SNUPPS

Standardized Nuclear Unit
Power Plant System

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July 8, 1983

SLNRC 83-0036 FILE: 0543
SUBJ: SNUPPS Technical Specifications

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Ref: 1. ULNRC-564, 6/29/82, Callaway Plant Technical Specifications
2. KMLNRC 82-230, 8/5/82, Draft Technical Specifications
3. ULNRC-614, 3/23/83, Technical Specification - Section 6.0
4. KMLNRC 83-012, 1/26/83, Draft Technical Specifications
5. KMLNRC 83-022, 3/4/83, Draft Technical Specifications

Dear Mr. Denton:

The referenced letters were transmittals of the SNUPPS Technical Specifications for NRC review. In January, 1983, the NRC returned to SNUPPS the initial NRC draft of the Wolf Creek and Callaway Technical Specifications. A meeting was held on February 3, 1983 to discuss the differences between the NRC draft and the SNUPPS submittals. As a result of this meeting, SNUPPS committed to provide to the NRC a resubmittal of the specifications which should be revised from the NRC draft, and a justification for each change. This submittal is attached hereto. Except where specifically noted otherwise, the changes apply to both plants.

The following confirmatory issues and license conditions listed in the Callaway and Wolf Creek Safety Evaluation Reports (NUREG-0830 and NUREG-0881, Section 1.8) contain Technical Specification requirements:

Confirmatory Issues

Callaway	Wolf Creek	Issues
2*	B.2*	Analysis of Steam Generator
6	B.6	Steam Generator Inservice Inspection
18	B.19	Low and/or Degraded Grid Voltage
24	B.25	Compliance with Position 1 of Regulatory Guide 1.63
28	B.28	TMI Action Plan - II.E.4.2 Containment Isolation Dependability

* Item Number in Section 1.8

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Licensing Condition

Callaway	Wolf Creek	Issues
8	B.8	Indicator, Alarms, and Test Features Provided for Instrumentation and Safety Functions

In order to resolve these items, requirements are to be included in the Technical Specifications for the SNUPPS plants. It is therefore requested that the Technical Specification requirements for these items be removed from the Safety Evaluation Reports and that the responsibility for resolution and tracking of all Technical Specification SER items be transferred to the NRC Technical Specification reviewer.

There are several items in the Technical Specifications which SNUPPS intends to change, but which are not attached hereto. These items include:

1. Cold overpressure specifications - Westinghouse is recalculating the SNUPPS cold overpressure temperature limit. The affected specifications will be revised and forwarded under separate cover when the limit is available.

2. Radiological Effluent Technical Specifications (RETS) - SNUPPS is reviewing the RETS sections and will prepare a submittal similar to this one by the end of the summer.

3. Ultimate heat sink level and temperature control (Union Electric only). SNUPPS is obtaining specific level and temperature values to use as criteria for operability.

In addition, there are a few areas where SNUPPS is investigating the possibility of modifying the requirements presently in the Technical Specifications. These areas include:

1. Boron dilution system

2. Emergency diesel generator testing and fuel oil chemistry requirements

3. Use of the RHR suction relief valves for cold overpressurization protection

4. Boron Injection Tank

5. Reactor vessel level indication system

SNUPPS intends to make another submittal in August, 1983 which will close out the majority of the above open items.

SNUPPS requests that the NRC complete its review of this submittal before the end of September so as to allow adequate time for resolution of the remaining issues before the end of 1983.

Very truly yours,

N A Petrick

Nicholas A. Petrick

JHR/dck/6b19

Attachment

cc: G. Edison NRC
J. Holonich NRC

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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 CONTROLLED LEAKAGE shall be that seal water flow from ~~supplied to~~ the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any installed reactor vessel or core component within the reactor 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 (microCurie/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 [redacted] Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977. X

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 ~~$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample.~~ X

See Attachment

Technical Specification: Controlled Leakage
1.8 Definition
3.4.6.2.e
4.4.6.2.1.c
Bases 3/4.4.6.2

Justification:

Revise the above referenced sections of Technical Specifications to reflect changes as noted. The bases for the changes is to define controlled leakage as leakage from the seal, which is consistent from a human engineering point of view with the balance of defined RCS leakage terms. This also provides a limit and surveillance which will monitor actual seal performance. Seal injection does not measure performance and seal degradation.

The charging pump discharge supply to the RCP seal water injection is controlled by locked and throttled angle stop check valves BGV-198, 199, 200, 201. Controlled leak off return is limited by a locked and throttled angle stop check valve BGV-202. HCV-182 allows flow to be balanced between the normal charging flow path to the reactor coolant loop 1 cold leg and the reactor coolant pump seals. Controlled leak off from the #1 seal is isolated by building isolation valves BGHV-8100 and 8112 on a Phase A containment isolation signal. Technical Specification 3.5.2 for ECCS subsystems establishes operability for the centrifugal charging pumps and safety injection pumps by verifying proper line ups, pump capacities, and balanced flows to assure minimum flow rates to the reactor coolant system normal injection flow paths.

Since Technical Specification 3.5.2 assures ECCS system operability, the revision of surveillance requirements for specification 4.4.6.2.1 will allow consideration for leakage and correct the assumptions made on the design and bases of the pump seal control system.

Eight gpm controlled leakage per pump was choosen to be consistent with the requirement to isolate the #1 pump seal as recommended by Westinghouse.

This change is submitted in conjunction with the proposed change to ECCS subsystems 4.5.2.g.2.

Specification 1.9 Page 1-2

Justification -

The changes in this definition are intended to eliminate a source of confusion that has existed in the industry for some time. While it clearly limits those things which constitute a CORE ALTERATION, it provides adequate assurance that unexpected reactivity excursions and/or fuel damage will be precluded. The proposed definition will allow video taping for core verification, video inspections of reactor internals, and other remote inspection techniques to be performed without the additional Technical Specification limitations imposed on CORE ALTERATIONS.

Technical Specification: Definition 1.11

Revise $\overline{E}$ definition to read:

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

Justification:

Technically the proposed definition makes compliance difficult due to: 1) the lack of exclusion of nuclides that have a half-life less than 15 minutes with the exception of Xe-138; and 2) the absence of a provision to set a limit of 95% on the activity in the coolant sample to be identified. In addition iodine activity should be excluded due to the fact that if the radioiodine level in the coolant was at its limit, its contribution would only be approximately 1% of the total activity.

The specific activity LCO also limits Dose Equivalent I-131 and addresses a separate limit for same.

W

PROCESS CONTROL PROGRAM

1.22 The PROCESS CONTROL PROGRAM (PCP) 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 SOLIDIFICAITON 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 fresh water, at the time of disposal within the limits of 10 CFR Part 61 and of low level radioactive waste disposal sites. X

PURGE - PURGING

1.23 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.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total core heat transfer rate to the reactor coolant of 3411 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 loss of stationary gripper coil voltage.

REPORTABLE OCCURRENCE

1.27 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specifications 6.9.1.10 and 6.9.1.11

SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be:

a. The amount of reactivity by which the reactor is subcritical, or

b. The amount of reactivity by which the reactor shall become subcritical as a result of opening of the reactor trip breakers assuming the reactivity associated with the single rod cluster assembly of highest worth is unavailable for insertion.

Definition: 1.28

Justification:

Shutdown Margin Technical Specifications

Two issues were raised in the recent past concerning shutdown margin technical specifications. One issue is the requirement to assume that the most reactive rod is fully withdrawn. The other is a conflict between shutdown margin requirements and control rod insertion limit requirements. Technical specifications require via the shutdown margin definition that shutdown margin calculations assume the most reactive rod is fully withdrawn. This requirement can be justified for those instances when credit is being taken for withdrawn rods to meet shutdown margin requirements (for instance during critical operation). However in those instances when all rods are known to be fully inserted or the shutdown margin is met without relying on withdrawn rods, the requirement to assume the most reactive rod is withdrawn constitutes an unnecessary burden on plant operation and the needless processing of primary coolant. Considering the second issue, surveillance requirements which when performed verify that shutdown margin requirements are met when critical consist of verifying that control rod insertion limits are met. In other words, the shutdown margin is verified by verifying that control rod insertion limits are met. The problem is that action statements for violation of the two specifications are different. Violation of shutdown margin requirements requires immediate boration, however, action statements for control rod insertion limits require no such boration. These two different action statements create a situation in which the operation must respond to two different requirements for the same initial event. This is not a desirable practice.

In the opinion of Westinghouse changes to the Tech Specs to clarify these issues are justifiable and result in a benefit to the operating utility in ease of understanding of technical specifications and boration requirements while shutdown. Any revisions stand on their own and are not related to other issues such as the boron dilution accident.

Justification for Definition 1.28 continued:

Definition of Shutdown Margin

The definition of shutdown margin has been revised such that it is no longer required to account for the most reactive rod being withdrawn for shutdown conditions. The definition is written to allow the option of taking credit for withdrawn rods or having the reactor shutdown by an amount equal to the shutdown margin in which case it is not necessary nor no longer required to account for a stuck rod. Additionally the shutdown margin definition has been worded to imply that only available reactivity may be considered, that is, if a rod is known to be untrippable the definition of shutdown margin would preclude taking credit for the stuck rod to meet shutdown margin requirements since that rod cannot contribute to reactivity added as a result of opening trip breakers.

Related changes occur in specifications: 3/4.1.1.1, 3/4.1.1.2, 3/4.1.3.1, 3/4.1.3.6, 3/4.10.1, 3/4.10.2, 3/4.10.3.

K

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 core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.35 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST ~~shall~~ may include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.36 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.37 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

1.38 A VENTILATION EXHAUST TREATMENT SYSTEM is 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 Features (ESF) Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

(ATTACHMENT 2)

Definition: 1.35 Trip Actuating Device Operational Test

Justification:

The second sentence says the test shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint ...

The Trip Actuating Device Operational Test is used to describe the test used on both trip actuating devices that have a setpoint such as a pressure switch and on those that do not have a setpoint and therefore cannot be adjusted - for example P-4 permissive which is initiated simply as a back contact of the reactor trip breakers. Most, if not all of the devices that are tested by the operational test, cannot be tested to the level of actually testing for setpoint adjustment due to lack of proper test facilities. These facilities are normally provided to accomplish the Analog Channel Operational Test only. The definition needs to be changed to read, "may include adjustments as necesary," vs "shall include adjustments."

Definition 1.38

Justification -

Although charcoal adsorbers and HEPA filters are the most common filters used today, other filters (such as roughing filters) may also be used for this purpose. Deleting the words charcoal and HEPA would allow the use of another filter should a better one become available.

TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
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°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

*Excluding decay heat.
**Fuel in the reactor vessel with any ~~the~~ vessel head closure bolt ~~s~~ less than fully tensioned ~~or with the head removed.~~

Specification - Table 1.2 Page 1-9

Justification -

The proposed change makes it very clear that the unit enters Mode 6, from Mode 5 upon detensioning of the first closure bolt on the first pass, and remains in Mode 6 until (1) the last closure bolt is tensioned on the last pass, or (2) all fuel is removed from the vessel.

There have been instances of non-compliance resulting from ambiguity of the current wording.

K

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlocks Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

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

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, place the channel in the tripped condition within ~~1~~ 2 hours, and within the following 12 hours either:

   1. Determine that Equation 2.2-1 was satisfied for the affected channel and adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1, or

   2. 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 Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $$Z + R + S \leq TA$$

Where:

Z = the value for column Z of Table 2.2-1 for the affected channel,

R = the "as measured" value (in percent span) of rack error for the affected channel,

S = either the "as measured" value (in percent span) of the sensor error, or the value is column S of Table 2.2-1 for the affected channel, and
in

TA = the value for column TA of Table 2.2-1 for the affected channel.

Specification: 2.2.1.b

Justification:

The 1 hour criteria should be increased to 2 hours for the same reason as Table 3.3-1 of Specification 3.3.1, pages 3/4 3-7 through 3/4 3-9.

There is a typographical error in the "S = ..." In the second line, after "value", "is" should be "in".

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where: ΔT = Measured ΔT by RTD Manifold Instrumentation,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT,

τ_1, τ_2 = Time constants utilized in lead-lag ~~controller~~ compensator for ΔT, τ_1 = 8 sec., τ_2 = 3 sec,

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT,

τ_3 = Time constants utilized in the lag compensator for ΔT, τ_3 = 2 secs,

ΔT_0 = Indicated ΔT at RATED THERMAL POWER,

K_1 = ~~1.09~~, 1.10

K_2 = 0.0138,

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag ~~controller~~ Compensator for T_{avg} dynamic compensation,

τ_4, τ_5 = Time constants utilized in the lead-lag ~~controller~~ compensator for T_{avg}, τ_4 = 33 secs., τ_5 = 4 secs,

T = Average temperature, °F,

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg},

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 2 secs,

K-1

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

T'	$\leq$	588.5°F (Nominal T_{avg} at RATED THERMAL POWER),
K_3	=	0.00671,
P	=	Pressurizer pressure, psig,
P'	=	2235 psig (Nominal RCS operating pressure),
S	=	Laplace transform operator, sec^{-1},

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) for $q_t - q_b$ between -35% and + 7%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -35%, the ΔT Trip Setpoint shall be automatically reduced by 1.26% of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +7%, the ΔT Trip Setpoint shall be automatically reduced by 1.05% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.8%. OF SPAN

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 [T \left(\frac{1}{1 + \tau_6 S}\right) - T''] \cancel{+} - f_2(\Delta I)\}$$

Where:

ΔT = ~~As defined in Note 1.~~ Measured ΔT by RTD Manifold Instrumentation,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = ~~As defined in Note 1,~~ Lead-Lag compensator on measured ΔT

τ_1, τ_2 = ~~As defined in Note 1,~~ Time constants ~~used~~ utilized in lead-lag compensator for ΔT, τ_1 = 8 Secs, τ_2 = 3 secs,

$\frac{1}{1 + \tau_3 S}$ = ~~As defined in Note 1,~~ Lag compensator on measured ΔT

τ_3 = ~~As defined in Note 1,~~ Time constant utilized in the lag compensator for ΔT, τ_3 = 2 secs,

ΔT_o = ~~As defined in Note 1,~~ Indicated ΔT at RATED THERMAL POWER,

K_4 = 1.09,

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag ~~controller~~ compensator for T_{avg} dynamic compensation,

τ_7 = Time constant~~s~~ utilized in the rate-lag ~~controller~~ compensator for T_{avg}, τ_7 = 10 secs.,

$\frac{1}{1 + \tau_6 S}$ = ~~As defined in Note 1,~~ Lag compensator on measured Tavg,

τ_6 = ~~As defined in Note 1,~~ Time constant utilized in measured Tavg lag compensator, τ_6 = 2 secs,

K-1

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 3: (Continued)

K_6 = 0.00128/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$,

T = ~~As defined in Note 1,~~ Average temperature, °F,

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 588.5°F),

S = ~~As defined in Note 1, and~~ Laplace transform operator, sec^{-1}, and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.0%. of span.

K-1.

Table 2.2-1 Reactor Trip System Instrumentation Setpoints Notation, Pages 2-8, 2-9, 2-10 and 2-11.

Justification -

The terms Lead-Lag Compensator and Lead-Lag Controller are used indeterminably in the definitions on page 2-8. In each case the devices referred to are Dynamic Compensators, not controllers. In similar terms for Note 3 on page 2-10, the correct name for the circuits is rate-lag compensator.

Note 2 (pg. 2-9) and Note 4 (pg. 2-11) refer to the Allowable Value in %. For clarity the value should be in terms of % of span.

Note 3 (pg. 2-10) The positive sign in the formula for the Overpower ΔT S.P. in the term for the penalty factory f_2 (ΔI) is incorrect. The sign should be '-' as in the formula for Overtemperature ΔT S.P. The factor f_2 (ΔI) is presently set to zero, however, it could be reinstated in the future. This axial offset is always a penalty, if used, and would subtract from the setpoint.

In Note 3 the terms are referenced to the definitions in Note 1. To aid the operator in relating to the terms in the OPΔT-SP formula it would be better if he wasn't required to page back to Note 1.

JHR/dck/6b28

Pressurizer Pressure

In each of the pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full equivalent); and on increasing power, automatically reinstated by P-7.

Low Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately ~~49%~~ 48% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The steam generator water level low-low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified setpoint provides allowances for starting delays of the auxiliary feedwater system.

General Warning Alarm

A general warning alarm in both solid state protection system trains initiates a reactor trip. The general warning alarm is activated in each train of the solid state protective system when the train is being tested or is otherwise inoperable. The general warning alarm trip provides protection for conditions under which both trains of the protection system may be rendered inoperable.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide reactor core protection against DNB as a result of complete loss of forced coolant flow. The specified setpoints assure a reactor trip signal is generated before the low flow trip setpoint is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip setpoint is reached shall not exceed 0.3 seconds. On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine Trip initiates a reactor trip. On decreasing power the Reactor trip caused by the turbine trip is automatically blocked by P-9 (a power level of approximately ~~10%~~ 50% of RATED THERMAL POWER)~~with a turbine impulse chamber at approximately 10% of full power equivalent);~~ and on increasing power, reinstated automatically by P-9.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. The ESF instrumentation channels which initiate a safety injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range reactor trip (i.e., prevents premature block of source range trip), provides a backup block for source range neutron flux doubling, and de-energizes the high voltage to the detectors. On decreasing power, Source Range level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables reactor trips on low flow in more than one primary coolant loop, ~~more than one reactor coolant pump breaker open,~~ reactor coolant pump bus undervoltage and underfrequency, ~~turbine trip,~~ pressurizer low pressure and pressurizer high level. On decreasing power the above listed trips are automatically blocked.

P-8 On increasing power P-8 automatically enables reactor trips on low flow in one or more primary coolant loops, ~~and one or more reactor coolant pump breakers open.~~ On decreasing power the P-8 reset automatically blocks the ~~above listed trips.~~ single loop low flow trip.

P-9 On increasing power P-9 automatically enables reactor trip on turbine trip. On decreasing power P-9 automatically blocks reactor trip on turbine trip.

P-10 On increasing power P-10 allows the manual block of the Intermediate Range reactor trip and the ~~flow~~ low setpoint Power Range reactor trip; and automatically blocks the Source Range reactor trip and de-energizes the Source Range high voltage power. On decreasing power the Intermediate Range reactor trip and the low setpoint Power Range reactor trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

Bases: Section 2.0, pages B 2-6, 2-7, 2-8

Justification:

Changes were made to more clearly reflect the SNUPPS plant design.

W

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} >200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% delta k/k for four loop operation.

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

ACTION:

With the SHUTDOWN MARGIN less than 1.3% delta k/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 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 1.3% delta k/k:

~~a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(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 control bank withdrawal is within the limits of Specification 3.1.3.6;~~

a ~~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 control rod position is within the limits of Specification 3.1.3.6;

~~d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and~~

*See Special Test Exception 3.10.1.

Keff less than 1.0

SURVEILLANCE REQUIREMENTS (Continued)

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

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

X

~~4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% 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 Effective Full Power Days after each fuel loading.~~

Specification: 3/4.1.1.1

Justification:

This specification relates to shutdown margin/rod insertion limits and is modified for the same reason as in definition 1.28.

This specification has been modified such that there is no longer a shutdown margin requirement in Modes 1 and 2 critical. Verification of shutdown margin will be accomplished by compliance with rod insertion limits. This change eliminates the conflict between shutdown margin and control rod insertion limit specifications. Removal of Mode 1 and 2 requirements also required modification of surveillance requirements. All Mode 1 and 2 (critical) surveillance has been either removed to other locations or eliminated as appropriate.

The requirement to adjust shutdown margin for stuck rods has been eliminated since the new shutdown margin definition does not allow credit to be taken for stuck rods, hence, the operation to meet the new definition will have to correct shutdown margin. A separate surveillance action is therefore unnecessary.

Related changes occur in specifications: 1.28, 3/4.1.1.2, 3/4.1.3.1, 3/4.1.3.6, 3/4.10.1, 3/4.10.2, 3/4.10.3.

W

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% delta k/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 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 1.0% delta k/k:

~~a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and~~

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

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

Specification: 3/4.1.1.2

Justification:

This specification relates to shutdown margin/rod insertion limits and is modified for the same reason as in definition 1.28.

The requirement to revise the shutdown margin for stuck rods has been eliminated for reasons previously discussed. An additional consideration is that in Mode 5 the required Keff is .99 which is equivalent to a 1.0 percent shutdown margin. There is no situation then in which credit will be taken for withdrawn rods to meet shutdown margin requirements, hence, no need for a requirement to consider the impact of a stuck rod.

