure, number of Dewcell sensors, elapsed time and the accuracy of the various sensors used.

Program PLOT is designed to accept computed data from the programs: CALC, RELHUM, INERR, and display the results on the DEC VT55-FE graphics terminal.

[illegible]

C. Description of the Computer Program (Cont'd)

Program INLEAK is designed to calculate the value of the installed leak for the CLRT as measured by the ILRT system. The program requires the operator to enter the various leak rate parameters.

This program interacts with the user to convert the leak rates obtained in weight percent per day to standard cubic feet per minute. The conversion is obtained by calculating the initial containment mass and applying the measured leak rate to this mass. The program also calculates the installed leak.

## D. Error Analysis

The instrument system error analysis is based on the Instrument Selection Guide (ISG) formula ANSI/ANS 56.8-1981 "Containment System Leakage Testing Requirements." The formula is:

$$ISG = \pm \frac{2400}{t}\left[2\left(\frac{ep}{P}\right)^2 + 2\left(\frac{epv}{P}\right)^2 + 2\left(\frac{et}{T}\right)^2\right]^{\frac{1}{2}} \text{ \%/day}$$

where,

ep = absolute pressure measurement repeatability error divided by the square root of the number of sensors.
= $(.0005\%)\ (100 \text{ psia})/(1)^{\frac{1}{2}}$
= .0005 psia

epv = vapor pressure measurement accuracy error divided by the square root of the number of sensors.
= $(.54°F)\ (0.0124 \text{ psia}/°F)^*/(6)^{\frac{1}{2}}$
= .00273 psia
* From steam tables at dewpoint temperature range 69–71°F

et = drybulb temperature measurement repeatability error divided by the square root of the number of sensors.
= $(0.1°F)/(24)^{\frac{1}{2}} = .0204°F$

P = Test pressure
= 63.9 psia

T = Test temperature (nominal)
= 540° R

t = Test duration in hours

t = 8 hours

Therefore, the ISG is:

$$ISG = \frac{2400}{8}\left[2\left(\frac{.0005}{63.9}\right)^2 + 2\left(\frac{.00273}{63.9}\right)^2 + 2\left(\frac{.0204}{540}\right)^2\right]^{\frac{1}{2}} \text{ \%/day}$$

ISG = ± 0.0244% per day for 8 hour ILRT

Additional error calculations are discussed in Section III.C.

[illegible]

E. Description of Tests

The containment was made ready for the Integrated Leak Rate Test (ILRT) with final containment inspection, closure and exlusion areas established at 1745 hours on 9-11-86. It should be noted that the Structural Integrity Test (SIT) was scheduled and was performed in conjunction with the ILRT. The official start of this milestone was 1800 hours on 9-10-86 which was the first data set for SIT baseline at 0 psig taken twenty-four (24) hours prior to commencement of containment pressurization. The details concerning SIT performance such as instrumentation, data collection, acceptance criteria, etc. shall be transmitted by others under separate cover and shall not be specifically addressed in this report with the exception of interface areas such as pressure hold points, outgassing, etc. Prior to commencement of this milestone, various tasks were completed including ILRT instrument sensor installation, in-situ testing, temperature surveys, Type "B" and "C" testing, valve line-ups, etc. Various problems were encountered and resolved during this period. These problems primarily concerned the inability of the A-100 Data Acquisition System (DAS) to transmit in the automatic mode, pressurization equipment malfunction, etc. The details concerning these items can be found in preoperational test procedure 91PE-3CL02 (Rev. 0), Integrated Leak Rate Test and/or the corresponding Test Log on file at PVNGS. Dewcell numbers 4 and 5 (located at the 178' and 110' elevation respectively) exhibited erratic behavior during this period and were to be closely observed for possible future deletion if the conditions continued.

A pneumatic test (nitrogen) was satisfactorily performed prior to ILRT start on the steam generators (secondary-side) up to the MSIVs at aproximately 70 psig to identify and correct any resultant leakage detected. This pressure was then reduced to essentially atomospheric to assure no potential adverse effects on the ILRT test results with temporary pressure gauges (located on the main steam lines

[illegible]

E. <u>Description of Tests</u> (Cont'd)

external to containment) left in-place to monitor secondary-side pressure with the RCS dry and vented to containment atmosphere. Just prior to containment pressurization, additional tests were performed on the containment personnel lock (140' elevation) and the emergency lock (100' elevation) seals to reverify their integrity. Both locks tested satisfactorily with no measurable seal leakage. All non-essential loads in containment were de-energized at this time.

At 1800 hours, on 9-11-86, pressurization of the containment commenced with nine (9) of ten (10) mobile air compressors in-service having a total capacity of approximately 10,000 cfm (one compressor out of service due to defective starter). The compressors were oil-free, diesel-driven, rotary screw-type units. These units were connected to the containment as shown in Figures 9, 10, and 11. An average rate of approximately 3.2 psi/hr was achieved with an average air inlet temperature to containment of approximately 65°F to 75°F maintained by adjusting cooling water flow to the after-cooler and chiller-dryer units. With both containment ambient and outside ambient average temperatures of approximately 85°F, this inlet air temperature reduced the stabilization time by limiting the containment temperature gradient. This became a concern since the Purge System was not available prior to pressurization and portable circulating fans in containment were not being utilized. At 1815 hours another air compressor failed reducing the pressurization rate to approximately 2.4 psi/hr. Two (2) replacement compressors were ordered and were en route to PVNGS. The leak survey team was deployed at 1900 hours. At 2000 hours with the containment at 5.0 psig, a leak was evidenced within the 100' airlock (airlock pressure gauge indicated 5.0 psig) apparently due to leakage past the inner door equalizing valve. Consequently, two (2) leak survey team personnel were cleared for entry (medically and administratively) and at 2155 hours after a containment air sample was taken and found satisfactory, containment entry was made.

[illegible]

E. Description of Tests (Cont'd)

The equalizing valve was found to be not fully closed apparently due to the linkage arm nut position. The linkage arm nut was adjusted resulting in full closure of the valve. The containment was exited at 2200 hours at approximately 9.75 psig (preoperational test procedure limit for containment entry is 10.0 psig). Further external leak checking in the airlock with SNOOP revealed that the leakage had ceased. The airlock was then exited with no additional leakage or increase in airlock pressure observed. Pressurization continued with SIT holdpoints at 10, 25, 40 and 55 psig. Problems continued with loss of air compressors and the maximum pressurization rate achieved was approximately 3.2 psi/hr. Relative humidity increased considerably during this period but was significantly reduced by increasing and maintaining backpressure at the pressurization skid. Steam generator secondary-side pressure gauges continued to indicate no increase in pressure. The leak survey team remained active with no leaks of any consequence observed.

At 2358 hours on 9-12-86 the overpressure test plateau of 69.0 psig (115% of containment design pressure of 60 psig) was achieved. At 0200 hours on 9-13-86 depressurization to the outgassing plateau of 41.8 psig (85% of $P_a$) commenced with one (1) SIT holdpoint satisfied at 55 psig. Depressurization was maintained at a maximum rate of 10 psi/hr. Depressurization was secured at 0630 hours at approximately 41 psig with the pressurization line to containment vented to atmosphere. This started the 24 hour stabilization period at 85% $P_a$. During this period, no additional leaks were observed and airlock pressures remained at 0 psig. An unofficial "ILRT" was also performed during this period with excellent results. Consequently, the leak survey team was secured until further notice.

[illegible]

E. Description of Tests (Cont'd)

Pressurization to peak accident pressure ($P_a$) commenced at 0635 hours on 9-14-86 with the leak survey team again deployed. Pressurization was secured at 0858 hours at a pressure of approximately 49.7 psig. A 0.5 psig "buffer" was intentionally installed to assure pressure did not fall below $P_a$ due to temperature stabilization and/or potential leakage. The pressurization line to containment was again vented to atmosphere and the stabilization period commenced at 0915 hours with stabilization achieved and all criteria satisfied at 1330 hours. During the ensuing period, the leak survey team was reactivated and a complete analysis of all instrument sensors commenced. The analysis again revealed that Dewcell numbers 4 and 5 were exhibiting erratic behavior and were trending opposite their associated dewcells at the same elevation. After several more hours of analyzing and trending, it was concluded that Dewcell numbers 4 and 5 were erratic and adversely affecting the test results. Consequently, these sensors were deleted from the calculations and the volume fractions for the remaining four (4) dewcells were adjusted, accordingly.

Upon completion of this evaluation, the ILRT was officially started (time zero) at 1600 hours and was successfully completed at 2400 hours for a total duration of eight (8) hours. The reduced duration test was performed subsequent to discussions with on-site NRC representatives and upon satisfying all the requirements for both ANSI/ANS-56.8 (1981) and BN-TOP-1 (Revision 1-1972). The results yielded a calculated Total-Time leak rate of 0.0234% per day and 0.0521% per day at the 95% upper confidence limit (one-sided). With the test results being satisfactory, the leak survey team was terminated.

Minor problems were encountered in preparing for the CLRT concerning the input parameters for the induced flow rate. These were basically input errors that were corrected but resulted in minor delays due to

[illegible]

[illegible]

E. Description of Tests (Cont'd)

adjustments to the actual induced flow rate. Upon resolution and no further adjustments to the verification flow, the CLRT stabilization commenced at 0130 hours on 9-15-86. A fixed-orifice "leak" for verification of the ILRT data of 7.49 scfm was superimposed. This flow was approximately equivalent to 0.1% per day ($L_a$) at actual test pressure conditions. The actual CLRT commenced at 0245 hours and was successfully completed at 0703 hours for a total CLRT duration of 4.75 hours. The results yielded a calculated Total-Time leak rate of 0.103% per day. It should be noted that no data sets or individual data points were rejected for either the ILRT or CLRT.

Depressurization to atmosphere commenced at 0800 hours with SIT holdpoints satisfied at 40, 25 and 10 psig. Depressurization was again maintained at a maximum rate of 10 psi/hr. At 0045 hours on 9-16-86, 0 psig containment pressure was achieved followed by containment air sampling and personnel entry. All sumps were verified to be dry as they were prior to pressurization. The exclusion areas were removed and no abnormalities were noted. The nitrogen blanket on the steam generators (secondary-side) was confirmed to have no indicated pressure increase. The ILRT was satisfactorily completed and system/component restoration commenced.

[illegible]

## IV. RESULT AND VERIFICATION

The ILRT was conducted for a period of eight (8) hours with a total of thirty-three (33) samples or data sets taken. The results of a calculated least-squares statistical fit of all data yielded a Total-Time leak rate of 0.0234% per day with a 95% upper confidence limit (one sided) of 0.0521% per day.

Following satisfactory completion of the ILRT at $P_a$, a four and three-quarter (4.75) hour CLRT was performed with a total of twenty (20) samples or data sets taken. This test was conducted by superimposing a known fixed-orifice leak approximately equivalent to $L_a$ (0.1% per day) of 7.49 scfm. The calculated Total-Time leak rate for CLRT was 0.103% per day.

No data samples were rejected in computing the results for either the ILRT or the CLRT and all data were recorded at equal fifteen (15) minute intervals.

[illegible]

## V. CONCLUSIONS

The Integrated Leak Rate Test at peak accident pressure provided acceptable results as evidenced by the computer printouts in Appendix A of this report. The computed leak rate is well within the specified limit. The acceptance criteria for the ILRT is as follows:

1 - The maximum allowable operational leak rate shall not exceed 75% of $L_a$ (0.1% per day) at a pressure of not less than $P_a$ ( 49.2 psig):

° 0.075% per day

2 - The accuracy of the ILRT is verified by a supplemental test (CLRT) where a calibrated leak is imposed on the existing leaks ($L_{am}$) in the containment system. The superimposed leak rate ($L_o$) shall be between 75% and 125% of $L_a$. Acceptability is demonstrated if:

$$(L_o + L_{am} - 0.25\ L_a) \leq L_c \leq (L_o + L_{am} + 0.25\ L_a)$$
$$(0.1+0.0234-0.025) \leq L_c \leq (0.1+0.0234+0.025)$$
$$(0.0984) \leq L_c \leq (0.1484)$$

| | Leak Rate ($L_{am}$) % per 24 hrs by weight | |
|---|---|---|
| **ILRT** | Fitted | 95% UCL |
| °Total-Time Analysis | 0.0234 | 0.0521 |

**CLRT**

°Induced Flow — 7.49 scfm ($L_a$ or 0.1%)

## V. CONCLUSIONS (Cont'd)

| CLRT | Leak Rate ($L_c$) % per 24 hrs by weight |
|---|---|
| °Total-Time Analysis | 0.103 |

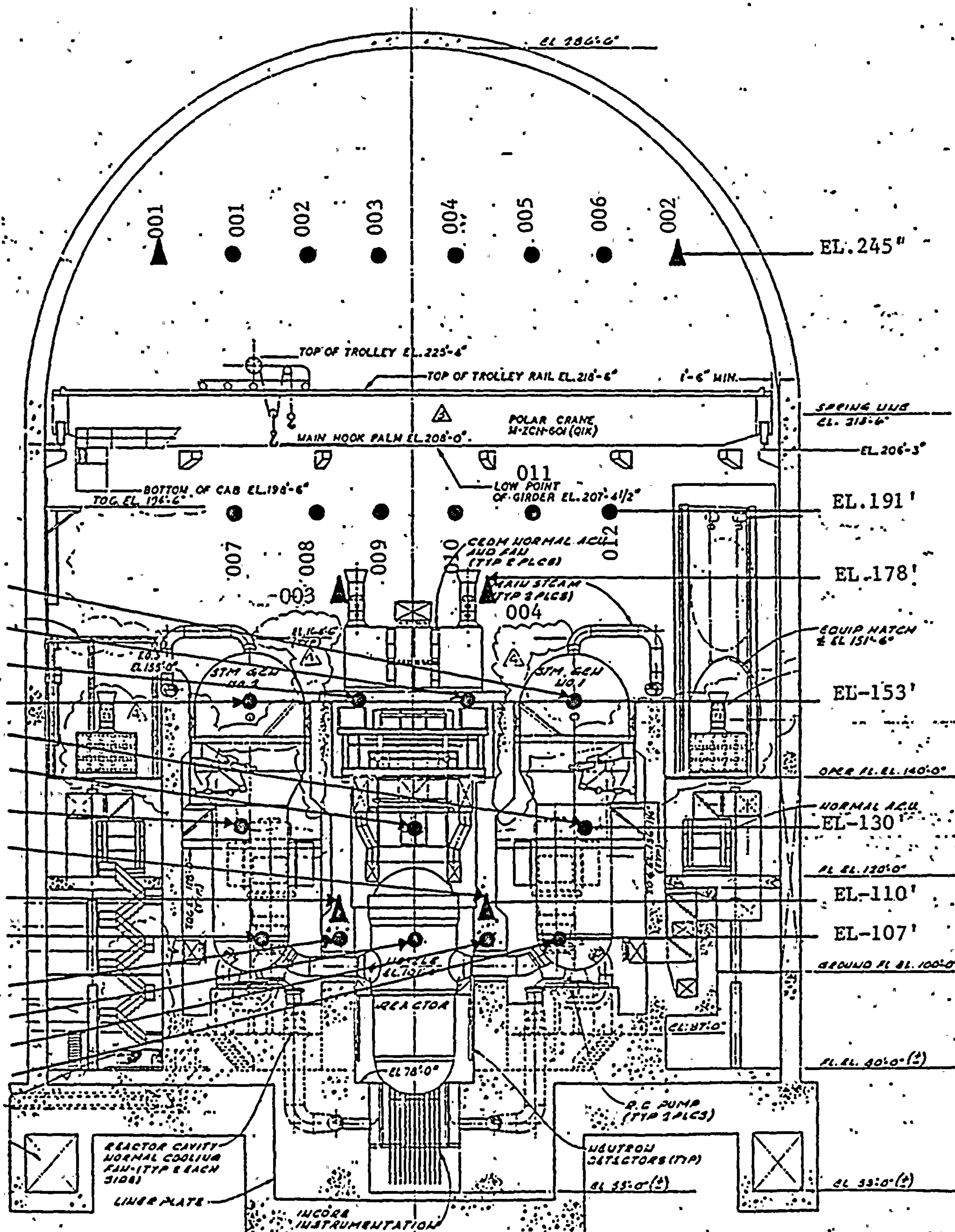
| CLRT LIMITS | CLRT Limits % per 24 hrs by weight |
|---|---|
| Total-Time Analysis | |
| °Upper Limit | 0.1484 |
| °Lower Limit | 0.0984 |

The computer generated reports based upon verified data substantiate for both the ILRT and CLRT that an acceptable test has been performed in accordance with 10 CFR 50, Appendix J, ANSI N45.4 (1972) and ANSI/ANS 56.8 (1981).

# VI. FIGURES

Figure 1

RTD & ME LOCATION




NOTE:
RTD's 1-6 are elevated above the polar crane at approximately elevation 245' (suspended from containment spray headers)

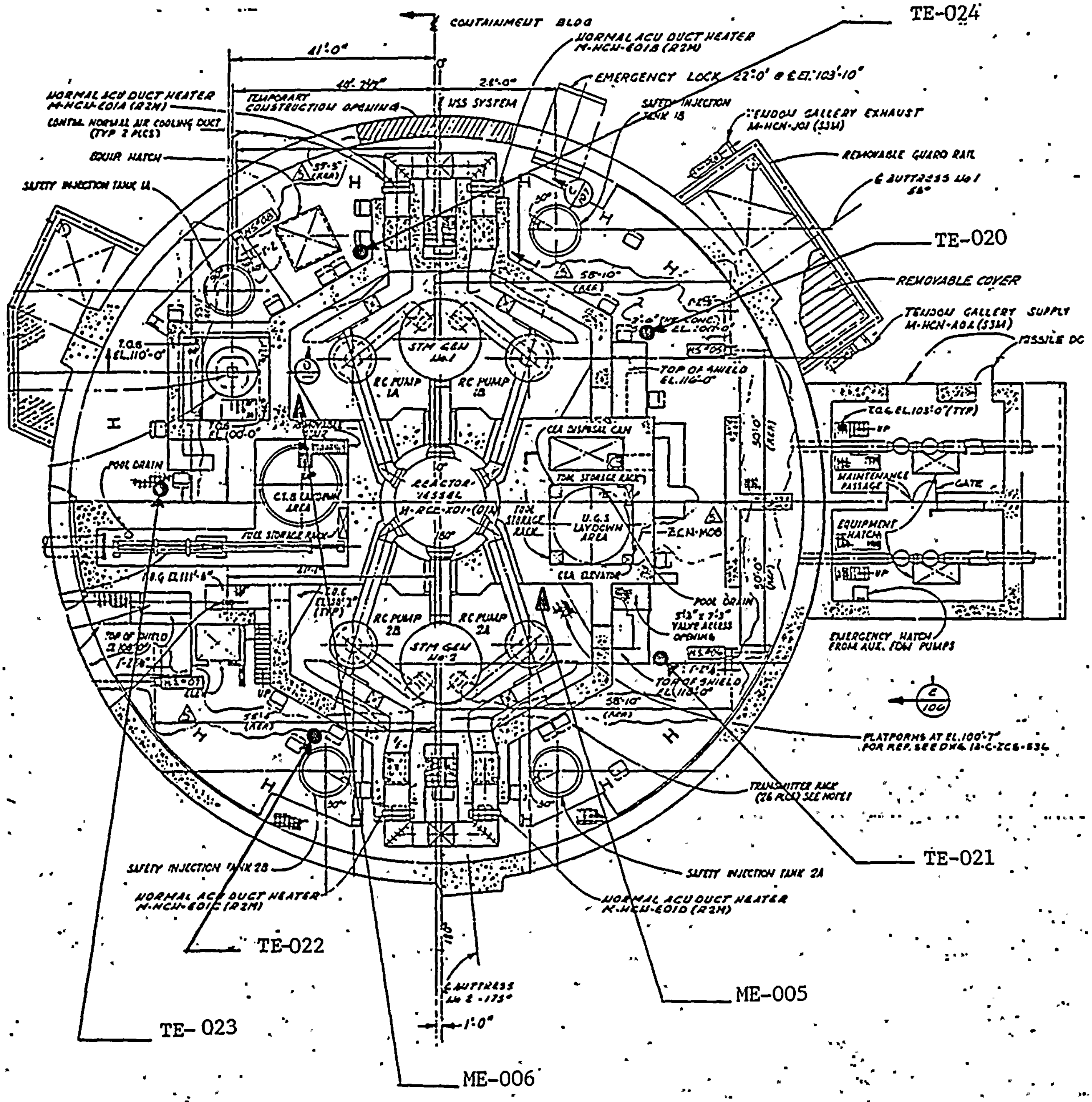
● TEMPERATURE ELEMENT (RTD)

▲ DEWCELL ELEMENT (ME)

[illegible]

Figure 2

RTD & ME LOCATION



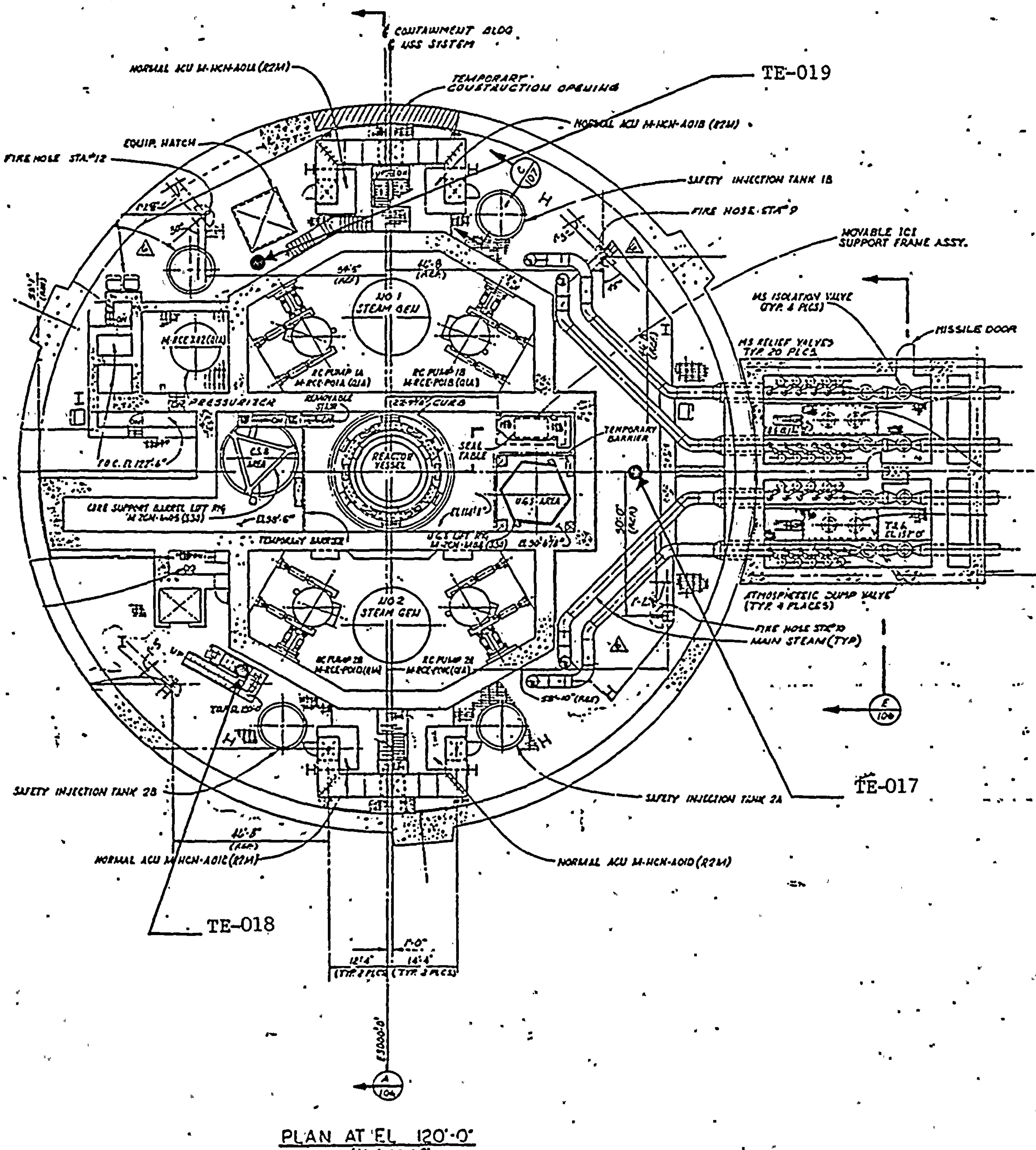

PLAN AT EL 100'-0"

TE's 020 through 024 are located at EL.107'
ME 005 and 006 are located at EL. 110'

[illegible]

Figure 3

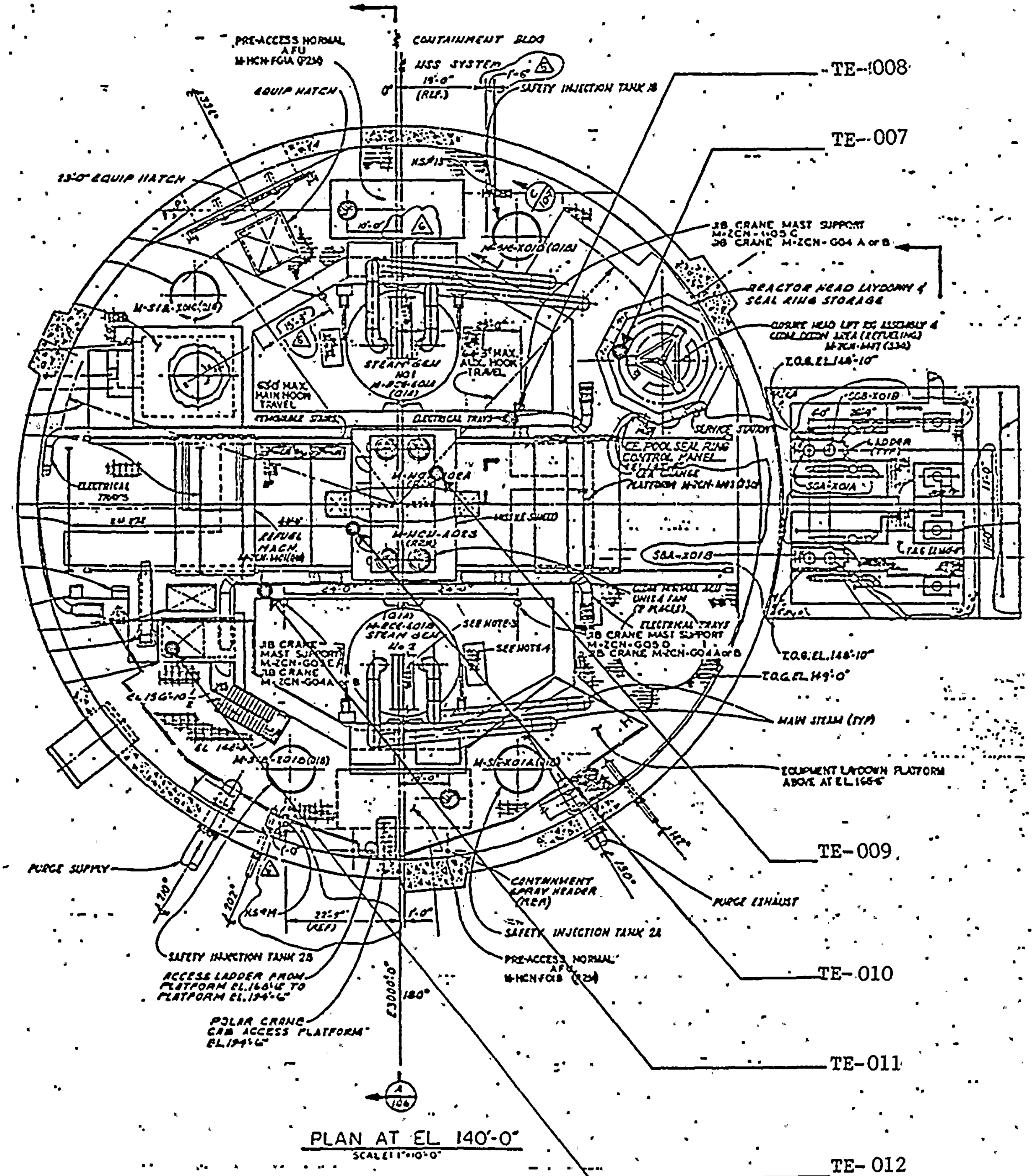
RTD & ME LOCATION




TE's 017 through 019
are located at EL.130'

Figure 4

RTD & ME LOCATION




PLAN AT EL. 140'-0"

NOTE:
RTD's 7-12 are lowered from the crane to an elevation approximately 30' below the top of the crane rail.

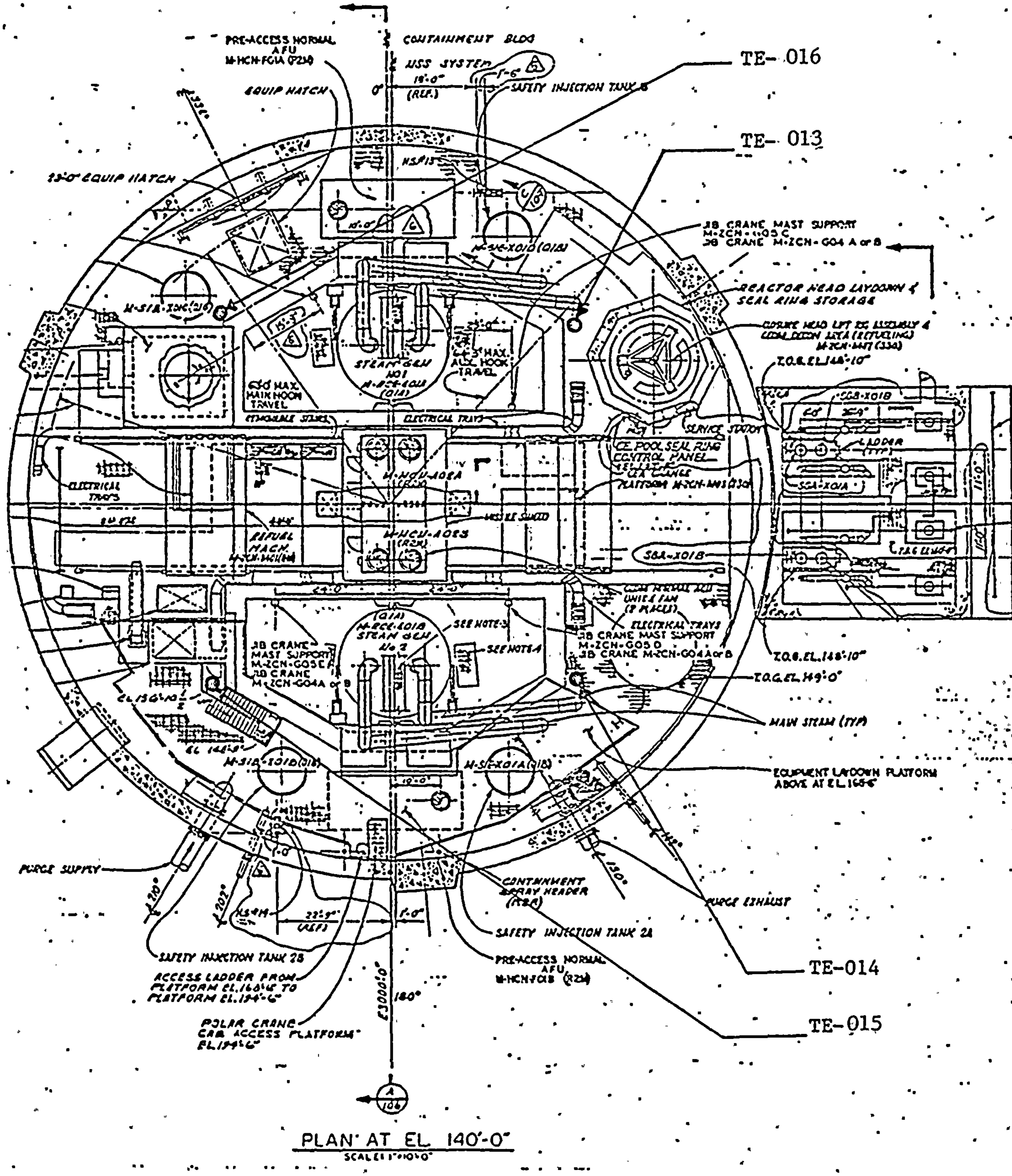
TE's 007 through 012 are located at EL.191'

[illegible]

Figure 5

RTD & ME LOCATION



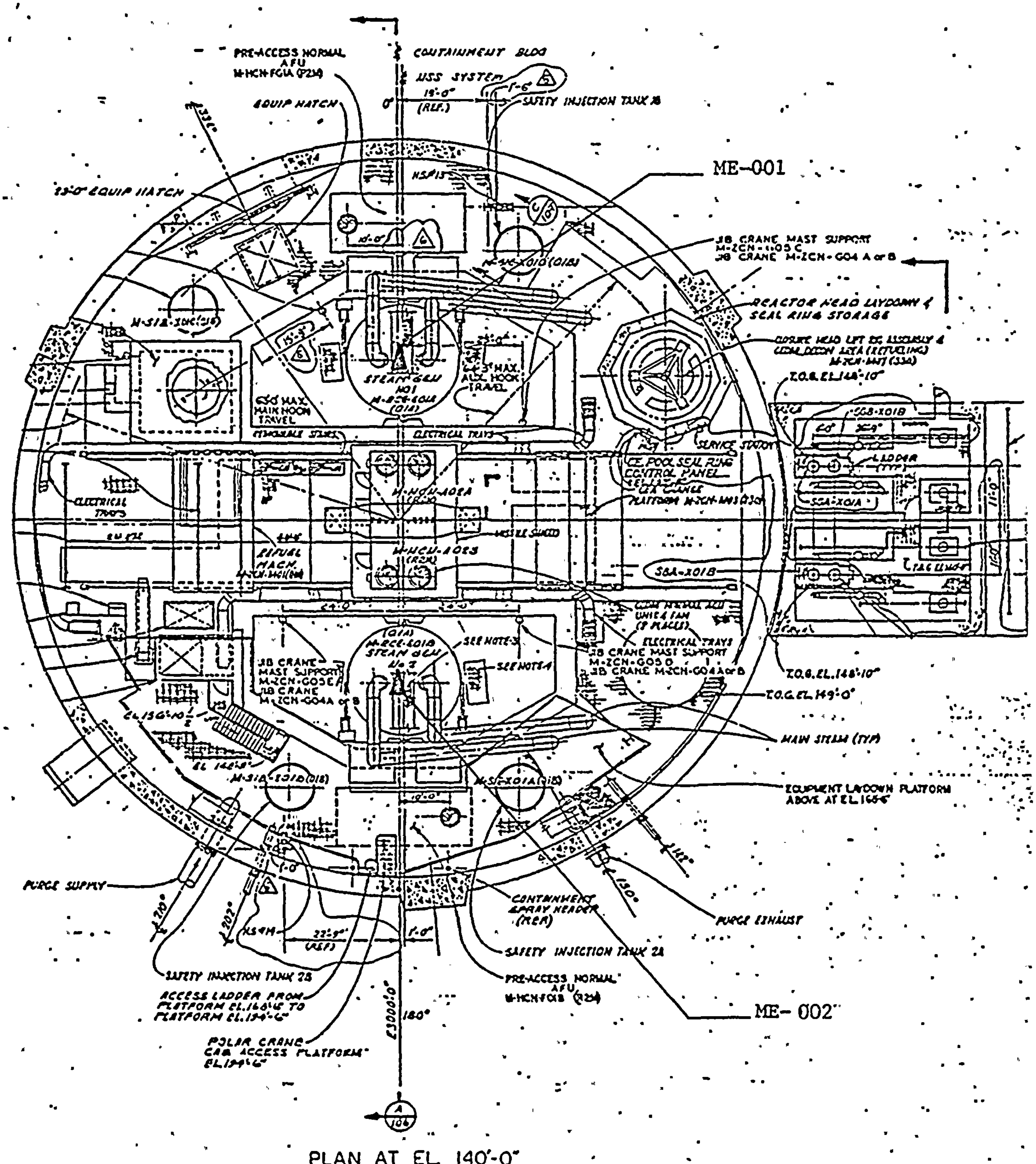



TE's 013, 014, 015 and 016 are located at EL.153"

[illegible]

Figure 6

RTD & ME LOCATION




PLAN AT EL 140'-0"

NOTE:

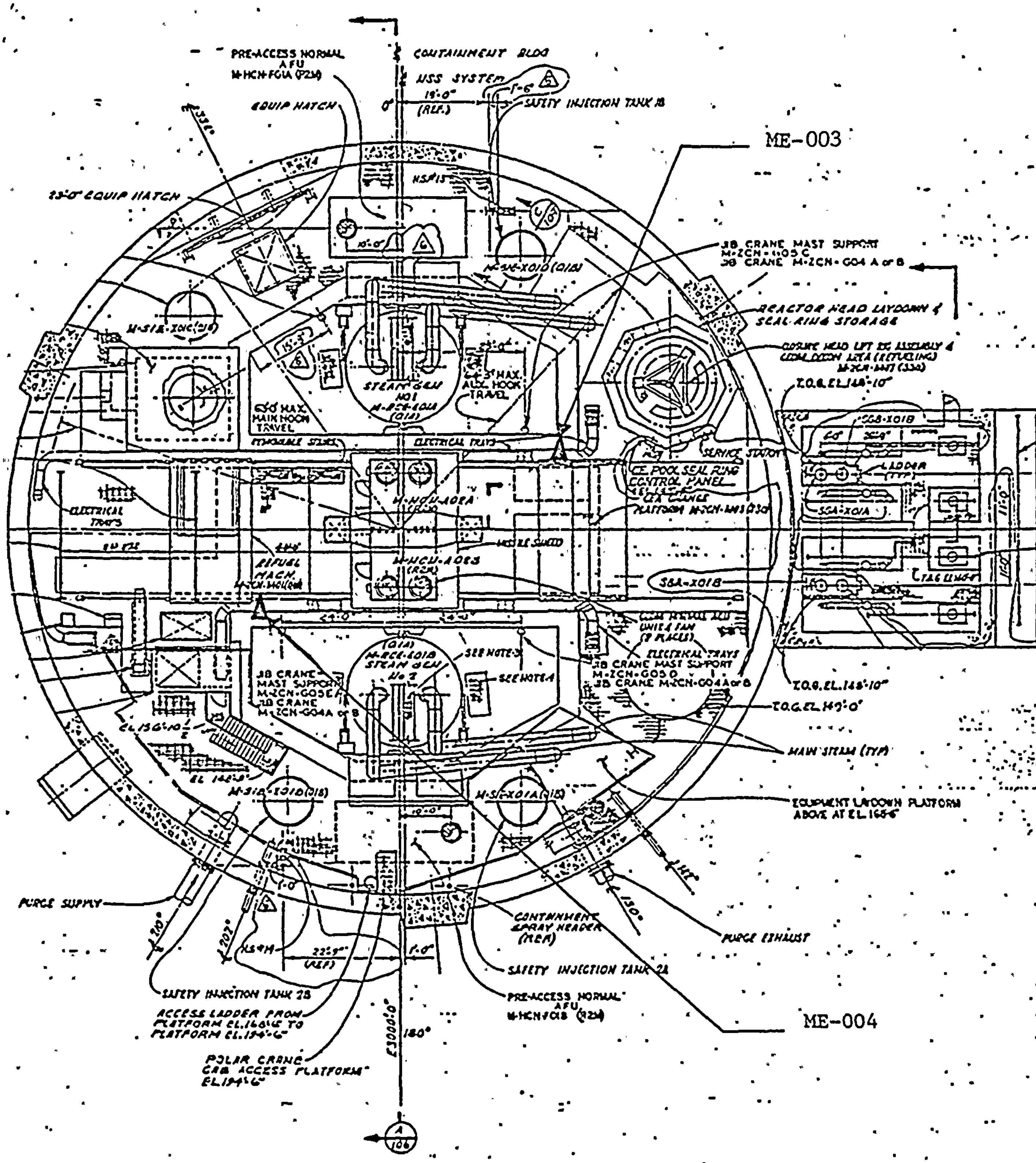
ME's 1 and 2 are elevated above the polar crane at approximately elevation 245' (suspended from containment spray headers)

ME's 001 and 002 are located at EL.245'

[illegible]

Figure 7

RTD & ME LOCATION




PLAN AT EL 140'-0"

NOTE:

ME's 3 and 4 are lowered from the crane to an elevation approximately 30' below the top of the crane rail.

ME's 003 and 004 are located at EL.178'

Figure 8

ILRT MEASUREMENT SYSTEM
SCHEMATIC

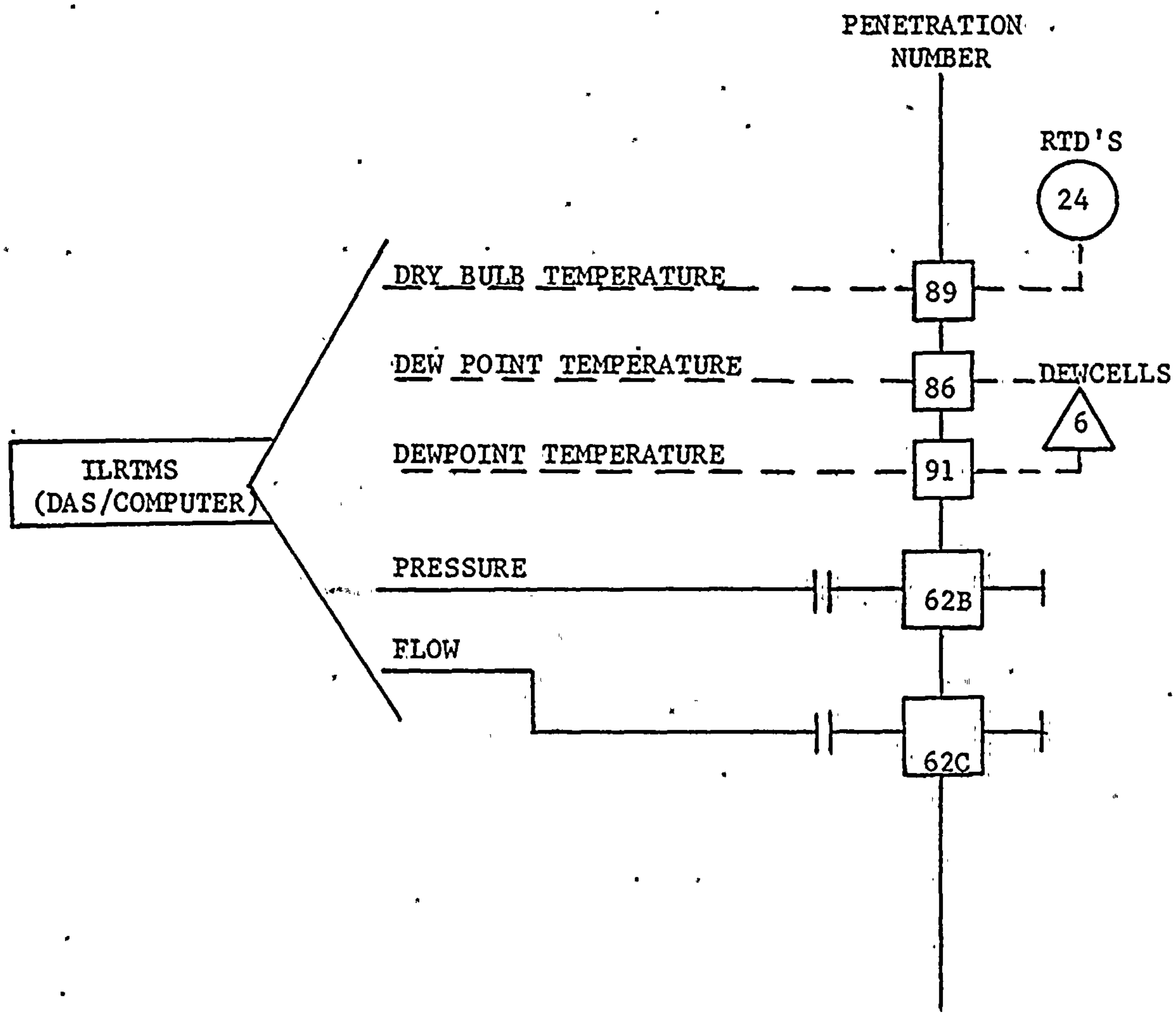


Figure 9

PRESSURIZATION SYSTEM

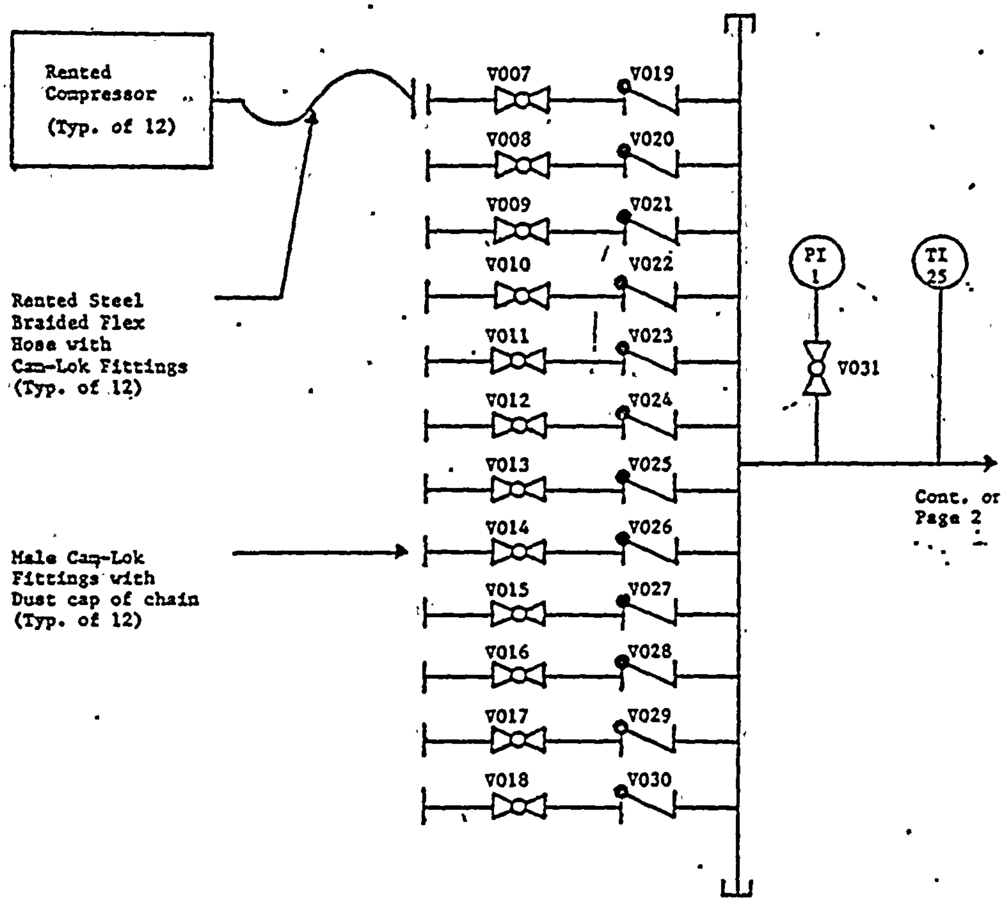


Figure 10

PRESSURIZATION SYSTEM

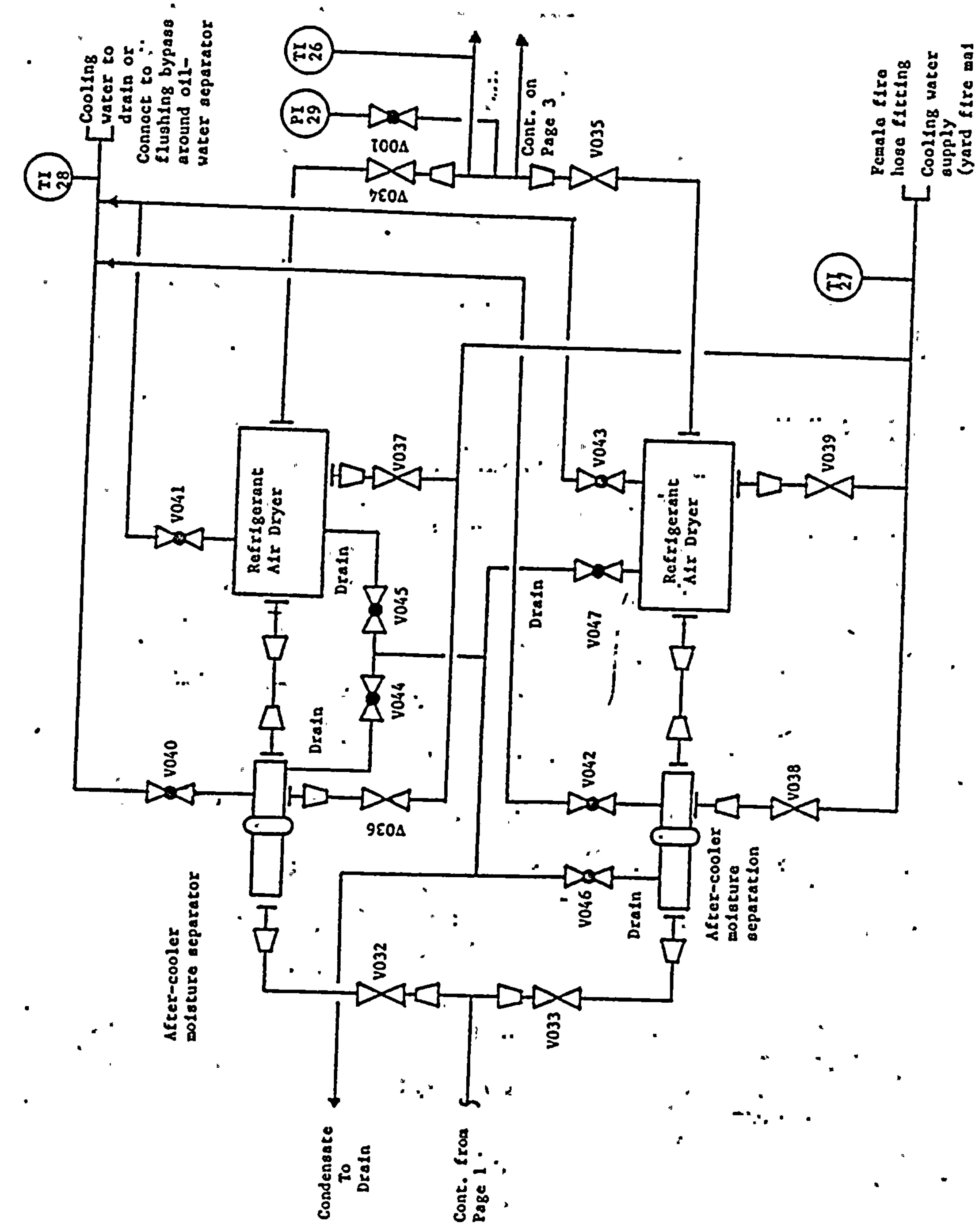
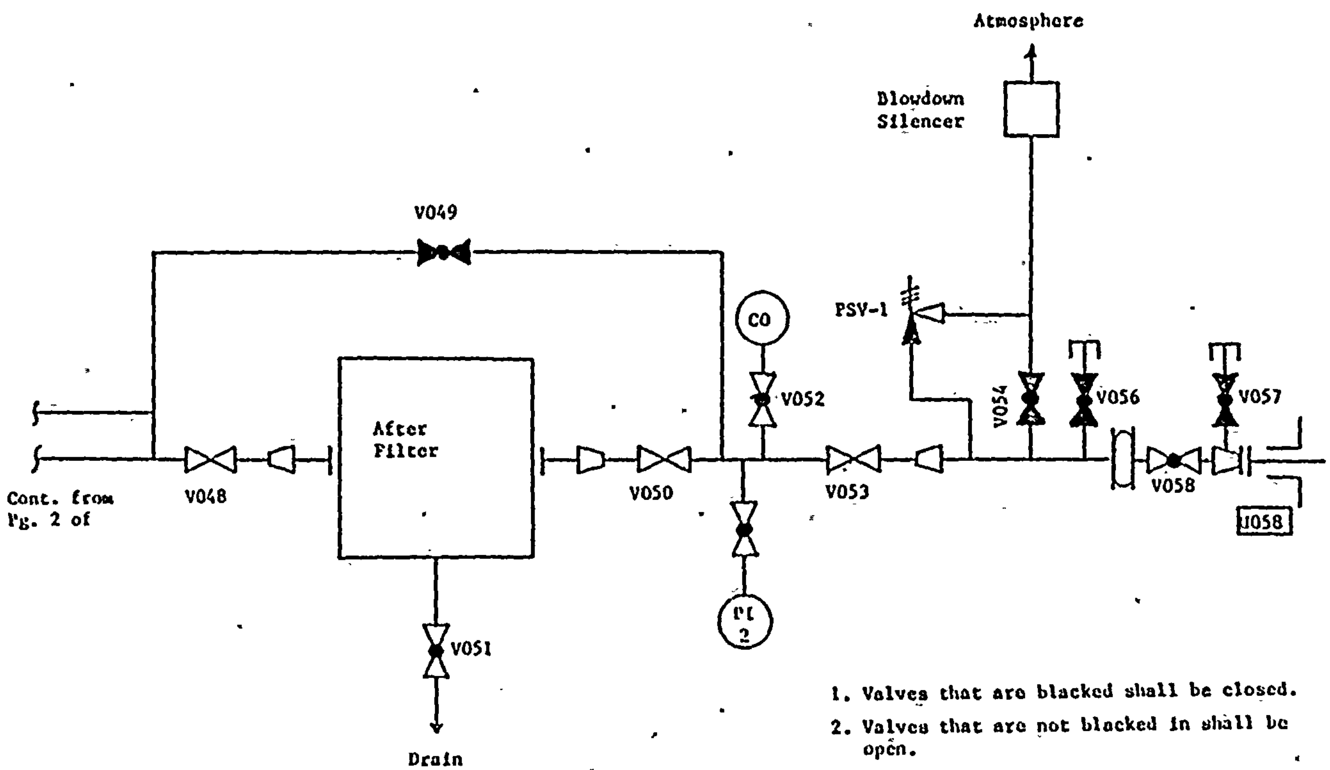


Figure 11
PRESSURIZATION SYSTEM




1. Valves that are blacked shall be closed.
2. Valves that are not blacked in shall be open.

# APPENDIX A

COMPUTER - GENERATED REPORT

# 1.

# STABILIZATION

ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
RAW DATA SUMMARY REPORT

| SCAN NO. | DATE | TIME | PRESS 1 | PRESS 2 | RTD #1 | RTD #2 | RTD #3 | RTD #4 | RTD #5 | RTD #6 | RTD #7 | RTD #8 | RTD #9 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| SD.101 | 257 | 9:15 | 62.915 | 63.009 | 100.280 | 100.640 | 100.260 | 101.360 | 100.840 | 100.230 | 97.355 | 97.232 | 97.395 |
| SD.102 | 257 | 9:30 | 62.843 | 62.937 | 99.156 | 99.487 | 99.165 | 100.020 | 99.656 | 99.099 | 96.426 | 96.292 | 96.441 |
| SD.103 | 257 | 9:45 | 62.795 | 62.889 | 98.300 | 98.626 | 98.266 | 99.040 | 98.765 | 98.288 | 95.802 | 95.620 | 95.809 |
| SD.104 | 257 | 10: 0 | 62.760 | 62.854 | 97.607 | 98.001 | 97.639 | 98.370 | 98.126 | 97.622 | 95.332 | 95.207 | 95.416 |
| SD.105 | 257 | 10:15 | 62.733 | 62.827 | 97.166 | 97.496 | 97.154 | 97.876 | 97.607 | 97.142 | 94.978 | 94.908 | 95.063 |
| SD.106 | 257 | 10:30 | 62.711 | 62.806 | 96.768 | 97.100 | 96.737 | 97.467 | 97.204 | 96.768 | 94.686 | 94.674 | 94.744 |
| SD.107 | 257 | 10:45 | 62.693 | 62.788 | 96.382 | 96.757 | 96.417 | 97.116 | 96.876 | 96.438 | 94.505 | 94.422 | 94.541 |
| SD.108 | 257 | 11: 0 | 62.678 | 62.773 | 96.136 | 96.476 | 96.130 | 96.836 | 96.603 | 96.116 | 94.352 | 94.233 | 94.406 |
| SD.109 | 257 | 11:15 | 62.665 | 62.760 | 95.907 | 96.252 | 95.906 | 96.595 | 96.359 | 95.904 | 94.218 | 94.091 | 94.241 |
| SD.110 | 257 | 11:30 | 62.653 | 62.749 | 95.710 | 96.069 | 95.716 | 96.400 | 96.138 | 95.752 | 94.099 | 93.975 | 94.094 |
| SD.111 | 257 | 11:45 | 62.643 | 62.740 | 95.532 | 95.918 | 95.541 | 96.231 | 95.953 | 95.552 | 93.983 | 93.853 | 93.978 |
| SD.112 | 257 | 12: 0 | 62.635 | 62.731 | 95.387 | 95.768 | 95.385 | 96.064 | 95.782 | 95.422 | 93.859 | 93.747 | 93.870 |
| SD.113 | 257 | 12:15 | 62.628 | 62.724 | 95.256 | 95.637 | 95.256 | 95.959 | 95.643 | 95.282 | 93.751 | 93.591 | 93.792 |
| SD.114 | 257 | 12:30 | 62.620 | 62.717 | 95.150 | 95.529 | 95.143 | 95.863 | 95.535 | 95.192 | 93.660 | 93.483 | 93.714 |
| SD.115 | 257 | 12:45 | 62.615 | 62.712 | 95.030 | 95.419 | 95.045 | 95.784 | 95.413 | 95.132 | 93.585 | 93.425 | 93.641 |
| SD.116 | 257 | 13: 0 | 62.609 | 62.706 | 94.894 | 95.311 | 94.949 | 95.661 | 95.312 | 95.025 | 93.548 | 93.392 | 93.563 |
| SD.117 | 257 | 13:15 | 62.604 | 62.701 | 94.804 | 95.257 | 94.883 | 95.553 | 95.274 | 94.970 | 93.498 | 93.333 | 93.499 |
| SD.118 | 257 | 13:30 | 62.599 | 62.696 | 94.729 | 95.159 | 94.812 | 95.474 | 95.172 | 94.853 | 93.429 | 93.281 | 93.431 |

ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
RAW DATA SUMMARY REPORT

| SCAN NO. | RTD #10 | RTD #11 | RTD #12 | RTD #13 | RTD #14 | RTD #15 | RTD #16. | RTD #17 | RTD #18 | RTD #19 | RTD #20 |
|---|---|---|---|---|---|---|---|---|---|---|---|
| SD.101 | 97.401 | 97.534 | 98.310 | 90.774 | 89.033 | 88.479 | 87.883 | 82.836 | 82.203 | 83.548 | 79.379 |
| SD.102 | 96.450 | 96.508 | 97.270 | 90.411 | 88.531 | 88.075 | 87.428 | 82.490 | 82.035 | 83.287 | 79.489 |
| SD.103 | 95.758 | 95.874 | 96.646 | 90.203 | 88.307 | 87.826 | 87.223 | 82.366 | 81.967 | 83.168 | 79.565 |
| SD.104 | 95.361 | 95.448 | 96.200 | 90.103 | 88.201 | 87.693 | 87.147 | 82.278 | 81.924 | 83.067 | 79.612 |
| SD.105 | 95.056 | 95.141 | 95.878 | 89.988 | 88.131 | 87.603 | 87.080 | 82.209 | 81.890 | 82.988 | 79.653 |
| SD.106 | 94.717 | 94.915 | 95.608 | 89.941 | 88.070 | 87.548 | 87.039 | 82.156 | 81.845 | 82.937 | 79.678 |
| SD.107 | 94.496 | 94.728 | 95.425 | 89.894 | 88.021 | 87.497 | 87.017 | 82.104 | 81.807 | 82.905 | 79.710 |
| SD.108 | 94.360 | 94.540 | 95.253 | 89.852 | 87.983 | 87.442 | 87.002 | 82.060 | 81.759 | 82.881 | 79.728 |
| SD.109 | 94.189 | 94.419 | 95.105 | 89.833 | 87.941 | 87.400 | 86.975 | 82.048 | 81.735 | 82.875 | 79.747 |
| SD.110 | 94.067 | 94.310 | 94.975 | 89.819 | 87.918 | 87.365 | 86.958 | 82.038 | 81.720 | 82.870 | 79.762 |
| SD.111 | 93.959 | 94.187 | 94.860 | 89.794 | 87.898 | 87.335 | 86.955 | 82.029 | 81.704 | 82.850 | 79.777 |
| SD.112 | 93.830 | 94.099 | 94.772 | 89.796 | 87.883 | 87.324 | 86.952 | 82.016 | 81.691 | 82.847 | 79.794 |
| SD.113 | 93.736 | 94.020 | 94.673 | 89.784 | 87.861 | 87.300 | 86.933 | 82.003 | 81.675 | 82.832 | 79.806 |
| SD.114 | 93.661 | 93.946 | 94.598 | 89.758 | 87.843 | 87.289 | 86.923 | 81.991 | 81.666 | 82.818 | 79.806 |
| SD.115 | 93.577 | 93.875 | 94.523 | 89.738 | 87.829 | 87.277 | 86.917 | 81.980 | 81.652 | 82.809 | 79.809 |
| SD.116 | 93.499 | 93.804 | 94.456 | 89.723 | 87.820 | 87.269 | 86.917 | 81.973 | 81.657 | 82.811 | 79.820 |
| SD.117 | 93.414 | 93.736 | 94.389 | 89.703 | 87.808 | 87.258 | 86.927 | 81.964 | 81.652 | 82.801 | 79.826 |
| SD.118 | 93.345 | 93.678 | 94.326 | 89.686 | 87.799 | 87.251 | 86.917 | 81.951 | 81.654 | 82.791 | 79.830 |

ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
RAW DATA SUMMARY REPORT

| SCAN NO. | RTD #21 | RTD #22 | RTD #23 | RTD #24 | DEW CELL #1 | DEW CELL #2 | DEW CELL #3 | DEW CELL #4 | DEW CELL #5 | DEW CELL #6 |
|---|---|---|---|---|---|---|---|---|---|---|
| SD.101 | 79.486 | 78.898 | 79.260 | 79.190 | 78.994 | 80.050 | 78.657 | 78.088 | 69.877 | 70.864 |
| SD.102 | 79.505 | 78.916 | 79.269 | 79.296 | 78.887 | 79.917 | 78.619 | 78.001 | 69.703 | 70.903 |
| SD.103 | 79.557 | 78.945 | 79.383 | 79.344 | 79.039 | 79.856 | 78.526 | 77.964 | 69.393 | 70.947 |
| SD.104 | 79.612 | 78.994 | 79.466 | 79.390 | 78.978 | 79.760 | 78.521 | 77.941 | 69.441 | 70.741 |
| SD.105 | 79.649 | 79.022 | 79.496 | 79.412 | 78.866 | 79.751 | 78.401 | 77.188 | 69.609 | 70.663 |
| SD.106 | 79.676 | 79.043 | 79.512 | 79.434 | 78.904 | 79.678 | 78.273 | 77.635 | 69.899 | 70.816 |
| SD.107 | 79.679 | 79.063 | 79.536 | 79.438 | 78.906 | 79.669 | 78.395 | 77.639 | 69.683 | 70.665 |
| SD.108 | 79.704 | 79.078 | 79.559 | 79.458 | 78.878 | 79.708 | 78.295 | 77.708 | 69.561 | 70.497 |
| SD.109 | 79.719 | 79.089 | 79.577 | 79.470 | 78.659 | 79.742 | 78.204 | 77.644 | 69.537 | 70.552 |
| SD.110 | 79.734 | 79.101 | 79.596 | 79.481 | 78.799 | 79.684 | 78.286 | 77.478 | 69.651 | 70.413 |
| SD.111 | 79.739 | 79.112 | 79.608 | 79.493 | 78.791 | 79.718 | 78.347 | 77.269 | 69.560 | 70.571 |
| SD.112 | 79.754 | 79.129 | 79.620 | 79.505 | 78.772 | 79.652 | 78.254 | 77.214 | 69.569 | 70.822 |
| SD.113 | 79.760 | 79.132 | 79.621 | 79.516 | 78.799 | 79.682 | 78.276 | 77.681 | 69.535 | 70.835 |
| SD.114 | 79.765 | 79.136 | 79.623 | 79.521 | 78.799 | 79.650 | 78.257 | 77.536 | 69.519 | 70.768 |
| SD.115 | 79.768 | 79.145 | 79.628 | 79.527 | 78.779 | 79.628 | 78.265 | 77.523 | 69.598 | 70.674 |
| SD.116 | 79.776 | 79.151 | 79.629 | 79.534 | 78.794 | 79.667 | 78.343 | 77.504 | 69.667 | 70.782 |
| SD.117 | 79.777 | 79.161 | 79.632 | 79.541 | 78.744 | 79.687 | 78.350 | 77.528 | 69.671 | 70.814 |
| SD.118 | 79.776 | 79.164 | 79.628 | 79.539 | 78.747 | 79.670 | 78.419 | 77.522 | 69.720 | 70.782 |

[illegible]

ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
ILRT PROGRAM REPORT

STARTING DAY - 257 STARTING TIME - 9:15: 0 STARTING SCAN - SD.101 ENDING SCAN - SD.118

| | | | | POINT TO POINT | | TOTAL TIME | | | MASS PLOT | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| SCAN NO. | ELAPSED TIME (HR) | AVERAGE TEMP. (F) | AVERAGE PRESSURE (PSIA) | MEASURED LEAK RATE | CALCULATED LEAK RATE | MEASURED LEAK RATE | CALCULATED LEAK RATE (WEIGHT PERCENT PER DAY) | UPPER CONFIDENCE | MEASURED LEAK RATE | CALCULATED LEAK RATE | UPPER CONFIDENCE |
| SD.101 | 0.00 | 92.03 | 63.736 | | | | | | | | |
| SD.102 | 0.25 | 91.36 | 63.663 | -0.803E+00 | -0.803E+00 | -0.803E+00 | -0.803E+00 | 0.000E+00 | -0.803E+00 | -0.803E+00 | 0.000E+00 |
| SD.103 | 0.50 | 90.90 | 63.615 | -0.615E+00 | -0.615E+00 | -0.709E+00 | -0.709E+00 | 0.000E+00 | -0.709E+00 | -0.709E+00 | 0.000E+00 |
| SD.104 | 0.75 | 90.59 | 63.579 | -0.282E+00 | -0.306E+00 | -0.566E+00 | -0.574E+00 | -0.496E+00 | -0.566E+00 | -0.447E+00 | 0.379E+00 |
| SD.105 | 1.00 | 90.35 | 63.552 | -0.223E+00 | -0.170E+00 | -0.480E+00 | -0.473E+00 | -0.434E+00 | -0.480E+00 | -0.363E+00 | -0.172E+00 |
| SD.106 | 1.25 | 90.15 | 63.530 | -0.916E-01 | -0.398E-01 | -0.403E+00 | -0.386E+00 | -0.342E+00 | -0.403E+00 | -0.292E+00 | -0.161E+00 |
| SD.107 | 1.50 | 89.99 | 63.512 | -0.217E-01 | 0.561E-01 | -0.339E+00 | -0.313E+00 | -0.257E+00 | -0.339E+00 | -0.233E+00 | -0.126E+00 |
| SD.108 | 1.75 | 89.87 | 63.497 | -0.164E+00 | 0.388E-01 | -0.314E+00 | -0.262E+00 | -0.178E+00 | -0.314E+00 | -0.205E+00 | -0.127E+00 |
| SD.109 | 2.00 | 89.76 | 63.484 | 0.114E-02 | 0.917E-01 | -0.275E+00 | -0.218E+00 | -0.119E+00 | -0.275E+00 | -0.177E+00 | -0.112E+00 |
| SD.110 | 2.25 | 89.67 | 63.472 | 0.300E+00 | 0.236E+00 | -0.211E+00 | -0.168E+00 | -0.695E-01 | -0.211E+00 | -0.130E+00 | -0.602E-01 |
| SD.111 | 2.50 | 89.58 | 63.462 | 0.244E+00 | 0.311E+00 | -0.165E+00 | -0.120E+00 | -0.213E-01 | -0.165E+00 | -0.870E-01 | -0.150E-01 |
| SD.112 | 2.75 | 89.51 | 63.454 | 0.366E-01 | 0.296E+00 | -0.147E+00 | -0.820E-01 | 0.243E-01 | -0.147E+00 | -0.574E-01 | 0.828E-02 |
| SD.113 | 3.00 | 89.45 | 63.447 | -0.504E-01 | 0.257E+00 | -0.139E+00 | -0.538E-01 | 0.660E-01 | -0.139E+00 | -0.405E-01 | 0.165E-01 |
| SD.114 | 3.25 | 89.39 | 63.439 | 0.197E+00 | 0.292E+00 | -0.113E+00 | -0.271E-01 | 0.102E+00 | -0.113E+00 | -0.219E-01 | 0.294E-01 |
| SD.115 | 3.50 | 89.34 | 63.433 | -0.166E+00 | 0.225E+00 | -0.117E+00 | -0.914E-02 | 0.134E+00 | -0.117E+00 | -0.149E-01 | 0.294E-01 |
| SD.116 | 3.75 | 89.29 | 63.427 | 0.244E+00 | 0.269E+00 | -0.928E-01 | 0.941E-02 | 0.161E+00 | -0.928E-01 | -0.363E-02 | 0.362E-01 |
| SD.117 | 4.00 | 89.26 | 63.422 | 0.124E+00 | 0.276E+00 | -0.792E-01 | 0.261E-01 | 0.185E+00 | -0.792E-01 | 0.701E-02 | 0.432E-01 |
| SD.118 | 4.25 | 89.21 | 63.417 | 0.195E-01 | 0.258E+00 | -0.734E-01 | 0.397E-01 | 0.207E+00 | -0.734E-01 | 0.143E-01 | 0.469E-01 |

ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
ILRT PROGRAM REPORT

STARTING DAY - 257 | STARTING TIME - 9:15: 0 | STARTING SCAN - SD.101 | ENDING SCAN - SD.118

ILRT RESULTS AFTER 4.25 HRS.

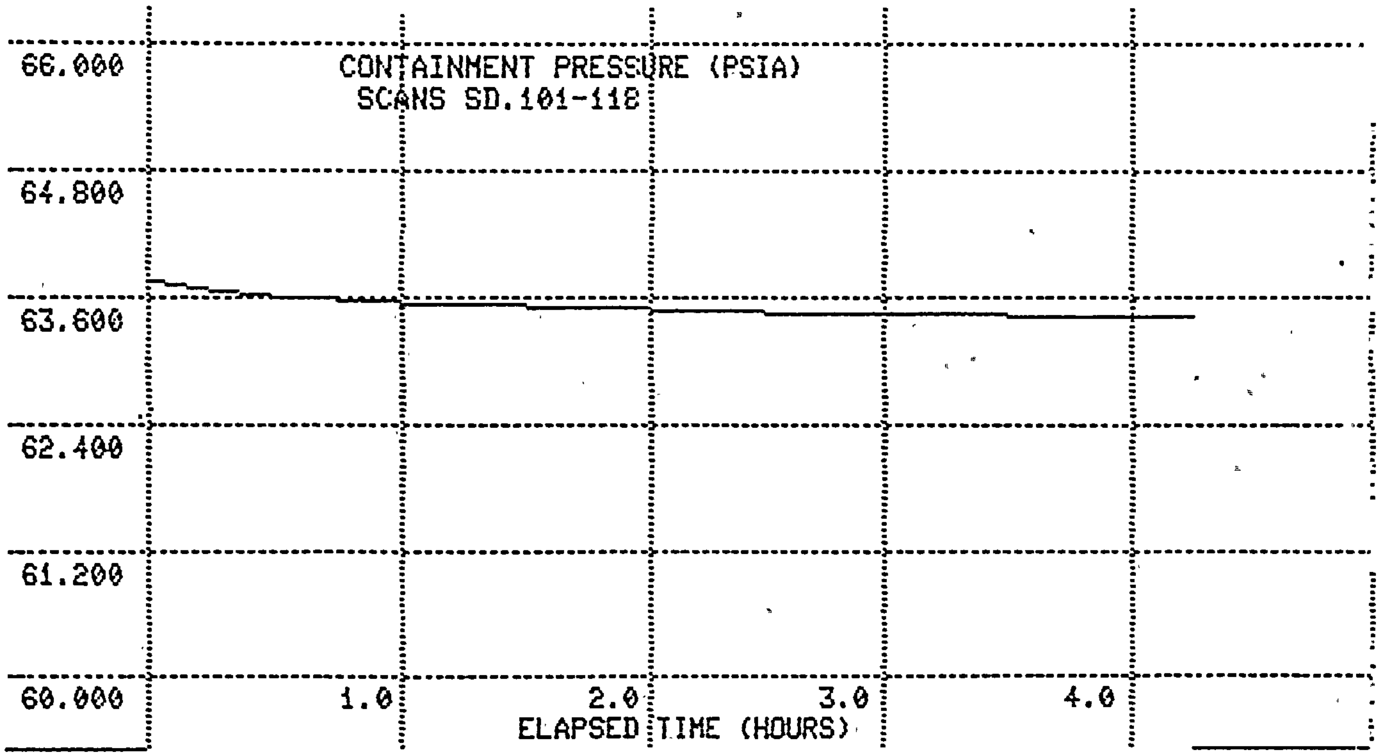
| POINT TO POINT | TOTAL TIME | | MASS PLOT | |
|---|---|---|---|---|
| | AVERAGE MEASURED LEAK RATES (WEIGHT PERCENT PER DAY) | | | |
| LEAK RATE | LEAK RATE | STD.DEV. | LEAK RATE | STD.DEV. |
| -0.735E-01 | -0.296E+00 | 0.231E+00 | -0.296E+00 | 0.231E+00 |

| POINT TO POINT | TOTAL TIME | | | MASS PLOT | | |
|---|---|---|---|---|---|---|
| | CALCULATED LEAK RATES (WEIGHT PERCENT PER DAY) | | | | | |
| LEAK RATE | LEAK RATE | STD.DEV. | UPPER CON. LIMIT | LEAK RATE | STD.DEV. | UPPER CON.LIMIT |
| 0.253E+00 | 0.397E-01 | 0.782E-01 | 0.206E+00 | 0.143E-01 | 0.186E-01 | 0.539E-01 |

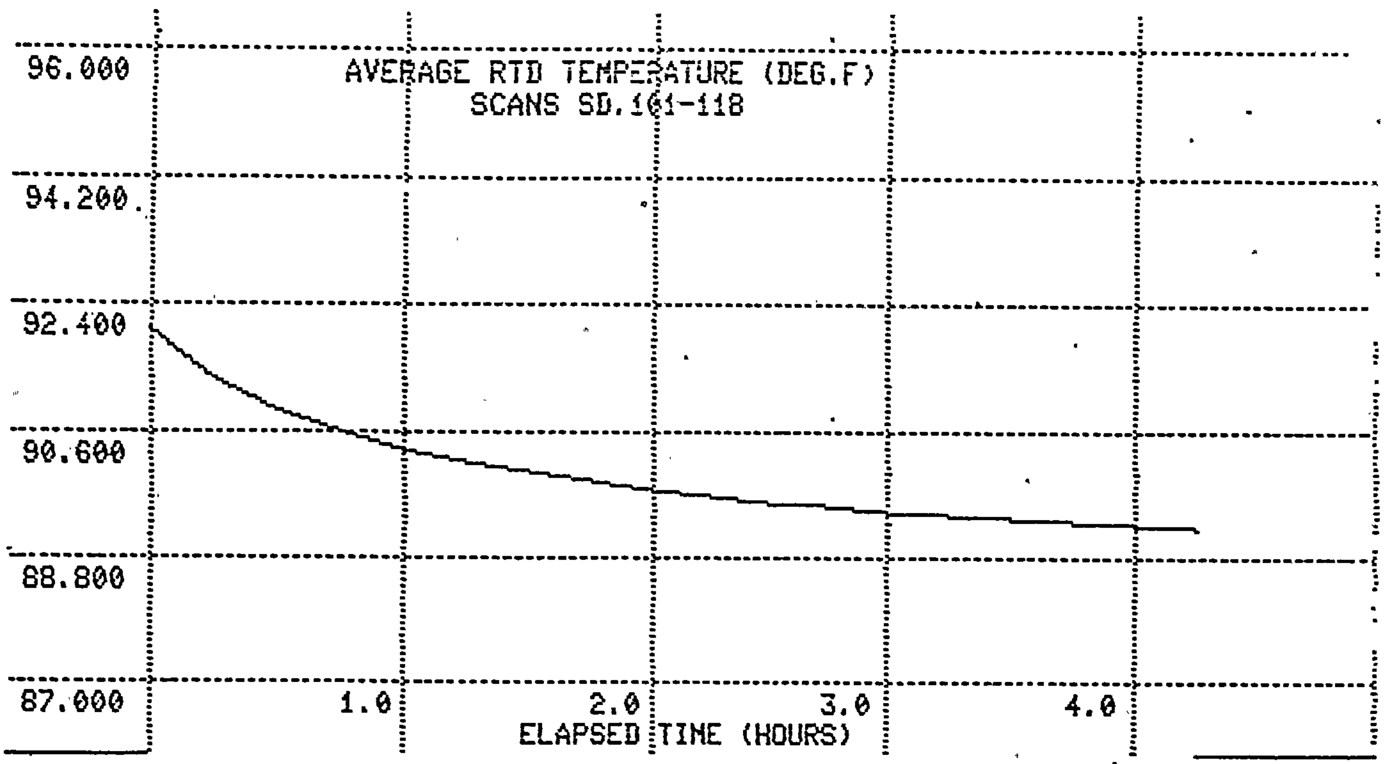
ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
RELATIVE HUMIDITY PROGRAM

| SCAN NO. | AVERAGE DEW POINT TEMPERATURE (F) | AVERAGE CONTAINMENT TEMPERATURE (F) | AVERAGE VAPOR PRESSURE (PSIA) | AVERAGE RELATIVE HUMIDITY (%) |
|---|---|---|---|---|
| SD.101 | 76.571 | 92.029 | 0.451 | 60.849 |
| SD.102 | 76.532 | 91.356 | 0.451 | 62.059 |
| SD.103 | 76.520 | 90.901 | 0.451 | 62.923 |
| SD.104 | 76.434 | 90.589 | 0.449 | 63.357 |
| SD.105 | 76.343 | 90.351 | 0.448 | 63.640 |
| SD.106 | 76.332 | 90.154 | 0.448 | 64.009 |
| SD.107 | 76.335 | 89.994 | 0.448 | 64.339 |
| SD.108 | 76.246 | 89.865 | 0.447 | 64.408 |
| SD.109 | 76.198 | 89.757 | 0.446 | 64.524 |
| SD.110 | 76.202 | 89.669 | 0.446 | 64.712 |
| SD.111 | 76.278 | 89.585 | 0.447 | 65.047 |
| SD.112 | 76.303 | 89.514 | 0.447 | 65.247 |
| SD.113 | 76.324 | 89.446 | 0.448 | 65.431 |
| SD.114 | 76.291 | 89.392 | 0.447 | 65.472 |
| SD.115 | 76.260 | 89.342 | 0.447 | 65.507 |
| SD.116 | 76.333 | 89.294 | 0.448 | 65.764 |
| SD.117 | 76.341 | 89.256 | 0.448 | 65.860 |
| SD.118 | 76.358 | 89.212 | 0.448 | 65.989 |

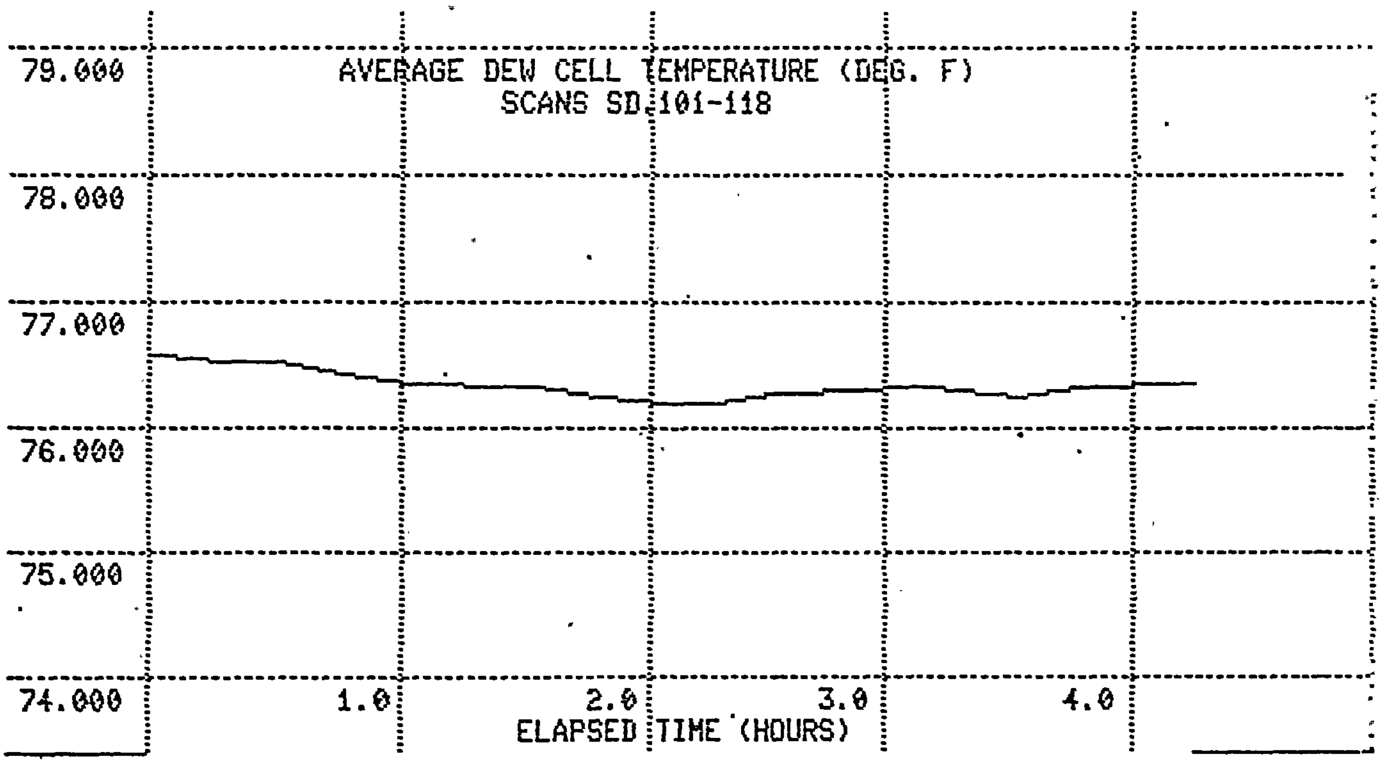
STABILIZATION



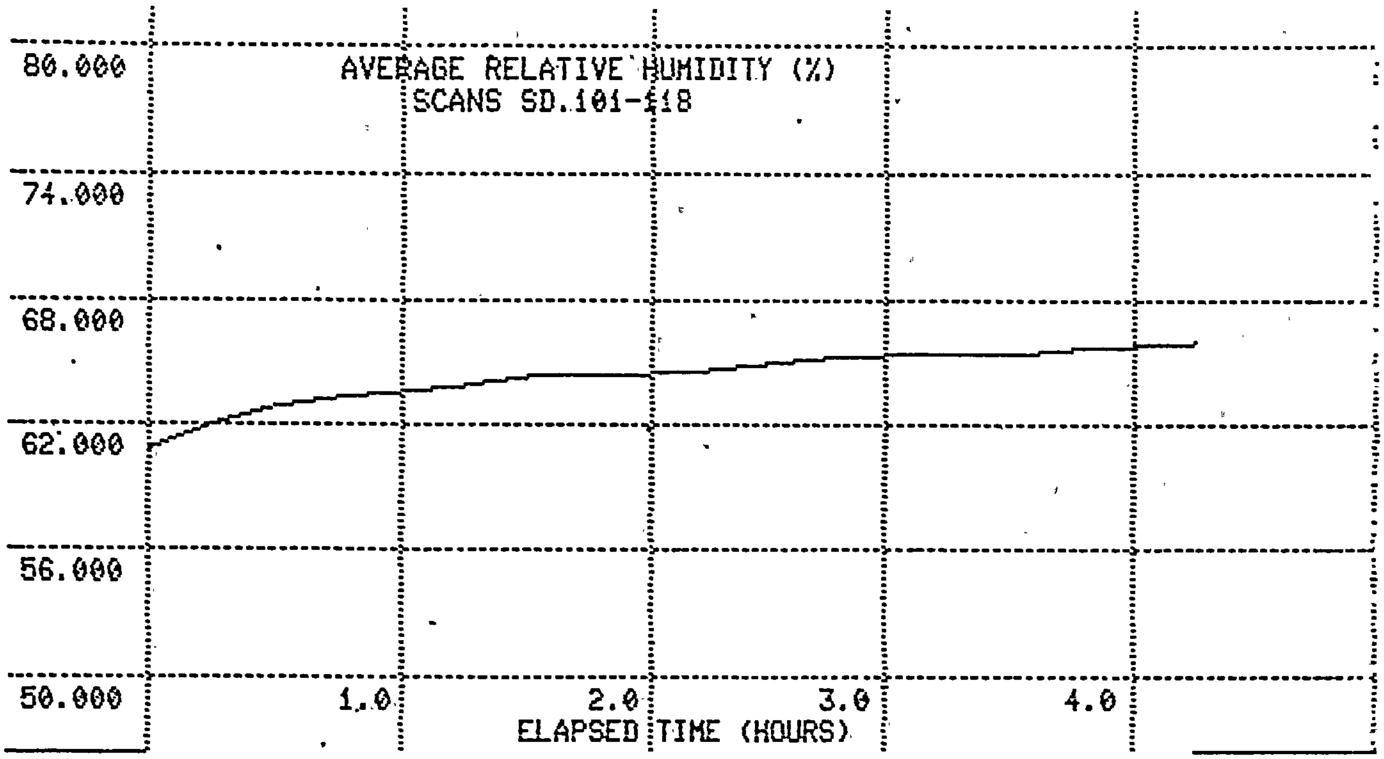
STABILIZATION



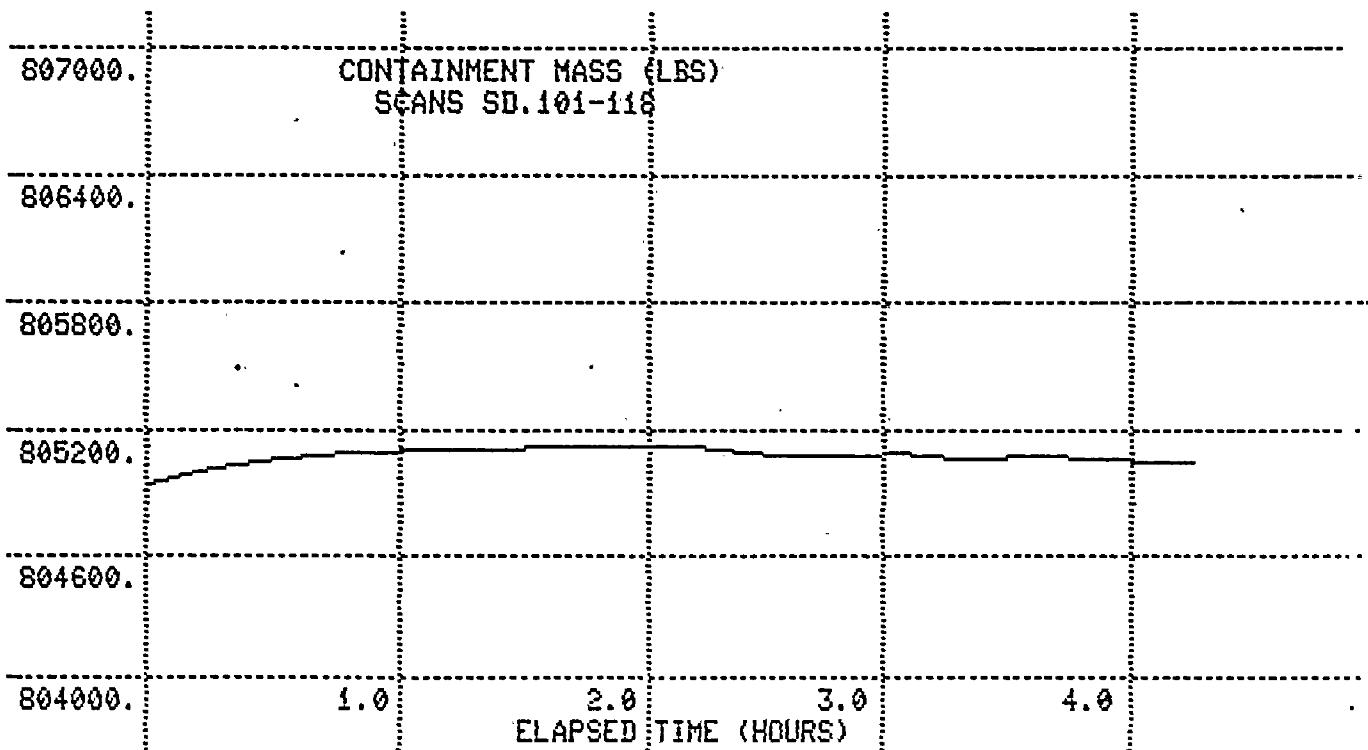
STABILIZATION



STABILIZATION



STABILIZATION



2.