Related changes occur in specifications: 1.28, 3/4.1.1.1, 3/4.1.3.1, 3/4.1.3.6, 3/4.10.1, 3/4.10.2, 3/4.10.3.

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

a. A flow path from the Boric Acid Storage System via a boric acid transfer pump and a Centrifugal charging pump to the Reactor Coolant System if the Boric Acid Storage System in Specification 3.1.2.5a. is OPERABLE, or

b. The flow path from the refueling water storage tank via a Centrifugal charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b. is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

~~a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path is greater than or equal to 65°F when a flow path from the Boric Acid Storage System is used, and~~

a. ~~b.~~ 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.

Specification: 3.1.2.1.a, b and 4.1.2.1.a, b

Justification:

The word "centrifugal" was inserted to ensure it is understood the positive displacement pump is non-safety related, i.e., it has no emergency power supply, and SNUPPS has no intention of testing it in the pump and valve program. Since the pump will not be tested per Section II of the ASME code, it would be impossible to determine it operable.

The SNUPPS design does not have heat tracing on the flow path from the boric acid storage system to the charging pump suction. Thus 4.1.2.1.a is non-applicable.

REACTIVITY CONTROL SYSTEMS

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. The flow path from the Boric Acid Storage System via a boric acid transfer pump and a centrifugal charging pump to the Reactor Coolant System, and

b. Two flow paths from the refueling water storage tank via centrifugal charging pumps 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 1% delta k/k at 200°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 At least two of the above required flow paths shall be demonstrated OPERABLE:

~~a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is greater than or equal to 65°F when it is a required water source;~~

a ~~b.~~ 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 ~~c.~~ At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal. ~~and~~

~~d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the Reactor Coolant System.~~

#Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 310°F.

Specification: 3.1.2.2.a, b and 4.1.2.2.a, b, c, d

Justification:

The word "centrifugal" was inserted for the same reason as in Specification 3.1.2.1.a, b.

Specification 4.1.2.2.a was deleted for the same reason as in Specification 4.1.2.1.a.

Specification 4.1.2.2.d was deleted as explained below:

> The surveillance requires verification that the flow path required by Specification 3.1.2.2.a delivers at least 30 gpm to the RCS. This path is from the boric acid storage system via a transfer pump and a charging pump to the RCS. SNUPPS believes it is unnecessary to verify this path at a specificed frequency because the RCS is borated during normal operation using this path. The SNUPPS control board design incorporates a boric acid flow recorder which would immediately tell the operator during normal operation that the makeup/ boric acid addition was not correct. Corrective action would be to verify the flow path and the proper operation of the recorder as a minimum.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One centrifugal charging pump in the boron injection flow path required by 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 centrifugal charging 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

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, ~~a differential pressure across the~~ the pump devel ops a discharge pressure ~~pump of~~ greater than or equal to 2390 psig ~~is developed~~ when tested pursuant to Specification 4.0.5.

4.1.2.3.2 ~~All~~ Both centrifugal charging pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable** by verifying that the motor circuit breakers ~~are secured in the open position~~ have been removed from their electrical power supply circuits at least once per 31 days, except when the reactor vessel head is removed.

** An inoperable pump may be energized for testing per Specification 4.0.5 provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

Specification: 3.1.2.3 and 4.1.2.3.2

Justification:

The word "centrifugal" was inserted for the same reason as in Specification 3.1.2.1.a, b; 3.1.2.2.a, b.

Specification 4.1.2.3.2 was rephrased to clarify how SNUPPS interprets "secured in the open position." SNUPPS will verify removal of the circuit breakers from their electrical power supply circuit as follows:

> 13.8 kV, 4.16 kV, 480V: Ensuring the breakers are racked from the operate position to either the test position or the disconnected position.
>
> Motor Control Center Molded Case Circuit Breakers: Ensuring the breaker is placed in the trip position. For certain of these it may be desirable to administratively control the breaker by using a lock.

Specification 4.1.2.3.2, **note - Specification 4.5.3.2 Page 3/4 5-8 has this note which affords the capability to test trains that are not required to be operable. Without this note it may not be possible to perform surveillances on the inoperable pump. The surveillance requirement and basis for both specifications 4.1.2.3.2 and 4.5.3.2 are identical.

JHR/dck/6b5

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 Both centrifugal ~~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 1% delta k/k at 200°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

4.1.2.4.1 Both centrifugal ~~At least two~~ charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, ~~a differential pressure across each~~ the pump develops a discharge pressure ~~pump of~~ greater than or equal to 2390 psi~~d~~g ~~is developed~~ when tested pursuant to Specification 4.0.5.

~~4.1.2.4.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable by verifying that the motor circuit breakers are secured in the open position at least once per 31 days whenever the temperature of one or more of the RCS cold legs is less than or equal to 310°F by verifying that the motor circuit breakers are secured in the open position.~~ (TBD)

~~#A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 310°F.~~

BORATION SYSTEMS (Continued)

~~The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 310°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.~~

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% delta k/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2713 gallons of 7000-ppm borated water from the boric acid storage tanks or 12,117 gallons of 2000-ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 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 OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

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 rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

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. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°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.

Specification: 3.1.2.4, 4.1.2.4.1, 4.1.2.4.2

Justification:

The word "centrifugal" was inserted for the same reason as in Specifications 3.1.2.1.a, b; 3.1.2.2.a, b; 3.1.2.3.

Delete reference to mode 4 in LCO and in surveillance requirement 4.1.2.4.2. This LCO addresses the operability requirement of two charging pumps which are required in modes 1, 2 and 3. The concern for a mass addition pressure transient occurs in mode 4. There is a specific LCO, 3.5.3 which addresses the requirements of 1 operable charging pump when in mode 4. It also addresses the maximum number of charging pumps allowed to be operable when the temperature in cold leg is below 310° F. Removing reference from Tech. Spec. 3.1.2.4 clarifies LCO and eliminates redundant surveillance requirements between specifications.

Recommend removal from Bases of 3/4 1.2 reference to mass addition pressure transients as it is covered in the bases of 3/4 5.3 and would not be referenced as part of 3/4 1.2.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A Boric Acid Storage System ~~and at least one associated Heat Tracing System~~ with:

   1) A minimum contained borated water volume of 2713 gallons,

   2) Between 7000 and 7700 ppm of boron, and

   3) A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

   1) A minimum contained borated water volume of ~~40,247~~ 53,500 gallons,

   2) A minimum boron concentration of 2000 ppm, and

   3) A minimum solution temperature of 37°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

   1) Verifying the boron concentration of the water,

   2) Verifying the contained borated water volume, and

   3) Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 37°F.

Specification: 3.1.2.5.a

Justification:

The SNUPPS Boric Acid Storage System has no heat tracing. Room heaters in each boric acid storage tank room maintain the temperature higher than $65^{\circ}F$.

B

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 16,142 gallons,

2) Between 7000 and 7700 ppm of boron, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1) A contained borated water volume of AT LEAST 113,476 gallons,

2) Between 2000 and 2100 ppm of boron,

3) A minimum solution temperature of 37°F, and

4) A maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the Boric Acid Storage System inoperable and being used as one of the above required borated water sources, restore the storage system 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 1% delta k/k at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage 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.

E

REV 0

Specification: 3.1.2.6.b.1 - page 3/4 1-12

Justification:

This LCO is applicable to modes 1, 2, 3 and 4 and provides requirements for the minimum borated water volume for reactivity control; however, the subject LCO places an additional requirement on the maximum level of the RWST. Specification of a maximum volume is not required since specification 3.5.5 addresses the same modes and establishes the water volumes for ECCS operation which govern for stored water volumes.

Therefore, only a minimum volume LCO is appropriate.

W

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All ~~full-length (~~shutdown and control~~)~~ rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more ~~full-length~~ rods 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 HOT STANDBY within 6 hours.

b. ~~With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.~~ Add Insert (B)

c. With one ~~full-length~~ rod ~~trippable but inoperable due to causes other than addressed by ACTION a. above, or~~ misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

1. The rod is restored to ~~OPERABLE status~~ within the above alignment requirements, or

2. ~~The rod is declared inoperable and~~ The remainder of the rods in the group with the misaligned rod are aligned to within ± 12 steps of the misaligned rod while maintaining the rod sequence and insertion limits of Figures 3.1-1~~, and 3.1-2.~~ The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

3. The rod is declared inoperable~~, and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied.~~ POWER OPERATION may then continue provided that:

   a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;

   b) The Rod Insertion Limits of Specification 3.1.3.6 are adjusted within 6 hours as required to ensure a SHUTDOWN MARGIN of at least (1.3%) delta k/k when all OPERABLE rods are above that limit and assuming the reactivity associated with the withdrawn worth of the inoperable rod is unavailable for insertion.

*See Special Test E

W

INSERT (B)

b. WITH ONE OR MORE ~~FULL LENGTH~~ RODS TRIPPABLE BUT INOPERABLE DUE TO CAUSES OTHER THAN ADDRESSED BY ACTION a., POWER OPERATION MAY CONTINUE PROVIDED THE INOPERABLE ROD(S) ARE POSITIONED WITHIN ± 12 STEPS (INDICATED POSITION) OF THEIR GROUP STEP COUNTER DEMAND POSITION.

Specification: 3/4.1.3.1

Justification:

This specification relates to shutdown margin/rod insertion limits and is modified for the same reason as in definition 1.28.

Action a. has been revised to remove referenced to the shutdown margin specification since there is no longer a Mode 1 and 2 critical requirement.

Action b. and c. were revised to eliminate redundancy and make the specifications easier to interpret. The new Action b. refers only to inoperable rods that are trippable. The new Action c. refers only to misaligned rods. The present version of these two actions confuses these two cases of "inoperable" rods and detracts from the specification.

Action statement c.3.b has been revised to refer to the rod insertion limit specifications rather than shutdown margin specifications which is appropriate considering there is no longer a Mode 1 or 2 critical shutdown margin requirement.

Related changes occur in specifications: 1.28, 3/4.1.1.1, 3/4.1.1.2, 3/4.1.3.6, 3/4.10.1, 3/4.10.2, 3/4.10.3.

U-2

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS-OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The ~~Shutdown And Control~~ Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the ~~control~~ rod positions within ± 12 steps.*

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one rod position indicator per bank inoperable either:

   1. Determine the position of the nonindicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

   1. Verify that all rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the Demand Position Indication System and the Digital Rod Position Indication System at least once per 4 hours.

* THE DIGITAL ROD POSITION INDICATION SYSTEM DOES NOT INDICATE THE ACTUAL POSITION OF THE SHUTDOWN BANK RODS BETWEEN 18 STEPS WITHDRAWN AND 210 STEPS WITHDRAWN.

Specification: 3.1.3.2

Justification:

SNUPPS has digital rod position indication incapable of indicating the step position of shutdown rods between 19 and 209 steps. The LCO should not require determination of rod position within ± 12 steps when operating within this range.

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEM-SHUTDOWN

LIMITING CONDITION FOR OPERATION

DIGITAL ROD POSITION INDICATION

3.1.3.3 ~~One rod position indicator~~ (excluding demand position indication) shall be OPERABLE and capable of determining the ~~control~~ rod position within ± 12 steps for each ~~shutdown and control~~ rod not fully inserted.**

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

ACTION:

With less than the above required position indicator(s) OPERABLE, immediately open the Reactor Trip System breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required rod position indicator(s) shall be determined to be OPERABLE by performance of an ANALOG CHANNEL OPERATIONAL TEST at least once per 18 months, and following each removal of the reactor vessel head.

*With the Reactor Trip System breakers in the closed position.
#See Special Test Exception 3.10.5.

** THE DIGITAL ROD POSITION INDICATION SYSTEM DOES NOT INDICATE THE ACTUAL POSITION OF THE SHUTDOWN BANK RODS BETWEEN 18 STEPS WITHDRAWN AND 210 STEPS WITHDRAWN.

Specification: 3.1.3.3 and 4.1.3.3

Justification:

The LCO was reworded for the same reason as in Specification 3.1.3.2.

The Analog Channel Operational Test specified in Specification 4.1.3.3 is inappropriate for digital rod position indication testing. Channel Calibration is a better choice because digital rod position indication is calibrated at 18 month intervals and a determination is made if a rod position channel responds properly to known input values.

REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual ~~full-length (shutdown and control)~~ rod drop time from the fully withdrawn position shall be less than or equal to 2.2 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 551°F, and

b. All Reactor Coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

~~b. With the rod drop times within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to either:~~

~~1. Less than or equal to (*)% of RATED THERMAL POWER when the reactor coolant stop valves in the non-operating loop are open, or~~

~~2. Less than or equal to (*)% of RATED THERMAL POWER when the reactor coolant stop valves in the non-operating loop are closed.~~

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of ~~full-length~~ rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per 18 months.

~~*These valves left blank pending NRC approval of three loop operation.~~

Technical Specification 3.1.3.4.b Page 3/4 1-19

Justification:

Delete reference to coolant stop valves. Callaway does not have stop valves in system.

W

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figure~~s~~ 3.1-1~~, and 3.1-2~~.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the above figures, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.6.1 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

4.1.3.6.2 and

4.1.3.6.3 on next page

*See Special Test Exceptions 3.10.2 and 3.10.3.
#With K_{eff} greater than or equal to 1.0.

SURVEILLANCE REQUIREMENTS (Continued)

W

4.1.3.6.2 Prior to initial operation above 5% of RATED THERMAL POWER the SHUTDOWN MARGIN shall be determined to be at least (1.3%) delta k/k with the reactor critical and the control banks at the maximum insertion limit of Specification 3.1.3.6, by consideration of the factors below.

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration,
6. Samarium concentration, and

4.1.3.6.3 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within + 1%Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least factors 1 through 6 stated in Specification 4.1.3.6.2 above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

Specification: 3/4.1.3.6

Justification:

This specification relates to the shutdown margin/rod insertion limits and is modified for the same reason as in definition 1.28.

Two surveillance requirements have been removed from specification 3/4.1.1.1 and placed here. This was necessary since these surveillance requirements are for Modes 1 and 2 critical and the shutdown margin specification is no longer applicable in those modes.

Related changes occur in specifications: 1.28, 3/4.1.1.1, 3/4.1.1.2, 3/4.1.3.1, 3/4.10.1, 3/4.10.2, 3/4.10.3.

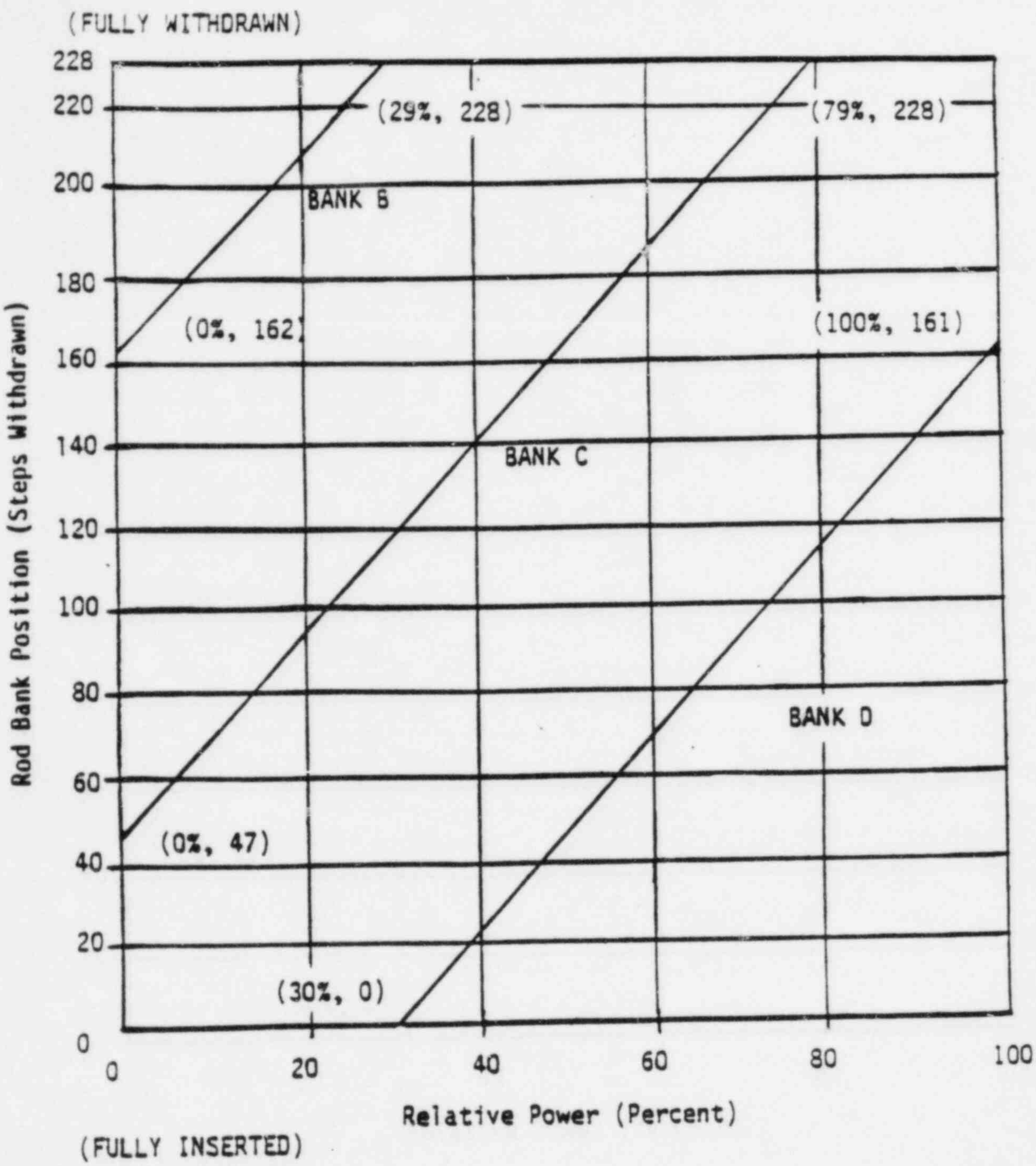

FIGURE 3.1-1 ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER FOUR LOOP OPERATION

Figure: 3.1-1

Justification:

The correct Control Bank Insertion Limits Figure 3.1-1 is attached. This figure has been revised to reflect the mechanical limitation to no less than a 113 step overlap for control banks. The 115 step difference in the insertion limits for the end points of the rods provides a minimum bank overlap of 113 steps.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.32}{P}\right] [K(Z)] \text{ for } P > 0.5\text{, and}$$

$$F_Q(Z) \leq [4.64] [K(Z)] \text{ for } P \leq 0.5.$$

Where:

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}\text{, and}$$

K(Z) = the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

not FQ

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower Delta T Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit; and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

#2

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during POWER OPERATION.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_O is depleated. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

F_Q not F_O

The 2 hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_O is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.0.

F_Q not F_O

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are (C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8). X

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters is maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

Technical Specification 3.2.2 and Bases 3/4.2.4 Pages 3/4 2-4, B 3/4 2-5

Justification:

$F_O(Z)$ should be $F_Q(Z)$

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	10
2. Power Range, Neutron Flux - High Setpoint	4	2	3	1, 2	2#
Low Setpoint	4	2	3	1###, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3*, 4*, 5*	10
c. Shutdown	2	0	1	3, 4, and 5	5
d. Boron Dilution (Doubling)	~~2~~	~~1 (Alarm)~~	~~2~~	~~2, 3, 4, 5~~	~~4#~~ (Later)
7. Overtemperature ΔT					
a. Four Loop Operation	4	2	3	1, 2	6#
b. Three Loop Operation	(**)	(**)	(**)	(**)	(**)

Table 3.3-1

Justification:

The Boron Dilution System and its Technical Specification requirements are under evaluation. A Technical Specification submittal will be made at a later date.

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the Reactor Trip System breakers in the closed position, and the Control Rod Drive System capable of rod withdrawal.

**Values left blank pending NRC approval of three loop operation.

#The provisions of Specification 3.0.4 are not applicable.

##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within ~~1 hour,~~ 6 hours;

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to ~~2 hours~~ 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. If the faulted channel involves loss of the Analog Signal, either ~~Either,~~ THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within ~~4~~ 6 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

Specification 3.3.1 (Table 3.3-1, Table Notation)

Justification:

Action 2c pertains only to the analog signal contribution in determining quadrant power tilt. A fault in the particular bistable driver or transformer which would render that trip function inoperable does not affect the ability of a power range channel to provide power tilt information to the operator. The wording added to the beginning of Action 2c is meant to clarify that this action should only apply to a faulted analog signal.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint; and

b. Above the P-6 (Intermediate Range Neutron Flux interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

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

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within ~~1~~ 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to ~~2~~ 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

INSERT (A)

~~ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.~~

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

INSERT (A) pg 3/4 3-8

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours.

b. An additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1 provided the inoperable channel is in the tripped condition.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to ~~2~~ 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers within the next hour.

ACTION 11 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 2 hour.

Table: 3.1-1 and 4.3-1

Specification: 3/4.3.1

Justification:

These sections were modified based on the specified surveillance intervals and surveillance and maintenance outage times recommended in WCAP-10271 "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System" and supplements to that report. The following are identified difficulties with the current testing and maintenance practices:

1. A significant manpower expenditure to accomplish and document testing.
2. Distraction of the operator and shift supervisor away from normal plant control.
3. A large percentage of time spent with portions of the RPS partially inoperable.
4. A large percentage of time spent in a partial trip condition.
5. Subjecting the operator to frequent false alarms and indication.

These changes to RPS surveillance requirements are beneficial to plant safety for the following reasons:

1. A reduction in the number of unnecessary plant transients and challanges to the protection systems.

2. A potential increase in equipment reliability with an associated decrease in equipment random failure probabilities which results in a factor of 3 to 5 increase in RPS availability.

3. Performance of testing and maintenance in a bypass condition for shorter total time which contributes to the increase in RPS availability.

4. An improvement in plant availability.

5. A potential decrease in testing and maintenance errors.

6. More effective use of the operating staff with the capability to redirect significant amounts of manpower to non-surveillance matters.

Changes in these sections are consistent with the modifications to the Westinghouse Standard Tech Specs, Rev. 4 provided in Appendix A of WCAP-10271.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	≤ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 second*
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range	
a. Neutron Flux	N.A.
b. Neutron Flux, Boron Dilution (Doubling)	~~≤ 30**~~ (Later) X
7. Overtemperature ΔT	≤ 3.0 seconds*
8. Overpower ΔT	N.A.
9. Pressurizer Pressure--Low	≤ 2.0 seconds
10. Pressurizer Pressure--High	≤ 2.0 seconds
11. Pressurizer Water Level--High	N.A.

~~*Diesel generator starting and sequence loading delay not included. Offsite power available. Response time limit includes actuation of valves to establish a boration path from the RWST and isolation of CVCS clean water flow paths.~~

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

Table 3.3-2

Justification:

The Boron Dilution System and its Technical Specification requirements are under evaluation. A Technical Specification submittal will be made at a later date.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3*, 4*,
2. Power Range, Neutron Flux High Setpoint	S(9)	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	~~M~~ Q	N.A.	N.A.	1, 2
Low Setpoint	S(9)	R(4)	~~M(9)~~ Q	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	~~M~~ Q	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	~~M~~ Q	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S(9)	R(4, 5)	~~S/U(1),M(9)~~ Q	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	S(9)	R(4, 5)	~~S/U(1),M(9)~~ Q(9)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	S	R	~~M~~ Q	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	~~M~~ Q	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	~~M~~ Q	N.A.	N.A.	1
10. Pressurizer Pressure--High	S	R	~~M~~ Q	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	S	R	~~M~~ Q	N.A.	N.A.	1
12. Low Reactor Coolant Flow	S	R	~~M~~ Q	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level--Low-Low	S	R	~~M~~ Q	N.A.	N.A.	1, 2
14. General Warning Alarm	N.A.	N.A.	N.A.	R ~~N.A.~~ NA	N.A.	1, 2
15. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	~~M~~ Q	N.A.	1
16. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	~~M~~ Q	N.A.	1
17. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
19. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	~~M~~ Q	N.A.	N.A.	2##
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	~~M~~ Q (8)	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	~~M~~ Q (8)	N.A.	N.A.	1
d. Power Range Neutron Flux, P-9	N.A.	R(4)	Q M(8)	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
e. Power Range Neutron Flux, P-10	N.A.	R(4)	~~M~~ Q (8)	N.A.	N.A.	1, 2
f. Turbine Impulse Chamber Pressure, P-13	N.A.	R	~~M~~ Q (8)	N.A.	N.A.	1
20. Reactor Trip Breaker	N.A.	N.A.	N.A.	~~M~~ SA (7, 11)	N.A.	1, 2, 3*, 4*, 5*
21. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	~~M~~ SA (7)	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With the Reactor Trip System breakers closed and the Control Rod Drive System capable of rod withdrawal.

- Below P-6 (Intermediate Range Neutron Flux interlock) Setpoint.

- Below P-10 (Low Setpoint Power Range Neutron Flux interlock) Setpoint.

(1) - If not performed in previous ~~7~~ 92 days.

(2) - Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) - Single point comparison of incore to excore axial flux difference above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

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

(5) - Detector plateau curves shall be obtained, evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) - Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) - Each train shall be tested at least every ~~62~~ 184 days on a STAGGERED TEST BASIS.

(8) - With power greater than or equal to the interlock setpoint the required OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.

(9) - Monthly surveillance in MODES 3*, 4* and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.

(10) - Setpoint verification is not required.

(11) - ~~At least once per 18 months and following maintenance or adjustment of the Reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the Undervoltage and Shunt trips.~~ (LATER). PENDING FURTHER WESTINGHOUSE EVALUATION

Table: 3.1-1 and 4.3-1

Specification: 3/4.3.1

Justification:

These sections were modified based on the specified surveillance intervals and surveillance and maintenance outage times recommended in WCAP-10271 "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System" and supplements to that report. The following are identified difficulties with the current testing and maintenance practices:

1. A significant manpower expenditure to accomplish and document testing.
2. Distraction of the operator and shift supervisor away from normal plant control.
3. A large percentage of time spent with portions of the RPS partially inoperable.
4. A large percentage of time spent in a partial trip condition.
5. Subjecting the operator to frequent false alarms and indication.

These changes to RPS surveillance requirements are beneficial to plant safety for the following reasons:

1. A reduction in the number of unnecessary plant transients and challanges to the protection systems.

2. A potential increase in equipment reliability with an associated decrease in equipment random failure probabilities which results in a factor of 3 to 5 increase in RPS availability.

3. Performance of testing and maintenance in a bypass condition for shorter total time which contributes to the increase in RPS availability.

4. An improvement in plant availability.

5. A potential decrease in testing and maintenance errors.

6. More effective use of the operating staff with the capability to redirect significant amounts of manpower to non-surveillance matters.

Changes in these sections are consistent with the modifications to the Westinghouse Standard Tech Specs, Rev. 4 provided in Appendix A of WCAP-10271.

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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 interlocks 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 or Interlock Setpoint trip less conservative than the value shown in the Trip Setpoint column of Table 3.3-4 adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, place the channel in the tripped condition within ~~1~~ 2 hour, and within the following 12 hours either:

1. Determine that Equation 2.2-1 was satisfied for the affected channel and adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-4, or

2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3.3 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $\qquad Z + R + S \leq TA$

Where:

Z = the value from column Z of Table 3.3-4 for the affected channel,

R = the "as measured" value (in percent span) of rack error for the affected channel,

S = either the "as measured" value (in percent span) of the sensor error, or the value in column S of Table 3.3-4 for the affected channel, and

TA = the value from column TA of Table 3.3-4 for the affected channel.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by the performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 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 train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per 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.

<u>Specification</u>: 3.3.2.b

<u>Justification</u>:

The time limit should be increased to 2 hours for the same reason as in 2.2.1, page 2-4 and Table 3.3-1 of Specification 3.3.1, pages 3/4 3-7 through 3/4 3-9.

TABLE 3.3-3 (Continued)

TABLE NOTATION

#Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure interlock) Setpoint.

##Trip function may be bypassed in this MODE above the P-11 (Pressurizer - Pressure interlock) Setpoint.

*The provisions of Specification 3.0.4 are not applicable.

**One in separation group 1 and one in separation group 2.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within ~~1~~ 2 hours.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the Containment purge supply and exhaust valves are maintained closed.

ACTION STATEMENTS (Continued)

ACTION 18 - With the number of OPERABLE channels one less than 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 in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within ~~1~~ 2 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable ~~one additional~~ channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 20 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 21 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, 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 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 22 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 at least HOT SHUTDOWN within the following 6 hours.

Specification: 3.3.2 (Table 3.3-3 Action Statements)

Justification:

The time limits for placing inoperable channels in the tripped condition was changed for the same reason as in Table 3.3-1.

Technical Specification: Table 3.3-3 Action 19.b Page 3/4 3-26

Justification:

Revise technical specification action statement 19 to allow bypassing the inoperable channel. This allows surveillance to continue while satisfying the minimum channels operable requirements.

SPECIFICATION 3/4 3.2 USES AS REFERENCE

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Containment Isolation	
Phase "A" Isolation	N.A.
Phase "B" Isolation	N.A.
Containment Purge Isolation	N.A.
d. Steam Line Isolation	N.A.
e. Feedwater Isolation	N.A.
f. Auxiliary Feedwater	N.A.
g. Essential Service Water	N.A.
h Containment Cooling Fans	N.A.
i. Control Room Isolation	N.A.
j. Reactor Trip	N.A.
k. Start Diesel GeneratorS	N.A.
2. Containment Pressure-High-1	
a. Safety Injection (ECCS)	$\leq 27.0^{(1)}/12^{(5)}$
~~b.~~ 1) Reactor Trip (from SI)	≤ 2.0
~~c.~~ 2) Feedwater Isolation	$\leq 7.0^{(3)}$
~~d.~~ 3) Containment Isolation-Phase "A"	$\leq$ 1.5 (7)
~~e.~~ 4) Containment Purge Isolation	≤ 4.5
~~f.~~ 5) MOTOR DRIVEN Auxiliary Feedwater PUMPS	≤ 60.0
~~g.~~ 6) Essential Service Water	$\leq 60.0^{(1)}$
~~h.~~ 7) Containment Cooling Fans	$\leq 60.0^{(1)}$
~~i.~~ 8) Control Room Isolation	N.A.
~~j.~~ 9) Start Diesel Generators	$\leq$ 11.5 (8)

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq 27.0^{(1)}/12.0^{(5)}$
~~b~~. 1. Reactor Trip (from SI)	≤ 2.0
~~c~~. 2. Feedwater Isolation	$\leq 7.0^{(3)}$
~~d~~. 3. Containment Isolation-Phase "A"	≤ 2.0 (7)
~~e~~. 4. Containment Purge Isolation	≤ 4.5
~~f~~. 5. MOTOR DRIVEN Auxiliary Feedwater Pumps	≤ 60.0
~~g~~. 6. Essential Service Water	$\leq 60.0^{(1)}$
~~h~~. 7. Containment Cooling Fans	$\leq 60.0^{(1)}$
~~i~~. 8. Control Room Isolation	N.A.
~~j~~. 9. Start Diesel Generators	≤ 12.0 (8)
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 22.0^{(4)}/12.0^{(5)}$
~~b~~. 1. Reactor Trip (from SI)	≤ 2.0
~~c~~. 2. Feedwater Isolation	$\leq 7.0^{(3)}$
~~d~~. 3. Containment Isolation-Phase "A"	≤ 2.0 (7)
~~e~~. 4. Containment Purge Isolation	≤ 5.0
~~f~~. 5. MOTOR DRIVEN Auxiliary Feedwater Pumps	≤ 60.0
~~g~~. 6. Essential Service Water	$\leq 60.0^{(1)}$
~~h~~. 7. Containment Cooling Fans	$\leq 60.0^{(1)}$
~~i~~. 8. Control Room Isolation	N.A.
~~j~~. 9. Steam Line Isolation	$\geq 1.5 \leq 5.0$
~~k~~. 10. Start Diesel Generators	≤ 12.0 (8)

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
5. Containment Pressure--High-3	
a. Containment Spray	$\leq$ 60.0(1)/50.0(2)/30.0(6)
b. Containment Isolation-Phase "B"	$\leq$ 31.5
6. Steam Generator Water Level--High-High	
a. Turbine Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 7.0(3)
7. Steam Generator Water Level - Low-Low	
a. Start Motor-Driven AUXILIARY FEEDWATER Pumps	$\leq$ 60.0
b. Start Turbine-Driven AUXILIARY FEEDWATER Pumps	$\leq$ 60.0
8. Containment Pressure--High-2	
a. Steam Line Isolation	$\leq$ 7.0
9. RWST Level-Low-Low-1 Coincident with Safety Injection	
a. Automatic Switchover to Containment Sump	$\leq$ 60.0
10. Station Blackout	
a. Start Turbine-Driven AUXILIARY FEEDWATER Pump	N.A.
11. Trip Main Feedwater Pumps	
a. Start Motor-Driven AUXILIARY FEEDWATER ~~and Turbine-Driven Pumps~~ PUMPS	N.A.

K

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
12. Loss of Power	
a. 4 kV Bus Undervoltage-Loss of Voltage	≤ 10.5
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	≤ 14.0/44.0
13. Steam Line Pressure-Negative Rate-High	
a. Steam Line Isolation	N.A.
~~14. Containment Isolation-Phase "A"~~	
~~a. Containment mini Purge Isolation~~	~~≤ 5.0~~
15. Auxiliary Feedwater Pump Suction Pressure-Low	
a. Transfer to Essential Service Water	N.A.

X

TABLE 3.3-5 (Continued)

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting and sequence loading delay not included. Offsite power available.

(3) Air operated valves.

(4) Diesel generator starting and sequence loading delay included. RHR pumps not included.

(5) Diesel generator starting and sequence loading delays not included. Offsite power available. RHR pumps not included.

(6) Sequence delays not included.

(7) DOES NOT INCLUDE VALVE CLOSURE TIME

(8) Includes time for Desel to reach full speed

Specification 3/4 3.2 Table 3.3-5 Page 3/4 3-35

Justification:

Revise table as indicated on attached copy. To revise tables to reflect logic and sequence as they occur in ESF system. Basically 4 signals initiate Safety Injections-

1. Low Pressurizer Pressure
2. Containment Pressure High - 1
3. Steam Line Pressure - Low
4. Manual actuation

Once Safety Injection is initiated, the Function as revised occur. Proposed change reflects actions as they occur in system. In some cases parallel signals are sent to same function for example, containment Pressure High - 1 initiates Safety Injection as well as initiating Containment Isolation - Phase "A".

Item 11 - Trip of Main Feed Water Pumps starts motor driven pumps only - not Turbine Driven Pump.

Item 14 - deleted, already covered in Safety Injection portion where Containment Isolation occurs subsequent to SI, & Phase A Isolation.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection surveillance requirements							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure--High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Containment Purge Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	R	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Isolation-Phase "A"	See Item 3.a. above for all Containment Isolation Phase "A" surveillance requirements.							

Table 4.3-2 Page 3/4 3-41

Justification

SNUPPS believes the requirements for the Trip Actuating Device Operational Test and the Actuation Logic Test for item 3.c. have been reversed.

INSTRUMENTATION

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.

restore the instrument(s) to an OPERABLE status within 30 days or, in lieu of any other reporting requirements,

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 ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-4.

INSERT-

~~4.3.3.3.2 Each of the above seismic monitoring instruments actuated during a seismic event greater than or equal to 0.01 g shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 5 days following the seismic event. 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.~~ 10

with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Structural Engineering Branch, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555,

Specification 3.3.3.3 Action a Page 3/4 3-52

Justification -

The proposed change will eliminate the submission of nuisance and duplicate reports for failure of systems with no immediate consequences to safety.

Technical Specification: 4.3.3.3.2 (Revise to read)

4.3.3.3.2 Each of the above seismic monitoring instruments actuated during a seismic event greater than or equal to 0.01 g shall:

a. be restored to OPERABLE status within 24 hours, and

b. a CHANNEL CALIBRATION performed within 10 days following the seismic event, and

c. data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion.

4.3.3.3.3 Following a seismic event greater than or equal to .01 g, a special report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Structural Engineering Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, within 10 days describing the magnitude, frequency spectrum and resultant effect upon the facility features important to safety.

Justification:

Proposed change is designed to clarify a surveillance with requirements which will impact several different plant sections. Dividing the surveillance also lends itself to the establishment of the surveillance matrix and tracking responsibility of same. This change will also reduce the posibility of a missed surveillance.

U-2

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Triaxial Peak Recording Accelerographs			
a. Radwaste Base Slab	N.A.	R	~~SA~~ N.A.
b. Control Room	N.A.	R	~~SA~~ N.A.
c. ESW Pump Facility	N.A.	R	~~SA~~ N.A.
d. Ctmt Structure	N.A.	R	~~SA~~ N.A.
e. Auxiliary Bldg. SI Pump Suction	N.A.	R	~~SA~~ N.A.
f. SGB Piping	N.A.	R	~~SA~~ N.A.
g. SGB Support	N.A.	R	~~SA~~ N.A.
2. Triaxial Time History and Response Spectrum Recording System, Monitoring the Following Accelerometers (Active)			
a. Ctmt. Base Slab	M	R	SA
b. Ctmt. Oper. Floor	M	R	SA
c. Reactor Support	M	R	SA
d. Aux. Bldg. Base Slab	M	R	SA
e. Aux. Bldg. Control Room Air Filters	M	R	SA
f. Free Field	M	R	SA
3. Triaxial Response-Spectrum Recorder (Passive)			
a. Ctmt. Base Slab	N.A.	R	SA
4. Triaxial Seismic Switches			
a. OBE Ctmt. Base Slab	M	R	SA
b. SSE Ctmt. Base Slab	M	R	SA
c. OBE Ctmt. Oper. Fl.	M	R	SA
d. SSE Ctmt. Oper. Fl.	M	R	SA
e. System Trigger	M	R	SA

Technical Specification: Table 4.3-4

Justification:

Delete the semi-annual Analog Channel Operational Test surveillance requirement. These instruments are self contained and measure peak amplitudes from seismic events on a magnetic tape. The channel calibration is the only surveillance which can be performed on this equipment.

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.

restore the inoperable channel(s) to OPERABLE status within 7 days, or in lieu of any other reporting requirements

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.

Specification 3.3.3.4 Action a Page 3/4 3-55

Justification -

The proposed change will eliminate the submission of nuisance LER's and the potential for duplicate reports for failures of equipment or systems with no consequences to safety.

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. RCS Pressure - Wide Range	M	R
2. Reactor Coolant Temperature - Cold Leg	M	R
3. Source Range Nuclear Flux	M	~~N.A.~~ R
4. Reactor Trip Breaker Indication	M	N.A.
5. Reactor Coolant Temperature - ~~Average~~ Hot Leg	M	R
6. Reactor Coolant Pump Breakers	N.A.	N.A.
7. Pressurizer Pressure	M	R
8. Pressurizer Level	M	R
9. Steam Generator Pressure	M	R
10. Steam Generator Level	M	R
11. Auxiliary Feedwater Flow Rate	M	R

K-1

Specification: 3.3.3.5 (Table 4.3-6)

Justification:

The Channel Calibration was changed from "NA" to "R". SNUPPS believes the "NA" is a typographical error.

Instrument #5 was changed to "Hot Leg" vice "Average" because SNUPPS Auxiliary Shutdown Panel has only "Hot Leg" and "Cold Leg" RCS temperatures.

W.C. ONLY

CHLORINE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.3.3.7 Two independent Chlorine Detection Systems, with their Alarm/Trip Setpoints adjusted to actuate at a chlorine concentration of less than or equal to 5 ppm, shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION:

a. With one Chlorine Detection System inoperable, restore the inoperable system to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the Control Room Emergency Ventilation System in the recirculation mode of operation.

b. With both Chlorine Detection Systems inoperable, within 1 hour initiate and maintain operation of the Control Room Emergency Ventilation System in the recirculation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each Chlorine Detection System shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and a CHANNEL CALIBRATION at least once per 18 months.

4.3.3.7.2 In addition, at least once per 18 months Control Room Ventilation Isolation shall be verified to occur on a high chlorine concentration.

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-~~11~~ 9 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

SEE INSERT

a. ~~With any, but not more than one-half the total in any fire zone, Function A fire detection instruments shown in Table 3.3-11 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within the next 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours (or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.6).~~

b. ~~With more than one-half of the Function A fire detection instruments in any fire zone shown in Table 3.3-11 inoperable, or with any Function B fire detection instruments shown in Table 3.3-11 inoperable, or with any two or more adjacent fire detection instruments shown in Table 3.3-11 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours (or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.6).~~

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

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.7.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

~~4.3.3.7.3 The nonsupervised circuits, associated with detector alarms, between the instrument and the control room shall be demonstrated OPERABLE at least once per 31 days.~~

INSERT

ACTION:

a. If one of the following conditions exists, restore the inoperable fire detection instruments(s) to OPERABLE status within 14 days or, within 24 hours, establish a fire watch patrol to inspect the area(s) containing the inoperable instrument(s) at least once per 8 hours. For instruments located inside Containment, it is acceptable to monitor Containment air temperature at least once per hour at the locations listed in Specification 4.6.1.6 in lieu of establishing a fire watch.