# INTEGRATED LEAK RATE TEST

(ILRT)

[illegible]

ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
RAW DATA SUMMARY REPORT

| SCAN NO. | DATE | TIME | PRESS 1 | PRESS 2 | RTD #1 | RTD #2 | RTD #3 | RTD #4 | RTD #5 | RTD. #6 | RTD #7 | RTD #8 | RTD #9 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| SD.128 | 257 | 16: 0 | 62.567 | 62.664 | 94.259 | 94.665 | 94.306 | 94.914 | 94.625 | 94.413 | 93.048 | 92.836 | 93.042 |
| SD.129 | 257 | 16:15 | 62.565 | 62.663 | 94.236 | 94.638 | 94.270 | 94.876 | 94.592 | 94.374 | 93.025 | 92.814 | 93.019 |
| SD.130 | 257 | 16:30 | 62.563 | 62.661 | 94.206 | 94.606 | 94.252 | 94.839 | 94.557 | 94.322 | 92.996 | 92.796 | 92.996 |
| SD.131 | 257 | 16:45 | 62.562 | 62.659 | 94.160 | 94.566 | 94.213 | 94.815 | 94.531 | 94.276 | 92.968 | 92.776 | 92.965 |
| SD.132 | 257 | 17: 0 | 62.560 | 62.657 | 94.133 | 94.528 | 94.194 | 94.787 | 94.462 | 94.262 | 92.938 | 92.756 | 92.938 |
| SD.133 | 257 | 17:15 | 62.558 | 62.656 | 94.097 | 94.509 | 94.184 | 94.764 | 94.410 | 94.261 | 92.910 | 92.747 | 92.915 |
| SD.134 | 257 | 17:30 | 62.556 | 62.654 | 94.076 | 94.496 | 94.148 | 94.741 | 94.410 | 94.230 | 92.875 | 92.739 | 92.895 |
| SD.135 | 257 | 17:45 | 62.555 | 62.653 | 94.055 | 94.467 | 94.128 | 94.715 | 94.378 | 94.215 | 92.848 | 92.724 | 92.872 |
| SD.136 | 257 | 18: 0 | 62.553 | 62.651 | 94.029 | 94.448 | 94.093 | 94.696 | 94.372 | 94.140 | 92.831 | 92.697 | 92.845 |
| SD.137 | 257 | 18:15 | 62.551 | 62.649 | 94.020 | 94.433 | 94.085 | 94.679 | 94.374 | 94.142 | 92.823 | 92.681 | 92.834 |
| SD.138 | 257 | 18:30 | 62.549 | 62.648 | 94.001 | 94.383 | 94.056 | 94.674 | 94.313 | 94.119 | 92.813 | 92.666 | 92.819 |
| SD.139 | 257 | 18:45 | 62.548 | 62.646 | 93.978 | 94.363 | 94.032 | 94.645 | 94.323 | 94.123 | 92.796 | 92.643 | 92.800 |
| SD.140 | 257 | 19: 0 | 62.546 | 62.645 | 93.962 | 94.354 | 94.000 | 94.613 | 94.297 | 94.090 | 92.782 | 92.633 | 92.784 |
| SD.141 | 257 | 19:15 | 62.545 | 62.643 | 93.954 | 94.358 | 93.980 | 94.587 | 94.258 | 94.073 | 92.770 | 92.625 | 92.765 |
| SD.142 | 257 | 19:30 | 62.543 | 62.641 | 93.930 | 94.325 | 93.966 | 94.570 | 94.258 | 94.052 | 92.764 | 92.616 | 92.755 |
| SD.143 | 257 | 19:45 | 62.541 | 62.639 | 93.898 | 94.299 | 93.975 | 94.540 | 94.226 | 94.009 | 92.746 | 92.601 | 92.733 |
| SD.144 | 257 | 20: 0 | 62.539 | 62.638 | 93.876 | 94.291 | 93.936 | 94.512 | 94.210 | 94.010 | 92.724 | 92.582 | 92.717 |
| SD.145 | 257 | 20:15 | 62.538 | 62.636 | 93.870 | 94.270 | 93.928 | 94.512 | 94.187 | 94.023 | 92.713 | 92.573 | 92.712 |
| SD.146 | 257 | 20:30 | 62.537 | 62.635 | 93.882 | 94.253 | 93.904 | 94.531 | 94.180 | 93.974 | 92.686 | 92.559 | 92.697 |
| SD.147 | 257 | 20:45 | 62.535 | 62.633 | 93.875 | 94.227 | 93.887 | 94.526 | 94.175 | 93.949 | 92.672 | 92.544 | 92.681 |
| SD.148 | 257 | 21: 0 | 62.534 | 62.632 | 93.843 | 94.216 | 93.862 | 94.479 | 94.168 | 93.943 | 92.654 | 92.521 | 92.662 |
| SD.149 | 257 | 21:15 | 62.533 | 62.631 | 93.783 | 94.177 | 93.832 | 94.448 | 94.154 | 93.913 | 92.616 | 92.506 | 92.648 |
| SD.150 | 257 | 21:30 | 62.531 | 62.629 | 93.786 | 94.169 | 93.823 | 94.458 | 94.163 | 93.891 | 92.607 | 92.497 | 92.633 |
| SD.151 | 257 | 21:45 | 62.529 | 62.627 | 93.756 | 94.139 | 93.812 | 94.442 | 94.108 | 93.870 | 92.608 | 92.468 | 92.619 |
| SD.152 | 257 | 22: 0 | 62.528 | 62.626 | 93.775 | 94.142 | 93.800 | 94.426 | 94.123 | 93.832 | 92.602 | 92.419 | 92.599 |
| SD.153 | 257 | 22:15 | 62.526 | 62.625 | 93.768 | 94.097 | 93.782 | 94.421 | 94.085 | 93.812 | 92.578 | 92.392 | 92.584 |
| SD.154 | 257 | 22:30 | 62.525 | 62.623 | 93.733 | 94.099 | 93.759 | 94.412 | 94.072 | 93.794 | 92.559 | 92.384 | 92.567 |
| SD.155 | 257 | 22:45 | 62.523 | 62.622 | 93.705 | 94.078 | 93.751 | 94.395 | 94.068 | 93.772 | 92.561 | 92.385 | 92.559 |
| SD.156 | 257 | 23: 0 | 62.523 | 62.621 | 93.672 | 94.052 | 93.734 | 94.381 | 94.044 | 93.756 | 92.547 | 92.370 | 92.538 |
| SD.157 | 257 | 23:15 | 62.521 | 62.620 | 93.649 | 94.030 | 93.724 | 94.363 | 94.046 | 93.757 | 92.526 | 92.353 | 92.524 |
| SD.158 | 257 | 23:30 | 62.520 | 62.619 | 93.615 | 94.021 | 93.701 | 94.342 | 94.020 | 93.724 | 92.503 | 92.330 | 92.509 |
| SD.159 | 257 | 23:45 | 62.519 | 62.617 | 93.606 | 94.006 | 93.690 | 94.296 | 94.023 | 93.718 | 92.483 | 92.318 | 92.498 |
| SD.160 | 258 | 0: 0 | 62.518 | 62.616 | 93.582 | 93.995 | 93.672 | 94.311 | 93.988 | 93.695 | 92.483 | 92.305 | 92.494 |

ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
RAW DATA SUMMARY REPORT

| SCAN NO. | RTD #10 | RTD #11 | RTD #12 | RTD #13 | RTD #14 | RTD #15 | RTD #16 | RTD #17 | RTD #18 | RTD #19 | RTD #20 |
|---|---|---|---|---|---|---|---|---|---|---|---|
| SD.128 | 93.003 | 93.328 | 93.957 | 89.694 | 87.960 | 87.381 | 86.932 | 81.901 | 81.654 | 82.800 | 79.876 |
| SD.129 | 92.977 | 93.298 | 93.937 | 89.740 | 87.992 | 87.405 | 86.918 | 81.901 | 81.655 | 82.800 | 79.884 |
| SD.130 | 92.958 | 93.278 | 93.907 | 89.761 | 88.061 | 87.435 | 86.952 | 81.901 | 81.662 | 82.798 | 79.885 |
| SD.131 | 92.930 | 93.252 | 93.870 | 89.762 | 88.110 | 87.458 | 87.051 | 81.895 | 81.652 | 82.778 | 79.888 |
| SD.132 | 92.904 | 93.231 | 93.847 | 89.796 | 88.125 | 87.484 | 87.074 | 81.893 | 81.658 | 82.788 | 79.892 |
| SD.133 | 92.889 | 93.217 | 93.826 | 89.828 | 88.128 | 87.500 | 87.085 | 81.893 | 81.660 | 82.789 | 79.898 |
| SD.134 | 92.871 | 93.196 | 93.798 | 89.827 | 88.102 | 87.512 | 87.110 | 81.892 | 81.652 | 82.778 | 79.901 |
| SD.135 | 92.845 | 93.179 | 93.775 | 89.837 | 88.101 | 87.524 | 87.146 | 81.890 | 81.651 | 82.780 | 79.905 |
| SD.136 | 92.820 | 93.154 | 93.753 | 89.831 | 88.090 | 87.527 | 87.150 | 81.886 | 81.660 | 82.775 | 79.904 |
| SD.137 | 92.804 | 93.139 | 93.733 | 89.837 | 88.087 | 87.529 | 87.187 | 81.887 | 81.658 | 82.771 | 79.908 |
| SD.138 | 92.791 | 93.119 | 93.722 | 89.854 | 88.084 | 87.536 | 87.193 | 81.884 | 81.657 | 82.766 | 79.917 |
| SD.139 | 92.771 | 93.106 | 93.708 | 89.856 | 88.081 | 87.538 | 87.158 | 81.886 | 81.655 | 82.768 | 79.916 |
| SD.140 | 92.759 | 93.083 | 93.699 | 89.839 | 88.069 | 87.524 | 87.121 | 81.886 | 81.655 | 82.768 | 79.911 |
| SD.141 | 92.738 | 93.063 | 93.675 | 89.801 | 88.058 | 87.501 | 87.081 | 81.884 | 81.663 | 82.782 | 79.925 |
| SD.142 | 92.720 | 93.046 | 93.643 | 89.799 | 88.040 | 87.475 | 87.042 | 81.883 | 81.658 | 82.769 | 79.930 |
| SD.143 | 92.700 | 93.029 | 93.631 | 89.787 | 88.020 | 87.460 | 87.028 | 81.884 | 81.665 | 82.775 | 79.931 |
| SD.144 | 92.683 | 93.013 | 93.618 | 89.730 | 88.000 | 87.442 | 86.999 | 81.886 | 81.660 | 82.778 | 79.924 |
| SD.145 | 92.659 | 93.002 | 93.608 | 89.721 | 87.980 | 87.428 | 86.978 | 81.883 | 81.663 | 82.774 | 79.924 |
| SD.146 | 92.652 | 92.974 | 93.588 | 89.677 | 87.953 | 87.411 | 86.950 | 81.881 | 81.669 | 82.768 | 79.927 |
| SD.147 | 92.631 | 92.958 | 93.577 | 89.703 | 87.934 | 87.399 | 86.927 | 81.884 | 81.668 | 82.772 | 79.924 |
| SD.148 | 92.622 | 92.941 | 93.562 | 89.666 | 87.912 | 87.374 | 86.911 | 81.886 | 81.665 | 82.774 | 79.927 |
| SD.149 | 92.602 | 92.919 | 93.544 | 89.650 | 87.890 | 87.353 | 86.885 | 81.883 | 81.662 | 82.769 | 79.927 |
| SD.150 | 92.581 | 92.900 | 93.531 | 89.640 | 87.870 | 87.335 | 86.879 | 81.881 | 81.663 | 82.769 | 79.922 |
| SD.151 | 92.538 | 92.898 | 93.505 | 89.613 | 87.855 | 87.323 | 86.856 | 81.884 | 81.662 | 82.775 | 79.925 |
| SD.152 | 92.517 | 92.881 | 93.495 | 89.598 | 87.838 | 87.312 | 86.853 | 81.886 | 81.665 | 82.778 | 79.927 |
| SD.153 | 92.506 | 92.865 | 93.483 | 89.602 | 87.822 | 87.292 | 86.843 | 81.884 | 81.669 | 82.777 | 79.925 |
| SD.154 | 92.511 | 92.860 | 93.475 | 89.575 | 87.815 | 87.281 | 86.837 | 81.889 | 81.675 | 82.775 | 79.930 |
| SD.155 | 92.511 | 92.825 | 93.460 | 89.573 | 87.796 | 87.268 | 86.819 | 81.886 | 81.669 | 82.768 | 79.934 |
| SD.156 | 92.501 | 92.817 | 93.446 | 89.559 | 87.789 | 87.260 | 86.814 | 81.887 | 81.666 | 82.772 | 79.927 |
| SD.157 | 92.488 | 92.813 | 93.434 | 89.564 | 87.780 | 87.254 | 86.807 | 81.886 | 81.669 | 82.775 | 79.931 |
| SD.158 | 92.478 | 92.794 | 93.421 | 89.543 | 87.771 | 87.245 | 86.808 | 81.886 | 81.677 | 82.772 | 79.939 |
| SD.159 | 92.469 | 92.785 | 93.418 | 89.561 | 87.765 | 87.240 | 86.807 | 81.889 | 81.677 | 82.780 | 79.936 |
| SD.160 | 92.462 | 92.770 | 93.409 | 89.529 | 87.757 | 87.236 | 86.795 | 81.892 | 81.686 | 82.777 | 79.945 |

ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
RAW DATA SUMMARY REPORT

| SCAN NO. | RTD #21 | RTD #22 | RTD #23 | RTD #24 | DEW CELL #1 | DEW CELL #2 | DEW CELL #3 | DEW CELL #4 | DEW CELL #5 | DEW CELL #6 |
|---|---|---|---|---|---|---|---|---|---|---|
| SD.128 | 79.827 | 79.217 | 79.672 | 79.568 | 78.648 | 79.597 | 78.347 | 77.336 | 70.179 | 70.913 |
| SD.129 | 79.837 | 79.226 | 79.681 | 79.570 | 78.733 | 79.586 | 78.321 | 77.288 | 70.193 | 70.988 |
| SD.130 | 79.840 | 79.229 | 79.684 | 79.576 | 78.701 | 79.594 | 78.363 | 77.252 | 70.190 | 70.964 |
| SD.131 | 79.841 | 79.231 | 79.682 | 79.579 | 78.721 | 79.586 | 78.292 | 77.246 | 70.257 | 71.058 |
| SD.132 | 79.844 | 79.237 | 79.686 | 79.579 | 78.710 | 79.568 | 78.262 | 77.313 | 70.254 | 71.019 |
| SD.133 | 79.855 | 79.240 | 79.690 | 79.589 | 78.654 | 79.562 | 78.205 | 77.272 | 70.274 | 70.985 |
| SD.134 | 79.861 | 79.248 | 79.696 | 79.585 | 78.691 | 79.574 | 78.311 | 77.235 | 70.390 | 71.060 |
| SD.135 | 79.863 | 79.248 | 79.698 | 79.588 | 78.712 | 79.557 | 78.280 | 77.204 | 70.347 | 71.083 |
| SD.136 | 79.861 | 79.245 | 79.692 | 79.591 | 78.624 | 79.559 | 78.276 | 77.191 | 70.417 | 71.077 |
| SD.137 | 79.858 | 79.249 | 79.695 | 79.588 | 78.709 | 79.554 | 78.308 | 77.201 | 70.390 | 71.023 |
| SD.138 | 79.863 | 79.251 | 79.704 | 79.591 | 78.703 | 79.538 | 78.266 | 77.214 | 70.393 | 71.119 |
| SD.139 | 79.873 | 79.255 | 79.707 | 79.591 | 78.698 | 79.524 | 78.207 | 77.201 | 70.472 | 71.087 |
| SD.140 | 79.876 | 79.263 | 79.704 | 79.596 | 78.663 | 79.525 | 78.099 | 77.204 | 70.483 | 71.075 |
| SD.141 | 79.884 | 79.267 | 79.711 | 79.608 | 78.695 | 79.519 | 78.167 | 77.227 | 70.433 | 71.124 |
| SD.142 | 79.884 | 79.267 | 79.715 | 79.608 | 78.665 | 79.541 | 78.120 | 77.223 | 70.593 | 71.078 |
| SD.143 | 79.885 | 79.269 | 79.719 | 79.609 | 78.660 | 79.522 | 78.082 | 77.230 | 70.549 | 71.046 |
| SD.144 | 79.893 | 79.272 | 79.724 | 79.611 | 78.648 | 79.525 | 78.068 | 77.258 | 70.518 | 71.023 |
| SD.145 | 79.892 | 79.275 | 79.725 | 79.611 | 78.675 | 79.516 | 78.154 | 77.287 | 70.588 | 71.124 |
| SD.146 | 79.893 | 79.275 | 79.730 | 79.618 | 78.646 | 79.516 | 78.152 | 77.307 | 70.495 | 71.044 |
| SD.147 | 79.899 | 79.277 | 79.725 | 79.611 | 78.681 | 79.519 | 78.308 | 77.299 | 70.631 | 71.026 |
| SD.148 | 79.896 | 79.280 | 79.727 | 79.615 | 78.714 | 79.505 | 78.089 | 77.319 | 70.568 | 70.986 |
| SD.149 | 79.901 | 79.278 | 79.725 | 79.615 | 78.671 | 79.487 | 78.108 | 77.354 | 70.501 | 71.025 |
| SD.150 | 79.902 | 79.280 | 79.725 | 79.620 | 78.685 | 79.493 | 78.169 | 77.383 | 70.507 | 70.932 |
| SD.151 | 79.907 | 79.284 | 79.736 | 79.617 | 78.625 | 79.509 | 78.196 | 77.414 | 70.524 | 70.933 |
| SD.152 | 79.908 | 79.284 | 79.728 | 79.609 | 78.625 | 79.513 | 78.134 | 77.410 | 70.603 | 70.983 |
| SD.153 | 79.908 | 79.286 | 79.730 | 79.623 | 78.659 | 79.473 | 78.117 | 77.423 | 70.607 | 70.965 |
| SD.154 | 79.911 | 79.290 | 79.731 | 79.626 | 78.654 | 79.516 | 78.103 | 77.482 | 70.576 | 70.932 |
| SD.155 | 79.916 | 79.290 | 79.733 | 79.623 | 78.642 | 79.507 | 78.065 | 77.456 | 70.637 | 70.874 |
| SD.156 | 79.916 | 79.290 | 79.736 | 79.620 | 78.616 | 79.510 | 78.079 | 77.433 | 70.570 | 70.828 |
| SD.157 | 79.917 | 79.295 | 79.748 | 79.634 | 78.660 | 79.504 | 78.129 | 77.410 | 70.579 | 70.895 |
| SD.158 | 79.921 | 79.295 | 79.745 | 79.626 | 78.584 | 79.493 | 78.038 | 77.395 | 70.622 | 70.848 |
| SD.159 | 79.921 | 79.295 | 79.736 | 79.637 | 78.674 | 79.496 | 77.980 | 77.400 | 70.632 | 70.837 |
| SD.160 | 79.930 | 79.299 | 79.747 | 79.631 | 78.593 | 79.493 | 77.940 | 77.368 | 70.571 | 70.756 |

[illegible]

ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
ILRT PROGRAM REPORT

STARTING DAY - 257 STARTING TIME - 16: 0: 0 STARTING SCAN - SD.128 ENDING SCAN - SD.160

| | | | | POINT TO POINT | | TOTAL TIME | | | MASS PLOT | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| SCAN NO. | ELAPSED TIME (HR) | AVERAGE TEMP. (F) | AVERAGE PRESSURE (PSIA) | MEASURED LEAK RATE | CALCULATED LEAK RATE | MEASURED LEAK RATE | CALCULATED LEAK RATE (WEIGHT PERCENT PER DAY) | UPPER CONFIDENCE | MEASURED LEAK RATE | CALCULATED LEAK RATE | UPPER CONFIDENCE |
| SD.128 | 0.00 | 88.99 | 63.385 | | | | | | | | |
| SD.129 | 0.25 | 88.98 | 63.383 | 0.169E+00 | 0.169E+00 | 0.169E+00 | 0.169E+00 | 0.000E+00 | 0.169E+00 | 0.169E+00 | 0.000E+00 |
| SD.130 | 0.50 | 88.97 | 63.381 | 0.161E+00 | 0.161E+00 | 0.165E+00 | 0.165E+00 | 0.000E+00 | 0.165E+00 | 0.165E+00 | 0.000E+00 |
| SD.131 | 0.75 | 88.95 | 63.380 | -0.652E-01 | -0.290E-01 | 0.887E-01 | 0.101E+00 | 0.215E+00 | 0.889E-01 | 0.497E-01 | 0.606E+00 |
| SD.132 | 1.00 | 88.94 | 63.378 | 0.509E-01 | -0.824E-02 | 0.793E-01 | 0.735E-01 | 0.120E+00 | 0.794E-01 | 0.394E-01 | 0.126E+00 |
| SD.133 | 1.25 | 88.94 | 63.376 | 0.750E-01 | 0.184E-01 | 0.784E-01 | 0.626E-01 | 0.109E+00 | 0.784E-01 | 0.439E-01 | 0.848E-01 |
| SD.134 | 1.50 | 88.92 | 63.374 | 0.277E+00 | 0.140E+00 | 0.112E+00 | 0.756E-01 | 0.146E+00 | 0.112E+00 | 0.791E-01 | 0.128E+00 |
| SD.135 | 1.75 | 88.91 | 63.373 | -0.332E-01 | 0.655E-01 | 0.909E-01 | 0.742E-01 | 0.136E+00 | 0.908E-01 | 0.789E-01 | 0.112E+00 |
| SD.136 | 2.00 | 88.90 | 63.371 | -0.229E-02 | 0.323E-01 | 0.792E-01 | 0.689E-01 | 0.123E+00 | 0.793E-01 | 0.721E-01 | 0.969E-01 |
| SD.137 | 2.25 | 88.89 | 63.369 | 0.251E+00 | 0.107E+00 | 0.982E-01 | 0.731E-01 | 0.128E+00 | 0.983E-01 | 0.806E-01 | 0.102E+00 |
| SD.138 | 2.50 | 88.88 | 63.367 | 0.138E+00 | 0.119E+00 | 0.102E+00 | 0.777E-01 | 0.132E+00 | 0.102E+00 | 0.884E-01 | 0.107E+00 |
| SD.139 | 2.75 | 88.87 | 63.366 | -0.893E-01 | 0.552E-01 | 0.847E-01 | 0.756E-01 | 0.126E+00 | 0.848E-01 | 0.839E-01 | 0.993E-01 |
| SD.140 | 3.00 | 88.86 | 63.364 | -0.504E-01 | 0.199E-01 | 0.735E-01 | 0.709E-01 | 0.118E+00 | 0.735E-01 | 0.765E-01 | 0.914E-01 |
| SD.141 | 3.25 | 88.85 | 63.363 | 0.607E-01 | 0.240E-01 | 0.725E-01 | 0.673E-01 | 0.111E+00 | 0.726E-01 | 0.713E-01 | 0.848E-01 |
| SD.142 | 3.50 | 88.84 | 63.361 | 0.515E-02 | 0.132E-01 | 0.679E-01 | 0.635E-01 | 0.105E+00 | 0.679E-01 | 0.666E-01 | 0.792E-01 |
| SD.143 | 3.75 | 88.82 | 63.359 | 0.744E-02 | 0.543E-02 | 0.639E-01 | 0.596E-01 | 0.991E-01 | 0.638E-01 | 0.618E-01 | 0.737E-01 |
| SD.144 | 4.00 | 88.81 | 63.357 | 0.292E-01 | 0.442E-02 | 0.616E-01 | 0.562E-01 | 0.939E-01 | 0.616E-01 | 0.578E-01 | 0.689E-01 |
| SD.145 | 4.25 | 88.80 | 63.356 | 0.175E+00 | 0.353E-01 | 0.683E-01 | 0.549E-01 | 0.919E-01 | 0.684E-01 | 0.574E-01 | 0.672E-01 |
| SD.146 | 4.50 | 88.79 | 63.355 | -0.119E+00 | 0.412E-03 | 0.579E-01 | 0.519E-01 | 0.875E-01 | 0.578E-01 | 0.539E-01 | 0.632E-01 |
| SD.147 | 4.75 | 88.78 | 63.353 | 0.292E+00 | 0.518E-01 | 0.702E-01 | 0.519E-01 | 0.876E-01 | 0.703E-01 | 0.553E-01 | 0.637E-01 |
| SD.148 | 5.00 | 88.77 | 63.352 | -0.316E+00 | -0.181E-01 | 0.509E-01 | 0.484E-01 | 0.828E-01 | 0.509E-01 | 0.510E-01 | 0.597E-01 |
| SD.149 | 5.25 | 88.75 | 63.351 | -0.153E+00 | -0.481E-01 | 0.412E-01 | 0.438E-01 | 0.771E-01 | 0.412E-01 | 0.451E-01 | 0.549E-01 |
| SD.150 | 5.50 | 88.74 | 63.349 | 0.195E+00 | -0.142E-01 | 0.482E-01 | 0.412E-01 | 0.736E-01 | 0.482E-01 | 0.423E-01 | 0.516E-01 |
| SD.151 | 5.75 | 88.73 | 63.347 | 0.629E-01 | -0.663E-02 | 0.488E-01 | 0.391E-01 | 0.709E-01 | 0.488E-01 | 0.409E-01 | 0.496E-01 |
| SD.152 | 6.00 | 88.72 | 63.346 | -0.217E-01 | -0.132E-01 | 0.459E-01 | 0.369E-01 | 0.681E-01 | 0.459E-01 | 0.389E-01 | 0.471E-01 |
| SD.153 | 6.25 | 88.71 | 63.344 | 0.675E-01 | -0.543E-02 | 0.468E-01 | 0.352E-01 | 0.661E-01 | 0.468E-01 | 0.378E-01 | 0.453E-01 |
| SD.154 | 6.50 | 88.70 | 63.343 | -0.801E-02 | -0.951E-02 | 0.447E-01 | 0.335E-01 | 0.640E-01 | 0.447E-01 | 0.363E-01 | 0.434E-01 |
| SD.155 | 6.75 | 88.69 | 63.341 | 0.618E-01 | -0.323E-02 | 0.454E-01 | 0.322E-01 | 0.625E-01 | 0.454E-01 | 0.352E-01 | 0.419E-01 |
| SD.156 | 7.00 | 88.68 | 63.341 | -0.224E+00 | -0.364E-01 | 0.357E-01 | 0.297E-01 | 0.594E-01 | 0.358E-01 | 0.327E-01 | 0.394E-01 |
| SD.157 | 7.25 | 88.67 | 63.339 | 0.309E+00 | 0.422E-02 | 0.451E-01 | 0.289E-01 | 0.587E-01 | 0.451E-01 | 0.326E-01 | 0.388E-01 |
| SD.158 | 7.50 | 88.66 | 63.338 | -0.214E+00 | -0.260E-01 | 0.364E-01 | 0.270E-01 | 0.565E-01 | 0.365E-01 | 0.306E-01 | 0.367E-01 |
| SD.159 | 7.75 | 88.65 | 63.337 | 0.132E-01 | -0.250E-01 | 0.357E-01 | 0.253E-01 | 0.545E-01 | 0.357E-01 | 0.291E-01 | 0.349E-01 |
| SD.160 | 8.00 | 88.64 | 63.336 | -0.973E-01 | -0.372E-01 | 0.316E-01 | 0.234E-01 | 0.521E-01 | 0.316E-01 | 0.270E-01 | 0.328E-01 |

ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
ILRT PROGRAM REPORT

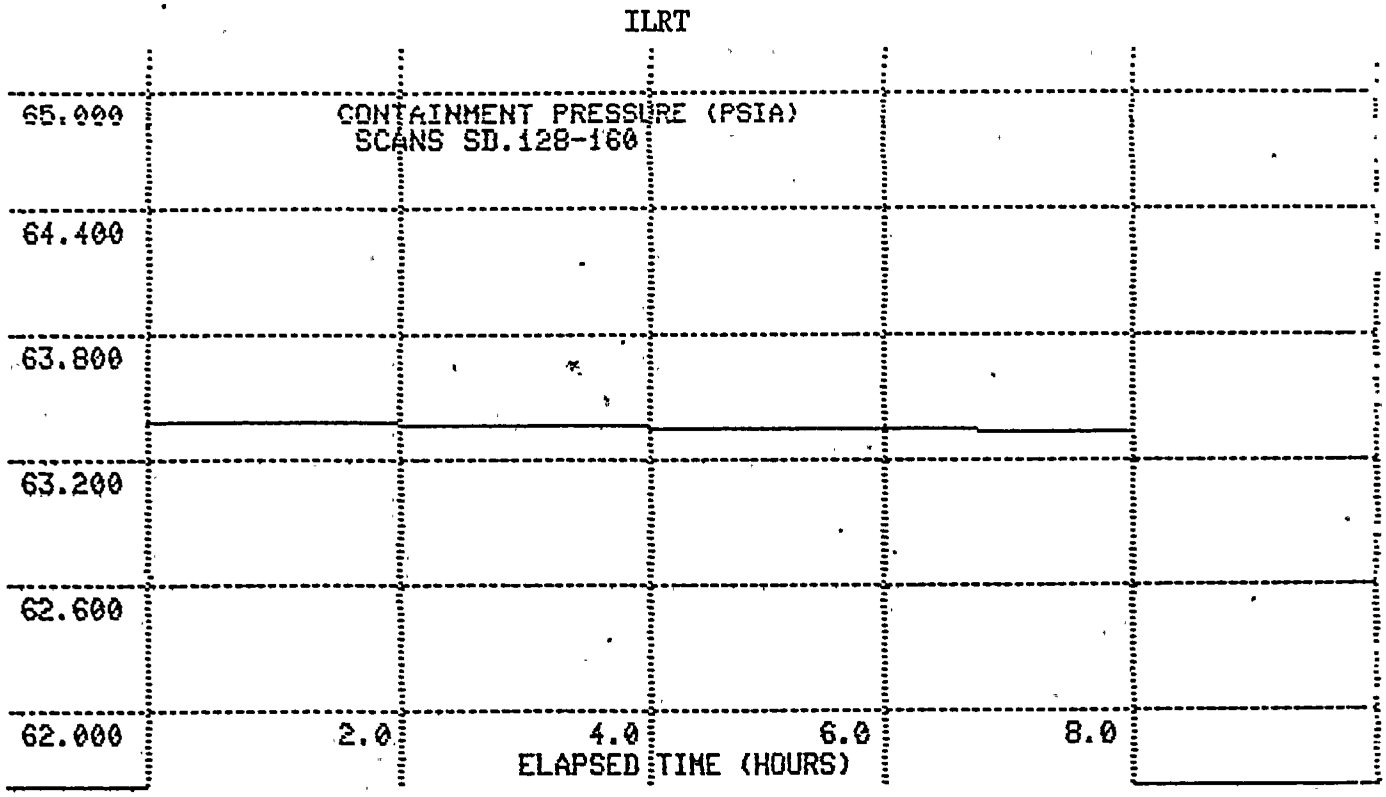
STARTING DAY - 257 STARTING TIME - 16: 0: 0 STARTING SCAN - SD.128 ENDING SCAN - SD.160

ILRT RESULTS AFTER 8.00 HRS.

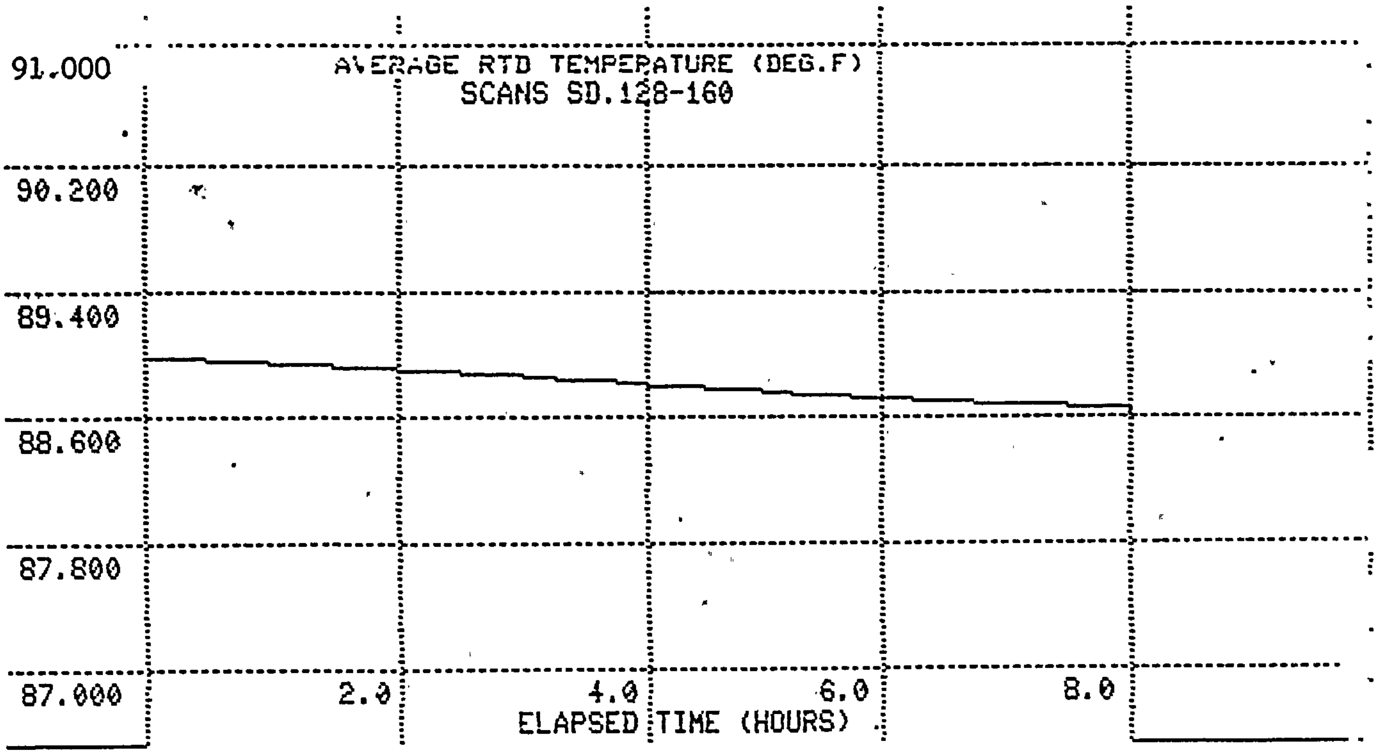
| POINT TO POINT | TOTAL TIME | | | MASS PLOT | | |
|---|---|---|---|---|---|---|
| | AVERAGE MEASURED LEAK RATES (WEIGHT PERCENT PER DAY) | | | | | |
| LEAK RATE | LEAK RATE | STD.DEV. | | LEAK RATE | STD.DEV. | |
| 0.315E-01 | 0.700E-01 | 0.348E-01 | | 0.700E-01 | 0.349E-01 | |
| | CALCULATED LEAK RATES (WEIGHT PERCENT PER DAY) | | | | | |
| LEAK RATE | LEAK RATE | STD.DEV. | UPPER CON. LIMIT | LEAK RATE | STD.DEV. | UPPER CON.LIMIT |
| -0.372E-01 | 0.234E-01 | 0.141E-01 | 0.521E-01 | 0.270E-01 | 0.344E-02 | 0.340E-01 |

ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
RELATIVE HUMIDITY PROGRAM

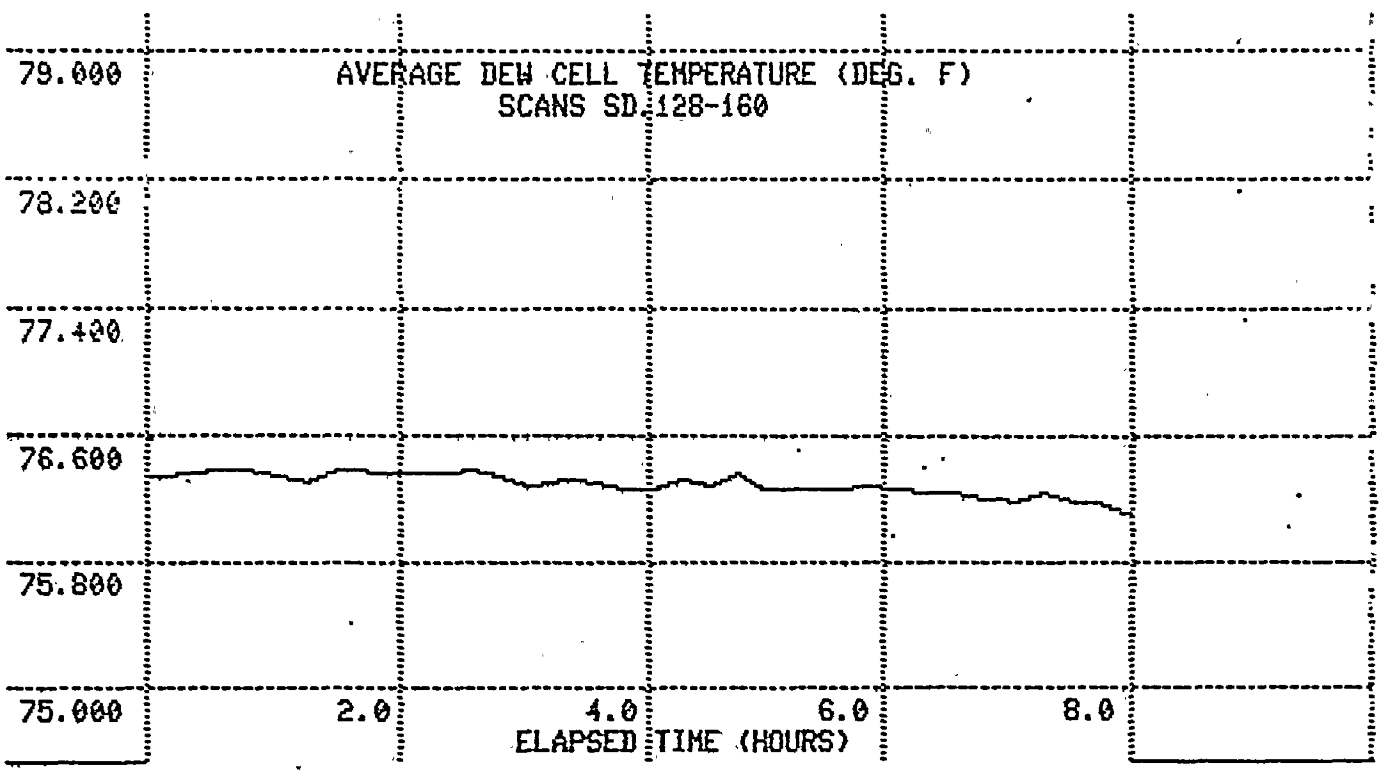
| SCAN NO. | AVERAGE DEW POINT TEMPERATURE (F) | AVERAGE CONTAINMENT TEMPERATURE (F) | AVERAGE VAPOR PRESSURE (PSIA) | AVERAGE RELATIVE HUMIDITY (%) |
|---|---|---|---|---|
| SD.128 | 76.342 | 88.987 | 0.448 | 66.425 |
| SD.129 | 76.365 | 88.976 | 0.448 | 66.498 |
| SD.130 | 76.371 | 88.966 | 0.449 | 66.532 |
| SD.131 | 76.372 | 88.954 | 0.449 | 66.560 |
| SD.132 | 76.344 | 88.943 | 0.448 | 66.520 |
| SD.133 | 76.301 | 88.935 | 0.447 | 66.442 |
| SD.134 | 76.375 | 88.924 | 0.449 | 66.627 |
| SD.135 | 76.369 | 88.914 | 0.448 | 66.637 |
| SD.136 | 76.353 | 88.898 | 0.448 | 66.634 |
| SD.137 | 76.362 | 88.894 | 0.448 | 66.662 |
| SD.138 | 76.370 | 88.883 | 0.448 | 66.704 |
| SD.139 | 76.334 | 88.874 | 0.448 | 66.642 |
| SD.140 | 76.281 | 88.861 | 0.447 | 66.553 |
| SD.141 | 76.327 | 88.849 | 0.448 | 66.680 |
| SD.142 | 76.293 | 88.837 | 0.447 | 66.631 |
| SD.143 | 76.264 | 88.823 | 0.447 | 66.595 |
| SD.144 | 76.250 | 88.809 | 0.447 | 66.594 |
| SD.145 | 76.319 | 88.801 | 0.448 | 66.761 |
| SD.146 | 76.290 | 88.790 | 0.447 | 66.722 |
| SD.147 | 76.354 | 88.780 | 0.448 | 66.884 |
| SD.148 | 76.255 | 88.766 | 0.447 | 66.693 |
| SD.149 | 76.265 | 88.748 | 0.447 | 66.756 |
| SD.150 | 76.265 | 88.741 | 0.447 | 66.770 |
| SD.151 | 76.270 | 88.726 | 0.447 | 66.812 |
| SD.152 | 76.260 | 88.718 | 0.447 | 66.809 |
| SD.153 | 76.247 | 88.706 | 0.447 | 66.804 |
| SD.154 | 76.237 | 88.698 | 0.447 | 66.798 |
| SD.155 | 76.201 | 88.689 | 0.446 | 66.738 |
| SD.156 | 76.190 | 88.677 | 0.446 | 66.736 |
| SD.157 | 76.236 | 88.671 | 0.446 | 66.851 |
| SD.158 | 76.172 | 88.659 | 0.446 | 66.736 |
| SD.159 | 76.158 | 88.652 | 0.445 | 66.719 |
| SD.160 | 76.105 | 88.645 | 0.445 | 66.617 |


ILRT
CONTAINMENT PRESSURE (PSIA)
SCANS SD.128-160
65.000
64.400
63.800
63.200
62.600
62.000
2.0
4.0
6.0
8.0
ELAPSED TIME (HOURS)

ILRT

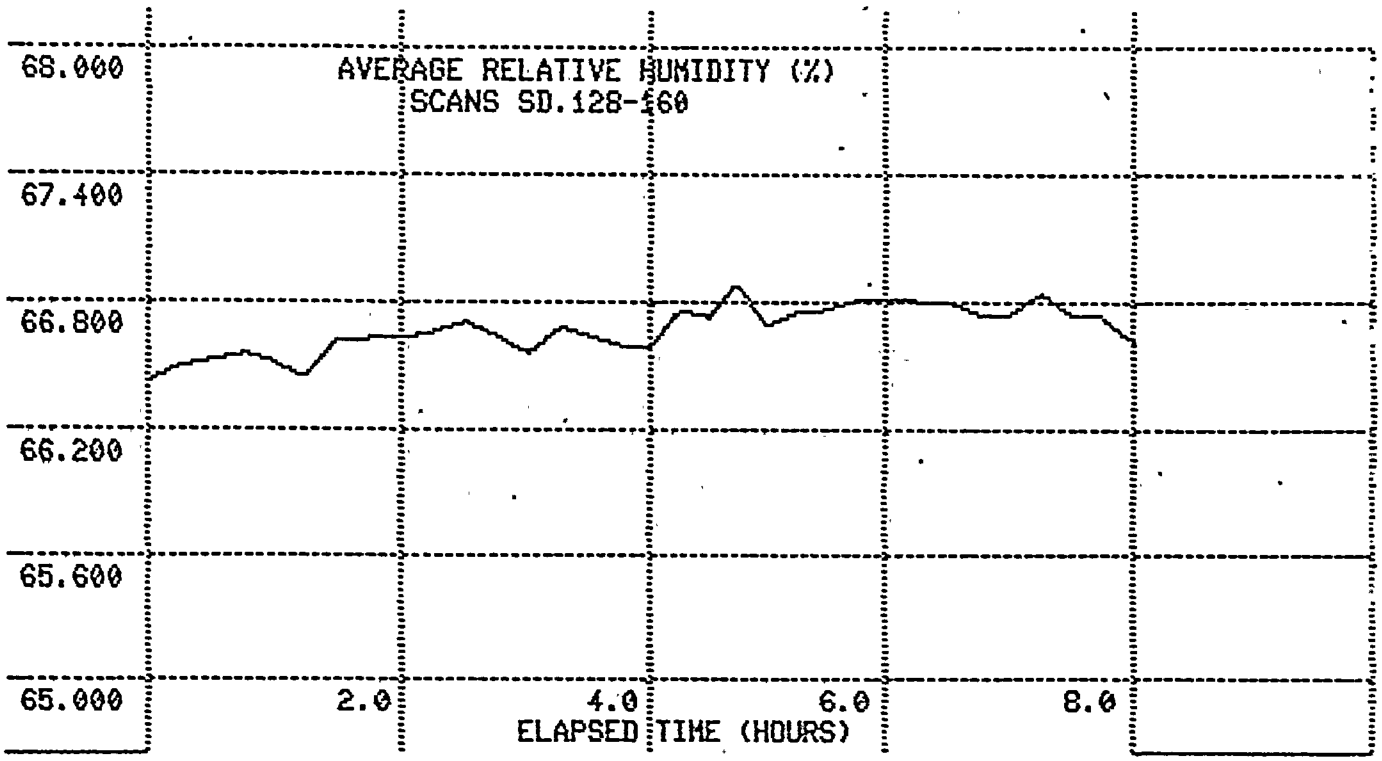

AVERAGE RTD TEMPERATURE (DEG.F)
SCANS SD.128-160
91.000
90.200
89.400
88.600
87.800
87.000
2.0
4.0
6.0
8.0
ELAPSED TIME (HOURS)

ILRT

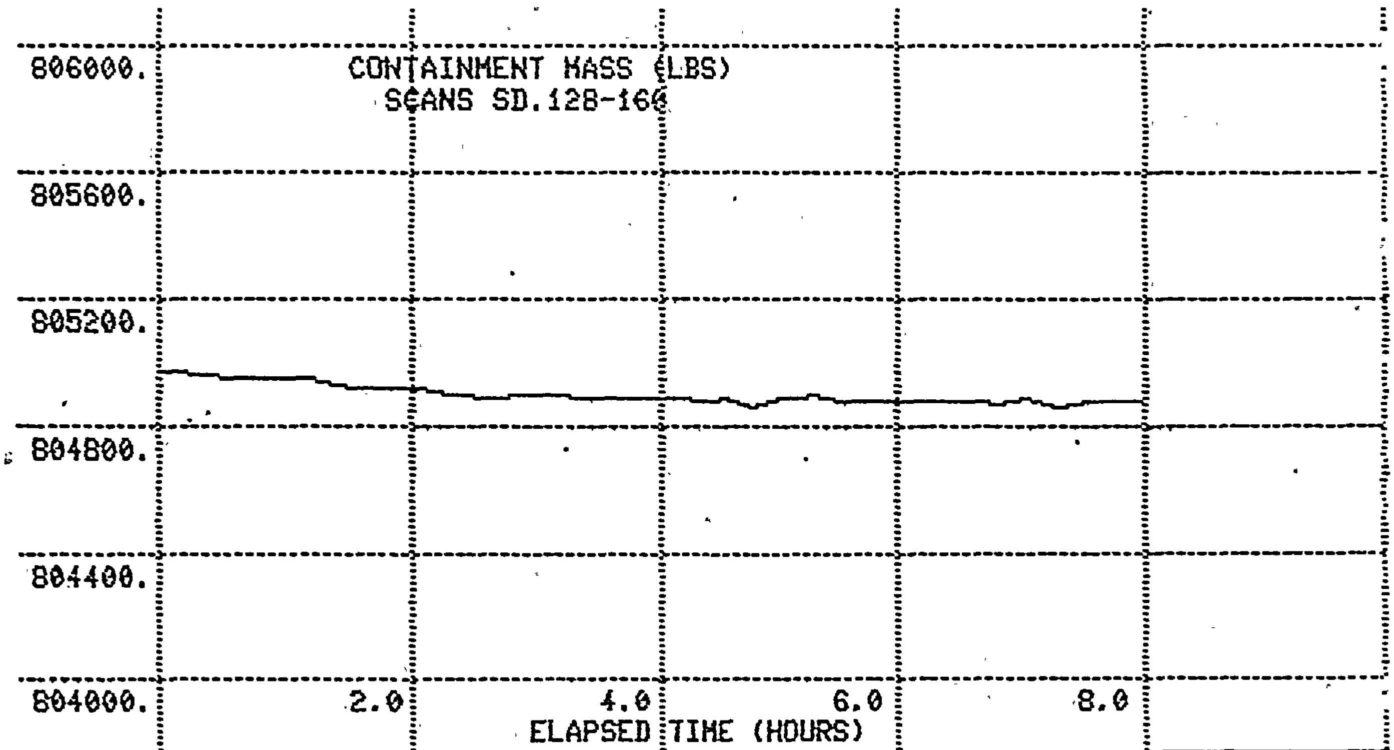

AVERAGE DEW CELL TEMPERATURE (DEG. F)
SCANS SD 128-160
79.000
78.200
77.400
76.600
75.800
75.000
2.0
4.0
6.0
8.0
ELAPSED TIME (HOURS)

[illegible]

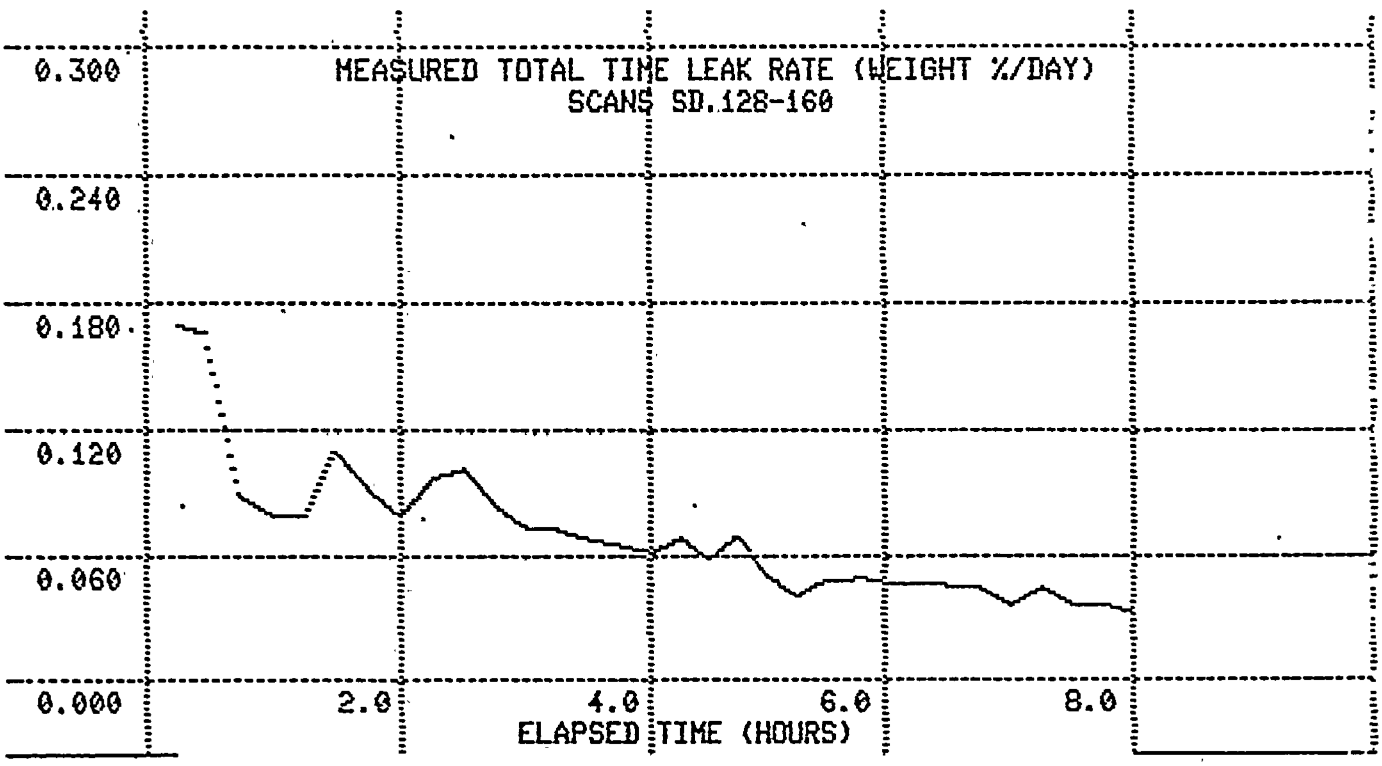
ILRT


AVERAGE RELATIVE HUMIDITY (%)
SCANS SD.128-160
68.000
67.400
66.800
66.200
65.600
65.000
2.0
4.0
6.0
8.0
ELAPSED TIME (HOURS)

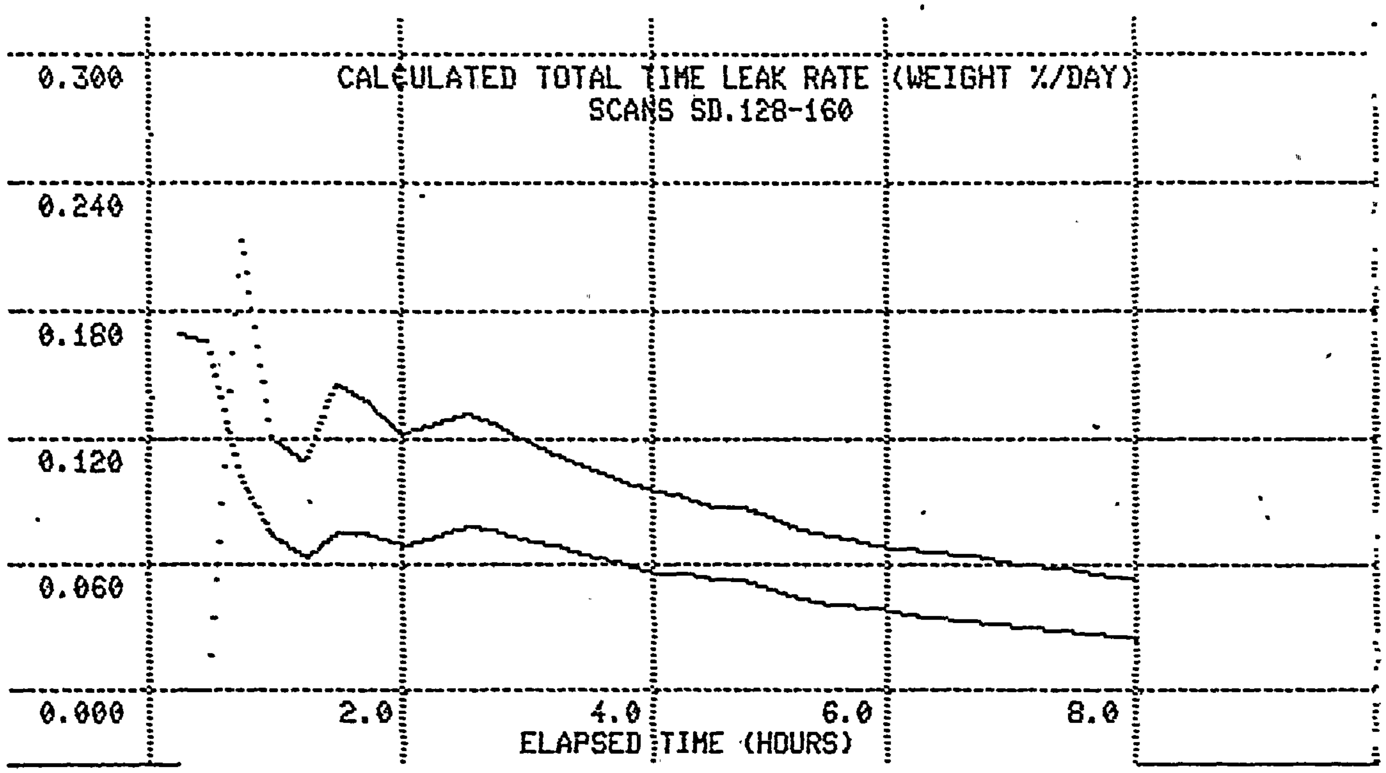
ILRT


CONTAINMENT MASS (LBS)
SCANS SD.128-160
806000.
805600.
805200.
804800.
804400.
804000.
2.0
4.0
6.0
8.0
ELAPSED TIME (HOURS)

ILRT


MEASURED TOTAL TIME LEAK RATE (WEIGHT %/DAY)
SCANS SD.128-160
0.300
0.240
0.180
0.120
0.060
0.000
2.0
4.0
6.0
8.0
ELAPSED TIME (HOURS)

ILRT


CALCULATED TOTAL TIME LEAK RATE (WEIGHT %/DAY)
SCANS SD.128-160
0.300
0.240
0.180
0.120
0.060
0.000
2.0
4.0
6.0
8.0
ELAPSED TIME (HOURS)

# 3.

# CONTROLLED LEAK RATE TEST

# (CLRT)

[illegible]

ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
RAW DATA SUMMARY REPORT

| SCAN NO. | DATE | TIME | PRESS 1 | PRESS 2 | RTD #1 | RTD #2 | RTD #3 | RTD #4 | RTD #5 | RTD #6 | RTD #7 | RTD #8 | RTD #9 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| SD.171 | 258 | 2:45 | 62.503 | 62.598 | 93.441 | 93.791 | 93.486 | 94.128 | 93.821 | 93.556 | 92.369 | 92.181 | 92.387 |
| SD.172 | 258 | 3: 0 | 62.502 | 62.597 | 93.406 | 93.774 | 93.478 | 94.117 | 93.835 | 93.522 | 92.344 | 92.164 | 92.375 |
| SD.173 | 258 | 3:15 | 62.502 | 62.595 | 93.379 | 93.757 | 93.452 | 94.116 | 93.821 | 93.507 | 92.323 | 92.155 | 92.369 |
| SD.174 | 258 | 3:30 | 62.500 | 62.594 | 93.357 | 93.748 | 93.457 | 94.094 | 93.814 | 93.501 | 92.308 | 92.140 | 92.359 |
| SD.175 | 258 | 3:45 | 62.497 | 62.592 | 93.347 | 93.742 | 93.441 | 94.052 | 93.800 | 93.476 | 92.298 | 92.149 | 92.340 |
| SD.176 | 258 | 4: 0 | 62.495 | 62.590 | 93.342 | 93.718 | 93.432 | 93.977 | 93.775 | 93.467 | 92.297 | 92.164 | 92.335 |
| SD.177 | 258 | 4:15 | 62.496 | 62.589 | 93.335 | 93.713 | 93.429 | 93.949 | 93.780 | 93.458 | 92.286 | 92.147 | 92.329 |
| SD.178 | 258 | 4:30 | 62.494 | 62.587 | 93.333 | 93.699 | 93.399 | 93.910 | 93.783 | 93.428 | 92.277 | 92.109 | 92.308 |
| SD.179 | 258 | 4:45 | 62.493 | 62.586 | 93.362 | 93.702 | 93.357 | 93.895 | 93.754 | 93.399 | 92.268 | 92.105 | 92.294 |
| SD.180 | 258 | 5: 0 | 62.491 | 62.584 | 93.336 | 93.692 | 93.351 | 93.876 | 93.711 | 93.379 | 92.253 | 92.096 | 92.282 |
| SD.181 | 258 | 5:15 | 62.490 | 62.583 | 93.298 | 93.678 | 93.353 | 93.855 | 93.722 | 93.383 | 92.236 | 92.086 | 92.268 |
| SD.182 | 258 | 5:30 | 62.488 | 62.581 | 93.292 | 93.646 | 93.306 | 93.827 | 93.682 | 93.374 | 92.213 | 92.077 | 92.254 |
| SD.183 | 258 | 5:45 | 62.487 | 62.580 | 93.287 | 93.621 | 93.324 | 93.843 | 93.692 | 93.376 | 92.211 | 92.067 | 92.265 |
| SD.184 | 258 | 6: 0 | 62.485 | 62.578 | 93.264 | 93.602 | 93.306 | 93.832 | 93.681 | 93.368 | 92.207 | 92.039 | 92.260 |
| SD.185 | 258 | 6:15 | 62.483 | 62.577 | 93.223 | 93.617 | 93.273 | 93.806 | 93.669 | 93.335 | 92.213 | 92.030 | 92.244 |
| SD.186 | 258 | 6:30 | 62.482 | 62.575 | 93.200 | 93.656 | 93.270 | 93.794 | 93.656 | 93.322 | 92.193 | 92.034 | 92.225 |
| SD.187 | 258 | 6:45 | 62.480 | 62.574 | 93.212 | 93.605 | 93.275 | 93.775 | 93.646 | 93.315 | 92.186 | 92.019 | 92.205 |
| SD.188 | 258 | 7: 0 | 62.478 | 62.573 | 93.183 | 93.554 | 93.234 | 93.788 | 93.634 | 93.273 | 92.175 | 91.993 | 92.193 |
| SD.189 | 258 | 7:15 | 62.477 | 62.571 | 93.168 | 93.537 | 93.220 | 93.782 | 93.612 | 93.272 | 92.166 | 92.001 | 92.187 |
| SD.190 | 258 | 7:30 | 62.476 | 62.570 | 93.162 | 93.525 | 93.212 | 93.766 | 93.608 | 93.270 | 92.143 | 92.009 | 92.170 |

ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
RAW DATA SUMMARY REPORT

| SCAN NO. | RTD #10 | RTD #11 | RTD #12 | RTD #13 | RTD #14 | RTD #15 | RTD #16 | RTD #17 | RTD #18 | RTD #19 | RTD #20 |
|---|---|---|---|---|---|---|---|---|---|---|---|
| SD.171 | 92.314 | 92.672 | 93.235 | 89.498 | 87.695 | 87.191 | 86.711 | 81.900 | 81.691 | 82.783 | 79.957 |
| SD.172 | 92.308 | 92.662 | 93.222 | 89.479 | 87.690 | 87.188 | 86.692 | 81.903 | 81.684 | 82.774 | 79.945 |
| SD.173 | 92.309 | 92.649 | 93.206 | 89.495 | 87.687 | 87.185 | 86.702 | 81.903 | 81.687 | 82.782 | 79.957 |
| SD.174 | 92.311 | 92.640 | 93.202 | 89.486 | 87.689 | 87.187 | 86.692 | 81.910 | 81.697 | 82.789 | 79.957 |
| SD.175 | 92.292 | 92.633 | 93.197 | 89.494 | 87.687 | 87.182 | 86.717 | 81.912 | 81.698 | 82.792 | 79.957 |
| SD.176 | 92.279 | 92.620 | 93.191 | 89.457 | 87.681 | 87.173 | 86.695 | 81.907 | 81.697 | 82.792 | 79.962 |
| SD.177 | 92.251 | 92.613 | 93.165 | 89.444 | 87.674 | 87.171 | 86.688 | 81.906 | 81.698 | 82.789 | 79.965 |
| SD.178 | 92.211 | 92.601 | 93.154 | 89.466 | 87.674 | 87.173 | 86.677 | 81.912 | 81.698 | 82.788 | 79.966 |
| SD.179 | 92.215 | 92.581 | 93.138 | 89.469 | 87.667 | 87.170 | 86.680 | 81.910 | 81.689 | 82.785 | 79.954 |
| SD.180 | 92.215 | 92.572 | 93.121 | 89.489 | 87.661 | 87.161 | 86.657 | 81.910 | 81.695 | 82.775 | 79.953 |
| SD.181 | 92.211 | 92.569 | 93.125 | 89.497 | 87.661 | 87.164 | 86.654 | 81.916 | 81.700 | 82.794 | 79.960 |
| SD.182 | 92.181 | 92.558 | 93.122 | 89.492 | 87.658 | 87.161 | 86.647 | 81.916 | 81.707 | 82.791 | 79.959 |
| SD.183 | 92.205 | 92.552 | 93.125 | 89.497 | 87.655 | 87.161 | 86.648 | 81.919 | 81.704 | 82.788 | 79.966 |
| SD.184 | 92.210 | 92.540 | 93.116 | 89.491 | 87.648 | 87.150 | 86.634 | 81.919 | 81.712 | 82.792 | 79.962 |
| SD.185 | 92.192 | 92.532 | 93.100 | 89.460 | 87.643 | 87.147 | 86.622 | 81.921 | 81.706 | 82.797 | 79.954 |
| SD.186 | 92.178 | 92.518 | 93.086 | 89.459 | 87.638 | 87.142 | 86.624 | 81.922 | 81.698 | 82.791 | 79.954 |
| SD.187 | 92.175 | 92.507 | 93.077 | 89.468 | 87.635 | 87.144 | 86.616 | 81.924 | 81.700 | 82.788 | 79.951 |
| SD.188 | 92.153 | 92.500 | 93.072 | 89.447 | 87.631 | 87.138 | 86.615 | 81.927 | 81.704 | 82.803 | 79.956 |
| SD.189 | 92.141 | 92.485 | 93.058 | 89.456 | 87.628 | 87.136 | 86.619 | 81.927 | 81.712 | 82.798 | 79.957 |
| SD.190 | 92.134 | 92.477 | 93.046 | 89.459 | 87.625 | 87.132 | 86.615 | 81.930 | 81.712 | 82.804 | 79.959 |

ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
RAW DATA SUMMARY REPORT

| SCAN NO. | RTD #21 | RTD #22 | RTD #23 | RTD #24 | DEW CELL #1 | DEW CELL #2 | DEW CELL #3 | DEW CELL #4 | DEW CELL #5 | DEW CELL #6 |
|---|---|---|---|---|---|---|---|---|---|---|
| SD.171 | 79.954 | 79.313 | 79.762 | 79.641 | 78.537 | 79.444 | 77.812 | 77.104 | 70.616 | 70.748 |
| SD.172 | 79.942 | 79.313 | 79.748 | 79.634 | 78.538 | 79.391 | 77.821 | 77.122 | 70.541 | 70.687 |
| SD.173 | 79.953 | 79.315 | 79.756 | 79.638 | 78.558 | 79.425 | 77.890 | 77.023 | 70.573 | 70.677 |
| SD.174 | 79.959 | 79.315 | 79.757 | 79.640 | 78.529 | 79.403 | 77.970 | 77.003 | 70.567 | 70.726 |
| SD.175 | 79.959 | 79.318 | 79.762 | 79.643 | 78.556 | 79.414 | 77.975 | 77.005 | 70.559 | 70.703 |
| SD.176 | 79.950 | 79.321 | 79.771 | 79.649 | 78.515 | 79.411 | 77.745 | 76.977 | 70.594 | 70.652 |
| SD.177 | 79.959 | 79.318 | 79.763 | 79.646 | 78.498 | 79.422 | 77.580 | 76.994 | 70.571 | 70.657 |
| SD.178 | 79.959 | 79.316 | 79.769 | 79.652 | 78.572 | 79.405 | 77.504 | 77.075 | 70.570 | 70.680 |
| SD.179 | 79.968 | 79.319 | 79.765 | 79.647 | 78.498 | 79.383 | 77.967 | 77.218 | 70.562 | 70.733 |
| SD.180 | 79.965 | 79.321 | 79.763 | 79.640 | 78.463 | 79.379 | 77.909 | 77.203 | 70.573 | 70.695 |
| SD.181 | 79.969 | 79.324 | 79.763 | 79.641 | 78.419 | 79.380 | 77.865 | 77.134 | 70.568 | 70.674 |
| SD.182 | 79.966 | 79.324 | 79.768 | 79.649 | 78.495 | 79.382 | 77.803 | 77.050 | 70.573 | 70.675 |
| SD.183 | 79.968 | 79.325 | 79.768 | 79.647 | 78.517 | 79.362 | 77.310 | 76.928 | 70.527 | 70.665 |
| SD.184 | 79.969 | 79.322 | 79.759 | 79.644 | 78.482 | 79.312 | 77.604 | 76.715 | 70.549 | 70.665 |
| SD.185 | 79.966 | 79.324 | 79.756 | 79.649 | 78.463 | 79.347 | 77.870 | 76.919 | 70.539 | 70.661 |
| SD.186 | 79.974 | 79.324 | 79.762 | 79.649 | 78.439 | 79.325 | 77.882 | 77.061 | 70.518 | 70.636 |
| SD.187 | 79.980 | 79.324 | 79.760 | 79.649 | 78.422 | 79.330 | 77.716 | 77.027 | 70.472 | 70.657 |
| SD.188 | 79.982 | 79.327 | 79.768 | 79.653 | 78.431 | 79.310 | 77.871 | 76.982 | 70.509 | 70.655 |
| SD.189 | 79.978 | 79.328 | 79.768 | 79.652 | 78.488 | 79.257 | 78.083 | 76.982 | 70.532 | 70.631 |
| SD.190 | 79.980 | 79.332 | 79.777 | 79.653 | 78.425 | 79.309 | 77.862 | 76.965 | 70.544 | 70.634 |

ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
ILRT PROGRAM REPORT

STARTING DAY - 258 STARTING TIME - 2:45: 0 STARTING SCAN - SD.171 ENDING SCAN - SD.190

| | | | | POINT TO POINT | | TOTAL TIME | | | MASS PLOT | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| SCAN NO. | ELAPSED TIME (HR) | AVERAGE TEMP. (F) | AVERAGE PRESSURE (PSIA) | MEASURED LEAK RATE | CALCULATED LEAK RATE | MEASURED LEAK RATE | CALCULATED LEAK RATE (WEIGHT PERCENT PER DAY) | UPPER CONFIDENCE | MEASURED LEAK RATE | CALCULATED LEAK RATE | UPPER CONFIDENCE |
| SD.171 | 0.00 | 88.56 | 63.321 | | | | | | | | |
| SD.172 | 0.25 | 88.55 | 63.320 | -0.104E+00 | -0.104E+00 | -0.104E+00 | -0.104E+00 | 0.000E+00 | -0.105E+00 | -0.105E+00 | 0.000E+00 |
| SD.173 | 0.50 | 88.54 | 63.320 | -0.126E-01 | -0.126E-01 | -0.584E-01 | -0.584E-01 | 0.000E+00 | -0.589E-01 | -0.589E-01 | 0.000E+00 |
| SD.174 | 0.75 | 88.54 | 63.318 | 0.319E+00 | 0.279E+00 | 0.671E-01 | 0.539E-01 | 0.183E+00 | 0.668E-01 | 0.155E+00 | 0.973E+00 |
| SD.175 | 1.00 | 88.53 | 63.315 | 0.357E+00 | 0.397E+00 | 0.139E+00 | 0.139E+00 | 0.187E+00 | 0.140E+00 | 0.231E+00 | 0.414E+00 |
| SD.176 | 1.25 | 88.52 | 63.313 | -0.138E+00 | 0.144E+00 | 0.840E-01 | 0.140E+00 | 0.270E+00 | 0.839E-01 | 0.172E+00 | 0.288E+00 |
| SD.177 | 1.50 | 88.51 | 63.314 | -0.438E+00 | -0.147E+00 | -0.324E-02 | 0.925E-01 | 0.281E+00 | -0.323E-02 | 0.682E-01 | 0.213E+00 |
| SD.178 | 1.75 | 88.50 | 63.312 | 0.967E-01 | -0.644E-01 | 0.110E-01 | 0.701E-01 | 0.245E+00 | 0.110E-01 | 0.341E-01 | 0.139E+00 |
| SD.179 | 2.00 | 88.49 | 63.311 | 0.454E+00 | 0.137E+00 | 0.664E-01 | 0.784E-01 | 0.229E+00 | 0.663E-01 | 0.551E-01 | 0.134E+00 |
| SD.180 | 2.25 | 88.48 | 63.309 | 0.401E-01 | 0.113E+00 | 0.636E-01 | 0.822E-01 | 0.217E+00 | 0.636E-01 | 0.629E-01 | 0.123E+00 |
| SD.181 | 2.50 | 88.48 | 63.308 | 0.389E-01 | 0.953E-01 | 0.612E-01 | 0.836E-01 | 0.207E+00 | 0.611E-01 | 0.671E-01 | 0.114E+00 |
| SD.182 | 2.75 | 88.47 | 63.306 | 0.664E-01 | 0.912E-01 | 0.616E-01 | 0.843E-01 | 0.199E+00 | 0.616E-01 | 0.688E-01 | 0.107E+00 |
| SD.183 | 3.00 | 88.47 | 63.305 | -0.274E+00 | -0.121E-01 | 0.336E-01 | 0.763E-01 | 0.188E+00 | 0.336E-01 | 0.565E-01 | 0.904E-01 |
| SD.184 | 3.25 | 88.46 | 63.303 | 0.405E+00 | 0.963E-01 | 0.623E-01 | 0.778E-01 | 0.183E+00 | 0.622E-01 | 0.607E-01 | 0.894E-01 |
| SD.185 | 3.50 | 88.45 | 63.301 | 0.362E+00 | 0.169E+00 | 0.836E-01 | 0.843E-01 | 0.183E+00 | 0.837E-01 | 0.719E-01 | 0.989E-01 |
| SD.186 | 3.75 | 88.45 | 63.300 | 0.498E-01 | 0.150E+00 | 0.814E-01 | 0.887E-01 | 0.182E+00 | 0.814E-01 | 0.785E-01 | 0.103E+00 |
| SD.187 | 4.00 | 88.44 | 63.298 | 0.566E-01 | 0.136E+00 | 0.798E-01 | 0.917E-01 | 0.181E+00 | 0.799E-01 | 0.822E-01 | 0.104E+00 |
| SD.188 | 4.25 | 88.43 | 63.296 | 0.252E+00 | 0.167E+00 | 0.900E-01 | 0.962E-01 | 0.182E+00 | 0.900E-01 | 0.883E-01 | 0.108E+00 |
| SD.189 | 4.50 | 88.43 | 63.295 | 0.240E+00 | 0.190E+00 | 0.984E-01 | 0.101E+00 | 0.183E+00 | 0.983E-01 | 0.952E-01 | 0.114E+00 |
| SD.190 | 4.75 | 88.42 | 63.294 | -0.125E+00 | 0.137E+00 | 0.866E-01 | 0.103E+00 | 0.183E+00 | 0.866E-01 | 0.967E-01 | 0.114E+00 |

ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
ILRT PROGRAM REPORT

STARTING DAY - 258 STARTING TIME - 2:45: 0 STARTING SCAN - SD.171 ENDING SCAN - SD.190

ILRT RESULTS AFTER 4.75 HRS.