   (1) One or more, but not more than one-half the total in any fire area, Function A instruments listed in Table 3.3-9 inoperable.
   (2) One or more, but not more than one-half the total in any fire area, Function B instruments not associated with a Halon-protected area and which are listed in Table 3.3-9 inoperable.
   (3) One or more instruments in one of two redundant Function B zone detection circuits for a Halon-protected area listed in Table 3.3-9 inoperable.

b. If one of the following conditions exists, within one hour establish a fire watch patrol to inspect the area(s) with the inoperable instrument(s) at least once per hour. For instruments located inside Containment, it is acceptable to inspect the affected area at least once per 8 hours or to monitor Containment air temperature at least once per hour at the locations listed in Specification 4.6.1.6 in lieu of establishing a fire watch.

   (1) More than one-half of the detection instruments in any fire area which is not protected by Halon and is listed in Table 3.3-9 inoperable.
   (2) Two or more adjacent detection instruments in any fire area which is not protected by Halon and is listed in Table 3.3-9 inoperable.
   (3) One or more instruments in both redundant zone detection circuits for a Halon-protected area listed in Table 3.3-9 inoperable.

Q-1

Specification 3.3.3.7, Page 3/4 3-64

Justification -

It is not necessary to enter Action Statement b on the loss of any single Function B fire detection instrument. The Function B instruments listed in Table 3.3-9 are associated with either a Halon fire suppression system or a pre-action sprinkler system.

Each Halon-protected area listed in Table 3.3-9 is cross-zoned which means the area contains two parallel and redundant detection circuits. If both detection circuits are operable, one or more detectors in each of the two detection circuits must be in alarm to initiate Halon release. If one detection circuit is inoperable, the systems are designed to initiate Halon release if one or more detectors in the remaining operable detection circuit are in alarm. The latter situation is the more conservative of the two in that only one detector must be in alarm to initiate a Halon release rather than one detector in each zone. For this reason, it is acceptable to have one of two redundant detection circuits for a Halon-protected area inoperable without requiring an hourly fire watch patrol of the area.

The pre-action systems located in the Control, Reactor and Auxiliary Buildings are designed to extinguish fires occurring in concentrations of cable trays. Cable fires are extremely smoky but produce relatively little heat. The fire detection zones which cover these areas consist almost exclusively of smoke detectors which are tightly spaced over the protected cable trays. A fire detection signal from any detector is required to charge the pre-action system piping with water, however, the fire must still grow large enough to fuse a sprinkler head before the system actually performs its function. Given the tight spacing of detectors and the characteristics of cable fires, the loss of up to half of the detectors in a zone will not significantly impair the ability of the fire detection system to function before the fire grows sufficiently large to open a sprinkler head.

All pre-action sprinkler piping is charged with a supervisory air pressure. Loss of this air pressure will produce a trouble alarm in the Control Room for the sprinkler system in question. Thus if a fire ever actuates a sprinkler head in a pre-action system before the detection system has functioned to open the control valve, an alarm will still be received in the Control Room. The operator dispatched to investigate the alarm may then manually operate the system control valve and water will be discharged through all opened sprinklers.

Each diesel generator room is equipped with a pre-action sprinkler system. The detection zone for each room consists of eight (8) heat detectors and four (4) infra-red flame detectors. A signal from any heat detector will open the system control valve. A fire involving either the lube oil or fuel oil from a diesel generator will develop very rapidly and produce large quantities of heat and smoke. The detectors are installed on a tighter spacing than for which they are listed. The conditions in the diesel generator rooms (smooth continuous ceilings) match the conditions under which heat detectors are tested to determine their listed spacing. Underwriters

Laboratories (UL) has rated these detectors to be spaced at intervals of 50 feet. This spacing can still be met with one-half of the heat detectors inoperable. The pre-action system installed in the Fuel Building is located in the railroad bay. The railroad bay is open to the remainder of the building. As a result, smoke from a fire in the railroad bay will alarm smoke detectors located in other zones of the Fuel Building. The pre-action system is equipped with heat detectors located on the ceiling of the railroad bay. A fire occurring in the railroad bay will have to develop into a relatively large size before it would have sufficient buoyancy to drive heated gases to the ceiling, a distance of 47 feet, to actuate a heat detector and trip the system control valve automatically.

In the meanwhile, smoke will migrate through the Fuel Building because of its open construction and the smoke detectors will alarm long in advance of the heat detectors on the ceiling of the railroad bay. It can be expected that operators will respond and be able to manually actuate the system control valve before the heat detectors respond automatically regardless of the number of operable heat detectors.

The discussion above demonstrates that there in no need to differentiate between pre-action detection zones and Function A fire detection zones because of the nature of the protected hazards and the suitability of the types of detection provided.

The requirement in Action Statement a to initiate an hourly fire watch patrol within one hour after a detection instrument has been inoperable for 14 days was modified to require a once-per-shift fire watch patrol to be initiated within 24 hours after the 14 days elapse. The conditions under which Action Statement a would be entered are not severe enough to justify an hourly fire watch. Since no two adjacent detectors in an area can be inoperable while under Action Statement a, the capability to detect a fire in the area will not be adversely affected. A once-per-shift fire watch patrol is more than adequate under these conditions.

The incorporation of the changes noted above into the Action Statements as written would have rendered them so wordy and complex that the requirements could have been misunderstood or misinterpreted. The Action Statements were rewritten to remedy this. The content of the Action Statements have not been changed except as noted above.

Specification 4.3.3.7.3, Page 3/4 3-64

Justification -

All fire detection circuits installed at Callaway are supervised therefore this requirement may be eliminated.

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	ZONE	TOTAL NUMBER OF INSTRUMENTS (SEE NOTE 1) HEAT	FLAME	SMOKE	COMMENTS
1101-Aux. Bldg. 1974' Gen. Flr. #1	100			(0/11)	
1102-Chiller & Surge Tks. Area	100			(0/4)	
1102-Chiller & Surge Tks. Area	101			(2/0)	
1107-Cent. Charg. Pmp. Rm. B	101			(2/0)	
1108-Safety Inj. Pmp. Rm. B	101			(2/0)	
1109-Res. Ht. Remov. Pmp. Rm. B	101			(1/0)	
1110-Ctmt. Spray Pmp. Rm. B	101			(1/0)	
1111-Res. Ht. Remov. Pmp. Rm. A	101			(1/0)	
1112-Ctmt. Spray Pmp. Rm. A	101			(1/0)	
1113-Safety Inj. Pmp. Rm. A	101			(2/0)	
1114-Cent. Charg. Pmp. Rm. A	101			(2/0)	
1115-Pos. Disp. Charg. Pmp. Rm.	101			(2/0)	
1116, 1117-Boric Acid Tk. Rms.	101			(2/0)	
1116, 1117-Boric Acid Tk. Rms.	101		(2/0)		
1120-Aux. Bldg. 1974' Gen. Flr. #2	101			(4/0)	
1122-Aux. Bldg. 1974' Gen. Flr. #3	100			(0/3)	
1122-Aux. Bldg. 1974' Gen. Flr. #3	101			(5/0)	
1126-Boron Inj. Tk. & Pmp. Rm.	101			(1/0)	
1127-Stair A-Z	109			(1/0)	
1128-	117			(2/0)	
1130-Aux. Bldg. 1974' N. Corr.	100			(0/2)	
1206-W. Pipe Chase Below AFWP Area	117			(2/0)	
-Aux. Bldg. Elec. Chase S. 1988'	117			(1/0)	
1301-Aux. Bldg. 2000' Corridor #1	103			(0/10)	
1301-Aux. Bldg. 2000' Corridor #1	117			(2/0)	
1311-Aux. Bldg. Sampling Rm.	117			(2/0)	
1312-Boron Meter/RC Activity Mon. Rm.	103			(0/1)	
1314-Aux. Bldg. 2000' Corridor #3	103			(0/3)	
1314-Aux. Bldg. 2000' Corridor #3	117			(2/0)	
1315-Ctmt. Spray Add. Tk. Area	103			(0/2)	
1316-Vlv. Rm. by Seal Wtr. Ht. Exch.	103			(0/1)	
1320-Aux. Bldg. 2000' Corridor #4	103			(0/3)	

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INSTRUMENT LOCATION	ZONE	TOTAL NUMBER OF INSTRUMENTS (SEE NOTE 1) HEAT	FLAME	SMOKE	COMMENTS
1321-Aux. Bldg. 2000' S. Exit Vest.	103			(0/1)	
1322-Pipe Pene. Rm. B	117			(5/0)	
1323-Pipe Pene. Rm. A	117			(6/0)	
1325-Aux. FW Pmp. Rm. B	117			(2/0)	
1326-Aux. FW Pmp. Rm. A	117			(2/0)	
1331-Aux. FW Pmp. Rm. C	111	(2/0)			
1331-Aux. FW Pmp. Rm. C	117			(1/0)	
1335-Aux. Bldg. Elec. Chase N. 2000'	117			(1/0)	
1336-Aux. Bldg. Elec. Chase S. 2000'	117			(1/0)	
1401-Comp. Cool. Pmp. & Ht. Exch. B	118			(5/0)	
1402-Aux. Bldg. 2026' Corridor #1	104			(0/6)	
1403-MG Set Rm.	105			(0/9)	See Note 2
1403-MG Set Rm.	112			(0/9)	See Note 2
1405-Chemical Stg. Area	118			(6/0)	
1406-Comp. Cool. Pmp. & Ht. Exch. A	104			(0/1)	
1406-Comp. Cool. Pmp. & Ht. Exch. A	118			(2/0)	
1408-Aux. Bldg. 2026' Corridor #2	104			(0/9)	
1408-Aux. Bldg. 2026' Corridor #2	118			(5/0)	
1409-Elec. Pene. Rm. B	106			(0/4)	See Note 2
1409-Elec. Pene. Rm. B	113			(0/4)	See Note 2
1410-Elec. Pene. Rm. A	107			(0/8)	See Note 2
1410-Elec. Pene. Rm. A	114			(0/8)	See Note 2
1413-Aux. Shutdown Pnl. Rm.	118			(4/0)	
1501-Ctrl. Rm. A/C & Filt. Units B	110			(10/0)	
1504-Ctmt. Purge Exh. & Mech. Equip. B	108			(18/0)	
1506-Ctmt. Purge Sup. AHU Rm. A	109			(18/0)	
1507-Personnel Hatch Area	108			(3/0)	
1512-Ctrl. Rm. A/C & Filt. Units A	110			(10/0)	
1513-Ctrl. Bldg. Vent Sup. A/C Unit Rm.	109			(3/0)	
Aux. Bldg. Duct 2047'6"	119			(1/0)	
Containment	201	(1/0)			See Note 3
Containment	202	(2/0)			See Note 3
Containment	203	(1/0)			See Note 3
Containment	204	(1/0)			See Note 3

INSTRUMENT LOCATION	ZONE	TOTAL NUMBER OF INSTRUMENTS (SEE NOTE 1) HEAT	FLAME	SMOKE	COMMENTS
Containment	206	(3/0)			See Note 3
Containment	215	(1/0)			See Note 3
Containment	216	(1/0)			See Note 3
Containment	217	(1/0)			See Note 3
Containment	218	(1/0)			See Note 3
Containment	219			(4/0)	
Containment	220	(1/0)			See Note 3
3101-Ctrl. Bldg. 1974' Pipe Space	300			(11/0)	
3105-Ctrl. Bldg. Elec. Chase S. 1974'	300			(1/0)	
3106-Ctrl. Bldg. Elec. Chase N. 1974'	300			(1/0)	
-Area Above Access Control	301			(12/0)	
3229-Ctrl. Bldg. Elec. Chase S. 1984'	300			(1/0)	
3230-Ctrl. Bldg. Elec. Chase N. 1984'	300			(1/0)	
3301-ESF Swgr. Rm. #1	314			(0/7)	See Note 2
3301-ESF Swgr. Rm. #1	315			(0/7)	See Note 2
3302-ESF Swgr. Rm. #2	316			(0/5)	See Note 2
3302-ESF Swgr. Rm. #2	317			(0/5)	See Note 2
3305-Ctrl. Bldg. Elec. Chase S. 2000'	301			(1/0)	
3306-Ctrl. Bldg. Elec. Chase N. 2000'	301			(1/0)	
3403-Non-Vit. Swgr. & Xfmr. Rm. #1	304			(0/1)	See Note 2
3403-Non-Vit. Swgr. & Xfmr. Rm. #1	305			(0/1)	See Note 2
3404-Switchboard Rm. #4	321			(0/2)	See Note 2
3404-Switchboard Rm. #4	322			(0/2)	See Note 2
3405-Battery Rm. #4	303			(2/0)	
3407-Battery Rm. #1	303			(2/0)	
3408-Switchboard Rm. #1	325			(0/2)	See Note 2
3408-Switchboard Rm. #1	326			(0/2)	See Note 2
3409-Non-Vit. Swgr. & Xfmr. Rm. #2	323			(0/1)	See Note 2
3409-Non-Vit. Swgr. & Xfmr. Rm. #2	327			(0/1)	See Note 2
3410-Switchboard Rm. #2	324			(0/2)	See Note 2
3410-Switchboard Rm. #2	328			(0/2)	See Note 2
3411-Battery Rm. #2	303			(2/0)	

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INSTRUMENT LOCATION	ZONE	HEAT	FLAME	SMOKE	COMMENTS
		TOTAL NUMBER OF INSTRUMENTS (SEE NOTE 1)			
3413-Battery Rm. #3	303			(1/0)	
3414-Switchboard Rm. #3	318			(0/2)	See Note 2
3414-Switchboard Rm. #3	320			(0/2)	See Note 2
3415-Acc. Ctrl. & Elec. Equip. A/C Units #1	303			(4/0)	
3416-Acc. Ctrl. & Elec. Equip. A/C Units #2	303			(4/0)	
3418-Ctrl. Bldg. Elec. Chase S. 2016'	303			(1/0)	
3419-Ctrl. Bldg. Elec. Chase N. 2016'	303			(1/0)	
-Ctrl. Bldg. Elec. Chase N. 2016'	303			(1/0)	
-Ctrl. Bldg. Elec. Chase S. 2016'	303			(1/0)	
3501-Lower Cable Spreading Rm.	306			(0/13)	
3504-Ctrl. Bldg. Elec. Chase N. 2032'	303			(1/0)	
3505-Ctrl. Bldg. Elec. Chase S. 2032'	303			(1/0)	
-Ctrl. Bldg. Elec. Chase N. 2032'	303			(1/0)	
-Ctrl. Bldg. Elec. Chase S. 2032'	303			(1/0)	
3601-Control Room	308			(4/0)	
3601-Control Room	309			(0/7)	See Note 2
3601-Control Room	319			(0/7)	See Note 2
3602-Pantry	308			(1/0)	
3603-Shift Supv. Office	308			(1/0)	
3605-Equipment Cabinet Area	308			(13/0)	
3606-Emerg. Equip. Storage Rm.	308			(1/0)	
3608-Janitor's Closet	308			(1/0)	
3609-SAS Rm.	308			(1/0)	
3617-Ctrl. Bldg. Elec. Chase S. 2047'6"	308			(1/0)	
3618-Ctrl. Bldg. Elec. Chase N. 2047'6"	308			(1/0)	
-Ctrl. Bldg. Elec. Chase S. 2047'6"	308			(1/0)	
3801-Upper Cable Spreading Rm.	307			(0/18)	
3804-Ctrl. Bldg. Elec. Chase S. 2073'6"	308			(1/0)	
-Ctrl. Bldg. Elec. Chase S. 2073'6"	308			(1/0)	
5201-W. Diesel Gen. Rm.	501		(4/0)		
5201-W. Diesel Gen. Rm.	502	(0/8)			
5203-E. Diesel Gen. Rm.	500		(4/0)		
5203-E. Diesel Gen. Rm.	503	(0/8)			

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INSTRUMENT LOCATION	ZONE	TOTAL NUMBER OF INSTRUMENTS (SEE NOTE 1) HEAT	FLAME	SMOKE	COMMENTS
6102-Fuel Bldg. Railroad Bay	600	(0/8)			
6104-Fuel Pool Cool. HX Rm. B	601			(6/0)	
6105-Fuel Pool Cool. HX Rm. A	601			(6/0)	
6202-Elec. Equipment Rm.	601			(3/0)	
6203-Air Handling Equip. Rm.	601			(3/0)	
6301-Fuel Bldg. 2047'6" Gen. Flr.	602		(2/0)		
6303-Fuel Bldg. Exh. Filt. Absorb. Rm. A	601			(2/0)	
6304-Fuel Bldg. Exh. Filt. Absorb. Rm. B	601			(2/0)	
-North ESW Pumphouse	002			(3/0)	
-South ESW Pumphouse	001			(3/0)	
-ESW Cooling Tower	001			(1/0)	
-ESW Cooling Tower	002			(1/0)	

NOTE 1: (x/y), where x is number of Function A (early warning fire detection and notification only) instruments and y is number of Function B (actuation of fire suppression systems and early warning and notification) instruments.

NOTE 2: Zone is associated with a Halon-protected space. Each space has two (2) separate detection circuits (zones). One zone, in its entirety, needs to remain operable.

NOTE 3: Line-type heat detector.

Justification for Revising Table 3.3-9 Page 3/4 3-64

The Table has been revised in accordance with the format provided in Standardized Technical Specifications. To this end, detector grouping is by room only (in lieu of smaller sub-groups) and detectors are identified by quantity in a room and function rather than by individual identification number.

LOOSE-PART DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The Loose-Part Detection System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2

ACITON:

in lieu of any other reporting requirement,

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

4.3.3.8 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. An ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and

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

Specification 3.3.3.8, Page 3/4 3-66

Justification :

The inoperability of the Loose Parts Monitoring System does not impair safe operation of the plant, therefore the preparation of both an LER for operating in a degraded mode and a Special Report is not justified.

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 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within ~~1~~ 6 hours.

SURVEILLANCE REQUIREMENT

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

*See Special Test Exception 3.10.4.

Specification 3.4.1.1 Page 3/4 4-1

Justification -

The safety analysis, Chapter 15.0.1.2 states that a partial loss of Reactor Coolant System flow is an ANS Condition II Fault which, at worst, will result in a reactor trip with the plant being capable of returning to power. Further, Chapter 15.3.2.2 demonstrates that for a full loss of flow from three loop conditions the resultant transient is similar to Condition II faults relative to fuel and Reactor Coolant System parameters.

In the event that a Reactor Coolant Pump is lost at power levels corresponding to Mode 1, a reactor trip will probably result from the induced SG water level transient, obviously in this situation proceeding to Mode 3 is virtually instantaneous. However, in the event that a reactor trip did not occur, it will take some time period for the Operating Shift to assure that plant conditions have stabilized and reasonable planning completed prior to initiating a shutdown transient with the plant already in an abnormal situation. Realistically, this would leave 30-40 minutes to conduct an evolution (≈30% power to 0% power) that normally requires 120 to 180 minutes with the plant in normal lineup.

In view of the fact that failure to meet the HOT STANDBY Condition in one hour automatically results in a Level III Violation, more emphasis is likely to be placed on meeting the legal obligation than on proceeding in a slow and controlled manner to assure safety.

W

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the reactor coolant and/or residual heat removal (RHR) loops listed below shall be OPERABLE and at least one of these reactor coolant and/or RHR loops shall be in operation:**

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,*

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,*

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,*

d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump,*

e. RHR Loop A,

f. RHR Loop B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required reactor coolant and/or RHR loops OPERABLE, immediately initiate corrective ACTION to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

b. With no reactor coolant or RHR 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.

*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 310°F unless: ~~(1) the pressurizer water volume is less than 92% (____ cubic feet) and/or~~ (2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

**All reactor coolant pumps and RHR pumps 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.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s) and/or RHR pump(s), if not in operation, shall be determined 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 secondary side wide range water level to be greater than or equal to 10% at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.

Specification: 3.4.1.3

Justification:

The Cold Overpressure Mitigation System (COMS) is part of the Wolfcreek and Callaway design and provides overpressure protection at low temperature conditions. COMS is designed to open the PORVs before the limiting RCS pressure for the given temperature is reached. This is accomplished through the use of a PORV setpoint programmed as a function of RCS temperature.

The requirement to maintain the pressurizer water volume below a limiting maximum value is not necessary with COMS.