AVERAGE MEASURED LEAK RATES
(WEIGHT PERCENT PER DAY)

| POINT TO POINT | TOTAL TIME | | MASS PLOT | |
|---|---|---|---|---|
| LEAK RATE | LEAK RATE | STD.DEV. | LEAK RATE | STD.DEV. |
| 0.866E-01 | 0.529E-01 | 0.568E-01 | 0.528E-01 | 0.570E-01 |

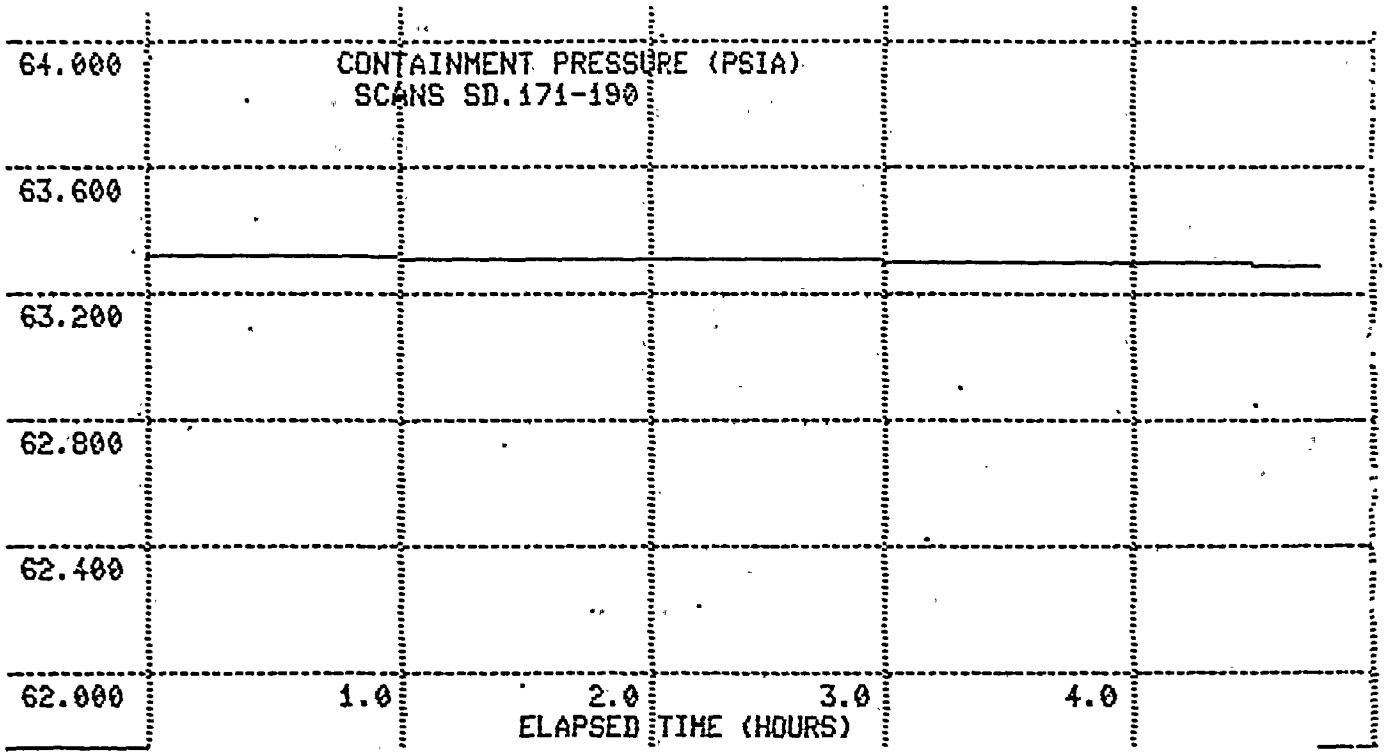
CALCULATED LEAK RATES
(WEIGHT PERCENT PER DAY)

| POINT TO POINT | TOTAL TIME | | | MASS PLOT | | |
|---|---|---|---|---|---|---|
| LEAK RATE | LEAK RATE | STD.DEV. | UPPER CON. LIMIT | LEAK RATE | STD.DEV. | UPPER CON.LIMIT |
| 0.137E+00 | 0.103E+00 | 0.375E-01 | 0.182E+00 | 0.967E-01 | 0.971E-02 | 0.117E+00 |

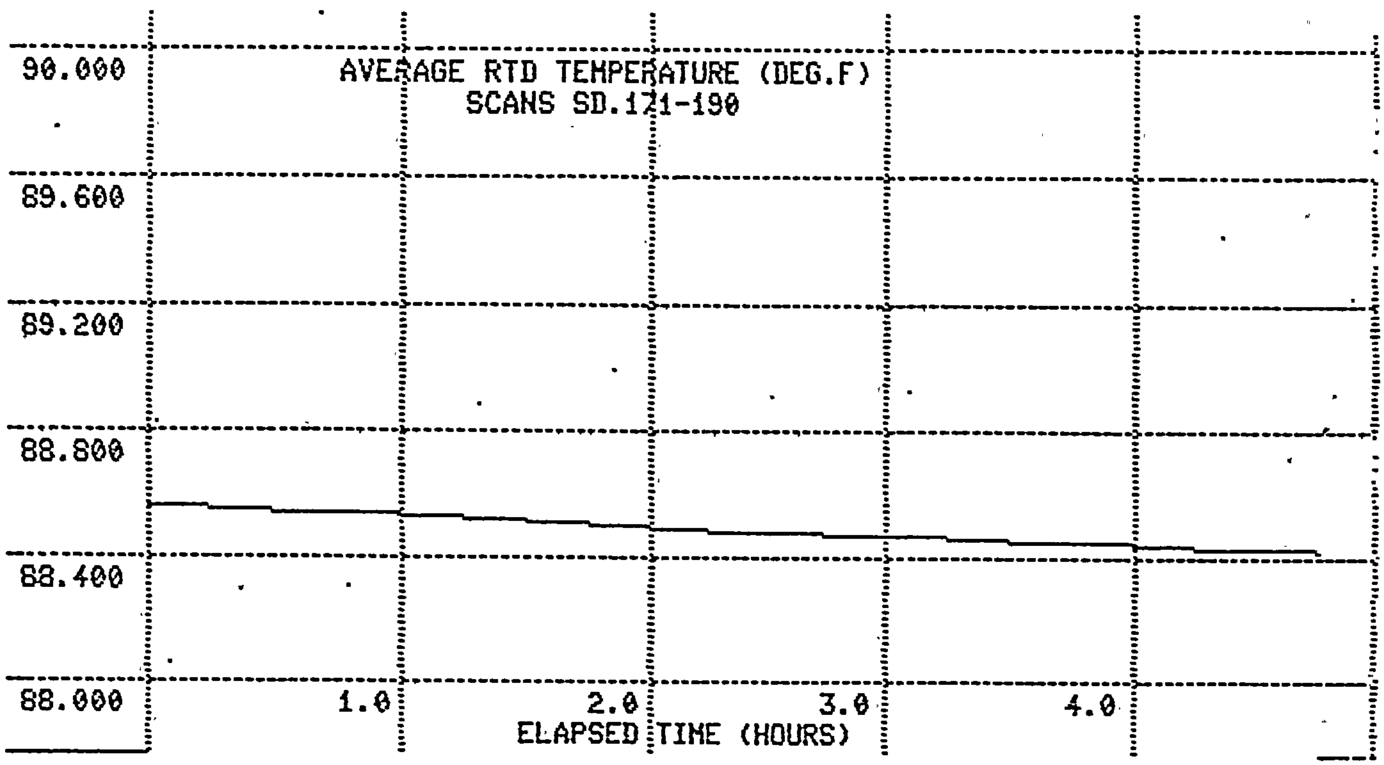
ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
RELATIVE HUMIDITY PROGRAM

| SCAN NO. | AVERAGE DEW POINT TEMPERATURE (F) | AVERAGE CONTAINMENT TEMPERATURE (F) | AVERAGE VAPOR PRESSURE (PSIA) | AVERAGE RELATIVE HUMIDITY (%) |
|---|---|---|---|---|
| SD.171 | 76.035 | 88.558 | 0.444 | 66.645 |
| SD.172 | 76.013 | 88.546 | 0.443 | 66.621 |
| SD.173 | 76.046 | 88.541 | 0.444 | 66.705 |
| SD.174 | 76.086 | 88.536 | 0.444 | 66.804 |
| SD.175 | 76.087 | 88.530 | 0.444 | 66.818 |
| SD.176 | 75.971 | 88.520 | 0.443 | 66.583 |
| SD.177 | 75.904 | 88.512 | 0.442 | 66.450 |
| SD.178 | 75.888 | 88.502 | 0.441 | 66.436 |
| SD.179 | 76.080 | 88.495 | 0.444 | 66.878 |
| SD.180 | 76.039 | 88.485 | 0.444 | 66.808 |
| SD.181 | 76.008 | 88.482 | 0.443 | 66.746 |
| SD.182 | 75.994 | 88.470 | 0.443 | 66.740 |
| SD.183 | 75.789 | 88.472 | 0.440 | 66.282 |
| SD.184 | 75.898 | 88.464 | 0.441 | 66.539 |
| SD.185 | 76.008 | 88.453 | 0.443 | 66.807 |
| SD.186 | 75.999 | 88.448 | 0.443 | 66.796 |
| SD.187 | 75.935 | 88.442 | 0.442 | 66.668 |
| SD.188 | 75.997 | 88.431 | 0.443 | 66.828 |
| SD.189 | 76.077 | 88.425 | 0.444 | 67.018 |
| SD.190 | 75.986 | 88.421 | 0.443 | 66.823 |

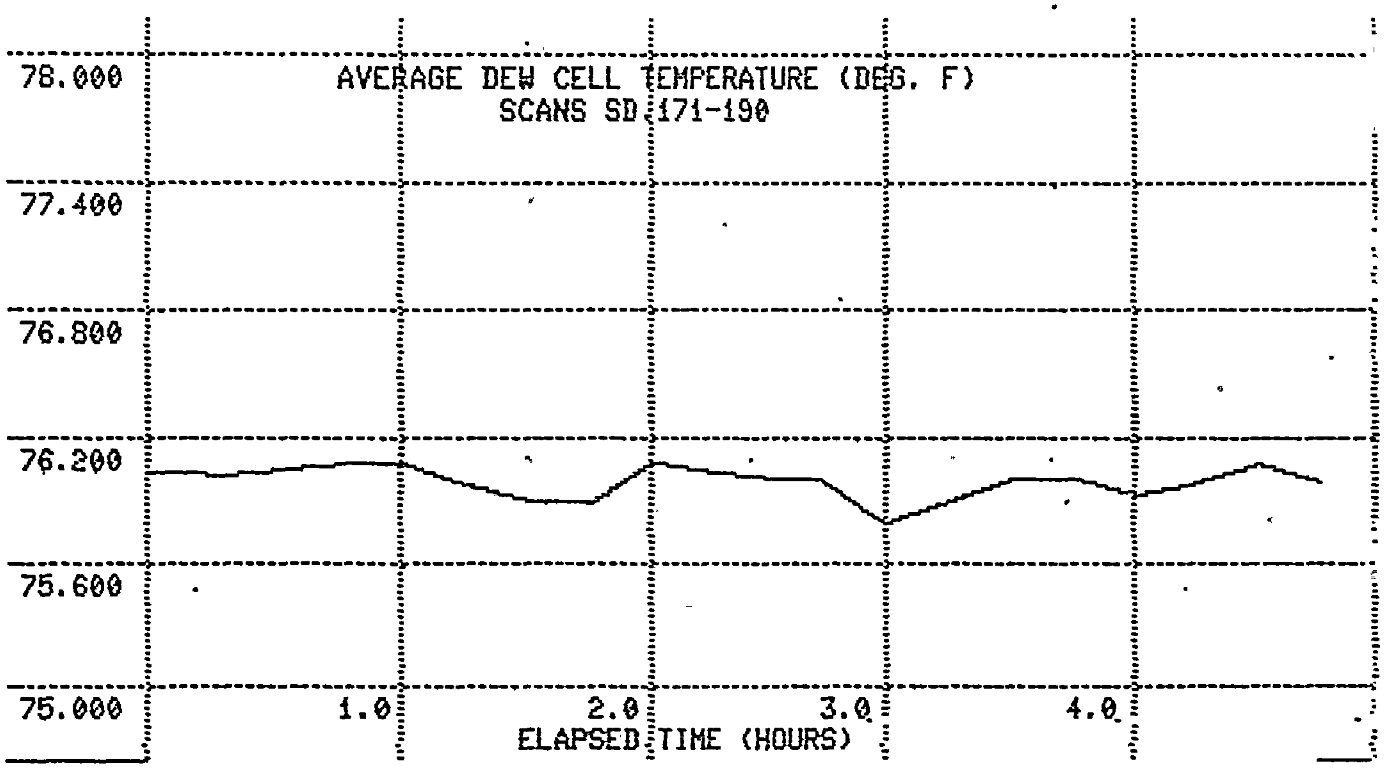
CLRT

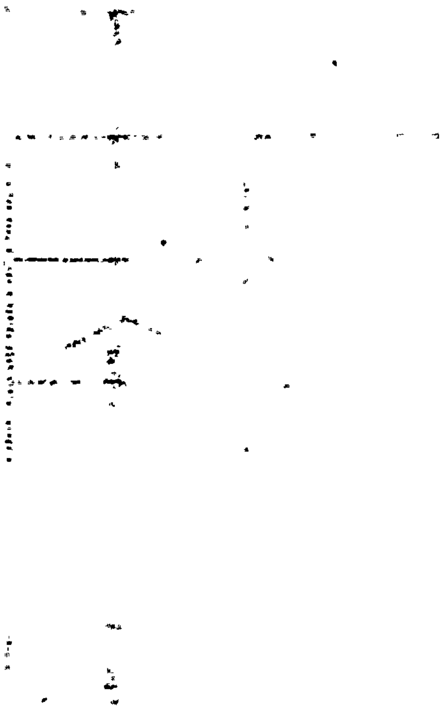

64.000
CONTAINMENT PRESSURE (PSIA)
SCANS SD.171-190
63.600
63.200
62.800
62.400
62.000
1.0
2.0
3.0
4.0
ELAPSED TIME (HOURS)

CLRT

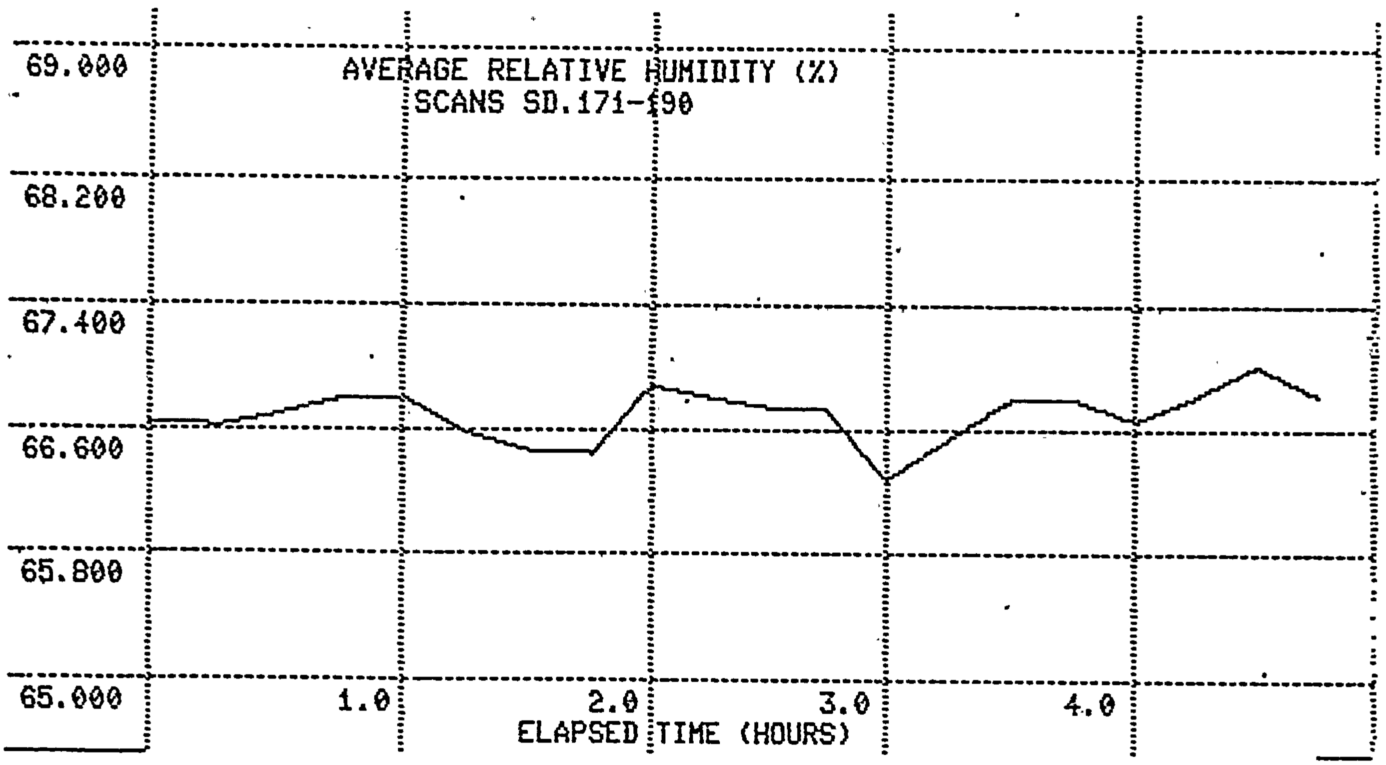

AVERAGE RTD TEMPERATURE (DEG.F)
SCANS SD.171-190
90.000
89.600
89.200
88.800
88.400
88.000
1.0
2.0
3.0
4.0
ELAPSED TIME (HOURS)

CLRT

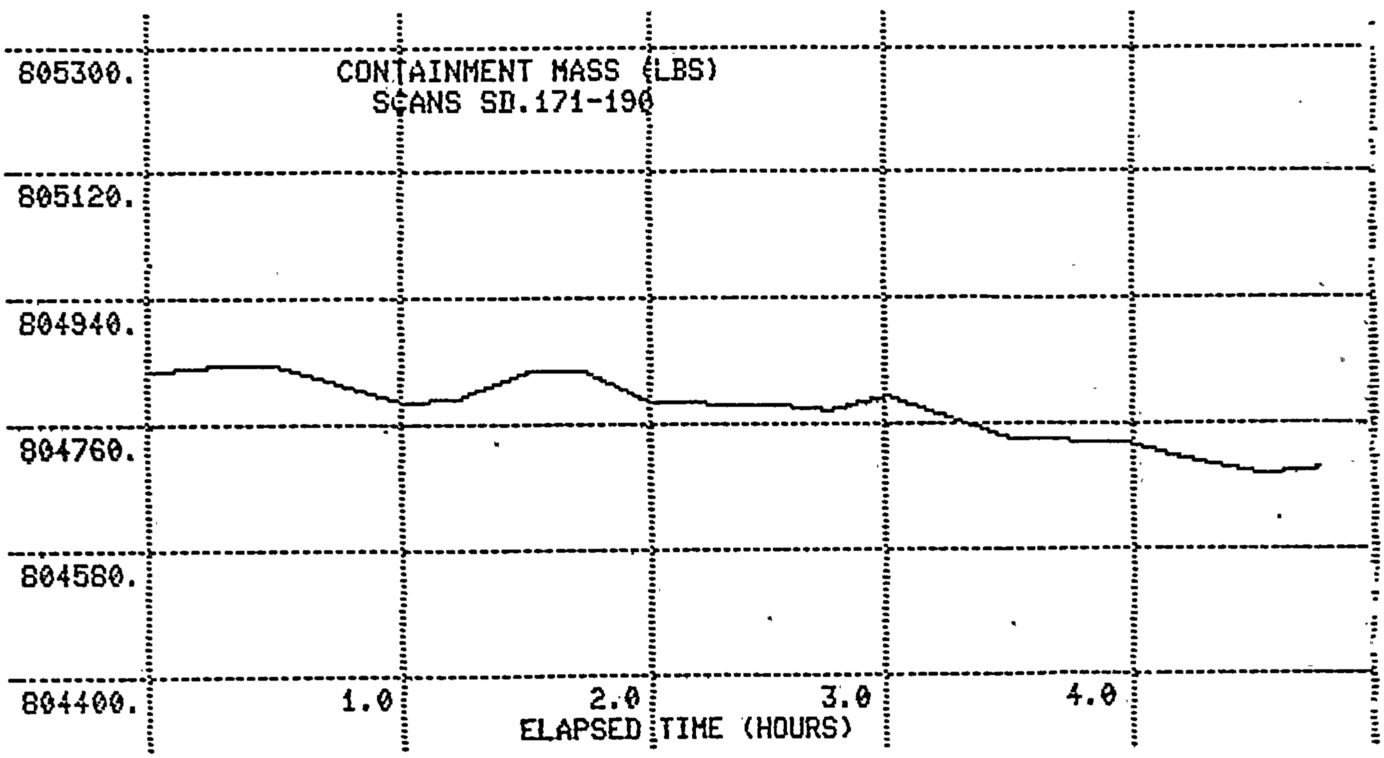

AVERAGE DEW CELL TEMPERATURE (DEG. F)
SCANS SD 171-190
78.000
77.400
76.800
76.200
75.600
75.000
1.0
2.0
3.0
4.0
ELAPSED TIME (HOURS)

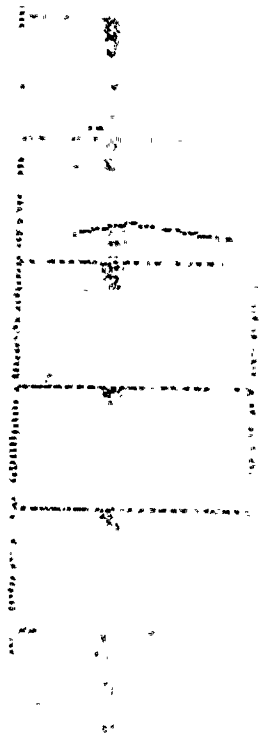


CLRT

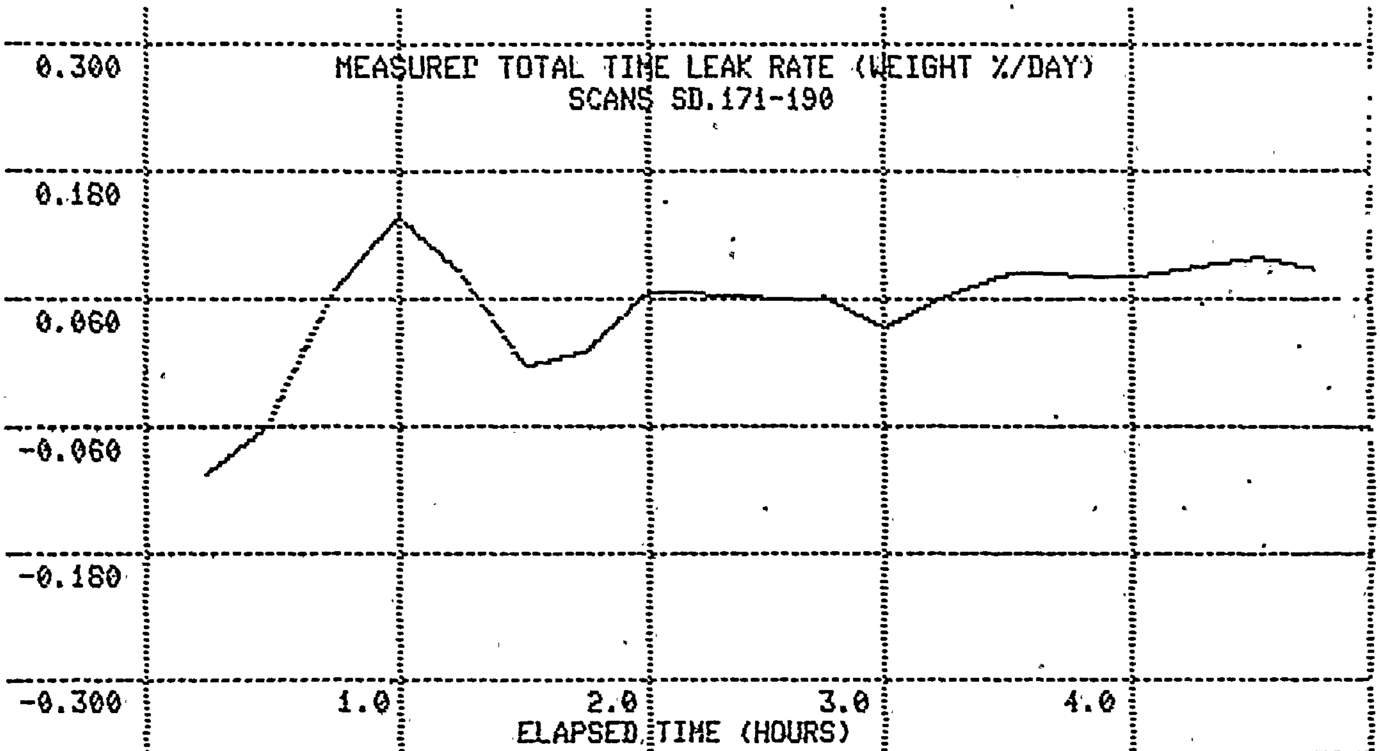

AVERAGE RELATIVE HUMIDITY (%)
SCANS SD.171-190
69.000
68.200
67.400
66.600
65.800
65.000
1.0
2.0
3.0
4.0
ELAPSED TIME (HOURS)

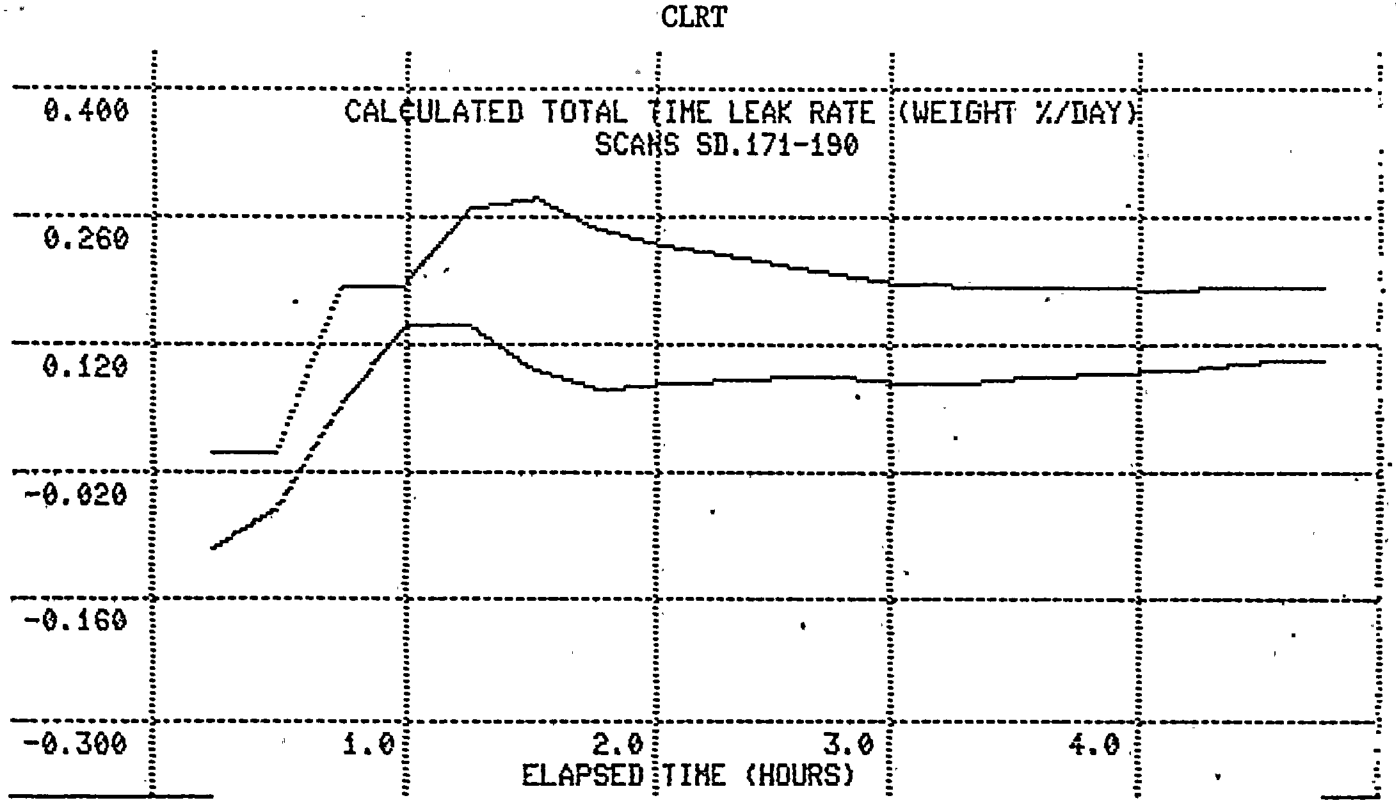
CLRT


CONTAINMENT MASS (LBS)
SCANS SD.171-190
805300.
805120.
804940.
804760.
804580.
804400.
1.0
2.0
3.0
4.0
ELAPSED TIME (HOURS)



CLRT


MEASURED TOTAL TIME LEAK RATE (WEIGHT %/DAY)
SCANS SD.171-190
0.300
0.180
0.060
-0.060
-0.180
-0.300
1.0
2.0
3.0
4.0
ELAPSED TIME (HOURS)


CLRT
CALCULATED TOTAL TIME LEAK RATE (WEIGHT %/DAY)
SCANS SD.171-190
0.400
0.260
0.120
-0.020
-0.160
-0.300
1.0
2.0
3.0
4.0
ELAPSED TIME (HOURS)

4.

# GENERAL

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| 31 | 99.990 | × 99.065 | | | × 99.135 | | |
| | | | 0.99900 | 1.024 | | 0.99641 | 1.211 |
| 20 | 94.995 | × 94.065 | | | × 94.135 | | |
| | | | 1.00040 | 0.893 | | 0.99861 | 1.003 |
| 19 | 89.951 | × 89.023 | | | × 89.071 | | |
| | | | 1.00291 | 0.714 | | 0.99900 | 0.969 |
| 18 | 84.955 | × 84.040 | | | × 84.072 | | |
| | | | 1.00221 | 0.730 | | 1.00181 | 0.758 |
| 17 | 79.952 | × 79.051 | | | × 79.086 | | |
| | | | 1.00281 | 0.651 | | 1.00203 | 0.478 |
| 1 | 74.958 | × 74.079 | | | × 74.117 | | |
| | | | 1.00283 | 0.501 | | [illegible] | 0.207 |
| [illegible] | [illegible] | × 69.110 | | | [illegible] | | |
| | | | [illegible] | [illegible] | | 1 [illegible] | [illegible] |

ARIZONA PUBLIC SERVICE COMPANY

PALO VERDE NUCLEAR GENERATING STATION

PRESSURE GAUGE CALIBRATION PROGRAM

| CALIBRATION POINT | TRUE PRESSURE | PRESSURE GAUGE 1 GAUGE READING | PRESSURE GAUGE 1 MULTIPLICATION FACTOR | PRESSURE GAUGE 1 CORRECTION CONSTANT | PRESSURE GAUGE 2 GAUGE READING | PRESSURE GAUGE 2 MULTIPLICATION FACTOR | PRESSURE GAUGE 2 CORRECTION CONSTANT |
|---|---|---|---|---|---|---|---|
| 1 | 0.000 | * 0.000 | | | * 0.000 | | |
| | | *--------------< | 1.01898 | 0.000 | *--------------< | 1.00869 | 0.000 |
| 2 | 4.994 | * 4.901 | | | * 4.951 | | |
| | | *--------------< | 1.01855 | 0.002 | *--------------< | 1.00746 | 0.006 |
| 3 | 9.990 | * 9.806 | | | * 9.910 | | |
| | | *--------------< | 1.01543 | 0.033 | *--------------< | 1.01010 | -0.020 |
| 4 | 14.990 | * 14.730 | | | * 14.860 | | |
| | | *--------------< | 1.01218 | 0.081 | *--------------< | 1.01218 | -0.051 |
| 5 | 19.978 | * 19.658 | | | * 19.788 | | |
| | | *--------------< | 1.01463 | 0.032 | *--------------< | 1.01278 | -0.063 |
| 6 | 24.972 | * 24.580 | | | * 24.719 | | |
| | | *--------------< | 1.01628 | -0.008 | *--------------< | 1.01319 | -0.073 |
| 7 | 29.966 | * 29.494 | | | * 29.648 | | |
| | | *--------------< | 1.01298 | 0.089 | *--------------< | 1.01236 | -0.049 |
| 8 | 34.961 | * 34.425 | | | * 34.582 | | |
| | | *--------------< | 1.01155 | 0.139 | *--------------< | 1.01134 | -0.013 |
| 9 | 39.955 | * 39.362 | | | * 39.520 | | |
| | | *--------------< | 1.00930 | 0.227 | *--------------< | 1.01278 | -0.070 |
| 10 | 44.949 | * 44.310 | | | * 44.451 | | |
| | | *--------------< | 1.01082 | 0.159 | *--------------< | 1.01509 | -0.173 |
| 11 | 49.993 | * 49.300 | | | * 49.420 | | |
| | | *--------------< | 1.01093 | 0.154 | *--------------< | 1.01524 | -0.180 |
| 12 | 54.988 | * 54.241 | | | * 54.340 | | |
| | | *--------------< | 1.00970 | 0.221 | *--------------< | 1.01236 | -0.024 |
| 13 | 59.985 | * 59.190 | | | * 59.276 | | |
| | | *--------------< | 1.00685 | 0.389 | *--------------< | 1.01175 | 0.013 |
| 14 | 64.979 | * 64.150 | | | * 64.212 | | |
| | | *--------------< | 1.00686 | 0.389 | *--------------< | 1.00828 | 0.235 |
| 15 | 69.973 | * 69.110 | | | * 69.165 | | |
| | | *--------------< | 1.00523 | 0.501 | *--------------< | 1.00868 | 0.207 |
| 16 | 74.968 | * 74.079 | | | * 74.117 | | |
| | | *--------------< | 1.00321 | 0.651 | *--------------< | 1.00503 | 0.478 |
| 17 | 79.962 | * 79.057 | | | * 79.086 | | |
| | | *--------------< | 1.00221 | 0.730 | *--------------< | 1.00181 | 0.733 |
| 18 | 84.956 | * 84.040 | | | * 84.071 | | |
| | | *--------------< | 1.00241 | 0.714 | *--------------< | 0.99900 | 0.969 |
| 19 | 89.951 | * 89.023 | | | * 89.071 | | |
| | | *--------------< | 1.00040 | 0.893 | *--------------< | 0.99861 | 1.003 |
| 20 | 94.995 | * 94.065 | | | * 94.122 | | |
| | | *--------------< | 0.99900 | 1.024 | *--------------< | 0.99641 | 1.211 |
| 21 | 99.990 | * 99.065 | | | * 99.135 | | |

************************************************************************************************************************

FAULTY PRESSURE GAUGE 1.
THE CONTAINMENT VOLUME IS 0.2600E+05 CUBIC FEET
THE NUMBER OF FANS IN USE ARE 2.
THE TOTAL PERCENT FOR RTDS IS 100.00 %
THE NUMBER OF DEW CELLS IN USE IS 4.
THE TOTAL PERCENT FOR DEW CELLS IS 100.000 %

ARIZONA PUBLIC SERVIC COMPANY

PALO VERDE NUCLEAR GENERATING STATION

ILRT SUB-VOLUME WEIGHTING PROGRAM

DATE: 11- 5-86 TIME: 14:33:49

| DAS CHANNEL NO. | TYPE OF SENSOR | CONTAINMENT WEIGHTING FACTOR(%) | DAS CHANNEL NO. | TYPE OF SENSOR | CONTAINMENT WEIGHTING FACTOR(%) |
|---|---|---|---|---|---|
| 1 | RTD# 1 | 4.83 | 2 | RTD# 2 | 4.83 |
| 3 | RTD# 3 | 4.83 | 4 | RTD# 4 | 4.83 |
| 5 | RTD# 5 | 4.83 | 6 | RTD# 6 | 4.83 |
| 7 | RTD# 7 | 4.33 | 8 | RTD# 8 | 4.33 |
| 9 | RTD# 9 | 4.33 | 10 | RTD#10 | 4.33 |
| 11 | RTD#11 | 4.33 | 12 | RTD#12 | 4.33 |
| 13 | RTD#13 | 3.77 | 14 | RTD#14 | 3.77 |
| 15 | RTD#15 | 3.77 | 16 | RTD#16 | 3.77 |
| 17 | RTD#17 | 3.41 | 18 | RTD#18 | 3.41 |
| 19 | RTD#19 | 3.41 | 20 | RTD#20 | 3.95 |
| 21 | RTD#21 | 3.95 | 22 | RTD#22 | 3.95 |
| 23 | RTD#23 | 3.95 | 24 | RTD#24 | 3.95 |
| 25 | DEW CELL# 1 | 14.49 | 26 | DEW CELL# 2 | 14.49 |
| 27 | DEW CELL# 3 | 41.04 | 28 | DEW CELL# 4 | 00.00 |
| 29 | DEW CELL# 5 | 00.00 | 30 | DEW CELL# 6 | 29.99 |

FAULTY PRESSURE GUAGE-- 2.
THE CONTAINMENT VOLUME IS 0.2600E+07 CUBIC FEET
THE NUMBER OF RTDS IN USE ARE 24.
THE TOTAL PERCENT FOR RTDS IS 100.000 %
THE NUMBER OF DEW CELLS IN USE IS 4.
THE TOTAL PERCENT FOR DEW CELLS IS 100.000 %

PALO VERDE NUCLEAR STATION

APS.

NUREG-1287

# Technical Specifications
## Palo Verde Nuclear Generating Station, Unit No. 3

Docket No. STN 50-530

Appendix "A" to
License No. NPF-74


**Issued by the**
**U.S. Nuclear Regulatory**
**Commission**

**Office of Nuclear Reactor Regulation**

November 1987

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NUREG-1287

# Technical Specifications
## Palo Verde Nuclear Generating Station, Unit No. 3

Docket No. STN 50-530

Appendix "A" to
License No. NPF-74

Issued by the
U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

November 1987

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## LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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# SECTION 1.0

# DEFINITIONS

## 1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a specification which prescribes remedial measures required under designated conditions.

AXIAL SHAPE INDEX

1.2 The AXIAL SHAPE INDEX shall be the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers.

AZIMUTHAL POWER TILT - $T_q$

1.3 AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.
b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.
c. Digital computer channels - the exercising of the digital computer hardware using diagnostic programs and the injection of simulated process data into the channel to verify OPERABILITY including alarm and/or trip functions.
d. Radiological effluent process monitoring channels - the CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is functionally tested.

The CHANNEL FUNCTIONAL TEST shall include adjustment, as necessary, of the alarm, interlock and/or trip setpoints such that the setpoints are within the required range and accuracy.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:
   1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or
   2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.
b. All equipment hatches are closed and sealed,
c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,
d. The containment leakage rates are within the limits of Specification 3.6.1.2, and
e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 Not Applicable.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURES RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE SYSTEM

1.14 A GASEOUS RADWASTE SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage into closed systems, other than reactor coolant pump controlled bleed-off flow, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the secondary system.

MEMBER(S) OF THE PUBLIC

1.16 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.17 The OFFSITE DOSE CALCULATION MANUAL shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

1.18 A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and cold leg reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PLANAR RADIAL PEAKING FACTOR - $F_{xy}$

1.21 The PLANAR RADIAL PEAKING FACTOR is the ratio of the peak to plane average power density of the individual fuel rods in a given horizontal plane, excluding the effects of azimuthal tilt.

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.23 The PROCESS CONTROL PROGRAM shall contain the provisions to assure that the SOLIDIFICATION of wet radioactive wastes results in a waste form with properties that meet the requirements of 10 CFR Part 61 and of low level radioactive waste disposal sites. The PCP shall identify process parameters influencing SOLIDIFICATION such as pH, oil content, $H_2O$ content, solids content, ratio of solidification agent to waste and/or necessary additives for each type of anticipated waste, and the acceptable boundary conditions for the process parameters shall be identified for each waste type, based on laboratory scale and full-scale testing or experience. The PCP shall also include an identification of conditions that must be satisfied, based on full-scale testing, to assure that dewatering of bead resins, powdered resins, and filter sludges will result in volumes of free water, at the time of disposal, within the limits of 10 CFR Part 61 and of low level radioactive waste disposal sites.

PURGE - PURGING

1.24 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3800 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Sections 50.72 and 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. No change in part-length control element assembly position, and

b. All full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.29 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOFTWARE

1.30 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation, and procedures.

SOLIDIFICATION

1.31 SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a homogeneous (uniformly distributed), monolithic, immobilized solid with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

SOURCE CHECK

1.32 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.33 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.34 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.35 UNIDENTIFIED LEAKAGE shall be all leakage which does not constitute either IDENTIFIED LEAKAGE or reactor coolant pump controlled bleed-off flow.

UNRESTRICTED AREA

1.36 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.37 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.38 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

## TABLE 1.1

## FREQUENCY NOTATION

| NOTATION | FREQUENCY |
|---|---|
| S | At least once per 12 hours. |
| D | At least once per 24 hours. |
| W | At least once per 7 days. |
| 4/M | At least 4 times per month at intervals no greater than 9 days and a minimum of 48 times per year. |
| M | At least once per 31 days. |
| Q | At least once per 92 days. |
| SA | At least once per 184 days. |
| R | At least once per 18 months. |
| P | Completed prior to each release. |
| S/U | Prior to each reactor startup. |
| N.A. | Not applicable. |

TABLE 1.2

OPERATIONAL MODES

| OPERATIONAL MODE | REACTIVITY CONDITION, $K_{eff}$ | % OF RATED THERMAL POWER* | COLD LEG TEMPERATURE ($T_{cold}$) |
|---|---|---|---|
| 1. POWER OPERATION | $\geq$ 0.99 | > 5% | $\geq$ 350°F |
| 2. STARTUP | $\geq$ 0.99 | $\leq$ 5% | $\geq$ 350°F |
| 3. HOT STANDBY | < 0.99 | 0 | $\geq$ 350°F |
| 4. HOT SHUTDOWN | < 0.99 | 0 | 350° > $T_{cold}$ > 210°F |
| 5. COLD SHUTDOWN | < 0.99 | 0 | $\leq$ 210°F |
| 6. REFUELING** | $\leq$ 0.95 | 0 | $\leq$ 135°F |

*Excluding decay heat.
**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

## 2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

### 2.1 SAFETY LIMITS

#### 2.1.1 REACTOR CORE

DNBR

2.1.1.1 The calculated DNBR of the reactor core shall be maintained greater than or equal to 1.231.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the calculated DNBR of the reactor has decreased to less than 1.231, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained less than or equal to 21 kW/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21 kW/ft, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.

## 2.2 LIMITING SAFETY SYSTEM SETTINGS

### REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint limits shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

## TABLE 2.2-1

## REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| I. TRIP GENERATION | | |
| A. Process | | |
| 1. Pressurizer Pressure - High | ≤ 2383 psia | ≤ 2388 psia |
| 2. Pressurizer Pressure - Low | ≥ 1837 psia (2) | ≥ 1822 psia (2) |
| 3. Steam Generator Level - Low | ≥ 44.2% (4) | ≥ 43.7% (4) |
| 4. Steam Generator Level - High | ≤ 91.0% (9) | ≤ 91.5% (9) |
| 5. Steam Generator Pressure - Low | ≥ 919 psia (3) | ≥ 912 psia (3) |
| 6. Containment Pressure - High | ≤ 3.0 psig | ≤ 3.2 psig |
| 7. Reactor Coolant Flow - Low | | |
| a. Rate | ≤ 0.115 psi/sec (6)(7) | ≤ 0.118 psi/sec (6)(7) |
| b. Floor | ≥ 11.9 psid(6)(7) | ≥ 11.7 psid (6)(7) |
| c. Band | ≤ 10.0 psid(6)(7) | ≤ 10.2 psid (6)(7) |
| 8. Local Power Density - High | ≤ 21.0 kW/ft (5) | ≤ 21.0 kW/ft (5) |
| 9. DNBR - Low | ≥ 1.231 (5) | ≥ 1.231 (5) |
| B. Excore Neutron Flux | | |
| 1. Variable Overpower Trip | | |
| a. Rate | ≤ 10.6%/min of RATED THERMAL POWER (8) | ≤ 11.0%/min of RATED THERMAL POWER (8) |
| b. Ceiling | ≤ 110.0% of RATED THERMAL POWER (8) | ≤ 111.0% of RATED THERMAL POWER (8) |
| c. Band | ≤ 9.8% of RATED THERMAL POWER (8) | ≤ 10.0% of RATED THERMAL POWER (8) |

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 2. Logarithmic Power Level - High (1) | | |
| a. Startup and Operating | < 0.798% of RATED THERMAL POWER | ≤ 0.895% of RATED THERMAL POWER |
| b. Shutdown | < 0.798% of RATED THERMAL POWER | ≤ 0.895% of RATED THERMAL POWER |
| C. Core Protection Calculator System | | |
| 1. CEA Calculators | Not Applicable | Not Applicable |
| 2. Core Protection Calculators | Not Applicable | Not Applicable |
| D. Supplementary Protection System | | |
| Pressurizer Pressure - High | ≤ 2409 psia | ≤ 2414 psia |
| II. RPS LOGIC | | |
| A. Matrix Logic | Not Applicable | Not Applicable |
| B. Initiation Logic | Not Applicable | Not Applicable |
| III. RPS ACTUATION DEVICES | | |
| A. Reactor Trip Breakers | Not Applicable | Not Applicable |
| B. Manual Trip | Not Applicable | Not Applicable |

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) Trip may be manually bypassed above $10^{-4}$% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to $10^{-4}$% of RATED THERMAL POWER.

(2) In MODES 3-4, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) In MODES 3-4, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

The approved DNBR limit is 1.231 which includes a partial rod bow penalty compensation. If the fuel burnup exceeds that for which an increased rod bow penalty is required, the DNBR limit shall be adjusted. In this case a DNBR trip setpoint of 1.231 is allowed provided that the difference is compensated by an increase in the CPC addressable constant BERR1 as follows:

$$BERR1_{new} = BERR1_{old} \left[1 + \frac{RB - RB_o}{100} \times \frac{d\ (\%\ POL)}{d\ (\%\ DNBR)}\right]$$

where $BERR1_{old}$ is the uncompensated value of BERR1; RB is the fuel rod bow penalty in % DNBR; $RB_o$ is the fuel rod bow penalty in % DNBR already accounted for in the DNBR limit; POL is the power operating limit; and d (% POL)/d (% DNBR) is the absolute value of the most adverse derivative of POL with respect to DNBR.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS (Continued)

(6) RATE is the maximum rate of decrease of the trip setpoint. There are no restrictions on the rate at which the setpoint can increase.
FLOOR is the minimum value of the trip setpoint.
BAND is the amount by which the trip setpoint is below the input signal unless limited by Rate or Floor.
Setpoints are based on steam generator differential pressure.

(7) The setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

(8) RATE is the maximum rate of increase of the trip setpoint. (The rate at which the setpoint can decrease is no slower than five percent per second.)
CEILING is the maximum value of the trip setpoint.
BAND is the amount by which the trip setpoint is above the steady state input signal unless limited by the rate or the ceiling.

(9) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

## NOTE

The BASES contained in the succeeding pages summarize the reasons for the specifications of Section 2.0 but in accordance with 10 CFR 50.36 are not a part of these Technical Specifications.

## 2.1.1 REACTOR CORE

The restrictions of these safety limits prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by (1) restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature, and (2) maintaining the dynamically adjusted peak linear heat rate of the fuel at or less than 21 kW/ft which will not cause fuel centerline melting in any fuel rod.

First, by operating within the nucleate boiling regime of heat transfer, the heat transfer coefficient is large enough so that the maximum clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in higher cladding temperatures and the possibility of cladding failure.

Correlations predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during normal operation and design basis anticipated operational occurrences is limited to 1.231 based upon a statistical combination of CE-1 CHF correlation and engineering factor uncertainties and is established as a Safety Limit. The DNBR limit of 1.231 includes a rod bow compensation of 0.8% on DNBR. For fuel burnups which exceed that for which an increased rod bow penalty is required, the DNBR limit shall be adjusted. In this case the DNBR trip setpoint of 1.231 is allowed if the required DNBR increase is compensated by an increase of the addressable constant BERR1.

Second, operation with a peak linear heat rate below that which would cause fuel centerline melting maintains fuel rod and cladding integrity. Above this peak linear heat rate level (i.e., with some melting in the center), fuel rod integrity would be maintained only if the design and operating conditions are appropriate throughout the life of the fuel rods. Volume changes which accompany the solid to liquid phase change are significant and require accommodation. Another consideration involves the redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of melting. Because of the above factors, the steady state value of the peak linear heat rate which would not cause fuel centerline melting is established as a Safety Limit. To account for fuel rod dynamics (lags), the directly indicated linear heat rate is dynamically adjusted by the CPC program.

Limiting Safety System Settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High Linear Power Level trips, and Limiting Conditions for Operation on DNBR and kW/ft margin are specified such that there is a high degree of confidence that the specified acceptable fuel design limits are not exceeded during normal operation and design basis anticipated operational occurrences.

## 2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The Reactor Coolant System components are designed to Section III, 1974 Edition, Summer 1975 Addendum, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

## 2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Safety Limits of 1.231 and 21 kW/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CESSAR System 80 applicable system descriptions and safety analyses.

REACTOR TRIP SETPOINTS (Continued)

The methodology for the calculation of the PVNGS trip setpoint values, plant protection system, is discussed in the CE Document No. CEN-286(V) dated July 31, 1984.

Manual Reactor Trip

The Manual reactor trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Variable Overpower Trip

A reactor trip on Variable Overpower is provided to protect the reactor core during rapid positive reactivity addition excursions. This trip function will trip the reactor when the indicated neutron flux power exceeds either a rate limited setpoint at a great enough rate or reaches a preset ceiling. The flux signal used is the average of three linear subchannel flux signals originating in each nuclear instrument safety channel. These trip setpoints are provided in Table 2.2-1.

Logarithmic Power Level - High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above $10^{-4}$% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to $10^{-4}$% of RATED THERMAL POWER.

Pressurizer Pressure - High

The Pressurizer Pressure - High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is below the nominal lift setting of the pressurizer safety valves and its operation minimizes the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure - Low

The Pressurizer Pressure - Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a decrease in Reactor Coolant System inventory and in the event of an increase in heat

Pressurizer Pressure - Low (Continued)

removal by the secondary system. During normal operation, this trip's setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached. The operator may manually bypass this trip when pressurizer pressure is below 400 psia. This bypass is automatically removed when the pressurizer pressure increases to 500 psia.

Containment Pressure - High

The Containment Pressure - High trip provides assurance that a reactor trip is initiated in the event of containment building pressurization due to a pipe break inside the containment building. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure - Low

The Steam Generator Pressure - Low trip provides protection in the event of an increase in heat removal by the secondary system and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the normal pressure trip setpoint is reached.

Steam Generator Level - Low

The Steam Generator Level - Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to a decrease in heat removal by the secondary system. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before auxiliary feedwater is required to prevent degraded core cooling.

Local Power Density - High

The Local Power Density - High trip is provided to prevent the linear heat rate (kW/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any design bases anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

Local Power Density - High (Continued)

a. Nuclear flux power and axial power distribution from the excore flux monitoring system;

b. Radial peaking factors from the position measurement for the CEAs;

c. Delta T power from reactor coolant temperatures and coolant flow measurements.

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the Peak Linear Heat Rate Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR - Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of design bases anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of 1861 psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

DNBR - Low (Continued)

The DNBR, the trip variable, calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the calculated core DNBR is sufficiently greater than 1.231 such that the decrease in calculated core DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

| | Parameter | Limiting Value |
|---|---|---|
| a. | RCS Cold Leg Temperature-Low | ≥ 470°F |
| b. | RCS Cold Leg Temperature-High | ≤ 610°F |
| c. | Axial Shape Index-Positive | Not more positive than + 0.5 |
| d. | Axial Shape Index-Negative | Not more negative than - 0.5 |
| e. | Pressurizer Pressure-Low | ≥ 1861 psia |
| f. | Pressurizer Pressure-High | ≤ 2388 psia |
| g. | Integrated Radial Peaking Factor-Low | ≥ 1.28 |
| h. | Integrated Radial Peaking Factor-High | ≤ 4.28 |
| i. | Quality Margin-Low | ≥ 0 |

Steam Generator Level - High

The Steam Generator Level - High trip is provided to protect the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excesssive moisture carryover. This trip's setpoint does not correspond to a safety limit, and provides protection in the event of excess feedwater flow. The setpoint is identical to the main steam isolation setpoint. Its functional capability at the specified trip setting enhances the overall reliability of the reactor protection system.

Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a four pump flow coastdown during a steam line break with loss of offsite power. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a variable setpoint. This variable setpoint stays a set amount below the pressure differential unless limited by a set maximum decrease rate or a set minimum value. The specified setpoint ensures that a reactor trip occurs to prevent violation of Peak Linear Heat Rate or DNBR Safety Limits under the stated conditions.

Pressurizer Pressure - High (SPS)

The Supplementary Protection System (SPS) augments reactor protection against overpressurization by utilizing a separate and diverse trip logic from the Reactor Protection System for initiation of reactor trip. The SPS will initiate a reactor trip when pressurizer pressure exceeds a predetermined value.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

## 3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

## 3/4.0 APPLICABILITY

### LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and/or associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour, action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours, and
2. At least COLD SHUTDOWN within the following 30 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODE 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION statements. Exceptions to these requirements are stated in the individual specifications.

## APPLICABILITY

## SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, and

b. The combined time interval for any three consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

| ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities | Required frequencies for performing inservice inspection and testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Yearly or annually | At least once per 366 days |

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

## 3/4.1 REACTIVITY CONTROL SYSTEMS

### 3/4.1.1 BORATION CONTROL

### SHUTDOWN MARGIN - $T_{cold}$ GREATER THAN 210°F

### LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 6.0% delta k/k.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than 6.0% delta k/k, immediately initiate and continue boration at greater than or equal to 26 gpm to reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

### SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 6.0% delta k/k:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable as a result of excessive friction or mechanical interference or known to be untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODE 1 or MODE 2 with $K_{eff}$ greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with $K_{eff}$ less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

*See Special Test Exception 3.10.1.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

e. When in MODE 3 or 4, at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

## REACTIVITY CONTROL SYSTEMS

## SHUTDOWN MARGIN - $T_{cold}$ LESS THAN OR EQUAL TO 210°F

### LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 4.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 4.0% delta k/k, immediately initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

### SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 4.0% delta k/k:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable as a result of excessive friction or mechanical interference or known to be untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of the following factors:

    1. Reactor Coolant System boron concentration,
    2. CEA position,
    3. Reactor Coolant System average temperature,
    4. Fuel burnup based on gross thermal energy generation,
    5. Xenon concentration, and
    6. Samarium concentration.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the area of Acceptable Operation shown on Figure 3.1-1.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With the moderator temperature coefficient outside the area of Acceptable Operation shown on Figure 3.1-1, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure of 40 EFPD burnup into the current cycle.

c. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure equivalent to two-thirds of the expected current cycle end-of-cycle core average burnup.

---

*With Keff greater than or equal to 1.0.

#See Special Test Exception 3.10.2.

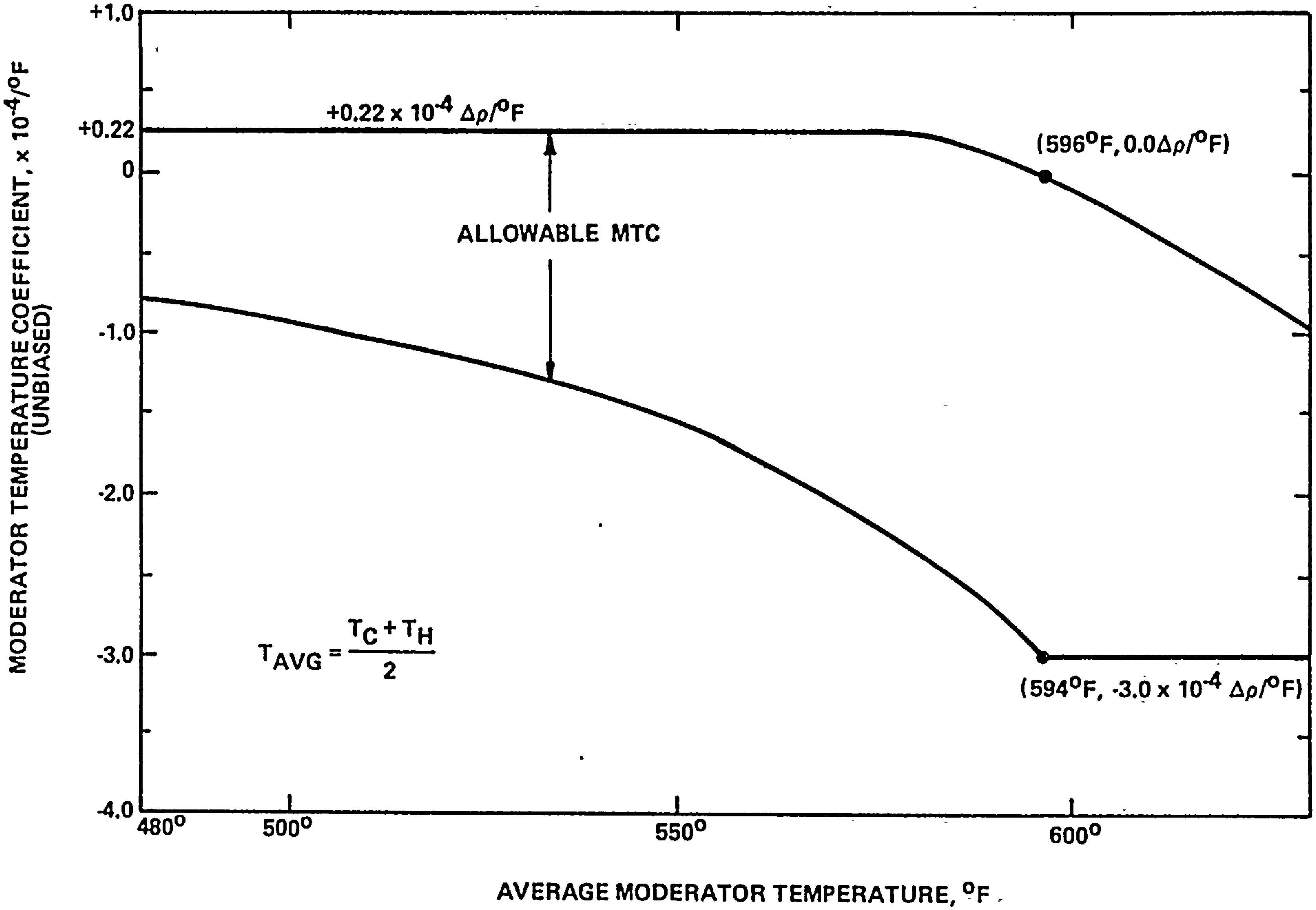



FIGURE 3.1-1

ALLOWABLE MTC MODES 1 AND 2 PALO VERDE UNIT 3 CYCLE 1

## MINIMUM TEMPERATURE FOR CRITICALITY

### LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature ($T_{cold}$) shall be greater than or equal to 552°F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature ($T_{cold}$) less than 552°F, restore $T_{cold}$ to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

### SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature ($T_{cold}$) shall be determined to be greater than or equal to 552°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System $T_{cold}$ is less than 557°F.

#With $K_{eff}$ greater than or equal to 1.0.

*See Special Test Exception 3.10.5.

## 3/4.1.2 BORATION SYSTEMS

### FLOW PATHS - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. If only the spent fuel pool in Specification 3.1.2.5a. is OPERABLE, a flow path from the spent fuel pool via a gravity feed connection and a charging pump to the Reactor Coolant System.

b. If only the refueling water tank in Specification 3.1.2.5b. is OPERABLE, a flow path from the refueling water tank via either a charging pump, a high pressure safety injection pump, or a low pressure safety injection pump to the Reactor Coolant System.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

### SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

## FLOW PATHS - OPERATING

### LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. A gravity feed flow path from either the refueling water tank or the spent fuel pool through CH-536 (RWT Gravity Feed Isolation Valve) and a charging pump to the Reactor Coolant System,

b. A gravity feed flow path from the refueling water tank through CH-327 (RWT Gravity Feed/Safety Injection System Isolation Valve) and a charging pump to the Reactor Coolant System,

c. A flow path from either the refueling water tank or the spent fuel pool through CH-164 (Boric Acid Filter Bypass Valve), utilizing gravity feed and a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

### SURVEILLANCE REQUIREMENTS

4.1.2.2.1 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months when the Reactor Coolant System is at normal operating pressure by verifying that the flow path required by Specification 3.1.2.2 delivers at least 26 gpm for 1 charging pump and 68 gpm for two charging pumps to the Reactor Coolant System.

4.1.2.2.2 The provisions of Specification 4.0.4 are not applicable for entry into Mode 3 or Mode 4 to perform the surveillance testing of Specification 4.1.2.2.1.b provided the testing is performed within 24 hours after achieving normal operating pressure in the reactor coolant system.

## CHARGING PUMPS - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump* or one high pressure safety injection pump or one low pressure safety injection pump in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump or low pressure safety injection pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

### SURVEILLANCE REQUIREMENTS

4.1.2.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*Whenever the reactor coolant level is below the bottom of the pressurizer in MODE 5, one and only one charging pump shall be OPERABLE, by verifying at least once per every 7 days that power is removed from the remaining charging pumps.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 No additional Surveillance Requirements other than those required by Specification 4.0.5.

## BORATED WATER SOURCES - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. The spent fuel pool with:

1. A minimum borated water volume of 33,500 gallons and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume of 33,500 gallons and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 5* and 6*.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

### SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water sources shall be demonstrated OPERABLE:

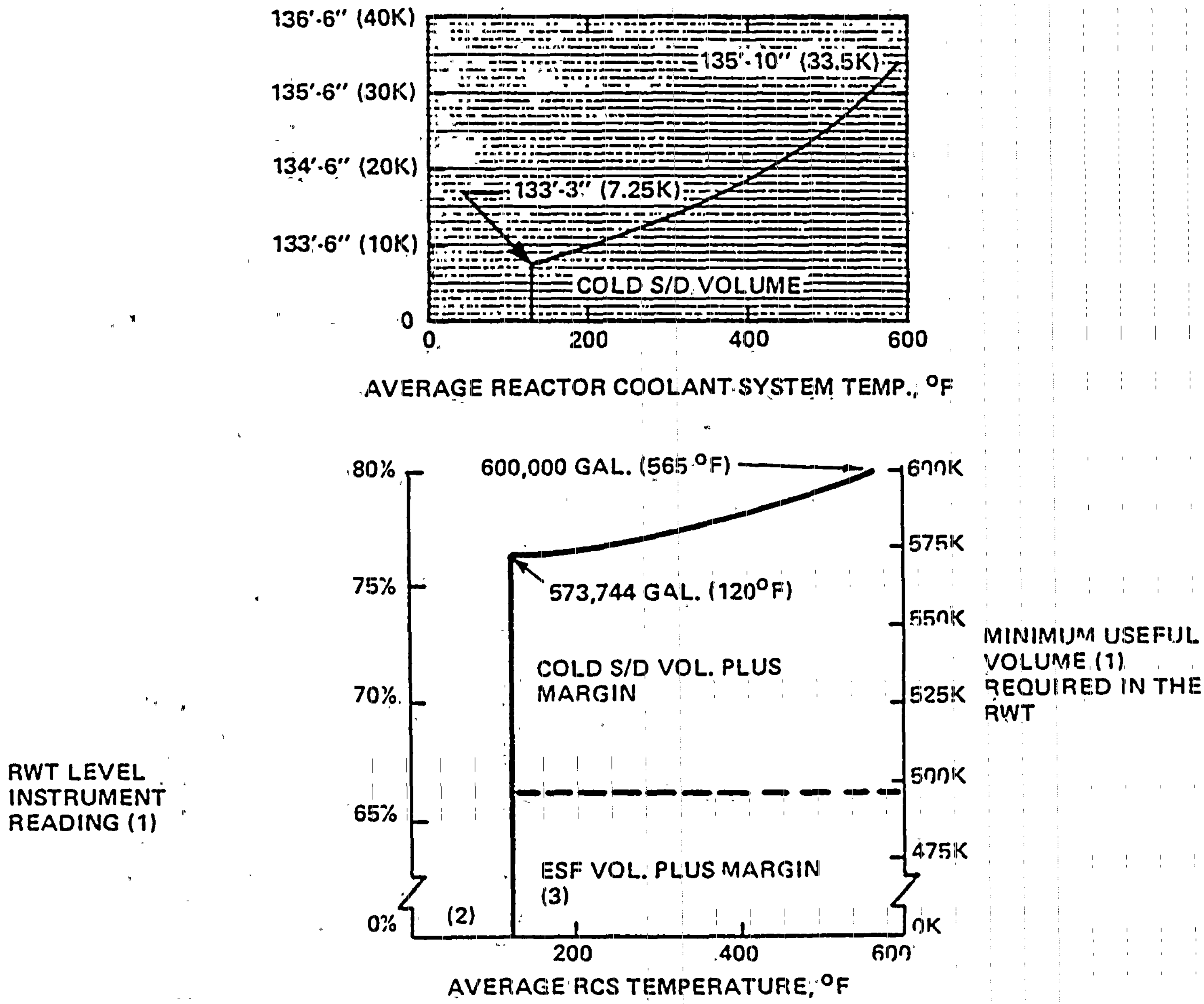
a. At least once per 7 days by:

1. Verifying the boron concentration of the water, and
2. Verifying the contained borated water volume of the refueling water tank or the spent fuel pool.

b. At least once per 24 hours by verifying the refueling water tank temperature when it is the source of borated water and the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when it is the source of borated water and irradiated fuel is present in the pool.

*See Special Test Exception 3.10.7.




(1) THE TANK LEVEL AND VOLUME SHOWN ARE THE USEFUL LEVEL AND VOLUME ABOVE THAT IN THE TANK WHICH IS REQUIRED FOR VORTEX CONSIDERATIONS

(2) DURING MODE 5 AND 6 ONE OF THESE BORATED SOURCES SHALL CONTAIN A MINIMUM OF 33,500 GALLONS

(3) THIS VOLUME IS NOT REQUIRED DURING MODE 6

FIGURE 3.1-2
MINIMUM BORATED WATER VOLUMES

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 Each of the following borated water sources shall be OPERABLE:

a. The spent fuel pool with:

1. A minimum borated water volume as specified in Figure 3.1-2, and

2. A boron concentration of between 4000 ppm and 4400 ppm boron, and

3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume as specified in Figure 3.1-2, and

2. A boron concentration of between 4000 and 4400 ppm of boron, and

3. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 1, 2,* 3,* and 4*.

ACTION:

a. With the above required spent fuel pool inoperable, restore the pool to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F, restore the above required spent fuel pool to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each of the above required borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water, and

2. Verifying the contained borated water volume of the water source.

b. At least once per 24 hours by verifying the refueling water tank temperature when the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when irradiated fuel is present in the pool.

*See Special Test Exception 3.10.7.

## BORON DILUTION ALARMS

### LIMITING CONDITION FOR OPERATION

3.1.2.7 Both startup channel high neutron flux alarms shall be OPERABLE.

APPLICABILITY: MODES 3*, 4, 5, and 6.

ACTION:

a. With one startup channel high neutron flux alarm inoperable:

1. Determine the RCS boron concentration when entering MODE 3, 4, 5, or 6 or at the time the alarm is determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable monitoring frequency in Tables 3.1-1 through 3.1-5 by either boronometer or RCS sampling.**

b. With both startup channel high neutron flux alarms inoperable:

1. Determine the RCS boron concentration by either boronmeter and RCS sampling** or by independent collection and analysis of two RCS samples when entering Mode 3, 4, or 5 or at the time both alarms are determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable monitoring frequency in Tables 3.1-1 through 3.1-5, as applicable, by either boronmeter and RCS sampling** or by collection and analysis of two independent RCS samples. If redundant determination of RCS boron concentration cannot be accomplished immediately, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the method for determining and confirming RCS boron concentration is restored.

2. When in MODE 5 with the RCS level below the centerline of the hotleg or MODE 6, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one startup channel high neutron flux alarm is restored to OPERABLE status.

c. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.1.2.7 Each startup channel high neutron flux alarm shall be demonstrated OPERABLE by performance of:

---

*Within 1 hour after the neutron flux is within the startup range following a reactor shutdown.

**With one or more reactor coolant pumps (RCP) operating the sample should be obtained from the hot leg. With no RCP operating, the sample should be obtained from the discharge line of the low pressure safety injection (LPSI) pump operating in the shutdown cooling mode.

a. A CHANNEL CHECK:

1. At least once per 12 hours.

2. When initially setting setpoints at the following times:

a) One hour after a reactor trip.
b) After a controlled reactor shutdown: Within 1 hour after the neutron flux is within the startup range in MODE 3.

b. A CHANNEL FUNCTIONAL TEST every 31 days of cumulative operation during shutdown.

# TABLE 3.1-1

## REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $K_{eff} > 0.98$

| OPERATIONAL MODE | Number of Operating Charging Pumps | | | |
|---|---|---|---|---|
| | 0 | 1 | 2 | 3 |
| 3 | 12 hours | 1 hour | ONA | ONA |
| 4 | 12 hours | 1 hour | ONA | ONA |
| 5 RCS filled | 8 hours | 1 hour | ONA | ONA |
| 5 RCS partially drained | ONA | ONA | ONA | ONA |
| 6 | 24 hours | 8 hours | 4 hours | 2 hours |

Note: ONA = operation not allowed

TABLE 3.1-2

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $0.98 \geq K_{eff} > 0.97$

| OPERATIONAL MODE | Number of Operating Charging Pumps | | | |
|---|---|---|---|---|
| | 0 | 1 | 2 | 3 |
| 3 | 12 hours | 2.5 hours | 1 hour | 0.5 hours |
| 4 | 12 hours | 2.5 hours | 1 hour | 0.5 hours |
| 5 RCS filled | 8 hours | 2.5 hours | 1 hour | 0.5 hours |
| 5 RCS partially drained | 8 hours | 0.5 hours | Operation not allowed | |
| 6 | 24 hours | 8 hours | 4 hours | 2 hours |

## TABLE 3.1-3

## REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $0.97 \geq K_{eff} > 0.96$

| OPERATIONAL MODE | Number of Operating Charging Pumps | | | |
|---|---|---|---|---|
| | 0 | 1 | 2 | 3 |
| 3 | 12 hours | 3.5 hours | 1.5 hours | 1 hour |
| 4 | 12 hours | 3.5 hours | 1.5 hours | 1 hour |
| 5 RCS filled | 8 hours | 3.5 hours | 1.5 hours | 1 hour |
| 5 RCS partially drained | 8 hours | 1 hour | Operation not allowed | |
| 6 | 24 hours | 8 hours | 4 hours | 2 hours |

TABLE 3.1-4

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $0.96 \geq K_{eff} > 0.95$

| OPERATIONAL MODE | Number of Operating Charging Pumps 0 | 1 | 2 | 3 |
|---|---|---|---|---|
| 3 | 12 hours | 5 hours | 2 hours | 1 hour |
| 4 | 12 hours | 5 hours | 2 hours | 1 hour |
| 5 RCS filled | 8 hours | 5 hours | 2 hours | 1 hour |
| 5 RCS partially drained | 8 hours | 1.5 hours | Operation not allowed | |
| 6 | 24 hours | 8 hours | 4 hours | 2 hours |

## TABLE 3.1-5

## REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $K_{eff} \leq 0.95$

| OPERATIONAL MODE | Number of Operating Charging Pumps 0 | 1 | 2 | 3 |
|---|---|---|---|---|
| 3 | 12 hours | 6 hours | 3 hours | 1.5 hours |
| 4 | 12 hours | 6 hours | 3 hours | 1.5 hours |
| 5 RCS filled | 8 hours | 6 hours | 3 hours | 1.5 hours |
| 5 RCS partially drained | 8 hours | 2 hours | Operation not allowed | |
| 6 | 24 hours | 8 hours | 4 hours | 2 hours |

## 3/4.1.3 MOVABLE CONTROL ASSEMBLIES

### CEA POSITION

### LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length (shutdown and regulating) CEAs, and all part-length CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 6.6 inches (indicated position) of all other CEAs in its group. In addition, the position of the part length CEAs Groups shall be limited to the insertion limits shown in Figure 3.1-2A.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full-length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

b. With more than one full-length or part-length CEA inoperable or misaligned from any other CEA in its group by more than 19 inches (indicated position), be in at least HOT STANDBY within 6 hours.

c. With one or more full-length or part-length CEAs misaligned from any other CEAs in its group by more than 6.6 inches, operation in MODES 1 and 2 may continue, provided that core power is reduced in accordance with Figure 3.1-2B and that within 1 hour the misaligned CEA(s) is either:

   1. Restored to OPERABLE status within its above specified alignment requirements, or

   2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA(s) inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

      a) Within 1 hour the remainder of the CEAs in the group with the inoperable CEA(s) shall be aligned to within 6.6 inches of the inoperable CEA(s) while maintaining the allowable CEA sequence and insertion limits shown on Figures 3.1-2A, 3.1.3, and 3.1-4; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

*See Special Test Exceptions 3.10.2 and 3.10.4.

ACTION: (Continued)

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within 6 hours.

d. With one full-length CEA inoperable due to causes other than addressed by ACTION a., above, but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

e. With one part-length CEA inoperable and inserted in the core, operation may continue provided the alignment of the inoperable part length CEA is maintained within 6.6 inches (indicated position) of all other part-length CEAs in its group.

f. With part length CEAs inserted beyond insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 2 hours either:

1. Restore the part length CEAs to within their limits, or
2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by part length CEA group position using Figure 3.1-2A.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length and part-length CEA shall be determined to be within 6.6 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when one CEAC is inoperable or when both CEACs are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each full-length CEA not fully inserted and each part-length CEA which is inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 31 days.