Related changes occur in specifications: 3.4.1.4.1; 3.4.9.3; Figure 3.4-4; 3.8.1.2; 3.8.2.2; 3.8.3.2; B3/4.4.1; B3/4.4.9.

Justification 4.4.1.3.1, Page 3/4 4-4

Section 3.4.1.2 requires two RCP's operable in Mode 3. The associated surveillance requirement 4.4.1.2.1 addresses the same pumps.

Section 3.4.1.3 requires two of the following in any combination RCP/RHR pumps to be operable in Mode 4. The associated surveillance requirement 4.4.1.3.1 only addresses the RCP's.

Section 3/4.4.1 (the bases) discuss the required pumps for Mode 3 and Mode 4.

Careful examination shows that an omission is possible. Since the bases define the required pumps in Mode 4 to be either RHR or RCS, then Section 4.4.1.3.1 should be rewritten to include the option of using RHR pumps. Otherwise, if we go by the strict intent of 4.4.1.3.1, it is possible to run RHR in Mode 4 and violate Tech Specs surveillance requirements. In addition it makes the associated surveillance requirement unnecessary.

Specification: 4.4.1.3.2

Justification:

The words "wide range" were inserted to be consistent with the use of these words in Specification 4.4.1.2.2.

W

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*,* and either:

a. One additional RHR loop shall be OPERABLE#, or

b. The secondary side wide range water level of at least two steam generators shall be greater than 10%.

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

a. With one of the RHR loops inoperable and with less than the required steam generator level, immediately initiate corrective ACTION to return the inoperable RHR loop to OPERABLE status or restore the required steam generator level as soon as possible.

b. With no RHR 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 RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side wide range water level of at least two 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 RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR 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 310°F unless: ~~(1) the pressurizer water volume is less than 92% (____ cubic feet) and/or~~ ~~(2)~~ the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

**The RHR 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.

Specification: 3.4.1.4.1

Justification:

The LCO is modified for the same reason as in specification 3.4.1.3. The requirement to maintain the pressurizer water volume below a limiting maximum value is not necessary with COMS.

Related changes occur in specifications: 3.4.1.3; 3.4.9.3; Figure 3.4-4; 3.8.1.2; 3.8.2.2; 3.8.3.2; B3/4.4.1; B3/4.4.9.

<u>Specification</u>: 3.4.1.4.1.b, 4.4.1.4.1.1

<u>Justification</u>:

The words "wide range" were inserted for the same reason as Specification 4.4.1.3.2 and 4.4.1.2.2.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with at least two groups of backup pressurizer heaters each having a capacity of at least ~~(150)~~ kW and a water level of less than or equal to 92% (1657 cubic feet).

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one group of backup pressurizer heaters inoperable, restore at least two groups of backup heaters 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, 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 The pressurizer water level shall be determined to be within its limit at least once per 12 hours.

~~4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.~~

~~4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by manually transferring power from the normal to the emergency power supply and energizing the heaters.~~

Specification 4.4.3.3 Page 3/4 4-9

Justification -

The normal supply for the pressurizer backup heaters is a safety related bus. Due to the plant design of the normal supply meeting the criteria of an emergency power supply, and each of the two groups of backup heaters being supplied by a redundant emergency (normal) power source, this surveillance does not apply and should be deleted

Specification 4.4.3.2 Page 3/4 4-9

Justification -

The installed capacity of each group of pressurizer backup heaters is 700 KW. We have ground fault indication and alarms on the heaters. Normal operation of the plant will insure sufficient heater capacity is available to meet the L.C.O. due to the large reserve of installed capacity along with the ground fault detection system without this surveillance.

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more block valve(s) inoperable, within 1 hour: restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 ~~In addition to the requirements of Specification 4.0.5,~~ Each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Performance of a CHANNEL CALIBRATION, and

b. Operating the valve through one complete cycle of full travel.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of
ACTION a. of Specification 3.4.4.

~~4.4.4.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by:~~

~~a. Manually transferring motive and control power from the normal to the emergency power supply, and~~

~~b. Operating the valves through a complete cycle of full travel.~~

Specification 4.4.4.1 Page 3/4 4-10

Justification -

Since testing requirements of ASME Section XI do not apply to the Power Operated Relief Valves, no additional surveillance requirements are invoked by the words, "In addition to the requirements of Specification 4.0.5."

Specification 4.4.4.3 Page 3/4 4-10

Justification -

The PORV and the block valves are powered from an emergency power source at all times. Hence, this surveillance would be redundant to Specification 4.4.4.1.b and 4.4.4.2.

R

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 ~~Each steam~~ Steam generator tube structural integrity shall be ~~OPERABLE.~~ maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, due to tube leakage, ~~restore~~ repair the ~~inoperable generator(s) to OPERABLE status~~ tube leak(s) prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 In addition to the requirements of Specification 4.4.10, ~~Each~~ each steam generator shall be demonstrated ~~OPERABLE~~ to have satisfactory tube structural integrity by performance of the following augmented inservice inspection program. ~~and the requirements of Specification 4.0.5.~~

4.4.5.1 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.

4.4.5.2 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 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. 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:

Specification: 3.4.5

Justification:

The wording of the LCO, ACTION, and 4.4.5.0 was changed to indicate that this Specification is dealing with tube structural integrity. A steam generator could be inoperable for other reasons (e.g., low water level, high water level, etc.). By changing the wording, the operator instantly knows he is dealing with tube integrity rather than finding out as he reads through the specification.

Specification 4.4.5.0 Page 3/4 4-11

Justification -

Specification 4.4.10 requires the performance of Specification 4.0.5 for all ASME class 1, 2 or 3 components. The use of specification 4.0.5 in this spec (4.4.5.0) is unnecessary.

4.4.5.3 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 calendar 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 in 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 4.4.5.3a; the interval may then be extended to a maximum of once per 40 months; and

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 Class IV conditions or a seismic occurrence greater than the Operating Basis Earthquake:

   1) Primary ~~-to-secondary~~ -to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2,

   ~~2) A seismic occurrence greater than the Operating Basis Earthquake,~~

   2 ~~3)~~ A loss-of-coolant accident requiring actuation of the engineered safety features, and

   3 ~~4)~~ A main steam line or feedwater line break.

Specification: 4.4.5.3

Justification:

Class IV was added to clarify the definitions of the transients in item C. The purpose of this addition is to eliminate any ambiguity by more clearly defining the intent of this specification which is to provide surveillance for conditions which involve severe primary side transients. Revised text is attached.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Particulate Radioactivity Monitoring System,

b. The Containment Normal Sump Level Measurement ~~And Flow Monitoring~~ System, and △

c. Either the Containment Air Cooler Condensate Flow Rate or 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 for gaseous and particulate radioacitivity at least once per 24 hours when the required Gaseous or Particulate Radioactivity Monitoring System is operable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. X X

SEE INSERT FOLLOWING THIS PAGE

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Gaseous and Particulate Monitoring System-performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment Normal Sump Level Measurement ~~and Flow Monitoring~~ System-performance of CHANNEL CALIBRATION at least once per 18 months,

c. Containment Air Cooler Condensate Flow Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months. X

INSERT FOR ACTION IN SPECIFICATION 3.4.6.1

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 for gaseous and particulate radioactivity or a gamma isotopic analysis of the containment atmosphere is performed using the post accident sampling system at least once per 24 hours when the required gaseous 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.

Specification: 3.4.6.1, 4.4.6.1.b

Justification:

SNUPPS has no flow monitoring capability for the flow coming from the containment normal sumps; however, SNUPPS instrumentation does monitor normal sump level.

The action was changed to allow the post accident sampling system to be used in lieu of grab samples. As described in the FSAR, Chapter 18.2, this system can perform on-line gamma isotopic analysis of the containment atmosphere.

LIMITING CONDITION FOR OPERATION

3.4.6.2 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 not isolated from the Reactor Coolant System and (500) gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. ~~40~~ 8 gpm per pump CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. LEAKAGE AS SPECIFIED IN TABLE 3.4-1 ~~1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig~~ 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 above 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, reduce the leakage rate to within limits within 4 hours, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours with an RCS pressure of less than 600 psig.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere (gaseous or particulate) radioactivity monitor at least once per 12 hours.

b. Monitoring the containment normal sump inventory and discharge at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE ~~to~~ from the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days~~, with the modulating valve fully open~~. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours → Insert A below

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.6.2.2 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. FOR THOSE VALVES IDENTIFIED ON TABLE 3.4-1, Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

Insert A:

except when Tavg is being changed by greater than 5 F/hr or when diverting reactor coolant to the liquid hold up tanks, in which case the required inventory balance should be performed within **12** hours after completion of the operation.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION	NOTES
BBV8948A,B,C,D	Accum Injection Cold Leg	(a)(b)(d)
BBV8949B,C	SI/RHR Hot Leg Injection	(a)(b)(d)
BBV8949A,D	SI Hot Leg Injection	(a)(b)
BBV001, 022, 040, 059	BIT Disch to Cold Leg	(a)(b)
BBPV8702A,B	RHR Normal Suction	(a)(b)
EJV8841A,B	RHR Hot Leg Recirc Ctmt Iso	(a)(c)(d)
EJHV8701A,B	RHR Normal Suction	(a)(c)
EMV001, 002, 003, 004	SI Hot Leg Inj Ctmt Iso	(a)(c)
EM8815	BIT Inj Ctmt Isolation	(a)(c)
EPV010, 020, 030, 040	SI Accum Inj Ctmt Iso	(a)(c)
EPV8818A,B,C,D	RHR Accum Inj Ctmt Iso	(a)(c)(d)
EPV8956A,B,C,D	Accum Inj Isolation	(a)(c)(d)

NOTES:

(a) Maximum Allowable Leakage (each valve):

1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. 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 the previous test by 1.0 gpm or by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or more.

3. Leakage rates greater than 5.0 gpm are considered unacceptable.

(b) Test Pressure shall be 2235 ± 20 psig.

(c) Test Pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted to a test pressure of 2235 assuming leakage to be directly proportional to pressure differential to the one-half power.

(d) Leakage test required for these valves within 24 hours following valve actuation or manual action or flow through the valve.

Justification:

(a) Maximum Allowable Leakage

A Maximum Allowable Leakage of 5.0 gpm is reasonable in that:

1. The relieving capacities of the systems under question are greater than 20 gpm.
2. Leakages greater than 1 gpm are trended to ensure that leak rate increases between subsequent tests do not exceed 1 gpm.
3. For leakage rates over 3 gpm, leak rate increases between subsequent tests are limited to 50% of the margin between last measured leakage and the maximum allowable of 5.0 gpm. This trending will ensure repairs are made before leakage exceeds the 5.0 gpm limit.

(b) These are the first isolation valves on the injection lines and see full RCS pressure during operation. They will be tested at full RCS pressure.

(c) These are the second isolation valves out from the RCS. Due to the difficulty of applying full RCS pressure to these valves, testing at reduced pressures will be accomplished in accordance with IWV-3423 of ASME Section XI. Leakage rates will be adjusted upwards to a corresponding test pressure of 2235 psig before applying the acceptance criteria of (a) above.

(d) Valves with this note serve to isolate systems with design ratings of less than 50% of RCS design pressure. These valves will be tested following valve actuation or manual action or flow through the valve. The RHR suction valves are not included in this category since valve closure can be verified via remote position indication in the control room.

Technical Specification: Controlled Leakage
1.8 Definition
3.4.6.2.e
4.4.6.2.1.c
Bases 3/4.4.6.2

Justification:

Revise the above referenced sections of Technical Specifications to reflect changes as noted. The bases for the changes is to define controlled leakage as leakage from the seal, which is consistent from a human engineering point of view with the balance of defined RCS leakage terms. This also provides a limit and surveillance which will monitor actual seal performance. Seal injection does not measure performance and seal degradation.

The charging pump discharge supply to the RCP seal water injection is controlled by locked and throttled angle stop check valves BGV-198, 199, 200, 201. Controlled leak off return is limited by a locked and throttled angle stop check valve BGV-202. HCV-182 allows flow to be balanced between the normal charging flow path to the reactor coolant loop 1 cold leg and the reactor coolant pump seals. Controlled leak off from the #1 seal is isolated by building isolation valves BGHV-8100 and 8112 on a Phase A containment isolation signal. Technical Specification 3.5.2 for ECCS subsystems establishes operability for the centrifugal charging pumps and safety injection pumps by verifying proper line ups, pump capacities, and balanced flows to assure minimum flow rates to the reactor coolant system normal injection flow paths.

Since Technical Specification 3.5.2 assures ECCS system operability, the revision of surveillance requirements for specification 4.4.6.2.1 will allow consideration for leakage and correct the assumptions made on the design and bases of the pump seal control system.

Eight gpm controlled leakage per pump was choosen to be consistent with the requirement to isolate the #1 pump seal as recommended by Westinghouse.

This change is submitted in conjunction with the proposed change to ECCS subsystems 4.5.2.g.2.

Specification 4.4.6.2.1.d Page 3/4 4-20

Justification -

The water inventory balance is only accurate under stable thermodynamic conditions. The suggested modification to the specification is designed to prevent technical violation of the specification e.g. normally this surveillance requires taking one to four hours of data then doing computations. Further, this type surveillance is normally scheduled for a particular shift for every third day, therefore, if immediately prior to completing the data taking the plant underwent a load reduction transient and stable conditions were not reestablished for eighteen hours, then the Technical Specification would have been violated.

In addition to the situation described above, return to power following forced outages can easily be delayed until stable thermodynamic conditions can be established to facilitate a water inventory balance. Such restrictions seem unnecessary for a backup method of leakage detection.

See LER 82-027, Rev. 1, Docket 50-311 (Copy of summary attached).

(NSIC 179144) DURING ROUTINE POWER ESCALATION, THE CONTROL ROOM OPERATOR INFORMED THE SHIFT SUPERVISOR THAT THE RCS WATER INVENTORY SP(O)4.4.7.2.D HAD NOT BEEN PERFORMED WITHIN THE TIME INTERVAL OF 72 HOURS PLUS 25% REQUIRED BY THE TECH SPECS. STEADY STATE CONDITIONS, WHICH ARE REQUIRED FOR PERFORMANCE OF THE SURVEILLANCE, COULD NOT BE OBTAINED DUE TO CHANGES IN XENON REACTIVITY. THIS OCCURRENCE CONSTITUTED OPERATION IN A DEGRADED MODE IN ACCORDANCE WITH TECH SPEC 6.9.1.9.B. SURVEILLANCE OF THE CONTAINMENT SUMP PUMP AND OF THE CONTAINMENT RADIOACTIVITY MONITORS WAS INCREASED, AND RCS LEAKAGE WAS DEMONSTRATED TO BE WITHIN SPECIFICATION LIMITS. THE RCS WATER INVENTORY WAS PERFORMED, AND AT 0135 HOURS, APRIL 20, 1982, ACTION STATEMENT 3.4.7.2.A WAS TERMINATED. LICENSE CHANGE REQUEST 82-14 WAS SUBMITTED TO ELIMINATE THE RCS WATER INVENTORY REQUIREMENT DURING NON-STEADY STATE OPERATION.

INDICATION FOR NO. 15 CONTAINMENT FAN COIL UNIT (CFCU) WAS ERRATIC. THE UNIT WAS DECLARED INOPERABLE AND ACTION STATEMENT 3.6.2.3.A WAS ENTERED. BOTH CONTAINMENT SPRAY SYSTEMS WERE OPERABLE. THE EVENT CONSTITUTED OPERATION IN A DEGRADED MODE IN ACCORDANCE WITH TECH SPEC 6.9.1.9.B. SEE LERS: 82-067, 82-061, 82-037, 82-029, 82-024. THE ERRATIC FLOW INDICATION WAS CAUSED BY SILT PLUGGING THE TRANSMITTER SENSING LINES. THE SENSING LINES WERE BLOWN DOWN TO REMOVE THE SILT, THE CFCU WAS SATISFACTORILY TESTED, AND THE ACTION STATEMENT WAS TERMINATED. A PROGRAM TO BLOW DOWN TRANSMITTERS WEEKLY HAS EFFECTIVELY REDUCED THE FREQUENCY OF THIS PROBLEM.

[264] SALEM 1 DOCKET 50-272 LER 82-079
BATTERY AND CHARGER TESTING MISSED.
EVENT DATE: 100682 REPORT DATE: 102782 NSSS: WE TYPE: PWR
SYSTEM: DC ONSITE POWER SYS & CONTROLS COMPONENT: BATTERIES & CHARGERS
CAUSE: ADMINISTRATIVE ERROR.

(NSIC 179083) FOLLOWING SATISFACTORY COMPLETION OF MAINTENANCE PROCEDURE M3M, IT WAS DISCOVERED THAT THE PROCEDURE HAD NOT BEEN PERFORMED THE PRECEDING WEEK AS REQUIRED BY TECH SPEC SURVEILLANCE REQUIREMENTS 4.8.2.3.2.A, AND 4.8.2.5.2.A. THE SURVEILLANCE IS REQUIRED TO DEMONSTRATE THE OPERABILITY OF SALEM GENERATING STATION UNITS 1 AND 2 28VDC AND 125VDC BATTERIES AND CHARGERS. ASSIGNMENT OF THE SURVEILLANCE TO A STATION ELECTRICIAN WAS OVERLOOKED BY THE ELECTRICAL SUPERVISOR. SCHEDULING OF THE PROCEDURE HAD NOT BEEN INCORPORATED INTO THE INSPECTION ORDER SYSTEM. A MONTHLY INSPECTION ORDER CARD WAS WRITTEN TO PERFORM THE SURVEILLANCE WEEKLY, AND A SUPERVISOR WAS DESIGNATED TO TRACK COMPLETION.

[265] SALEM 2 DOCKET 50-311 LER 82-002 REV 1
UPDATE ON BATTERY CHARGER TRIPS.
EVENT DATE: 011182 REPORT DATE: 102082 NSSS: WE TYPE: PWR
SYSTEM: DC ONSITE POWER SYS & CONTROLS COMPONENT: BATTERIES & CHARGERS
CAUSE: DIFFERENCE BETWEEN BATTERY AND CHARGER VOLTAGES.

(NSIC 179185) THE CONTROL ROOM OPERATOR NOTICED NO. 2A1 BATTERY CHARGER HAD TRIPPED. UPON RESETTING THE BREAKER, IT TRIPPED AGAIN. NO. 2A2 BATTERY CHARGER WAS PUT IN SERVICE. NO. 2A1 BATTERY CHARGER WAS DECLARED INOPERABLE AND ACTION STATEMENT 3.8.2.3.C WAS ENTERED. THE TRIPS WERE CAUSED BY A DIFFERENCE BETWEEN THE BATTERY AND BATTERY CHARGER VOLTAGES. PERSONNEL WERE INSTRUCTED AS TO THE PROBLEM. NO. 2A1 BATTERY CHARGER WAS BACK IN SERVICE AND ACTION STATEMENT 3.8.2.3.C WAS TERMINATED. THE BATTERY CHARGER OPERATING INSTRUCTION WAS REVISED TO HAVE AN ELECTRICIAN ADJUST THE FLOAT VOLTAGE PRIOR TO CHARGING THE BATTERY.

[266] SALEM 2 DOCKET 50-311 LER 82-027 REV 1
UPDATE ON MISSED RCS WATER INVENTORY.
EVENT DATE: 041882 REPORT DATE: 102082 NSSS: WE TYPE: PWR
SYSTEM: COOLANT RECIRC SYS & CONTROLS COMPONENT: COMPONENT CODE NOT APPLICABLE
CAUSE: CHANGES IN XENON REACTIVITY.

(NSIC 179144) DURING ROUTINE POWER ESCALATION, THE CONTROL ROOM OPERATOR INFORMED THE SHIFT SUPERVISOR THAT THE RCS WATER INVENTORY SP(0)4.4.7.2.D HAD NOT BEEN PERFORMED WITHIN THE TIME INTERVAL OF 72 HOURS PLUS 25% REQUIRED BY THE TECH SPECS. STEADY STATE CONDITIONS, WHICH ARE REQUIRED FOR PERFORMANCE OF THE SURVEILLANCE, COULD NOT BE OBTAINED DUE TO CHANGES IN XENON REACTIVITY. THIS OCCURRENCE CONSTITUTED OPERATION IN A DEGRADED MODE IN ACCORDANCE WITH TECH SPEC 6.9.1.9.B. SURVEILLANCE OF THE CONTAINMENT SUMP PUMP AND OF THE CONTAINMENT RADIOACTIVITY MONITORS WAS INCREASED, AND RCS LEAKAGE WAS DEMONSTRATED TO BE WITHIN SPECIFICATION LIMITS. THE RCS WATER INVENTORY WAS PERFORMED, AND AT 0135 HOURS, APRIL 20, 1982, ACTION STATEMENT 3.4.7.2.A WAS TERMINATED. LICENSE CHANGE

REQUEST 82-14 WAS SUBMITTED TO ELIMINATE THE RCS WATER INVENTORY REQUIREMENT DURING NON-STEADY STATE OPERATION.