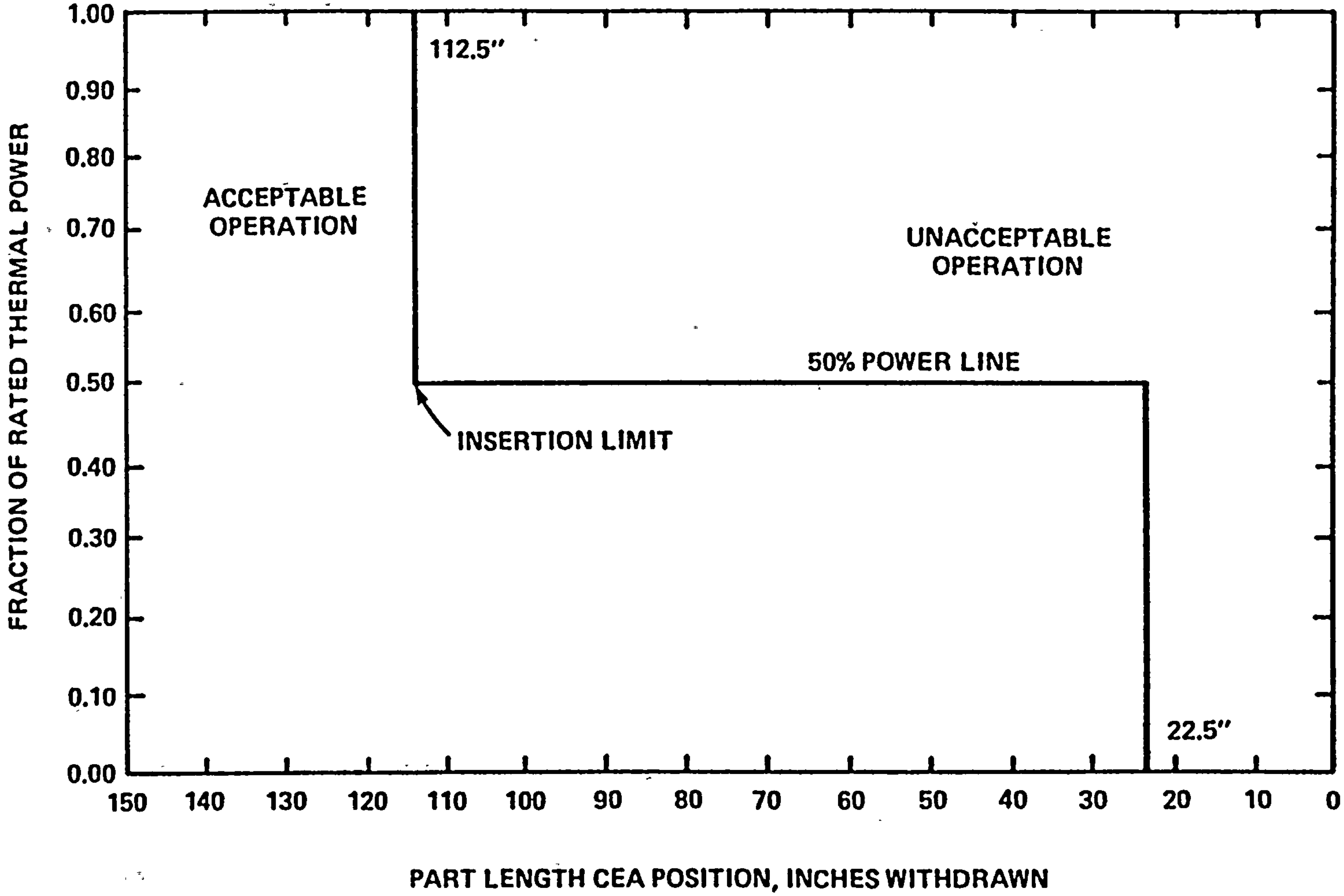
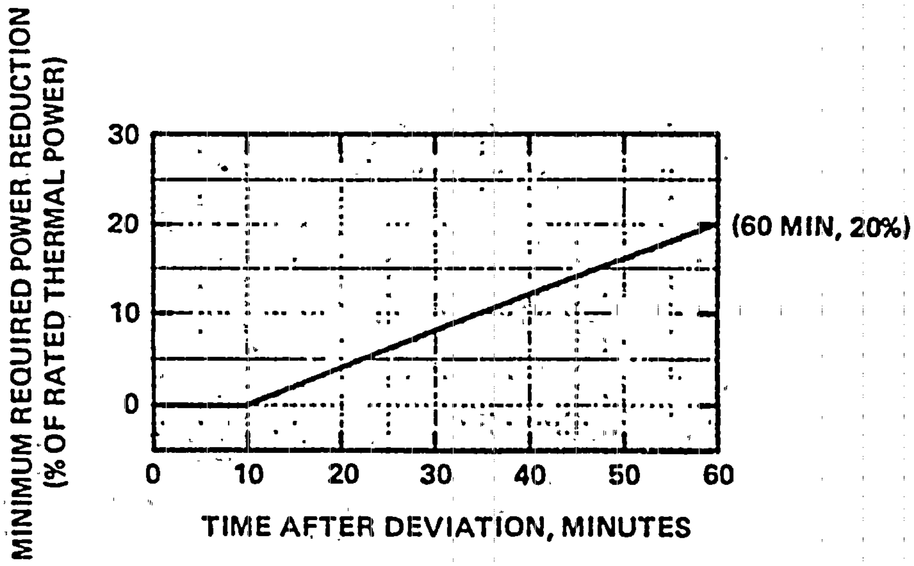



FIGURE 3.1-2A

PART LENGTH CEA INSERTION LIMIT VS. THERMAL POWER




*WHEN CORE POWER IS REDUCED TO 55% OF RATED THERMAL POWER PER THIS LIMIT CURVE, FURTHER REDUCTION IS NOT REQUIRED

FIGURE 3.1-2B

CORE POWER LIMIT AFTER CEA DEVIATION*

POSITION INDICATOR CHANNELS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA:

a. CEA Reed Switch Position Transmitter (RSPT 1) with the capability of determining the absolute CEA positions within 5.2 inches,

b. CEA Reed Switch Position Transmitter (RSPT 2) with the capability of determining the absolute CEA positions within 5.2 inches, and

c. The CEA pulse counting position indicator channel.

APPLICABILITY: MODES 1 and 2.

ACTION:

With a maximum of one CEA per CEA group having only one of the above required CEA position indicator channels OPERABLE, within 6 hours either:

a. Restore the inoperable position indicator channel to OPERABLE status, or

b. Be in at least HOT STANDBY, or

c. Position the CEA group(s) with the inoperable position indicator(s) at its fully withdrawn position while maintaining the requirements of Specifications 3.1.3.1 and 3.1.3.6. Operation may then continue provided the CEA group(s) with the inoperable position indicator(s) is maintained fully withdrawn, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2, and each CEA in the group(s) is verified fully withdrawn at least once per 12 hours thereafter by its "Full Out" limit*.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each of the above required position indicator channels shall be determined to be OPERABLE by verifying that for the same CEA, the position indicator channels agree within 5.2 inches of each other at least once per 12 hours.

*CEAs are fully withdrawn (Full Out) when withdrawn to at least 144.75 inches.

## POSITION INDICATOR CHANNELS - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one CEA Reed Switch Position Transmitter indicator channel shall be OPERABLE for each shutdown, regulating or part-length CEA not fully inserted.

APPLICABILITY: MODES 3*, 4*, and 5*.

ACTION:

With less than the above required position indicator channel(s) OPERABLE, immediately open the reactor trip breakers.

### SURVEILLANCE REQUIREMENTS

4.1.3.3 The above required CEA Reed Switch Position Transmitter indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months.

*With the reactor trip breakers in the closed position.

## CEA DROP TIME

### LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and regulating) CEA drop time, from a fully withdrawn position, shall be less than or equal to 4 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90% insertion position with:

a. $T_{cold}$ greater than or equal to 552°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full-length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

### SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full-length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal and reinstallation of the reactor vessel head,

b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per 18 months.

## SHUTDOWN CEA INSERTION LIMIT

### LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to at least 144.75 inches.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than 144.75 inches, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Withdraw the CEA to at least 144.75 inches, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

### SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to at least 144.75 inches:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exception 3.10.2.

#With $K_{eff}$ greater than or equal to 1.

## REGULATING CEA INSERTION LIMITS

## LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence, and to the insertion limits## shown on Figure 3.1-3** when the COLSS is in service or shown on Figure 3.1-4** when the COLSS is not in service. The CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits is restricted to:

a. Less than or equal to 4 hours per 24 hour interval,

b. Less than or equal to 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. Less than or equal to 14 Effective Full Power Days per 18 Effective Full Power Months.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 2 hours either:

   1. Restore the regulating CEA groups to within the limits, or

   2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position using Figures 3.1-3 or 3.1-4.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 4 hours per 24 hour interval, operation may proceed provided either:

   1. The Short Term Steady State Insertion Limits of Figure 3.1-3 or Figure 3.1-4 are not exceeded, or

   2. Any subsequent increase in THERMAL POWER is restricted to less than or equal to 5% of RATED THERMAL POWER per hour.

*See Special Test Exceptions 3.10.2 and 3.10.4.

#With $K_{eff}$ greater than or equal to 1.

**CEAs are fully withdrawn in accordance with Figure 3.1-3 or Figure 3.1-4 when withdrawn to at least 144.75 inches.

##A reactor power cutback will cause either (Case 1) Regulating Group 5 or Regulating Group 4 and 5 to be dropped with no sequential insertion of additional Regulating Groups (Groups 1, 2, 3, and 4) or (Case 2) Regulating Group 5 or Regulating Group 4 and 5 to be dropped with all or part of the remaining Regulating Groups (Groups 1, 2, 3, and 4) being sequentially inserted. In either case, the Transient Insertion Limit and the withdrawal sequence of Figure 3.1-3 or Figure 3.1-4 can be exceeded for up to 2 hours.

ACTION: (Continued)

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per 18 Effective Full Power Months, either:

   1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within 2 hours, or

   2. Be in at least HOT STANDBY within 6 hours.

## SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted beyond the Long Term Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 hours.

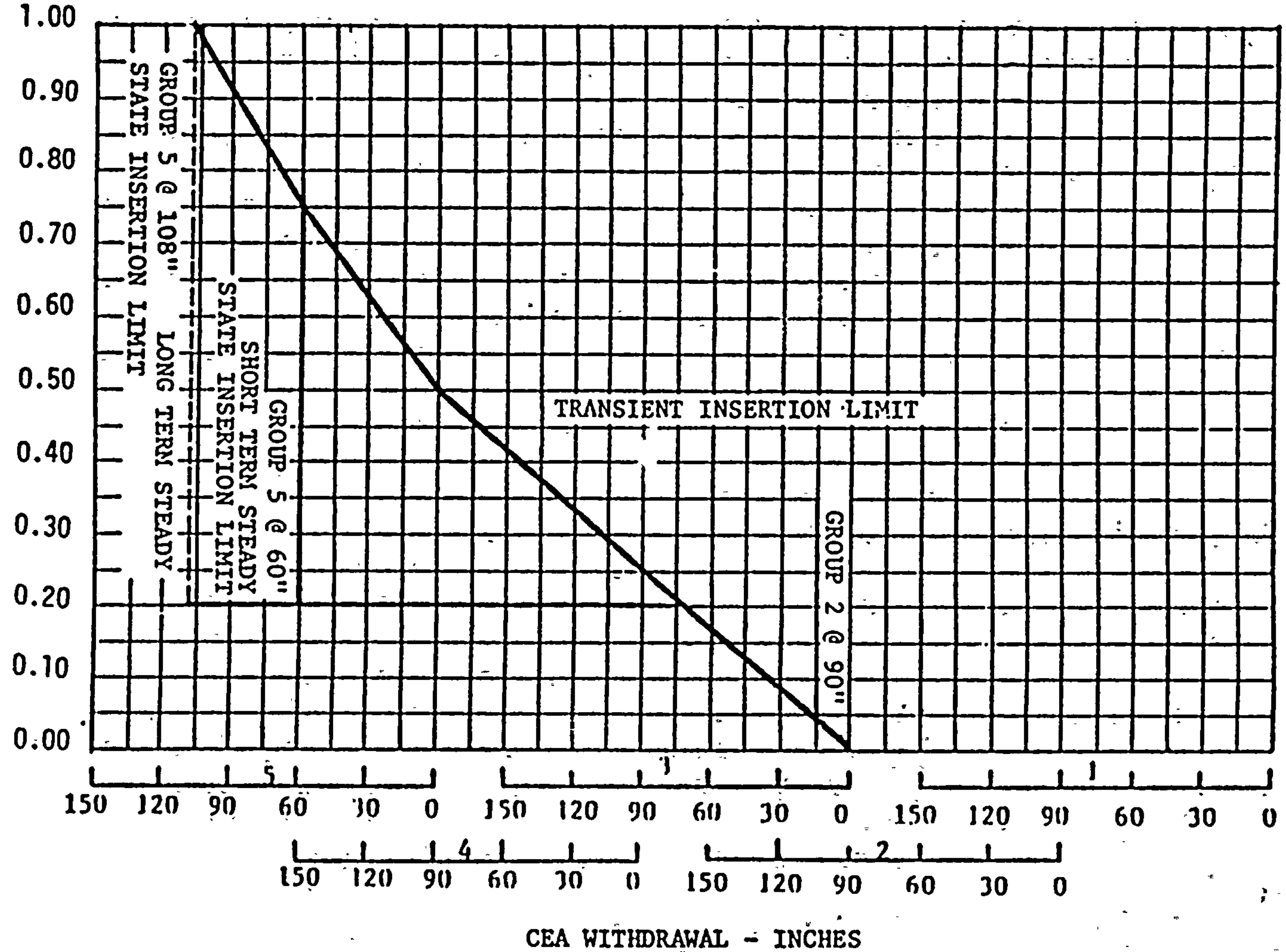



FIGURE 3.1-3

CEA INSERTION LIMITS VS THERMAL POWER (COLSS IN SERVICE)

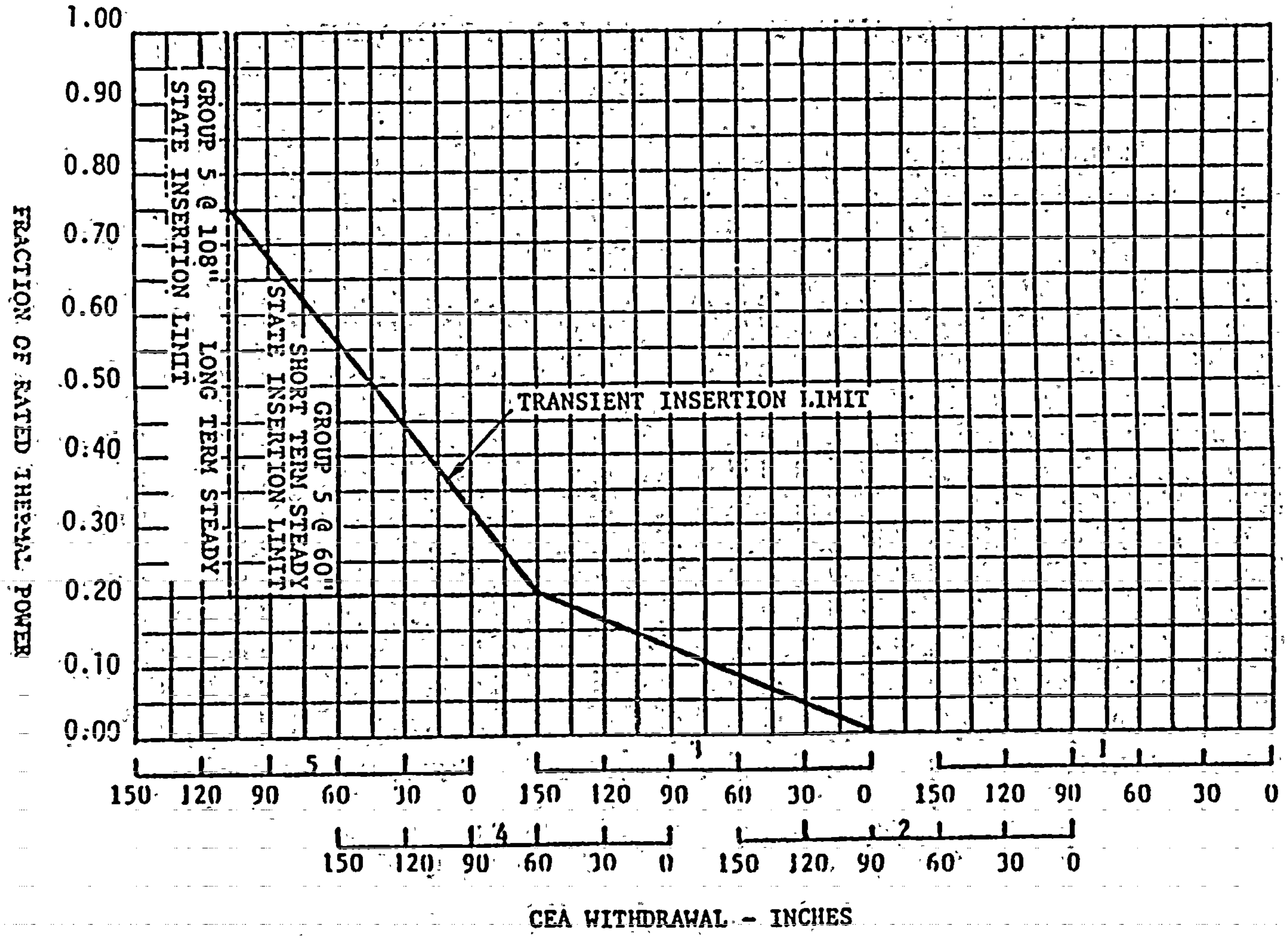



FIGURE 3.1-4

CEA INSERTION LIMITS VS THERMAL POWER (COLSS OUT OF SERVICE)

## 3/4.2 POWER DISTRIBUTION LIMITS

### 3/4 2.1 LINEAR HEAT RATE

#### LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed 14.0 kW/ft.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With the linear heat rate exceeding its limits, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on kW/ft; or (2) when the COLSS is not being used, any OPERABLE Local Power Density channel exceeding the linear heat rate limit, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within 1 hour, or

b. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

#### SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the linear heat rate, as indicated on all OPERABLE Local Power Density channels, is less than or equal to 14.0 kW/ft.

4.2.1.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on 14.0 kW/ft.

## 3/4.2.2 PLANAR RADIAL PEAKING FACTORS - $F_{xy}$

### LIMITING CONDITION FOR OPERATION

3.2.2 The measured PLANAR RADIAL PEAKING FACTORS ($F^m_{xy}$) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS ($F^c_{xy}$) used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPC).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER*.

ACTION:

With an $F^m_{xy}$ exceeding a corresponding $F^c_{xy}$, within 6 hours either:

a. Adjust the CPC addressable constants to increase the multiplier applied to planar radial peaking by a factor equivalent to greater than or equal to $F^m_{xy}/F^c_{xy}$ and restrict subsequent operation so that a margin to the COLSS operating limits of at least $[(F^m_{xy}/F^c_{xy}) - 1.0]$ x 100% is maintained; or

b. Adjust the affected PLANAR RADIAL PEAKING FACTORS ($F^c_{xy}$) used in the COLSS and CPC to a value greater than or equal to the measured PLANAR RADIAL PEAKING FACTORS ($F^m_{xy}$) or

c. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER.

### SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured PLANAR RADIAL PEAKING FACTORS ($F^m_{xy}$) obtained by using the incore detection system, shall be determined to be less than or equal to the PLANAR RADIAL PEAKING FACTORS ($F^c_{xy}$), used in the COLSS and CPC at the following intervals:

a. After each fuel loading with THERMAL POWER greater than 40% but prior to operation above 70% of RATED THERMAL POWER, and

b. At least once per 31 Effective Full Power Days.

---

*See Special Test Exception 3.10.2.

## 3/4.2.3 AZIMUTHAL POWER TILT - $T_q$

### LIMITING CONDITION FOR OPERATION

3.2.3 The AZIMUTHAL POWER TILT ($T_q$) shall be less than or equal to the AZIMUTHAL POWER TILT Allowance used in the Core Protection Calculators (CPCs).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER*.

ACTION:

a. With the measured AZIMUTHAL POWER TILT determined to exceed the AZIMUTHAL POWER TILT Allowance used in the CPCs but less than or equal to 0.10, within 2 hours either correct the power tilt or adjust the AZIMUTHAL POWER TILT Allowance used in the CPCs to greater than or equal to the measured value.

b. With the measured AZIMUTHAL POWER TILT determined to exceed 0.10:

   1. Due to misalignment of either a part-length or full-length CEA, within 30 minutes verify that the Core Operating Limit Supervisory System (COLSS) (when COLSS is being used to monitor the core power distribution per Specifications 4.2.1 and 4.2.4) is detecting the CEA misalignment.

   2. Verify that the AZIMUTHAL POWER TILT is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and verify that the Variable Overpower Trip Setpoint has been reduced as appropriate within the next 4 hours.

   3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the AZIMUTHAL POWER TILT is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception 3.10.2.

## SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit above 20% of RATED THERMAL POWER by:

a. Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.

b. Calculating the tilt at least once per 12 hours when the COLSS is inoperable.

c. Verifying at least once per 31 days, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT less than or equal to the AZIMUTHAL POWER TILT Allowance used in the CPCs.

d. Using the incore detectors at least once per 31 EFPD to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

POWER DISTRIBUTION LIMITS

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the Region of Acceptable Operation of Figure 3.2-1 or 3.2-2, as applicable, or in accordance with the requirements of Action 6 of Table 3.3-1.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel below the DNBR limit, within 15 minutes initiate corrective action to restore either the DNBR core power operating limit or the DNBR to within the limits and either:

a. Restore the DNBR core power operating limit or DNBR to within its limits within 1 hour, or

b. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR margin, as indicated on all OPERABLE DNBR margin channels, is within the limit shown on Figure 3.2-2.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

4.2.4.4 The following DNBR or equivalent penalty factors shall be verified to be included in the COLSS and CPC DNBR calculations at least once per 31 EFPD.

| Burnup $(\frac{GWD}{MTU})$ | DNBR Penalty (%)* |
|---|---|
| 0-10 | 0.5 |
| 10-20 | 1.0 |
| 20-30 | 2.0 |
| 30-40 | 3.5 |
| 40-50 | 5.5 |

*The penalty for each batch will be determined from the batch's maximum burnup assembly and applied to the batch's maximum radial power peak assembly. A single net penalty for COLSS and CPC will be determined from the penalties associated with each batch accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

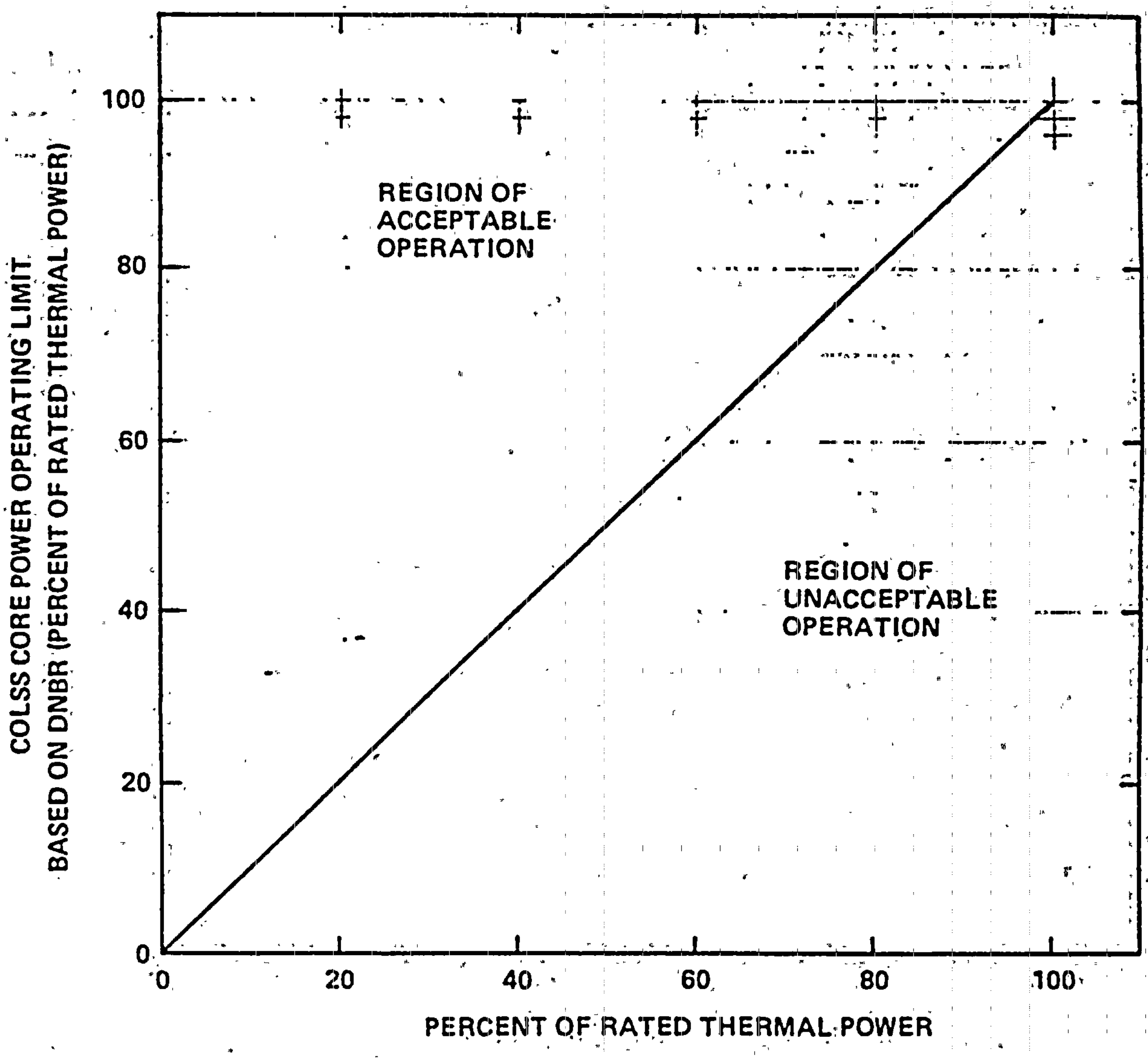
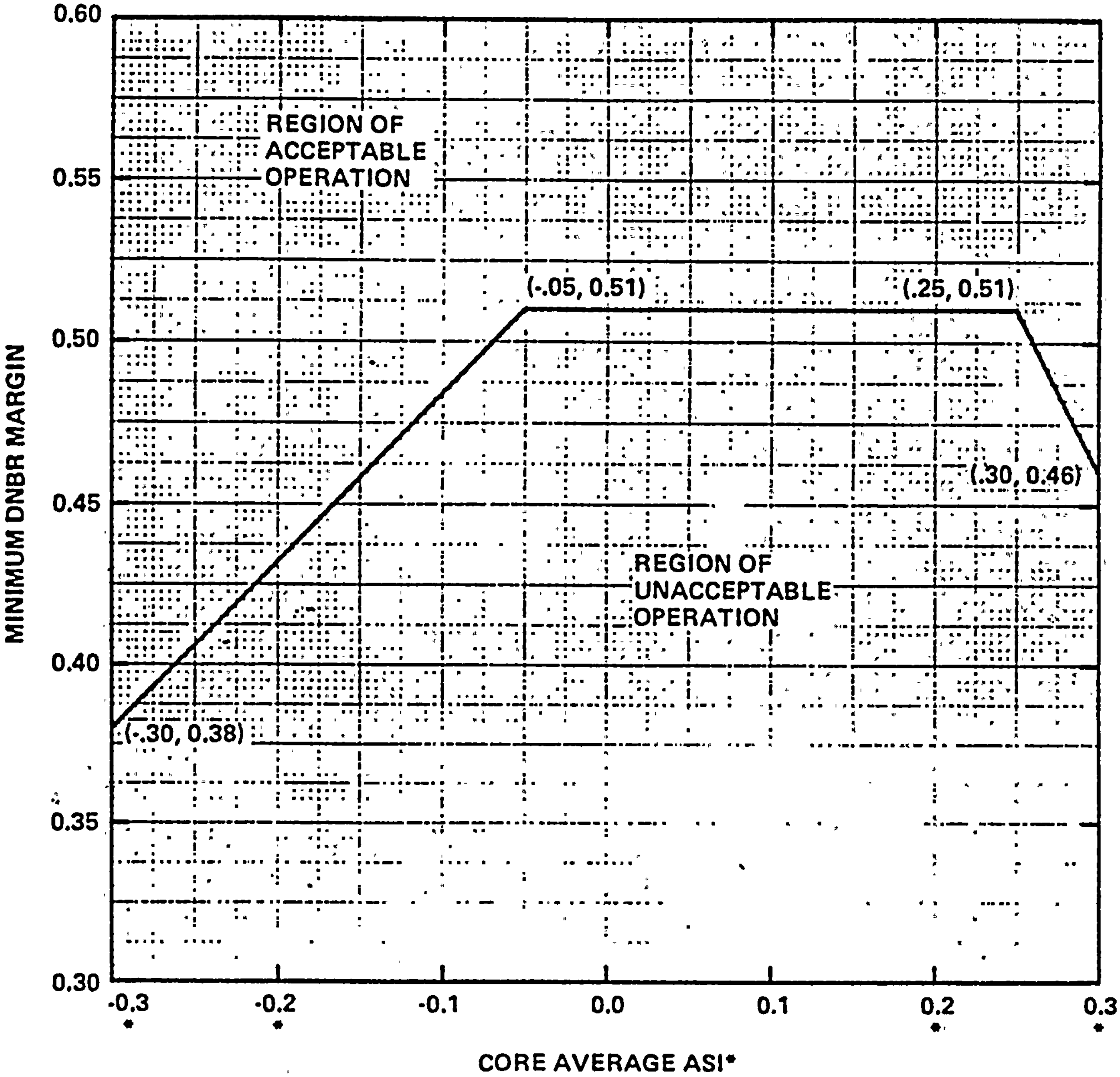



FIGURE 3.2-1

DNBR MARGIN OPERATING LIMIT BASED ON COLSS (COLSS IN SERVICE)




*SEE SECTION 3.2.7 FOR THE ASI OPERATING LIMITS

FIGURE 3.2-2

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS

(COLSS OUT OF SERVICE)

## 3/4.2.5 RCS FLOW RATE

### LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to $164.0 \times 10^6$ lbm/hr.

APPLICABILITY: MODE 1.

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be greater than or equal to its limit at least once per 12 hours.

## 3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

### LIMITING CONDITION FOR OPERATION

3.2.6 The reactor coolant cold leg temperature ($T_c$) shall be within the Area of Acceptable Operation shown in Figure 3.2-3.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the reactor coolant cold leg temperature exceeding its limit, restore the temperature to within its limit within 2 hours or be in HOT STANDBY within the next 6 hours.

### SURVEILLANCE REQUIREMENTS

4.2.6 The reactor coolant cold leg temperature shall be determined to be within its limit at least once per 12 hours.

*See Special Test Exception 3.10.4.
#With $K_{eff}$ greater than or equal to 1

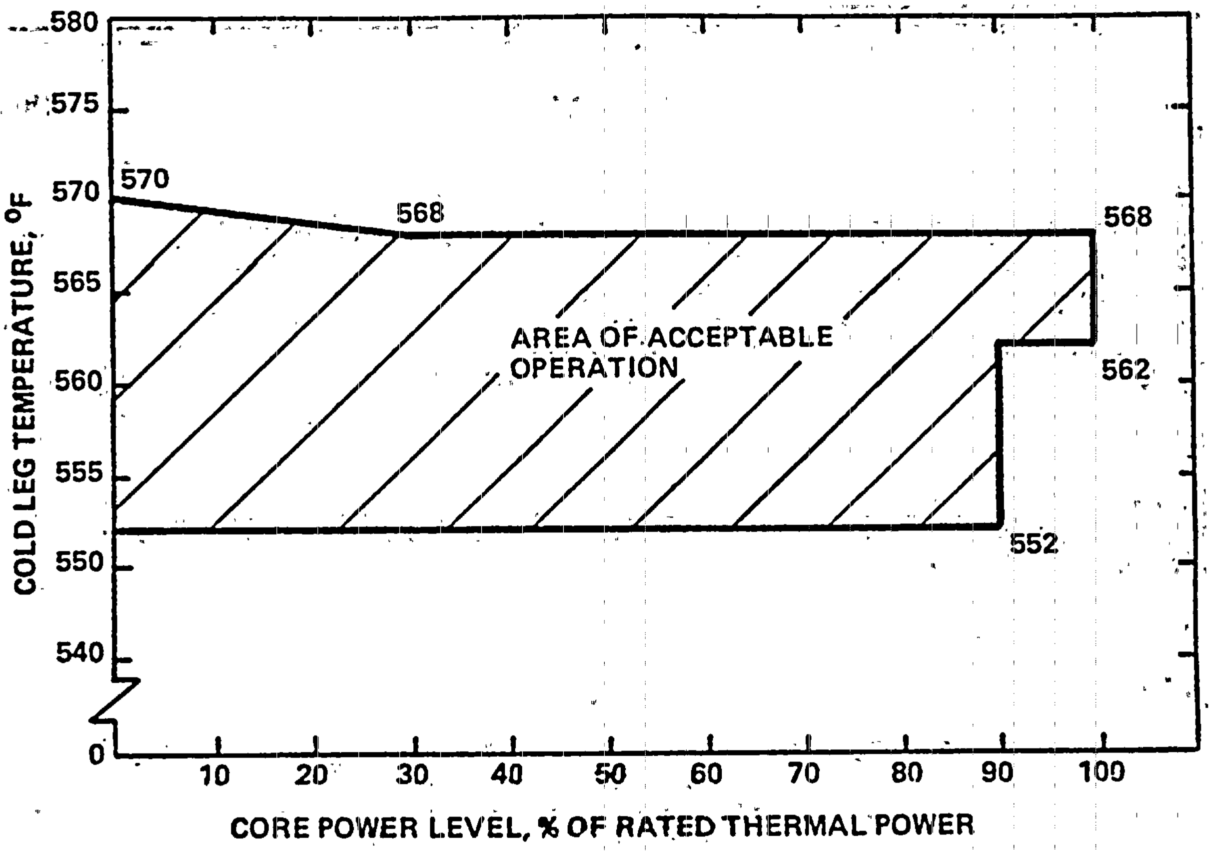



FIGURE 3.2-3

REACTOR COOLANT COLD LEG TEMPERATURE VS. CORE POWER LEVEL

## 3/4.2.7 AXIAL SHAPE INDEX

### LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.28 \leq \text{ASI} \leq 0.28$

b. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq \text{ASI} \leq +0.20$

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER*.

ACTION:

With the core average AXIAL SHAPE INDEX outside its above limits, restore the core average ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limit at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

## 3/4.2.8 PRESSURIZER PRESSURE

### LIMITING CONDITION FOR OPERATION

3.2.8 The pressurizer pressure shall be maintained between 1815 psia and 2370 psia.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the pressurizer pressure outside its above limits, restore the pressure to within its limit within 2 hours or be in at least HOT STANDBY within the next 6 hours.

### SURVEILLANCE REQUIREMENTS

4.2.8 The pressurizer pressure shall be determined to be within its limit at least once per 12 hours.

*See Special Test Exception 3.10.5

## 3/4.3 INSTRUMENTATION

### 3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

#### SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

4.3.1.4 The isolation characteristics of each CEA isolation amplifier shall be verified at least once per 18 months during the shutdown per the following tests for the CEA position isolation amplifiers:

a. With 120 volts A.C. (60 Hz) applied for at least 30 seconds across the output, the reading on the input does not change by more than 0.015 volt D.C. with an applied input voltage of 5-10 volts D.C.

SURVEILLANCE REQUIREMENTS (Continued)

b. With 120 volts A.C. (60 Hz) applied for at least 30 seconds across the input, the reading on the output does not exceed 15 volts D.C.

4.3.1.5 The Core Protection Calculators shall be determined OPERABLE at least once per 12 hours by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours. The auto restart periodic tests Restart (Code 30) and Normal System Load (Code 33) shall not be included in this total.

4.3.1.6 The Core Protection Calculators shall be subjected to a CHANNEL FUNCTIONAL TEST to verify OPERABILITY within 12 hours of receipt of a High CPC Cabinet Temperature alarm.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| I. TRIP GENERATION | | | | | |
| A. Process | | | | | |
| 1. Pressurizer Pressure - High | 4 | 2 | 3 | 1, 2 | $2^{\#}$, $3^{\#}$ |
| 2. Pressurizer Pressure - Low | 4 | 2 (b) | 3 | 1, 2 | $2^{\#}$, $3^{\#}$ |
| 3. Steam Generator Level - Low | 4/SG | 2/SG | 3/SG | 1, 2 | $2^{\#}$, $3^{\#}$ |
| 4. Steam Generator Level - High | 4/SG | 2/SG | 3/SG | 1, 2 | $2^{\#}$, $3^{\#}$ |
| 5. Steam Generator Pressure - Low | 4/SG | 2/SG | 3/SG | 1, 2, 3*, 4* | $2^{\#}$, $3^{\#}$ |
| 6. Containment Pressure - High | 4 | 2 | 3 | 1, 2 | $2^{\#}$, $3^{\#}$ |
| 7. Reactor Coolant Flow - Low | 4/SG | 2/SG | 3/SG | 1, 2 | $2^{\#}$, $3^{\#}$ |
| 8. Local Power Density - High | 4 | 2 (c)(d) | 3 | 1, 2 | $2^{\#}$, $3^{\#}$ |
| 9. DNBR - Low | 4 | 2 (c)(d) | 3 | 1, 2 | $2^{\#}$, $3^{\#}$ |
| B. Excore Neutron Flux | | | | | |
| 1. Variable Overpower Trip | 4 | 2 | 3 | 1, 2 | $2^{\#}$, $3^{\#}$ |
| 2. Logarithmic Power Level - High | | | | | |
| a. Startup and Operating | 4 | 2 (a)(d) | 3 | 1, 2 | $2^{\#}$, $3^{\#}$ |
| | 4 | 2 | 3 | 3*, 4*, 5* | 8 |
| b. Shutdown | 4 | 0 | 2 | 3, 4, 5 | 4 |
| C. Core Protection Calculator System | | | | | |
| 1. CEA Calculators | 2 | 1 | 2 (e) | 1, 2 | 6, 7 |
| 2. Core Protection Calculators | 4 | 2 (c)(d) | 3 | 1, 2 | $2^{\#}$, $3^{\#}$, 7 |

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| D. Supplementary Protection System | | | | | |
| Pressurizer Pressure - High | 4 (f) | 2 | 3 | 1, 2 | 8 |
| II. RPS LOGIC | | | | | |
| A. Matrix Logic | 6 | 1 | 3 | 1, 2 | 1 |
| | 6 | 1 | 3 | 3*, 4*, 5* | 8 |
| B. Initiation Logic | 4 | 2 | 4 | 1, 2 | 5 |
| | 4 | 2 | 4 | 3*, 4*, 5* | 8 |
| III. RPS ACTUATION DEVICES | | | | | |
| A. Reactor Trip Breaker | 4 (f) | 2 | 4 | 1, 2 | 5 |
| | 4 (f) | 2 | 4 | 3*, 4*, 5* | 8 |
| B. Manual Trip | 4 (f) | 2 | 4 | 1, 2 | 5 |
| | 4 (f) | 2 | 4 | 3*, 4*, 5* | 8 |

## TABLE 3.3-1 (Continued)

## REACTOR PROTECTIVE INSTRUMENTATION

### TABLE NOTATIONS

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above $10^{-4}$% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to $10^{-4}$% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(c) Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) There are four channels, each of which is comprised of one of the four reactor trip breakers, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

### ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6.g. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

ACTION STATEMENTS

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

| Process Measurement Circuit | Functional Unit Bypassed/Tripped |
|---|---|
| 1. Linear Power (Subchannel or Linear) | Variable Overpower (RPS)<br>Local Power Density - High (RPS)<br>DNBR - Low (RPS) |
| 2. Pressurizer Pressure - High (Narrow Range) | Pressurizer Pressure - High (RPS)<br>Local Power Density - High (RPS)<br>DNBR - Low (RPS) |
| 3. Steam Generator Pressure - Low | Steam Generator Pressure - Low<br>Steam Generator Level 1-Low (ESF)<br>Steam Generator Level 2-Low (ESF) |
| 4. Steam Generator Level - Low (Wide Range) | Steam Generator Level - Low (RPS)<br>Steam Generator Level 1-Low (ESF)<br>Steam Generator Level 2-Low (ESF) |
| 5. Core Protection Calculator | Local Power Density - High (RPS)<br>DNBR - Low (RPS) |

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

| Process Measurement Circuit | Functional Unit Bypassed/Tripped |
|---|---|
| 1. Linear Power (Subchannel or Linear) | Variable Overpower (RPS)<br>Local Power Density - High (RPS)<br>DNBR - Low (RPS) |
| 2. Pressurizer Pressure - High (Narrow Range) | Pressurizer Pressure - High (RPS)<br>Local Power Density - High (RPS)<br>DNBR - Low (RPS) |

REACTOR PROTECTIVE INSTRUMENTATION

ACTION STATEMENTS

| | | |
|---|---|---|
| 3. | Steam Generator Pressure - Low | Steam Generator Pressure - Low<br>Steam Generator Level 1-Low (ESF)<br>Steam Generator Level 2-Low (ESF) |
| 4. | Steam Generator Level - Low (Wide Range) | Steam Generator Level - Low (RPS)<br>Steam Generator Level 1-Low (ESF)<br>Steam Generator Level 2-Low (ESF) |
| 5. | Core Protection Calculator | Local Power Density - High (RPS)<br>DNBR - Low (RPS) |

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breaker of the inoperable channel is placed in the tripped condition within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; however, the trip breaker associated with the inoperable channel may be closed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

ACTION 6 - a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 6.6 inches (indicated position) of all other CEAs in its group. After 7 days, operation may continue provided that the conditions of Action Item 6.b or 6.c are met.

b. With both CEACs inoperable and COLSS in service, operation may continue provided that:

1. Within 1 hour:

   a) Operation is restricted to the limits shown in Figure 3.3-1. The DNBR margin required by Specification 3.2.4 is replaced by this restriction when both CEAC's are inoperable and COLSS is in operation.

   b) The Linear Heat Rate Margin required by Specification 3.2.1 is maintained.

   c) The Reactor Power Cutback System is placed out of service.

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

ACTION STATEMENTS

2. Within 4 hours:

   a) All full-length and part-length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to indicate that both CEAC's are inoperable.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full-length and part-length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

4. Following a CEA misalignment with both CEAC's inoperable and COLSS in operation, operation may continue provided that within 1 hour:

   The power is reduced to 85% of the pre-misaligned power but need not be reduced to less than 50% of RATED THERMAL POWER. This power restriction replaces the power restriction of Specification 3.1.3.1, Figure 3.1-2B, otherwise Specification 3.1.3.1 remains applicable.

c. With both CEACs inoperable and COLSS out-of-service, operation may continue provided that:

   1. Within 1 hour:

      a) The existing CPC value of the CPC addressable constant "BERR1" is multiplied by 1.19 and the resulting value is re-entered into the CPCs.

      b) The Reactor Power Cutback System is placed out of service

      c) The COLSS out of service Limit Line, on Figure 3.2-2 of Specification 3.2.4, is not applicable to this mode of operation.

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

ACTION STATEMENTS

2. Within 4 hours:

   a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to indicate that both CEAC's are inoperable.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

4. Following a CEA misalignment with both CEAC's and COLSS inoperable, operation may continue provided that within 1 hour:

   The power is reduced to 85% of the pre-misaligned power but need not be reduced to less than 50% of RATED THERMAL POWER. This power restriction replaces the power restriction of Specification 3.1.3.1, Figure 3.1-2B, otherwise Specification 3.1.3.1 remains applicable.

ACTION 7 - With three or more auto restarts, excluding periodic auto restarts (Code 30 and Code 33), of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore an inoperable channel to OPERABLE status within 48 hours or open an affected reactor trip breaker within the next hour.

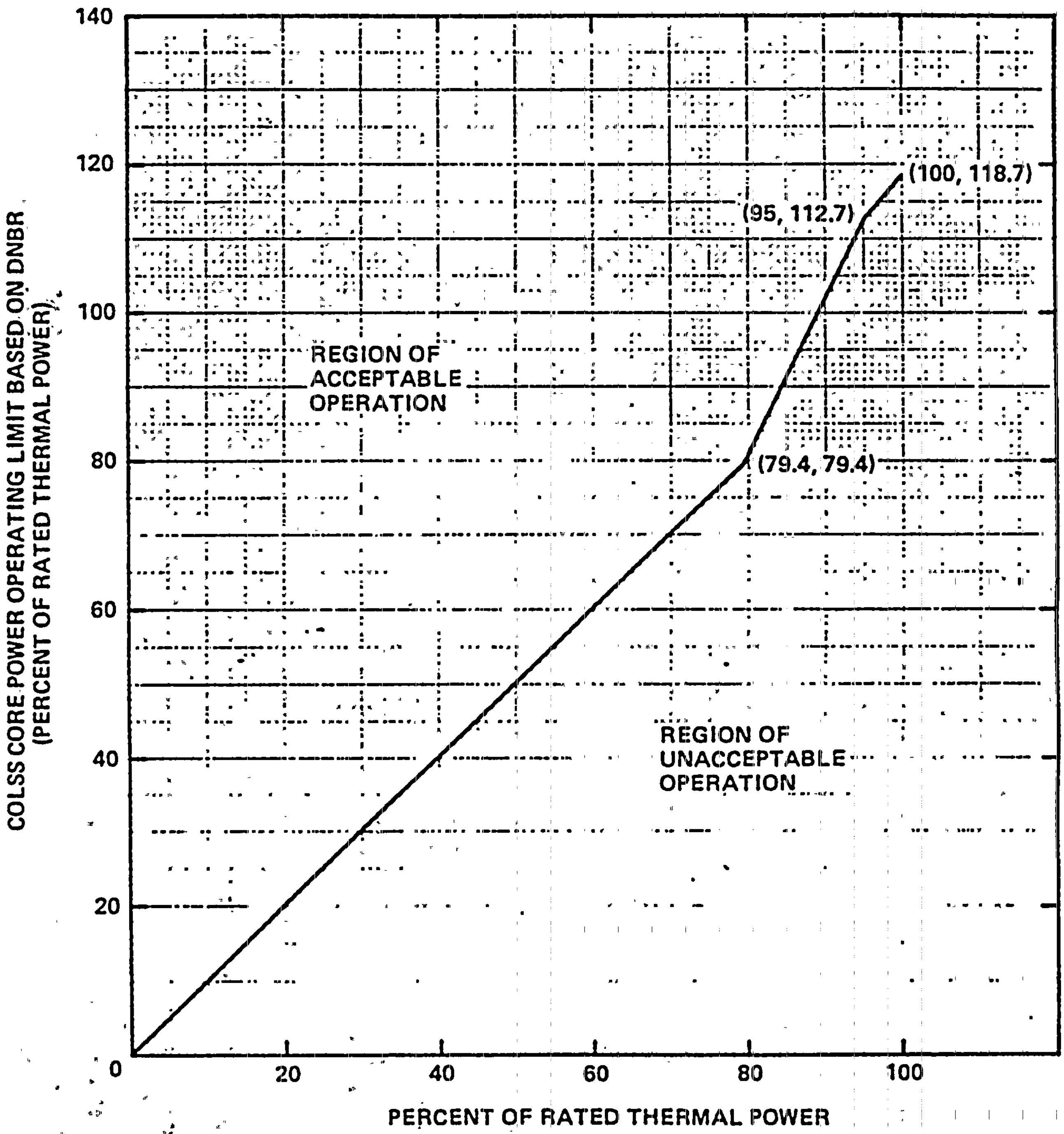



FIGURE 3.3-1

DNBR MARGIN OPERATING LIMIT BASED ON COLSS

FOR BOTH CEACs INOPERABLE

## TABLE 3.3-2

## REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

| FUNCTIONAL UNIT | RESPONSE TIME |
|---|---|
| I. TRIP GENERATION | |
| A. Process | |
| 1. Pressurizer Pressure - High | $\leq$ 1.15 seconds |
| 2. Pressurizer Pressure - Low | $\leq$ 1.15 seconds |
| 3. Steam Generator Level - Low | $\leq$ 1.15 seconds |
| 4. Steam Generator Level - High | $\leq$ 1.15 seconds |
| 5. Steam Generator Pressure - Low | $\leq$ 1.15 seconds |
| 6. Containment Pressure - High | $\leq$ 1.15 seconds |
| 7. Reactor Coolant Flow - Low | $\leq$ 0.58 second |
| 8. Local Power Density - High | |
| a. Neutron Flux Power from Excore Neutron Detectors | $\leq$ 0.75 second* |
| b. CEA Positions | $\leq$ 1.35 second** |
| c. CEA Positions: CEAC Penalty Factor | $\leq$ 0.75 second** |
| 9. DNBR - Low | |
| a. Neutron Flux Power from Excore Neutron Detectors | $\leq$ 0.75 second* |
| b. CEA Positions | $\leq$ 1.35 second** |
| c. Cold Leg Temperature | $\leq$ 0.75 second## |
| d. Hot Leg Temperature | $\leq$ 0.75 second## |
| e. Primary Coolant Pump Shaft Speed | $\leq$ 0.75 second# |
| f. Reactor Coolant Pressure from Pressurizer | $\leq$ 0.75 second### |
| g. CEA Positions: CEAC Penalty Factor | $\leq$ 0.75 second** |
| B. Excore Neutron Flux | |
| 1. Variable Overpower Trip | $\leq$ 0.55 second* |
| 2. Logarithmic Power Level - High | |
| a. Startup and Operating | $\leq$ 0.55 second* |
| b. Shutdown | $\leq$ 0.55 second* |

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

| FUNCTIONAL UNIT | RESPONSE TIME |
|---|---|
| C. Core Protection Calculator System | |
| 1. CEA Calculators | Not Applicable |
| 2. Core Protection Calculators | Not Applicable |
| D. Supplementary Protection System | |
| Pressurizer Pressure - High | ≤ 1.15 second |
| II. RPS LOGIC | |
| A. Matrix Logic | Not Applicable |
| B. Initiation Logic | Not Applicable |
| III. RPS ACTUATION DEVICES | |
| A. Reactor Trip Breakers | Not Applicable |
| B. Manual Trip | Not Applicable |

*Neutron detectors are exempt from response time testing. The response time of the neutron flux signal portion of the channel shall be measured from the detector output or from the input of first electronic component in channel.

**Response time shall be measured from the output of the sensor. Acceptable CEA sensor response shall be demonstrated by compliance with Specification 3.1.3.4.

#The pulse transmitters measuring pump speed are exempt from response time testing. The response time shall be measured from the pulse shaper input.

##Response time shall be measured from the output of the resistance temperature detector (sensor). RTD response time shall be measured at least once per 18 months. The measured response time of the slowest RTD shall be less than or equal to 13 seconds. Adjustments to the CPC addressable constants given in Table 3.3-2a shall be made to accommodate current values of the RTD time constants. If the RTD time constant for a CPC channel exceeds the value corresponding to the penalties currently in use, the affected channel(s) shall be declared inoperable until penalties appropriate to the new time constant are installed.

###Response time shall be measured from the output of the pressure transmitter. The transmitter response time shall be less than or equal to 0.7 second.

## TABLE 3.3-2a

## INCREASES IN BERR0, BERR2, AND BERR4 VERSUS RTD DELAY TIMES

| RTD DELAY TIME ($\tau$) | BERR0 INCREASE (%) | BERR2 INCREASE (%) | BERR4 INCREASE (%) |
|---|---|---|---|
| $\tau \le 8.0$ sec | 0 | 0 | 0 |
| $8.0$ sec $< \tau \le 10.0$ sec | 2.5 | 2.0 | 1.0 |
| $10.0$ sec $< \tau \le 13.0$ sec | 6.0 | 4.0 | 6.0 |

NOTE: BERR term increases are not cumulative. For example, if the time constant changes from the range of $8.0 < \tau \le 10.0$ sec to the range $10.0 < \tau \le 13.0$, the BERR0 increase from its original ($\tau \le 8.0$ sec) value is 6.0 not 2.5 + 6.0.

## TABLE 4.3-1

## REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| I. TRIP GENERATION | | | | |
| A. Process | | | | |
| 1. Pressurizer Pressure - High | S | R | M | 1, 2 |
| 2. Pressurizer Pressure - Low | S | R | M | 1, 2 |
| 3. Steam Generator Level - Low | S | R | M | 1, 2 |
| 4. Steam Generator Level - High | S | R | M | 1, 2 |
| 5. Steam Generator Pressure - Low | S | R | M | 1, 2, 3*, 4* |
| 6. Containment Pressure - High | S | R | M | 1, 2 |
| 7. Reactor Coolant Flow - Low | S | R | M | 1, 2 |
| 8. Local Power Density - High | S | D (2, 4), R (4, 5) | M, R (6) | 1, 2 |
| 9. DNBR - Low | S | D (2, 4), R (4, 5) M (8), S (7) | M, R (6) | 1, 2 |
| B. Excore Neutron Flux | | | | |
| 1. Variable Overpower Trip | S | D (2, 4), M (3, 4) Q (4) | M | 1, 2 |
| 2. Logarithmic Power Level - High | S | R (4) | M and S/U (1) | 1, 2, 3, 4, 5 and * |
| C. Core Protection Calculator System | | | | |
| 1. CEA Calculators | S | R | M, R (6) | 1, 2 |
| 2. Core Protection Calculators | S | D (2, 4), R (4, 5) M (8), S (7) | M (9), R (6) | 1, 2 |

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| D. Supplementary Protection System | | | | |
| Pressurizer Pressure - High | S | R | M | 1, 2 |
| II. RPS LOGIC | | | | |
| A. Matrix Logic | N.A. | N.A. | M | 1, 2, 3*, 4*, 5* |
| B. Initiation Logic | N.A. | N.A. | M | 1, 2, 3*, 4*, 5* |
| III. RPS ACTUATION DEVICES | | | | |
| A. Reactor Trip Breakers | N.A. | N.A. | M, R(10) | 1, 2, 3*, 4*, 5* |
| B. Manual Trip | N.A. | N.A. | M | 1, 2, 3*, 4*, 5* |

## TABLE 4.3-1 (Continued)

## REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

### TABLE NOTATIONS

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

(1) - Each STARTUP or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the linear power level, the CPC delta T power and CPC nuclear power signals to agree with the calorimetric calculation if absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differentral pressure instrumentation and the ultrasonic flow meter adjusted pump curves or calorimetric calculations.

(9) - The monthly CHANNEL FUNCTIONAL TEST shall include verification that the correct (current) values of addressable constants are installed in each OPERABLE CPC.

(10) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage and shunt trips.

INSTRUMENTATION

## 3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

### SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| ESFA SYSTEM FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| I. SAFETY INJECTION (SIAS) | | | | | |
| A. Sensor/Trip Units | | | | | |
| 1. Containment Pressure - High | 4 | 2 | 3 | 1, 2, 3, 4 | 13*, 14* |
| 2. Pressurizer Pressure - Low | 4 | 2 | 3 | 1, 2, 3(a), 4(a) | 13*, 14* |
| B. ESFA System Logic | | | | | |
| 1. Matrix Logic | 6 | 1 | 3 | 1, 2, 3, 4 | 17 |
| 2. Initiation Logic | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| 3. Manual SIAS (Trip Buttons) | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| C. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 16 |
| II. CONTAINMENT ISOLATION (CIAS) | | | | | |
| A. Sensor/Trip Units | | | | | |
| 1. Containment Pressure - High | 4 | 2 | 3 | 1, 2, 3 | 13*, 14*, |
| 2. Pressurizer Pressure - Low | 4 | 2 | 3 | 1, 2, 3(a) | 13*, 14* |
| B. ESFA System Logic | | | | | |
| 1. Matrix Logic | 6 | 1 | 3 | 1, 2, 3 | 17 |
| 2. Initiation Logic | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| ESFA SYSTEM FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| II. CONTAINMENT ISOLATION (Continued) | | | | | |
| 3. Manual CIAS (Trip Buttons) | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| 4. Manual SIAS (Trip Buttons) | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| C. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 16 |
| III. CONTAINMENT SPRAY (CSAS) | | | | | |
| A. Sensor/Trip Units | | | | | |
| Containment Pressure -- High - High | 4 | 2 | 3 | 1, 2, 3 | 13*, 14* |
| B. ESFA System Logic | | | | | |
| 1. Matrix Logic | 6 | 1 | 3 | 1, 2, 3 | 17 |
| 2. Initiation Logic | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| 3. Manual CSAS (Trip Buttons) | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| C. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 16 |

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| ESFA SYSTEM FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| IV. MAIN STEAM LINE ISOLATION (MSIS) | | | | | |
| A. Sensor/Trip Units | | | | | |
| 1. Steam Generator Pressure - Low | 4/steam generator | 2/steam generator | 3/steam generator | 1, 2, 3(b), 4(b) | 13*, 14* |
| 2. Steam Generator Level - High | 4/steam generator | 2/steam generator | 3/steam generator | 1, 2, 3, 4 | 13*, 14* |
| 3. Containment Pressure - High | 4 | 2 | 3 | 1, 2, 3, 4 | 13*, 14* |
| B. ESFA System Logic | | | | | |
| 1. Matrix Logic | 6 | 1 | 3 | 1, 2, 3, 4 | 17 |
| 2. Initiation Logic | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| 3. Manual MSIS (Trip Buttons) | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| C. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 16 |

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| ESFA SYSTEM FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| V. RECIRCULATION (RAS) | | | | | |
| A. Sensor/Trip Units | | | | | |
| Refueling Water Storage Tank - Low | 4 | 2 | 3 | 1, 2, 3 | 13*, 14* |
| B. ESFA System Logic | | | | | |
| 1. Matrix Logic | 6 | 1 | 3 | 1, 2, 3 | 17 |
| 2. Initiation Logic | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| 3. Manual RAS | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| C. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 16 |
| VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1) | | | | | |
| A. Sensor/Trip Units | | | | | |
| 1. Steam Generator #1 Level - Low | 4 | 2 | 3 | 1, 2, 3 | 13*, 14* |
| 2. Steam Generator Δ Pressure - SG2 > SG1 | 4 | 2 | 3 | 1, 2, 3 | 13*, 14* |

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| ESFA SYSTEM FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1) (Continued) | | | | | |
| B. ESFA System Logic | | | | | |
| 1. Matrix Logic | 6 | 1 | 3 | 1, 2, 3 | 17 |
| 2. Initiation Logic | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| 3. Manual AFAS | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 15 |
| C. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 16 |
| VII. AUXILIARY FEEDWATER (SG-2)(AFAS-2) | | | | | |
| A. Sensor/Trip Units | | | | | |
| 1. Steam Generator #2 Level - Low | 4 | 2 | 3 | 1, 2, 3 | 13*, 14* |
| 2. Steam Generator Δ Pressure - SG1 > SG2 | 4 | 2 | 3 | 1, 2, 3 | 13*, 14* |
| B. ESFA System Logic | | | | | |
| 1. Matrix Logic | 6 | 1 | 3 | 1, 2, 3 | 17 |
| 2. Initiation Logic | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 12 |
| 3. Manual AFAS | 4(c) | 2(d) | 4 | 1, 2, 3, 4 | 15 |
| C. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 16 |
| VIII. LOSS OF POWER (LOV) | | | | | |
| A. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage) | 4/Bus | 2/Bus | 3/Bus | 1, 2, 3 | 13*, 14* |
| B. 4.16 kV Emergency Bus Under-voltage (Degraded Voltage) | 4/Bus | 2/Bus | 3/Bus | 1, 2, 3 | 13*, 14* |
| IX. CONTROL ROOM ESSENTIAL FILTRATION | 2 | 1 | 1 | All Modes | 18* |

## TABLE 3.3-3 (Continued)

## ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

### TABLE NOTATIONS

(a) In MODES 3-4, the value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(b) In MODES 3-4, the value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(c) Four channels provided, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

(d) The proper two-out-of-four combination.

* The provisions of Specification 3.0.4 are not applicable.

### ACTION STATEMENTS

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6.g. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.

| Process Measurement Circuit | |
|---|---|
| 1. Steam Generator Pressure - Low | Steam Generator Pressure - Low<br>Steam Generator Level 1-Low (ESF)<br>Steam Generator Level 2-Low (ESF) |
| 2. Steam Generator Level (Wide Range) | Steam Generator Level - Low (RPS)<br>Steam Generator Level 1-Low (ESF)<br>Steam Generator Level 2-Low (ESF) |

## TABLE 3.3-3 (Continued)

## ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

## ACTION STATEMENTS

ACTION 14 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

| Process Measurement Circuit | Functional Unit Bypassed/Tripped |
|---|---|
| 1. Steam Generator Pressure - Low | Steam Generator Pressure - Low<br>Steam Generator Level 1 - Low (ESF)<br>Steam Generator Level 2 - Low (ESF) |
| 2. Steam Generator Level - Low (Wide Range) | Steam Generator Level - Low (RPS)<br>Steam Generator Level 1 - Low (ESF)<br>Steam Generator Level 2 - Low (ESF) |

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 13 are satisfied.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE.

ACTION 17 - With the number of OPERABLE channels one less than the Minimum Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 18 - With the number of OPERABLE channels one less than the Minimum Number of Channels, operation may continue for up to 6 hours. After 6 hours operation may continue provided at least 1 train of essential filtration is in operation, otherwise, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

| ESFA SYSTEM FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| I. SAFETY INJECTION (SIAS) | | |
| A. Sensor/Trip Units | | |
| 1. Containment Pressure - High | ≤ 3.0 psig | ≤ 3.2 psig |
| 2. Pressurizer Pressure - Low | ≥ 1837 psia[1] | ≥ 1822 psia[1] |
| B. ESFA System Logic | Not Applicable | Not Applicable |
| C. Actuation Systems | Not Applicable | Not Applicable |
| II. CONTAINMENT ISOLATION (CIAS) | | |
| A. Sensor/Trip Units | | |
| 1. Containment Pressure - High | ≤ 3.0 psig | ≤ 3.2 psig |
| 2. Pressurizer Pressure - Low | ≥ 1837 psia[1] | ≥ 1822 psia[1] |
| B. ESFA System Logic | Not Applicable | Not Applicable |
| C. Actuation Systems | Not Applicable | Not Applicable |
| III. CONTAINMENT SPRAY (CSAS) | | |
| A. Sensor/Trip Units | | |
| Containment Pressure High - High | ≤ 8.5 psig | ≤ 8.9 psig |
| B. ESFA System Logic | Not Applicable | Not Applicable |
| C. Actuation Systems | Not Applicable | Not Applicable |
| IV. MAIN STEAM LINE ISOLATION (MSIS) | | |
| A. Sensor/Trip Units | | |
| 1. Steam Generator Pressure - Low | ≥ 919 psia[3] | ≥ 912 psia[3] |
| 2. Steam Generator Level - High | ≤ 91.0% NR[2] | ≤ 91.5% NR[2] |
| 3. Containment Pressure - High | ≤ 3.0 psig | ≤ 3.2 psig |
| B. ESFA System Logic | Not Applicable | Not Applicable |
| C. Actuation Systems | Not Applicable | Not Applicable |

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

| ESFA SYSTEM FUNCTIONAL UNIT | TRIP VALUES | ALLOWABLE VALUES |
|---|---|---|
| V. RECIRCULATION (RAS) | | |
| A. Sensor/Trip Units | | |
| Refueling Water Storage Tank - Low | $\geq$ 7.4% of Span | 7.9 $\geq$ % of Span $\geq$ 6.9 |
| B. ESFA System Logic | Not Applicable | Not Applicable |
| C. Actuation System | Not Applicable | Not Applicable |
| VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1) | | |
| A. Sensor/Trip Units | | |
| 1. Steam Generator #1 Level - Low | $\geq$ 25.8% WR[4] | $\geq$ 25.3% WR[4] |
| 2. Steam Generator Δ Pressure - SG2 > SG1 | $\leq$ 185 psid | $\leq$ 192 psid |
| B. ESFA System Logic | Not Applicable | Not Applicable |
| C. Actuation Systems | Not Applicable | Not Applicable |
| VII. AUXILIARY FEEDWATER (SG-2)(AFAS-2) | | |
| A. Sensor/Trip Units | | |
| 1. Steam Generator #2 Level - Low | $\geq$ 25.8% WR[4] | $\geq$ 25.3% WR[4] |
| 2. Steam Generator Δ Pressure - SG1 > SG2 | $\leq$ 185 psid | $\leq$ 192 psid |
| B. ESFA System Logic | Not Applicable | Not Applicable |
| C. Actuation Systems | Not Applicable | Not Applicable |
| VIII. LOSS OF POWER | | |
| A. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage) | $\geq$ 3250 volts | $\geq$ 3250 volts |
| B. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage) | 2930 to 3744 volts with a 35-second maximum time delay | 2930 to 3744 volts with a 35-second maximum time delay |
| IX. CONTROL ROOM ESSENTIAL FILTRATION | $\leq 2 \times 10^{-5}$ µCi/cc | $\leq 2 \times 10^{-5}$ µCi/cc |

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(1) In MODES 3-4, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(2) % of the distance between steam generator upper and lower level narrow range instrument nozzles.

(3) In MODES 3-4, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 1. Manual | |
| a. SIAS | |
| Safety Injection (ECCS) | Not Applicable |
| Containment Isolation | Not Applicable |
| Containment Purge Valve Isolation | Not Applicable |
| b. CSAS | |
| Containment Spray | Not Applicable |
| c. CIAS | |
| Containment Isolation | Not Applicable |
| d. MSIS | |
| Main Steam Isolation | Not Applicable |
| e. RAS | |
| Containment Sump Recirculation | Not Applicable |
| f. AFAS | |
| Auxiliary Feedwater Pumps | Not Applicable |

## TABLE 3.3-5 (Continued)

## ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 2. Pressurizer Pressure - Low | |
| a. Safety Injection (HPSI) | ≤ 30*/30** |
| b. Safety Injection (LPSI) | ≤ 30*/30** |
| c. Containment Isolation | |
| 1. CIAS actuated mini-purge valves | ≤ 10.6*/10.6** |
| 2. Other CIAS actuated valves | ≤ 31*/31** |
| 3. Containment Pressure - High | |
| a. Safety Injection (HPSI) | ≤ 30*/30** |
| b. Safety Injection (LPSI) | ≤ 30*/30** |
| c. Containment Isolation | |
| 1. CIAS actuated mini-purge valves | ≤ 10.6*/10.6** |
| 2. Other CIAS actuated valves | ≤ 31*/31** |
| d. Main Steam Isolation | |
| 1. MSIS actuated MSIV's | ≤ 5.6*/5.6** |
| 2. MSIS actuated MFIV's# | ≤ 10.6*/10.6** |
| e. Containment Spray Pump | ≤ 33*/23** |
| 4. Containment Pressure - High-High | |
| a. Containment Spray | ≤ 33*/23** |
| 5. Steam Generator Pressure - Low | |
| a. Main Steam Isolation | |
| 1. MSIS actuated MSIV's | ≤ 5.6*/5.6** |
| 2. MSIS actuated MFIV's# | ≤ 10.6*/10.6** |
| 6. Refueling Water Tank - Low | |
| a. Containment Sump Recirculation | ≤ 45*/45** |
| 7. Steam Generator Level - Low | |
| a. Auxiliary Feedwater (Motor Drive) | ≤ 46*/23** |
| b. Auxiliary Feedwater (Turbine Drive) | ≤ 30*/30** |

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 8. Steam Generator Level - High | |
| a. Main Steam Isolation | |
| 1. MSIS actuated MSIV's | ≤ 5.6*/5.6** |
| 2. MSIS actuated MFIV's# | ≤ 10.6*/10.6** |
| 9. Steam Generator ΔP-High-Coincident With Steam Generator Level Low | |
| a. Auxiliary Feedwater Isolation from the Ruptured Steam Generator | ≤ 16*/16** |
| 10. Control Room Essential Filtration Actuation | ≤ 180*/180**## |
| 11. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage) | |
| Loss of Power 90% system voltage | ≤ 35.0 |
| 12. 4.16 kV Emergency Bus Undervoltage (loss of Voltage) | |
| Loss of Power | ≤ 2.4 |

TABLE NOTATIONS

*Diesel generator starting and sequence loading delays included. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

**Diesel generator starting delays not included. Offsite power available. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

#MFIV valves tested at simulated operating conditions; valves tested at static flow conditions to ≤ 8.6*/8.6** seconds.

##Radiation detectors are exempt from response time testing. The response time of the radiation signal portion of the channel shall be measured from the detector output or from the input of first electronic component in channel to closure of dampers M-HJA-M01, M-HJA-M52, M-HJB-M01 and M-HJB-M55.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| ESFA SYSTEM FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|
| I. SAFETY INJECTION (SIAS) | | | | |
| A. Sensor/Trip Units | | | | |
| 1. Containment Pressure - High | S | R | M | 1, 2, 3, 4 |
| 2. Pressurizer Pressure - Low | S | R | M | 1, 2, 3, 4 |
| B. ESFA System Logic | | | | |
| 1. Matrix Logic | N.A. | N.A. | M | 1, 2, 3, 4 |
| 2. Initiation Logic | N.A. | N.A. | M | 1, 2, 3, 4 |
| 3. Manual SIAS | N.A. | N.A. | M | 1, 2, 3, 4 |
| C. Automatic Actuation Logic | N.A. | N.A. | M(1) (2) (3) | 1, 2, 3, 4 |
| II. CONTAINMENT ISOLATION (CIAS) | | | | |
| A. Sensor/Trip Units | | | | |
| 1. Containment Pressure - High | S | R | M | 1, 2, 3 |
| 2. Pressurizer Pressure - Low | S | R | M | 1, 2, 3 |
| B. ESFA System Logic | | | | |
| 1. Matrix Logic | N.A. | N.A. | M | 1, 2, 3, 4 |
| 2. Initiation Logic | N.A. | N.A. | M | 1, 2, 3, 4 |
| 3. Manual CIAS | N.A. | N.A. | M | 1, 2, 3, 4 |
| 4. Manual SIAS | N.A. | N.A. | M | 1, 2, 3, 4 |

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| ESFA SYSTEM FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|
| II. CONTAINMENT ISOLATION (Continued) | | | | |
| C. Automatic Actuation Logic | N.A. | N.A. | M(1) (2) (3) | 1, 2, 3, 4 |
| III. CONTAINMENT SPRAY (CSAS) | | | | |
| A. Sensor/Trip Units | | | | |
| 1. Containment Pressure -- High - High | S | R | M | 1, 2, 3 |
| B. ESFA System Logic | | | | |
| 1. Matrix Logic | N.A. | N.A. | M | 1, 2, 3, 4 |
| 2. Initiation Logic | N.A. | N.A. | M | 1, 2, 3, 4 |
| 3. Manual CSAS | N.A. | N.A. | M | 1, 2, 3, 4 |
| C. Automatic Actuation Logic | N.A. | N.A. | M(1) (2) (3) | 1, 2, 3, 4 |

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| ESFA SYSTEM FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|
| IV. MAIN STEAM LINE ISOLATION (MSIS) | | | | |
| A. Sensor/Trip Units | | | | |
| 1. Steam Generator Pressure - Low | S | R | M | 1, 2, 3, 4 |
| 2. Steam Generator Level - High | S | R | M | 1, 2, 3, 4 |
| 3. Containment Pressure - High | S | R | M | 1, 2, 3, 4 |
| B. ESFA System Logic | | | | |
| 1. Matrix Logic | N.A. | N.A. | M | 1, 2, 3, 4 |
| 2. Initiation Logic | N.A. | N.A. | M | 1, 2, 3, 4 |
| 3. Manual MSIS | N.A. | N.A. | M | 1, 2, 3, 4 |
| C. Automatic Acutation Logic | N.A. | N.A. | M(1) (2) (3) | 1, 2, 3, 4 |

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| ESFA SYSTEM FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|
| V. RECIRCULATION (RAS) | | | | |
| A. Sensor/Trip Units | | | | |
| Refueling Water Storage Tank - Low | S | R | M | 1, 2, 3 |
| B. ESFA System Logic | | | | |
| 1. Matrix Logic | N.A. | N.A. | M | 1, 2, 3, 4 |
| 2. Initiation Logic | N.A. | N.A. | M | 1, 2, 3, 4 |
| 3. Manual RAS | N.A. | N.A. | M | 1, 2, 3, 4 |
| C. Automatic Acutation Logic | N.A. | N.A. | M(1) (2) (3) | 1, 2, 3, 4 |
| VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1) | | | | |
| A. Sensor/Trip Units | | | | |
| 1. Steam Generator #1 Level - Low | S | R | M | 1, 2, 3 |
| 2. Steam Generator Δ Pressure SG2 > SG1 | S | R | M | 1, 2, 3 |

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| ESFA SYSTEM FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|
| VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1) (Continued) | | | | |
| B. ESFA System Logic | | | | |
| 1. Matrix Logic | N.A. | N.A. | M | 1, 2, 3, 4 |
| 2. Initiation Logic | N.A. | N.A. | M | 1, 2, 3, 4 |
| 3. Manual AFAS | N.A. | N.A. | M | 1, 2, 3, 4 |
| C. Automatic Actuation Logic | N.A. | N.A. | M(1) (2) (3) | 1, 2, 3, 4 |
| VII. AUXILIARY FEEDWATER (SG-2)(AFAS-2) | | | | |
| A. Sensor/Trip Units | | | | |
| 1. Steam Generator #2 Level - Low | S | R | M | 1, 2, 3 |
| 2. Steam Generator Δ Pressure SG1 > SG2 | S | R | M | 1, 2, 3 |
| B. ESFA System Logic | | | | |
| 1. Matrix Logic | N.A. | N.A. | M | 1, 2, 3, 4 |
| 2. Initiation Logic | N.A. | N.A. | M | 1, 2, 3, 4 |
| 3. Manual AFAS | N.A. | N.A. | M | 1, 2, 3, 4 |
| C. Automatic Actuation Logic | N.A. | N.A. | M(1) (2) (3) | 1, 2, 3, 4 |
| VIII. LOSS OF POWER (LOV) | | | | |
| A. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage) | S | R | R | 1, 2, 3, 4 |
| B. 4.16 kV Emergency Bus Under-voltage (Degraded Voltage) | S | R | R | 1, 2, 3, 4 |

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(1) Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) Testing of automatic actuation logic shall include energization/deenergization of each initiation relay and verification of proper operation of each initiation relay.

(3) A subgroup relay test shall be performed which shall include the energization/deenergization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. Relays listed below are exempt from testing during POWER OPERATION but shall be tested at least once per 18 months during REFUELING and during each COLD SHUTDOWN condition unless tested within the previous 62 days.

ACTUATION DEVICES THAT CANNOT BE TESTED AT POWER

| TRAIN A | | TRAIN B | |
|---|---|---|---|
| ESF FUNCTION | ACTUATION DEVICE | ESF FUNCTION | ACTUATION DEVICE |
| SIAS A | K108 | SIAS B | K108 |
| SIAS A | K409 | SIAS B | K409 |
| CIAS A | K202 | CIAS B | K204 |
| CIAS A | K204 | CSAS B | K304 |
| CSAS A | K304 | MSIS B | K305 |
| MSIS A | K305 | MSIS B | K404 |
| MSIS A | K404 | AFAS 1B | K113 |
| AFAS 1A | K211 | AFAS 1B | K211 |
| AFAS 2A | K112 | AFAS 2B | K112 |

In the case of the following relays <u>which are tested during power operation</u>, one or more pieces of equipment cannot be actuated, but can be racked out, bypassed or etc., which will not preclude the relay from being tested but will not actuate the locked out equipment associated with the relay:

| | | | |
|---|---|---|---|
| SIAS A | K401 | SIAS B | K301 |
| SIAS A | K410 | SIAS B | K308 |
| SIAS A | K412 | CIAS B | K203 |
| CIAS A | K203 | CIAS B | K210 |
| CIAS A | K210 | RAS B | K104 |
| RAS A | K104 | RAS B | K312 |
| RAS A | K312 | RAS B | K405 |
| RAS A | K405 | | |
| AFAS 1A | K113 | | |

## 3/4.3.3 MONITORING INSTRUMENTATION

### RADIATION MONITORING INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

| | INSTRUMENT | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ALARM/TRIP SETPOINT | MEASUREMENT RANGE | ACTION |
|---|---|---|---|---|---|---|
| 1. | Area Monitors | | | | | |
| | A. Fuel Pool Area RU-31 | 1 | ** | $\leq$15mR/hr | $10^{-1}$ to $10^{4}$mR/hr | 22 & 24 |
| | B. New Fuel Area RU-19 | 1 | * | $\leq$15mR/hr | $10^{-1}$ to $10^{4}$mR/hr | 22 |
| | C. Containment RU-148 & RU-149 | 2 | 1,2,3,4 | $\leq$10R/hr | 1R/hr to $10^{7}$R/hr | 27 |
| | D. Containment Power Access Purge Exhaust RU-37 & RU-38 | 1 | # | $\leq$2.5mR/hr | $10^{-1}$ to $10^{4}$mR/hr | 25 |
| | E. Main Steam | | | | | |
| | 1) RU-139 A&B | 1 | 1,2,3,4 | ## | $10^{-3}$ to $10^{4}$R/hr | 27 |
| | 2) RU-140 A&B | 1 | 1,2,3,4 | ## | $10^{-3}$ to $10^{4}$R/hr | 27 |
| 2. | Process Monitors | | | | | |
| | A. Containment Building Atmosphere RU-1 | 2 | 1,2,3,4 | | | 23 & 27 |
| | 1) Particulate | | | $\leq 2.3\times10^{-6}$μCi/cc Cs-137 | $10^{-9}$ to $10^{-4}$μCi/cc | |
| | 2) Gaseous | | | $\leq 6.6\times10^{-2}$μCi/cc Xe-133 | $10^{-6}$ to $10^{-1}$μCi/cc | |
| | B. Noble Gas Monitors Control Room Ventilation Intake RU-29 & RU-30 | 1 | ALL MODES | $\leq 2\times10^{-5}$μCi/cc | $10^{-6}$ to $10^{-1}$μCi/cc | 26 |
| 3. | Post Accident Sampling System | 1### | 1,2,3 | N.A. | N.A. | 28 |

*With fuel in the storage pool or building.
**With irradiated fuel in the storage pool.
#When purge is being used.
##Three (3) times background in Rem/hour.
###The Minimum Channels Operable will be defined in the Preplanned Alternate Sampling Program.

## TABLE 3.3-6 (Continued)

## ACTION STATEMENTS

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.5.1.

ACTION 24 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12 or operate the fuel building essential ventilation system while handling irradiated fuel.

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the essential filtration mode of operation.

ACTION 27 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

1. For area monitors RU-139 A and B, RU-140 A and B, RU-148 and RU-149, initiate a preplanned alternate program to monitor the appropriate parameters.

2. For process monitors, place moveable air monitor in-line.

3. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

ACTION 28 - With the number of OPERABLE Channels one less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 7 days, or:

1. Initiate the Preplanned Alternate Sampling Program to monitor the appropriate parameter(s).

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action(s) taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| INSTRUMENT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|
| 1. Area Monitors | | | | |
| A. Fuel Pool Area RU-31 | S | R | M | ** |
| B. New Fuel Area RU-19 | S | R | M | * |
| C. Containment Power Access Purge Exhaust RU-37 & RU-38 | P# | R | P###, W### | ## |
| D. Containment RU-148 & RU-149 | S | R | M | 1,2,3,4 |
| E. Main Steam RU-139 A&B RU-140 A&B | S | R | M | 1,2,3,4 |
| 2. Process Monitors | | | | |
| A. Containment Building Atmosphere RU-1 | | | | |
| 1) Particulate | S | R | M | 1,2,3,4 |
| 2) Gaseous | S | R | M | 1,2,3,4 |
| B. Control Room Ventilation Intake RU-29 & RU-30 | S | R | M | All MODES |
| 3. Post Accident Sampling System | N.A. | R | M*** | 1,2,3 |

*With fuel in the storage pool or building.

**With irradiated fuel in the storage pool.

***The functional test should consist of, but not be limited to, a verification of system sampling capabilities.

#If purge is in service for greater than 12 hours, perform once per 12-hour period.

##When purge system is in operation.

###The functional test should consist of, but not be limited to, a verification of system isolation capability by the insertion of a simulated alarm condition.

INSTRUMENTATION

INCORE DETECTORS

## LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and 75% of all detectors, with at least one detector in each quadrant at each level; and

b. A minimum of six tilt estimates, with at least one at each of three levels.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of three OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

APPLICABILITY: When the incore detection system is used for monitoring:

a. AZIMUTHAL POWER TILT,

b. Radial Peaking Factors,

c. Local Power Density,

d. DNB Margin.

ACTION:

a. With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

## SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use if the system has just been returned to OPERABLE status or if 7 days or more have elasped since last use and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin:

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The fixed incore neutron detectors shall be calibrated prior to installation in the reactor core.