[267] SALEM 2 DOCKET 50-311 LER 82-074
CFCU INOPERABLE DUE TO LEAKAGE.
EVENT DATE: 081382 REPORT DATE: 081882 NSSS: WE TYPE: PWR
SYSTEM: CNTNMNT HEAT REMOV SYS & CONT COMPONENT: HEAT EXCHANGERS
CAUSE: TUBING EROSION DUE TO SILT PARTICLES.

(NSIC 178166) THE CONTROL OPERATOR DISCOVERED LEAKAGE TO THE CONTAINMENT SUMP HAD INCREASED. ACTION STATEMENT 3.4.7.2.B WAS ENTERED DUE TO UNIDENTIFIED LEAKAGE BEING GREATER THAN 1 GPM. A CONTAINMENT ENTRY WAS MADE, AND A 1 GPM LEAK WAS OBSERVED ON NO. 25 CFCU. WITH UNIDENTIFIED LEAKAGE LESS THAN 1 GPM, ACTION STATEMENT 3.4.7.2.B WAS TERMINATED. THE LEAK WAS ISOLATED AND ACTION STATEMENT 3.6.2.3.A WAS ENTERED. A SIMILAR LEAK OCCURRED AUGUST 14, ON NO. 22 CFCU AND ACTION STATEMENT 3.6.2.3.B WAS ENTERED. SEE LERS: 82-073, 82-070, 82-040, 82-039, 82-028. THE LEAKAGE IN BOTH CASES WAS DUE TO COOLING COIL FAILURES RESULTING FROM EROSION BY SILT PARTICLES IN THE SERVICE WATER. THE LEAK WAS REPAIRED BY BLANKING THE FLANGES IN THE LINES TO THE COIL. NO. 25 CFCU WAS DECLARED OPERABLE, AND ACTION STATEMENT 3.6.2.3.A REMAINED IN EFFECT UNTIL AUGUST 15, 1982, WHEN ALL CFCU'S WERE RETURNED TO OPERABLE STATUS, SEE LER 82-075.

[268] SALEM 2 DOCKET 50-311 LER 82-075
CFCU INOPERABLE DUE TO TUBING LEAK.
EVENT DATE: 081482 REPORT DATE: 081882 NSSS: WE TYPE: PWR
SYSTEM: CNTNMNT HEAT REMOV SYS & CONT COMPONENT: HEAT EXCHANGERS
CAUSE: EROSION DUE TO SILT PARTICLES.

(NSIC 178169) ON AUGUST 14, 1982, A 1 GPM SERVICE WATER LEAK WAS DISCOVERED ON NO. 22 CONTAINMENT FAN COIL UNIT (CFCU). NO. 25 CFCU WAS INOPERABLE AT THE TIME, AND ACTION STATEMENT 3.6.2.3.A WAS IN EFFECT. WITH AN ADDITIONAL CFCU INOPERABLE, ACTION STATEMENT 3.6.2.3.B WAS ENTERED. THE UNIT WAS IMMEDIATELY ISOLATED. NO. 25 CFCU WAS REPAIRED LATER THAT DAY, AND ACTION STATEMENT 3.6.2.3.B WAS TERMINATED. (SEE LER 82-074.) ON AUGUST 15, A SIMILAR LEAK WAS DISCOVERED ON NO. 24 CFCU; THIS UNIT IS IN THE SAME GROUP AS NO. 22, HENCE ACTION STATEMENT 3.6.2.3.A STILL APPLIED. THE EVENTS CONSTITUTED DEGRADATION OF CONTAINMENT IN ACCORDANCE WITH TECH SPEC 6.9.1.8.C. NO. 22 CFCU WAS REPAIRED BY BLANKING THE FLANGES IN THE SERVICE WATER PIPING TO THE COIL. THE LEAK ON NO. 24 CFCU WAS REPAIRED WITH BELZONA METAL FILLER. WITH BOTH UNITS OPERABLE, ACTION STATEMENT 3.6.2.3.A WAS TERMINATED. NEW DESIGN COILS WILL BE INSTALLED DURING THE NEXT REFUELING. SEE LERS: 82-074, 82-073, 82-070, 82-040, 82-028.

[269] SALEM 2 DOCKET 50-311 LER 82-072 REV 1
UPDATE ON RPS TRIP BREAKER FAILURE.
EVENT DATE: 082082 REPORT DATE: 102082 NSSS: WE TYPE: PWR
SYSTEM: REACTOR TRIP SYSTEMS COMPONENT: CIRCUIT CLOSERS/INTERRUPTERS
CAUSE: BINDING OF UNDERVOLTAGE COIL.

(NSIC 179184) DURING TESTING, IT WAS DISCOVERED THAT REACTOR TRIP BREAKER B WOULD NOT TRIP AS REQUIRED. THE CHANNEL WAS DECLARED INOPERABLE AND ACTION STATEMENT 3.3.1 ACTION 1 WAS ENTERED. A POWER REDUCTION WAS COMMENCED IN COMPLIANCE WITH THE ACTION STATEMENT. THE CAUSE WAS BINDING OF THE UNDERVOLTAGE (UV) COIL. THE REACTOR TRIP BREAKER WAS REPLACED WITH AN A TRAIN BYPASS BREAKER AND THE SURVEILLANCE WAS SATISFACTORILY PERFORMED. ACTION STATEMENT 3.3.1 ACTION 1 WAS TERMINATED. THE UV COIL WAS REPLACED AND REACTOR TRIP BREAKER B WAS REINSTALLED AND SATISFACTORILY TESTED.

TABLE 4.4-4

REACTOR 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 Specific 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 Ē Determination***	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-132 I-133, I-134 and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram DOSE EQUIVALENT I-131 or 100/Ē 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.	1, 2, 3

#Until the specific activity of the Reactor 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.

**A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the reactor coolant sample except for radionuclides with half-lives less than ~~10~~ 15 minutes and all radioiodines. The total specific activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities in the sample within 2 hours after the sample is taken extrapolated back to when the sample was taken.

***A radiochemical analysis shall consist of the quantitative measurement of the specific activity for each radionuclide except for radionuclides with half-lives less than ~~10~~ 15 minutes and all radioiodines which ~~is~~ are identified in the reactor coolant ~~with 2 hours after the sample is taken~~. The specific activities for these individual radionuclides shall be used in the determination of Ē, and shall make up at least 95% of the total non-iodine activity in the coolant sample. + Sample within 2 hours after the sample is taken.

Technical Specification: Table 4.4-4

Justification:

Item 1 Gross Specific Activity

These changes are requested due to the constraints put on this analysis. These constraints are: 1) activity can only be measured in the sample; and 2) the time necessary to sample, prepare, and analyze in this situation does not lend itself to allow an accurate analysis of nuclides with half-lives less than 10 minutes. It is estimated that 90 minutes would be required to perform this sample-analysis sequence. This would then be trying to analyze for nuclides that have decayed through 9 half-lives. This ~~which~~ would make further analysis to obtain statistically meaningful data i.e., longer count times meaningless due to further decay during the additional count time. The radio nuclides in a typical reactor coolant sample have half-lives of less than 4 minutes or greater than 15 minutes. The only noteable exception to this rule is Xe-138 with a half-life of 14.17 minutes.

Item 3 Radiochemical for $\overline{E}$ Determination

In addition to the arguments stated for Item 1 as to the 15 minute cut off for nuclides half-lives, these changes need to be made so that this corresponds to the $\overline{E}$ definition in Section 1.11.

Item 4 Isotopic Analysis for Iodine

These changes are requested so that this section corresponds to the definition in Section 1.10 for dose equivalent iodine.

MATERIAL PROPERTY BASIS

COPPER CONTENT : CONSERVATIVELY ASSUMED TO BE 0.10 WT%

RT_{NDT} INITIAL : 40°F

RT_{NDT} AFTER 16 EFPY: 1/4T, 110°F
3/4T, 87°F

CURVE APPLICABLE FOR HEATUP RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

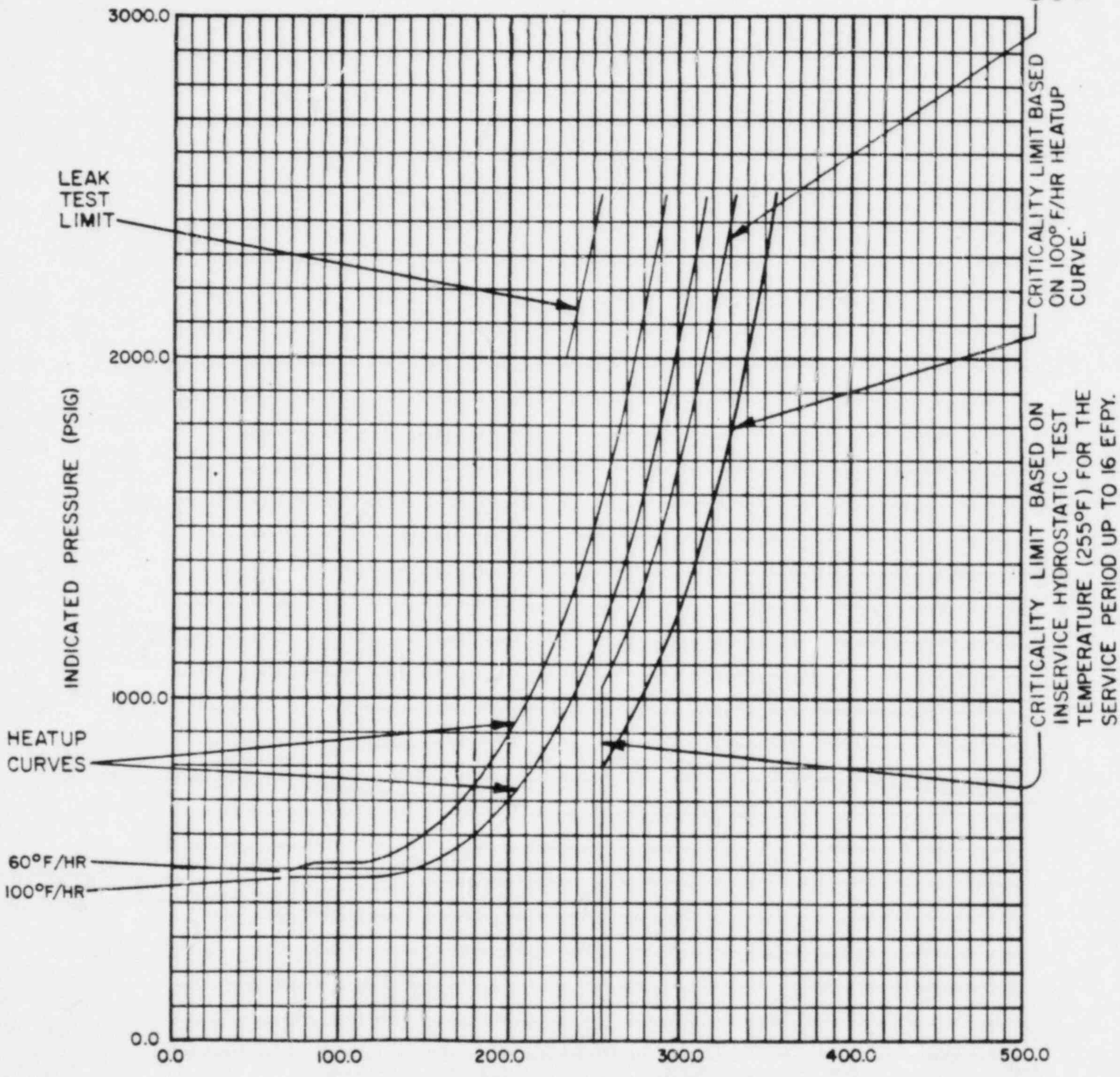

FIGURE 3.4-2 WOLF CREEK REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE UP TO 16 EFPY

MATERIAL PROPERTY BASIS

COPPER CONTENT : CONSERVATIVELY ASSUMED TO BE 0.10 WT%

RT_{NDT} INITIAL : 50°F

RT_{NDT} AFTER 7 EFPY: 1/4T, 110°F
3/4T, 87°F

CURVE APPLICABLE FOR HEATUP RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TC 7 EFPY AND CONTAINS MARGINS OF 10°F. AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

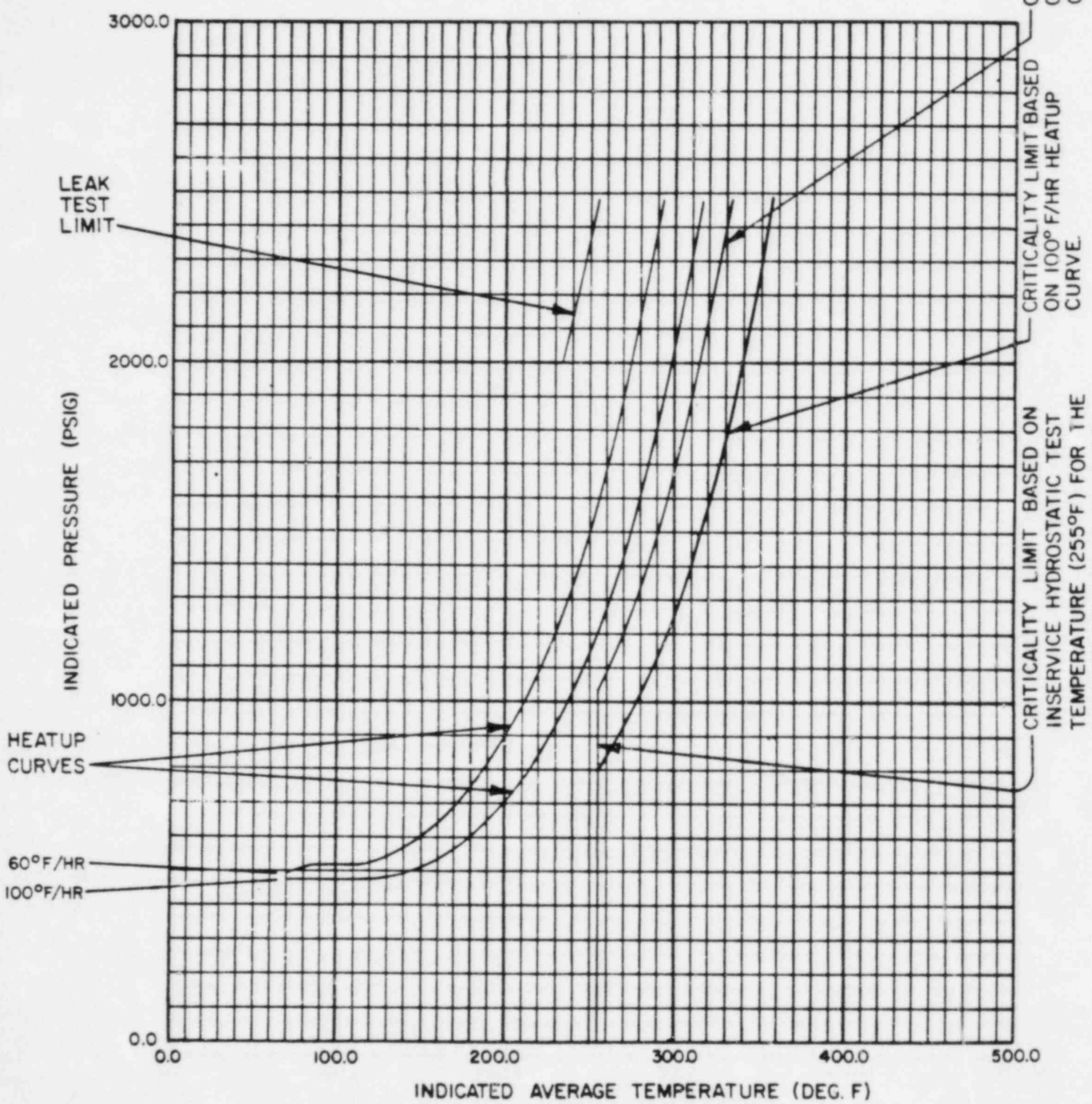

FIGURE 3.4-2 CALLAWAY REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE UP TO 7 EFPY

Figure 3.4-2

Justification:

The attached curves have been modified to reflect the addition of a 100°F/hr heatup curve and (Figure 3.4-2) replace any existing Tech Spec figures. Curves for both 60°F/hr and 100°F/hr are included for both Wolfcreek and Callaway.

W

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two power operated relief valves PORVs with ~~nominal~~ Setpoints which do not ~~vary with RCS temperature as shown on~~ exceed the limit established by Figure 3.4-4, or

b. The Reactor Coolant System (RCS) depressurized with ~~an RCS vent of~~ one OR MORE ~~greater than or equal to 2 square inches~~ RCS vents which are capable of relieving a combined TOTAL OF AT LEAST 460 GPM PRIMARY COOLANT AT 560PSIG RCS PRESSure.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 310°F, MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through ~~at least a 2 square inch vent~~ THE VENT(S) DESCRIBED IN PARAGRAPH 3.4.9.3.b within the next 8 hours.

b. With both PORVs inoperable, depressurize and vent the RCS through ~~at least a 2 square inch vent~~ THE VENT(S) DESCRIBED IN PARAGRAPH 3.4.9.3.b within 8 hours.

c. In the event either the PORVs or the 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 PORVs or vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

Specification: 3.4.9.3

Justification:

The LCO is modified for the same reason as in specification 3.4.1.3.

The PORV setpoint limit for COMS is specified by Figure 3.4-4. Vents are more appropriately specified by a flow rate requirement than a vent area.

Related changes occur in specifications: 3.4.1.3, 3.4.1.4.1, Figure 3.4-4, 3.8.1.2, 3.8.2.2, 3.8.3.2, B3/4.4.1, B3/4.4.9.

W

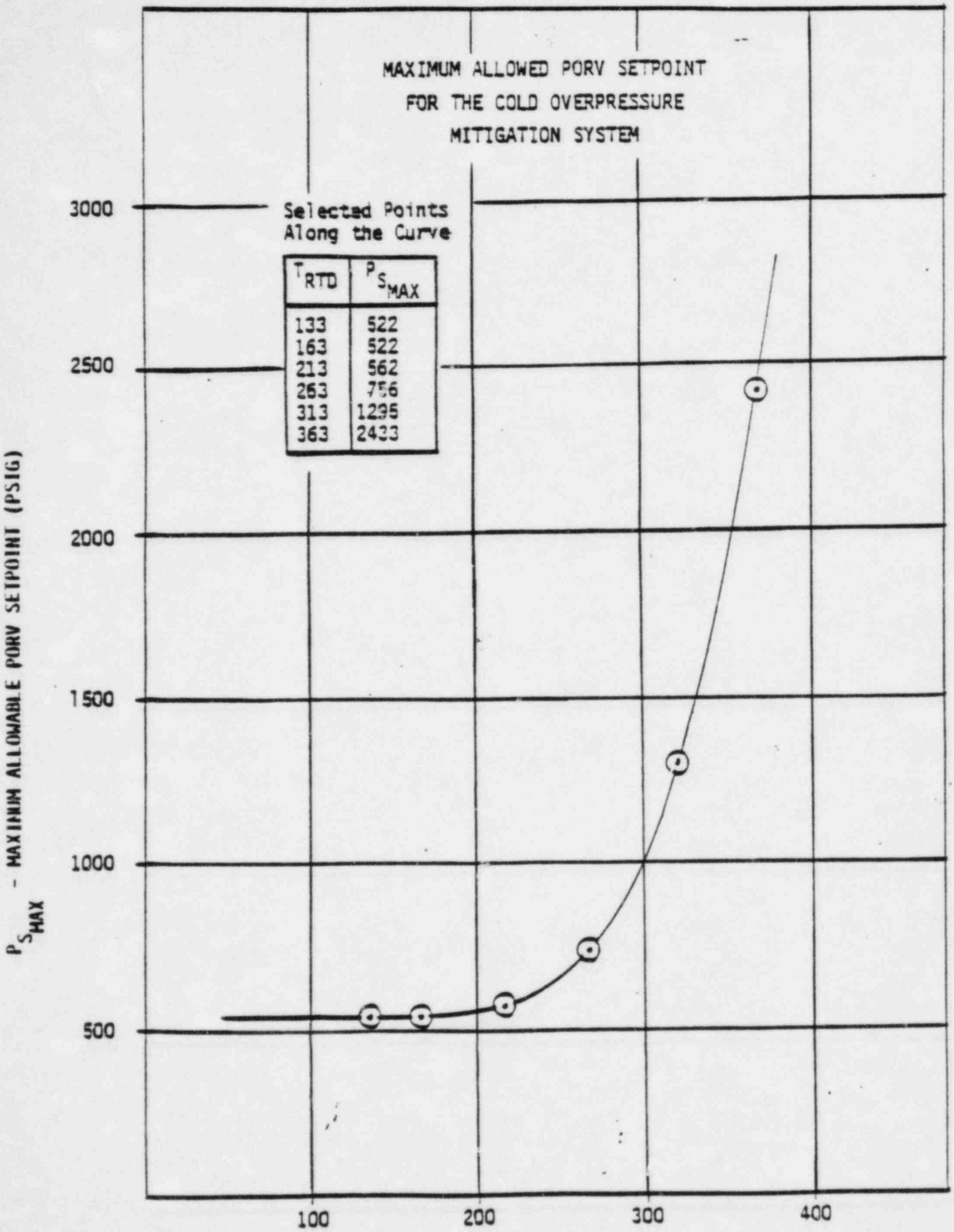

FIGURE 3.4-4

T_{RTD} - LOWEST COLD OVERPRESSURE MITIGATION SYSTEM RTD TEMPERATURE (°F)

Figure 3.4-4

Justification:

The Figure is modified for the same reason as in specification 3.4.1.3.