## SEISMIC INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above seismic monitoring instruments actuated during a seismic event (greater than or equal to 0.02g) shall have a CHANNEL CALIBRATION performed within 5 days. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

## TABLE 3.3-7

## SEISMIC MONITORING INSTRUMENTATION

| INSTRUMENTS AND SENSOR LOCATIONS | MINIMUM INSTRUMENT OPERABLE |
|---|---|
| 1. Triaxial Accelerometers | |
| a. Tendon Gallery Floor, 55' level | 1 |
| b. R.C.P., Motor Housing, 129'6" level | 1 |
| c. Steam Generator Base, 101'9" level | 1 |
| d. Control Building Floor, 74' level | 1 |
| e. Auxiliary Building Floor 40' level | 1 |
| f. 25' E. of Turbine Bldg. W. side x 189'9" S. of Turbine Bldg. S. Side on ground (Ref. Plant N.) | 1 |
| 2. Peak Reading Accelerograph | |
| a. Aux. Bldg., Valve Gallery, Class 1 Pipe, 78'7" level | 1 |
| 3. Seismic Triggers | |
| a. Tendon Gallery Floor, 55' level (Setpoint 0.010 g) | 1 |
| b. Containment Operating Floor, 140' level (Setpoint 0.020 g) | 1 |
| 4. Digital Cassette Recorders | |
| a. Control Room Area, 140' level | 1 |
| b. Control Room Area, 140' level | 1 |
| c. Control Room Area, 140' level | 1 |
| d. Control Room Area, 140' level | 1 |
| e. Control Room Area, 140' level | 1 |
| f. Control Room Area, 140' level | 1 |
| 5. Seismic Switches | |
| a. Tendon Gallery Floor, 55' level | 1 |

| | Horizontal | Vertical |
|---|---|---|
| Setpoint OBE | 0.18 g | 0.17 g |
| Setpoint SSE | 0.31 g | 0.34 g |

## TABLE 4.3-4

## SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| INSTRUMENTS AND SENSOR LOCATIONS | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST |
|---|---|---|---|
| 1. Triaxial Accelerometers | | | |
| a. Tendon Gallery Floor, 55' level | N.A. | R | SA |
| b. R.C.P., Motor Housing, 129'6" level | N.A. | R | SA |
| c. Steam Generator Base, 101'9" level | N.A. | R | SA |
| d. Control Building Floor, 74' level | N.A. | R | SA |
| e. Auxiliary Building Floor 40' level | N.A. | R | SA |
| f. 25' E. of Turbine Bldg. W. side x 189'9" S. of Turbine Bldg. S. Side on ground (Ref. Plant N.) | N.A. | R | SA |
| 2. Peak Reading Accelerograph | | | |
| a. Aux. Bldg., Valve Gallery, Class 1 Pipe, 78'7" level | N.A. | R | NA |
| 3. Seismic Triggers | | | |
| a. Tendon Gallery Floor, 55' level | N.A. | R | SA |
| b. Containment Operating Floor, 140' level | N.A. | R | SA |
| 4. Digital Cassette Recorders | | | |
| a. Control Room Area, 140' level | M | R | SA |
| b. Control Room Area, 140' level | M | R | SA |
| c. Control Room Area, 140' level | M | R | SA |
| d. Control Room Area, 140' level | M | R | SA |
| e. Control Room Area, 140' level | M | R | SA |
| f. Control Room Area, 140' level | M | R | SA |
| 5. Seismic Switches | | | |
| a. Tendon Gallery Floor, 55' level | M | R | SA |

INSTRUMENTATION

METEOROLOGICAL INSTRUMENTATION

## LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

## SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-5.

## TABLE 3.3-8

## METEOROLOGICAL MONITORING INSTRUMENTATION

| INSTRUMENT | | LOCATION | MINIMUM OPERABLE |
|---|---|---|---|
| 1. WIND SPEED | | | |
| | a. 0* to 50 mph, | Nominal Elev. 35 feet | 1 |
| | b. 0* to 50 mph, | Nominal Elev. 200 feet | 1 |
| 2. WIND DIRECTION | | | |
| | a. 0°-360°-180°, | Nominal Elev. 35 feet | 1 |
| | b. 0°-360°-180°, | Nominal Elev. 200 feet | 1 |
| 3. AIR TEMPERATURE - DELTA T | | | |
| | a. -6°F to 6°F, | Nominal Elev. 35 feet-200 feet | 1 |

*Wind speeds less than 0.6 MPH will be reported as 0.

## TABLE 4.3-5

## METEOROLOGICAL MONITORING INSTRUMENTATION

## SURVEILLANCE REQUIREMENTS

| INSTRUMENT | CHANNEL CHECK | CHANNEL CALIBRATION |
|---|---|---|
| 1. WIND SPEED | | |
| a. Nominal Elev. 35 feet | D | SA |
| b. Nominal Elev. 200 feet | D | SA |
| 2. WIND DIRECTION | | |
| a. Nominal Elev. 35 feet | D | SA |
| b. Nominal Elev. 200 feet | D | SA |
| 3. AIR TEMPERATURE - DELTA T | | |
| a. Nominal Elev. 35 feet - 200 feet | D | SA |

INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown system disconnect switches, power, controls and monitoring instrumentation channels shown in Tables 3.3-9A-C shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than required by Table 3.3-9A, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

b. With one or more remote shutdown system disconnect switches or power or control circuits inoperable, (listed in Tables 3.3-9B and 3.3-9C) restore the inoperable switch(s)/circuit(s) to OPERABLE status or issue procedure changes per Specification 6.8.3 that identifies alternate disconnect methods or power or control circuits for remote shutdown within 7 days, or be in HOT STANDBY within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5 The Remote Shutdown System shall be demonstrated operable:

a. By performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6 for each remote shutdown monitoring instrumentation channel.

b. By operation of each remote shutdown system disconnect switch and power and control circuit including the actuated components at least once per 18 months.

## TABLE 3.3-9A

## REMOTE SHUTDOWN INSTRUMENTATION

| INSTRUMENTATION | READOUT LOCATION | MINIMUM CHANNELS OPERABLE |
|---|---|---|
| 1. Log Neutron Power Level | Remote Shutdown Panel | 2 |
| 2. Reactor Coolant Hot Leg Temperature | Remote Shutdown Panel | 1/loop |
| 3. Reactor Coolant Cold Leg Temperature | Remote Shutdown Panel | 1/loop |
| 4. Pressurizer Pressure | Remote Shutdown Panel | 1 |
| 5. Pressurizer Level | Remote Shutdown Panel | 2 |
| 6. Steam Generator Pressure | Remote Shutdown Panel | 2/steam generator |
| 7. Steam Generator Level | Remote Shutdown Panel | 2/steam generator |
| 8. Refueling Water Tank Level | Remote Shutdown Panel | 2 |
| 9. Charging Line Pressure | Remote Shutdown Panel | 1 |
| 10. Charging Line Flow | Remote Shutdown Panel | 1 |
| 11. Shutdown Cooling Heat Exchanger Temperatures | Remote Shutdown Panel | 2 |
| 12. Shutdown Cooling Flow | Remote Shutdown Panel | 2 |
| 13. Auxiliary Feedwater Flow Rate | Remote Shutdown Panel | 2/steam generator |

## TABLE 3.3-9B

## REMOTE SHUTDOWN DISCONNECT SWITCHES

| DISCONNECT SWITCHES | SWITCH LOCATION |
|---|---|
| 1. SG 1 line 2 Atmospheric Dump Valve Solenoid Air Isolation Valves SGB-HY-178A and SGB-HY-178R | RSP |
| 2. SG 2 line 1 Atmospheric Dump Valve Solenoid Air Isolation Valves SGB-HY-185A and SGB-HY-185R | RSP |
| 3. Auxiliary Spray Valve CHB-HV-203 | RSP |
| 4. Letdown to Regenerative Heat Exchanger Isolation, CHB-UV-515 | RSP |
| 5. Reactor Coolant Pump Controlled Bleedoff, CHB-UV-505 | RSP |
| 6. Auxiliary Feedwater Pump B to SG 1 Control Valve, AFB-HV-30 | RSP |
| 7. Auxiliary Feedwater Pump B to SG 2 Control Valve, AFB-HV-31 | RSP |
| 8. Auxiliary Feedwater Pump B to SG 1 Block Valve, AFB-UV-34 | RSP |
| 9. Auxiliary Feedwater Pump B to SG 2 Block Valve, AFB-UV-35 | RSP |
| 10. Pressurizer Backup Heaters Banks B10, B18, A05 Control | RSP |
| 11. Safety Injection Tank 2A Vent Control SIB-HV-613 | RSP |
| 12. Safety Injection Tank 2B Vent Control SIB-HV-623 | RSP |
| 13. Safety Injection Tank 1A Vent Control SIB-HV-633 | RSP |
| 14. Safety Injection Tank 1B Vent Control SIB-HV-643 | RSP |
| 15. Safety Injection Tank Vent Valves Power Supply SIB-HS-18A | RSP |
| 16. SG 1 line 2 Atmospheric Dump Valve Solenoid Air Isolation Valves SGD-HY-178B and SGD-HY-178S | RSP |
| 17. SG 2 line 1 Atmospheric Dump Valve Solenoid Air Isolation Valves SGD-HY-185B and SGD-HY-185S | RSP |
| 18. Control BLDG Battery Room D Essential Exhaust Fan 'HJB-J01A' | PHB-M3205 |
| 19. Control BLDG Battery Room B Essential Exhaust Fan 'HJB-J01B' | PHB-M3205 |
| 20. Battery Charger D Control Room Circuits PKD-H14 | PHB-M3209 AND PKD-H14 |
| 21. ESF Switchgear Room Essential AHU HJB-Z03 | PHB-M3205 |
| 22. LPSI Pump SIB-P01 Breaker Control | PBB-S04F |
| 23. Diesel Generator B Breaker Control | PBB-S04B |
| 24. Essential Spray Pond Pump SPB-P01 Breaker Control | PBB-S04C |

TABLE 3.3-9B (Continued)

REMOTE SHUTDOWN DISCONNECT SWITCHES

| DISCONNECT SWITCHES | SWITCH LOCATION |
|---|---|
| 25. Essential Chiller ECB-E01 Breaker Control | PBB-S04G |
| 26. E-PBB-S04J 4.16KV Feeder Breaker to 480V Load Center PGB-L32 | PBB-S04J |
| 27. E-PBB-S04H 4.16KV Feeder Breaker to 480V Load Center PGB-L34 | PBB-S04H |
| 28. E-PBB-S04N 4.16KV Feeder Breaker to 480V Load Center PGB-L36 | PBB-S04N |
| 29. Auxiliary Feedwater Pump AFB-P01 Breaker Control | PBB-S04S |
| 30. Essential Cooling Water Pump EWB-P01 Breaker Control | PBB-S04M |
| 31. E-PGB-L32B2 480V Main Supply Breaker to Load Center PGB-L32 | PGB-L32B2 |
| 32. E-PGB-L34B2 480V Main Supply Breaker to Load Center PGB-L34 | PGB-L34B2 |
| 33. E-PGB-L36B2 480V Main Supply Breaker to Load Center PGB-L36 | PGB-L36B2 |
| 34. Charging Pump No. 2 CHB-P01 Supply Breaker CHB-P01 | PGB-L32C1 |
| 35. Diesel Engine Control Switch HS-2A | DGB-C01 |
| 36. Diesel Engine Control Switch HS-2B | DGB-C01 |
| 37. Diesel Generator Control Switch HS-2 | DGB-C01 |
| 38. Diesel Generator Essential Exhaust Fan HDB-J01 | DGB-C01 |
| 39. Diesel Generator Fuel Oil Transfer Pump DFB-P01 | DGB-C01 |
| 40. Battery Charger BD Control Room Circuits PKB-H16 | PHB-M3425 |
| 41. Battery Charger B Control Room Circuits PKB-H12 | PHB-M3627 |
| 42. 125 VDC Battery B Breaker Control Room Circuits | PKB-M4201 |
| 43. 125 VDC Battery D Breaker Control Room Circuits | PKD-M4401 |
| 44. CS Pump B Discharge to SD HX B SIB-HV-689 | PHB-M3804 |
| 45. Shutdown Cooling LPSI Suction SIB-UV-656 | PHB-M3611 |
| 46. LPSI-CS from SD HX B X-Tie SIB-HV-695 | PHB-M3810 |
| 47. Shutdown Cooling Warmup Bypass SIB-HV-690 | PHB-M3806 |
| 48. LPSI-CS to SD HX B Crosstie SIB-HV-694 | PHB-M3416 |

## TABLE 3.3-9B (Continued)

## REMOTE SHUTDOWN DISCONNECT SWITCHES

| DISCONNECT SWITCHES | SWITCH LOCATION |
|---|---|
| 49. SD HX "B" to RC Loops 2A/2B SIB-HV-696 | PHB-M3416 |
| 50. LPSI-SD HX "B" Bypass SIB-HV-307 | PHB-M3803 |
| 51. LPSI Pump "B" Recirc SIB-UV-668 | PHB-M3611 |
| 52. LPSI Pump "B" Suction from RWT SIB-HV-692 | PHB-M3805 |
| 53. SD Cooling LPSI Pump "B" Suction SIB-UV-652 | PHB-M3611 |
| 54. SD Cooling LPSI Pump "B" Suction SID-UV-654 | PKD-B44 |
| 55. LPSI Header "B" to RC Loop 2A SIB-UV-615 | PHB-M3611 |
| 56. LPSI Header "B" to RC Loop 2B SIB-UV-625 | PHB-M3640 |
| 57. VCT Outlet Isolation CHN-UV-501 | NHN-M7208 |
| 58. RWT Gravity Feed CHE-HV-536 | NHN-M7209 |
| 59. Shutdown Cooling Temperature Control SIB-UV-658 | PHB-M3416 |
| 60. Shutdown Cooling Heat Exchanger Bypass Valve SIB-HV-693 | PHB-M3416 |
| 61. 4.16 KV Bus PBB-S04 Feeder from XFMR NBN-X04 | PBB-S04K |
| 62. 4.16 KV Bus PBB-S04 Feeder from XFMR NBN-X03 | PBB-S04L |
| 63. Electrical Penetration Room B ACU HAB-Z06 | PHB-M3640 |
| 64. Control Room HVAC Isolation Dampers HJB-M01/HJB-M55 | RSP |
| 65. O.S.A. Supply Damper HJB-M02 | RSP |
| 66. O.S.A. Supply Damper HJB-M03 | RSP |
| 67. R.C.S. Sample Isolation Valve SSA-UV-203 | SSA-J04 |
| 68. R.C.S. Sample Isolation Valve SSB-UV-200 | RSP |
| 69. 125 VDC Battery A Breaker Control Room Circuits | PKA-M4101 |

## TABLE 3.3-9C

## REMOTE SHUTDOWN CONTROL CIRCUITS

| CONTROL CIRCUITS | SWITCH LOCATION |
|---|---|
| 1. Auxiliary Feedwater Pump B to S/G 1 Isolation Valve AFB-UV-34 | RSP |
| 2. Auxiliary Feedwater Pump B to S/G 1 Control Valve AFB-HV-30 | RSP |
| 3. Auxiliary Feedwater Pump B to S/G 2 Isolation Valve AFB-UV-35 | RSP |
| 4. Auxiliary Feedwater Pump B to S/G 2 Control Valve AFB-HV-31 | RSP |
| 5. Auxiliary Feedwater Pump AFB-P01 | PBB-S04S |
| 6. Charging Pump No. 2 CHB-P01 | PGB-L32C4 |
| 7. Pressurizer Auxiliary Spray Valve CHB-HV-203 | RSP |
| 8. Pressurizer Backup Heater Bank | RSP |
| 9. Letdown to Regen HX Isolation Valve CHB-UV-515 | RSP |
| 10. RCP Cont Bleedoff Valve CHB-UV-505 | RSP |
| 11. Volume Control Tank Outlet Isolation Valve CHN-UV-501 | NHN-M7208 |
| 12. RWT Gravity Feed Isolation Valve CHE-HV-536 | NHN-M7209 |
| 13. S/G 1 line 2 Atmospheric Dump Valve Controller SGB-HIC-178B | RSP |
| 14. S/G 1 line 2 Atmospheric Dump Valve Solenoid Air Isolation Valves SGB-HY-178A and SGB-HY-178R | RSP |
| 15. S/G 1 line 2 Atmospheric Dump Valve Solenoid Air Isolation Valves SGD-HY-178B and SGD-HY-178S | RSP |
| 16. S/G 2 line 2 Atmospheric Dump Valve Controller SGB-HIC-185B | RSP |
| 17. S/G 2 line 1 Atmospheric Dump Valve Solenoid Air Isolation Valves SGB-HY-185A and SGB-HY-185R | RSP |
| 18. S/G 2 line 1 Atmospheric Dump Valve Solenoid Air Isolation Valves SGD-HY-185B and SGD-HY-185S | RSP |
| 19. Diesel Generator B Output Breaker | PBB-S04B |
| 20. Diesel Generator Building Essential Exhaust Fan HDB-J01 | DGB-B01 |
| 21. Diesel Generator B Fuel Oil Transfer Pump DFB-P01 | DGB-B01 |
| 22. E-PBB-S04H 4.16 KV Feeder Breaker to 480V Load Center PGB-L34 | PBB-S04H |
| 23. E-PBB-S04J 4.16KV Feeder Breaker to 480V Load Center PGB-L32 | PBB-S04J |
| 24. E-PBB-S04N 4.16KV Feeder Breaker to 480V Load Center PGB-L36 | PBB-S04N |

## TABLE 3.3-9C (Continued)

## REMOTE SHUTDOWN CONTROL CIRCUITS

| CONTROL CIRCUITS | SWITCH LOCATION |
|---|---|
| 25. E-PGB L32B2 480V Main Supply Breaker To Load Center PGB-L32 | PGB-L32B1 |
| 26. E-PGB-L34B2 480V Main Supply Breaker To Load Center PGB-L34 | PGB-L34B1 |
| 27. E-PGB-L36 480V Supply Breaker To Load Center PGB-L36 | PGB-L36B1 |
| 28. Battery Charger PKB-H12 Supply Breaker | PHB-M3627 |
| 29. Battery Charger PKD-H14 Supply Breaker | PHB-M3209 |
| 30. Backup Battery Charger PKB-H16 Supply Breaker | PHB-M3425 |
| 31. Essential Spray Pond Pump SPB-P01 | PBB-S04C |
| 32. Essential Cooling Water Pump EWB-P01 | PBB-S04M |
| 33. Essential Chilled Water Chiller ECB-E01 | PBB-S04G |
| 34. Battery Room D Essential Exhaust Fan HJB-J01A | PHB-M3206 |
| 35. Battery Room B Essential Exhaust Fan HJB-J01B | PHB-M3207 |
| 36. ESF Switchgear Room B Essential AHU HJB-Z03 | PHB-M3203 |
| 37. Electrical Penetration Room B ACU Fan HAB-Z06 | PHB-M3631 |
| 38. SIT Vent Valves Power Supply SIB-HS-18A | RSP |
| 39. SIT 2A Vent Valve SIB-HV-613 | RSP |
| 40. SIT 2B Vent Valve SIB-HV-623 | RSP |
| 41. SIT 1A Vent Valve SIB-HV-633 | RSP |
| 42. SIT 1B Vent Valve SIB-HV-643 | RSP |
| 43. LPSI Pump B SIB-P01 | PBB-S04F |
| 44. Containment Spray Pump B Discharger to SD HX "B" Valve SIB-HV-689 | PHB-M3804 |
| 45. LPSI Containment Spray from SD HX "B" X-tie Valve SIB-HV-695 | PHB-M3810 |
| 46. Shutdown Cooling LPSI Suction Valve SIB-UV-656 | PHB-M3605 |
| 47. Shutdown Cooling Warmup Bypass Valve SIB-HV-690 | PHB-M3806 |
| 48. LPSI Containment Spray to SD HX "B" X-tie Valve SIB-HV-694 | PHB-M3414 |

## TABLE 3.3-9C (Continued)

## REMOTE SHUTDOWN CONTROL CIRCUITS

| CONTROL CIRCUITS | SWITCH LOCATION |
|---|---|
| 49. SD HX "B" to RC Loops 2A/2B Valve SIB-HV-696 | PHB-M3415 |
| 50. LPSI SD HX "B" Bypass Valve SIB-HV-307 | PHB-M3803 |
| 51. LPSI Pump B Recirc. Valve SIB-UV-668 | PHB-M3609 |
| 52. LPSI Pump B Suction From RWT SIB-HV-692 | PHB-M3805 |
| 53. RC Loop to Shutdown Cooling Valve SIB-UV-652 | PHB-M3604 |
| 54. RC Loop to Shutdown Cooling Valve SID-UV-654 | PKD-B44 |
| 55. LPSI Header B to RC Loop 2A Valve SIB-UV-615 | PHB-M3606 |
| 56. LPSI Header B to RC Loop 2B Valve SIB-UV-625 | PHB-M3621 |
| 57. SDC "B" Temperature Control Valve SIB-HV-658 | PHB-M3412 |
| 58. Control Room Ventilation Isolation Dampers HJB-M01/HJB-M55 | RSP |
| 59. O.S.A. Supply Damper HJB-M02 | RSP |
| 60. O.S.A. Supply Damper HJB-M03 | RSP |
| 61. Diesel Generator "B" Emergency Start | DGB-B01 |
| 62. Normal Offsite Power Supply Breaker | PBB-S04K |
| 63. Alternate Offsite Power Supply Breaker | PBB-S04L |
| 64. Battery "B" Breaker | PKB-M4201 |
| 65. Battery "D" Breaker | PKD-M4401 |
| 66. RCS Sample Isolation Valve SSA-UV-203 | SSA-J04 |
| 67. RCS Sample Isolation Valve SSB-UV-200 | SSB-J04 |
| 68. Train "B" Pumps Combined Recirc to RWT Valve SIB-UV-659 | RSP |
| 69. Shutdown Cooling Heat Exchanger Bypass Valve SIB-UV-693 | PHB-M3413 |
| 70. Battery "A" Breaker | PKA-M4101 |

## TABLE 4.3-6

## REMOTE SHUTDOWN INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| INSTRUMENT | | CHANNEL CHECK | CHANNEL CALIBRATION |
|---|---|---|---|
| 1. | Log Neutron Power Level | M | R |
| 2. | Reactor Coolant Hot Leg Temperature (2) | M | R |
| 3. | Reactor Coolant Cold Leg Temperature (2) | M | R |
| 4. | Pressurizer Pressure | M | R |
| 5. | Pressurizer Level | M | R |
| 6. | Steam Generator Pressure | M | R |
| 7. | Steam Generator Level | M | R |
| 8. | Refueling Water Tank Level | M | R |
| 9. | Charging Line Pressure | M | R |
| 10. | Charging Line Flow | M | R |
| 11. | Shutdown Cooling Heat Exchanger Temperatures | M | R |
| 12. | Shutdown Cooling Flow | M | R |
| 13. | Auxiliary Feedwater Flow Rate | M | R |

INSTRUMENTATION

POST-ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The post-accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more accident monitoring instrumentation channels inoperable, take the action shown in Table 3.3-10.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

## TABLE 3.3-10

## POST-ACCIDENT MONITORING INSTRUMENTATION

| INSTRUMENT | REQUIRED NUMBER OF CHANNELS | MINIMUM CHANNELS OPERABLE | ACTION |
|---|---|---|---|
| 1. Containment Pressure | 2 | 1 | 29,30 |
| 2. Reactor Coolant Outlet Temperature - $T_{hot}$ (Wide Range) | 2 | 1/loop | 29,30 |
| 3. Reactor Coolant Inlet Temperature - $T_{cold}$ (Wide Range) | 2 | 1/loop | 29,30 |
| 4. Pressurizer Pressure - Wide Range | 2 | 1 | 29,30 |
| 5. Pressurizer Water Level | 2 | 1 | 29,30 |
| 6. Steam Generator Pressure | 2/steam generator | 1/steam generator | 29,30 |
| 7. Steam Generator Water Level - Wide Range | 2/steam generator | 1/steam generator | 29,30 |
| 8. Refueling Water Storage Tank Water Level | 2 | 1 | 29,30 |
| 9. Auxiliary Feedwater Flow Rate | 2 | 1 | 29,30 |
| 10. Reactor Cooling System Subcooling Margin Monitor | 2 | 1 | 29,30 |
| 11. Pressurizer Safety Valve Position Indicator | 1/valve | 1/valve | 29,30 |
| 12. Containment Water Level (Narrow Range) | 2 | 1 | 29,30 |
| 13. Containment Water Level (Wide Range) | 2 | 1 | 29,30 |
| 14. Core Exit Thermocouples | 4/core quadrant | 2/core quadrant | 29,30 |
| 15. Reactor Vessel Water Level | 2* | 1* | 31,32 |
| 16. Neutron Flux Monitor (Power Range) | 2 | 1 | 29,30 |

*A channel is eight sensors in a probe. A channel is OPERABLE if four or more sensors, two or more in the upper four and two or more in the lower four, are OPERABLE.

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 29 - With the number of OPERABLE Channels one less than the Required Number of Channels in Table 3.3-10, either restore the Inoperable Channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 30 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE in Table 3.3-10, either restore the Inoperable Channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 31 - With the number of OPERABLE Channels one less than the Required Number of Channels either restore the system to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission Pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 32 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE in Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternative method of monitoring the reactor vessel inventory:

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

3. Restore the system to OPERABLE status at the next scheduled refueling.

TABLE 4.3-7

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| INSTRUMENT | CHANNEL CHECK | CHANNEL CALIBRATION |
|---|---|---|
| 1. Containment Pressure | M | R |
| 2. Reactor Coolant Outlet Temperature - $T_{hot}$ (Wide Range) | M | R |
| 3. Reactor Coolant Inlet Temperature -$T_{cold}$ (Wide Range) | M | R |
| 4. Pressurizer Pressure - Wide Range | M | R |
| 5. Pressurizer Water Level | M | R |
| 6. Steam Generator Pressure | M | R |
| 7. Steam Generator Water Level - Wide Range | M | R |
| 8. Refueling Water Storage Tank Water Level | M | R |
| 9. Auxiliary Feedwater Flow Rate | M | R |
| 10. Reactor Coolant System Subcooling Margin Monitor | M | R |
| 11. Pressurizer Safety Valve Position Indicator | M | R |
| 12. Containment Water Level (Narrow Range) | M | R |
| 13. Containment Water Level (Wide Range) | M | R |
| 14. Core Exit Thermocouples | M | R |
| 15. Reactor Vessel Water Level | M | R |
| 16. Neutron Flux Monitor (Power Range) | M | R |

## LOOSE-PART DETECTION INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.3.3.7 The loose-part detection system shall be OPERABLE with all sensors specified in Table 3.3-11.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one or more loose-part detection system channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.3.3.7 Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of:

a. a CHANNEL CHECK at least once per 24 hours,

b. a CHANNEL FUNCTIONAL TEST at least once per 31 days, and

c. a CHANNEL CALIBRATION at least once per 18 months.

## TABLE 3.3-11

## LOOSE PARTS SENSOR LOCATIONS

| INSTRUMENT NO. | LOCATION |
|---|---|
| JSVNYE - 1 | UPPER VESSEL A (STUD BOLTS) |
| JSVNYE - 2 | UPPER VESSEL B (STUD BOLTS) |
| JSVNYE - 3 | LOWER VESSEL A (INCORE NOZZLE) |
| JSVNYE - 4 | LOWER VESSEL B (INCORE NOZZLE) |
| JSVNYE - 5 | SG-1A (HOT LEG) |
| JSVNYE - 6 | SG-1B (COLD LEG 1A) |
| JSVNYE - 7 | SG-2A (HOT LEG) |
| JSVNYE - 8 | SG-2B (COLD LEG 2A) |

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

## LIMITING CONDITION FOR OPERATION

3.3.3.8 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-12.

ACTION:

a. With a low range radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why this inoperability was not corrected within the time specified.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

## SURVEILLANCE REQUIREMENTS

4.3.3.8 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-8.

TABLE 3.3-12

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

| INSTRUMENT | MINIMUM CHANNELS OPERABLE | APPLICABILITY | ACTION |
|---|---|---|---|
| 1. GASEOUS RADWASTE SYSTEM | | | |
| a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release #RU-12 | 1 | # | 35 |
| b. Flow Rate Monitor | 1 | # | 36 |
| 2. GASEOUS RADWASTE SYSTEM EXPLOSIVE GAS MONITORING SYSTEM | | | |
| a. Hydrogen Monitor | 2 | ** | 39 |
| b. Oxygen Monitor | 2 | ** | 39 |

TABLE 3.3-12 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

| INSTRUMENT | MINIMUM CHANNELS OPERABLE | APPLICABILITY | ACTION |
|---|---|---|---|
| 3. CONDENSER EVACUATION SYSTEM | | | |
| A. Low Range Monitors | | | |
| a. Noble Gas Activity Monitor #RU-141 | 1 | 1, 2, 3***, 4*** | 37 |
| b. Iodine Sampler | 1 | 1, 2, 3***, 4*** | 40 |
| c. Particulate Sampler | 1 | 1, 2, 3***, 4*** | 40 |
| d. Flow Rate Monitor | 1 | 1, 2, 3***, 4*** | 36 |
| e. Sampler Flow Rate Measuring Device | 1 | 1, 2, 3***, 4*** | 36 |
| B. High Range Monitors | | | |
| a. Noble Gas Activity Monitor #RU-142 | 1 | 1, 2, 3***, 4*** | 42 |
| b. Iodine Sampler | 1 | 1, 2, 3***, 4*** | 42 |
| c. Particulate Sampler | 1 | 1, 2, 3***, 4*** | 42 |
| d. Sampler Flow Rate Measuring Device | 1 | 1, 2, 3***, 4*** | 42 |
| 4. PLANT VENT SYSTEM | | | |
| A. Low Range Monitors | | | |
| a. Noble Gas Activity Monitor #RU-143 | 1 | * | 37 |
| b. Iodine Sampler | 1 | * | 40 |
| c. Particulate Sampler | 1 | * | 40 |
| d. Flow Rate Monitor | 1 | * | 36 |
| e. Sampler Flow Rate Measuring Device | 1 | * | 36 |

## TABLE 3.3-12 (Continued)

## RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

| INSTRUMENT | MINIMUM CHANNELS OPERABLE | APPLICABILITY | ACTION |
|---|---|---|---|
| 4. PLANT VENT SYSTEM (Continued) | | | |
| B. High Range Monitors | | | |
| a. Noble Gas Activity Monitor #RU-144 | 1 | * | 42 |
| b. Iodine Sampler | 1 | * | 42 |
| c. Particulate Sampler | 1 | * | 42 |
| d. Sampler Flow Rate Measuring Device | 1 | * | 42 |
| 5. FUEL BUILDING VENTILATION SYSTEM | | | |
| A. Low Range Monitors | | | |
| a. Noble Gas Activity Monitor #RU-145 | 1 | ## | 37,41 |
| b. Iodine Sampler | 1 | ## | 40 |
| c. Particulate Sampler | 1 | ## | 40 |
| d. Flow Rate Monitor | 1 | ## | 36 |
| e. Sampler Flow Rate Measuring Device | 1 | ## | 36 |
| B. High Range Monitors | | | |
| a. Noble Gas Activity Monitor #RU-146 | 1 | ## | 41,42 |
| b. Iodine Sampler | 1 | ## | 42 |
| c. Particulate Sampler | 1 | ## | 42 |
| d. Sampler Flow Rate Measuring Device | 1 | ## | 42 |

## TABLE 3.3-12 (Continued)

## TABLE NOTATION

* At all times.
** During GASEOUS RADWASTE SYSTEM operation.
*** Whenever the condenser air removal system is in operation, or whenever turbine glands are being supplied with steam from sources other than the auxiliary boiler(s).
\# During waste gas release.
\#\# In MODES 1, 2, 3, and 4 or when irradiated fuel is in the fuel storage pool.

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the actions of (a) or (b) or (c) are performed:

a. Initiate the Preplanned Alternate Sampling Program to monitor the appropriate parameter(s).

b. Place moveable air monitors in-line.

c. Take grab samples at least once per 12 hours.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

ACTION 39 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of the GASEOUS RADWASTE SYSTEM may continue provided grab samples are taken and analyzed daily. With both channels inoperable operation may continue provided grab samples are taken and analyzed (1) every 4 hours during degassing operations, and (2) daily during other operations.

## TABLE 3.3-12 (Continued)

## TABLE NOTATION

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the effected pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2 within one hour after the channel has been declared inoperable.

ACTION 41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, comply with the ACTION of Specification 3.9.12 or operate the fuel building essential ventilation system while moving irradiated fuel. b

ACTION 42 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement restore the channel to OPERABLE status within 72 hours or:

a. Initiate the Preplanned Alternate Sampling Program to monitor the appropriate parameter(s) when it is needed.

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action(s) taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

## TABLE 4.3-8

## RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| INSTRUMENT | CHANNEL CHECK | SOURCE CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|
| 1. GASEOUS RADWASTE SYSTEM | | | | | |
| a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release RU-12 | P | P | R(3) | Q(1),(2),P### | # |
| b. Flow Rate Monitor | P | N.A. | R | Q,P### | # |
| 2. GASEOUS RADWASTE SYSTEM EXPLOSIVE GAS MONITORING SYSTEM | | | | | |
| a. Hydrogen Monitor (continuous) | D | N.A. | Q(4) | M | ** |
| b. Hydrogen Monitor (sequential) | D | N.A. | Q(4) | M | ** |
| c. Oxygen Monitor (continuous) | D | N.A. | Q(5) | M | ** |
| d. Oxygen Monitor (sequential) | D | N.A. | Q(5) | M | ** |

TABLE 4.3-8 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| INSTRUMENT | CHANNEL CHECK | SOURCE CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|
| 3. CONDENSER EVACUATION SYSTEM (RU-141 and RU-142) | | | | | |
| a. Noble Gas Activity Monitor | D(6) | M | R(3) | Q(2) | 1, 2, 3***, 4*** |
| b. Iodine Sampler | N.A. | N.A. | N.A. | N.A. | 1, 2, 3***, 4*** |
| c. Particulate Sampler | N.A. | N.A. | N.A. | N.A. | 1, 2, 3***, 4*** |
| d. Flow Rate Monitor | D(7) | N.A. | R | Q | 1, 2, 3***, 4*** |
| e. Sampler Flow Rate Measuring Device | D(7) | N.A. | R | Q | 1, 2, 3***, 4*** |
| 4. PLANT VENT SYSTEM (RU-143 and RU-144) | | | | | |
| a. Noble Gas Activity Monitor | D(6) | M | R(3) | Q(2) | * |
| b. Iodine Sampler | N.A. | N.A. | N.A. | N.A. | * |
| c. Particulate Sampler | N.A. | N.A. | N.A. | N.A. | * |
| d. Flow Rate Monitor | D(7) | N.A. | R | Q | * |
| e. Sampler Flow Rate Measuring Device | D(7) | N.A. | R | Q | * |

## TABLE 4.3-8 (Continued)

## RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| INSTRUMENT | CHANNEL CHECK | SOURCE CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|
| 5. FUEL BUILDING VENTILATION SYSTEM (RU-145 and RU-146) | | | | | |
| a. Noble Gas Actvity Monitor | D(6) | M | R(3) | Q(2) | ## |
| b. Iodine Sampler | N.A. | N.A. | N.A. | N.A. | ## |
| c. Particulate Sampler | N.A. | N.A. | N.A. | N.A. | ## |
| d. Flow Rate Monitor | D(7) | N.A. | R | Q | ## |
| e. Sampler Flow Rate Measuring Device | D(7) | N.A. | R | Q | ## |

## TABLE 4.3-8 (Continued)

## TABLE NOTATIONS

* At all times.

** During GASEOUS RADWASTE SYSTEM operation.

*** Whenever the condenser air removal system is in operation, or whenever turbine glands are being supplied with steam from sources other than the auxiliary boiler(s).

# During waste gas release.

## During MODES 1, 2, 3 or 4 or with irradiated fuel in the fuel storage pool.

### Functional test should consist of, but not be limited to, a verification of system isolation capability by the insertion of a simulated alarm condition.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway occurs if the instrument indicates measured levels above the alarm/trip setpoint.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

(6) The channel check for channels in standby status shall consist of verification that the channel is "on-line and reachable."

(7) Daily channel check not required for flow monitors in standby status.

## 3/4.4 REACTOR COOLANT SYSTEM

### 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

### STARTUP AND POWER OPERATION

#### LIMITING CONDITION FOR OPERATION

3.4.1.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With less than the above required reactor coolant pumps in operation, be in at least HOT STANDBY within 1 hour.

#### SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exception 3.10.3.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 The reactor coolant loops listed below shall be OPERABLE and at least one of these reactor coolant loops shall be in operation*.

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump.

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump.

APPLICABILITY: MODE 3.

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one reactor coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$ 25% indicated wide range level at least once per 12 hours.

*All reactor coolant pumps may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one reactor coolant and/or shutdown cooling loops shall be in operation*.

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump**,

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump**,

c. Shutdown Cooling Train A,

d. Shutdown Cooling Train B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required reactor coolant and/or shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 24 hours.

b. With no reactor coolant or shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

*All reactor coolant pumps and shutdown cooling pumps may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 255°F during cooldown, or 295°F during heatup, unless the secondary water temperature (saturation temperature corresponding to steam generator pressure) of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

## SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$ 25% indicated wide range level at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate greater than or equal to 4000 gpm at least once per 12 hours.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling loop shall be OPERABLE and in operation*, and either:

a. One additional shutdown cooling loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 25% indicated wide range level.

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

a. With less than the above required loops OPERABLE or with less than the required steam generator level, immediately initiate corrective action to return the required loops to OPERABLE status or to restore the required level as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of both steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

*The shutdown cooling pump may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

##A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 255°F during cooldown, or 295°F during heatup, unless the secondary water temperature saturation temperature corresponding to steam generator pressure) of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

## LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two shutdown cooling loops shall be OPERABLE# and at least one shutdown cooling loop shall be in operation*.

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation.

## SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

---

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

*The shutdown cooling pump may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 psia ± 1%*.

APPLICABILITY: MODE 4.

ACTION:

a. With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

b. The provisions of Specification 3.0.4 may be suspended for up to 12 hours for entering into and during operation in MODE 4 for purposes of setting the pressurizer code safety valves under ambient (HOT) conditions provided a preliminary cold setting was made prior to heatup.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

## LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 psia ± 1%*.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours with the shutdown cooling system suction line relief valves aligned to provide overpressure protection for the Reactor Coolant System.

## SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

## 3/4.4.3 PRESSURIZER

### PRESSURIZER

### LIMITING CONDITION FOR OPERATION

3.4.3.1 The pressurizer shall be OPERABLE with a minimum steady-state water level of greater than or equal to 27% indicated level (425 cubic feet) and a maximum steady-state water level of less than or equal to 56% indicated level (948 cubic feet) and at least two groups of pressurizer heaters capable of being powered from Class 1E buses each having a minimum capacity of 125 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of the above required pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, restore the pressurizer to OPERABLE status within 1 hour, or be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

### SURVEILLANCE REQUIREMENTS

4.4.3.1.1 The pressurizer water volume shall be determined to be within its limits at least once per 12 hours.

4.4.3.1.2 The capacity of the above required groups of pressurizer heaters shall be verified to be at least 125 kW at least once per 92 days.

4.4.3.1.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by verifying that on an Engineered Safety Features Actuation test signal concurrent with a loss-of-offsite power:

a. The pressurizer heaters are automatically shed from the emergency power sources, and

b. The pressurizer heaters can be reconnected to their respective buses manually from the control room.

## AUXILIARY SPRAY

### LIMITING CONDITION FOR OPERATION

3.4.3.2 Both auxiliary spray valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With only one of the above required auxiliary spray valves OPERABLE, restore both valves to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With none of the above required auxiliary spray valves OPERABLE, restore at least one valve to OPERABLE status within the next 6 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

### SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The auxiliary spray valves shall be verified to have power available to each valve every 24 hours.

4.4.3.2.2 CH-HV-524 and CH-HV-532 shall be verified locked open at least once per 31 days.

4.4.3.2.3 The auxiliary spray valves shall be cycled at least once per 18 months.

## 3/4.4.4 STEAM GENERATORS

### LIMITING CONDITION FOR OPERATION

3.4.4 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing $T_{cold}$ above 210°F.

### SURVEILLANCE REQUIREMENTS

4.4.4.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

4.4.4.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.4.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.4.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.4.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

SURVEILLANCE REQUIREMENTS (Continued)

1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%).

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.4.4a.8.) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

| Category | Inspection Results |
|---|---|
| C-1 | Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective. |
| C-2 | One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes. |
| C-3 | More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective. |

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.4.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calender months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.4.3a.; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.5.2.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

   4. A main steam line or feedwater line break.

4.4.4.4 Acceptance Criteria

a. As used in this Specification

1. Imperfection means an exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.4.3c., above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline

condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.4.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This Special Report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of an imperfection.

   3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

## TABLE 4.4-1

## MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

| Preservice Inspection | No | Yes |
|---|---|---|
| No. of Steam Generators per Unit | Two | Two |
| First Inservice Inspection | All | One |
| Second & Subsequent Inservice Inspection | One* | One* |

### TABLE NOTATION

*The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

## TABLE 4.4-2

## STEAM GENERATOR TUBE INSPECTION

<table>
<tr><th colspan="3">1ST SAMPLE INSPECTION</th><th colspan="2">2ND SAMPLE INSPECTION</th><th colspan="2">3RD SAMPLE INSPECTION</th></tr>
<tr><th>Sample Size</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th></tr>
<tr><td rowspan="9">A minimum of S Tubes per S. G.</td><td>C–1</td><td>None</td><td>N. A.</td><td>N. A.</td><td>N. A.</td><td>N. A.</td></tr>
<tr><td rowspan="5">C–2</td><td rowspan="5">Plug defective tubes and inspect additional 2S tubes in this S. G.</td><td>C–1</td><td>None</td><td>N. A.</td><td>N. A.</td></tr>
<tr><td rowspan="3">C–2</td><td rowspan="3">Plug defective tubes and inspect additional 4S tubes in this S. G.</td><td>C–1</td><td>None</td></tr>
<tr><td>C–2</td><td>Plug defective tubes</td></tr>
<tr><td>C–3</td><td>Perform action for C–3 result of first sample</td></tr>
<tr><td>C–3</td><td>Perform action for C–3 result of first sample</td><td>N. A.</td><td>N. A.</td></tr>
<tr><td rowspan="3">C–3</td><td rowspan="3">Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G.<br><br>Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50</td><td>All other S. G.s are C–1</td><td>None</td><td>N. A.</td><td>N. A.</td></tr>
<tr><td>Some S. G.s C–2 but no additional S. G. are C–3</td><td>Perform action for C–2 result of second sample</td><td>N. A.</td><td>N. A.</td></tr>
<tr><td>Additional S. G. is C–3</td><td>Inspect all tubes in each S. G. and plug defective tubes. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50</td><td>N. A.</td><td>N. A.</td></tr>
</table>

$S = 3 \frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

## 3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

### LEAKAGE DETECTION SYSTEMS

### LIMITING CONDITION FOR OPERATION

3.4.5.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. A containment atmosphere particulate radioactivity monitoring system,

b. The containment sump level and flow monitoring system, and

c. The containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring system is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.4.5.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere gaseous and particulate monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment sump level and flow monitoring system-performance of CHANNEL CALIBRATION at least once per 18 months.

## OPERATIONAL LEAKAGE

### LIMITING CONDITION FOR OPERATION

3.4.5.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total primary-to-secondary leakage through all steam generators, and 720 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 1 gpm leakage at a Reactor Coolant System pressure of 2250 ± 20 psia from any Reactor Coolant System pressure isolation valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System pressure isolation valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one closed manual or deactivated automatic valve, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With RCS leakage alarmed and confirmed in a flow path with no flow rate indicators, commence an RCS water inventory balance within 1 hour to determine the leak rate.

### SURVEILLANCE REQUIREMENTS

4.4.5.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

SURVEILLANCE REQUIREMENTS (Continued)

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours**.

d. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.5.2.2 Each Reactor Coolant System pressure isolation valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit**:

a. At least once per 18 months,

b.* Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve,

d.* Within 24 hours following valve actuation due to automatic or manual action or flow through the valve,

e.* Within 72 hours following a system response to an Engineered Safety Feature actuation signal.

*The provisions of Specifications 4.4.5.2.2.b, 4.4.5.2.2.d, and 4.4.5.2.2.e are not applicable for valves UV 651, UV 652, UV 653 and UV 654 due to position indication of valves in the control room.

**The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

## TABLE 3.4-1

## REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

| VALVE | DESCRIPTION |
|---|---|
| 1) SIE-V237 | LOOP 1A RC/SI CHECK |
| 2) SIE-V247 | LOOP 1B RC/SI CHECK |
| 3) SIE-V217 | LOOP 2A RC/SI CHECK |
| 4) SIE-V227 | LOOP 2B RC/SI CHECK |
| 5) SIE-V235 | LOOP 1A SIT CHECK |
| 6) SIE-V245 | LOOP 1B SIT CHECK |
| 7) SIE-V215 | LOOP 2A SIT CHECK |
| 8) SIE-V225 | LOOP 2B SIT CHECK |
| 9) SIE-V542 | LOOP 1A SI HEADER CHECK |
| 10) SIE-V543 | LOOP 1B SI HEADER CHECK |
| 11) SIE-V540 | LOOP 2A SI HEADER CHECK |
| 12) SIE-V541 | LOOP 2B SI HEADER CHECK |
| 13) SIA-V522 | LOOP 1 HP LONG TERM RECIRCULATION CHECK |
| 14) SIA-V523 | LOOP 1 HP LONG TERM RECIRCULATION CHECK |
| 15) SIB-V532 | LOOP 2 HP LONG TERM RECIRCULATION CHECK |
| 16) SIB-V533 | LOOP 2 HP LONG TERM RECIRCULATION CHECK |
| 17) SIA-UV651*,# | LOOP 1 SHUTDOWN COOLING ISOLATION |
| 18) SIB-UV652*,# | LOOP 2 SHUTDOWN COOLING ISOLATION |
| 19) SIC-UV653*,# | LOOP 1 SHUTDOWN COOLING ISOLATION |
| 20) SID-UV654*,# | LOOP 2 SHUTDOWN COOLING ISOLATION |

*Testing per Specification 4.4.5.2.2.d is not applicable due to positive indication of valve position in the control room.

#1. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 5.0 gpm are considered unacceptable.

## 3/4.4.6 CHEMISTRY

### LIMITING CONDITION FOR OPERATION

3.4.6 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, restore the parameter to within its Steady State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psia, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psia or prior to proceeding to MODE 4.

### SURVEILLANCE REQUIREMENTS

4.4.6 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

## TABLE 3.4-2

## REACTOR COOLANT SYSTEM CHEMISTRY

| PARAMETER | STEADY STATE LIMIT | TRANSIENT LIMIT |
|---|---|---|
| DISSOLVED OXYGEN* | ≤ 0.10 ppm | ≤ 1.00 ppm |
| CHLORIDE | ≤ 0.15 ppm | ≤ 1.50 ppm |
| FLUORIDE | ≤ 0.10 ppm | ≤ 1.00 ppm |

*Limit not applicable with $T_{cold}$ less than or equal to 250°F.

## TABLE 4.4-3

## REACTOR COOLANT SYSTEM

## CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

| PARAMETER | SAMPLE AND ANALYSIS FREQUENCY |
|---|---|
| DISSOLVED OXYGEN* | At least once per 72 hours |
| CHLORIDE | At least once per 72 hours |
| FLUORIDE | At least once per 72 hours |

*Not required with $T_{cold}$ less than or equal to 250°F

REACTOR COOLANT SYSTEM

## 3/4.4.7 SPECIFIC ACTIVITY

### LIMITING CONDITION FOR OPERATION

3.4.7 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/$\bar{E}$ microcuries/gram.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with $T_{cold}$ less than 500°F within 6 hours.

b. With the specific activity of the primary coolant greater than 100/$\bar{E}$ microcuries/gram, be in at least HOT STANDBY with $T_{cold}$ less than 500°F within 6 hours.

MODES 1, 2, 3, 4 and 5:

With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ microcuries/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

### SURVEILLANCE REQUIREMENTS

4.4.7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

*With $T_{cold}$ greater than or equal to 500°F.

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

| TYPE OF MEASUREMENT AND ANALYSIS | SAMPLE AND ANALYSIS FREQUENCY | MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED |
|---|---|---|
| 1. Gross Activity Determination | At least once per 72 hours | 1, 2, 3, 4 |
| 2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration | 1 per 14 days | 1 |
| 3. Radiochemical for $\bar{E}$ Determination | 1 per 6 months* | 1 |
| 4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135 | (a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram, DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram, and | 1#, 2#, 3#, 4#, 5# |
| | (b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period. One sample is sufficient if plant has gone through a SHUTDOWN or if transient is complete in 6 hours. | 1, 2, 3 |

# Until the specific activity of the primary coolant system is restored within its limits.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

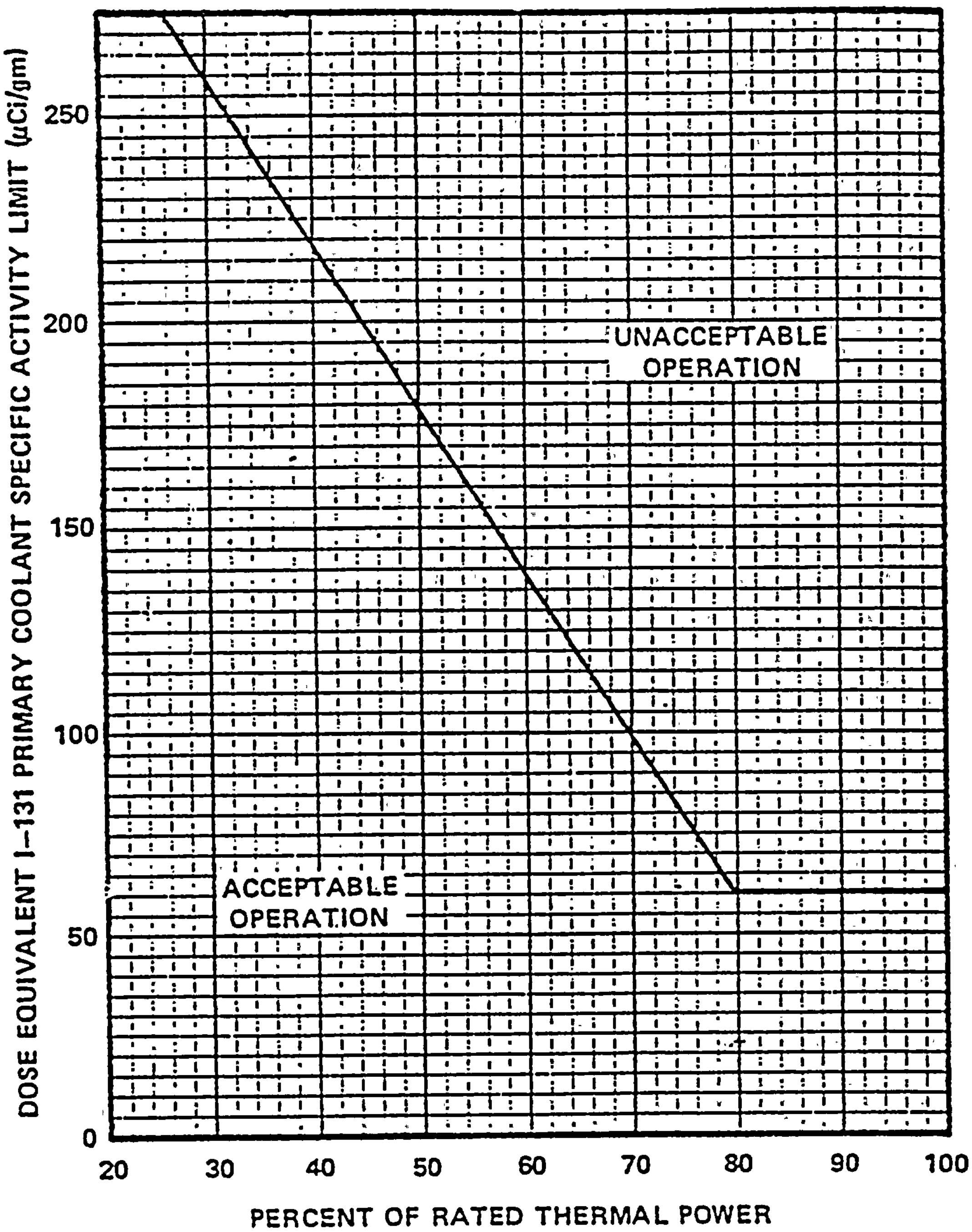



FIGURE 3.4-1

DOSE EQUIVALENT I-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY > 1.0 μCi/GRAM DOSE EQUIVALENT I-131

## 3/4.4.8 PRESSURE/TEMPERATURE LIMITS

### REACTOR COOLANT SYSTEM

### LIMITING CONDITION FOR OPERATION

3.4.8.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate of 20°F per hour with the RCS cold leg temperature less than or equal to 95°F, 40°F per hour with RCS cold leg temperature greater than 95°F but less than or equal to 400°F, and 100°F per hour with RCS cold leg temperature greater than 400°F.

b. A maximum cooldown rate of 10°F per hour with RCS cold leg temperature less than or equal to 100°F, 40°F per hour with RCS cold leg temperature greater than 100°F but less than or equal to 130°F, and 100°F per hour with RCS cold leg temperature greater than 130°F.

c. A maximum temperature change of 10°F in any 1-hour period during inservice hydrostatic and leak testing operations.

APPLICABILITY: At all times*.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS $T_{cold}$ and pressure to less than 210°F and 500 psia, respectively, within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR Part 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figure 3.4-2.

*See Special Test Exception 3.10.5.

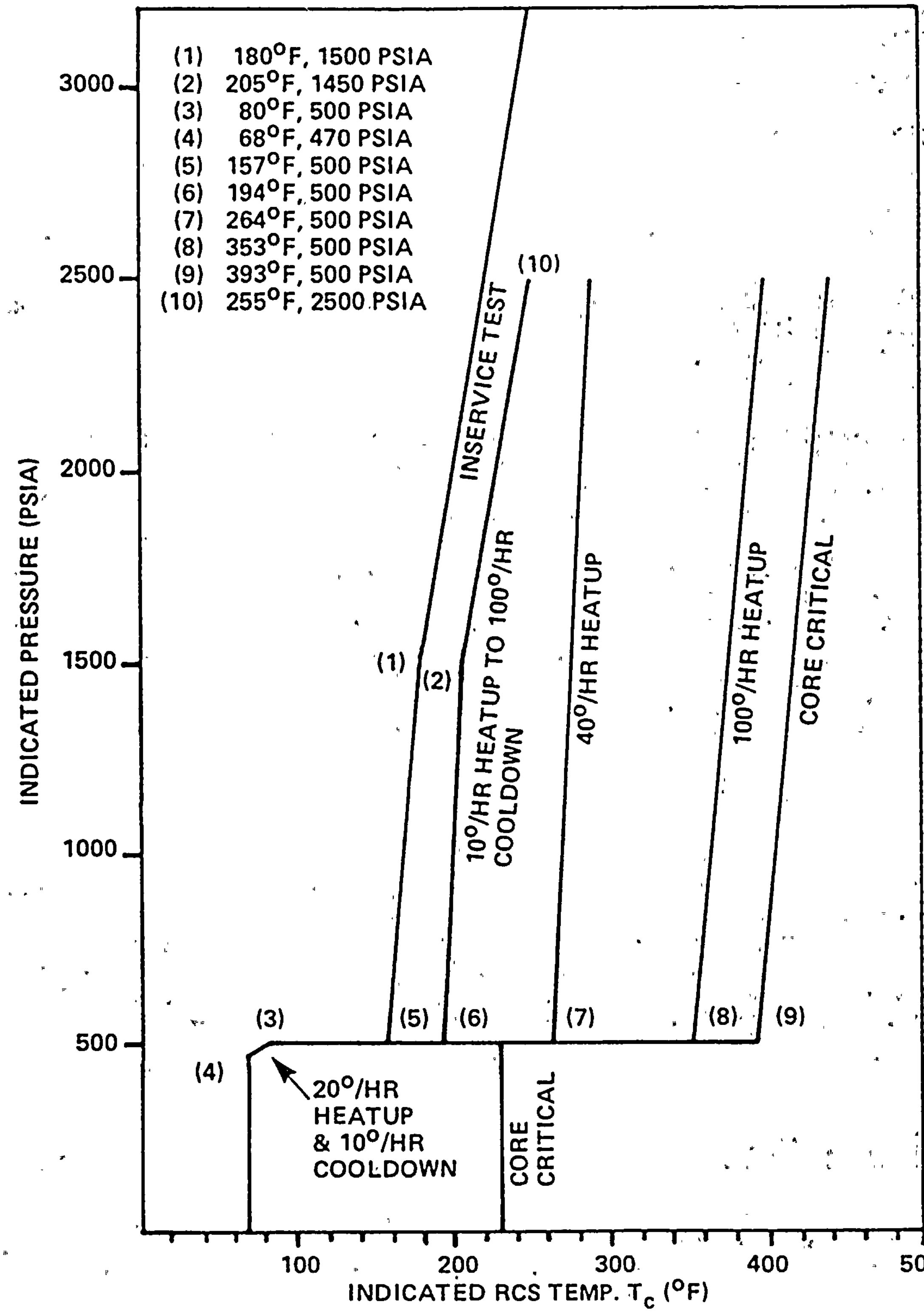



FIGURE 3.4-2

REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITATIONS FOR 0 TO 10 YEARS OF FULL POWER OPERATION

## TABLE 4.4-5

## REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

| CAPSULE NUMBER | VESSEL LOCATION | LEAD FACTOR (LF) | WITHDRAWAL TIME (EFPY) |
|---|---|---|---|
| 1 | 38° | 1.0<LF<1.5 | Standby |
| 2 | 43° | 1.0<LF<1.5 | Standby |
| 3 | 137° | 1.0<LF<1.5 | 4 - 6 |
| 4 | 142° | 1.0<LF<1.5 | Standby |
| 5 | 230° | 1.0<LF<1.5 | 12 - 15 |
| 6 | 310° | 1.0<LF<1.5 | 18 - 24 |

REACTOR COOLANT SYSTEM

PRESSURIZER HEATUP/COOLDOWN LIMITS

LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer temperature shall be limited to:

a. A maximum heatup rate of 200°F per hour, and

b. A maximum cooldown rate of 200°F per hour.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.8.2.2 The spray water temperature differential shall be determined for use in Table 5.7-2 for each cycle of main spray with less than four reactor coolant pumps operating and for each cycle of auxiliary spray operation.

## LIMITING CONDITION FOR OPERATION

3.4.8.3 Both shutdown cooling system (SCS) suction line relief valves with lift settings of less than or equal to 467 psig shall be OPERABLE and aligned to provide overpressure protection for the Reactor Coolant System.

APPLICABILITY: When the reactor vessel head is installed and the temperature of one or more of the RCS cold legs is less than or equal to:

a. 255°F during cooldown

b. 295°F* during heatup

ACTION:

a. With one SCS relief valve inoperable, restore the inoperable valve to OPERABLE status within seven days or reduce $T_{cold}$ to less than 200°F and, depressurize and vent the RCS through a greater than or equal to 16 square inch vent(s) within the next eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

b. With both SCS relief valves inoperable, reduce $T_{cold}$ to less than 200°F and, depressurize and vent the RCS through a greater than or equal to 16 square inch vent(s) within eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

c. In the event either the SCS suction line relief valves or an RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SCS suction line relief valves or RCS vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

*255° during heatup provided the heatup rate is limited to 10°F/hr or less for RCS temperature greater than 255°F and less than or equal to 295°F.

## SURVEILLANCE REQUIREMENTS

4.4.8.3.1 Each SCS suction line relief valve shall be verified to be aligned to provide overpressure protection for the RCS once every 8 hours during

a. Cooldown with the RCS temperature less than or equal to 255°F.

b. Heatup with the RCS temperature less than or equal to 295°F.

4.4.8.3.2 The SCS suction line relief valves shall be verified OPERABLE with the required setpoint at least once per 18 months.

## 3/4.4.9 STRUCTURAL INTEGRITY

### LIMITING CONDITION FOR OPERATION

3.4.9 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.9.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 210°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from service.

d. The provisions of Specification 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.4.9 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 0, October 27, 1971.

## 3/4.4.10 REACTOR COOLANT SYSTEM VENTS

### LIMITING CONDITION FOR OPERATION

3.4.10 Both reactor coolant system vent paths shall be OPERABLE and closed at each of the following locations:

a. Reactor vessel head, and
b. Pressurizer steam space.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With only one of the above required reactor coolant system vent paths OPERABLE, from either location restore both paths at that location to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With none of the above required reactor coolant system vent paths OPERABLE, from either location restore at least one path at that location to OPERABLE status within the next 6 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

### SURVEILLANCE REQUIREMENTS

4.4.10 Each reactor coolant system vent path shall be demonstrated OPERABLE at least once per 18 months, when in MODES 5 or 6, by:

a. Verifying all manual isolation valves in each vent path are locked in the open position.

b. Cycling each vent through at least one complete cycle from the control room.

c. Verifying flow through the reactor coolant system vent paths during venting.

## 3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### 3/4.5.1 SAFETY INJECTION TANKS

### LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System safety injection tank shall be OPERABLE with:

a. The isolation valve key-locked open and power to the valve removed,

b. A contained borated water level of between 1802 cubic feet (28% narrow range indication) and 1914 cubic feet (72 % narrow range indication),

c. A boron concentration between 2300 and 4400 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 625 psig.

e. Nitrogen vent valves closed and power removed**.

f. Nitrogen vent valves capable of being operated upon restoration of power.

APPLICABILITY: MODES 1*, 2*, 3,*†, and 4*†.

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 12 hours.

†With pressurizer pressure greater than or equal to 1837 psia. When pressurizer pressure is less than 1837 psia, at least three safety injection tanks must be OPERABLE, each with a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 1415 cubic feet (60% wide range indication) and 1914 cubic feet (83% wide range indication). With all four safety injection tanks OPERABLE, each tank shall have a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 962 cubic feet (39% wide range indication) and 1914 cubic feet (83% wide range indication). In MODE 4 with pressurizer pressure less than 430 psia, the safety injection tanks may be isolated.

*See Special Test Exceptions 3.10.6 and 3.10.8.

**Nitrogen vent valves may be cycled as necessary to maintain the required nitrogen cover pressure per Specification 3.5.1d.

## SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks is within the above limits, and

2. Verifying that each safety injection tank isolation valve is open and the nitrogen vent valves are closed.

b. At least once per 31 days and whenever the tank is drained to maintain the contained borated water level within the limits of Specification 3.5.1b, by verifying the boron concentration of the safety injection tank solution is between 2300 and 4400 ppm.

c. At least once per 31 days when the pressurizer pressure is above 430 psia, by verifying that power to the isolation valve operator is removed.

d. At least once per 18 months by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

1. When an actual or simulated RCS pressure signal exceeds 515 psia, and

2. Upon receipt of a safety injection actuation (SIAS) test signal.

e. At least once per 18 months by verifying OPERABILITY of RCS-SIT differential pressure alarm by simulating RCS pressure > 715 psia with SIT pressure < 600 psig.

f. At least once per 18 months, when SITs are isolated, by verifying the SIT nitrogen vent valves can be opened.

g. At least once per 31 days, by verifying that power is removed from the nitrogen vent valves.

EMERGENCY CORE COOLING SYSTEMS

## 3/4.5.2 ECCS SUBSYSTEMS - $T_{cold}$ GREATER THAN OR EQUAL TO 350°F

### LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection pump,

b. One OPERABLE low-pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*With pressurizer pressure greater than or equal to 1837 psia.

## SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with the valves key-locked shut:

| Valve Number | Valve Function | Valve Position |
|---|---|---|
| 1. SIA HV-604 | 1. HOT LEG INJECTION | 1. SHUT |
| 2. SIC HV-321 | 2. HOT LEG INJECTION | 2. SHUT |
| 3. SIB HV-609 | 3. HOT LEG INJECTION | 3. SHUT |
| 4. SID HV-331 | 4. HOT LEG INJECTION | 4. SHUT |

b. At least once per 31 days by:

1. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and

2. Verifying that the ECCS piping is full of water by venting the accessible discharge piping high points.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. For all the affected areas within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

SURVEILLANCE REQUIREMENTS (Continued)

1. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

2. Verifying that a minimum total of 464 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

3. Verifying that when a representative sample of 0.055 ± 0.001 lb of TSP from a TSP storage basket is submerged, without agitation, in 1.0 ± 0.05 gallons of 77 ± 9 °F borated water from the RWT, the pH of the mixed solution is raised to greater than or equal to 7 within 4 hours.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on (SIAS and RAS) test signal(s).

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection actuation test signal:

   a. High pressure safety injection pump.

   b. Low pressure safety injection pump.

3. Verifying that on a recirculation actuation test signal, the containment sump isolation valves open, the HPSI, LPSI and CS pump minimum bypass recirculation flow line isolation valves and combined SI mini-flow valve close, and the LPSI pumps stop.

4. Conducting an inspection of all ECCS piping outside of containment, which is in contact with recirculation sump inventory during LOCA conditions, and verifying that the total measured leakage from piping and components is less than 1 gpm when pressurized to at least 40 psig.

f. By verifying that each of the following pumps develops the indicated differential pressure at or greater than their respective minimum allowable recirculation flow when tested pursuant to Specification 4.0.5:

1. High pressure safety injection pump greater than or equal to 1761 psid.

2. Low pressure safety injection pump greater than or equal to 165 psid.

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

   1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

   2. At least once per 18 months.

| LPSI System Valve Number | Hot Leg Injection Valve Number |
|---|---|
| 1. SIB-UV 615, SIA-UV 306 | 1. SIC-HV 321 |
| 2. SIB-UV 625, SIB-UV 307 | 2. SID-HV 331 |
| 3. SIA-UV 635 | |
| 4. SIA-UV 645 | |

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying the following flow rates:

   HPSI System - Single Pump

   The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 816 gpm.

   LPSI System - Single Pump

   1. Injection Loop 1, total flow equal to 4900 ± 100 gpm
   2. Injection Legs 1A and 1B when tested individually, with the other leg isolated, shall be within 100 gpm of each other.
   3. Injection Loop 2, total flow equal to 4900 ± 100 gpm
   4. Injection Legs 2A and 2B when tested individually, with the other leg isolated, shall be within 100 gpm of each other.

   Simultaneous Hot Leg and Cold Leg Injection - Single Pump

   1. Hot Leg, flow equal to 545 ± 20 gpm
   2. Cold Leg, flow equal to 545 ± 20 gpm

## 3/4.5.3 ECCS SUBSYSTEMS - $T_{cold}$ LESS THAN 350°F

### LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. An OPERABLE high pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal.

APPLICABILITY: MODES 3* AND 4.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

### SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystem shall be demonstrated OPERABLE per the applicable surveillance requirements of Specification 4.5.2.

*With pressurizer pressure less than 1837 psia.

## 3/4.5.4 REFUELING WATER TANK

### LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank (RWT) shall be OPERABLE with:

a. A minimum borated water volume as specified in Figure 3.1-2 of Specification 3.1.2.5, and

b. A boron concentration between 4000 and 4400 ppm of boron, and

c. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature when the (outside) air temperature is outside the 60°F to 120°F range.

## 3/4.6 CONTAINMENT SYSTEMS

## 3/4.6.1 PRIMARY CONTAINMENT

### CONTAINMENT INTEGRITY

### LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions except as provided in Table 3.6-1 of Specification 3.6.3.

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

c. After each closing of each penetration subject to Type B testing, except containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at $P_a$ 49.5 psig and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 $L_a$.

*Except valves, blind flanges and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. Less than or equal to $L_a$, 0.10% by weight of the containment air per 24 hours at $P_a$, 49.5 psig, or

2. Less than or equal to $L_t$, 0.05% by weight of the containment air per 24 hours at a reduced pressure of $P_t$, 24.8 psig.

b. A combined leakage rate of less than or equal to 0.60 $L_a$ for all penetrations and valves subject to Type B and C tests, when pressurized to $P_a$.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 $L_a$ or 0.75 $L_t$, as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 $L_a$, restore the overall integrated leakage rate to less than or equal to 0.75 $L_a$, or less than or equal to 0.75 $L_t$, as applicable, and the combined leakage rate for all penetrations and valves subject to Type B and C tests to less than or equal to 0.60 $L_a$ prior to increasing the Reactor Coolant System temperature above 210°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4 - 1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at either $P_a$ 49.5 psig or at $P_t$ 24.8 psig during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet either 0.75 $L_a$ or 0.75 $L_t$, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either 0.75 $L_a$ or 0.75 $L_t$, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either 0.75 $L_a$ or 0.75 $L_t$ at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the Type A test by verifying that the supplemental test result, $L_c$, minus the sum of the Type A test result, $L_{am}$, and the superimposed leak rate, $L_o$, is equal to or less than 0.25 $L_a$.

2. Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 $L_a$ and 1.25 $L_a$.

d. Type B and C tests shall be conducted with gas at $P_a$, 49.5 psig, at intervals no greater than 24 months except for tests involving:

1. Air locks,

2. Purge supply and exhaust isolation valves with resilient material seals.

e. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Specifications 4.6.1.7.2 and 4.6.1.7.3.

f. Air locks shall be tested and demonstrated OPERABLE per Specification 4.6.1.3.

g. The provisions of Specification 4.0.2 are not applicable.

## CONTAINMENT AIR LOCKS

### LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 $L_a$ at $P_a$, 49.5 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed* and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days, or

2. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage to be less than or equal to 0.01 $L_a$ when determined with the volume between the door seals pressurized to greater than or equal to 14.5 ± 0.5 psig, for at least 15 minutes,

*Except during entry to repair an inoperable inner door, for a cumulative time not to exceed 1 hour per year.

SURVEILLANCE REQUIREMENTS (Continued)

b. By conducting overall air lock leakage tests at not less than $P_a$, 49.5 psig, and verifying the overall air lock leakage rate is within its limit:

1. At least once per 6 months#, and

2. Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability*.

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

#The provisions of Specification 4.0.2 are not applicable.

*This constitutes an exemption to Appendix J of 10 CFR Part 50.

## INTERNAL PRESSURE

### LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -0.3 and 2.5 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment average air temperature greater than 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at any five of the following locations and shall be determined at least once per 24 hours:

Location

a. Nominal Elevation 85'0"

b. Nominal Elevation 85'0"

c. Nominal Elevation 126'0"

d. Nominal Elevation 126'0"

e. Nominal Elevation 145'0"

f. Nominal Elevation 188'0"

g. Nominal Elevation 188'0"

## CONTAINMENT VESSEL STRUCTURAL INTEGRITY

### LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the structural integrity at a level below the acceptance criteria of Specification 4.6.1.6 except for Specification 4.6.1.6.2a.4), restore the containment vessel to the required level of integrity within 15 days, perform an engineering evaluation of the containment vessel structural integrity, and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2; or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the structural integrity at a level below the acceptance criteria of Specification 4.6.1.6.2a.4), restore the containment vessel to the required level of integrity within 72 hours, perform an engineering evaluation of the containment vessel structural integrity and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2; or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.1.6.1 The structural integrity of the containment vessel shall be demonstrated at the end of 1, 3, and 5 years following the initial containment vessel structural integrity test and at 5-year intervals thereafter. All of the acceptance testing of tendon and visual examinations of end anchorages, adjacent concrete surfaces, and containment vessel surfaces shall be performed sequentially and within the same time frame.

4.6.1.6.2 The structural integrity of the tendons shall be demonstrated by:

a. Determining from a random but representative sample of at least 10 tendons (6 hoop and 4 inverted U) that each group (hoop and inverted U) has an observed lift-off force within the predicted limits for that group. For each subsequent inspection one tendon from each group shall be kept unchanged to develop a history and to correlate the observed data. The procedure of inspection and the tendon acceptance criteria shall be as follows:

SURVEILLANCE REQUIREMENTS (Continued)

1) If the measured prestressing force of the selected tendon in a group lies above the prescribed lower limit, the lift-off test is considered to be a positive indication of the sample tendon's acceptability;

2) If the measured prestressing force of the selected tendon in a group lies between the prescribed lower limit and 90% of the prescribed lower limit, two tendons, one on each side of this tendon, shall be checked for their prestressing forces. If the prestressing forces of these two tendons are above 95% of the prescribed lower limits for tendons, all three tendons shall be restored to the required level of integrity, and the tendon group shall be considered acceptable. If the measured prestressing force of any two tendons falls below 95% of the prescribed lower limits of the tendons, additional lift-off testing shall be done to detect the cause and extent of such occurrence;

3) If the measured prestressing force of any tendon lies below 90% of the prescribed lower limit, the defective tendon shall be completely detensioned and additional lift-off testing shall be performed to determine the cause and extent of such occurrence;

4) If the average of all measured prestressing forces for each group (corrected for average condition) is found to be less than the minimum required prestress level at anchorage location for that group, the condition shall be considered as below the acceptance criteria for containment vessel structural integrity; and

5) Unless there is degradation of the containment vessel below the acceptance criteria during the first three inspections, the sample population for subsequent inspections shall include at least 6 tendons (3 hoop and 3 inverted U).

b. Performing tendon detensioning, inspections, and material tests on a previously stressed tendon from each group. A randomly selected tendon from each group shall be completely detensioned in order to identify broken or damaged wires. A previously stressed tendon wire or strands from one tendon of each group shall be removed for testing and examination over the entire length to determine (which should include the broken wire if so identified) that:

1) The tendon wires are free of corrosion, cracks, and damage;
2) There are no changes in the presence or physical appearance of the sheathing filler-grease; and

SURVEILLANCE REQUIREMENTS (Continued)

3) A minimum tensile strength of 240,000 psi (guaranteed ultimate strength of the tendon material) exists for at least three wire samples (one from each end and one at mid-length) cut from each removed wire. Failure of any one of the wire samples to meet the minimum tensile strength test is evidence that structural integrity is below the acceptance criteria.

c. Performing tendon retensioning of those tendons detensioned for inspection to at least force level recorded prior to detensioning or the predicted value, whichever is greater, with the tolerance within minus zero to plus 6% except that the final seating force shall be such that the stress in the wire or strand shall not exceed 70% of the guaranteed ultimate tensile strength of the tendons. During retensioning of these tendons, the stress in the tendon shall not exceed 80% of its ultimate strength, and the changes in load and elongation shall be measured simultaneously at a minimum of three approximately equally spaced levels of force between zero and the seating force. If the elongation corresponding to a specific load differs by more than 10% from that recorded during installation, an investigation shall be made to ensure that the difference is not related to wire failures or slips of wires in anchorages; and

d. Verifying the OPERABILITY of the sheathing filler-grease by assuring:

1) No voids in excess of 5% of the net duct volume,

2) Minimum grease coverage exists for the different parts of the anchorage system, and

3) The chemical properties of the filler material are within the tolerance limits specified as follows:

| | |
|---|---|
| Water content | 0 - 5% by wt. |
| Chlorides | 0 - 10 ppm |
| Nitrates | 0 - 10 ppm |
| Sulfides | 0 - 5 ppm |
| Reserved alkalinity | 0 - 50% of the installed value |
| (Base numbers) | (installed value 0-5 for older grease). |

4.6.1.6.3 As an assurance of the structural integrity of the containment vessel, tendon anchorage assembly hardware (such as bearing plates, stressing washers, wedges, and buttonheads) of all tendons selected for inspection shall be visually examined. For those containments in multiple unit plants for which only visual inspection need be performed, tendon anchorages selected for inspection shall be visually examined to the extent practical without dismantling the load-bearing components of the anchorages. The surrounding concrete shall also be checked visually for indication of any abnormal condition.

CONTAINMENT SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.6.4 The exterior surface of the containment vessel shall be visually examined to detect areas of large spall, severe scaling, D-cracking in an area of 25 sq. ft. or more, other surface deterioration or disintegration, or grease leakage, each of which can be considered as evidence that the structural integrity is below the acceptance criteria.

4.6.1.6.5 Reports: Any abnormal degradation of the containment structure detected during the above required tests and inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 30 days. This report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure, the tolerances on cracking, and the corrective actions taken.

## TABLE 4.6-1

## TENDON SURVEILLANCE - FIRST YEAR

| Tendon No. | Visual Inspection | Monitor Forces | Detension Tendon | Remove Wire | Test Wire |
|---|---|---|---|---|---|
| V16* | X | X | No | No | No |
| V28 | X | X | X | X | X |
| V09 | X | X | No | No | No |
| V49 | X | X | No | No | No |
| H13-010 | X | X | No | No | No |
| H13-036* | X | X | No | No | No |
| H13-044 | X | X | No | No | No |
| H21-007 | X | X | X | X | X |
| H32-013 | X | X | No | No | No |
| H32-021 | X | X | No | No | No |

Notes:

1. "X" means the tendon shown shall be inspected for the stated requirements during this surveillance.
2. "No" means that inspection is not required for that tendon.
3. "*" means control tendon.

TABLE 4.6-2

TENDON LIFT-OFF FORCE - FIRST YEAR

U-TENDONS

| TENDON NUMBER | TENDON END | MAXIMUM (kips) | MINIMUM (kips) |
|---|---|---|---|
| V16 | Shop | 1469 | 1349 |
| | Field | 1576 | 1448 |
| V28 | Shop | 1489 | 1367 |
| | Field | 1524 | 1400 |
| V09 | Shop | 1524 | 1399 |
| | Field | 1524 | 1399 |
| V49 | Shop | 1438 | 1318 |
| | Field | 1519 | 1394 |

HOOP TENDONS

| TENDON NUMBER | TENDON END | MAXIMUM (kips) | MINIMUM (kips) |
|---|---|---|---|
| H13-010 | Shop | 1507 | 1369 |
| | Field | 1484 | 1348 |
| H13-036 | Shop | 1433 | 1302 |
| | Field | 1388 | 1259 |
| H13-044 | Shop | 1507 | 1381 |
| | Field | 1495 | 1370 |
| H21-007 | Shop | 1541 | 1405 |
| | Field | 1472 | 1340 |
| H32-013 | Shop | 1488 | 1355 |
| | Field | 1465 | 1334 |
| H32-021 | Shop | 1442 | 1312 |
| | Field | 1512 | 1377 |

## CONTAINMENT VENTILATION SYSTEM

### LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

a. Each 42-inch containment purge supply and exhaust isolation valve shall be sealed closed.

b. The 8-inch containment purge supply and exhaust isolation valves shall be sealed closed to the maximum extent practicable but may be open for purge system operation for pressure control, for ALARA and respirable air quality considerations for personnel entry and for surveillance tests that require the valve to be open.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With a 42-inch containment purge supply and/or exhaust isolation valve(s) open or not sealed closed, close and/or seal closed the open valve(s) or isolate the penetration within 4 hours otherwise be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With an 8-inch containment purge supply and/or exhaust isolation valve(s) open for reasons other than given in 3.6.1.7.b above, close the open 8-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits of Specifications 4.6.1.7.2 and/or 4.6.1.7.3, restore the inoperable valve(s) to OPERABLE status or isolate the penetrations such that the measured leakage rate does not exceed the limits of Specifications 4.6.1.7.2 and/or 4.6.1.7.3 within 24 hours, otherwise be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 42-inch containment purge supply and exhaust isolation valve shall be verified to be sealed closed at least once per 31 days.

4.6.1.7.2 At least once per 6 months on a STAGGERED TEST BASIS each sealed closed 42-inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 $L_a$ when pressurized to $P_a$.

4.6.1.7.3 At least once per 92 days each 8-inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.01 $L_a$ when pressurized to $P_a$.

4.6.1.7.4 Each 8-inch containment purge supply and exhaust isolation valve shall be verified to be sealed closed or open in accordance with Specification 3.6.1.7.b at least once per 31 days.

## 3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

### CONTAINMENT SPRAY SYSTEM

### LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWT on a containment spray actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal. Each spray system flow path from the containment sump shall be via an OPERABLE shutdown cooling heat exchanger.

APPLICABILITY: MODES 1, 2, 3, and 4*.

ACTION:

With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is positioned to take suction from the RWT on a containment spray actuation (CSAS) test signal.

b. By verifying that each pump develops an indicated differential pressure of greater than or equal to 257 psid at greater than or equal the minimum allowable recirculation flowrate when tested pursuant to Specification 4.0.5.

c. At least once per 31 days by verifying that the system piping is full of water to the 60 inch level in the containment spray header ($\geq$115 foot level).

d. At least once per 18 months, during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation (CSAS) and recirculation actuation (RAS) test signal.

*Only when shutdown cooling is not in operation.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that upon a recirculation actuation test signal, the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established.

3. Verifying that each spray pump starts automatically on a safety injection actuation (SIAS) and on a containment spray actuation (CSAS) test signal.

e. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

## IODINE REMOVAL SYSTEM

### LIMITING CONDITION FOR OPERATION

3.6.2.2 The iodine removal system shall be OPERABLE with:

a. A spray chemical addition tank containing a level of between 90% and 100% (816 and 896 gallons) of between 33% and 35% by weight $N_2H_4$ solution, and

b. Two spray chemical addition pumps each capable of adding $N_2H_4$ solution from the spray chemical addition tank to a containment spray system pump flow.

APPLICABILITY: MODES 1, 2, 3, and 4*.

ACTION:

With the iodine removal system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the iodine removal system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.2.2 The iodine removal system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked sealed, or otherwise secured in position, is in its correct position.

b. At least once per 6 months by:

1. Verifying the contained solution volume in the tank, and

2. Verifying the concentration of the $N_2H_4$ solution by chemical analysis.

c. By verifying that on recirculation flow, each spray chemical addition pump develops a discharge pressure of 100 psig when tested pursuant to Specification 4.0.5.

*When the containment spray system is required to be OPERABLE.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months, during shutdown, by

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation (CSAS) test signal, and

   2. Verifying that each spray chemical addition pump starts automatically on a CSAS test signal.

e. At least once per 5 years by verifying each solution flow rate from the following drain connections in the iodine removal system:

   1. SIA-V253 pump discharge line 0.63 ± 0.02 gpm.

   2. SIB-V254 pump discharge line 0.63 ± 0.02 gpm.

## 3/4.6.3 CONTAINMENT ISOLATION VALVES

### LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

1. With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

   a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

   b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position*, or

   c. Isolate the affected penetration within 4 hours by use of at least one closed manual valve or blind flange*; or

   d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit.

4.6.3.2 Each isolation valve specified in Sections A, B, and C of Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

   a. Verifying that on a CIAS, CSAS or SIAS test signal, each isolation valve actuates to its isolation position.

   b. Verifying that on a CPIAS test signal, all containment purge valves actuate to their isolation position.

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*The inoperable isolation valve(s) may be part of a system(s). Isolating the affected penetration(s) may affect the use of the system(s). Consider the technical specification requirements on the affected system(s) and act accordingly.

## SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.3 The isolation time of each power operated or automatic valve of Sections A, B and C of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 The check valves specified in Section D of Table 3.6-1 shall be demonstrated OPERABLE pursuant to 10 CFR 50, Appendix J, with the exception of those check valves footnoted as "Not Type C Tested."

4.6.3.5 The isolation valves specified in Sections E, F, and G of Table 3.6-1 shall be demonstrated OPERABLE as required by Specification 4.0.5 and the Surveillance Requirements associated with those Limiting Conditions for Operation pertaining to each valve or system in which it is installed. Valves secured** in their actuated position are considered operable pursuant to this specification.

4.6.3.6 The manual isolation valves specified in Section H of Table 3.6-1 shall be demonstrated OPERABLE pursuant to Surveillance Requirement 4.6.1.1.a of Specification 3.6.1.1.

**Locked, sealed, or otherwise prevented from unintentional operation.

## TABLE 3.6-1

## CONTAINMENT ISOLATION VALVES

| VALVE NUMBER | PENETRATION NUMBER | FUNCTION | MAXIMUM ACTUATION TIME (SECONDS) |
|---|---|---|---|
| | | A. CONTAINMENT ISOLATION (CIAS) | |
| RDA-UV 023 | 9 | Containment radwaste sump pump to LRS holdup tank | 30 |
| RDB-UV 024 | 9 | Containment radwaste sump pump to LRS holdup tank | 5 |
| RDB-UV 407 | 9 | Containment radwaste sump post-accident sampling system | 5 |
| SGB-HV 200# | 11 | Downcomer feedwater chemical injection | 1 |
| SGB-HV 201# | 12 | Downcomer feedwater chemical injection | 1 |
| SIA-UV 708# | 23 | Containment recirc sump to post-accident sampling system | 5 |
| HCB-UV 044 | 25A | Containment air radioactivity monitor (inlet) | 12 |
| HCA-UV 045 | 25A | Containment air radioactivity monitor (inlet) | 12 |
| HCA-UV 046 | 25B | Containment air radioactivity monitor (outlet) | 12 |
| HCB-UV 047 | 25B | Containment air radioactivity monitor (outlet) | 12 |
| GAA-UV 002 | 29 | $N_2$ to steam generator and reactor drain tank | 10 |
| GAA-UV 001 | 30 | $N_2$ to SI tanks | 10 |

#Not Type C tested.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

| VALVE NUMBER | PENETRATION NUMBER | FUNCTION | MAXIMUM ACTUATION TIME (SECONDS) |
|---|---|---|---|
| | | A. CONTAINMENT ISOLATION (CIAS) (Continued) | |
| HPA-UV 001 | 35 | Containment to hydrogen recombiner | 12 |
| HPA-UV 003 | 35 | Containment to hydrogen recombiner | 12 |
| HPA-UV 024 | 35 | $H_2$ control system | 5 |
| HPB-UV 002 | 36 | Containment to hydrogen recombiner | 12 |
| HPA-UV 005 | 38 | Containment to hydrogen recombiner | 12 |
| HPB-UV 004 | 36 | $H_2$ recombiner return to containment (inlet) | 12 |
| HPA-UV 023 | 38 | $H_2$ control system | 5 |
| HPB-UV 006 | 39 | $H_2$ recombiner return to containment (inlet) | 12 |
| CHA-UV 516 | 40 | Letdown line from RC loop 2B to regenerative heat exchanger and letdown heat exchanger | 5 |
| CHB-UV 523 | 40 | Letdown line from RC loop 2B to regenerative heat exchanger and letdown heat exchanger | 5 |
| CHB-UV 924 | 40 | Letdown line to post-accident sampling system | 5 |
| SSB-UV 201 | 42A | Pressurizer liquid sample line | 5 |
| SSA-UV 204 | 42A | Pressurizer liquid sample line | 5 |
| SSB-UV 202 | 42B | Pressurizer steam space sample line | 5 |
| SSA-UV 205 | 42B | Pressurizer steam space sample line | 5 |
| SSB-UV 200 | 42C | Hot leg sample line | 5 |
| SSA-UV 203 | 42C | Hot leg sample line | 5 |

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

| VALVE NUMBER | PENETRATION NUMBER | FUNCTION | MAXIMUM ACTUATION TIME (SECONDS) |
|---|---|---|---|
| | | A. CONTAINMENT ISOLATION (CIAS) (Continued) | |
| CHA-UV 560 | 44 | Reactor Drain tank to pre-holdup ion exchanger | 5 |
| CHB-UV 561 | 44 | Reactor Drain tank to pre-holdup ion exchanger | 5 |
| CHA-UV 580 | 45 | Makeup to reactor drain tank | 5 |
| CHA-UV 715 | 45 | Makeup to reactor drain tank post-accident sampling system | 5 |
| GRA-UV 001 | 52 | RDT vent to WG surge tank | 12 |
| GRB-UV 002 | 52 | RDT vent to WG surge tank | 10 |
| WCB-UV 63 | 60 | Normal chilled water to containment ACU (inlet) | 10 |
| WCB-UV 61 | 61 | Normal chilled water to containment ACU (outlet) | 10 |
| WCA-UV 62 | 61 | Normal chilled water to containment ACU (outlet) | 10 |

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

| VALVE NUMBER | PENETRATION NUMBER | FUNCTION | MAXIMUM ACTUATION TIME (SECONDS) |
|---|---|---|---|
| | | B. CONTAINMENT PURGE (CPIAS)* | |
| CPA-UV 002A | 56 | Containment purge (inlet) | 12 |
| CPB-UV 003A | 56 | Containment purge (inlet) | 12 |
| CPA-UV 002B | 57 | Containment purge (outlet) | 12 |
| CPB-UV 003B | 57 | Containment purge (outlet) | 12 |
| CPA-UV 004A | 78 | Containment purge (inlet) | 8 |
| CPB-UV 005A | 78 | Containment purge (inlet) | 8 |
| CPA-UV 004B | 79 | Containment purge (outlet) | 8 |
| CPB-UV 005B | 79 | Containment purge (outlet) | 8 |

*Also isolated on CIAS.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

| VALVE NUMBER | PENETRATION NUMBER | FUNCTION | MAXIMUM ACTUATION TIME (SECONDS) |
|---|---|---|---|
| | | C. CONTAINMENT SPRAY (CSAS) | |
| IAA-UV-002 | 31 | Service air to reactor containment inst. air | 10 |
| NCB-UV-401 | 33 | NC water to RCP motor bearing lube oil and air coolers | 10 |
| NCB-UV-403 | 34 | NC water to RCP motor bearing lube oil and air coolers | 10 |
| NCA-UV-402 | 34 | NC water to RCP motor bearing lube oil and air coolers | 10 |
| CHB-UV-505 | 43 | RC pump seal bleedoff | 5 |
| CHA-UV-506 | 43 | RC pump seal bleedoff | 5 |

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

| VALVE NUMBER | PENETRATION NUMBER | FUNCTION | MAXIMUM ACTUATION TIME (SECONDS) |
|---|---|---|---|
| | | D. CHECK VALVES | |
| SGE-V 642# | 11 | Feedwater downcomer | N.A. |
| SGE-V 652# | 11 | Feedwater downcomer | N.A. |
| SGE-V 653# | 12 | Feedwater downcomer | N.A. |
| SGE-V 693# | 12 | Feedwater downcomer | N.A. |
| GAE-V 015 | 29 | $N_2$ to steam generator and reactor drain tank | N.A. |
| GAE-V 011 | 30 | $N_2$ to SI tanks | N.A. |
| IAE-V 021 | 31 | Service air to reactor containment instrument air header | N.A. |
| NCE-V 118 | 33 | NC water to RCP motor bearing lube oil and air coolers | N.A. |
| HPA-V 002 | 38 | $H_2$ recombiner return to containment | N.A. |
| HPB-V 004 | 39 | $H_2$ recombiner return to containment | N.A. |
| CHE-V 494 | 45 | Makeup to reactor drain tank | N.A. |
| WCE-V 039 | 60 | Normal chilled water to containment ACU | N.A. |

#Not Type C tested.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

| VALVE NUMBER | PENETRATION NUMBER | FUNCTION | MAXIMUM ACTUATION TIME (SECONDS) |
|---|---|---|---|
| | | D. CHECK VALVES (Continued) | |
| FPE-V 090 | 7 | Containment fire protection | N.A. |
| SGE-V 003# | 8 | Steam generator feedwater | N.A. |
| SGE-V 007# | 8 | Steam generator feedwater | N.A. |
| SGE-V 005# | 10 | Steam generator feedwater | N.A. |
| SGE-V 006# | 10 | Steam generator feedwater | N.A. |
| SIE-V 113# | 13 | HPSI to RC loop 2A | N.A. |
| SIE-V 123# | 14 | HPSI to RC loop 2B | N.A. |
| SIE-V 133# | 15 | HPSI to RC loop 1A | N.A. |
| SIE-V 143# | 16 | HPSI to RC loop 1B | N.A. |
| SIE-V 114# | 17 | LPSI to RC loop 2A | N.A. |
| SIE-V 124# | 18 | LPSI to RC loop 2B | N.A. |
| SIE-V 134# | 19 | LPSI to RC loop 1A | N.A. |
| SIE-V 144# | 20 | LPSI to RC loop 1B | N.A. |
| SIA-V 164 | 21 | Shutdown cooling heat exchanger 1 to containment spray header 1 | N.A. |
| SIB-V 165 | 22 | Shutdown cooling heat exchanger 2 to containment spray header 2 | N.A. |

#Not Type C tested.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

| VALVE NUMBER | PENETRATION NUMBER | FUNCTION | MAXIMUM ACTUATION TIME (SECONDS) |
|---|---|---|---|
| | | D. CHECK VALVES (Continued) | |
| CHE-V M70 | 41 | Regenerative heat exchanger to RC loop 2A | N.A. |
| IAE-V 073 | 59 | Containment service air utility station | N.A. |
| SIB-V 533 | 67 | Long term recirculation loop 2 | N.A. |
| CHE-V 835 | 72 | RC pump seal injection water to RCP 1A, 1B, 2A, 2B | N.A. |
| AFE-V 079# | 75 | Steam generator 1 auxiliary feedwater | N.A. |
| AFE-V 080# | 76 | Steam generator 2 auxiliary feedwater | N.A. |
| SIA-V 523 | 77 | Long term recirculation loop 1 | N.A. |

#Not Type C tested.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

| VALVE NUMBER | PENETRATION NUMBER | FUNCTION | MAXIMUM ACTUATION TIME (SECONDS) |
|---|---|---|---|
| | | E. SAFETY/RELIEF VALVES | |
| SIA-PSV 151# | 23 | Containment recirculation sump to containment spray, LPSI and HPSI headers 1A & 1B | N.A. |
| SIB-PSV 140# | 24 | Containment recirculation sump to containment spray, LPSI and HPSI headers 2A & 2B | N.A. |
| SIB-PSV 189 | 26 | From shutdown cooling RC Loop 2 | N.A.* |
| SIA-PSV 179 | 27 | From shutdown cooling RC Loop 1 | N.A.* |
| SIE-PSV 474 | 28 | Safety injection drain relief | N.A. |

*Valves also covered by Specification 3/4.4.8.3.

#Not Type C tested.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

| VALVE NUMBER | PENETRATION NUMBER | FUNCTION | MAXIMUM ACTUATION TIME (SECONDS) |
|---|---|---|---|
| | | F. NORMALLY OPEN - ESF ACTUATED CLOSED | |
| SGE-UV 169# | 1 & 2 | Main steam isolation bypass | N.A. |
| SGE-UV 183# | 3 & 4 | Main steam isolation bypass | N.A. |
| SGA-UV 1133# | 1-4 | Steam trap/bypass | N.A. |
| SGA-UV 1134# | 1-4 | Steam trap/bypass | N.A. |
| SGB-UV 1135A# | 1-4 | Steam trap/bypass | N.A. |
| SGB-UV 1135B# | 1-4 | Steam trap/bypass | N.A. |
| SGB-UV 1136A# | 1-4 | Steam trap/bypass | N.A. |
| SGB-UV 1136B# | 1-4 | Steam trap/bypass | N.A. |
| SGA-UV 174# | 8 | Steam generator feedwater | N.A. |
| SGB-UV 132# | 8 | Steam generator feedwater | N.A. |
| SGB-UV 137# | 10 | Steam generator feedwater | N.A. |
| SGA-UV 177# | 10 | Steam generator feedwater | N.A. |
| SGB-UV 130# | 11 | Downcomer FIV | N.A. |
| SGA-UV 172# | 11 | Downcomer FIV | N.A. |
| SGB-UV 135# | 12 | Downcomer FIV | N.A. |

#Not Type C tested.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

| VALVE NUMBER | PENETRATION NUMBER | FUNCTION | MAXIMUM ACTUATION TIME (SECONDS) |
|---|---|---|---|
| | F. NORMALLY OPEN - ESF ACTUATED CLOSED (Continued) | | |
| SGA-UV 175# | 12 | Downcomer FIV | N.A. |
| SIA-UV 682# | 28 | SI drain from drain tank | N.A. |
| SGA-UV 211# | 37A | Steam generator blowdown sample | N.A. |
| SGB-UV 228# | 37A | Steam generator blowdown sample | N.A. |
| SGA-UV 204# | 37B | Steam generator blowdown sample | N.A. |
| SGB-UV 219# | 37B | Steam generator blowdown sample | N.A. |
| SGA-UV 500P# | 46 | Steam generator blowdown to SCCS | N.A. |
| SGB-UV 500Q# | 46 | Steam generator blowdown to SCCS | N.A. |
| SGB-UV 500R# | 47 | Steam generator blowdown to SCCS | N.A. |
| SGA-UV 500S# | 47 | Steam generator blowdown to SCCS | N.A. |
| SGB-UV 226# | 48 | Steam generator blowdown to downcomer blowdown sample | N.A. |
| SGA-UV 227# | 48 | Steam generator blowdown to downcomer blowdown sample | N.A. |
| SGA-UV 220# | 49 | Steam generator blowdown to downcomer blowdown sample | N.A. |
| SGB-UV 221# | 49 | Steam generator blowdown to downcomer blowdown sample | N.A. |
| SGB-UV 224# | 63A | SG2 blowdown sample | N.A. |
| SGA-UV 225# | 63A | SG2 blowdown sample | N.A. |
| SGB-UV 222# | 63B | SG2 blowdown sample | N.A. |
| SGA-UV 223# | 63B | SG2 blowdown sample | N.A. |

#Not Type C tested.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

| VALVE NUMBER | PENETRATION NUMBER | FUNCTION | MAXIMUM ACTUATION TIME (SECONDS) |
|---|---|---|---|
| | | G. REQUIRED OPEN DURING ACCIDENT CONDITIONS | |
| SID-UV 654 | 26 | From shutdown cooling RC loop 2 | N.A. |
| SIB-UV 656 | 26 | From shutdown cooling RC loop 2 | N.A. |
| SIB-HV 690 | 26 | From shutdown cooling RC loop 2 | N.A. |
| SIC-UV 653 | 27 | From shutdown cooling RC loop 1 | N.A. |
| SIA-UV 655 | 27 | From shutdown cooling RC loop 1 | N.A. |
| SIA-HV 691 | 27 | From shutdown cooling RC loop 1 | N.A. |
| HCC-HV 076# | 32A | Containment pressure monitor | N.A. |
| HPA-HV 007A | 35 | Containment to hydrogen monitor | N.A. |
| HPB-HV 008A | 36 | Containment to hydrogen monitor | N.A. |
| HPA-HV 007B | 38 | Hydrogen monitor to containment | N.A. |
| HPB-HV 008B | 39 | Hydrogen monitor to containment | N.A. |
| CHA-HV 524 | 41 | Regenerative heat exchanger to RC loop 2A | N.A. |
| HCA-HV 074# | 54A | Containment pressure monitor | N.A. |
| HCB-HV 075# | 55A | Containment pressure monitor | N.A. |
| HCD-HV 077# | 62A | CB pressure monitor | N.A. |
| SID-HV 331 | 67 | Long-term recirculation loop 2 | N.A. |
| CHB-HV 255 | 72 | RC pump seal injection water to RCP 1A, 1B 2A, 2B | N.A. |
| SIC-HV 321 | 77 | Long-term recirculation loop 1 | N.A. |
| SGA-UV 134# | 2 | Main steam to auxiliary feedwater turbine | N.A. |

#Not Type C tested.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

| VALVE NUMBER | PENETRATION NUMBER | FUNCTION | MAXIMUM ACTUATION TIME (SECONDS) |
|---|---|---|---|
| | G. | REQUIRED OPEN DURING ACCIDENT CONDITIONS (Continued) | |
| SGA-UV 134A# | 2 | Main steam to auxiliary feedwater turbine bypass | N.A. |
| SGA-UV 138# | 3 | Main steam to auxiliary feedwater turbine | N.A. |
| SGA-UV 138A# | 3 | Main steam to auxiliary feedwater turbine bypass | N.A. |
| SIB-UV 616# | 13 | HPSI to RC loop 2A | N.A. |
| SIA-UV 617# | 13 | HPSI to RC loop 2A | N.A. |
| SIB-UV 626# | 14 | HPSI to RC loop 2B | N.A. |
| SIA-UV 627# | 14 | HPSI to RC loop 2B | N.A. |
| SIB-UV 636# | 15 | HPSI to RC loop 1A | N.A. |
| SIA-UV 637# | 15 | HPSI to RC loop 1A | N.A. |
| SIB-UV 646# | 16 | HPSI to RC loop 1B | N.A. |
| SIA-UV 647# | 16 | HPSI to RC loop 1B | N.A. |
| SIB-UV 615# | 17 | LPSI to RC loop 2A | N.A. |
| SIB-UV 625# | 18 | LPSI to RC loop 2B | N.A. |
| SIA-UV 635# | 19 | LPSI to RC loop 1A | N.A. |
| SIA-UV 645# | 20 | LPSI to RC loop 1B | N.A. |
| SIA-UV 672 | 21 | Shutdown cooling heat exchanger 1 to containment spray header 1 | N.A. |
| SIB-UV 671 | 22 | Shutdown cooling heat exchanger 2 to containment spray header 2 | N.A. |

#Not Type C tested.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

| VALVE NUMBER | PENETRATION NUMBER | FUNCTION | MAXIMUM ACTUATION TIME (SECONDS) |
|---|---|---|---|
| | G. REQUIRED OPEN DURING ACCIDENT CONDITIONS (Continued) | | |
| SIA-UV 673# | 23 | Containment recirculation sump to containment spray, LPSI and HPSI headers 1A & 1B | N.A. |
| SIA-UV 674# | 23 | Containment recirculation sump to containment spray, LPSI and HPSI headers 1A & 1B | N.A. |
| SIB-UV 675# | 24 | Containment recirculation sump to containment spray, LPSI and HPSI headers 2A & 2B | N.A. |
| SIB-UV 676# | 24 | Containment recirculation sump to containment spray, LPSI and HPSI headers 2A & 2B | N.A. |
| AFB-UV 034# | 75 | Steam generator 1 auxiliary feedwater | N.A. |
| AFC-UV 036# | 75 | Steam generator 1 auxiliary feedwater | N.A. |
| AFB-UV 035# | 76 | Steam generator 2 auxiliary feedwater | N.A. |
| AFA-UV 037# | 76 | Steam generator 2 auxiliary feedwater | N.A. |

#Not Type C tested.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

| VALVE NUMBER | PENETRATION NUMBER | FUNCTION | MAXIMUM ACTUATION TIME (SECONDS) |
|---|---|---|---|
| | H. NORMALLY CLOSED/POSTACCIDENT CLOSED VALVES | | |
| SGE-V-603# | 1 | $N_2$ blanket supply/$N_2$ vent | N.A. |
| SGE-V-611# | 3 | $N_2$ blanket supply/$N_2$ vent | N.A. |
| DWE-V 061* | 6 | Containment demineralized water stations | N.A. |
| DWE-V 062* | 6 | Containment demineralized water stations | N.A. |
| FPE-V 089 | 7 | Fire protection containment | N.A. |
| SIE-V 463* | 28 | Safety injection tank drain | N.A. |
| CHE-V 854* | 41 | Chemical addition unit to regenerative heat exchanger | N.A. |
| PCE-V 070 | 50 | Fuel pool cooling | N.A. |
| PCE-V 071 | 50 | Fuel pool cooling | N.A. |
| PCE-V 075 | 51 | Refueling pool cleanup | N.A. |
| PCE-V 076 | 51 | Refueling pool cleanup | N.A. |
| IAE-V 072* | 59 | Containment service air utility station | N.A. |

*May be opened on an intermittent basis under administrative control.

#Not Type C tested.

## 3/4.6.4 COMBUSTIBLE GAS CONTROL

### HYDROGEN MONITORS

### LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen monitors shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one hydrogen monitor inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

b. With both hydrogen monitors inoperable, restore at least one monitor to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

### SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing a nominal:

a. One volume percent hydrogen, balance nitrogen.

b. Four volume percent hydrogen, balance nitrogen.

## ELECTRIC HYDROGEN RECOMBINERS

### LIMITING CONDITION FOR OPERATION

3.6.4.2 Two portable independent containment hydrogen recombiner systems shared among the three units shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or meet the requirements of Specification 3.6.4.3, or be in at least HOT STANDBY within the next 6 hours.

### SURVEILLANCE REQUIREMENTS

4.6.4.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by:

1. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure and control console.

2. Operating the recombiner to include the air blast heat exchanger fan motor and enclosed blower motor continuously for at least 30 minutes at a temperature of approximately 800°F reaction chamber temperature.

b. At least once per year by performing a CHANNEL CALIBRATION of recombiner instrumentation to include a functional test of the recombiner at 1200°F (± 50°F) for at least four hours.

CONTAINMENT SYSTEMS

HYDROGEN PURGE CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 A containment hydrogen purge cleanup system, shared among the three units, shall be OPERABLE and capable of being powered from a minimum of one OPERABLE emergency bus.

APPLICABILITY: MODES 1* and 2*.

ACTION:

With the containment hydrogen purge cleanup system inoperable and one hydrogen recombiner OPERABLE as determined by Specification 4.6.4.2, restore the hydrogen purge cleanup system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 The hydrogen purge cleanup system shall be demonstrated OPERABLE.

a. At least once per 31 days by initiating flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 50 scfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978**, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978**.

*With less than two hydrogen recombiners OPERABLE.

**ANSI N509-1980 is applicable for this specification.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying a system flow rate of 50 scfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978*, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978*.

d. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters, pre-filters and charcoal adsorber banks is less than 8.4 inches Water Gauge while operating the system at a flow rate of 50 scfm ± 10%.

   2. Verifying that the heaters dissipate at least 0.5 kW when tested in accordance with ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 50 scfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.0% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 50 scfm ± 10%.

*ANSI N509-1980 is applicable for this specification.

## 3/4.7 PLANT SYSTEMS

### 3/4.7.1 TURBINE CYCLE

### SAFETY VALVES

### LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam safety valves shall be OPERABLE with lift settings as specified in Table 3.7-1.

APPLICABILITY: MODES 1, 2, 3, and 4*.

ACTION:

a. With both reactor coolant loops and associated steam generators in operation and with one or more** main steam safety valves inoperable per steam generator, operation in MODES 1 and 2 may proceed provided that within 4 hours, either all the inoperable valves are restored to OPERABLE status or the Variable Overpower trip setpoint ceiling and the Maximum Allowable Steady State Power Level are reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. Operation in MODES 3 and 4* may proceed with at least one reactor coolant loop and associated steam generator in operation, provided that there are no more than four inoperable main steam safety valves associated with the operating steam generator; otherwise, be in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

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*Until the steam generators are no longer required for heat removal.
**The maximum number of inoperable safety valves on any operating steam generator is four (4).

## TABLE 3.7-1

## STEAM LINE SAFETY VALVES PER LOOPS

| VALVE NUMBER | S/G No. 1 | S/G No. 2 | LIFT SETTING (±1%) * | MINIMUM RATED CAPACITY** |
|---|---|---|---|---|
| a. | SGE PSV 572 | SGE PSV 554 | 1250 psig | 941,543 lb/hr |
| b. | SGE PSV 579 | SGE PSV 561 | 1250 psig | 941,543 lb/hr |
| c. | SGE PSV 573 | SGE PSV 555 | 1290 psig | 971,332 lb/hr |
| d. | SGE PSV 578 | SGE PSV 560 | 1290 psig | 971,332 lb/hr |
| e. | SGE PSV 574 | SGE PSV 556 | 1315 psig | 989,950 lb/hr |
| f. | SGE PSV 575 | SGE PSV 557 | 1315 psig | 989,950 lb/hr |
| g. | SGE PSV 576 | SGE PSV 558 | 1315 psig | 989,950 lb/hr |
| h. | SGE PSV 577 | SGE PSV 559 | 1315 psig | 989,950 lb/hr |
| i. | SGE PSV 691 | SGE PSV 694 | 1315 psig | 989,950 lb/hr |
| j. | SGE PSV 692 | SGE PSV 695 | 1315 psig | 989,950 lb/hr |

*The lift setting pressure shall correspond to ambient conditions at the valve at nominal operating temperature and pressure.

**Capacity is rated at lift setting +3% accumulation.

## TABLE 3.7-2

## MAXIMUM ALLOWABLE STEADY STATE POWER LEVEL AND VARIABLE OVERPOWER TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

| MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR | VARIABLE OVERPOWER TRIP SETPOINT CEILING (% OF RATED THERMAL POWER) | MAXIMUM ALLOWABLE STEADY STATE POWER LEVEL (% OF RATED THERMAL POWER |
|---|---|---|
| 1 | 108.0 | 98.2 |
| 2 | 97.1 | 87.3 |
| 3 | 86.2 | 76.4 |
| 4 | 75.3 | 65.5 |

## AUXILIARY FEEDWATER SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate OPERABLE emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, 3, and 4*.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

### SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Testing the turbine-driven pump and both motor-driven pumps pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for the turbine-driven pump for entry into MODE 3.

2. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

3. Verifying that all manual valves in the suction lines from the primary AFW supply tank (condensate storage tank CTE-T01) to each essential AFW pump, and the manual discharge line valve of each AFW pump are locked, sealed or otherwise secured in the open position.

*Until the steam generators are no longer required for heat removal.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months during shutdown by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an auxiliary feedwater actuation test signal.

   2. Verifying that each pump that starts automatically upon receipt of an auxiliary feedwater actuation test signal will start automatically upon receipt of an auxiliary feedwater actuation test signal.

c. Prior to startup following any refueling shutdown or cold shutdown of 30 days or longer, by verifying on a STAGGERED TEST BASIS (by means of a flow test) that the normal flow path from the condensate storage tank to each of the steam generators through one of the essential auxiliary feedwater pumps delivers at least 750 gpm at 1270 psia or equivalent.

d. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or MODE 4 for the turbine-driven pump.

## CONDENSATE STORAGE TANK

### LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a level of at least 25 feet (300,000 gallons).

APPLICABILITY: MODES 1, 2, 3,# and 4*#.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the reactor makeup water tank as a backup supply to the essential auxiliary feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN with a OPERABLE shutdown cooling loop in operation within the following 6 hours.

### SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the level (contained water volume) is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The reactor makeup water tank shall be demonstrated OPERABLE at least once per 12 hours whenever the reactor makeup water tank is the supply source for the essential auxiliary feedwater pumps by verifying:

a. That the reactor makeup water tank supply line to the auxiliary feedwater system isolation valve is open, and

b. That the reactor makeup water tank contains a water level of at least 26 feet (300,000 gallons).

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*Until the steam generators are no longer required for heat removed.

#Not applicable when cooldown is in progress.

PLANT SYSTEMS

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be less than or equal to 0.10 microcurie/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the specific activity of the secondary coolant system greater than 0.10 microcurie/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

## TABLE 4.7-1

## SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

| TYPE OF MEASUREMENT AND ANALYSIS | SAMPLE AND ANALYSIS FREQUENCY |
|---|---|
| 1. Gross Activity Determination | At least once per 72 hours |
| 2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration | (a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit.<br>(b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. |

## MAIN STEAM LINE ISOLATION VALVES

### LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

MODE 1:

With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, be in at least MODE 2 within the next 6 hours.

MODES 2, 3, and 4:

With one main steam line isolation valve inoperable, subsequent operation in MODE 2, 3, or 4 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.7.1.5.1 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 4.6 seconds when tested pursuant to Specification 4.0.5.

4.7.1.5.2 The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or MODE 4 to perform the surveillance testing of Specification 4.7.1.5.1 provided the testing is performed within 12 hours after achieving normal operating steam pressure and normal operating temperature for the secondary side to perform the test.

## ATMOSPHERIC DUMP VALVES

### LIMITING CONDITION FOR OPERATION

3.7.1.6 The atmospheric dump valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4*.

ACTION:

With less than one atmospheric dump valve per steam generator OPERABLE, restore the required atmospheric dump valve to OPERABLE status within 72 hours; or be in at least HOT STANDBY within the next 6 hours.

### SURVEILLANCE REQUIREMENTS

4.7.1.6 Each atmospheric dump valve shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that the nitrogen accumulator tank is at a pressure $\geq$ 400 psig.

b. Prior to startup following any refueling shutdown or cold shutdown of 30 days or longer, verify that all valves will open and close fully.

*When steam generators are being used for decay heat removal.

## 3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

### LIMITING CONDITION FOR OPERATION

3.7.2 The temperature of the secondary coolant in the steam generators shall be greater than 120°F when the pressure of the secondary coolant in the steam generator is greater than 230 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure to less than or equal to 230 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

### SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in the secondary side of the steam generators shall be determined to be less than 230 psig at least once per 12 hours when the temperature of the secondary coolant is less than 120°F.

## 3/4.7.3 ESSENTIAL COOLING WATER SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent essential cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one essential cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.7.3 At least two essential cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on an SIAS test signal.

c. At least once per 18 months during shutdown, by verifying that the essential cooling water pumps start on an SIAS test signal.

d. At least once per 18 months during shutdown, by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is locked, sealed, or otherwise secured in position, is in its correct position.

## 3/4.7.4 ESSENTIAL SPRAY POND SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent essential spray pond loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one essential spray pond loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.7.4.1 At least two essential spray pond loops shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

4.7.4.2 Once per 18 months during shutdown, verify that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is locked, sealed, or otherwise secured in position, is in its correct position.

## 3/4.7.5 ULTIMATE HEAT SINK

### LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink shall be OPERABLE with two essential spray ponds each with:

a. A minimum usable water depth of 12 feet, and

b. An average water temperature of less than or equal to 89°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.7.5 The ultimate heat sink shall be determined OPERABLE at least once per 24 hours by verifying the average water temperature and water depth to be within their limits for each essential spray pond.

PLANT SYSTEMS

3/4.7.6 ESSENTIAL CHILLED WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 At least two independent essential chilled water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With only one essential chilled water loop OPERABLE, restore at least two loops to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With only one essential chilled water system OPERABLE:

   1. Within 1 hour verify that the normal HVAC system is providing space cooling to the vital power distribution rooms that depend on the inoperable essential chilled water system for space cooling, and

   2. Within 8 hours establish OPERABILITY of the safe shutdown systems which do not depend on the inoperable essential chilled water system (one train each of boration, pressurizer heaters and auxiliary feedwater), and

   3. Within 24 hours establish OPERABILITY of all required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE essential chilled water system for space cooling.

   If these conditions are not satisfied within the specified time, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.6.1 At least two essential chilled water loops shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

4.7.6.2 Once per 18 months during shutdown, verify that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is locked, sealed, or otherwise secured in position, is in its correct position.

## 3/4.7.7 CONTROL ROOM ESSENTIAL FILTRATION SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.7 Two independent control room essential filtration systems shall be OPERABLE.

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3, and 4:

With one control room essential filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one control room essential filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE control room essential filtration system.

b. With both control room essential filtration systems inoperable, or with the OPERABLE control room essential filtration system, required to be OPERABLE by ACTION a., not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

### SURVEILLANCE REQUIREMENTS

4.7.7 Each control room essential filtration system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 28,600 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978*, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978*.

3. Verifying a system flow rate of 28,600 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978*, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978*.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters, pre-filters, and charcoal adsorber banks is less than 8.4 inches Water Gauge while operating the system at a flow rate of 28,600 cfm ± 10%.

2. Verifying that on a Control Room Essential Filtration Actuation Signal and on a SIAS, the system is automatically placed into a filtration mode of operation with flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8-inch Water Gauge relative to adjacent areas during system operation at a makeup flow rate to the control room of less than or equal to 1000 cfm.

4. Verifying that the emergency chilled water system will maintain the control room environment at a temperature less than or equal to 80°F for a period of 30 minutes.

*ANSI N509-1980 is applicable for this specification.

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 28,600 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.0% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 28,600 cfm ± 10%.

## 3/4.7.8 ESF PUMP ROOM AIR EXHAUST CLEANUP SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.8* Two independent ESF pump room air exhaust cleanup systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one ESF pump room air exhaust cleanup system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.7.8 Each ESF pump room air exhaust cleanup system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

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*CAUTION - Reference Specification 3.9.12 page 3/4 9-14

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 6000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

3. Verifying a system flow rate of 6000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters, pre-filters, and charcoal adsorber banks is less than 8.4 inches Water Gauge while operating the system at a flow rate of 6000 cfm ± 10%.

2. Verifying that the system starts on an SIAS test signal.

e. After each complete or partial replacement of an HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.0% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.

*ANSI N509-1980 is applicable for this specification.

## 3/4.7.9 SNUBBERS

### LIMITING CONDITION FOR OPERATION

3.7.9 All hydraulic and mechanical snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

### SURVEILLANCE REQUIREMENTS

4.7.9 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Snubber Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all hydraulic and mechanical snubbers. If all snubbers of each type are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection of that type shall be performed at the first refueling outage. Otherwise, subsequent visual inspections of a given type shall be performed in accordance with the following schedule:

SURVEILLANCE REQUIREMENTS (Continued)

| No. of Inoperable Snubbers of Each Type per Inspection Period | Subsequent Visual Inspection Period*# |
|---|---|
| 0 | 18 months ± 25% |
| 1 | 12 months ± 25% |
| 2 | 6 months ± 25% |
| 3,4 | 124 days ± 25% |
| 5,6,7 | 62 days ± 25% |
| 8 or more | 31 days ± 25% |

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type on that system that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specifications 4.7.9f. When a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and cannot be determined OPERABLE via functional testing unless the test is started with the piston in the as-found setting, extending the piston rod in the tension mode direction. Snubbers which appear inoperable during an area post maintenance inspection, area walkdown, or Transient Event Inspection shall not be considered inoperable for the purpose of establishing the Subsequent Visual Inspection Period provided that the cause of the inoperability is clearly established and remedied for that particular snubber and for the other snubbers, irrespective of type, that may be generally susceptible.

d. Transient Event Inspection

An inspection shall be performed of all hydraulic and mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data. A visual inspection of the systems shall be made within 6 months following such an event. In addition

*The inspection interval for each type of snubber on a given system shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found on that system.

#The provisions of Specification 4.0.2 are not applicable.

to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.9f., an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.9f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Reject" region all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation N = 55(1 + C/2). Each snubber point should be plotted as soon as the snubber

is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

SURVEILLANCE REQUIREMENTS (Continued)

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.9e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the functional test must have been performed within 12 months before being installed in the unit.

i. Snubber Seal Replacement Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

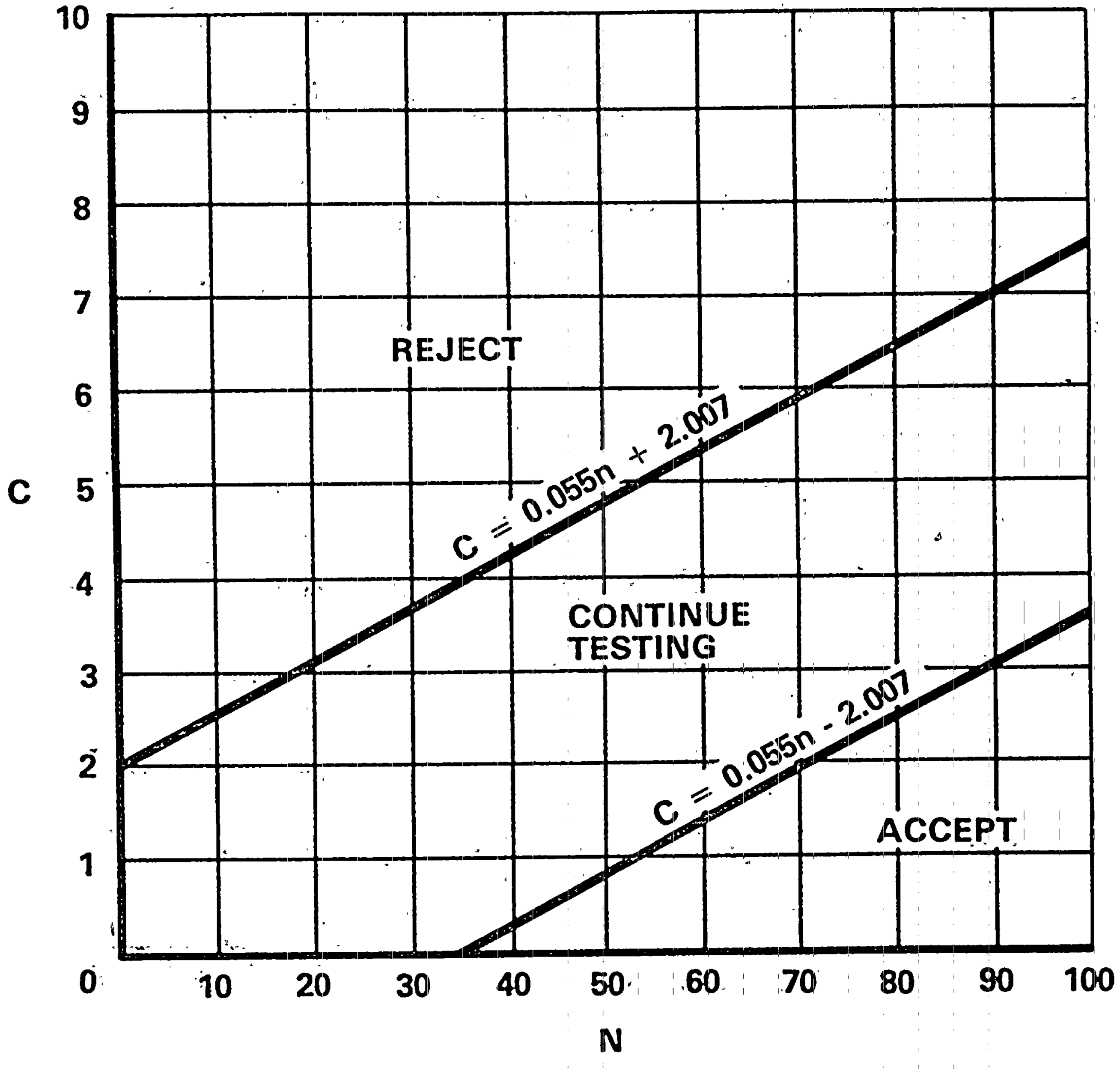


FIGURE 4.7-1

SAMPLING PLAN FOR SNUBBER FUNCTIONAL TEST

## 3/4.7.10 SEALED SOURCE CONTAMINATION

### LIMITING CONDITION FOR OPERATION

3.7.10 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, immediately withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.7.10.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcurie per test sample.

4.7.10.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequencies described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive material:

   1. With a half-life greater than 30 days (excluding Hydrogen 3), and

   2. In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source or detector.

4.7.10.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcurie of removable contamination.

PLANT SYSTEMS

## 3/4.7.11 SHUTDOWN COOLING SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.11 Two independent shutdown cooling subsystems shall be OPERABLE, with each subsystem comprised of:

a. One OPERABLE low pressure safety injection pump, and

b. An independent OPERABLE flow path capable of taking suction from the RCS hot leg and discharging coolant through the shutdown cooling heat exchanger and back to the RCS through the cold leg injection lines.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one shutdown cooling subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within 1 hour, be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the next 30 hours and continue action to restore the required subsystem to OPERABLE status.

b. With both shutdown cooling subsystems inoperable, restore one subsystem to OPERABLE status within 1 hour or be in at least HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 6 hours and continue action to restore the required subsystems to OPERABLE status.

c. With both sutdown cooling subsystems inoperable and both reactor coolant loops inoperable, initiate action to restore the required subsystems to OPERABLE status.

### SURVEILLANCE REQUIREMENTS

4.7.11 Each shutdown cooling subsystem shall be demonstrated OPERABLE:

a. At least once per 18 months, during shutdown, by establishing shutdown cooling flow from the RCS hot legs, through the shutdown cooling heat exchangers, and returning to the RCS cold legs.

b. At least once per 18 months, during shutdown, by testing the automatic and interlock action of the shutdown cooling system connections from the RCS. The shutdown cooling system suction valves shall not open when RCS pressure is greater than 410 psia. The shutdown cooling system suction valves located outside containment shall close automatically when RCS pressure is greater than 500 psia. The shutdown cooling system suction valve located inside containment shall close automatically when RCS pressure is greater than 700 psia.

## 3/4.7.12 CONTROL ROOM AIR TEMPERATURE

### LIMITING CONDITION OF OPERATION

3.7.12 The control room air temperature shall be maintained less than or equal to 80°F.

APPLICABILITY: ALL MODES

ACTION:

With the control room air temperature greater than 80°F, reduce the air temperature to less than or equal to 80°F within 30 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.7.12 At least once per 12 hours, verify that the control room air temperature is less than or equal to 80°F.

## 3/4.8 ELECTRICAL POWER SYSTEMS

### 3/4.8.1 A.C. SOURCES

#### OPERATING

#### LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits from the offsite transmission network to the switchyard and two physically independent circuits from the switchyard to the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

   1. Separate day fuel tank with a minimum level of 2.75 feet (550 gallons of fuel), and

   2. A separate fuel storage system with a minimum level of 80% (71,500 gallons of fuel), and

   3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one offsite circuit of 3.8.1.1.a inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If either EDG has not been successfully tested within the past 24 hours, demonstrate its OPERABILITY by performing Surveillance Requirement 4.8.1.1.2.a.4 separately for each such EDG, unless it is already operating, within 24 hours. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one emergency diesel generator of 3.8.1.1.b inoperable, demonstrate the OPERABILITY of the A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2.a.4 within 24 hours*; restore the diesel generator to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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*This test is required to be completed regardless of when the inoperable EDG is restored to OPERABILITY.

ACTION (Continued)

c. With one offsite circuit and one diesel generator inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2.a.4, unless it is already operating, within 8 hours*; restore one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 Action Statement "a" or "b", as appropriate with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable A.C. power source. A successful test of diesel OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 performed under this Action Statement for an OPERABLE diesel or a restored to OPERABLE diesel satisfies the EDG test requirement of Action Statement "a" or "b".

d. With two of the required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by sequentially performing Surveillance Requirement 4.8.1.1.2.a.4 on both diesels within 8 hours, unless the diesel generators are already operating; restore one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. Following restoration of one offsite source, follow Action Statement "a" with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable offsite A.C. circuit. A successful test(s) of diesel OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 performed under this Action Statement for the OPERABLE diesels satisfies the EDG test requirement of Action Statement "a".

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Following restoration of one diesel generator unit, follow Action Statement "b" with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable diesel generator. A successful test of diesel OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 performed under this Action Statement for a restored to OPERABLE diesel satisfies the EDG test requirement of Action Statement "b".

*This test is required to be completed regardless of when the inoperable EDG is restored to OPERABILITY.

## SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignment indicating power availability

b. Demonstrated OPERABLE at least once per 18 months during shutdown by manually transferring the onsite Class 1E power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

   1. Verifying the fuel level in the day tank.

   2. Verifying the fuel level in the fuel storage tank.

   3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

   4. Verifying the diesel generator can start** and accelerate to generator voltage and frequency at 4160 ± 420 volts and 60 ± 1.2 Hz in less than or equal to 10 seconds. Subsequently, the generator shall be manually synchronized to its appropriate bus and gradually loaded** to an indicated 5200-5400 kW*** and operates for at least 60 minutes. The diesel generator shall be started for this test**** using one of the following signals on a STAGGERED TEST BASIS:

      a) Manual
      b) Simulated loss of offsite power by itself.
      c) Simulated loss of offsite power in conjunction with an ESF actuation test signal.
      d) An ESF actuation test signal by itself.

   5. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

---

**This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

***This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring of the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

****Until the first refueling outage, the diesel generator shall be test started only manually.

4.8.1.1.2 (Continued)

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank obtained in accordance with ASTM-D4176-82, is within the acceptable limits specified in Table 1 of ASTM D975-81 when checked for viscosity, water and sediment.

c. At least once per 184 days the diesel generator shall be started** and accelerated to generator voltage and frequency at 4160 ± 420 volts and 60 ± 1.2 Hz in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The generator shall be manually synchronized to its appropriate emergency bus, loaded to an indicated 5200-5400*** kW in less than or equal to 60 seconds, and operate for at least 60 minutes.

   This test, if it is performed so it coincides with the testing required by Surveillance Requirement 4.8.1.1.2.a.4, may also serve to concurrently meet those requirements as well.

d. At least once per 18 months during shutdown by:

   1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

   2. Verifying the generator capability to reject a single largest load of greater than or equal to 839 kW (Train B AFW pump) for emergency diesel generator B or 696 kW for emergency diesel generator A (Train A HPSI pump) while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 1.2 Hz.

   3. Verifying that the automatic load sequencers are OPERABLE with the interval between each load block within ± 1 second of its design interval.

   4. Simulating a loss of offsite power by itself, and:

      a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

      b) Verifying the diesel starts** on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is

**This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

***This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring of the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

4.8.1.1.2 (Continued)

loaded with the shutdown loads. After energization of these loads, the steady state voltage and frequency shall be maintained at 4160 ± 420 volts and 60 + 1.2/-0.3 Hz.

5. Verifying that on an ESF actuation test signal (without loss of power) the diesel generator starts* on the auto-start signal and operates on standby for greater than or equal to 5 minutes.

6. Simulating a loss-of-offsite power in conjunction with an ESF actuation test signal, and

   a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts* on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer, and operates for greater than or equal to 5 minutes and maintains the steady-state voltage and frequency at 4160 ± 420 volts and 60 + 1.2/-0.3 Hz.

   c) Verifying that all automatic diesel generator trips, except engine overspeed, generator differential, and low lube oil pressure, are automatically bypassed upon loss of voltage on the emergency bus, upon a safety injection actuation signal or upon AFAS.

7. Verifying the diesel generator operates* for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to an indicated 5800-6000 kW** and during the remaining 22 hours of this test, the diesel generator shall be loaded to an indicated 5200-5400 kW**. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2.d.6.b).***

*This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

**This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring of the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

***If Specification 4.8.1.1.2.d.6.b) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated at 5200-5400 kW** for 1 hour or until operating temperature has stabilized.

8. Verifying that the auto-connected loads to each diesel generator do not exceed the continuous rating of 5500 kW.

9. Verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,
   b) Transfer its loads to the offsite power source, and
   c) Proceed through its shutdown sequence.

10. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a) turning gear engaged
    b) emergency stop

e. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting** both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to generator voltage and frequency at 4160 ± 420 volts and 60 ± 1.2 Hz in less than or equal to 10 seconds.

4.8.1.1.3 Reports - All diesel generator failures, valid or nonvalid, shall be reported to the Commission within 30 days in a Special Report pursuant to Specification 6.9.2. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

**This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

## TABLE 4.8-1

## DIESEL GENERATOR TEST SCHEDULE

| Number of Failures In Last 20 Valid Tests* | Number of Failures in Last 100 Valid Tests* | Test Frequency |
|---|---|---|
| $\leq 1$ | $\leq 4$ | Once per 31 days |
| $\geq 2$** | $\geq 5$ | Once per 7 days |

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, but determined on a per diesel generator basis.

For the purposes of determing the required test frequency, the previous test failure count may be reduced to zero if a complete diesel overhaul to like-new conditions is completed, provided that the overhaul including appropriate post-maintenance operation and testing, is specifically approved by the manufacturer <u>and</u> if acceptable reliability has been demonstrated. The reliability criterion shall be the successful completion of 14 consecutive tests in a single series. Ten of these tests shall be in accordance with Surveillance Requirement 4.8.1.1.2.a.4; four tests, in accordance with Surveillance Requirement 4.8.1.1.2.c. If this criterion is not satisfied during the first series of tests, any alternate criterion to be used to transvalue the failure count to zero requires NRC approval.

**The associated test frequency shall be maintained until seven consecutive failure free demands have been performed <u>and</u> the number of failures in the last 20 valid demands has been reduced to one.

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

## LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. Day tank with a minimum level of 2.75 feet (550 gallons of fuel),

2. A fuel storage system with a minimum level of 80% (71,500 gallons of fuel), and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

## SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1, 4.8.1.1.2, and 4.8.1.1.3.

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

CATHODIC PROTECTION

LIMITING CONDITIONS FOR OPERATION

3.8.1.3 The Cathodic Protection System associated with the Diesel Generator Fuel Oil Storage Tanks shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With Cathodic Protection System inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of malfunction and the plans for restoring the system to OPERABLE status.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.3 Verify that the Cathodic Protection System is OPERABLE at the following time intervals:

1. Verify at least once per 61 days that the Cathodic Protection rectifiers are OPERABLE and have been inspected in accordance with Regulatory Guide 1.137.

2. Verify at least once per 12 months that the Cathodic Protection is OPERABLE and providing adequate protection against corrosion in accordance with Regulatory Guide 1.137.

## 3/4.8.2 D.C. SOURCES

### OPERATNG

### LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum the D.C. trains listed in Table 3.8-1 shall be OPERABLE and energized.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required D.C. trains inoperable, restore the inoperable D.C. trains to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required chargers inoperable, either provide charging capability to the affected channel with the associated backup battery charger, or demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1a.1. within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

### SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 129 volts on float charge.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 105 volts, or battery overcharge with battery terminal voltage above 145 volts, by verifying that:

1. The parameters in Table 4.8-2 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than $150 \times 10^{-6}$ ohms, and

3. The average electrolyte temperature of six connected cells is above 60°F.

c. At least once per 18 months by verifying that:

1. The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,

3. The resistance of each cell-to-cell and terminal connection is less than or equal to $150 \times 10^{-6}$ ohms, and

4. The battery charger will supply at least 400 amperes for batteries A and B and 300 amperes for batteries C and D at 125 volts for at least 8 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.1d.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

TABLE 3.8-1

D.C. ELECTRICAL SOURCES

Train A

| CHANNEL A | CHANNEL C |
|---|---|
| 125V bus E-PKA-M41 | 125V D.C. bus E-PKC-M43 |
| 125V D.C. battery bank E-PKA-F11 | 125 V D.C. battery bank E-PKC-F13 |
| Battery charger E-PKA-H11 | Battery charger E-PKC-H13 |
| or | or |
| Backup battery charger E-PKA-H15 (AC) | Backup battery charger E-PKA-H15 (AC) |

Train B

| CHANNEL B | CHANNEL D |
|---|---|
| 125V D.C. bus E-PKB-M42 | 125V D.C. bus E-PKD-M44 |
| 125V D.C. battery bank E-PKB-F12 | 125V D.C. battery bank E-PKD-F14 |
| Battery charger E-PKB-H12 | Battery charger E-PKD-H14 |
| or | or |
| Backup battery charger E-PKB-H16 (BD) | Backup battery charger E-PKB-H16 (BD) |

## TABLE 4.8-2

## BATTERY SURVEILLANCE REQUIREMENTS

| | CATEGORY A(1) | CATEGORY B(2) | |
|---|---|---|---|
| Parameter | Limits for each designated pilot cell | Limits for each connected cell | Allowable(3) value for each connected cell |
| Electrolyte Level | >Minimum level indication mark, and ≤ ¼" above maximum level indication mark | >Minimum level indication mark, and ≤ ¼" above maximum level indication mark | Above top of plates, and not overflowing |
| Float Voltage | ≥ 2.13 volts | ≥ 2.13 volts(a) | > 2.07 volts |
| Specific Gravity(b) | | ≥ 1.195 | Not more than 0.020 below the average of all connected cells |
| | ≥ 1.200(c) | Average of all connected cells ≥ 1.205 | Average of all connected cells ≥ 1.195(c) |

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value, declare the battery inoperable.

(a) Corrected for average electrolyte temperature.

(b) Corrected for electrolyte temperature and level.

(c) Or battery charging current is less than 2 amps when on charge.

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one D.C. train as listed in Table 3.8-1 shall be OPERABLE and energized.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With a required battery bank inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes or movement of irradiated fuel; initiate corrective action to restore the required D.C. train to OPERABLE status as soon as possible.

b. With a required charger inoperable, either provide charging capability to the affected channel with the associated backup battery charger, or demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1a.1. within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt battery banks and chargers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

## 3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

### OPERATING

### LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner with tie breakers open between redundant busses within the unit.

a. Train "A" A.C. emergency busses consisting of:

1. 4160-volt ESF Bus #E-PBA-S03
2. 480-volt ESF Load Center #E-PGA-L31
   a. MCC E-PHA-M31
3. 480-volt ESF Load Center #E-PGA-L33
   a. MCC E-PHA-M33
   b. MCC E-PHA-M37
4. 480-volt ESF Load Center #E-PGA-L35
   a. MCC E-PHA-M35

b. Train "B" A.C. emergency busses consisting of:

1. 4160-volt ESF Bus #E-PBB-S04
2. 480-volt ESF Load Center #E-PGB-L32
   a. MCC E-PHB-M32
   b. MCC E-PHB-M38
3. 480-volt ESF Load Center #E-PGB-L34
   a. MCC E-PHB-M34
4. 480-volt ESF Load Center #E-PGB-L36
   a. MCC E-PHB-M36

c. 120-volt Channel A Vital A.C. Bus #E-PNA-D25 energized from its associated inverter connected to D.C. Channel A*.

d. 120-volt Channel B Vital A.C. Bus #E-PNB-D26 energized from its associated inverter connected to D.C. Channel B*.

e. 120-volt Channel C Vital A.C. Bus #E-PNC-D27 energized from its associated inverter connected to D.C. Channel C*.

f. 120-volt Channel D Vital A.C. Bus #E-PND-D28 energized from its associated inverter connected to D.C. Channel D*.

g. 125-volt D.C. Channel A energized from Battery Bank E-PKA-F11.

h. 125-volt D.C. Channel B energized from Battery Bank E-PKB-F12.

i. 125-volt D.C. Channel C energized from Battery Bank E-PKC-F13.

j. 125-volt D.C. Channel D energized from Battery Bank E-PKD-F14.

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*Two inverters may be disconnected from their D.C. bus for up to 24 hours, as necessary, for the purpose of performing an equalizing charge on their associated battery bank provided (1) their vital busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. bus.

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required divisions of A.C. ESF busses not fully energized, reenergize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. vital bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours and (2) reenergize the A.C. vital bus from its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. bus not energized from its associated battery bank, reenergize the D.C. bus from its associated battery bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

## ONSITE POWER DISTRIBUTION SYSTEMS

### SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One train of A.C. emergency busses consisting of one 4160-volt A.C. ESF bus, and three 480-volt A.C. load centers and their associated four class 1E-MCCs.

b. Two 120-volt A.C. channel vital busses energized from their associated inverters connected to their respective D.C. channels.

c. One 125-volt D.C. train with both required channels energized from their associated battery banks.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

### SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All containment penetration conductor overcurrent protective devices shown in Table 3.8-2 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the above required containment penetration conductor overcurrent protective devices shown in Table 3.8-2 inoperable:

a. Restore the protection device(s) to OPERABLE status or deenergize the circuits(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable device within 72 hours and declare the affected system or component inoperable and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.1 All containment penetration conductor overcurrent protective devices (except fuses) shown in Table 3.8-2 shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. By verifying that the medium voltage (4-15 kV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level, and performing the following:

(a) A CHANNEL CALIBRATION of the associated protection relays, and

(b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed and as specified in Table 3.8-2.

SURVEILLANCE REQUIREMENTS (Continued)

(c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current with a value equal to 300% of the setpoint (pickup) of the long-time delay trip element and 150% of the setpoint (pickup) of the short-time delay trip element, and verifying that the circuit breaker operates within the time delay band width for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current for a frame size of 250 amps or less with tolerances of +40%/-25% and a frame size of 400 amps or greater of ±25% and verifying that the circuit breaker trips instantaneously with no apparent time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

## TABLE 3.8-2

## CONTAINMENT PENETRATION CONDUCTOR

## OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| E-NHN-M1006 | E-NHN-M1002B | SG WET LAYUP RECIRC. PUMP M-SGN-P01B |
| E-NHN-M1017 | E-NHN-M1002B | CTMT/RADWASTE SUMP PUMP M-RDN-P03 |
| E-NHN-M1003 | E-NHN-M1002A | RCP 1B CONTROLLED BLEEDOFF VLV J-RCE-HV-431 |
| E-NHN-M1004 | E-NHN-M1002A | RCP 1B HP COOLER INLET VLV J-RCN-HV-447 |
| E-NHN-M1005 | E-NHN-M1002A | RCP 1B HP COOLER OUTLET VLV J-RCN-HV-451 |
| E-NHN-M1010 | E-NHN-M1002A | REACTOR CAVITY FAN B DISCHARGE DAMPER M-HCN-M02B |
| E-NHN-M1014 | E-NHN-M1002A | REACTOR CAVITY SUMP PUMP M-RDN-P01A |
| E-NHN-M2808 | E-NHN-M2832C | RCP 2B CONTROL BLEEDOFF VLV J-RCE-HV-433 |
| E-NHN-M2813 | E-NHN-M2832C | RCP 2B HI PRESSURE COOLER INLET VLV J-RCN-HV-449 |
| E-NHN-M1009 | E-NHN-M1002A | RCP 2B HI PRESSURE COOLER OUTLET VLV J-RCN-HV-453 |
| E-NHN-M1306 | E-NHN-M1314A | SG 2 HOT LEG BLDWN ISO VLV J-SGE-HV-42 |
| E-NHN-M1307 | E-NHN-M1314A | SG 2 COLD LEG BLDWN ISO VLV J-SGE-HV-44 |
| E-NHN-M1311 | E-NHN-M1314D | WET LAY UP RECIRC PUMP M-SGN-P01A |
| E-NHN-M1316 | E-NHN-M1314C | RCPT (30A) FOR SEAL CRANE ASSY MOTOR E-NHN-122A; E-NHN-122B |
| E-NHN-M1339 | E-NHN-M1314C | MOVABLE INCORE DETECTOR DRIVE MACHINE M-RIN-M03A |

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| E-NHN-M1321 | E-NHN-M1344B | WELDING RCPT'S E-NHN-I07A B, C, D |
| E-NHN-M1331 | E-NHN-M1314B | REACTOR CAVITY SUMP PUMP M-RDN-P01B |
| E-NHN-M1341 | E-NHN-M1314B | REACTOR CAVITY FAN C DISCH DAMPER M-HCN-M02C |
| E-NHN-M1342 | E-NHN-M1314B | CEDM ACU A INTAKE DAMPER M-HCN-M03A |
| E-NHN-M1343 | E-NHN-M1314B | CEDM ACU B INTAKE DAMPER M-HCN-M03B |
| E-NHN-M1323 | E-NHN-M1344A | REACTOR COOLANT OIL LIFT PUMP 2A M-RCN-P02C |
| E-NHN-M1332 | E-NHN-M1344A | CTMT RADWASTE SUMP EAST M-RDN-P02 |
| E-NHN-M1503 | E-NHN-M1502A | RCP 1A CONTROL BLEEDOFF VLV J-RCE-HV-430 |
| E-NHN-M1504 | E-NHN-M1502A | RCP 2A CONTROL BLEEDOFF VLV J-RCE-HV-432 |
| E-NHN-M1505 | E-NHN-M1502A | RCP 1A HI PRESSURE COOLER INLET VLV J-RCN-HV-446 |
| E-NHN-M1506 | E-NHN-M1502A | RCP 2A HI PRESSURE COOLER INLET VLV J-RCN-HV-448 |
| E-NHN-M1507 | E-NHN-M1502A | RCP 1A HI PRESSURE COOLER OUTLET VLV J-RCN-HV-450 |
| E-NHN-M1511 | E-NHN-M1535A | WELDING RCPT'S E-NHN-I12A, B, C |
| E-NHN-M1508 | E-NHN-M1502B | RCP 2A HI PRESSURE COOLER OUTLET VLV J-RCN-HV-452 |
| E-NHN-M1509 | E-NHN-M1502B | REACTOR CAVITY FAN A DISCH DAMPER M-HCN-M02A |

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| E-NHN-M1533 | E-NHN-M1502B | REACTOR CAVITY FAN D DISCH DAMPER M-HCN-M02D |
| E-NHN-M1534 | E-NHN-M1535 | CTMT BLDG MONO HOIST 1 TON M-ZCN-G09 |
| E-NHN-M1517 | E-NHN-M1535 | REACTOR COOLANT OIL LIFT PUMP M-RCN-P02A |
| E-NHN-M1902 | E-NHN-M1917A | REACTOR CAVITY NORM CLG FAN M-HCN-A03A |
| E-NHN-M1904 | E-NHN-M1917B | REACTOR CAVITY NORM CLG FAN M-HCN-A03C |
| E-NHN-M1907 | E-NHN-M1917 | CEDM NORM ACU-A HEXCH OUTLET VLV J-NCN-HV-485 |
| E-NHN-M1911 | E-NHN-M1917 | CTMT NORM ACU-C CHILLED WTR INLET VLV J-WCN-HV-59 |
| E-NHN-M1912 | E-NHN-M1917 | CTMT NORM ACU-A CHILLED WTR INLET VLV J-WCN-HV-57 |
| E-NHN-M2008 | E-NHN-M2010 | CEDM NORM ACU-B HEXCH OUTLET VLV J-NCN-HV-486 |
| E-NHN-M2003 | E-NHN-M2010 | CTMT NORM ACU-B CHILL WATER INLET VLV J-WCN-HV-58 |
| E-NHN-M2004 | E-NHN-M2010 | CTMT NORM ACU-D CHILL WATER INLET VLV J-WCN-HV-60 |
| E-NHN-M2006 | E-NHN-M2010A | REACTOR CAVITY NORM CLG FAN M-HCN-A03B |
| E-NHN-M2007 | E-NHN-M2016 | REACTOR CAVITY NORM CLG FAN M-HCN-A03D |
| E-NHN-M2803 | E-NHN-M2827A | CEDM ACU C INTAKE DAMPER M-HCN-M03C |
| E-NHN-M2804 | E-NHN-M2827A | CEDM ACU D INTAKE DAMPER M-HCN-M03D |

## TABLE 3.8-2 (Continued)

## CONTAINMENT PENETRATION CONDUCTOR

## OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| E-NHN-M2805 | E-NHN-M2827A | SG1 COLD LEG BLOWDOWN ISO VLV J-SGE-HV-41 |
| E-NHN-M2806 | E-NHN-M2827B | SG HOT LEG BLOWDOWN ISOLATION VALVE J-SGE-HV-43 |
| E-NHN-M2827 | E-NHN-M2827A | REACTOR COOLANT PUMP OIL LIFT PUMP 1B M-RCN-P02BP |
| E-NHN-M2828 | E-NHN-M2827A | REACTOR COOLANT PUMP OIL LIFT PUMP 2B M-RCN-P02DP |
| E-NHN-M2809 | E-NHN-M2827C | CONTAINMENT EQUIP HATCH J-ZCN-E02 |
| E-NHN-M2811 | E-NHN-M2832A | 30A RECEPTACLES FOR CTMT BLDG. JIB CRANE M-ZCN-G04A, B |
| E-NHN-M2818 | E-NHN-M2832A | 30A RECEPTACLES FOR SEAL CRANE ASSY MOT |
| E-NHN-M2817 | E-NHN-M2832B | CTMT BLDG MONORAIL HOIST 1 TON M-ZCN-G03 |
| E-NHN-M2819 | E-NHN-M2832B | 30A RECEPTACLES FOR CTMT BLDG JIB CRANE M-ZCN-G04 A, B |
| E-NHN-M2820 | E-NHN-M2832D | CTMT BLDG ELEV #2 CONTROLLER J-ZCN-E01 |
| E-NHN-M2821 | E-NHN-M2828C | MULTIPLE STUD TENSIONER M-ZCN-M15 |
| E-NHN-M2822 | E-NHN-M2828B | WELDING RECPTS E-NHN-109 B, C, D |
| E-NHN-M2801A | E-NHN-M2827B | FUEL TRANSFER SYS CONTROL CONSOLE E-PCN-D02 |
| E-NHN-M2833 | E-NHN-M2827B | REFUELING MACHINE E-PCE-J02 |
| E-NHN-M2833A | E-NHN-M2827B | CEA CHANGE PLATFORM E-PCE-J01 |

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| E-NHN-M7102 | E-NHN-M7104 | CONTAINMENT NORMAL ACU A DISCHARGE DAMPER M-HCN-M01A |
| E-NHN-M7103 | E-NHN-M7104 | CONTAINMENT NORMAL ACU C DISCHARGE DAMPER M-HCN-M01C |
| E-NHN-M7114 | E-NHN-7113 | PZR NORMAL COOLING FAN M-HCN-A06A |
| E-NHN-M2816 | E-NHN-M2832C | CTMT BLDG MONORAIL HOIST-2 TON M-ZCN-G08 |
| E-NHN-M2834A | E-NHN-M2832C | MOVABLE INCORE DETECTOR DRIVE MACH #2 M-RIN-M03B |
| E-NHN-M7202 | E-NHN-M7204 | CTM NORM ACU B DISCH DAMPER M-HCN-M01B |
| E-NHN-M7203 | E-NHN-M7204 | CTM NORM ACU D DISCH DAMPER M-HCN-M01D |
| E-NHN-M7214 | E-NHN-M7213 | PZR NORMAL COOLING FAN M-HCN-A06B |
| E-PGA-L31E2 | E-NGN-B31E2 (FUSE) | CONTAINMENT NORMAL ACU FAN M-HCN-A01A |
| E-PGA-L31E3 | E-NGN-B31E3 (FUSE) | CEDM NORMAL ACU FAN M-HCN-A02A |
| E-PGB-L32E3 | E-NGN-B32E3 (FUSE) | PRESSURIZER BACKUP HEATERS M-RCE-B18, B10, A5 |
| E-PGB-L32E2 | E-NGN-B32E2 (FUSE) | CEDM NORMAL ACU FAN M-HCN-A02B |
| E-PGA-L33D2 | E-NGN-B33D2 (FUSE) | CONTAINMENT NORMAL ACU FAN M-HCN-A01C |
| E-PGA-L33D4 | E-NGN-B33D4 (FUSE) | PRESSURIZER BACKUP HTR, M-RCE B1, B9, A14 |
| E-PGA-L33D3 | E-NGN-B33D3 (FUSE) | CEDM NORMAL ACU FAN M-HCN-A02C |

## TABLE 3.8-2 (Continued)

## CONTAINMENT PENETRATION CONDUCTOR

## OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| E-PGB-L34D2 | E-NGN-B34D2 (FUSE) | CTMT NORMAL ACU FAN M-HCN-A01D |
| E-PGB-L34D3 | E-NGN-B34D3 (FUSE) | CEDM NORMAL ACU FAN M-HCN-A02D |
| E-PGB-L36D3 | E-NGN-B36D3 (FUSE) | CTMT NOR ACU FAN M-HCN-A01B |
| E-PHA-M3318 | E-PHA-M3334 | SAFETY INJECT TANK 4 ISOL VLV J-SIA-UV-644 |
| E-PHA-M3316 | E-PHA-M3316A | SAFETY INJECT TANK 3 ISOL VLV J-SIA-UV-634 |
| E-PHB-M3404 | E-PHB-M3405B | NCWS RET INT CTMT ISOL VLV J-NCB-UV-403 |
| E-PHA-M3517 | E-PHA-M3521 | CTMT PRG RFL MODE ISO VLV J-CPA-UV-2B |
| E-PHA-M3503 | E-PHA-M3507A | SHUT DN CLG ISOL LOOP 1 VLV J-SIA-UV-651 |
| E-PHA-M3508 | E-PHA-M3511A | CTMT/RAD SUMP CTMT INT ISO VLV J-RDA-UV-23 |
| E-PHA-M3512 | E-PHA-M3513A | CTMT SUMP ISOL TRAIN A VLV J-SIA-UV-673 |
| E-PHB-M3622 | E-PHB-M3629 | CTMT PRG REFULING MODE ISO VLV J-CPB-UV-3A |
| E-PHB-M3604 | E-PHB-M3604A | SHUT DN CLG ISOL LOOP 2 VLV J-SIB-UV-652 |
| E-PHB-M3619 | E-PHB-M3641A | SAFETY INJECTION TANK ISOL VLV J-SIB-UV-614 |

## TABLE 3.8-2 (Continued)

## CONTAINMENT PENETRATION CONDUCTOR

## OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| E-PHB-M3613 | E-PHB-M3613A | CTMT SUMP ISOL TRAIN B VLV J-SIB-UV-675 |
| E-PHB-M3618 | E-PHB-M3641 | SAFETY INJECTION TANK 2 ISO VLV J-SIB-UV-624 |
| E-PHA-M3704 | E-PHA-M3703A | WASTE GAS HEADER CONTAINMENT ISOLATION VALVE J-GRA UV1 |
| E-PHA-M3715 | E-PHA-M3719 | $H_2$ CONT TRAIN A UPSTM SUP ISO VLV J-HPA-UV-1 |
| E-PHB-M3816 | E-PHB-M3836 | $H_2$ CTMT TRAIN B UPSTM SUP ISO VLV J-HPB-UV-2 |
| E-PHB-M3811 | E-PHB-M3813A | NORM CHIL WTR RETURN CTMT ISO VLV J-WCB-UV-61 |
| E-PKD-B44 | E-PKD-M4411 | SHUTDOWN CLG ISOL VLV J-SID-UV-654 |
| E-PKC-B43 | E-PKC-M4311 | SHUTDOWN COOLING ISOL VLV J-SIC-UV-653 |
| E-NNN-D1113 | E-NNN-D11 | MOVABLE INCORE DRIVE SYS #I 800VA, M-RIN-M03A VIA E-RIN-J01A |
| E-NNN-D1213 | E-NNN-D12 | MOVABLE INCORE DRIVE SYS #II 800VA, M-RIN-M03B VIA E-RIN-J01A |
| E-NNN-D1526 | E-NNN-D15 | RCP INSTM LOCAL PNL J-RCN-E02 |
| E-NNN-D1525 | E-NNN-D15 | RCP INSTM LOCAL PNL J-RCN-E01 |
| E-NNN-D1626 | E-NNN-D16 | RCP INSTM LOCAL PNL J-RCN-E04 |
| E-NNN-D1625 | E-NNN-D16 | RCP INSTM LOCAL PNL J-RCN-E03 |
| E-QAN-D05B | E-QAN-B02 | LIGHTING PANEL E-QAN-D05B CTMT BLDG EL 100' |

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| E-QAN-D05C | E-QAN-B03 | LIGHTING PANEL E-QAN-D05C CTMT BLDG EL 100' |
| E-QAN-D05D | E-QAN-B04 | LIGHTING PANEL E-QAN-D05D CTMT BLDG EL 140' |
| E-QAN-D05F | E-QAN-B05 | LIGHTING PANEL E-QAN-D05F CTMT BLDG EL 140' |
| E-QAN-D05E | E-QAN-B06 | LIGHTING PANEL E-QAN-D05E CTMT BLDG EL 140' |
| E-QBN-B01 | E-QBN-D91 | LIGHTING PANEL E-QBN-D73A CTMT BLDG EL 100' |
| E-QBN-B02 | E-QBN-D91 | LIGHTING PANEL E-QBN-D73B CTMT BLDG EL 140' |
| E-NHN-D1514 | E-NHN-M1526 | TO OPERATION CAMERA JB# 2 |
| E-RCN-D0102 | E-NGN-L11C2 | PZR BU HTR M-RCE-B07, B13, A01 |
| E-NHN-D2614 | E-NHN-M2618 | TO OPERATION CAMERA JB# 1 |
| E-RCN-D0101 | E-NGN-L11C2 | PZR BU HTR M-RCE-B03, A09, A15 |
| E-RCN-D0301 | E-NGN-L11C3 | PZR BU HTR M-RCE-B04, A11, A16 |
| E-RCN-D0302 | E-NGN-L11C3 | PZR BU HTR M-RCE-A02, A07, A13 |
| E-RCN-D0201 | E-NGN-L12C2 | PZR BU HTR M-RCE-B06, B12, A18 |
| E-RCN-D0202 | E-NGN-L12C2 | PZR BU HTR M-RCE-B16, A04, A08 |
| E-RCN-D0401 | E-NGN-L12C3 | PZR BU HTR M-RCE-B15, A03, A10 |
| E-RCN-D0402 | E-NGN-L12C3 | PZR BU HTR M-RCE-A17, A06, A12 |

## TABLE 3.8-2 (Continued)

## CONTAINMENT PENETRATION CONDUCTOR

## OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| E-NAN-S01M | E-NAN-S01A<br>E-NAN-S03B | RCP M-RCE-P01A (C.E. NO. 1A) |
| E-NAN-S01L | E-NAN-S01A<br>E-NAN-S03B | RCP M-RCE-P01C (C.E. NO. 2A) |
| E-NAN-S02L | E-NAN-S02A<br>E-NAN-S04B | RCP M-RCE-P01B (C.E. NO. 1B) |
| E-NAN-S02M | E-NAN-S02A<br>E-NAN-S04B | RCP M-RCE-P01D (C.E. NO. 2B) |
| E-NGN-L03C2 | FUSE IN BKR. | CTMT NOR DUCT HTR M-HCN-E01C |
| E-NGN-L03C3 | FUSE IN BKR. | CTMT NOR DUCT HTR M-HCN-E01D |
| E-NGN-L03D2 | FUSE IN BKR. | CTMT POLAR CRANE M-ZCN-G01 |
| E-NGN-L06C2 | E-NGN-B06C2<br>(FUSE) | CTMT PRE-ACCESS NORM AFU FAN M-HCN-F01A |
| E-NGN-L09C4 | E-NGN-B09C4<br>(FUSE) | CTMT PRE-ACCESS NORM AFU FAN M-HCN-F01B |
| E-NGN-L10C2 | FUSE IN BKR. | CTMT NORM DUCT HTR M-HCN-E01A |
| E-NGN-L10C3 | FUSE IN BKR. | CTMT NORM DUCT HTR M-HCN-E01B |
| J-RCN-PC100A<br>(FUSE) | E-NGN-L11C4 | PROPORTIONAL HTR BANK M-RCE-B2, B8, B14 |
| J-RCN-PC100B<br>(FUSE) | E-NGN-L12C4 | PROPORTIONAL HTR BANK M-RCE-B5, B11, B17 |
| CEA 06 CB101 | F101, F102, F103 | CEA 06 |
| CEA 08 CB102 | F104, F105, F106 | CEA 08 |
| CEA 10 CB103 | F107, F108, F109 | CEA 10 |

## TABLE 3.8-2 (Continued)

## CONTAINMENT PENETRATION CONDUCTOR

## OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| CEA 12 CB104 | F110, F111, F112 | CEA 12 |
| CEA 07 CB101 | F101, F102, F103 | CEA 07 |
| CEA 09 CB102 | F104, F105, F106 | CEA 09 |
| CEA 11 CB103 | F107, F108, F109 | CEA 11 |
| CEA 13 CB104 | F110, F111, F112 | CEA 13 |
| CEA 74 CB101 | F101, F102, F103 | CEA 74 |
| CEA 76 CB102 | F104, F105, F106 | CEA 76 |
| CEA 78 CB103 | F107, F108, F109 | CEA 78 |
| CEA 80 CB104 | F110, F111, F112 | CEA 80 |
| CEA 75 CB101 | F101, F102, F103 | CEA 75 |
| CEA 77 CB102 | F104, F105, F106 | CEA 77 |
| CEA 79 CB103 | F107, F108, F109 | CEA 79 |
| CEA 81 CB104 | F110, F111, F112 | CEA 81 |
| CEA 22 CB101 | F101, F102, F103 | CEA 22 |
| CEA 24 CB102 | F104, F105, F106 | CEA 24 |
| CEA 26 CB103 | F107, F108, F109 | CEA 26 |
| CEA 28 CB104 | F110, F111, F112 | CEA 28 |
| CEA 23 CB101 | F101, F102, F103 | CEA 23 |
| CEA 25 CB102 | F104, F105, F106 | CEA 25 |

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| CEA 27 CB103 | F107, F108, F109 | CEA 27 |
| CEA 29 CB104 | F110, F111, F112 | CEA 29 |
| CEA 34 CB101 | F101, F102, F103 | CEA 34 |
| CEA 36 CB102 | F104, F105, F106 | CEA 36 |
| CEA 38 CB103 | F107, F108, F109 | CEA 38 |
| CEA 40 CB104 | F110, F111, F112 | CEA 40 |
| CEA 35 CB101 | F101, F102, F103 | CEA 35 |
| CEA 37 CB102 | F104, F105, F106 | CEA 37 |
| CEA 39 CB103 | F107, F108, F109 | CEA 39 |
| CEA 41 CB104 | F110, F111, F112 | CEA 41 |
| CEA 55 CB101 | F101, F102, F103 | CEA 55 |
| CEA 58 CB102 | F104, F105, F106 | CEA 58 |
| CEA 61 CB103 | F107, F108, F109 | CEA 61 |
| CEA 64 CB104 | F110, F111, F112 | CEA 64 |
| CEA 54 CB101 | F101, F102, F103 | CEA 54 |
| CEA 57 CB102 | F104, F105, F106 | CEA 57 |
| CEA 60 CB103 | F107, F108, F109 | CEA 60 |
| CEA 63 CB104 | F110, F111, F112 | CEA 63 |
| CEA 56 CB101 | F101, F102, F103 | CEA 56 |
| CEA 59 CB102 | F104, F105, F106 | CEA 59 |