This figure specifies the maximum allowed PORV setpoint for COMS.

Related changes occur in specifications: 3.4.1.3, 3.4.1.4.1, 3.4.9.3, 3.8.1.2, 3.8.2.2, 3.8.3.2, B3/4.4.1, B3/4.4.9.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 6122 and 6594 gallons,

c. A boron concentration of between ~~(1900)~~ and ~~(2100)~~ ppm, and X

d. A nitrogen cover-pressure of between 602 and 648 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator 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 accumulator 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 in HOT SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, ~~by the absence of alarms,~~ the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 70 gallons by verifying the boron concentration of the accumulator solution;

c. At least once per 31 days when the RCS pressure is above 2000 psig by verifying that the circuit breaker supplying power to the isolation valve operator is open. ~~disconnected by removal of the breaker from the circuit; and~~

~~d. At least once per 18 months by verifying that each accumulator isolation valve opens automatically under each of the following conditions:~~

~~1) When an actual or a simulated RCS pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint, and~~

~~2) Upon receipt of a Safety Injection test signal.~~

~~4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE:~~

~~a. At least once per 31 days by the performance of an ANALOG CHANNEL OPERATIONAL TEST, and~~

~~b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.~~

4.5.1.1.a (1)

Justification:

Remove reference to "absence of alarms", redundant pressure and level indications exist which will be used to verify surveillance requirements. Annunciator alarms or problems with this system may not be an indication of volume and pressure in these tanks.

Technical Specification: 4.5.1.1.c

Justification:

The SNUPPS accumulator isolation valves are powered from 480 volt motor control centers. These breakers are of the molded case design and cannot be removed from the circuit. They can be verified in the tripped (open) condition.

Technical Specification: 4.5.1.1.d

Justification:

Delete reference to surveillance requirements associated with P-11 and SI signal. The accumulator isolation valves are required to be open and power removed from the valve when the plant is in mode 3 and > 1000 psig and at all times in modes 1 and 2. The accumulator isolation valves are not required to move during power operation or in a post accident situation. These valves are positioned prior to startup and then have power removed from the motor operator. These valves are not required to change position after a LOCA. Postulated failures do not present problems for either short or long term ECCS operations, containment isolation, or safety-related functions.

Technical Specifications: 4.5.1.2

Justification:

Delete reference to surveillance requirements. Basically, channel checks will be performed each shift ~~shiftly~~ of redundant level and pressure indicatiors for the determination of 4.5.1.1.a.1, water volume and cover pressure. Those components and modules do not generate any protective signals. These channels also do not satisfy the definition of channel as defined in IEEE-279-1971. The instrument strings will be included in the refueling interval preventive maintenance program for calibration.

W

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 power to the valve operators removed:

Valve Number	Valve Function	Valve Position
a. BN-HV-8813	a. Safety Injection to RWST Isolation Vlv	a. Open
b. EM-HV-8802A(B)	b. SI Pump Discharge Hot Leg Iso Vlvs	b. Closed
c. EM-HV-8835	c. Safety Injection Cold Leg Iso Valve	c. Open
d. ~~EF~~ EJ-HV-8840	d. RHR/SI Hot Leg Recirc Iso Valve	d. Closed
e. EJ-HV-8809A	e. RHR to Accum Inj Loops 1 & 2 Iso Vlv	e. Open
f. EJ-HV-8809B	f. RHR to Accum Inj Loops 3 & 4 Iso Vlv	f. Open

b. At least once per 31 days by:

~~1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points; and~~

~~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.

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) Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1) Verifying automatic isolation and interlock action of the RHR System from the Reactor Coolant System by ensuring that:

a) with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig the interlocks prevent the valves from being opened, and

b) with a simulated or actual Reactor Coolant System pressure signal less than or equal to ~~600~~ psig the interlocks will cause the valves to automatically close.

 750

Specification: 4.5.2.b.1

Justification:

The required venting every 31 days exposes personnel to potential radioactive liquid, creates unnecessary liquid radwaste, and violates the ALARA concept. SNUPPS performs surveillance at required intervals on all ECCS pumps and valves. These surveillances should be sufficient to detect any problems leading to pump cavitation and overheating without relying on the 31 day venting criteria.

W

4.5.2.d.1

The change of this setpoint to 750 psig had been approved for TVA and documented in a letter from T. Novak, 1/16/80 (attached). This approval was based on the fact that the automatic closure interlock setpoint on RCS pressure for RHR pump suction does not protect the RHR system from overpressurization events because of the relatively slow closure times of the valves. Overpressurization of the RHR system is prevented by the independent prevent-open interlock of the isolation valves set at 425 psig. The basis for the automatic closure interlock setpoint is to ensure the closure of both valves when ascending in power to an operating condition; thereby ensuring compliance with RCS pressure boundary isolation criteria. Because the proposed change in automatic closure interlock does not adversely affect protection against overpressurization of the RHR system, W recommends the change for the interlock setting to 750 psig.

W

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C.

JAN 1 6 1980

ket Nos.: 50-327/328

MEMORANDUM FOR: L. Rubenstein, Acting Chief, LWR #4, DPM

FROM: T. Novak, Chief, Reactor Systems Branch, DSS

SUBJECT: TECHNICAL SPECIFICATION - ISOLATION INTERLOCK

lant Name: Sequoyah
ocket Nos.: 50-327/328
icensing Stage: OL
esponsible Branch and Project Manager: LWR-4; C. Stahle
ystems Safety Branch Involved: Reactor Systems Branch
escription of Review: Tech Spec Change
equested Completion Date: January 11, 1980
iew Status: Complete

he Reactor Systems Branch has reviewed the Sequoyah applicant's proposal
o change the automatic closure interlock setpoint on reactor coolant system
ressure for residual heat removal pump suction isolation valves from 600 psig
o 750 psig in the Technical Specifications for Sequoyah, Units 1 and 2. We
ind this change acceptable as discussed in the enclosure.

Sanford Israel

T. Novak, Chief
Reactor Systems Branch
Division of Systems Safety

closure:
stated

: w/enclosure
~~M. Rubin~~
F. Orr
C. Stahle
F. Schroeder
R. Denise
S. Varga
D. Vassallo
. Mattson
M. Virgilio
P. Wagner

tact: Frank Orr, Ext. 27591

W

The applicant has proposed that the automatic closure interlock setpoint on reactor coolant system pressure for residual heat removal pump suction isolation valves be raised from 600 pounds per square inch gauge to 750 pounds per square inch gauge in the Technical Specifications for Sequoyah, Units 1 and 2. He has stated that the interlock does not protect the residual heat removal system from overpressurization events from shutdown or starting up conditions because of the relatively slow closure time of the valves. The applicant has identified that the reason for the automatic closure interlock is to ensure closure of both valves (rather than just one) when ascending in pressure to an operating condition, thereby ensuring compliance with reactor coolant pressure boundary isolation criteria. He has also pointed out that protection against overpressurization of the residual heat removal system while at full pressure or while depressurizing is provided by the independent prevent-open interlock of the isolation valves set at 425 pounds per square inch gauge. Because the proposed change in automatic closure interlock does not adversely affect protection against overpressurization of the residual heat removal system we find the Technical Specification change for the interlock setting (750 psig) acceptable.

In Safety Evaluation Report section 5.3.2 item 1 (page 5-14), last line, we identified the automatic closure interlock setting to be 600 pounds per square inch gauge. To maintain consistency with the Technical Specification setting this value should be amended to 750 pounds per square inch gauge.

SURVEILLANCE REQUIREMENTS (Continued)

2) 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 abnormal corrosion.

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 a Safety Injection test signal and an RHR automatic switchover to RWST Level-Low-Low 1 test signal; and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) RHR pump.

f. By verifying that each of the following pumps develops the indicated discharge pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1)	Centrifugal charging pump	≥ 2390 psig,
2)	Safety Injection pump	≥ 1500 psig, and
3)	RHR pump	≥ ~~210~~ 190 psig.

g. By verifying the correct position of each 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, and

2) At least once per 18 months.

HPSI System Valve Number

[redacted]. EMV095
[redacted]. EMV096
[redacted]. EMV097
[redacted]. EMV098
[redacted]. EMV107
[redacted]. EMV108

[redacted]. EMV109
[redacted]. EMV110
[redacted]. EMV089
[redacted]. EMV090
[redacted]. EMV091
[redacted]. EMV092

CVCS system valve number

BGV - 198
BGV - 199
BGV - 200
BGV - 201
BGV - 202

Technical Specification: 4.5.2.g.2

Justification:

Add the valves in the CVC System to the mechanically restrained valves list to assure injection line flow rates are maintained equal to or above those assumed in the HCCS LOCA analysis. This change is submitted in conjunction with the proposed change to the controlled leakage portion of 3.4.6.2.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

4.5.3.2 All charging pumps and Safety Injection pumps, except the above required OPERABLE pumps, shall be demonstrated inoperable** ~~by verifying that the motor circuit breakers are secured in the open position~~ at least once per 31 days ~~12 hours~~ whenever the temperature of one or more of the RCS cold legs is less than or equal to 310°F by verifying that the motor circuit breakers ~~are secured in the open position~~ have been removed from their electrical power supply circuits.

X

**An inoperable pump may be energized for testing per Specification 4.0.5 or for filling accumulators provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

Technical Specification: 4.5.3.2 Page 3/4 5-8

Justification:

Revise surveillance interval from 12 hours to 31 days. This surveillance applies to assuring breaker is racked out and secured to prevent inadvertant pump operation. Procedures control position of breakers and pump status on a cool down. Monthly surveillance agrees with tagging requirements of clearance procedure for reverification of system status. This precaution is also redundant to PORV's and RHR reliefs. Surveillance 4.1.2.3.2 which addresses same pumps establishes a 31 day interval for basically the same surveillance.

The motor circuit breaker wording was changed for the same reason as in Specification 4.1.2.3.2 and 4.1.2.4.2

3/4.5.4 BORON INJECTION SYSTEM

BORON INJECTION TANK

LIMITING CONDITION FOR OPERATION

3.5.4.1 The boron injection tank shall be OPERABLE with:

~~a. A contained borated water volume of between ____ and 900 gallons, and~~

~~b.~~ A boron concentration of between 2000 and ~~2100~~ 7000 ppm.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the boron injection tank inoperable, restore the tank to OPERABLE status within 72 hours or be in HOT STANDBY ~~and borated to a SHUTDOWN MARGIN equivalent to 1% delta k/k at 200°F~~ within the next 6 hours; restore the tank to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.1 The boron injection tank shall be demonstrated OPERABLE by:

~~a. Verifying the contained borated water volume at least once per 7 days, and~~

~~b.~~ Verifying the boron concentration of the water in the tank at least once per ~~7~~ 90 days.

Technical Specification: 3.5.4.1 Boron Injection Tank

Tech. Spec. 3.5.4.1.a

Justification:

Deletion of the contained volume is assumed as this volume is no longer needed to counteract any positive increases in reactivity associated with a RCS cool down. Filling and venting of the High Pressure Injection System will be accomplished as part of the normal line up of this system and will be controlled per procedure.

Tech. Spec. 3.5.4.1 ACTION Statement

Justification:

Reduction of the proposed action statement requirements and deletion of the boration requirements is proposed as the concentration is no longer as critical to ensure assumptions used in the steam line break analysis are met.

Tech. Spec. 4.5.4.1.a

Removal of the contained water volume is assumed based on the fact the concentration is the same as the balance of the High Pressure Injection System and the volume is no longer needed for the accident use. Filling and venting will be accomplished per the Operating Procedures controlling this system.

Tech. Spec. 4.5.4.1.b

Justification:

Extension of the boron concentration surveillance interval is based on the concentration being the same as the balance of the High Pressure Injection System. The source of fill water to this system is the RWST with boron concentrations of 2000 to 2100 PPMb. Plans are to also lock closed the valves from the boron injection make up pump to eliminate the concern for dilution.

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained borated water volume of at least 394,000 ~~between 407,000 and ___~~ gallons,

b. A boron concentration of between 2000 and (2100) ppm of boron,

c. A minimum solution temperature of 37°F, and

d. A maximum solution temperature of 100°F.

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

ACTION:

With the RWST 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

4.5.5 The RWST 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 RWST temperature when the outside air temperature is either less than 37°F or greater than 100°F.

(I)
Rev. 0

Specification: 3.5.5.a - See Attached Markup of Page 3/4.5-10

Justification:

This specification provides both a minimum and maximum RWST volume for ECCS operation. There is no need for a maximum water volume based on the SNUPPS design.

The RWST minimum assured volume requirement 394,000 gallons accounts for the location of the tank discharge line, instrumentation allowances and other physical characteristics.

The setpoint for the automatic switchover of the RHR pumps (LO-LO-1) ensures that sufficient water is accumulated in the containment to permit recirculation flow from the sump.

Additional water will be potentially available between the assured volume and the bottom of the RWST overflow. Recirculation phase sump pH calculations show that the addition of this water above the assured volume will not adversely affect the pH of the recirculated fluids (i.e., the pH remains between 8.5 and 11.0).

Tank overflow is not addressed in the Bases for this specification; however, any overflow is processed by the liquid radwaste system and will not be discharged to the environs prior to processing.

The RWST (shown on FSAR Figure 6.3-1) is provided with a 6" overflow which is permanently piped to the waste holdup tank (WHT) of the liquid radwaste system. The WHT is located in the radwaste building. The WHT (shown on FSAR Figure 11.2-1) is provided with high and high-high level alarms which are annunciated in the radwaste building control room. The waste evaporator feed pump takes suction from the WHT. Operation of this pump is activated by the WHT level switch and is monitored by the balance of plant computer.

Any overflow of the RWST will not adversely affect the operation of the ECCS nor will it result in an uncontrolled water discharge to the environs. Therefore, there are no basis for an LCO on the maximum RWST level.

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, 0.20% by weight of the containment air per 24 hours at P_a, 48 psig, or

2. Less than or equal to L_t, 0.10% by weight of the containment air per 24 hours at a reduced pressure of P_t, 24 psig.

b. A combined leakage rate of less than 0.60 L_a, 0.20% for all penetrations and valves subject to Type B and C tests, when pressurized to P_a, 48 psig. (Valves requiring Type C testing are indicated in Table 3.6-1.)

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 0.75 L_a or less than 0.75 L_t, as applicable, and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°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 50 using the methods and provisions of ANSI N45.4-~~(~~1972~~)~~: X

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at either P_a, 48 psig, or at P_t, 24 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;

Specification 3.6.1.2, Page 3/4 6-2

Justification -

The sentence is being added to point out that Table 3.6-1 shows the valves that require Type C testing.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

that the seal leakage is less than or equal to $.01 L_A$

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 ~~no detectable seal leakage by pressure decay~~ when the volume between the door seals is pressurized ~~to greater than or equal to P_a, 48 psig, for at least 15 minutes;~~ and stabalized at not less than 10 psig.

b. By conducting overall air lock leakage tests at not less than P_a, 48 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 50.

Specification 4.6.1.3.a Page 3/4 6-5

Justification -

The basis for Tech. Spec. 4.6.3 states: "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."

The seal test is more conservative than the overall airlock test in that 1) the pressure is applied across one seal as opposed to both seals during the overall test, 2) during the overall test, pressure is applied against the seat of the outside door and the strongbacks are installed on the inside door. Taking this conservatism into account, and the fact that the seal test is a check for physical damage and not overall seal degradation, a reduced test pressure of 10 psig would be adequate in establishing seal integrity. The limit of $0.01L_A$ is in keeping with the air lock limit of $\leq 0.05L_A$.

This spec will allow testing via either pressure decay, in leakage, or air flow.

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:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits 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

REPLACE WITH INSERT

4.6.1.6.1 Containment Tendons. The containment tendons' structural integrity shall be demonstrated at the end of 1, 3 and 5 years following the initial containment structural integrity test and at 5-year intervals thereafter. The tendons' structural integrity shall be demonstrated by:

a. Determining that a representative sample* of at least 4% but no less than four, of the U tendons each have a lift-off force of between _____ (minimum) and _____ (maximum) pounds at the first year inspection and that a representative sample* of at least 4%, but no less than nine, of the hoop tendons each have a lift off force of between _____ (minimum) and _____ (maximum) pounds at the first year inspection. For subsequent inspections, the maximum allowable lift-off forces shall be decreased from the value determined at the first year inspection by the amount: _____ log t and the minimum allowable lift-off force shall be decreased from the value determined at the first year inspection by the amount: _____ log t where t is the time interval in years from initial tensioning of the tendon to the current testing date. This test shall include an unloading cycle in which each of these tendons is detensioned to determine if any wires or strands are broken or damaged. Tendons found acceptable during this test shall be retensioned to their observed lift-off

*For each inspection, the tendons shall be selected on a random but representative basis so that the sample group will change somewhat for each inspection; however, to develop a history of tendon performance and to correlate the observed data, one tendon from each group (U and hoop) may be kept unchanged after the initial selection.

Rev.0

SURVEILLANCE REQUIREMENTS (Continued)

force, ±3%. During retensioning of these tendons, the change in load and elongation shall be measured simultaneously. If the lift-off force of any one tendon in the total sample population is out of the predicted bounds (less than minimum or greater than maximum), an adjacent tendon on each side of the defective tendon shall also be checked for lift-off force. If both of these adjacent tendons are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. This single tendon shall be restored to the required level of integrity. More than one defective tendon out of the original sample population is evidence of abnormal degradation of the containment structure. Unless there is evidence of abnormal degradation of the containment tendons during the first three tests of the tendons, the number of tendons checked for lift-off force and change in elongation during subsequent tests may be reduced to a representative sample of at least 2%, but no less than two, of the U tendons and a representative sample of at least 2%, but no less than three, of the hoop tendons; and

b. Removing one wire or strand from one U tendon and one hoop tendon checked for lift-off force and determining that over the entire length of the removed wire or strand that:

1) The tendon wires or strands are free of corrosion, cracks and damage,

2) There are not changes in the presence or physical appearance of the sheathing filler grease, and

3) A minimum tensile strength value of 240,000 psi (guaranteed ultimate strength of the tendon material) for at least three wire or strand samples (one from each end and one at mid-length) cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

REPLACE WITH INSERT

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SURVEILLANCE REQUIREMENTS (Continued)

REPLACE WITH INSERT

~~4.6.1.6.2 End Anchorages and Adjacent Concrete Surfaces. The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.6.1 and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages. Inspections of the concrete shall be performed during the Type A containment leakage rate tests (reference Specification 4.6.1.2) while the containment is at its maximum test pressure.~~

4.6.1.6.~~3~~2 Containment ~~Surfaces~~ Liner Plate System. The structural integrity of ~~the exposed accessible interior and exterior surfaces of the containment, including~~ the containment liner plate system, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.6.~~4~~3 Reports. Any abnormal degradation of the containment structure detected during the above required tests and inspections shall be reported to the Commission pursuant to Specification 6.9.1. 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.

(B)

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4.6.1.6.1 Containment Tendons. The structural integrity of the containment post-tensioning system shall be demonstrated by inservice inspection of tendons at one, two and three years following the initial structural integrity test (ISIT) and every five years thereafter. The tendons' structural integrity shall be demonstrated by:

a. Visual Inspections

1. The concrete exterior surface of the containment shall be visually examined to detect areas of wide-spread cracking, spalling, and grease leakage.