## TABLE 3.8-2 (Continued)

## CONTAINMENT PENETRATION CONDUCTOR

## OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| CEA 62 CB103 | F107, F108, F109 | CEA 62 |
| CEA 65 CB104 | F110, F111, F112 | CEA 65 |
| CEA 66 CB101 | F101, F102, F103 | CEA 66 |
| CEA 68 CB102 | F104, F105, F106 | CEA 68 |
| CEA 70 CB103 | F107, F108, F109 | CEA 70 |
| CEA 72 CB104 | F110, F111, F112 | CEA 72 |
| CEA 67 CB101 | F101, F102, F103 | CEA 67 |
| CEA 69 CB102 | F104, F105, F106 | CEA 69 |
| CEA 71 CB103 | F107, F108, F109 | CEA 71 |
| CEA 73 CB104 | F110, F111, F112 | CEA 73 |
| CEA 02 CB101 | F101, F102, F103 | CEA 02 |
| CEA 03 CB102 | F104, F105, F106 | CEA 03 |
| CEA 04 CB103 | F107, F108, F109 | CEA 04 |
| CEA 05 CB104 | F110, F111, F112 | CEA 05 |
| CEA 42 CB101 | F101, F102, F103 | CEA 42 |
| CEA 43 CB102 | F104, F105, F106 | CEA 43 |
| CEA 44 CB103 | F107, F108, F109 | CEA 44 |
| CEA 45 CB104 | F110, F111, F112 | CEA 45 |
| CEA 82 CB101 | F101, F102, F103 | CEA 82 |
| CEA 83 CB102 | F104, F105, F106 | CEA 83 |

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| CEA 84 CB103 | F107, F108, F109 | CEA 84 |
| CEA 85 CB104 | F110, F111, F112 | CEA 85 |
| CEA 18 CB101 | F101, F102, F103 | CEA 18 |
| CEA 19 CB102 | F104, F105, F106 | CEA 19 |
| CEA 20 CB103 | F107, F108, F109 | CEA 20 |
| CEA 21 CB104 | F110, F111, F112 | CEA 21 |
| CEA 86 CB101 | F101, F102, F103 | CEA 86 |
| CEA 87 CB102 | F104, F105, F106 | CEA 87 |
| CEA 88 CB103 | F107, F108, F109 | CEA 88 |
| CEA 89 CB104 | F110, F111, F112 | CEA 89 |
| CEA 14 CB101 | F101, F102, F103 | CEA 14 |
| CEA 15 CB102 | F104, F105, F106 | CEA 15 |
| CEA 16 CB103 | F107, F108, F109 | CEA 16 |
| CEA 17 CB104 | F110, F111, F112 | CEA 17 |
| CEA 46 CB101 | F101, F102, F103 | CEA 46 |
| CEA 48 CB102 | F104, F105, F106 | CEA 48 |
| CEA 50 CB103 | F107, F108, F109 | CEA 50 |
| CEA 52 CB104 | F110, F111, F112 | CEA 52 |
| CEA 47 CB101 | F101, F102, F103 | CEA 47 |
| CEA 49 CB102 | F104, F105, F106 | CEA 49 |

## TABLE 3.8-2 (Continued)

## CONTAINMENT PENETRATION CONDUCTOR

## OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| CEA 51 CB103 | F107, F108, F109 | CEA 51 |
| CEA 53 CB104 | F110, F111, F112 | CEA 53 |
| CEA 30 CB101 | F101, F102, F103 | CEA 30 |
| CEA 31 CB102 | F104, F105, F106 | CEA 31 |
| CEA 32 CB103 | F107, F108, F109 | CEA 32 |
| CEA 33 CB104 | F110, F111, F112 | CEA 33 |
| CEA 01 CB101 | F101, F102, F103 | CEA 01 |
| E-PHA-D33-03 | E-PHA-M3332 | INDICATING LIGHTS FOR VLV J-SIA-UV-634 |
| E-PHA-D33-04 | E-PHA-M3332 | INDICATING LIGHTS FOR VLV J-SIA-UV-644 |
| E-PHB-D36-01 | E-PHB-M3638 | INDICATING LIGHTS FOR VLV J-SIB-UV-614 |
| E-PHB-D36-02 | E-PHA-M3638 | INDICATING LIGHTS FOR VLV J-SIB-UV-624 |
| E-NHN-D28-04 | E-NHN-M2830 | CONTAINMENT PREACCESS NORMAL AFU MOTOR SPACE HEATER FOR M-HCN-F01AH |
| E-NHN-D28-14 | E-NHN-M2830 | FLOW SWITCH J-HCN-FSL-29 FOR DUCT HEATERS M-HCN-E01A AND B |
| E-NHN-D28-16 | E-NHN-M2830 | CONTAINMENT ACU DUCT HEATERS M-HCN-E01A AND B TEMPERATURE CONTROL J-HCN-TC-29 |
| E-NHN-D28-18 | E-NHN-M2830 | FLOW SWITCH J-HCN-FSL-31 FOR DUCT HEATERS M-HCN-E01C AND D |
| E-NHN-D13-04 | E-NHN-M1329 | CONTAINMENT ACU DUCT HEATERS M-HCN-E01C AND D TEMPERATURE CONTROLLER J-HCN-TC-31 |

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| E-NHN-D13-22 | E-NHN-M1329 | STEAM GENERATOR WET LAYUP PUMP MOTOR SPACE HEATER M-SGN-P01AH |
| E-NHN-D15-01 | E-NHN-M1526 | REACTOR COOLANT PUMP MOTOR SPACE HEATER M-RCE-P01BH |
| E-NHN-D15-02 | E-NHN-M1526 | REACTOR COOLANT PUMP MOTOR SPACE HEATER M-RCE-P01DH |
| E-NHN-D15-06 | E-NHN-M1526 | CONTAINMENT PREACCESS NORMAL AFU FAN MOTOR SPACE HEATER M-HCN-F01BH |
| E-NHN-D10-01 | E-NHN-M1027 | REACTOR COOLANT PUMP MOTOR SPACE HEATER M-RCE-P01AH |
| E-NHN-D10-02 | E-NHN-M1027 | REACTOR COOLANT PUMP MOTOR SPACE HEATER M-RCE-P01CH |
| E-NHN-D10-20 | E-NHN-M1027 | STEAM GENERATOR WET LAYUP PUMP MOTOR SPACE HEATER M-SGN-P01BH |
| E-NHN-D19-05 | E-NHN-M1914 | CEDM NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A02AH |
| E-NHN-D19-06 | E-NHN-M1914 | CEDM NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A02CH |
| E-NHN-D19-07 | E-NHN-M1914 | CONTAINMENT NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A01AH |
| E-NHN-D19-08 | E-NHN-M1914 | CONTAINMENT NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A01CH |
| E-NHN-D19-10 | E-NHN-M1914 | REACTOR CAVITY NORMAL COOLING FAN MOTOR SPACE HEATER M-HCN-A03AH |
| E-NHN-D19-12 | E-NHN-M1914 | REACTOR CAVITY NORMAL COOLING FAN MOTOR SPACE HEATER M-HCN-A03CH |

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| E-NHN-D20-05 | E-NHN-M2013 | CEDM NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A02BH |
| E-NHN-D20-06 | E-NHN-M2013 | CEDM NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A02DH |
| E-NHN-D20-07 | E-NHN-M2013 | CONTAINMENT NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A01DH |
| E-NHN-D20-08 | E-NHN-M2013 | CONTAINMENT NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A01BH |
| E-NHN-D20-10 | E-NHN-M2013 | REACTOR CAVITY NORMAL COOLING FAN MOTOR SPACE HEATER M-HCN-A03BH |
| E-NHN-D20-12 | E-NHN-M2013 | REACTOR CAVITY NORMAL COOLING FAN MOTOR SPACE HEATER M-HCN-A03DH |

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| E-ZAB-C06 (FUSE) | E-PKB-D2221 | SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIB-HV-622 |
| E-ZAB-C06 (FUSE) | E-PKB-D2221 | SAFETY INJ TANK VENT VALVE J-SIB-HV-613 |
| E-ZAB-C06 (FUSE) | E-PKB-D2221 | SAFETY INJ TANK VENT VALVE J-SIB-HV-623 |
| E-ZAB-C06 (FUSE) | E-PKB-D2221 | SAFETY INJ TANK VENT VALVE J-SIB-HV-633 |
| E-ZAB-C06 (FUSE) | E-PKB-D2221 | SAFETY INJ TANK VENT VALVE J-SIB-HV-643 |
| E-ZAB-C06 (FUSE) | E-PKB-D2221 | REACTOR COOLANT VENT VALVE J-RCB-HV-105 |
| E-ZAB-C06 (FUSE) | E-PKB-D2221 | SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIB-UV-612 |
| E-ZJA-C01 (FUSE) | E-PKA-D2101 | SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIA-HV-639 |
| E-ZJA-C01 (FUSE) | E-PKA-D2101 | SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIA-HV-649 |
| E-ZJA-C03 (FUSE) | E-PKA-D2111 | RCP CONTROLLED BLEEDOFF TO RDT VALVE J-CHA-HV-507 |
| E-ZJA-C03 (FUSE) | E-PKA-D2111 | LETDOWN LINE TO REGEN HEAT EXCH CTMT ISO VALVE J-CHA-HV-516 |
| E-ZJA-C03 (FUSE) | E-PKA-D2111 | RCP CONTROLLED BLEEDOFF TO VCT VALVE J-CHA-UV-506 |
| E-ZJB-C01 (FUSE) | E-PKB-D2201 | SAFETY INJ TANK FILL AND DRAIN VALVE J-SIB-UV-641 |
| E-ZJB-C01 (FUSE) | E-PKB-D2201 | SI TANK CHECK VALVE LEAKAGE ISO VALVE J-SIB-UV-648 |
| E-ZJB-C01 (FUSE) | E-PKB-D2201 | HOT LEG INJECT CHECK VLV LEAKAGE ISO VLV J-SIB-UV-322 |
| E-ZJB-C01 (FUSE) | E-PKB-D2201 | SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIB-HV-632 |

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| E-ZJB-C01 (FUSE) | E-PKB-D2201 | SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIB-HV-642 |
| E-ZJB-C03 (FUSE) | E-PKB-D2211 | LETDOWN LINE TO REGEN HEAT EXCH VALVE J-CHB-UV-515 |
| E-ZJB-C03 (FUSE) | E-PKB-D2211 | SAFETY INJ TANK FILL AND DRAIN VALVE J-SIB-UV-631 |
| E-ZJB-C03 (FUSE) | E-PKB-D2211 | SI TANK CHECK VLV LEAKAGE LINE ISO VALVE J-SIB-UV-638 |
| E-ZJB-C03 (FUSE) | E-PKB-D2211 | HOT LEG INJECT CHECK VLV LEAKAGE LINE ISO VALVE J-SIB-UV-332 |
| E-ZAA-C03 (FUSE) | E-PKA-D2109 | REACTOR DRAIN TANK OUTLET ISOLATION VALVE J-CHA-UV-560 |
| E-ZAA-C03 (FUSE) | E-PKA-D2109 | SI TANK RWT HDR CTMT ISOLATION VALVE J-SIA-UV-682 |
| E-ZAA-C03 (FUSE) | E-PKA-D2109 | REACTOR COOLANT VENT VALVE J-RCA-HV-101 |
| E-ZAA-C03 (FUSE) | E-PKA-D2109 | REGENERATIVE HEAT EXCH TO AUX SPRAY VALVE J-CHA-HV-205 |
| E-ZAA-C01 (FUSE) | E-PKA-D2110 | SAMPLE CONTAINMENT ISOLATION VALVE J-SSA-UV-203 |
| E-ZAA-C01 (FUSE) | E-PKA-D2110 | SAMPLE CONTAINMENT ISOLATION VALVE J-SSA-UV-204 |
| E-ZAA-C01 (FUSE) | E-PKA-D2110 | SAMPLE CONTAINMENT ISOLATION VALVE J-SSA-UV-205 |
| E-ZAA-C04 (FUSE) | E-PKA-D2102 | PRESSURIZER VENT VALVE J-RCA-HV-103 |
| E-ZAA-C04 (FUSE) | E-PKA-D2130 | CTMT PRG PWR ACCESS MODE ISO VLV J-CPA-UV-4B |
| E-ZAA-C04 (FUSE) | E-PKA-D2130 | CONTAINMENT PURGE POWER ACCESS MODE ISOLATION VALVE J-CPA-UV-4A |

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| E-ZAA-C05 (FUSE) | E-PKA-D2114 | STEAM GEN BLOWDOWN CTMT ISOLATION VALVE J-SGA-UV-500P |
| E-ZAA-C05 (FUSE) | E-PKA-D2114 | BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGA-UV-204 |
| E-ZAA-C05 (FUSE) | E-PKA-D2114 | BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGA-UV-211 |
| E-ZAA-C05 (FUSE) | E-PKA-D2114 | BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGA-UV-220 |
| E-ZAA-C06 (FUSE) | E-PKA-D2121 | SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIA-HV-619 |
| E-ZAA-C06 (FUSE) | E-PKA-D2121 | SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIA-HV-629 |
| E-ZAA-C06 (FUSE) | E-PKA-D2121 | SAFETY INJ TANK VENT VALVE J-SIA-HV-605 |
| E-ZAA-C06 (FUSE) | E-PKA-D2121 | SAFETY INJ TANK VENT VALVE J-SIA-HV-606 |
| E-ZAA-C06 (FUSE) | E-PKA-D2121 | SAFETY INJ TANK VENT VALVE J-SIA-HV-607 |
| E-ZAA-C06 (FUSE) | E-PKA-D2121 | SAFETY INJ TANK VENT VALVE J-SIA-HV-608 |
| E-ZAA-C06 (FUSE) | E-PKA-D2121 | RC SYSTEM VENT TO CTMT VALVE J-RCA-HV-106 |
| E-ZAB-C03 (FUSE) | E-PKB-D2209 | REGEN HEAT EXCH TO AUX SPRAY VALVE J-CHB-HV-203 |
| E-ZAB-C03 (FUSE) | E-PKB-D2209 | REACTOR COOLANT VENT VALVE J-RCB-HV-102 |
| E-ZAB-C03 (FUSE) | E-PKB-D2209 | SAFETY INJ TANK FILL AND DRAIN VALVE J-SIB-UV-611 |
| E-ZAB-C03 (FUSE) | E-PKB-D2209 | SI TANK CHECK VALVE LEAKAGE LINE ISO VALVE J-SIB-UV-618 |
| E-ZAB-C01 (FUSE) | E-PKB-D2210 | CTMT ATM RADIATION MONITORING ISO VALVE J-HCB-UV-44 |

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| E-ZAB-C01 (FUSE) | E-PKB-D2210 | CTMT ATM RADIATION MONITORING ISO VALVE J-HCB-UV-47 |
| E-ZAB-C01 (FUSE) | E-PKB-D2210 | CONTAINMENT POWER ACCESS PURGE MODE ISOLATION VALVE J-CPB-UV-5A |
| E-ZAB-C01 (FUSE) | E-PKB-D2210 | CONTAINMENT POWER ACCESS PURGE MODE ISOLATION VALVE J-CPB-UV-5B |
| E-ZAB-C04 (FUSE) | E-PKB-D2202 | REACTOR COOLANT VENT VALVE J-RCB-HV-108 |
| E-ZAB-C04 (FUSE) | E-PKB-D2202 | SAFETY INJ TANK FILL AND DRAIN VALVE J-SIB-UV-621 |
| E-ZAB-C04 (FUSE) | E-PKB-D2202 | SI TANK CHECK VALVE LEAKAGE LINE ISO VALVE J-SIB-UV-628 |
| E-ZAB-C05 (FUSE) | E-PKB-D2214 | REACTOR COOLANT VENT VALVE J-RCB-HV-109 |
| E-ZAB-C05 (FUSE) | E-PKB-D2214 | STEAM GEN BLOWDOWN CTMT ISOLATION VALVE J-SGB-UV-500R |
| E-ZAB-C05 (FUSE) | E-PKB-D2214 | BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGB-UV-222 |
| E-ZAB-C05 (FUSE) | E-PKB-D2214 | BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGB-UV-224 |
| E-ZAB-C05 (FUSE) | E-PKB-D2214 | BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGB-UV-226 |
| E-ZAN-C01 (FUSE) | E-NKN-D4226 | SEAL INJECT VALVES TO RCP J-CHE-FV-241 |
| E-ZAN-C01 (FUSE) | E-NKN-D4224 | SEAL INJECT VALVES TO RCP J-CHE-FV-242 |
| E-ZAN-C01 (FUSE) | E-NKN-D4222 | SEAL INJECT VALVES TO RCP J-CHE-FV-244 |
| E-ZAN-C01 (FUSE) | E-NKN-D4224 | POST ACDT SMPLG SYS ISO VALVE J-CHN-HV-923 |

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

| PRIMARY DEVICE NUMBER | BACKUP DEVICE NUMBER | SERVICE DESCRIPTION |
|---|---|---|
| E-ZAN-C01 (FUSE) | E-NKN-D4224 | REACTOR VESSEL SEAL DRAIN TO RDT VALVE J-RCE-HV-403 |
| E-ZAN-C01 (FUSE) | E-NKN-D4224 | SI DRAIN TO REACTOR DRAIN TANK VALVE J-SIE-HV-661 |
| E-ZAN-C02 (FUSE) | E-NKN-D4216 | SEAL INJECT VALVES TO RCP J-CHE-FV-243 |
| E-ZAN-C02 (FUSE) | E-NKN-D4216 | REGEN HEAT EXCH TO CHARGING LINE VALVE J-CHE-PDV-240 |
| E-PGB-L32E2 | E-NGN-B32E2 (FUSE) | CEDM NORM ACU FAN - B M-HCN-A02B |
| E-PGB-L34D2 | E-NGN-B34D2 (FUSE) | CTMT NORM ACU FAN - D M-HCN-A01D |
| E-PNC-D2719 (FUSE) | E-PNC-D27 | SAFETY INJECTION SHUTDOWN COOLING ISOLATION VALVE POSITION INDICATION J-SIC-UV-653 |
| E-PND-D2819 (FUSE) | E-PND-D28 | SAFETY INJECTION SHUTDOWN COOLING ISOLATION VALVE POSITION INDICATION J-SID-UV-654 |
| E-PNA-D2519 (FUSE) | E-PNA-D25 | MAIN PANEL BREAKER/SHUTDOWN COOLING ISOLATION VALVE J-SIA-UV-651 - POSITION INDICATION |
| E-PNB-D2619 (FUSE) | E-PNB-D26 | MAIN PANEL BREAKER/SHUTDOWN COOLING ISOLATION VALVE J-SIB-UV-652 - POSITION INDICATION |
| E-NHN-D1506 | E-NHN-M1526 | CTMT PRE-ACCESS NORMAL AFU FAN MOTOR HEATER M-HCN-F01BH |

## MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

### LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection of each valve shown in Table 3.8-3 shall be bypassed continuously or under accident conditions, as applicable, by an OPERABLE device integral with the motor starter.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not bypassed continuously or under accident conditions, as applicable, by an OPERABLE integral bypass device, take administrative action to continuously bypass the thermal overload within 8 hours or declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) for the affected valve(s).

### SURVEILLANCE REQUIREMENTS

4.8.4.2.1 The thermal overload protection for the above required valves shall be verified to be bypassed continuously or under accident conditions, as applicable, by an OPERABLE integral bypass device by the performance of a CHANNEL FUNCTIONAL TEST of the bypass circuitry for those thermal overloads which are normally in force during plant operation and bypassed under accident conditions and by verifying that the thermal overload protection is bypassed for those thermal overloads which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing:

a. At least once per 18 months, and

b. Following maintenance on the motor starter.

4.8.4.2.2 The thermal overload protection for the above required valves which are continuously bypassed shall be verified to be bypassed following testing during which the thermal overload protection was temporarily placed in force.

# TABLE 3.8-3

## MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND/OR BYPASS DEVICES

| VALVE NUMBER | BYPASS DEVICE (Accident Conditions) | SYSTEM(S) AFFECTED |
|---|---|---|
| J-SIA-UV-647 | HPSI A Flow Control to Reactor Coolant Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-UV-637 | HPSI A Flow Control to Reactor Coolant Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-HV-604 | HPSI Pump A Long Term Cooling Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-HV-609 | HPSI Pump B Long Term Cooling Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-HV-657 | Shutdown Clg. Temp. Control Train A Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-HV-658 | Shutdown Clg. Temp. Control Train B Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-HV-685 | LPSI - Ctmt Spray Pump Cross Connect A Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-HV-694 | LPSI- Ctmt Spray Pump Cross Connect B Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-HV-686 | Ctmt Spray A Cross Connect Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-HV-696 | Ctmt Spray B Cross Connect Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-HV-688 | Shutdown Clg. Heat Exchange A Bypass Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-HV-693 | Shutdown Clg. Heat Exchange B Bypass Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-UV-617 | HPSI A Flow Control To React Coolant 2A Valve | Safety Injection Shutdown Clg. Sys. |

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

| VALVE NUMBER | BYPASS DEVICE (Accident Conditions) | SYSTEM(S) AFFECTED |
|---|---|---|
| J-SIA-UV-627 | HPSI A Flow Control To React Coolant 2B Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-UV-645 | LPSI Flow Control To React Coolant 1B Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-UV-635 | LPSI Flow Control To React Coolant 1A Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-UV-644 | Safety Injection Tank 1B Isolation Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-UV-634 | Safety Injection Tank 1A Isolation Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-UV-616 | HPSI B Flow Control To React Coolant 2A Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-UV-626 | HPSI B Flow Control To React Coolant 2B Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-UV-636 | HPSI B Flow Control To React Coolant 1A Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-UV-646 | HPSI B Flow Control To React Coolant 1B Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-UV-655 | Shutdown Clg. Ctmt Isolation Loop 1 Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-UV-656 | Shutdown Clg. Ctmt Isolation Loop 2 Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-UV-664 | Ctmt Spray Pump A To Refueling Water Tank Isolation Vlv. | Safety Injection Shutdown Clg. Sys. |

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

| VALVE NUMBER | BYPASS DEVICE (Accident Conditions) | SYSTEM(S) AFFECTED |
|---|---|---|
| J-SIB-UV-665 | Ctmt Spray Pump B To Refueling Water Tank Isolation Vlv. | Safety Injection Shutdown Clg. Sys. |
| J-SIB-UV-615 | LPSI Flow Control To React Coolant 2A Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-UV-625 | LPSI B Flow Control To React Coolant 2B Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-UV-666 | HPSI Pump A to Refueling Water Tank Isolation | Safety Injection Shutdown Clg. Sys. |
| J-SIB-UV-667 | HPSI Pump B to Refueling Water Tank Isolation | Safety Injection Shutdown Clg. Sys. |
| J-SIA-UV-669 | LPSI Pump A To Refueling Water Tank Isolation | Safety Injection Shutdown Clg. Sys. |
| J-SIB-UV-668 | LPSI Pump B to Refueling Water Tank Isolation | Safety Injection Shutdown Clg. Sys. |
| J-SIA-UV-672 | Ctmt Spray Control Train A Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-UV-671 | Ctmt Spray Control Train B Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-UV-674 | Ctmt Sump Isolation Train A Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-UV-676 | Ctmt Sump Isolation Train B Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-UV-651 | Shutdown Clg. Isolation Loop 1 Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-UV-652 | Shutdown Clg. Isolation Loop 2 Valve | Safety Injection Shutdown Clg. Sys. |

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

| VALVE NUMBER | BYPASS DEVICE (Accident Conditions) | SYSTEM(S) AFFECTED |
|---|---|---|
| J-SIA-UV-673 | Ctmt Sump Isolation Train A Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-UV-675 | Ctmt Sump Isolation Train B Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-UV-614 | Safety Injection Tank 2A Isolation Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-UV-624 | Safety Injection Tank 2B Isolation Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-HV-684 | Shutdown Clg. Heat Exchange Isolation Train A | Safety Injection Shutdown Clg. Sys. |
| J-SIB-HV-689 | Shutdown Clg. Heat Exchange Isolation Train B | Safety Injection Shutdown Clg. Sys. |
| J-SIA-HV-683 | LPSI Pump A Isolation Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-HV-692 | LPSI Pump B Isolation Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-HV-691 | Shutdown Clg. Loop 2 Warm-Up Bypass Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-HV-690 | Shutdown Clg. Loop 1 Warm-Up Bypass Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-HV-698 | HPSI Pump A Discharge Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-HV-699 | HPSI Pump B Discharge Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-HV-306 | LPSI Pump A Header Discharge Valve | Safety Injection Shutdown Clg. Sys. |

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

| VALVE NUMBER | BYPASS DEVICE (Accident Conditions) | SYSTEM(S) AFFECTED |
|---|---|---|
| J-SIB-HV-307 | LPSI Pump B Header Discharge Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-HV-687 | Ctmt Spray Isolation Train A Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIB-HV-695 | Ctmt Spray Isolation Train B Valve | Safety Injection Shutdown Clg. Sys. |
| J-SIA-HV-678 | Shutdown Clg. Heat Exchange Isolation Train A | Safety Injection Shutdown Clg. Sys. |
| J-SIB-HV-679 | Shutdown Clg. Heat Exchange Isolation Train B | Safety Injection Shutdown Clg. Sys. |
| J-SIC-UV-653 | Shutdown Clg. Isolation Valve | Safety Injection Shutdown Clg. Sys. |
| J-SID-UV-654 | Shutdown Clg. Isolation Valve | Safety Injection Shutdown Clg. Sys. |
| J-EWA-UV-65 | ECW Loop A To/From NCW Cross Tie Valve | Essential Cooling Water System |
| J-EWA-UV-145 | ECW Loop A To/From NCW Cross Tie Valve | Essential Cooling Water System |
| J-CTA-HV-1 | Condensate Tank to Aux. Feedwater Pump Valve | Condensate Transfer & Storage Sys. |
| J-CTA-HV-4 | Condensate Tank to Aux. Feedwater Pump Valve | Condensate Transfer & Storage Sys. |
| J-SGA-UV-134 | SG-1 Aux. Feedwater Pump A Steam Supply | Main Steam System |
| J-SGA-UV-138 | SG-2 Aux. Feedwater Pump A Steam Supply | Main Steam System |

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

| VALVE NUMBER | BYPASS DEVICE (Accident Conditions) | SYSTEM(S) AFFECTED |
|---|---|---|
| J-NCB-UV-401 | NCWS Ctmt Isolation Valve | Nuclear Cooling Water System |
| J-NCA-UV-402 | NCWS Ctmt Isolation Valve | Nuclear Cooling Water System |
| J-NCB-UV-403 | NCWS Ctmt Isolation Valve | Nuclear Cooling Water System |
| J-AFB-HV-30 | Aux. Feedwater Regulating Valve | Auxiliary Feed-water System |
| J-AFB-HV-31 | Aux. Feedwater Regulating Valve | Auxiliary Feed-water System |
| J-AFB-UV-34 | Aux. Feedwater Isolation Valve | Auxiliary Feed-water System |
| J-AFB-UV-35 | Aux. Feedwater Isolation Valve | Auxiliary Feed-water System |
| J-AFA-HV-32 | Aux. Feedwater Regulating Valve | Auxiliary Feed-water System |
| J-AFA-UV-37 | Aux. Feedwater Isolation Valve | Auxiliary Feed-water System |
| J-AFC-UV-36 | Aux. Feedwater Isolation Valve | Auxiliary Feed-water System |
| J-AFC-HV-33 | Aux. Feedwater Regulating Valve | Auxiliary Feed-water System |
| J-CPA-UV-2A | Ctmt Purge Refueling Mode Isolation Valve | Containment Purge System |
| J-CPB-UV-3B | Ctmt Purge Refueling Mode Isolation Valve | Containment Purge System |
| J-CPA-UV-2B | Ctmt Purge Refueling Mode Isolation Valve | Containment Purge System |

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

| VALVE NUMBER | BYPASS DEVICE (Accident Conditions) | SYSTEM(S) AFFECTED |
|---|---|---|
| J-CPB-UV-3A | Ctmt Purge Refueling Mode Isolation Valve | Containment Purge System |
| J-WCA-UV-62 | Normal Chill Water Return Ctmt Isolation | Chilled Water System |
| J-WCB-UV-63 | Normal Chill Water Supply Ctmt Isolation | Chilled Water System |
| J-WCB-UV-61 | Normal Chill Water Return Ctmt Isolation | Chilled Water System |
| J-RDA-UV-23 | Ctmt Radwaste Sumps Internal Isolation | Radioactive Waste Drain System |
| J-HPA-UV-3 | $H_2$ Ctmt Train A Downstream Supply Isolation | Containment Hydrogen Control Sys. |
| J-HPA-UV-5 | $H_2$ Ctmt Train A Return Isolation Valve | Containment Hydrogen Control Sys. |
| J-HPB-UV-4 | $H_2$ Ctmt Train B Downstream Supply Isolation | Containment Hydrogen Control Sys. |
| J-HPB-UV-6 | $H_2$ Ctmt Train B Return Isolation Valve | Containment Hydrogen Control Sys. |
| J-HPB-UV-2 | $H_2$ Ctmt Train B Upstream Supply Isolation | Containment Hydrogen Control Sys. |
| J-HPA-UV-1 | $H_2$ Ctmt Train A Upstream Supply Isolation | Containment Hydrogen Control Sys. |
| J-GRA-UV-1 | Radioactive Drain Tk Gas Surge Hdr Internal Containment Isolation | Gaseous Radwaste System |

## 3/4.9 REFUELING OPERATIONS

### 3/4.9.1 BORON CONCENTRATION

#### LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a $K_{eff}$ of 0.95 or less, or

b. A boron concentration of greater than or equal to 2150 ppm.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 26 gpm of a solution containing $\geq$ 4000 ppm boron or its equivalent until $K_{eff}$ is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2150 ppm, whichever is the more restrictive.

#### SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

## 3/4.9.2 INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two startup channel neutron flux monitors shall be OPERABLE and operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

### SURVEILLANCE REQUIREMENTS

4.9.2 Each startup channel neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL FUNCTIONAL TEST at least once per 7 days.

## 3/4.9.3 DECAY TIME

### LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 100 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

### SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

## 3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

### LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

   1. Closed by an isolation valve, blind flange, or manual valve, or

   2. Be capable of being closed by an OPERABLE automatic containment purge valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

### SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment purge valve within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment purge valves per the applicable portions of Specification 4.9.9.

## 3/4.9.5 COMMUNICATIONS

### LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

### SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

## 3/4.9.6 REFUELING MACHINE

### LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of fuel assemblies and shall be OPERABLE with:

a. A minimum capacity of 3590 pounds and an overload cut off limit of less than or equal to 1556 (1727)* pounds for the refueling machine.

APPLICABILITY: During movement of fuel assemblies within the refueling cavity.

ACTION:

With the above requirements for the refueling machine not satisfied, suspend use of the refueling machine from operations involving the movement of fuel assemblies.

### SURVEILLANCE REQUIREMENTS

4.9.6.1 The refueling machine used for movement of fuel assemblies shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 3590 pounds and demonstrating an automatic load cut off when the refueling machine load exceeds 1556 (1727)* pounds.

*For initial fuel load only.

## 3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

### LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition.

### SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of 2000 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

## 3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

## HIGH WATER LEVEL

### LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling loop shall be OPERABLE and in operation*.

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is greater than or equal to 23 feet.

ACTION:

With no shutdown cooling loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

### SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

*The shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs or during surveillance testing of ECCS pumps.

REFUELING OPERATIONS

LOW WATER LEVEL

## LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent shutdown cooling loops shall be OPERABLE and at least one shutdown cooling loop shall be in operation*.

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor pressure vessel flange, as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

## SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

*The shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs or during surveillance testing of ECCS pumps.

## 3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

### LIMITING CONDITION FOR OPERATION

3.9.9 The containment purge valve isolation system shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the containment purge valve isolation system inoperable, close each of the containment purge penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.9 The containment purge valve isolation system shall be demonstrated OPERABLE within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment purge valve isolation occurs on manual initiation and on CPIAS.

## 3/4.9.10 WATER LEVEL - REACTOR VESSEL

### FUEL ASSEMBLIES

### LIMITING CONDITION FOR OPERATION

3.9.10.1 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies within the reactor pressure vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies within the pressure vessel.

### SURVEILLANCE REQUIREMENTS

4.9.10.1 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies.

## LIMITING CONDITION FOR OPERATION

3.9.10.2 At least 23 feet of water shall be maintained over the top of the fuel seated in the reactor pressure vessel.

APPLICABILITY: During movement of CEAs within the reactor pressure vessel, when the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of CEAs within the pressure vessel.

## SURVEILLANCE REQUIREMENTS

4.9.10.2 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of CEAs.

## 3/4.9.11 WATER LEVEL - STORAGE POOL

### LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

## 3/4.9.12 FUEL BUILDING ESSENTIAL VENTILATION SYSTEM

### LIMITING CONDITION FOR OPERATION

3.9.12* Two independent fuel building essential ventilation systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one fuel building essential ventilation system inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE fuel building essential ventilation system is capable of being powered from an OPERABLE emergency power source. Restore the inoperable fuel building essential ventilation system to OPERABLE status within 7 days or suspend all operations involving movement of fuel within the storage pool or operation of the fuel handling machine over the storage pool.

b. With no fuel building essential ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one fuel building essential ventilation system is restored to OPERABLE status.

c. The provisions of Specification 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel building essential ventilation systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

*CAUTION - Reference Specification 3.7.8 page 3/4 7-19

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 6000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978*, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978*.

3. Verifying a system flow rate of 6000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978*, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978*.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters, pre-filters, heaters, and charcoal adsorber banks is less than 8.4 inches Water Gauge while operating the system at a flow rate of 6000 cfm ± 10%.

2. Verifying that on a high radiation test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the fuel building at a measurable negative pressure relative to the outside atmosphere during system operation.

*ANSI N509-1980 is applicable for this specification.

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.0% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.0% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.

## 3/4.10 SPECIAL TEST EXCEPTIONS

### 3/4.10.1 SHUTDOWN MARGIN

### LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s), or the reactor is subcritical by at least the reactivity equivalent of the highest CEA worth.

APPLICABILITY: MODES 2, 3* and 4*#.

ACTION:

a. With any full-length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 26 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 26 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

### SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length and part-length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

4.10.1.3 When in MODE 3 or MODE 4, the reactor shall be determined to be subcritical by at least the reactivity equivalent of the highest estimated CEA worth or the reactivity equivalent of the highest estimated CEA worth is available for trip insertion from OPERABLE CEAs at least once per 2 hours by consideration of at least the following factors:

a. Reactor Coolant System boron concentration,

b. CEA position,

c. Reactor Coolant System average temperature,

d. Fuel burnup based on gross thermal energy generation,

e. Xenon concentration, and

f. Samarium concentration.

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*Operation in MODE 3 and MODE 4 shall be limited to 6 consecutive hours.

#Limited to low power PHYSICS TESTING at the 320°F plateau.

## 3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

### LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient, group height, insertion, and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

### SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.2 and 3.3.3.2 during PHYSICS TESTS above 20% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended.

SPECIAL TEST EXCEPTIONS

## 3/4.10.3 REACTOR COOLANT LOOPS

### LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification 3.4.1.1 and noted requirements of Tables 2.2-1 and 3.3-1 may be suspended during the performance of startup PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at less than or equal to 20% of RATED THERMAL POWER.

c. Both reactor coolant loops and at least one reactor coolant pump in each loop are in operation.

APPLICABILITY: During startup PHYSICS TESTS.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER or with less than the above required reactor coolant loops in operation and circulating reactor coolant, immediately trip the reactor.

### SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during startup PHYSICS TESTS.

4.10.3.2 Each logarithmic and variable overpower level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup PHYSICS TESTS.

4.10.3.3 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

## 3/4.10.4 CEA POSITION, REGULATING CEA INSERTION LIMITS AND REACTOR COOLANT COLD LEG TEMPERATURE

### LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1, 3.1.3.6 and 3.2.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient, and power coefficient provided the limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.6 and 3.2.6 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

### SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1, 3.1.3.6 and/or 3.2.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 3.3.3.2 during PHYSICS TESTS above 20% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1, 3.1.3.6 and/or 3.2.6 are suspended.

SPECIAL TEST EXCEPTIONS

## 3/4.10.5 MINIMUM TEMPERATURE AND PRESSURE FOR CRITICALITY

### LIMITING CONDITION FOR OPERATION

3.10.5 The minimum temperature and pressure for criticality limits of Specifications 3.1.1.4 and 3.2.8 may be suspended during low temperature PHYSICS TESTS to a minimum temperature of 300°F and a minimum pressure of 500 psia provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER.

b. The reactor trip setpoints on the OPERABLE Variable Overpower trip channels are set at ≤ 20% of RATED THERMAL POWER, and

c. The Reactor Coolant System temperature and pressure relationship is maintained within the acceptable region of operation required by Specification 3.4.8 except that the core critical line shown on Figure 3.4-2 does not apply.

APPLICABILITY: MODE 2*.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With the Reactor Coolant System temperature and pressure relationship within the region of unacceptable operation on Figure 3.4-2, immediately open the reactor trip breakers and restore the temperature-pressure relationship to within its limit within 30 minutes; perform the engineering evaluation required by Specification 3.4.8.1 prior to the next reactor criticality.

### SURVEILLANCE REQUIREMENTS

4.10.5.1 The Reactor Coolant System temperature and pressure relationship shall be verified to be within the acceptable region for operation of Figure 3.4-2 at least once per hour.

4.10.5.2 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour.

4.10.5.3 The Reactor Coolant System temperature shall be verified to be greater than or equal to 300°F at least once per hour.

4.10.5.4 Each Logarithmic Power Level and Variable Overpower channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

*First core only, prior to first exceeding 5% RATED THERMAL POWER.

## 3/4.10.6 SAFETY INJECTION TANKS

### LIMITING CONDITION FOR OPERATION

3.10.6 The safety injection tank isolation valve requirement of Specification 3.5.1a. may be suspended during partial stroke testing of the low pressure safety injection check valves (SI-114, SI-124, SI-134, SI-144) provided:

a. That power to the isolation valve is restored and the SIAS signal is not overridden.

b. Only one isolation valve at a time is closed during the testing for no longer than 1 hour.

c. That the valve is key locked opened with power removed before the next isolation valve is closed.

APPLICABILITY:

While partial stroke testing of the low pressure injection check valves during normal plant operation.

ACTION:

If the requirement of Specification 3.5.1a. was suspended to perform the Specification 3.10.6 partial stroke test and if any of the Specification 3.10.6 requirements are not met during the Specification 3.10.6 partial stroke testing, the Limiting Condition for Operation shall revert to Specification 3.5.1 and the 3.5.1 ACTION shall be applicable.

### SURVEILLANCE REQUIREMENTS

4.10.6.1 A valve alignment shall be performed within 4 hours following completion of testing to verify that all valves operated during this testing are restored to their normal positions and that power is removed to the SIT isolation valves.

## 3/4.10.7 SPENT FUEL POOL LEVEL

### LIMITING CONDITION FOR OPERATION

3.10.7 The borated water source of Specifications 3.1.2.5a. and 3.1.2.6a. may be suspended during initial fuel load and startup provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at less than or equal to 20% of RATED THERMAL POWER.

APPLICABILITY: MODES 2, 3, 4, 5, and 6.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately trip the reactor.

### SURVEILLANCE REQUIREMENTS

4.10.7.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during startup and PHYSICS TESTS.

4.10.7.2 Each logarithmic and variable overpower level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

## 3/4.10.8 SAFETY INJECTION TANK PRESSURE

### LIMITING CONDITION FOR OPERATION

3.10.8 The safety injection tank (SIT) pressure of Specification 3.5.1d. may be suspended for low temperature PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER;

b. The SITs have been filled per Specification 3.5.1b. and pressurized to 175 to 225 psig below the RCS pressure or not to go below 254 psig;

c. All valves in the injection lines from the SITs to the RCS are open and the SITs are capable of injecting into the RCS if there is a decrease in RCS pressure.

APPLICABILITY: MODES 2 and 3.

ACTION:

If all the SITs do not meet the level and pressure requirements of Specification 3.10.8, restore all the SITs to meet these requirements or be in HOT STANDBY within 6 hours and be in HOT SHUTDOWN within the following 6 hours.

### SURVEILLANCE REQUIREMENTS

4.10.8.1 The THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER at least once per hour during low pressure PHYSICS TESTS.

4.10.8.2 Every 8 hours verify:

a. All the SITs levels meet the requirements of Specification 3.5.1b.

b. All the SITs pressures meet the requirements of Specification 3.10.8.

c. The valve alignment from the SITs to the RCS has not changed.

## 3/4.11 RADIOACTIVE EFFLUENTS

## 3/4.11.1 SECONDARY SYSTEM LIQUID WASTE DISCHARGES TO ONSITE EVAPORATION PONDS

### CONCENTRATION

### LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material discharged from secondary system liquid waste to the onsite evaporation ponds shall be limited to the lower limit of detectability (LLD) defined as $5 \times 10^{-7}$ µCi/ml for the principal gamma emitters or $1 \times 10^{-6}$ µCi/ml for I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

When any secondary system liquid waste discharge pathway concentration determined in accordance with the surveillance requirements given below exceeds the specified LLD, divert that discharge pathway to the liquid radwaste system without delay.

### SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes collected in the chemical waste neutralizer tank shall be sampled and analyzed prior to their batchwise discharge to the onsite evaporation pond in accordance with the sampling and analysis program specified in Table 4.11-1.

4.11.1.1.2 With the concentration of radioactive material in the chemical waste neutralizer tank exceeding the specified LLD, sample and analyze other secondary system discharge pathways in accordance with the sampling and analysis program specified in Table 4.11-1.

## TABLE 4.11-1

## RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

| SECONDARY SYSTEM LIQUID RELEASE PATHWAY | SAMPLING FREQUENCY | MINIMUM ANALYSIS FREQUENCY | TYPE OF ACTIVITY ANALYSIS | LOWER LIMIT OF DETECTION (LLD)[a] (μCi/mL) |
|---|---|---|---|---|
| A. Batch discharges[b] | | | | |
| 1. Chemical Waste Neutralizer Tank | P Each Batch | P Each Batch | Principal Gamma Emitters[c] | $5\times10^{-7}$ |
| | | | I-131 | $1\times10^{-6}$ |
| 2. Steam Generator Blowdown Low TDS Sump* | P Each Batch | P Each Batch | Principal Gamma Emitters[c] | $5\times10^{-7}$ |
| | | | I-131 | $1\times10^{-6}$ |
| 3. Condensate Polishing Low TDS Sump* | P Each Batch | P Each Batch | Principal Gamma Emitters[c] | $5\times10^{-7}$ |
| | | | I-131 | $1\times10^{-6}$ |
| B. Continuous Releases[d] | | | | |
| 1. Turbine Building Sump* | D Grab Sample | D Grab Sample | Principal Gamma Emitters[c] | $5\times10^{-7}$ |
| | | | I-131 | $1\times10^{-6}$ |
| 2. Condenser Area Sumps* | D Grab Sample | D Grab Sample | Principal Gamma Emitters[c] | $5\times10^{-7}$ |
| | | | I-131 | $1\times10^{-6}$ |

*Sampling and analysis for pathways 2 and 3 under batch discharges and 1 and 2 under continuous releases are required only when concentration for chemical waste neutralizer tank pathway exceeds the LLD.

TABLE 4.11-1 (Continued)

TABLE NOTATION

[a]The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

$s_b$ is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

$2.22 \times 10^6$ is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

$\lambda$ is the radioactive decay constant for the particular radionuclide, and

$\Delta t$ for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

Typical values of E, V, Y, and $\Delta t$ should be used in the calculation.

It should be recognized that the LLD is defined as an _a priori_ (before the fact) limit representing the capability of a measurement system and not as an _a posteriori_ (after the fact) limit for a particular measurement.

[b]A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

## TABLE 4.11-1 (Continued)

## TABLE NOTATION

[c]The principal gamma emitters for which the LLD specification applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137 and Ce-141. Ce-144 shall also be measured, but with an LLD of $5 \times 10^{-6}$. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

[d]A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

RADIOACTIVE EFFLUENTS

DOSE

## LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each reactor unit, to areas at and beyond the SITE BOUNDARY (See Figure 5.1-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

## SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.3 The quantity of radioactive material contained in each outside temporary tank and the reactor makeup water tank shall be limited to less than or equal to 60 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any outside temporary tank or the reactor makeup water tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3 The quantity of radioactive material contained in each outside temporary tank and the reactor makeup water tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site (see Figures 5.1-1 and 5.1-3) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

b. For I-131 and I-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately decrease the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM.

4.11.2.1.2 The dose rate due to I-131, I-133, Tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

## TABLE 4.11-2

## RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

| GASEOUS RELEASE TYPE | SAMPLING FREQUENCY | MINIMUM ANALYSIS FREQUENCY | TYPE OF ACTIVITY ANALYSIS | LOWER LIMIT OF DETECTION (LLD) (μCi/ml)[a] |
|---|---|---|---|---|
| A. Waste Gas Storage Tank | P Each Tank Grab Sample | P Each Tank | Principal Gamma Emitters[g] | $1\times10^{-4}$ |
| B. Containment Purge | P Each Purge[b,c] Grab Sample | P Each Purge[b,c] | Principal Gamma Emitters[g] | $1\times10^{-4}$ |
| | | | H-3 | $1\times10^{-6}$ |
| C. 1. Condenser Vacuum Pump Exhaust<br>2. Plant Vent<br>3. Fuel Bldg. Exhaust | M[b,e] Grab Sample | M[b] | Principal Gamma Emitters[g] | $1\times10^{-4}$ |
| | | | H-3 | $1\times10^{-6}$ |
| | Continuous[f] | 4/M[d] Charcoal Sample | I-131 | $1\times10^{-12}$ |
| | | | I-133 | $1\times10^{-10}$ |
| | Continuous[f] | 4/M[d] Particulate Sample | Principal Gamma Emitters[g] (I-131, Others) | $1\times10^{-11}$ |
| | Continuous[f] | M Composite Particulate Sample | Gross Alpha | $1\times10^{-11}$ |
| | Continuous[f] | Q Composite Particulate Sample | Sr-89, Sr-90 | $1\times10^{-11}$ |
| D. All Radwaste Types as listed in A., B., and C. above. | Continuous[f] | Noble Gas Monitor | Noble Gases Gross Beta or Gamma | $1\times10^{-6}$ |

TABLE 4.11-2 (Continued)

TABLE NOTATION

[a]The LLD is the smallest concentration of radioactive material in a sample that will yield a net count above background that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as µCi per unit mass or volume). Current literature defines the LLD as the detection capability for the instrumentation only and the MDC minimum detectable concentration, as the detection capability for a given instrument procedure and type of sample.

$s_b$ is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22 is the number of transformations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

$\lambda$ is the radioactive decay constant for the particular radionuclide, and

$\Delta t$ is the elapsed time between the midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of $s_b$ used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry the background should include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y, and $\Delta t$ should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement*.

*For a more complete discussion of the LLD, and other detection limits, see the following:
(1) HASL Procedures Manual, HASL-300 (revised annually).
(2) Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968).
(3) Hartwell, J. K., "Detection Limits for Radioisotopic Counting Techniques," Atlantic Richfield Hanford Company Report ARH-2537 (June 22, 1972).

## TABLE NOTATION

[b]Analyses shall also be performed following SHUTDOWN, STARTUP, or a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period if 1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased more than a factor of 3; and 2) the noble gas activity monitor on the plant vent shows that effluent activity has increased by more than a factor of 3. If the associated noble gas vent monitor is inoperable, samples must be obtained as soon as possible. Analyses shall be performed within a four-hour period. This requirement does not apply to the Fuel Building Exhaust.

[c]Sampling and analyses shall also be performed at least once per 31 days when purging time exceeds 30 days continuous.

[d]Samples shall be changed at least 4 times a month and analyses shall be completed within 48 hours after changing (or after removal from sampler). When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10.

[e]Tritium grab samples shall be taken at least monthly from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

[f]The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

[g]The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measureable and identifiable, together with the above nuclides, shall also be identified and reported in the Semiannual Radioactive Effluent Release Report.

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

## LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figures 5.1-1 and 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation and,

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

## SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

## DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

### LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figures 5.1-1 and 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT

## LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases, from each reactor unit, from the site (see Figures 5.1-1 and 5.1-3), when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figures 5.1-1 and 5.1-3) when averaged over 31 days would exceed 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

ACTION:

a. With radioactive gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

    1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

    2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

    3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

## SURVEILLANCE REQUIREMENTS

4.11.2.4 Doses due to gaseous releases from the site shall be projected at least once per 31 days, in accordance with the methodology and parameters in the ODCM.

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limit within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than 4% by volume within 6 hours.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of hydrogen or oxygen in the waste gas holdup system shall be determined to be within the above limits by monitoring the waste gases in the waste gas holdup system in accordance with Specification 3.3.3.8.

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 170,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 7 days when radioactive materials are being added to the tank and the quantity of radioactivity contained in the tank is less than or equal to one-half of the above limit; otherwise, determine the quantity of radioactive material contained in the tank at least once per 24 hours during addition.

## 3/4.11.3 SOLID RADIOACTIVE WASTE

### LIMITING CONDITION FOR OPERATION

3.11.3 The solid radwaste system shall be OPERABLE and used, as applicable in accordance with a PROCESS CONTROL PROGRAM, for the SOLIDIFICATION and packaging of radioactive wastes to ensure meeting the requirements of 10 CFR Part 20 and of 10 CFR Part 71 prior to shipment of radioactive wastes from the site.

APPLICABILITY: At all times.

ACTION:

a. With the packaging requirements of 10 CFR Part 20 and/or 10 CFR Part 71 not satisfied, suspend shipments of defectively packaged solid radioactive wastes from the site.

b. With the solid radwaste system inoperable for more than 31 days, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability.

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status,

   3. A description of the alternative used for SOLIDIFICATION and packaging of radioactive wastes, and

   4. Summary description of action(s) taken to prevent a recurrence.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.11.3.1 The solid radwaste system shall be demonstrated OPERABLE at least once per 92 days by:

a. Operating the solid radwaste system at least once in the previous 92 days in accordance with the PROCESS CONTROL PROGRAM, or

b. Verification of the existence of a valid contract for SOLIDIFICATION to be performed by a contractor in accordance with a PROCESS CONTROL PROGRAM.

SURVEILLANCE REQUIREMENTS (Continued)

4.11.3.2 THE PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., spent resins, evaporator bottoms, and boric acid solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid and gaseous effluents exceeding twice the limits of Specifications 3.11.1.2a., 3.11.1.2b., 3.11.2.2a., 3.11.2.2b., 3.11.2.3a., or 3.11.2.3b., calculations should be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in Specification 3.11.4a.

## 3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

### 3/4.12.1 MONITORING PROGRAM

### LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.7, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to A MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific

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*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.9.1.8, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1, and the detection capabilities required by Table 4.12-1.

## TABLE 3.12-1

## RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

| EXPOSURE PATHWAY AND/OR SAMPLE | NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[a] | SAMPLING AND COLLECTION FREQUENCY[a] | TYPE AND FREQUENCY OF ANALYSIS[e] |
|---|---|---|---|
| Airborne | | | |
| Radioiodine and particulates | Samples from 5 locations: 3 samples at or near the SITE BOUNDARIES (#14A, 15, 21) in different sectors of the highest calculated annual average ground level D/Q.* | Continuous sampling collected weekly,[c] or more frequently if required by dust loading | Gross beta weekly;[d] I-131 weekly; gamma isotopic analysis of composite (by location) quarterly |
| | 1 sample (#40) from areas of special interest, which is from the vicinity of a community having the highest calculated annual average D/Q. | | |
| | 1 sample (#6) from a control location 15-30 km (10-20 mi) distant and in the least prevalent wind direction.[f] | | |
| Direct radiation[b] | 40 stations (#6-45) with two or more dosimeters for measuring dose rate continuously, placed as follows: an inner ring of stations at the site boundary and an outer ring in the 4-to-5 mi range from the site with a station in each sector of each ring, except the WNW sector, which is inaccessible (16 sectors x 2 rings minus 1 = 31 stations). 7 additional stations are in local schools and population centers; 2 other stations are used as controls. | Quarterly | Gamma dose quarterly |

*D/Q refers to average annual relative ground deposition rate.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

| EXPOSURE PATHWAY AND/OR SAMPLE | NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[a] | SAMPLING AND COLLECTION FREQUENCY[a] | TYPE AND FREQUENCY OF ANALYSIS[e] |
|---|---|---|---|
| Waterborne | | | |
| Surface | Water storage reservoir (#60) evaporation pond (#59) | Monthly composite of weekly grab sample | Gamma isotopic analysis monthly; tritium quarterly |
| Ground | 2 onsite wells[g] (#57, 58) | Quarterly grab sample | Tritium and gamma isotopic analysis quarterly |
| Drinking (well) | 3 wells from surrounding residences (#46, 48, 49) that would be affected by its discharge | Composite sample of weekly grab samples over 2-week period when I-131 analysis is performed, monthly composite of weekly grab samples otherwise | I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.[h] Composite for gross beta and gamma isotopic analyses monthly. Composite for tritium analysis quarterly. |
| Ingestion | | | |
| Milk | Samples from milking animals in 3 locations within 5 km distance having the highest dose potential. If there are none, 1 sample from milking animals in each of 3 areas (#50, 51, 53) between 5 and 8 km distant where doses are calculated to be greater than 1 mrem per year.[h]<br>One sample from milking animals at a control location (#56), 15 to 30 km distant and in the least prevalent wind direction. | Semimonthly for animals on pasture; otherwise, monthly | Gamma isotopic and I-131 analysis semi-monthly when animals are on pasture or monthly at other times |

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

| EXPOSURE PATHWAY AND/OR SAMPLE | NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[a] | SAMPLING AND COLLECTION FREQUENCY[a] | TYPE AND FREQUENCY OF ANALYSIS[e] |
|---|---|---|---|
| Food products* | Samples (#47, 52) of 3 different kinds of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground-level D/Q if milk sampling is not performed | Monthly during growing season | Gamma isotopic and I-131 analysis. |
| | 1 sample (#62) of each of the similar broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction if milk sampling is not performed | Monthly during growing season | Gamma isotopic and I-131 analysis. |

*When broad leaf vegetation samples are not available, reports from 4 existing supplemental airborne radioiodine sample locations will be substituted.

## TABLE NOTATIONS

[a]The number, media, frequency, and location of sampling may vary from site to site. It is recognized that, at times, it may not be possible or practical to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and submitted for acceptance. Actual locations (distance and direction) from the site shall be provided in Table 7-1 and Figure 7-1 in the ODCM. Refer to Regulatory Guide 4.1, "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants."

[b]Regulatory Guide 4.13 provides guidance for thermoluminescence dosimetry (TLD) systems used for environmental monitoring. One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter may be considered to be one phosphor, and two or more phosphors in a packet may be considered as two or more dosimeters. Film badges should not be used for measuring direct radiation.

[c]Canisters for the collection of radioiodine in air are subject to channeling. These devices should be carefully checked before operation in the field or several should be mounted in series to prevent loss of iodine.

[d]Particulate sample filters shall be analyzed for gross beta 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air or water is greater than 10 times the yearly mean of control samples for any medium, gamma isotopic analysis should be performed on the individual samples.

[e]Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

[f]The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites that provide valid background data may be substituted.

[g]Groundwater samples should be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

[h]The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

## TABLE 3.12-2

## REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

| ANALYSIS | WATER (pCi/ℓ) | AIRBORNE PARTICULATE OR GASES (pCi/$m^3$) | MILK (pCi/ℓ) | FOOD PRODUCTS (pCi/kg, wet) |
|---|---|---|---|---|
| H-3 | 20,000* | | | |
| Mn-54 | 1,000 | | | |
| Fe-59 | 400 | | | |
| Co-58 | 1,000 | | | |
| Co-60 | 300 | | | |
| Zn-65 | 300 | | | |
| Zr-Nb-95 | 400 | | | |
| I-131 | 2** | 0.9 | 3 | 100 |
| Cs-134 | 30 | 10 | 60 | 1,000 |
| Cs-137 | 50 | 20 | 70 | 2,000 |
| Ba-La-140 | 200 | | 300 | |

*For drinking water samples. This is 40 CFR Part 141 value. If no drinking pathway exists, a value of 30,000 pCi/ℓ may be used.

**If no drinking pathway exists, a reporting level of 20 pCi/ℓ may be used.

## TABLE 4.12-1

## DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a]

### LOWER LIMIT OF DETECTION (LLD)[b]

| ANALYSIS | WATER (pCi/ℓ) | AIRBORNE PARTICULATE OR GAS (pCi/$m^3$) | MILK (pCi/ℓ) | FOOD PRODUCTS (pCi/kg,wet) |
|---|---|---|---|---|
| Gross beta | 4 | 0.01 | | |
| H-3 | 2000* | | | |
| Mn-54 | 15 | | | |
| Fe-59 | 30 | | | |
| Co-58,-60 | 15 | | | |
| Zn-65 | 30 | | | |
| Zr-95 | 30 | | | |
| Nb-95 | 15 | | | |
| I-131 | 1** | 0.07 | 1 | 60 |
| Cs-134 | 15 | 0.05 | 15 | 60 |
| Cs-137 | 18 | 0.06 | 18 | 80 |
| Ba-140 | 60 | | 60 | |
| La-140 | 15 | | 15 | |

Note: This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measureable and identifiable, together with the above nuclides, shall also be identified and reported.

*If no drinking water pathway exists, a value of 3000 pCi/ℓ may be used.

**If no drinking water pathway exists, a value of 15 pCi/ℓ may be used.

## TABLE 4.12-1 (Continued)

## TABLE NOTATION

[a]Guidance for detection capabilities for thermoluminescent dosimeters used for environmental measurements is given in Regulatory Guide 4.13.

[b]Table 4.12-1 indicates acceptable detection capabilities for radioactive materials in environmental samples. These detection capabilities are tabulated in terms of the lower limits of detection (LLDs). The LLD is defined, for purposes of this guide, as the smallest concentration of radioactive material in a sample that will yield a net count (above system background) that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as picocuries per unit mass or volume).

$s_b$ is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute)

E is the counting efficiency (as counts per disintegration)

V is the sample size (in units of mass or volume)

2.22 is the number of disintegrations per minute per picocurie

Y is the fractional radiochemical yield (when applicable)

$\lambda$ is the radioactive decay constant for the particular radionuclide

$\Delta t$ for environmental samples is the elapsed time between sample collection (or end of the sample collection period) and time of counting

TABLE 4.12-1 (Continued)

TABLE NOTATION

In calculating the LLD for a radionuclide determined by gamma-ray spectrometry the background should include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y, and $\Delta t$ should be used in the calculation.

It should be recognized that the LLD is defined as an <u>a priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a posteriori</u> (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report.

RADIOLOGICAL ENVIRONMENTAL MONITORING

## 3/4.12.2 LAND USE CENSUS

### LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 $m^2$ (500 $ft^2$) producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.8.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after (October 31) of the year in which this land use census was conducted. Pursuant to Specification 6.9.1.8, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the SITE BOUNDARY in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1 shall be followed, including analysis of control samples.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

## LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission that correspond to samples required by Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

## SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program and in accordance with the methodology and parameters in the ODCM shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

BASES

FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

## NOTE

The BASES contained in the succeeding pages summarize the reasons for the specifications of Sections 3.0 and 4.0 but in accordance with 10 CFR 50.36 are not a part of these Technical Specifications.

## 3/4.0 APPLICABILITY

## BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the measures to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For example, Specification 3.6.2.1 requires two containment spray systems to be OPERABLE and provides explicit ACTION requirements if one spray system is inoperable. Under the terms of Specification 3.0.3, if both of the required containment spray systems are inoperable, within 1 hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN in the following 30 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment, or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to ensure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this specification have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

4.0.1 This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual surveillance requirements. Surveillance requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over three test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under these criteria, equipment, systems, or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems, or components OPERABLE, when such items are found or known to be inoperable although still meeting the surveillance requirements.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals thoughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to 1 week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

BASES

## 3/4.1.1 BORATION CONTROL

### 3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits assuming the insertion of the regulating CEAs are within the limits of Specification 3.1.3.6, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS $T_{cold}$. The most restrictive condition occurs at EOL, with $T_{cold}$ at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 6.0% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with the criteria used to establish the power dependent CEA insertion limits and with the assumptions used in the FSAR Safety Analysis.

With $T_{cold}$ less than or equal to 210°F, the reactivity transients resulting from uncontrolled RCS cooldown are minimal and a 4% $\Delta k/k$ SHUTDOWN MARGIN requirement is set to ensure that reactivity transients resulting from an inadvertent single CEA withdrawal event are minimal.

### 3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

## 3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 552°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, and (3) to ensure consistency with the FSAR safety analysis.

## 3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) an emergency power supply from OPERABLE diesel generators, and (5) the volume control tank (VCT) outlet valve CH-UV-501, capable of isolating the VCT from the charging pump suction line. The nominal capacity of each charging pump is 44 gpm at its discharge. Up to 16 gpm of this may be diverted to the volume control tank via the RCP control bleedoff. Instrument inaccuracies and pump performance uncertainties are limited to 2 gpm yielding the 26 gpm value.

With the RCS temperature above 210°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 4% delta k/k after xenon decay and cooldown to 210°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 23,800 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

With the RCS temperature below 210°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable. The restrictions of one and only one operable charging pump whenever reactor coolant level is below the bottom of the pressurizer is based on the assumptions used in the analysis of the boron dilution event.

The boron capability required below 210°F is based upon providing a 4% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 210°F to 120°F. This condition requires 9,700 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

BORATION SYSTEMS (Continued)

The values of water volumes, temperatures, and boron concentration in the refueling water tank are provided to ensure that the assumptions used in the initial conditions of the LOCA Safety Analysis remain valid.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

With the RCS temperature below 210°F while in MODES 5 and 6, a source of borated water is required to be available for reactivity control and makeup for losses due to contraction and evaporation. The requirement of 33,500 gallons of 4000 ppm borated water in either the refueling water tank or spent fuel pool ensures that this source is available.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between 7.0 and 8.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.1.2.7 BORON DILUTION ALARMS

The startup channel high neutron flux alarms alert the operator to an inadvertent boron dilution. Both channels must be operating to assure detection of a boron dilution event by the high neutron flux alarms. If one or both of the alarms are inoperable at any time, the bases for ACTION statements are as follows:

a. One startup channel high neutron flux alarm not operating:

   With only one startup channel high neutron flux alarm OPERABLE while in MODE 3, 4, 5, or 6, a single failure to the alarm could prevent detection of boron dilution. By periodic monitoring of the RCS boron concentration by either boronometer or RCS sampling, a decrease in the boron concentration during an inadvertent boron dilution event will be observed. This provides alternate methods of detection of boron dilution with sufficient time for termination of the event before complete loss of SHUTDOWN MARGIN and return to criticality.

b. Both startup channel high neutron flux alarms not operating:

   When both startup channel high neutron flux alarms are inoperable, there is no means of alarming on high neutron flux when subcritical. Therefore, either simultaneous use of the boronmeter and RCS sampling or independent collection and analysis of two RCS samples to monitor the RCS boron concentration provides alternate indications of inadvertent boron dilution. This will allow detection with sufficient time for termination of boron dilution before complete loss of shutdown margin and return to criticality.

## 3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA, to two or more inoperable CEAs, and to a large misalignment (greater than or equal to 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 19 inches) of the CEAs, there is (1) a small effect on the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, (2) a small effect on the available SHUTDOWN MARGIN, and (3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a 1-hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The 1-hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs, and (3) minimize the effects of xenon redistribution.

The CPCs provide protection to the core in the event of a large misalignment (greater than or equal to 19 inches) of a CEA by applying appropriate penalty factors to the calculation to account for the misaligned CEA. However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on (1) the available SHUTDOWN MARGIN, (2) the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, and (3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in (1) local burnup, (2) peaking factors, and (3) available SHUTDOWN MARGIN which are more adverse than the conditions assumed to exist in the safety analyses and LCO

MOVABLE CONTROL ASSEMBLIES (Continued)

and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. Measurement with $T_{cold}$ greater than or equal to 552°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Several design steps were employed to accommodate the possible CEA guide tube wear which could arise from CEA vibrations when fully withdrawn. Specifically, a programmed insertion schedule will be used to cycle the CEAs between the full out position ("FULL OUT" LIMIT) and 3.0 inches inserted over the fuel cycle. This cycling will distribute the possible guide tube wear over a larger area, thus minimizing any effects. To accommodate this programmed insertion schedule, the fully withdrawn position was redefined, in some cases, to be 144.75 inches or greater.

The establishment of LSSS and LCOs requires that the expected long- and short-term behavior of the radial peaking factors be determined. The long-term behavior relates to the variation of the steady-state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed and the expected power level variation throughout the cycle. The short-term behavior relates to transient perturbations to the steady-state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions

MOVABLE CONTROL ASSEMBLIES (Continued)

and load maneuvering. Analyses are performed based on the expected mode of operation of the NSSS (base load maneuvering, etc.) and from these analyses CEA insertions are determined and a consistent set of radial peaking factors defined. The Long Term Steady State and Short Term Insertion Limits are determined based upon the assumed mode of operation used in the analyses and provide a means of preserving the assumptions on CEA insertions used. The limits specified serve to limit the behavior of the radial peaking factors within the bounds determined from analysis. The actions specified serve to limit the extent of radial xenon redistribution effects to those accommodated in the analyses. The Long and Short Term Insertion Limits of Specification 3.1.3.6 are specified for the plant which has been designed for primarily base loaded operation but which has the ability to accommodate a limited amount of load maneuvering.

The Transient Insertion Limits of Specification 3.1.3.6 and the Shutdown CEA Insertion Limits of Specification 3.1.3.5 ensure that (1) the minimum SHUTDOWN MARGIN is maintained, and (2) the potential effects of a CEA ejection accident are limited to acceptable levels. Long-term operation at the Transient Insertion Limits is not permitted since such operation could have effects on the core power distribution which could invalidate assumptions used to determine the behavior of the radial peaking factors.

The PVNGS CPC and COLSS systems are responsible for the safety and monitoring functions, respectively, of the reactor core. COLSS monitors the DNB Power Operating Limit (POL) and various operating parameters to help the operator maintain plant operation within the limiting conditions for operation (LCO). Operating within the LCO guarantees that in the event of an Anticipated Operational Occurrence (AOO), the CPCs will provide a reactor trip in time to prevent unacceptable fuel damage.

The COLSS reserves the Required Overpower Margin (ROPM) to account for the Loss of Flow (LOF) transient which is the limiting AOO for the PVNGS plants. When the COLSS is Out of Service (COOS), the monitoring function is performed via the CPC calculation of DNBR in conjunction with a Technical Specification COOS Limit Line (Figure 3.2-2) which restricts the reactor power sufficiently to preserve the ROPM.

The reduction of the CEA deviation penalties in accordance with the CEAC (Control Element Assembly Calculator) sensitivity reduction program has been performed. This task involved setting many of the inward single CEA deviation penalty factors to 1.0. An inward CEA deviation event in effect would not be accompanied by the application of the CEA deviation penalty in either the CPC DNB and LHR (Linear Heat Rate) calculations for those CEAs with the reduced penalty factors. The protection for an inward CEA deviation event is thus accounted for separately.

MOVABLE CONTROL ASSEMBLIES (Continued)

If an inward CEA deviation event occurs, the current CPC algorithm applies two penalty factors to each of the DNB and LHR calculations. The first, a static penalty factor, is applied upon detection of the event. The second, a xenon redistribution penalty, is applied linearly as a function of time after the CEA drop. The expected margin degradation for the inward CEA deviation event for which the penalty factor has been reduced is accounted for in two ways. The ROPM reserved in COLSS is used to account for some of the margin degradation. If the combination of the static and xenon redistribution penalties exceeds the reserved ROPM, a power reduction in accordance with the curve in Figure 3.1-2B is required. In addition, the part length CEA maneuvering is restricted in accordance with Figure 3.1-2A to justify reduction of the PLR deviation penalty factors.

The technical specification permits plant operation if both CEACs are considered inoperable for safety purposes after this period.

# 3/4.2 POWER DISTRIBUTION LIMITS

## BASES

### 3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the Local Power Density channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate. Reactor operation at or below this calculated power level assures that the limits of 14.0 kW/ft are not exceeded.

The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the linear heat rate operating limit for normal steady-state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the linear heat rate includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the maximum linear heat rate calculated by COLSS is conservative with respect to the actual maximum linear heat rate existing in the core. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering heat flux uncertainty, axial densification, software algorithm modelling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the operating limit power level based on linear heat rate, margin to DNB, and total core power are also monitored by the CPCs (assuming minimum core power of 20% of RATED THERMAL POWER). The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty and penalty factors plus those associated with the CPC startup test acceptance criteria are also included in the CPCs.

### 3/4.2.2 PLANAR RADIAL PEAKING FACTORS

Limiting the values of the PLANAR RADIAL PEAKING FACTORS ($F_{xy}^{c}$) used in the COLSS and CPCs to values equal to or greater than the measured PLANAR RADIAL PEAKING FACTORS ($F_{xy}^{m}$) provides assurance that the limits calculated by COLSS and the CPCs remain valid. Data from the incore detectors are used for determining the measured PLANAR RADIAL PEAKING FACTORS. A minimum core power at 20% of RATED THERMAL POWER is assumed in determining the PLANAR RADIAL PEAKING FACTORS. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The periodic surveillance requirements for determining the measured PLANAR RADIAL PEAKING FACTORS provides assurance that the PLANAR RADIAL PEAKING FACTORS used in COLSS and the CPCs remain valid throughout the fuel cycle. Determining the measured PLANAR RADIAL PEAKING FACTORS after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

### 3/4.2.3 AZIMUTHAL POWER TILT - $T_q$

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. An AZIMUTHAL POWER TILT greater than 0.10 is not expected and if it should occur, operation is restricted to only those conditions required to identify the cause of the tilt. The tilt is normally calculated by COLSS. A minimum core power of 20% of RATED THERMAL POWER is assumed by the CPCs in its input to COLSS for calculation of AZIMUTHAL POWER TILT. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The surveillance requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady-state tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

The AZIMUTHAL POWER TILT is equal to $(P_{tilt}/P_{untilt})-1.0$ where:

AZIMUTHAL POWER TILT is measured by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at the location is of the form:

$$P_{tilt}/P_{untilt} = 1 + T_q \, g \cos(\theta - \theta_0)$$

where:

$T_q$ is the peak fractional tilt amplitude at the core periphery

$g$ is the radial normalizing factor

$\theta$ is the azimuthal core location

$\theta_0$ is the azimuthal core location of maximum tilt

AZIMUTHAL POWER TILT - $T_q$ (Continued)

$P_{tilt}/P_{untilt}$ is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

The AZIMUTHAL POWER TILT allowance used in the CPCs is defined as the value of CPC addressable constant TR-1.0.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-1 are not violated. The COLSS calculation of core power operating limit based on DNBR includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limits calculated by COLSS (based on the minimum DNBR Limit) is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering heat flux, state parameter measurement, software algorithm modelling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings.

The DNBR penalty factors listed in Specification 4.2.4.4 are penalties used to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. The penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

## 3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the safety analyses.

## 3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses.

## 3/4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of the core average AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses.

## 3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses.

## 3/4.3 INSTRUMENTATION

## BASES

### 3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the reactor protective and Engineered Safety Features Actuation Systems instrumentation and bypasses ensures that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses.

Response time testing of resistance temperature devices, which are a part of the reactor protective system, shall be performed by using in-situ loop current test techniques or another NRC approved method.

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications of power level, RCS flow rate, axial flux shape, radial peaking factors and CEA deviation penalties. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1 and 6.8.1) ensure that inadvertent misloading of addressable constants into the CPCs is unlikely.

The design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEACs become inoperable. If one CEAC is in test or inoperable, verification of CEA position is performed at least every 4 hours. If the second CEAC fails, the CPCs in conjunction with plant Technical Specifications will use DNBR and LPD penalty factors and increased DNBR and LPD margin to restrict reactor operation to a power level that will ensure safe operation of the plant. If the margins are not maintained, a reactor trip will occur.

The value of the DNBR in Specification 2.1 is conservatively compensated for measurement uncertainties. Therefore, the actual RCS total flow rate determined by the reactor coolant pump differential pressure instrumentation or by calorimetric calculations does not have to be conservatively compensated for measurement uncertainties.

An analysis was done to specify a minimum power level below which an additional power reduction is unnecessary even if there is a CEA misalignment with CEACs out of service.

REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The analysis determined a Power Operating Limit (POL) power and assumed a CEA misalignment occurred from this power level. The power penalty factor would accommodate changes in radial peaks and one hour xenon redistribution that would occur if there were a CEA misalignment with CEACs out of service. The quotient of the POL power and the CEA misalignment Power Penalty factor is the maximum power (50% power) at which DNBR SAFDL violation will occur even if there is a CEA misalignment from POL conditions. Below this power, extra thermal margin will be available to the plant. Thus, for CEA misalignment, power reduction below this limiting power is unnecessary.

The lowest core power for a POL was calculated to be 70% of rated power. This was based on the following worst COLSS fluid conditions.

| | | |
|---|---|---|
| High Temperature | : | 580°F |
| Low Pressure | : | 1785 psia |
| ASI | : | -.3 |
| Underflow fraction | : | 0.865 |
| Low Flow | : | 95% of full flow |
| High Radial Peak | : | 1.70 (Bank 5+4+PLR; PDIL = 40% Power) |

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. The response times in Table 3.3-2 are made up of the time to generate the trip signal at the detector (sensor response time) and the time for the signal to interrupt power to the CEA drive mechanism (signal or trip delay time). The response times are taken from the sequence-of-events Tables in Section 15 of CESSAR.

Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) in place, onsite, or offsite test measurements or (2) utilizing replacement sensors with certified response times.

## 3/4.3.3 MONITORING INSTRUMENTATION

### 3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: (1) the radiation levels are continually measured in the areas served by the

individual channels and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974 as identified in the PVNGS FSAR. The seismic instrumentation for the site is listed in Table 3.3-7.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February 1972. Wind speeds less than 0.6 MPH cannot be measured by the meteorological instrumentation.

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the remote shutdown system ensures that sufficient capability is available to permit safe shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

The parameters selected to be monitored ensure that (1) the condition of the reactor is known, (2) conditions in the RCS are known, (3) the steam generators are available for residual heat removal, (4) a source of water is available for makeup to the RCS, and (5) the charging system is available to makeup water to the RCS.

The OPERABILITY of the remote shutdown system insures that a fire will not preclude achieving safe shutdown. The remote shutdown system instrumentation, control and power circuits and disconnect switches necessary to eliminate

REMOTE SHUTDOWN SYSTEM (Continued)

effects of the fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shutdown the reactor. This capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR 50.

The alternate disconnect methods or power or control circuits ensure that sufficient capability is available to permit shutdown and maintenance of cold shutdown of the facility by relying on additional operator actions at local control stations rather than at the RSP.

3/4.3.3.6 POST-ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the post-accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

The containment high range area monitors (RU-148 & RU-149) and the main steamline radiation monitors (RU-139 A&B and RU-140 A&B) are in Table 3.3-6. The high range effluent monitors and samplers (RU-142, RU-144 and RU-146) are in Table 3.3-13. The containment hydrogen monitors are in Specification 3/4.6.4.1. The Post Accident Sampling System (RCS coolant) is in Table 3.3-6.

The Subcooled Margin Monitor (SMM), the Heat Junction Thermocouple (HJTC), and the Core Exit Thermocouples (CET) comprise the Inadequate Core Cooling (ICC) instrumentation required by Item II.F.2 NUREG-0737, the Post TMI-2 Action Plan. The function of the ICC instrumentation is to enhance the ability of the plant operator to diagnose the approach to existance of, and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37. These are not required by the accident analysis, nor to bring the plant to Cold Shutdown.

In the event more than four sensors in a Reactor Vessel Level channel are inoperable, repairs may only be possible during the next refueling outage. This is because the sensors are accessible only after the missile shield and reactor vessel head are removed. It is not feasible to repair a channel except during a refueling outage when the missile shield and reactor vessel head are removed to refuel the core. If both channels are inoperable, the channels shall be restored to OPERABLE status in the nearest refueling outage. If only one channel is inoperable, it is intented that this channel be restored to OPERABLE status in a refueling outage as soon as reasonably possible.

## 3/4.3.3.7 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

## 3/4.3.3.8 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the GASEOUS RADWASTE SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

There are two separate radioactive gaseous effluent monitoring systems: the low range effluent monitors for normal plant radioactive gaseous effluents and the high range effluent monitors for post-accident plant radioactive gaseous effluents. The low range monitors operate at all times until the concentration of radioactivity in the effluent becomes too high during post-accident conditions. The high range monitors only operate when the concentration of radioactivity in the effluent is above the setpoint in the low range monitors.

## 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.231 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either shutdown cooling or RCS) be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires that two shutdown cooling loops be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling loops be OPERABLE.

The operation of one reactor coolant pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 4000 gpm will circulate one equivalent Reactor Coolant System volume of 12,097 cubic feet in approximately 23 minutes. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump in MODES 4 and 5, with one or more RCS cold legs less than or equal to 255°F during cooldown or 295°F during heatup are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

## 3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve a minimum of 460,000 lb per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

SAFETY VALVES (Continued)

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the system pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., there is no direct reactor trip on the loss of turbine) and also assuming no operation of the steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the Reactor Coolant System during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an Engineered Safety Features Actuation test signal concurrent with a loss-of-offsite power the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class 1E heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

The auxiliary pressurizer spray is required to depressurize the RCS by cooling the pressurizer steam space to permit the plant to enter shutdown cooling. The auxiliary pressurizer spray is required during those periods when normal pressurizer spray is not available, such as during natural circulation and during the later stages of a normal RCS cooldown. The auxiliary pressurizer spray also distributes boron to the pressurizer when normal pressurizer spray is not available. Use of the auxiliary pressurizer spray is required during the recovery from a steam generator tube rupture and a small loss of coolant accident.

## 3/4.4.4 STEAM GENERATORS

The surveillance requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion.

Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 gpm per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 gpm per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior to the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

## 3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

### 3/4.4.5.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. Containment sump flow is provided by monitoring the rate of sump level increase prior to the sump being pumped down, and is alarmed at the equivalent of 1 gpm leakage into the sump. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

### 3/4.4.5.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value. A threshold value of less than 1 gpm is sufficiently low to ensure early detection of additional leakage.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowances for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The surveillance requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowable limit.

The total steam generator tube leakage limit of 1 gpm for both steam generators ensures that the dosage contribution from the tube leakage will be limited to less than Part 100 guidelines for infrequent and limiting fault events. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. Section 15.4.1 of the PVNGS SER dated November 11, 1981, stated that the primary-to-secondary leakage from the steam generators should be less than or equal to 0.3 gpm. This was based on the bounding accident analysis in Section 15 of CESSAR. The PVNGS meteorological parameters are sufficiently less than the parameters assumed in CESSAR to allow the Limiting Condition for Operation to be 1 gpm (instead of the 0.3 gpm) total primary-to-secondary leakage through all steam generators and 720 gallons per day through any one steam generator. The 0.5 gpm leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude may be indicative of an impending failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

## 3/4.4.6 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

## 3/4.4.7 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2-hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm and a concurrent loss-of-offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Palo Verde site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

SPECIFIC ACTIVITY (Continued)

Reducing $T_{cold}$ to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Chapters 3 and 5 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so as not to exceed the limit lines of Figure 3.4-2. This ensures that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses at the inner wall tend to alleviate the tensile stresses induced by the internal pressure.

At the outer wall of the vessel, these thermal stresses are additive to the pressure induced tensile stresses. The magnitude of the thermal stresses at either location is dependent on the rate of heatup. Consequently, each heatup rate of interest must be analyzed on an individual basis for both the inner and outer wall.

The heatup and cooldown limit curve (Figure 3.4-2) is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup or cooldown rates of up to 100°F per hour. The heatup and cooldown curve was prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figure 3.4-2.

The reactor vessel materials have been tested to determine their initial $RT_{NDT}$; the results of these test are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the $RT_{NDT}$. Therefore, an adjusted reference temperature, based

PRESSURE/TEMPERATURE LIMITS (Continued)

upon the fluence and residual element content, can be predicted using Figure B 3/4.4-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curve Figure 3.4-2 includes predicted adjustments for this shift in $RT_{NDT}$ at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in $RT_{NDT}$ of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and Appendix H of 10 CFR 50, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta $RT_{NDT}$ determined from the surveillance capsule is different from the calculated delta $RT_{NDT}$ for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50. The reactor vessel material irradiation surveillance specimens are removed and examined to determine changes in material properties. The results of these examinations shall be used to update Figure 3.4-2 based on the greater of the following:

(1) the actual shift in reference temperature for plate F-6411-2 and weld 101-142 as determined by impact testing, or

(2) the predicted shift in reference temperature for the limiting weld and plate as determined by RG 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

The maximum $RT_{NDT}$ for all Reactor Coolant System pressure-retaining materials has been determined to be 40°F. The Lowest Service Temperature limit is based upon this $RT_{NDT}$ since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be $RT_{NDT}$ + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia. However, based upon the 10 CFR Part 50 Appendix G analysis, the isothermal condition for the reactor vessel is more restrictive than the Lowest Service Temperature line. Therefore, only the isothermal line is shown on Figure 3.4-2.

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing these capsules are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

## TABLE B 3/4.4-1
## REACTOR VESSEL TOUGHNESS
## (FORGINGS)

| PIECE NO. | CODE NO. | MATERIAL | VESSEL LOCATION | DROP WEIGHT RESULTS (°F) | $RT_{NDT}$ (a)(b) (°F) | TEMPERATURE OF CHARPY V-NOTCH* @ 30 ft - lb | TEMPERATURE OF CHARPY V-NOTCH* @ 50 ft - lb | MINIMUM UPPER SHELF $C_V$ ENERGY ft-lb |
|---|---|---|---|---|---|---|---|---|
| 128-3201 | F-6409-01 | SA 508-CL3 | Inlet Nozzle | -50 | -50 | -22 | +5 | N.A. |
| 128-3201 | F-6409-02 | SA 508-CL3 | Inlet Nozzle | -60 | -60 | +4 | +29 | N.A. |
| 128-3201 | F-6409-03 | SA 508-CL3 | Inlet Nozzle | -30 | -30 | -10 | +17 | N.A. |
| 128-3201 | F-6409-04 | SA 508-CL3 | Inlet Nozzle | -40 | -40 | -16 | +12 | N.A. |
| 131-3302 | F-6405-01 | SA 508-CL1 | Outlet Nozzle Safe End | -30 | +10 | +34 | +76 | N.A. |
| 131-3302 | F-6405-02 | SA 508-CL1 | Outlet Nozzle Safe End | -30 | +10 | +34 | +76 | N.A. |
| 128-3301 | F-6404-01 | SA 508-CL3 | Outlet Nozzle | -20 | +10 | +40 | +75 | N.A. |
| 128-3301 | F-6404-02 | SA 508-CL3 | Outlet Nozzle | -20 | +10 | +40 | +75 | N.A. |
| 131-3301 | F-6406-01 | SA 508-CL1 | Inlet Nozzle Safe End | -20 | +20 | -4 | +41 | N.A. |
| 131-3301 | F-6406-02 | SA 508-CL1 | Inlet Nozzle Safe End | -20 | +20 | -4 | +41 | N.A. |
| 131-3301 | F-6406-03 | SA 508-CL1 | Inlet Nozzle Safe End | -20 | +20 | -4 | +41 | N.A. |
| 131-3301 | F-6406-04 | SA 508-CL1 | Inlet Nozzle Safe End | -20 | +20 | -4 | +41 | N.A. |
| 126-101 | F-6402-01 | SA 508-CL2 | Vessel Flange | -40 | -40 | -43 | -22 | N.A. |
| 106-101 | F-6401-01 | SA 508-CL2 | Closure Head Flange | -10 | -10 | -42 | 0 | N.A. |

N.A. = Not Applicable (no minimum upper shelf requirement).
* = Lower bound curve values of transverse specimens.
(a) = Determined per applicable ASME-BPV-Code Sect. III, Subsection NB, Article NB-2331-(a-1,2,3).
(b) = 0° and 180° specimens had the same values.

TABLE B 3/4.4-1 (Continued)

REACTOR VESSEL TOUGHNESS

(PLATES)

| PIECE NO. | CODE NO. | MATERIAL | VESSEL LOCATION | DROP WEIGHT RESULTS (°F) | $RT_{NDT}$ (a)(b) (°F) | TEMPERATURE OF CHARPY V-NOTCH* @ 30 ft - lb | TEMPERATURE OF CHARPY V-NOTCH* @ 50 ft - lb | MINIMUM UPPER SHELF $C_V$ ENERGY ft-lb |
|---|---|---|---|---|---|---|---|---|
| 142-102 | F-6411-01 | SA 533-GRB-CL1 | Lower Shell Plate | -40 | -40 | -56 | -37 | 156 |
| 142-102 | F-6411-02 | SA 533-GRB-CL1 | Lower Shell Plate | -10 | 0 | 0 | +40 | 111 |
| 142-102 | F-6411-03 | SA 533-GRB-CL1 | Lower Shell Plate | -60 | -60 | -2 | +30 | 107 |
| 124-102 | F-6407-04 | SA 533-GRB-CL1 | Intermed. Shell Plate | -30 | -30 | +10 | +30 | 129 |
| 124-102 | F-6407-05 | SA 533-GRB-CL1 | Intermed. Shell Plate | -20 | -20 | +30 | +60 | 114 |
| 124-102 | F-6407-06 | SA 533-GRB-CL1 | Intermed. Shell Plate | -20 | -20 | +15 | +40 | 133 |
| 122-102 | F-6407-01 | SA 533-GRB-CL1 | Upper Shell Plate | -20 | -20 | +16 | +36 | N.A |
| 122-102 | F-6407-02 | SA 533-GRB-CL1 | Upper Shell Plate | -30 | -30 | -5 | +27 | N.A |
| 122-102 | F-6407-03 | SA 533-GRB-CL1 | Upper Shell Plate | -20 | -20 | +7 | +32 | N.A. |
| 102-102 | F-6414-01 | SA 533-GRB-CL1 | Closure Head Dome | -60 | 0 | +22 | +57 | N.A. |
| 102-102 | F-6414-02 | SA 533-GRB-CL1 | Closure Head Dome | -40 | -10 | +12 | +46 | N.A. |
| 150-102 | F-6410-01 | SA 533-GRB-CL1 | Bottom Head Dome | -70 | -60 | -19 | +5 | N.A. |
| 150-102 | F-6410-02 | SA 533-GRB-CL1 | Bottom Head Dome | -70 | -70 | -32 | -17 | N.A. |

N.A. = Not Applicable (no minimum upper shelf requirement).
* = Lower bound curve values of transverse specimens.
(a) = Determined per applicable ASME-BPV-Code Sect. III, Subsection NB, Article NB-2331-(a-1,2,3).
(b) = 0° and 180° specimens had the same values.

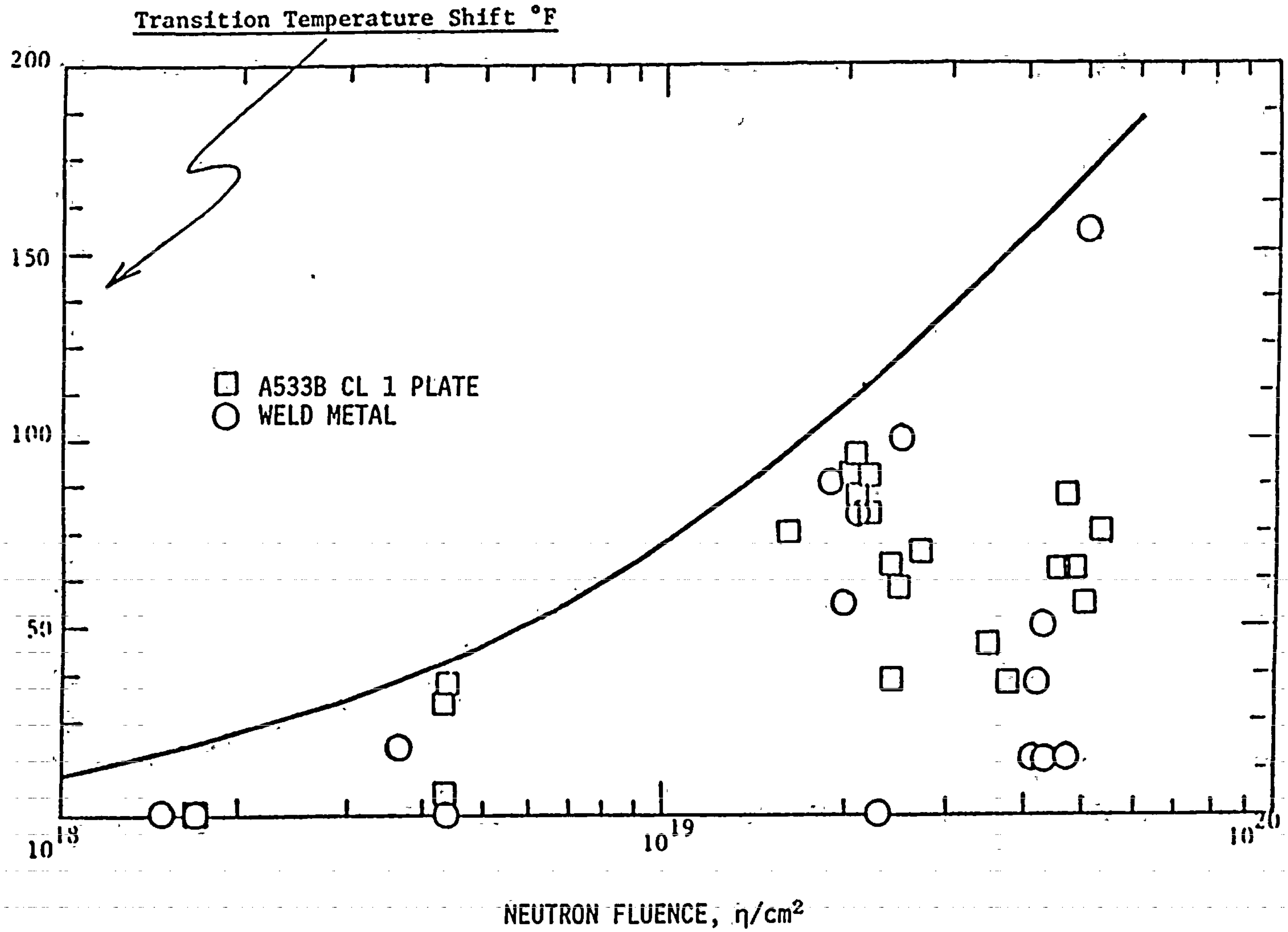


FIGURE B 3/4.4-1

NIL-DUCTILITY TRANSITION TEMPERATURE INCREASE AS A FUNCTION OF FAST (E > 1 MeV) NEUTRON FLUENCE (550°F IRRADIATION)

PRESSURE/TEMPERATURE LIMITS (Continued)

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two shutdown cooling suction line relief valves, one located in each shutdown cooling suction line, while maintaining the limits imposed on the RCS heatup and cooldown rates, ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 255°F during cooldown and 295°F during heatup. Either one of the two SCS suction line relief valves provides relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) the inadvertent safety injection actuation with two HPSI pumps injecting into a water-solid RCS with full charging capacity and with letdown isolated. These events are the most limiting energy and mass addition transients, respectively, when the RCS is at low temperatures.

The limitations imposed on the RCS heatup and cooldown rates are provided to assure low temperature overpressure protection (LTOP) with the two shutdown cooling suction line relief valves operable. At low temperatures with the relief valves aligned to the RCS, it is necessary to restrict heatup and cooldown rates to assure that the P/T limits are not exceeded. During worst case transients, RCS peak pressures can reach the relief valve setpoint, 467 psig, plus accumulation. At temperatures greater than 255°F during cooldown and 295°F during heatup, the heatup and cooldown rate limitations assure the limits of Appendix G to 10 CFR 50 will not be exceeded with overpressure protection provided by the primary safety valves.

3/4.4.9 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a (g) (6) (i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

## 3/4.4.10 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737.

## 3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the Safety Injection System (SIS) safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the RCS provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration, and pressure ensure that the safety injection tanks will adequately perform their function in the event of a LOCA in MODE 1, 2, 3, or 4.

A minimum of 25% narrow range corresponding to 1790 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet of borated water are used in the safety analysis as the volume in the SITs. To allow for instrument accuracy, 28% narrow range corresponding to 1802 cubic feet and 72% narrow range corresponding to 1914 cubic feet, are specified in the Technical Specification.

A minimum of 593 psig and a maximum pressure of 632 psig are used in the safety analysis. To allow for instrument accuracy, 600 psig minimum and 625 psig maximum are specified in the Technical Specification.

A boron concentration of 2000 ppm minimum and 4400 ppm maximum are used in the safety analysis. The Technical Specification lower limit of 2300 ppm in the SIT assures that the backleakage from RCS will not dilute the SITs below the 2000 ppm limit assumed in the safety analysis prior to the time when draining of the SIT is necessary.

The SIT isolation valves are not single failure proof; therefore, whenever the valves are open power shall be removed from these valves and the switch keylocked open. These precautions ensure that the SITs are available during a Limiting Fault.

The SIT nitrogen vent valves are not single failure proof against depressurizing the SITs by spurious opening. Therefore, power to the valves is removed while they are closed to ensure the safety analysis assumption of four pressurized SITs.

All of the SIT nitrogen vent valves are required to be operable so that, given a single failure, all four SITs may still be vented during post-LOCA long-term cooling. Venting the SITs provides for SIT depressurization capability which ensures the timely establishment of shutdown cooling entry conditions as assumed by the safety analysis for small break LOCAs.

The limits for operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a MODE where this capability is not required.

For MODES 3 and 4 operation with pressurizer pressure less than 1837 psia the Technical Specifications require a minimum of 57% wide range corresponding

SAFETY INJECTION TANKS (Continued)

to 1361 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet of borated water per tank, when three safety injection tanks are operable and a minimum of 36% wide range corresponding to 908 cubic feet and a maximum of 75% narrow range corresponding to 1927 cubic feet per tank, when four safety injection tanks are operable at a minimum pressure of 235 psig and a maximum pressure of 625 psig. To allow for instrument inaccuracy, 60% wide range instrument corresponding to 1415 cubic feet, and 72% narrow range instrument corresponding to 1914 cubic feet, when three safety injection tanks are operable, and 39% wide range instrument corresponding to 962 cubic feet, and 72% narrow range instrument corresponding to 1914 cubic feet, when four SITs are operable, are specified in the Technical Specifications. To allow for instrument inaccuracy 254 psig is specified in the Technical Specifications.

The instrumentation vs. volume correlation for the SITs is as follows:

| Volume | Narrow Range | Wide Range |
|---|---|---|
| 962 $ft^3$ | <0% | 39% |
| 1415 $ft^3$ | <0% | 60% |
| 1802 $ft^3$ | 28% | 78% |
| 1914 $ft^3$ | 72% | 83% |

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems with the RCS temperatures greater than or equal to 350°F ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double-ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provided this protection by dissolving in the sump water and causing its final pH to be raised to greater than or equal to 7.0.

The surveillance requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide

ECCS SUBSYSTEMS (Continued)

assurance that proper ECCS flows will be maintained in the event of a LOCA.* Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The requirement to dissolve a representative sample of TSP in a sample of RWT water provides assurance that the stored TSP will dissolve in borated water at the postulated post-LOCA temperatures.

The term "minimum bypass recirculation flow," as used in Specification 4.5.2e.3. and 4.5.2f., refers to that flow directed back to the RWT from the ECCS pumps for pump protection. Testing of the ECCS pumps under the condition of minimum bypass recirculation flow in Specification 4.5.2f. verifies that the performance of the ECCS pumps supports the safety analysis minimum RCS pressure assumption at zero delivery to the RCS.

## 3/4.5.4 REFUELING WATER TANK

The OPERABILITY of the refueling water tank (RWT) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that (1) sufficient water plus 10% margin is available to permit 20 minutes of engineered safety features pump operation, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

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*The following test conditions, which apply during flow balance tests, ensure that the ECCS subsystems are adequately tested.

1. The pressurizer pressure is at atmospheric pressure.
2. The miniflow bypass recirculation lines are aligned for injection.
3. For LPSI system, (add/subtract) 6.4 gpm (to/from) the 4900 gpm requirement for every foot by which the difference of RWT water level above the RWT RAS setpoint level (exceeds/is less than) the difference of RCS water level above the cold leg centerline.

REFUELING WATER TANK (Continued)

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between 7.0 and 8.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The limit on the RWT solution temperature ensures that the assumptions used in the LOCA analyses remain valid.

## 3/4.6.1 PRIMARY CONTAINMENT

### 3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

### 3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, $P_a$. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 $L_a$ or less than or equal to 0.75 $L_t$, as applicable during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.

### 3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

## 3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 4 psig and (2) the containment peak pressure does not exceed the design pressure of 60 psig during LOCA conditions.

The maximum peak pressure expected to be obtained from a LOCA event is 49.5 psig. The limit of 2.5 psig for initial positive containment pressure will limit the total pressure to 49.5 psig which is less than the design pressure (60 psig) and is consistent with the safety analyses.

## 3/4.6.1.5 AIR TEMPERATURE

The limitation on containment average air temperature ensures that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis.

## 3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 49.5 psig in the event of a LOCA. The containment design pressure is 60 psig. The measurement of containment tendon lift-off force; the tensile tests of the tendon wires or strands; the examination and testing of the sheathing filler grease; and the visual examination of tendon anchorage assembly hardware, surrounding concrete and the exterior surfaces of the containment are sufficient to demonstrate this capability. The tendon wire or strand samples will also be subjected to tests. All of the required testing and visual examinations should be performed in a time frame that permits a comparison of the results for the same operating history.

The Surveillance Requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," Revision 1, 1974.

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, the results of the engineering evaluation, and the corrective actions taken.

## 3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. To provide assurance that the 42-inch valves cannot be inadvertently opened, they are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevent power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 42-inch valves, the 8-inch valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations.

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 $L_a$ leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

## 3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

## 3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The containment spray system and the containment cooling system are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the containment spray system also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

## 3/4.6.2.2 IODINE REMOVAL SYSTEM

The OPERABILITY of the iodine removal system ensures that sufficient $N_2H_4$ is added to the containment spray in the event of a LOCA. The limits on $N_2H_4$ volume and concentration ensure adequate chemical available to remove iodine from the containment atmosphere following a LOCA.

## 3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment automatic isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through GDC 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The only valves in Table 6.2.4-1 of the PVNGS FSAR that are not required to be listed in Table 3.6-1 are the following: main steam safety valves, main steam atmospheric dump valves, and main steam isolation valves. The main steam safety valves have very high pressure setpoints to actuate and are covered by Specification 3/4.7.1.1. The atmospheric dump valves and the main steam isolation valves are covered by Specifications 3/4.7.1.6 and 3/4.7.1.5, respectively.

## 3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit (or the purge system) is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water and (3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

The use of ANSI Standard N509 (1980) in lieu of ANSI Standard N509 (1976) to meet the guidance of Regulatory Guide 1.52, Revision 2, Positions C.6.a and C.6.b, has been found acceptable as documented in Revision 2 to Section 6.5.1 of the Standard Review Plan (NUREG-0800).

## 3/4.7.1 TURBINE CYCLE

### 3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam safety valves (MSSVs) limit secondary system pressure to within 110% (1397 psia) of the design pressure (1270 psia) during the most severe anticipated operational transient. For design purposes the valves are sized to pass a minimum of 102% of the RATED THERMAL POWER at 102% of design power. The adequacy of this relieving capacity is demonstrated by maintaining the Reactor Coolant System pressure below NRC acceptance criteria (120% of design pressure for large feedwater line breaks, CEA ejection and 110% of design pressure for all overpressurization events).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition including the Summer 1975 Addenda. The total relieving capacity for all twenty MSSVs at 110% of system design pressure (adjusted for 50 psi pressure drop to valves inlet) is $19.44 \times 10^6$ lbm/hr. This capacity is less than the total rated capacity as the MSSVs are operating at an inlet pressure below rated conditions. At these same secondary pressure conditions, the total steam flow at 102% (2% uncertainty) of 3817 MWt (RATED THERMAL POWER plus 17 MWt pump heat input) is $17.83 \times 10^6$ lbm/hr. The ratio of this total steam flow to the total capacity is 109.2%.

STARTUP and/or POWER OPERATION is allowable with MSSVs inoperable if the maximum allowable power level is reduced to a value equal to the product of the ratio of the number of MSSVs available per steam generator to the total number of MSSVs per steam generator with the ratio of the total steam flow to available relieving capacity.

$$\text{Allowable Power Level} = \left(\frac{10-N}{10}\right) \times 109.2$$

The ceiling on the variable over power reactor trip is also reduced to an amount over the allowable power level equal to the BAND given for this trip in Table 2.2-1.

$$SP = \text{Allowable Power Level} + 9.8$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER. This is a ratio of the available relieving capacity over the total steam flow at rated power.

SAFETY VALVES (continued)

| | | |
|---|---|---|
| 10 | = | total number of secondary safety valves for one steam generator. |
| N | = | number of inoperable main steam safety valves on the steam generator with the greater number of inoperable valves. |
| 109.2 | = | ratio of main steam safety valve relieving capacity of 110% steam generator design pressure to calculated steam flow rate at 100% plant power + 2% uncertainty (see above text) |
| 9.8 | = | BAND between the maximum thermal power and the variable over-power trip setpoint ceiling |

## 3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

Each electric-driven auxiliary feedwater pump is capable of delivering a minimum feedwater flow of 750 gpm at a pressure of 1270 psia to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a minimum feedwater flow of 750 gpm at a pressure of 1270 psia to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the shutdown cooling system may be placed into operation.

## 3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank ensures that a minimum water volume of 300,000 gallons is available to maintain the Reactor Coolant System at HOT STANDBY for 8 hours followed by an orderly cooldown to the shutdown cooling entry (350°F) temperature with concurrent total loss-of-site power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

## 3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm primary-to-secondary tube leak in the steam generator of the affected steam line and a concurrent loss-of-offsite electrical power. These values are consistent with the assumptions used in the safety analyses.

## 3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the safety analyses.

## 3/4.7.1.6 ATMOSPHERIC DUMP VALVES

The limitation on maintaining the nitrogen accumulator at a pressure $\geq$ 400 psig is to ensure that a sufficient volume of nitrogen is in the accumulator to operate the associated ADV which holds the plant at hot standby while dissipating core decay heat or which allows a flow of sufficient steam to maintain a controlled reactor cooldown rate. A pressure of 400 psig retains sufficient nitrogen volume for 4 hours of operation at hot standby plus 6.5 hours of operation to reach cold shutdown under natural circulation conditions in the event of failure of the normal control air system.

## 3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations to 120°F and 230 psig are based on a steam generator $RT_{NDT}$ of 40°F and are sufficient to prevent brittle fracture.

## 3/4.7.3 ESSENTIAL COOLING WATER SYSTEM

The OPERABILITY of the essential cooling water system ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

## 3/4.7.4 ESSENTIAL SPRAY POND SYSTEM

The OPERABILITY of the essential spray pond system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

## 3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either (1) provide normal cooldown of the facility, or (2) to mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 27-day cooling water supply to safety-related equipment without exceeding their design basis temperature and is consistent with the intent of the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

## 3/4.7.6 ESSENTIAL CHILLED WATER SYSTEM

The OPERABILITY of the essential chilled water system ensures that sufficient cooling capacity is available for continued operation of equipment and control room habitability during accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

The Essential Chilled Water System (ECWS), in conjunction with respective emergency HVAC units, is required in accordance with Specification Definition 1.18 to provide heat removal in maintaining the various Engineered Safety Features (ESFs) room space design temperatures below the associated equipment qualification limits for the range of Design Basis Accident conditions. The normal HVAC system is redundant to the emergency HVAC system in maintaining the space design conditions of required safety systems during normal operating conditions and Design Basis Accident Conditions not involving seismic events or loss of offsite power. A seven (7) day Action requirement is for a single ECWS out of service, based on the high reliability of offsite power and availability of the normal HVAC system. The normal HVAC system contains two 100% redundant chillers. Action requirements are provided to ensure operability of the vital bus inverters and emergency battery chargers, by verifying within one hour that the normal HVAC system is providing space cooling to the vital power distribution rooms. The Action requirement is provided to establish within 8 hours operability of the safe shutdown systems which do not depend on the inoperable ECWS. The 8 hour period provides a reasonable time in which to establish operability of this complement of key safety systems. This requirement ensures that a functional train of safe shutdown equipment is available to put the plant in a safe, stable condition for the most probable abnormal operational occurences. An Action requirement of 24 hours is provided to establish operability of the remaining required safety systems which do not depend on the inoperable ECWS.

## 3/4.7.7 CONTROL ROOM ESSENTIAL FILTRATION SYSTEM

The OPERABILITY of the control room essential filtration system ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50.

The use of ANSI Standard N509 (1980) in lieu of ANSI Standard N509 (1976) to meet the guidance of Regulatory Guide 1.52, Revision 2, Positions C.6.a and C.6.b, has been found acceptable as documented in Revision 2 to Section 6.5.1 of the Standard Review Plan (NUREG-0800).

## 3/4.7.8 ESF PUMP ROOM AIR EXHAUST CLEANUP SYSTEM

The OPERABILITY of the ESF pump room air exhaust cleanup system ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses.

The use of ANSI Standard N509 (1980) in lieu of ANSI Standard N509 (1976) to meet the guidance of Regulatory Guide 1.52, Revision 2, Positions C.6.a and C.6.b, has been found acceptable as documented in Revision 2 to Section 6.5.1 of the Standard Review Plan (NUREG-0800).

## 3/4.7.9 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Plant Review Board. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommedations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection. Therefore, the required inspection interval varies

SNUBBERS (Continued)

inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. In order to establish the inspection frequency for each type of snubber, it was assumed that the frequency of failures and initiating events is constant with time and that the failure of any snubber could cause that system to be unprotected and to result in failure during an assumed initiating event. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers.

To provide assurance of snubber functional reliability one of three functional testing methods are used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7-1, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

## 3/4.7.10 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shield mechanism.

## 3/4.7.11 SHUTDOWN COOLING SYSTEM

The OPERABILITY of two separate and independent shutdown cooling subsystems ensures that the capability of initiating shutdown cooling in the event of an accident exists even assuming the most limiting single failure occurs. The safety analysis assumes that shutdown cooling can be initiated when conditions permit.

The limits of operation with one shutdown cooling inoperable for any reason minimize the time exposure of the plant to an accident event occurring concurrent with the failure of a component on the other shutdown cooling subsystem.

## 3/4.7.12 CONTROL ROOM AIR TEMPERATURE

Maintaining the control room air temperature less than or equal to 80°F ensures that (1) the ambient air temperature does not exceed the allowable air temperature for continuous duty rating for the equipment and instrumentation in the control room and (2) the control room will remain habitable for operations personnel during plant operation. The 30 days to return the control room air temperature to less than or equal to 80°F in the Action Statement is consistent with the equipment qualification program for the control room.

## 3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source.

The required steady state frequency for the emergency diesels is 60 + 1.2/-0.3 Hz to be consistent with the safety analysis to provide adequate safety injection flow.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that (1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The surveillance requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977.

A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

The surveillance requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.010 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

If any other metallic structures (e.g., buildings, new or modified piping systems, conduit) are placed in the ground in the vicinity of the fuel oil storage system or if the original system is modified, the adequacy and frequency of inspections of the cathodic protection system shall be re-evaluated and adjusted in accordance with Regulatory Guide 1.137.

## 3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance. The circuit breakers will be tested in accordance with NEMA Standard Publication No. AB-2-1980. For a frame size of 250 amperes or less, the field tolerances of the high and low setting of the injected current will be within +40%/-25% of the setpoint (pickup) value. For a frame size of 400 amperes or greater, the field tolerances will be ±25% of the setpoint (pickup) value. The circuit breakers should not be affected when tested within these tolerances.

The surveillance requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes. There are no surveillance requirements on fuses. For in-line fuses, the applicable surveillance would require removing the fuses from the circuit which would destroy the fuse. The test data for surveillance on the other fuses would not indicate whether the fuse was degrading which has been stated by the fuse manufacturer and Idaho National Engineering Laboratory.

The OPERABILITY of the motor-operated valves thermal overload protection and/or bypass devices ensures that these devices will not prevent safety related valves from performing their function. The surveillance requirements for demonstrating the OPERABILITY of these devices are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

## 3/4.9 REFUELING OPERATIONS

## BASES

### 3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The value of 0.95 or less for $K_{eff}$ includes a 1% delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of 2150 ppm or greater also includes a conservative uncertainty allowance of 50 ppm boron.

### 3/4.9.2 INSTRUMENTATION

The OPERABILITY of the startup channel neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

### 3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

### 3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

### 3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

## 3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) the machine will be used for movement of fuel assemblies, (2) the machine has sufficient load capacity to lift a fuel assembly, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

## 3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

## 3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling loop be in operation, and circulating reactor coolant at a flow rate equal to or greater than 4000 gpm ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 135°F as required during the REFUELING MODE, (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification, and (3) the $\Delta T$ across the core will be maintained at less than 75°F during the REFUELING MODE. The required flowrate of $\geq$ 4000 gpm ensures that 240 hours after reactor shutdown sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 135°F as required during REFUELING MODE; this assumes a shutdown cooling heat exchanger cooling water flowrate of 14000 gpm, a cooling water inlet temperature of $\leq$ 105°F at $\geq$ 27 1/2 hours after reactor shutdown, and the decay heat curve of CESSAR-F Figure 6.2.1-1 and reactor operation for two years at 4000 MWt.

Without a shutdown cooling train in operation steam may be generated; therefore, the containment should be sealed off to prevent escape of any radioactivity, and any operations that would cause an increase in decay heat should be secured.

The requirement to have two shutdown cooling loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange, ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool the core.

A shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during surveillance testing of ECCS pumps. This is necessary to meet Surveillance 4.5.2, flow testing of the HPSI pumps without other pumps running, and 4.3.3.5, testing of the containment spray pumps and LPSI pumps during surveillance of the remote shutdown system.

## 3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge valves will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

## 3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth (at least 23 feet above the top of the spent fuel) is available to remove a nominal 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly for a maximum fuel rod pressurization of 1200 psig. The minimum water depth is consistent with the assumptions of the safety analysis.

## 3/4.9.12 FUEL BUILDING ESSENTIAL VENTILATION SYSTEM

The limitations on the fuel building essential ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses.

The use of ANSI Standard N509 (1980) in lieu of ANSI Standard N509 (1976) to meet the guidance of Regulatory Guide 1.52, Revision 2, Positions C.6.a and C.6.b, has been found acceptable as documented in Revision 2 to Section 6.5.1 of the Standard Review Plan (NUREG-0800).

## BASES

### 3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEAs worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations. Although testing will be initiated from MODE 2, temporary entry into MODE 3 is necessary during some CEA worth measurements. A reasonable recovery time is available for return to MODE 2 in order to continue PHYSICS TESTING.

### 3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to (1) measure CEA worth, (2) determine the reactor stability index and damping factor under xenon oscillation conditions, (3) determine power distributions for non-normal CEA configurations, (4) measure rod shadowing factors, and (5) measure temperature and power coefficients. Special test exception permits MTC to exceed limits in Specification 3.1.1.3 during performance of PHYSICS TESTS.

### 3/4.10.3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality with less than four reactor coolant pumps in operation and is required to perform certain STARTUP and PHYSICS TESTS while at low THERMAL POWER levels.

### 3/4.10.4 CEA POSITION, REGULATING CEA INSERTION LIMITS AND REACTOR COOLANT COLD LEG TEMPERATURE

This special test exception permits the CEAs to be positioned beyond the insertion limits and reactor coolant cold leg temperature to be outside limits during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

### 3/4.10.5 MINIMUM TEMPERATURE AND PRESSURE FOR CRITICALITY

This special test exception permits reactor criticality at low THERMAL POWER levels with $T_{cold}$ below the minimum critical temperature and pressure during PHYSICS TESTS which are required to verify the low temperature physics predictions and to ensure the adequacy of design codes for reduced temperature conditions. The Low Power Physics Testing Program at low temperature (300°F) and a pressure of 500 psia is used to perform the following tests:

1. Biological shielding survey test
2. Isothermal temperature coefficient tests
3. CEA group tests
4. Boron worth tests
5. Critical configuration boron concentration

## 3/4.10 SPECIAL TEST EXCEPTIONS

## BASES

### 3/4.10.6 SAFETY INJECTION TANKS

This special test exception permits testing the low pressure safety injection system check valves. The pressure in the injection header must be reduced below the head of the low pressure injection pump in order to get flow through the check valves. The safety injection tank (SIT) isolation valve must be closed in order to accomplish this. The SIT isolation valve is still capable of automatic operation in the event of an SIAS; therefore, system capability should not be affected.

### 3/4.10.7 SPENT FUEL POOL LEVEL

This special test exception permits loading of the initial core with the spent fuel pool dry.

### 3/4.10.8 SAFETY INJECTION TANK PRESSURE

This special test exception allows the performance of PHYSICS TESTS at low pressure/low temperature (600 psig, 320°F) conditions which are required to verify the low temperature physics predictions and to ensure the adequacy of design codes for reduced temperature conditions.

## 3/4.11.1 SECONDARY SYSTEM LIQUID WASTE DISCHARGE TO ONSITE EVAPORATION PONDS

### 3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that at any time during the life of the nuclear station, the annual total body dose due to ground contamination of an UNRESTRICTED AREA, arising from transportation and deposition by wind of the accumulated activity discharged to the pond from the secondary system of the plant (if the pond gets dried up) on the UNRESTRICTED AREA, is within the guidelines of 10 CFR Part 20 for the above-mentioned postulated event.

Restricting the concentrations of the secondary liquid wastes discharged to the onsite evaporation ponds will restrict the quantity of radioactive material that can get accumulated in the ponds. This, in turn, provides assurance that in the event of an uncontrolled release of the pond's contents to an UNRESTRICTED AREA, the resulting total body annual exposure from ground contamination to a MEMBER OF THE PUBLIC at the nearest exclusion area boundary will be within 0.5 rem.

This specification applies to the secondary system liquid waste discharges of radioactive materials from all reactor units to the onsite evaporation ponds. Since the chemical neutralizer tank concentrations will bound concentrations in other secondary waste discharges, surveillance requirements stipulate that sampling and analysis of other secondary waste discharges need be performed only if the sampling and analysis of the contents of the chemical neutralizer tank shows that the neutralizer tank concentration exceeds the specified LLD.

The required detection capabilities for radioactive materials in the secondary liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

### 3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with

DOSE (Continued)

the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This specification applies to the release of liquid effluents from each reactor at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

3/4.11.1.3 LIQUID HOLDUP TANKS

The tanks referred to in this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

The limit of 60 curies is based on the analyses given in Section 2.4 of the PVNGS FSAR and on the amount of soluble (not gaseous) radioactivity in the Refueling Water Tank in Table 2.4-26.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year

DOSE RATE (Continued)

to the total body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

This specification applies to the release of radioactive materials in gaseous effluents from all reactor units at the site.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The surveillance requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

This specification applies to the release of radioactive materials in gaseous effluents from each reactor unit at the site.

## 3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the surveillance requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat-producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

This specification applies to the release of radioactive materials in gaseous effluents from each reactor unit at the site.

## 3/4.11.2.4 GASEOUS RADWASTE TREATMENT

The OPERABILITY of the GASEOUS RADWASTE SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

GASEOUS RADWASTE TREATMENT (Continued)

This specification applies to the release of radioactive materials in gaseous effluents from each reactor unit at the site.

The minimum analysis frequency of 4/M (i.e. at least 4 times per month at intervals no greater than 9 days and a minimum of 48 times a year) is used for certain radioactive gaseous waste sampling in Table 4.11-2. This will eliminate taking double samples when quarterly and weekly samples are required at the same time.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. (Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, or injection of dilutants to reduce the concentration below the flammability limits.) Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

This specification considers postulated radioactive releases due to a waste gas system leak or failure, and limits the quantity of radioactivity contained in each pressurized gas storage tank in the GASEOUS RADWASTE SYSTEM to assure that a release would be substantially below the guidelines of 10 CFR Part 100 for a postulated event.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a MEMBER OF THE PUBLIC at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

3/4.11.3 SOLID RADIOACTIVE WASTE

This specification addresses the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and

TOTAL DOSE (Continued)

direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the reactor units and outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

## 3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

## BASES

### 3/4.12.1 MONITORING PROGRAM

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an _a_ _priori_ (before the fact) limit representing the capability of a measurement system and not as an _a_ _posteriori_ (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, _HASL-300_ (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," _Anal. Chem. 40_, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report _ARH-SA-215_ (June 1975).

## 3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 $m^2$ provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 $kg/m^2$.

## 3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

## 5.0 DESIGN FEATURES

### 5.1 SITE

SITE AND EXCLUSION BOUNDARIES

5.1.1 The site and exclusion boundaries shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

GASEOUS RELEASE POINTS

5.1.3 The gaseous release points shall be as shown in Figure 5.1-3.

### 5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, prestressed concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 146 feet.

b. Nominal inside height = 206.5 feet.

c. Minimum thickness of concrete walls = 3 feet, 8 inches.

d. Minimum thickness of concrete roof = 3 feet, 8 inches.

e. Minimum thickness of concrete floor pad = 10.5 feet.

f. Nominal thickness of steel liner = 0.25 inch.

g. Net free volume = $2.6 \times 10^6$ cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 60 psig and a temperature of 300°F.

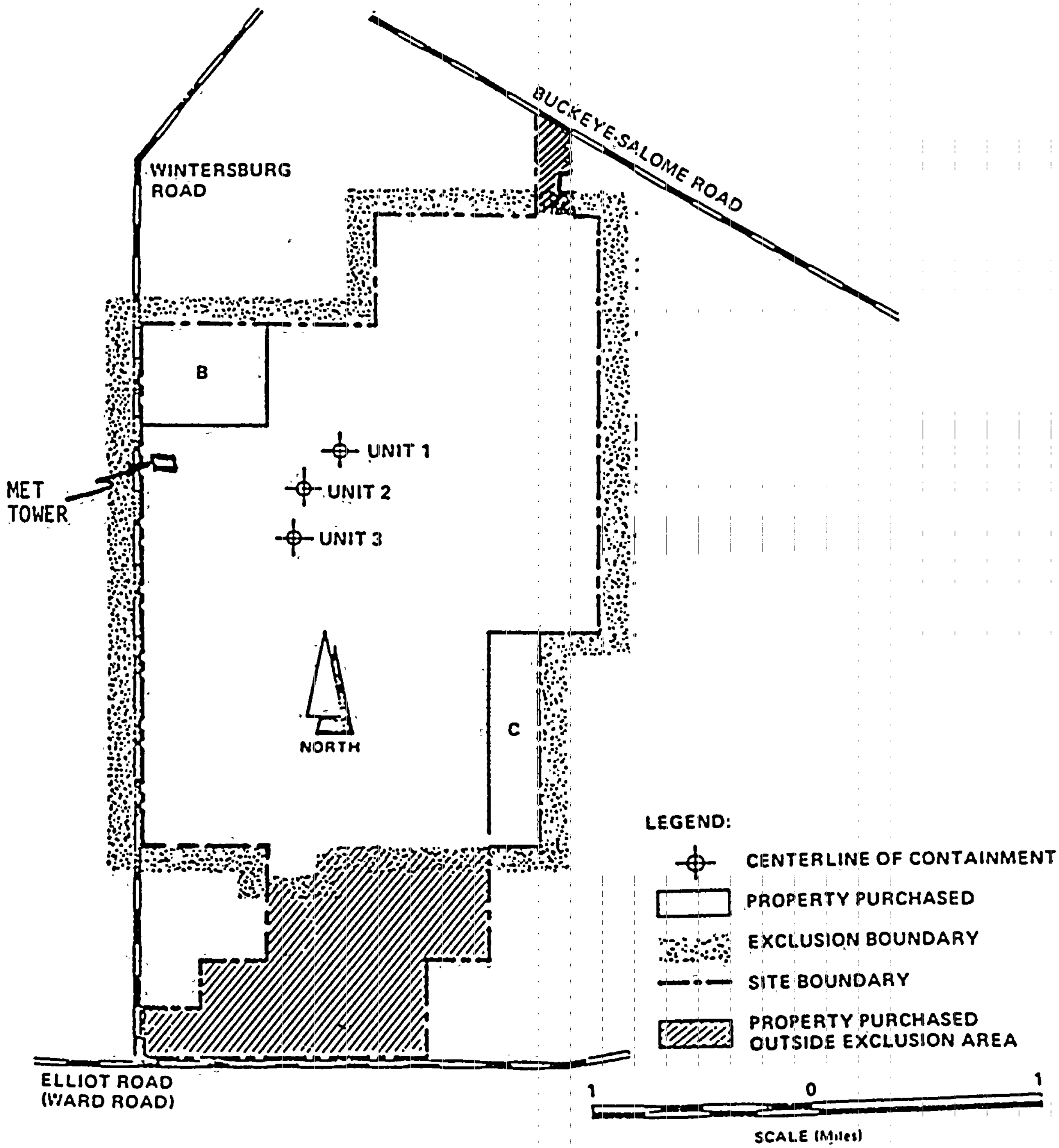



FIGURE 5.1-1

SITE AND EXCLUSION BOUNDARIES

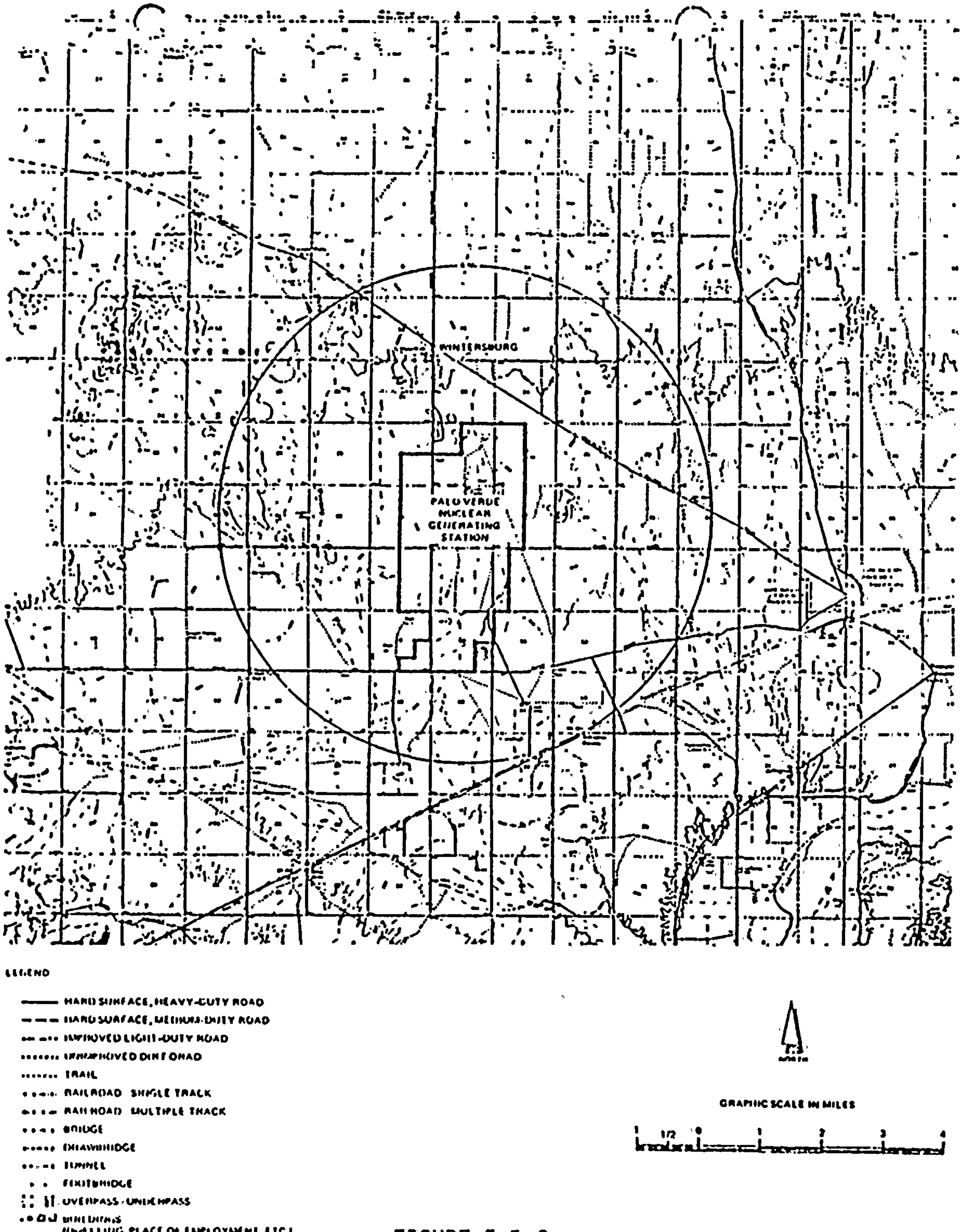
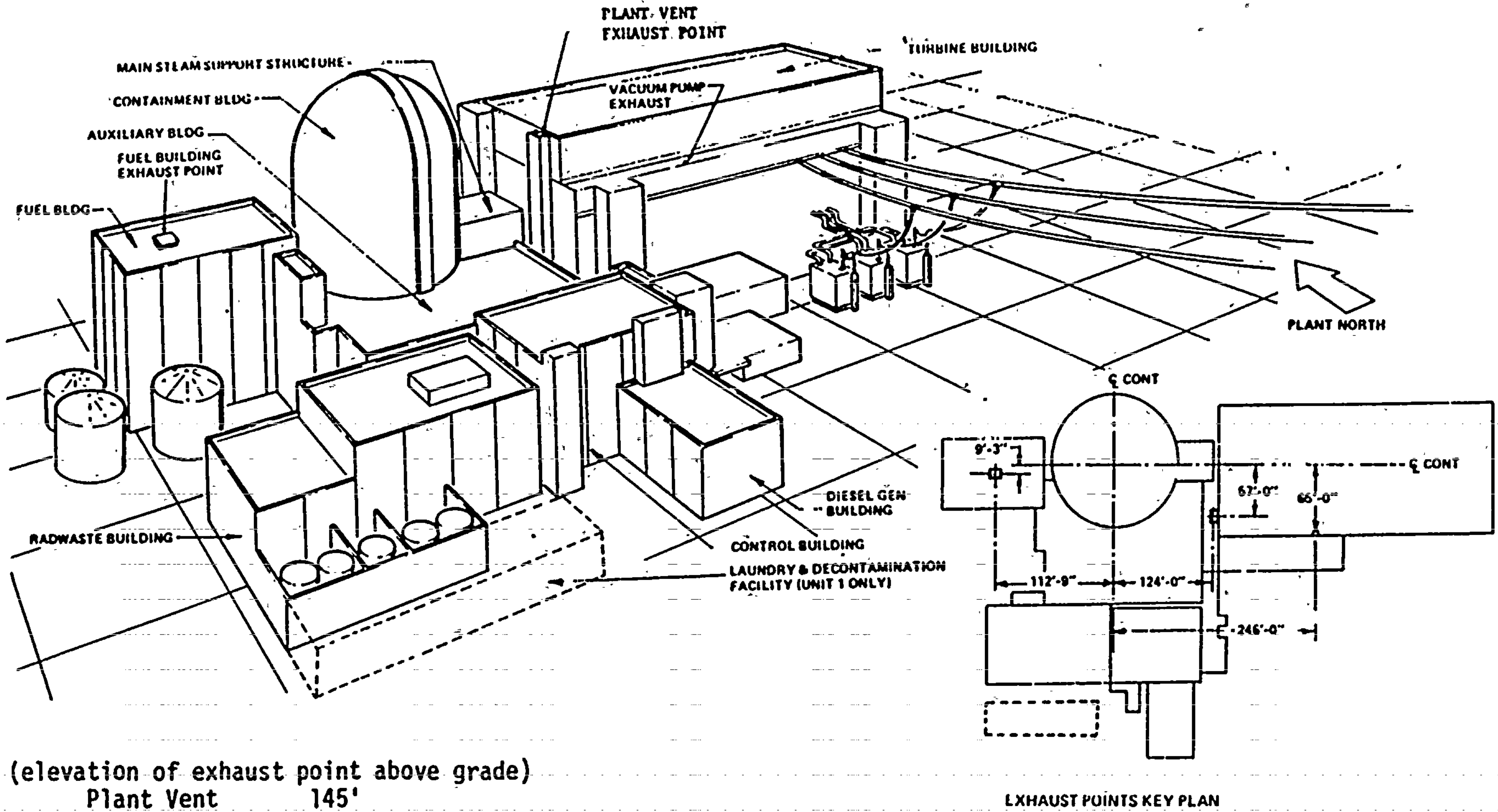



FIGURE 5.1-2

LOW POPULATION ZONE

PALO VERDE NUCLEAR GENERATING STATION




(elevation of exhaust point above grade)

| | |
|---|---|
| Plant Vent | 145' |
| Fuel Building | 116' |
| Vacuum Pump | 84' |

FIGURE 5.1-3

GASEOUS RELEASE POINTS

## 5.3 REACTOR CORE

### FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 241 fuel assemblies with each fuel assembly containing 236 fuel rods or burnable poison rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of approximately 1950 grams uranium. Each burnable poison rod shall have a nominal active poison length of 136 inches. The initial core loading shall have a maximum enrichment of 3.35 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4 weight percent U-235.

### CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 76 full-length and 13 part-length control element assemblies.

## 5.4 REACTOR COOLANT SYSTEM

### DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant of the applicable surveillance requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

### VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 13,900 + 300/-0 cubic feet at a nominal $T_{avg}$ of 593°F.

## 5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

## 5.6 FUEL STORAGE

### 5.6.1 CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A $k_{eff}$ equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 2.6% delta k/k for uncertainties as described in Section 9.1 of the FSAR.

b. A nominal 9.5 inch center-to-center distance between fuel assemblies placed in the storage racks in a high density configuration.

5.6.1.2 The $k_{eff}$ for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

### DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 137 feet - 6 inches.

### CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1329 fuel assemblies.

## 5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Tables 5.7-1 and 5.7-2.

## TABLE 5.7-1

## COMPONENT CYCLIC OR TRANSIENT LIMITS

| COMPONENT | CYCLIC OR TRANSIENT LIMIT | DESIGN CYCLE OR TRANSIENT |
|---|---|---|
| Reactor Coolant System | 500 system heatup and cooldown cycles at rates $\leq$ 100°F/hr. | Heatup cycle - Temperature from $\leq$ 70°F to $\geq$ 565°F; cooldown cycle - Temperature from $\geq$ 565°F to $\leq$ 70°F. |
| | 500 pressurizer heatup and cooldown cycles at rates $\leq$ 200°F/hr. | Heatup cycle - Pressurizer temperature from $\leq$ 70°F to $\geq$ 653°F; cooldown cycle - Pressurizer temperature from $\geq$ 653°F to $\leq$ 70°F. |
| | 10 hydrostatic testing cycles. | RCS pressurized to 3125 psia with RCS temperature between 120°F and 400°F. |
| | 480 reactor trip cycles, turbine trip cycles, and loss of reactor coolant flow. | Includes combinations of reactor trips due to operator errors, equipment malfunctions, and total loss of reactor coolant flow. |
| | 200 seismic stress cycles. | Subjection to a seismic event equal to one-half the design basis earthquake (DBE). |
| | 1 complete loss of secondary pressure cycle. | Loss of secondary pressure from either steam generator due to a complete double-ended break of a steam generator steam or feedwater nozzle. |

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

| COMPONENT | CYCLIC OR TRANSIENT LIMIT | DESIGN CYCLE OR TRANSIENT |
|---|---|---|
| | 200 primary system leak test cycles | Leak test primary system at a pressure of 2250 psia at a temperature from 120°F to 400°F. |
| Pressurizer Spray Nozzle | Calculate usage factor per Table 5.7-2. | Main spray (less than four RCP operating) with fluid $\Delta T_m$ > 200°F.<br><br>Auxiliary spray with fluid $\Delta T_a$ > 200°F. |

$\Delta T_m$ = The difference in temperature between the pressurizer and main spray water as adjusted by the instrument correction factor.

$\Delta T_a$ = The difference in temperature between the pressurizer and Auxiliary spray water as adjusted by the instrument correction factor.

## TABLE 5.7-2

## PRESSURIZER SPRAY NOZZLE USAGE FACTOR

| Main Spray | | | | Auxiliary Spray | | | |
|---|---|---|---|---|---|---|---|
| $\Delta T_m$ | $N_A$ | N | $N/N_A$ | $\Delta T_a$ | $N_A$ | N | $N/N_A$ |
| 201-250 | 7900 | | | 201-250 | 50000 | | |
| 251-300 | 4500 | | | 251-300 | 2200 | | |
| 301-350 | 2900 | | | 301-350 | 1300 | | |
| 351-400 | 1900 | | | 351-400 | 850 | | |
| 401-450 | 1200 | | | 401-450 | 550 | | |
| 451-500 | 850 | | | 451-500 | 375 | | |
| 501-550 | 555 | | | 501-550 | 225 | | |
| | | | | 551-600 | 150 | | |
| | | $\Sigma N/N_A$ = | ________ | | | $\Sigma N/N_A$ = | ________ |

Cumulative Usage Factor

$\Sigma N/N_A$ (Main Spray) ____________

$\Sigma N/N_A$ (Aux. Spray) ____________

Total ____________ = Cumulative Usage Factor

TABLE 5.7-2 (Continued)

Where:

$\Delta T_a = (T_{101} - T_{229}) + 60$

$\Delta T_m = (T_{101} - T_{103^* \text{ or } 104^*}) + 70$

NA = Allowable number of spray cycles

N = Number of cycles in $\Delta T$ range indicated

Calculational Method:

1. The spray cycle is defined as any initiation and termination of main or auxiliary spray flow throughout the pressurizer spray nozzle.

2. If the difference between pressurizer water temperature and the spray water temperature exceeds 200°F each spray cycle and the corresponding temperature difference is logged.

3. The spray nozzle usage factor shall be calculated as follows:

   A. Fill in Column "N" above.

   B. Calculate "$N/N_A$" (Divide N by $N_A$).

   C. Add Column "$N/N_A$" to find $\Sigma N/N_A$.

   $\Sigma N/N_A$ is the cumulative spray nozzle usage factor. If the cumulative usage factor is equal to or less than 0.65 no further action is required.

4. If the cumulative usage factor exceeds 0.65, subsequent pressurizer spray operation shall continue to be monitored and an engineering evaluation of nozzle fatigue shall be performed within 90 days. The evaluation shall determine that the nozzle remains acceptable for additional service beyond the 90 day period of subsequent spray operation shall be restricted so that the difference between the pressurizer water temperature and the spray water temperature shall be limited to less than or equal to 200°F when spray is operated.

*Use lower of two temperatures.

## 6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor, or during his absence from the Control Room, a designated individual per Table 6.2-1, shall be responsible for the Control Room command function. A management directive to this effect, signed by the Vice President-Nuclear Production shall be reissued to all station personnel on an annual basis.

## 6.2 ORGANIZATION

### OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown in Figure 6.2-1.

### UNIT STAFF

6.2.2.1 The unit organization shall be as shown in Figure 6.2-2 and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the Control Room when fuel is in the reactor. In addition, while the reactor is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator shall be in the Control Room.

c. A radiation protection technician* shall be onsite when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. A site Fire Team of at least five members shall be maintained onsite at all times*. The Fire Team shall not include the Shift Supervisor, the STA, nor the 3 other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

6.2.2.2 The unit staff working hours shall be as follows:

a. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., Senior Reactor Operators, Reactor Operators, radiation protection technicians, auxiliary operators, and key maintenance personnel.

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*The radiation protection technician and Fire Team composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

UNIT STAFF (Continued)

b. Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modifications, on a temporary basis, the following guidelines shall be followed (this excludes the STA working hours):

1) An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time.

3) A break of at least 8 hours should be allowed between work periods, including shift turnover time.

4) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

c. Any deviation from the above guidelines shall be authorized by the Assistant Vice President-Nuclear Production Support, Director, Standards and Technical Support or the Plant Manager or their designees who are at the manager level or above, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime in their respective groups shall be reviewed monthly by these authorized individuals or their designees to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

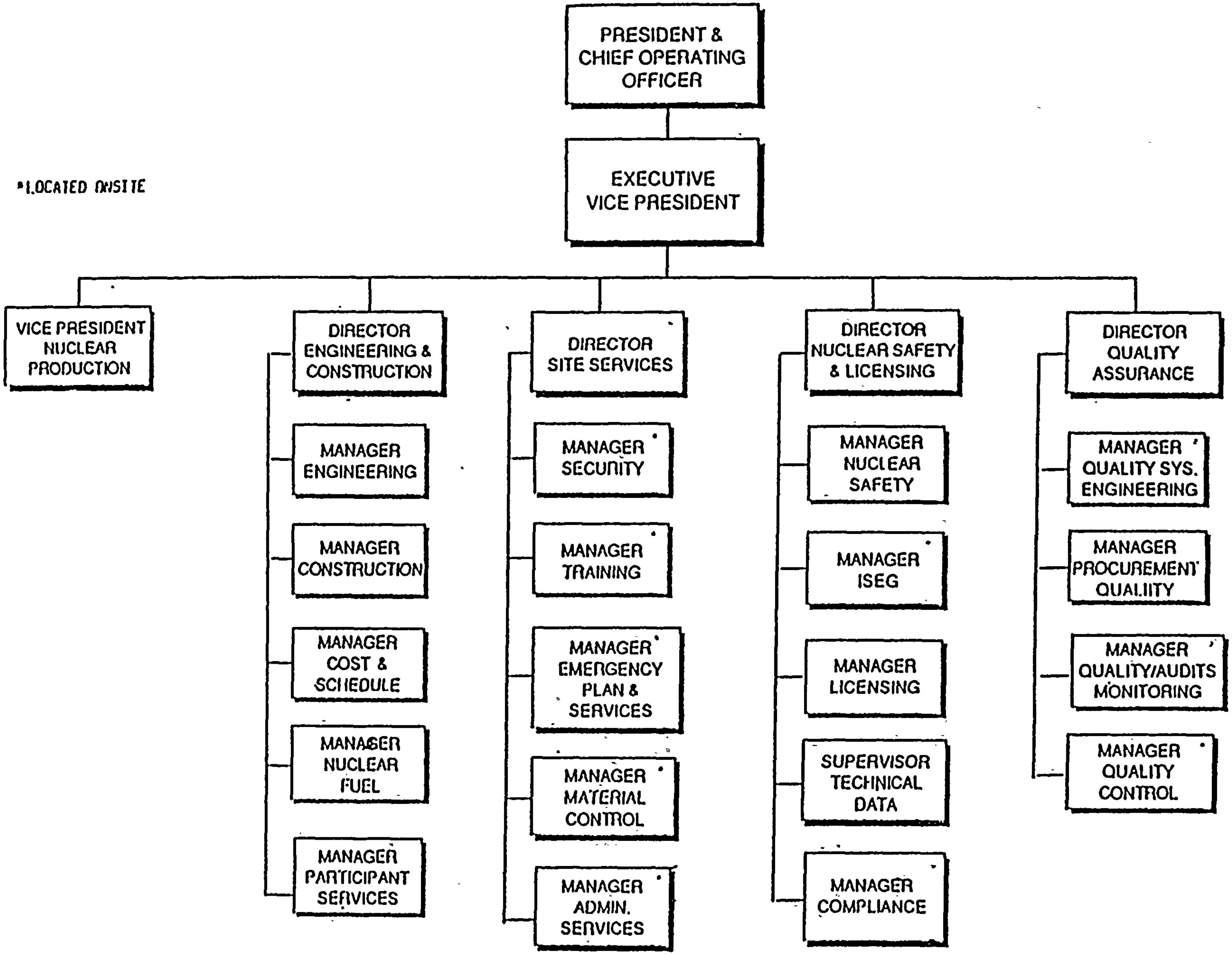
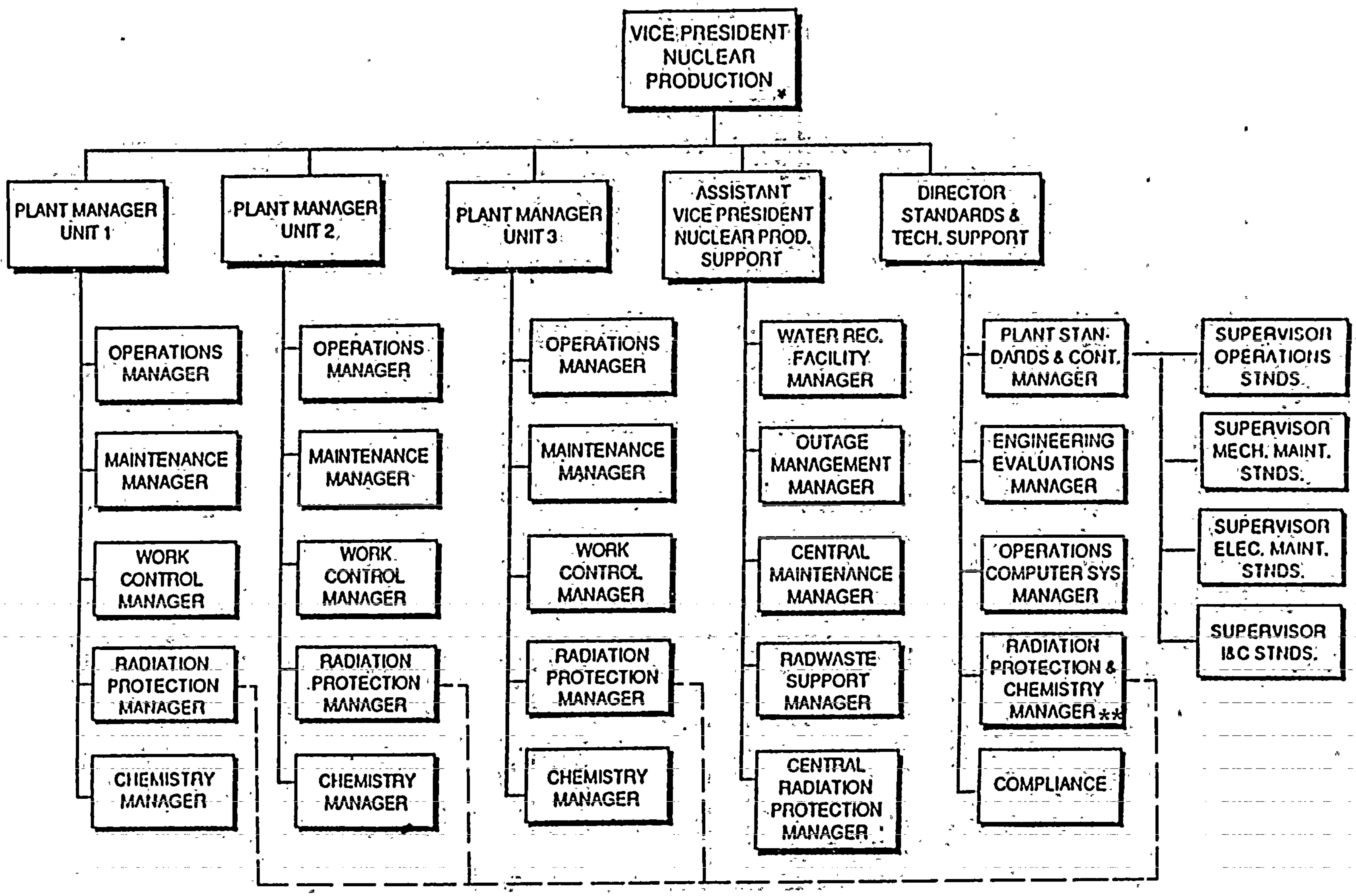



FIGURE 6.2-1

OFFSITE ORGANIZATION




—MATRIX
*Located Offsite
**Designated Regulatory Guide 1.8 "Radiation Protection Manager"
---Dotted Lines Indicate Programmatic Procedural Direction and Problem Resolution

FIGURE 6.2-2

ONSITE UNIT ORGANIZATION

## TABLE 6.2-1

## MINIMUM SHIFT CREW COMPOSITION

| POSITION | NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION | |
|---|---|---|
| | MODE 1, 2, 3, OR 4 | MODE 5 OR 6 |
| SS | 1 | 1 |
| SRO | 1 | None |
| RO | 2 | 1 |
| AO | 2 | 1 |
| STA | 1 | None |

SS - Shift Supervisor with a Senior Reactor Operators License
SRO - Individual with a Senior Reactor Operators License
RO - Individual with a Reactor Operators License
AO - Nuclear Operator I or II
STA - Shift Technical Advisor

The Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 1, 2, 3, or 4, an individual with a valid Senior Operator license shall be designated to assume the Control Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator or Operator license shall be designated to assume the Control Room command function.

## 6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

### FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety.

### COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a Bachelor's Degree in engineering or related science and at least two years professional level experience in his field.

### RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly to reduce human errors as much as practical, and to detect potential nuclear safety hazards.

### AUTHORITY

6.2.3.4 The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the Director, Nuclear Safety and Licensing, Plant Manager, and the Manager, Nuclear Safety Group (NSG).

### RECORDS

6.2.3.5 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Director, Nuclear Safety and Licensing.

## 6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA shall be onsite and shall be available in the control room within 10 minutes whenever one or more units are in MODE 1, 2, 3, or 4.

## 6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANS 3.1-1978 and Regulatory Guide 1.8, September 1975, except for the Radiation Protection and Chemistry Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and plant operating characteristics, including transients and accidents.

*Not responsible for sign-off function.

## 6.4 TRAINING

6.4.1 A training program for the unit staff shall be maintained under the direction of the Director, Site Services or his designee and shall meet or exceed the requirements and recommendations of Section 5.0 of ANS 3.1-1978 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

## 6.5 REVIEW AND AUDIT

### 6.5.1 PLANT REVIEW BOARD (PRB)

FUNCTION

6.5.1.1 The Plant Review Board shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PRB shall be composed of the following personnel:

| | |
|---|---|
| Member: | Engineering Evaluations Manager |
| Member: | Operations Standards Supervisor |
| Member: | Mechanical Maintenance Standards Supervisor |
| Member: | Electrical Maintenance Standards Supervisor |
| Member: | Operations Managers for Unit 1, Unit 2, Unit 3 |
| Member: | STA Supervisor |
| Member: | I&C Standards Supervisor |
| Member: | Radiation Protection and Chemistry Manager |
| Member: | Quality Systems/Engineering Manager |

The Vice President-Nuclear Production shall designate the Chairman and Vice-Chairmen in writing. The Chairman and Vice-Chairmen may be from outside the members listed above provided that they meet ANSI Standard 3.1, 1978.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PRB activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PRB shall meet at least once per calendar month and as convened by the PRB Chairman, Vice-Chairmen, or his designated alternate.

## QUORUM

6.5.1.5 The quorum of the PRB necessary for the performance of the PRB responsibility and authority provisions of these Technical Specifications shall consist of the Chairman, Vice-Chairmen, or his designated alternate and five members including alternates.

## RESPONSIBILITIES

6.5.1.6 The PRB shall be responsible for:

a. Review of all administrative control procedures and changes.

b. Review of all proposed changes to Appendix "A" Technical Specifications.

c. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Nuclear Safety Group (NSG).

d. Review of REPORTABLE EVENTS.

e. Review of unit operations to detect potential nuclear safety hazards.

f. Performance of special reviews, investigations or analyses and reports thereon as requested by the Vice President-Nuclear Production.

g. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last PRB meeting.

## AUTHORITY

6.5.1.7 The PRB shall:

a. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6c. above constitutes an unreviewed safety question.

b. Provide written notification within 24 hours to the Vice President-Nuclear Production, Plant Manager and NSG of disagreement between the PRB and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1 above.

## RECORDS

6.5.1.8 The PRB shall maintain written minutes of each PRB meeting that, at a minimum, document the results of all PRB activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to the Plant Manager, Vice President-Nuclear Production and NSG.

## 6.5.2 TECHNICAL REVIEW AND CONTROL ACTIVITIES

6.5.2.1 The Director, Standards and Technical Support shall assure that each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, is prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto.

6.5.2.2 Phase I - IV tests described in the FSAR that are performed by the plant operations staff shall be approved by the Director, Standards and Technical Support or the Engineering Evaluations Manager as previously designated by the Vice President-Nuclear Production. Test results shall be approved by the Director, Standards and Technical Support or the Engineering Evaluations Manager.

6.5.2.3 Proposed modifications to unit nuclear safety-related structures, systems and components shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to nuclear safety-related structures, systems and components shall be approved prior to implementation by the Plant Manager; or by the Director, Standards and Technical Support as previously designated by the Vice President-Nuclear Production.

6.5.2.4 Individuals responsible for reviews performed in accordance with 6.5.2.1, 6.5.2.2, and 6.5.2.3 shall be members of the station nuclear production supervisory staff, previously designated by the Vice President-Nuclear Production to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate designated review personnel.

6.5.2.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the Vice President-Nuclear Production.

6.5.2.6 The station security program and implementing procedures shall be reviewed. Recommended changes shall be approved by the Director, Site Services or designated alternate and transmitted to the Vice President-Nuclear Production and to the NSG.

6.5.2.7 The station emergency plan and implementing procedures shall be reviewed. Recommended changes shall be approved by the Director, Site Services or designated alternate and transmitted to the Vice President-Nuclear Production and to the NSG.

6.5.2.8 The Director, Standards and Technical Support shall assure the performance of a review by a qualified individual/organization of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering the evaluation, recommendations and disposition of the corrective action to prevent recurrence.

TECHNICAL REVIEW AND CONTROL ACTIVITIES (Continued)

6.5.2.9 The Director, Standards and Technical Support shall assure the performance of a review by a qualified individual/organization of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, radwaste treatment systems, and the Pre-planned Alternate Sampling Program.

6.5.2.10 Reports documenting each of the activities performed under Specifications 6.5.2.1 through 6.5.2.9 above shall be maintained. Copies shall be provided to the Vice President-Nuclear Production and the Nuclear Safety Group.

## 6.5.3 NUCLEAR SAFETY GROUP (NSG)

### FUNCTION

6.5.3.1 The NSG shall function to provide independent review and shall be responsible for the audit of designated activities in the areas of:

a. nuclear power plant operations
b. nuclear engineering
c. chemistry and radiochemistry
d. metallurgy
e. instrumentation and control
f. radiological safety
g. mechanical and electrical engineering
h. quality assurance practices

### COMPOSITION

6.5.3.2 The NSG shall consist of a Manager and at least four staff specialists. The Manager shall have a Bachelor's Degree in Engineering or the Physical Sciences. He will also have a minimum of 10 years experience in the power field with at least 3 of those years in the nuclear field. The NSG Manager will have at least 2 years of supervisor/managerial experience. Each staff specialist will have at least one of the following requirements:

a. Eight years experience in one of the designated areas in Specification 6.5.3.1. One of these 8 years will be at Palo Verde Nuclear Generating Station.

b. Bachelor's Degree in Engineering or a related science and 3 years of professional experience.

### CONSULTANTS

6.5.3.3 Consultants shall be utilized as determined by the NSG Manager to provide expert advice to the NSG.

### REVIEW

6.5.3.4 The NSG shall review:

a. The safety evaluations program and its implementation for (1) changes to procedures, equipment, systems or facilities within the power block, and (2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question;

REVIEW (Continued

b. Proposed changes to procedures, equipment, systems or facilities within the power block which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

g. All REPORTABLE EVENTS requiring 24 hours written notification;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the PRB.

AUDITS

6.5.3.5 Audits of unit activities shall be performed under the cognizance of the NSG. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training, and qualifications of the unit staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months.

e. Any other area of unit operation considered appropriate by the NSG or the Vice President-Nuclear Production.

f. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel.

AUDITS (Continued)

g. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year.

h. The radiological environmental monitoring program and the results thereof at least once per 12 months.

i. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

j. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.

k. The performance of activities required by the Operations Quality Assurance Criteria Manual to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975 at least once per 12 months.

AUTHORITY

6.5.3.6 The NSG shall report to and advise the Director, Nuclear Safety and Licensing on those areas of responsibility specified in Specifications 6.5.3.4 and 6.5.3.5.

RECORDS

6.5.3.7 Records of NSG activities shall be prepared and maintained. Report of reviews and audits shall be prepared monthly for the Director, Nuclear Safety and Licensing who will distribute it to the Vice President-Nuclear Production, Plant Manager, and to the management positions responsible for the areas audited.

6.6 REPORTABLE EVENT ACTION

The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified pursuant to the requirements of Section 50.72 to 10 CFR Part 50, and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PRB, and the results of this review shall be submitted to the Manager of Nuclear Safety Group and the Vice President-Nuclear Production.

## 6.7 SAFETY LIMIT VIOLATION

The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President-Nuclear Production, Plant Manager and Manager of Nuclear Safety Group shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PRB. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the Manager of the NSG and the Vice President-Nuclear Production within 30 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

## 6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978, and those required for implementing the requirements of NUREG-0737.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Modification of Core Protection Calculator (CPC) Addressable Constants--These procedures should include provisions to ensure that sufficient margin is maintained in CPC Type I Addressable Constants to avoid excessive operator interaction with the CPCs during reactor operation.

NOTES: (1) Modification to the CPC Addressable Constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the PRB.

(2) Modifications to the CPC software (including algorithm changes and changes in fuel cycle specific data) shall be performed in accordance with the most recent version of CEN-39(A)-P, "CPC Protection Algorithm Software Change Procedure," that has been determined to be applicable to the facility. Additions or deletions to CPC Addressable Constants or changes to Addressable Constant software limit values shall not be implemented without prior NRC approval.

PROCEDURES AND PROGRAMS (Continued)

h. PROCESS CONTROL PROGRAM implementation.

i. OFFSITE DOSE CALCULATION MANUAL implementation.

j. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975.

k. Pre-planned Alternate Sampling Program implementation.

l. Secondary water chemistry program implementation.

NOTE: The licensee shall perform a secondary water chemistry monitoring and control program that is in conformance with the program discussed in Section 10.3.4.1 of the CESSAR FSAR or another NRC approved program.

m. Post-Accident Sampling System implementation.*

n. Settlement Monitoring Program implementation.

NOTE: The licensee shall maintain a settlement monitoring program throughout the life of the plant in accordance with the program presented in Table 2.5-18 of the PVNGS FSAR or another NRC approved program.

o. CEA Reactivity Integrity Program implementation

NOTE: The licensee shall perform, after initial fuel load or after each reload, either a CEA symmetry test or worth measurements of all full-length CEA groups to address Section 4.2.2 of the PVNGS SER dated November 11, 1981.

p. Fuel Assembly Surveillance Program Implementation

NOTE: The licensee shall perform a fuel assembly surveillance program in conformance with the program discussed in Section 4.2.4 of the PVNGS SER dated November 11, 1981.

6.8.2 Each program or procedure of Specification 6.8.1, and changes thereto, shall be reviewed as specified in Specification 6.5 and approved prior to implementation. Programs, administrative control procedures and implementing procedures shall be approved by the Vice President-Nuclear Production, or designated alternate who is at supervisory level or above. Programs and procedures of Specification 6.8.1 shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant supervisory staff, at least one of whom is a Shift Supervisor or Assistant Shift Supervisor with an SRO on the affected unit.

c. The change is documented, reviewed in accordance with Specification 6.5.2 and approved by the Director, Standards and Technical Support or cognizant department head, as designated by the Vice President-Nuclear Production, within 14 days of implementation.

*Not required until prior to exceeding 5% of RATED THERMAL POWER.

PROCEDURES AND PROGRAMS (Continued)

6.8.4 The following programs shall be established, implemented, maintained, and shall be audited under the cognizance of the NSG at least once per 24 months:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the recirculation portion of the high pressure safety injection system, the shutdown cooling portion of the low pressure safety injection system, the post-accident sampling subsystem of the reactor coolant sampling system, the containment spray system, the post-accident sample return piping of the radioactive waste gas system, the post-accident sampling return piping of the liquid radwaste system, and the post-accident containment atmosphere sampling piping of the hydrogen monitoring subsystem. The program shall include the following:

(1) Preventive maintenance and periodic visual inspection requirements, and

(2) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

(1) Training of personnel,

(2) Procedures for monitoring, and

(3) Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(1) Identification of a sampling schedule for the critical variables and control points for these variables,

(2) Identification of the procedures used to measure the values of the critical variables,

(3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(4) Procedures for the recording and management of data,

PROCEDURES AND PROGRAMS (Continued)

(5) Procedures defining corrective actions for all off-control point chemistry conditions, and

(6) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

(1) Training of personnel, and

(2) Procedures for monitoring.

e. Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(1) Training of personnel,

(2) Procedures for sampling and analysis,

(3) Provisions for maintenance of sampling and analysis equipment.

f. Spray Pond Monitoring

A program which will identify and describe the parameters and activities used to control and monitor the Essential Spray Pond and Piping. The program shall be conducted in accordance with station manual procedures.

## 6.9 REPORTING REQUIREMENTS

### ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

### STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel

REPORTING REQUIREMENTS (Continued)

supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The Startup Report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrems/yr and their associated man-rem exposure according to work and job functions,** e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

Annual reports shall also include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.7. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

**This tabulation supplements the requirements of §20.407 of the 10 CFR Part 20.

ANNUAL REPORTS (Continued)

after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.7 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps**

*A single submittal may be made for a multiple unit station.

**One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (Continued)

covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 3.12-1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.8 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability**. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL.

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*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

**In lieu of submission with the first half year Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Semiannual Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM and to the OFFSITE DOSE CALCULATION MANUAL, as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be reported in accordance with 10 CFR 50.73.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

RECORD RETENTION (Continued)

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities; inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures of Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the FSAR.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released beyond the SITE BOUNDARY.

e. Records of transient or operational cycles for those unit components identified in Tables 5.7-1 and 5.7-2.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of quality assurance activities required by the QA Manual not listed in Section 6.10.1.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of PRB meetings and of NSG activities.

RECORD RETENTION (Continued)

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.9 including the date at which the service life commences and associated installation and maintenance records.

m. Records of audits performed under the requirements of Specifications 6.5.3.5 and 6.8.4.

n. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

o. Meteorological data, summarized and reported in a format consistent with the recommendations of Regulatory Guides 1.21 and 1.23.

p. Records of secondary water sampling and water quality.

## 6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

## 6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

*Radiation Protection personnel or personnel escorted by Radiation Protection personnel shall be exempt from the REP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA (Continued)

c. A radiation protection qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Radiation Protection Supervisor or his designated alternate in the REP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved REP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrems*, that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the REP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

## 6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the PCP:

Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

1) Sufficiently detailed information to totally suport the rationale for the change without benefit of additional or supplemental information; and

2) A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes.

*Measurement made at 18 inches from source of radioactivity.

## 6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s); and

2) A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations.

## 6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS*

6.15.1 Licensee-initiated major changes to the radioactive waste systems (liquid, gaseous, and solid):

Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PRB. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

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*Licensees may chose to submit the information called for in this specification as part of the annual FSAR update.

MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS (Continued)

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made; and

7) An estimate of the exposure to plant operating personnel as a result of the change.

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

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13. ABSTRACT (200 words or less)

The Palo Verde, Unit 3 Technical Specifications were prepared by the U.S. Nuclear Regulatory Commission to set forth the limits, operating conditions, and other requirements applicable to a nuclear reactor facility as set forth in Section 50.36 of 10 CFR Part 50 for the protection of the health and safety of the public.

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