2. The anchorage assembly hardware including bearing plates, stressing washers, shims, and buttonheads of all surveillance tendons selected, as described in the Tendon Surveillance Program, shall be visually examined for abnormal material behavior according to procedures contained therein.

b. Prestress Monitoring Tests

1. All surveillance tendons selected, as described in the Tendon Surveillance Program, shall be subjected to liftoff tests to monitor their prestressing forces according to procedures contained therein.

REV.1

The prestressing force of each surveillance tendon in the te[illegible] shall lie above the prescribed lower limit for the time of the test, as indicated in the stress-relaxation curve for this tendon shown in the Tendon Surveillance Program .

If the prestressing force of a surveillance tendon lies between the prescribed lower limit and 90 percent 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 lie above the prescribed lower limits for these tendons, the single deficiency may be considered as unique and all three tendons shall be considered as acceptable. If the prestressing force of any of the adjacent tendons falls below the prescribed lower limits for the tendons, the condition shall be considered as reportable.

If the prestressing force of a surveillance tendon lies below 90 percent of the prescribed lower limit, the defective tendon shall be completely detensioned and a determination shall be made as to the cause of such occurrence. Such an occurrence shall be considered as a reportable condition.

(E) (E).1

2. One Tendon from each tendon group during each surveillance, as identified in the

Tendon Surveillance Program , shall be subjected to complete detensioning to identify broken or damaged wires according to procedures contained therein. One wire from each detensioned surveillance tendon shall be removed for testing and examination in accordance with Specification 4.6.1.6.1.c.

c. Tendon Material Tests and Inspections

1. All wires removed from each detensioned surveillance tendon shall be examined over the entire length to determine if evidence of corrosion or other deleterious effects are present.

2. Tensile tests shall be made on three samples cut from each removed wire (one at each end and one at mid-length). Failure in the tensile test at a strength or elongation value less than the minimum requirements of the tendon material shall be considered as reportable.

3. A grease sample obtained from each surveillance tendon which requires complete detensioning shall be analyzed for the amount of contaminants and water content. The presence of significant voids within the grease filler

(B) REV.1

material, the presence of the water, or chemical or physical properties outside the tolerance limits, as indicated in the Tendon Surveillance Program , shall be considered as reportable.

4. Other conditions found by tests or visual examination that indicate possible effects on the integrity of two or more tendons shall be considered as reportable.

Specification: 4.6.1.6.1, 4.6.1.6.2, 4.6.1.6.3 (B)
Pages 3/4 6-8 to 3/4 6-10

justification:

The proposed changes in SNUPPS Technical Specification Section 4.6.1.6 for the Callaway Plant and the Wolf Creek Plant are based on the following considerations:

1. Technical Specification Section 4.6.1.6.1, "Containment Tendon", as presently written, requires the use of a single maximum value and a single minimum value as the acceptance criteria for the lift-off tests for the entire population of surveillance tendons. This requirement necessitates using the lift-off value of the most optimum tendon as the maximum acceptable limit (upper limit) and the lift-off value of the least optimum tendon as the minimum acceptable limit (lower limit) for all the surveillance tendons. The jacking forces applied during stressing and the total shim thickness used in the lock-off of a tendon, as well as the number of effective wires and their length all have direct influence on the values of the subsequent lift-off tests for a tendon. The use of a single set of acceptance limits for all surveillance tendons, therefore, would not necessarily reflect any abnormal change in the tendons except those that were used in establishing the acceptance limits. This concern has been recognized by the NRC and is reflected in the proposed Revision 3 to Regulatory Guide 1.35, "In-Service Inspection of Ungrouted Tendons in Prestressed Concrete Containments".

   In order to properly account for the various factors affecting the tendon lift-off forces, and to establish reasonable tolerance limits for the acceptance criteria, stress relaxation curves with a band of tolerance for each surveillance tendon should be developed as recommended by the proposed Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments". Consequently, a change in Technical Specification Section 4.6.1.6.1 is required.

2. Technical Specification Section 4.6.1.6.1 requires that inspection of the surveillance tendons shall include an unloading cycle in which each of these tendons is detensioned to determine if any wires are broken or damaged. Since the effects of broken or damaged wires will most likely be reflected in the lift-off tests, complete detensioning of all the surveillance tendons seems to be unnecessarily restrictive, and the usefulness of such a method in identifying defective tendons is quite marginal and the cost is high. Consequently, the proposed Revision 3 to Regulatory Guide 1.35 requires detensioning of only one tendon for inspection of broken, damaged, or corroded wires. This approach appears to be appropriate and reasonable, and would result in considerable savings without reducing the effectiveness of the surveillance program. A revision to Technical Specification Section 4.6.1.6.1 is therefore required to incorporate this new requirement.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and~~:~~

~~a.~~ ~~E~~ach 36-inch containment shutdown purge supply and exhaust isolation valve shall be closed and blank flanged. ~~and~~

~~b. The 18-inch containment mini-purge supply and exhaust isolation valve(s) may be open for up to 500 hours during a calendar year.~~

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

ACTION:

a. With a 36-inch containment purge supply and/or exhaust isolation valve open or not blank flanged, close and/or blank flange that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~b. With the 18-inch containment mini-purge supply and/or exhaust isolation valve(s) open for more than 500 hours during a calendar year, close the open 18-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.~~

c. With a containment mini purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specifications 4.6.1.7.3 and/or 4.6.1.7.4, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours. THE PROVISIONS OF SPECIFICATION 3.0.4 DO NOT APPLY.

(OR INSTALL ITS' BLIND FLANGE)

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 36-inch containment shutdown purge supply and exhaust isolation valve(s) shall be verified blank flanged and closed ~~at least once per 31 days~~. PER SURVEILLANCE REQUIREMENTS OF 4.6.1.1.

~~4.6.1.7.2 The cumulative time that all 18-inch containment mini-purge supply and exhaust isolation valves have been open during a calendar year shall be determined at least once per 7 days.~~

~~4.6.1.7.3 At least once per 6 months on a STAGGERED TEST BASIS each blank flanged 36-inch containment shutdown purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.05 L_a when pressurized to P_a.~~

4.6.1.7.~~4~~ 2 At least once per ~~3~~ 6 months each 18-inch containment mini-purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.~~05~~ ~~01~~ .05 L_a when pressurized to P_a.

X

Specification 3.6.1.7 and 4.6.1.2 - Containment Systems
Page 3/4 6-3 and 3/4 6-11,12

Justification -

3.6.1.7.b deletion -

This specification places a maximum duration of 500 hours during which the 18" purge valves may be open during modes 1, 2, 3 and 4. Previous FSAR submittals justified the design basis of the system and the need for continuous purging of the containment. The SER supports the conclusions presented in the FSAR. This proposed LCO places an undue restriction on the safe operation of the plant and is contrary to the philosophy of Regulatory Guide 8.15 and 10CFR20.

Deletion of this specification is requested to allow the licensee to operate the low volume purge as required (up to 8760 hours or continuously) to maintain airborne activities and thus occupational exposures to as low as reasonably achievable (ALARA) levels.

The SNUPPS utilities expected that a specification restricting the operation of the low volume purge system might be promulgated due to various pressures as well as technical concerns for ensuring containment integrity following a design basis event. Hence, SNUPPS initiated an early review of the mini purge design and provided detailed justification on the design bases and the need for the system in FSAR Revision 6 dated August, 1981. Refer to Attachments A and B.

This information demonstrates the continuous use of the system helps ensure that the objectives of NRC regulations and guidelines are met. The assumptions used were in strict accordance with mandated assumptions previously provided by the NRC in NUREG 0017 for use in calculating airborne activities, effluent releases, doses to the public and cost-benefit analyses for radwaste systems. The NRC for the purposes of this specification calls the results "theoretical" in that the calculations do not reflect the first year of operation of the SNUPPS plants.

As stated in Attachment A, NUREG 0017 was used to predict the airborne activities. Although the NUREG 0017 assumptions may prove to be somewhat non representative of the SNUPPS actual operational experience, they do reflect substantial operating experience of PWRS. The industry has been required to use its findings and assumptions in the design and licensing of the plants currently in the design phase. The blanket terminology of "theoretical" is not considered appropriate.

There is no strictly safety related bases for annual purge durations of 500 hours (or larger as approved in other plant technical specifications).

There is no quantitative justification based on current licensing requirements or PRA methodology, for a 500 hour purge duration limit. The overall risk to the health and safety of the public is not significantly increased from even a qualitative standpoint by allowing the purge valves to be open up to a continuous basis. Technically, one would have to agree the purge valves will close: or operation for any period of time with the valves open would not be allowed. Therefore, the SNUPPS utilities view this issue as being one which ignores other NRC regulations and guidelines which protect the public and the plant operational staff.

By requesting that this specification be deleted, the SNUPPS utilities do not intend to imply that the low volume purge would necessarily be operated continuously. Its operation will be on an as required basis, however, the requirements for operation may change with time as fuel assemblies are replaced and leakage rates vary. As noted in the FSAR, the SNUPPS utilities plan to maintain a very clean plant and plan to maintain airborne activity levels to ALARA levels through frequent containment entries for inspection and maintenance.

In order to maintain airborne activities low by eliminating leaks, the ability to use the low volume purge on an as required basis is necessary; however, this practice is expected to result in the purge systems use on a less than continuous basis. The utilities request that they be allowed to use their best judgement and actual plant conditions to determine the frequency and duration of operation of the low volume purge system.

The utilities are prepared to submit the data/report described in Attachment C. This report, as stated in Attachment C, could be used by the NRC to help form a basis for the NRC determination of "either approval of continuous (low-volume) purge operation or a limit in the plant T.S. deferring the maximum number of hours per year during which the (low volume) purge system may be operated.

The SNUPPS utilities do not believe that the data from one year's operation would be statistically significant nor representative of any other year's operational requirements for the low-volume purge system. As noted above, many variables enter into the resultant airborne activities and the need to enter the containment. These variables and the need to enter the containment will change with time; however, they cannot be accurately predicted: nor can they be assumed to be representative during the first 12 months of operation.

Specification 4.6.1.7.1 Page 3/4 6-12
4.6.1.7.3 Page 3/4 6-12

Justification -

Primary containment integrity to ensure restricted release of radioactive materials from the containment atmosphere is, for all valves, blind flanges, automatic and deactivated automatic valves, assured by LCO 3.6.1.1 and the surveillance requirements of 4.6.1.1. Since above mode 3, the shutdown purge isolation valves are blind flanged, deactivated and leak rate tested, the LCO and Surveillance Requirements of 3/4 6.1.1 is adequate to prove operability.

Specification 4.6.1.7.4

Justification -

Seal materials for these valves are qualified and have a planned replacement program of a maximum of every 4 years. The proposed surveillance interval for these valves which is still four times more restrictive than maximum surveillance intervals for type B and C tests, will result in a reduction in man hour requirements, containment entries, preparation for work in a contaminated area and potential exposure to workers. The proposed interval is also consistent with surveillance requirements for containment air locks.

The use of the originally NRC proposed leakage rate acceptance criterion of .05 la is desired in lieu of the most recently proposed-NRC acceptance criterion of .01 la.

The total acceptable leakage rate from type B and C penetrations (defined in LCO 3.6.1.2.b) is .6 La as required by 10CFR50 Appendix J. These tests are required to be conducted at intervals no greater than 24 months except for air locks and purge valves (surveillance requirement 4.6.1.2.d). These leakage rates help ensure that the total leakage from the containment is less than La (0.2%) during the first day following a LOCA.

The more frequent testing of the purge valves is required by the NRC to ensure that the resilent seals are not deteriorating due to aging processes. The use of .05 La for the purge valve penetration is more than adequate for detecting seal deterioration.

The seal material properties and aging mechanisms have been reviewed in detail. These seals are conservatively qualified for 4 years of power operation followed by a LOCA in accordance with the assumptions required by NUREG 0588 as described in the SNUPPS Environmental Qualification Report. The seals will be replaced in accordance with their qualified life requirements.

(4.6.1.7.4 Justification Con't.)

From a radiological standpoint a leakage rate of .05 La for the purge valve penetration is extremely conservative and acceptable. The SNUPPS radiological consequences calculation for a LOCA assume that 0.2% leakage is released to the atmosphere at ground level. No credit is taken for any filtration and elevated release by the emergency exhaust filtration system which draws a negative pressure on the entire auxiliary building. The conservatism of this assumption is confirmed by industry experience with type A testing of the containment structures which show no leakage through the containment liner plate. All measured leakage has been through containment penetrations, however, the NRC has not accepted qualitative arguments for leakage partition factors for partial secondary containment designs since the partition factors can be based only on engineering judgement.

In the case of purge valve leakage, any leakage through the purge valves will be processed by the auxiliary building emergency exhaust filtration system which is required to be operable in accordance with LCO 3.7.7 with a filtration efficiency of 95% and penetration and bypass leakage rate of less than 1%.

In addition, since two valves are provided in series, the leakage through the penetration is the amount that can escape through the valve with the lowest leak rate. No leakage would occur, for example, if one valve leaked at .05 La and the other were leak tight.

In summary .05 La is more than adequate from both a radiological standpoint and for detecting the deterioration of the resilent seal material.

MINI PURGE
DESIGN BASIS
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9.4.6.1 Design Bases

9.4.6.1.1 Safety Design Bases

Except for an associated containment penetration, the hydrogen mixing fans, and the containment cooling system, and the containment HVAC systems are not safety related. A complete description of the design of the containment cooling system and containment hydrogen mixing system is provided in Section 6.2.2.2.

SAFETY DESIGN BASIS ONE - The containment isolation valves in the system are selected, tested, and located in accordance with the requirements of 10 CFR 50, Appendix A, General Design Criteria 54 and 56, and 10 CFR 50, Appendix J, Type C testing.

SAFETY DESIGN BASIS TWO - The containment purge system containment isolation valves are capable of rapid closure, following their respective DBA (FHA for the shutdown purge valves and LOCA for the minipurge valves), to limit the escape of fission products from the containment.

9.4.6.1.2 Power Generation Design Bases

POWER GENERATION DESIGN BASIS ONE - The containment shutdown purge system is designed to maintain a containment ambient air temperature between 50 and 90 F, when the reactor is shut down. The shutdown purge system supplies fresh air into the containment at a rate of approximately one containment volume air change per every 2 hours for fresh air ventilation.

POWER GENERATION DESIGN BASIS TWO - The design basis flow of 4,000 cfm is based on continuous system operation with an assumed weekly occupancy of 5 hours for any one individual.

The assumptions used in determining the flow rate and resultant airborne activities are consistent with NUREG-0017, Reference 7, Section 12.2. Table 12.2-11 provides the assumed RCS specific activities, failed fuel percentages, RCS leakage rates, and partition factors. Table 12.2-12 provides the airborne concentrations within the containment, assuming a continuous 4,000 cfm purge.

Any individual is allowed to be exposed to the concentrations of Table I Column I of 10 CFR 20, Appendix B, for 40 hours per week or to greater concentrations for a corresponding lesser amount of time. The design bases for the minipurge results in the most limiting factor being approximately 7 times those listed in Table I, Column I and therefore occupancy for an individual would be allowed for nearly 6 hours. In addition, the philosophy of Regulatory Guide 8.15 is to minimize the requirement for wearing respirators through improved ventilation. Therefore, not using the minipurge would be contrary

DESIGN BASIS
2 of 2

to the philosophy of both 10 CFR 20 and Regulatory Guide 8.15. In order to pass the required flow of 4,000 cfm and use only one set of valves in accordance with the recommendations of BTP CSB-4, an 18-inch isolation valve was utilized in lieu of the recommended 8-inch valve.

Good engineering practice limits the flow velocities and pressure drops through system valves. With an 18-inch line (velocity = 2,264 fpm), the design flow can be maintained by the supply and exhaust fans designed for a differential pressure of 4.25 and 5.0 inches w.g., respectively. If the system lines remained as designed and the isolation valves were replaced with 8-inch valves with reducers on either side, the supply and exhaust system pressure drops would increase to 9.02 and 10.5 inches w.g. at the design flow. Since the fans cannot create these high differential pressures, the design flow would not be realized and the system would not perform its design function.

The charcoal adsorbers in the discharge of the system comply with Regulatory Guide 1.140, to the extent discussed in Table 9.4-3.

POWER GENERATION DESIGN BASIS THREE - The CRDM cooling system is designed to limit the normal ambient temperature within the CRDM shroud to approximately 170 F by inducing 120 F containment air for cooling. The cooling of this air is provided by the containment coolers.

NEED FOR CONTINUOUS PURGE

and through the neutron detector wells. One operating fan has the capability to provide the necessary airflow. The ultimate cooling is provided by the containment coolers. The effluent air temperature from one reactor vessel support, from one detector well, and in one upper cavity region exhaust leg is monitored by the plant computer. In addition, temperature elements are embedded in the cavity, below each reactor vessel support, to monitor concrete temperature.

The pressurizer cooling fan is located near the bottom of the pressurizer compartment. The fan takes suction from the lower region of the pressurizer compartment (and therefore the coolest) and through the ductwork and discharges it in the area immediately below the pressurizer skirt. The fan will operate only when the associated containment cooler is out of service.

The machine room exhaust fan is located on the roof of the machinery equipment room and takes suction from the room. Makeup air is induced from the containment through transfer grilles located in the walls of the room. The machine room exhaust fan will operate during normal plant operations and during shutdown. It should not be operated during ILRT, to prevent overloading of the fan motor.

Cooling water for the shutdown purge supply unit is supplied by the central chilled water system (Section 9.4.10). Hot water for both the containment shutdown purge supply unit and the containment minipurge supply unit is supplied by the plant heating system (Section 9.4.9).

Discussed below are the power generation operations and shutdown operations of the containment HVAC systems. Because the emergency operation consists only of closing the containment isolation valves, it is discussed under the power generation and shutdown operations.

POWER GENERATION OPERATION - The minipurge system is designed to minimize occupational exposures to as-low-as-reasonably-achievable (ALARA) levels. Instead of personnel entering the containment with airborne activities much greater than MPC and at odds with the philosophy of Regulatory Guide 8.15, the containment will be purged to reduce airborne radioactivity concentrations and exposures in line with the philosophy of 10 CFR 20 and Regulatory Guide 8.15. The minipurge system is designed to be operated continuously to achieve these objectives. The need for continuous operation includes consideration for planned and unplanned entires into the containment and the need to periodically vent excess air from the containment to maintain the pressure near atmospheric conditions.

NEED FOR
CONTINUOUS
PURGE 2 of 2

a. Preplanned Entries

During the first years of commercial operation, daily entry into the containment is planned. This frequency is used by other PWRs. These entries would be from ½ to 1½ hours in length, depending on the conditions found within the containment. This type of operation would allow correction of leaks, (much smaller than the Technical Specification limits). Early correction, prior to the formation of large mounds of boric acid crystals and the release of significant amounts of radioactivity, will enhance the overall ALARA program at the plant.

b. Unplanned Entries

Unplanned entires inlcude those responding to abnormal indications from within the containment. These indications include leaks, equipment malfunctions, and instrumentation failures. Since these failures could have a significant impact on the continued safe operation of the plant, immediate response is most preferable. Without the continuous operation of the minipurge, the doses received from containment entries will be much higher, unless entries are delayed for significant amounts of time. For instance, if the containment had not been purged for 2 weeks, it would take 65 hours to bring the airborne activity down to the same levels as those maintained with its operation.

c. Containment Pressure Reduction

Instrument air is continuously being vented to the containment from air-operated valves. These valves also dump the air from their accumulators upon actuation. In order to maintain the containment pressure near atmospheric conditions, the minipurge system will be used to release excess air.

One operating plant has experienced over a 1 psig pressure buildup in 24 hours. If this rate were experienced at the SNUPPS units, the containment would have to be vented at least every other day to maintain the containment pressure within the Technical Specification limit of +2 psig.

SEP 11 1981

Lic.
ATTACH. C
1 of 4

MEMORANDUM FOR: R. Tedesco, Assistant Director for Licensing, DL

FROM: L. Rubenstein, Assistant Director for Core and Containment Systems, DSI

SUBJECT: SNUPPS: INPUT FOR SAFETY EVALUATION REPORT

Plant Name: Callaway Plant, units 1 and 2
Wolf Creek, Unit 1
Docket Nos.: 50-483, -486 and -482
Licensing Stage: OL
Responsible Branch: LB #1
Project Manager: G. Edison
Review Status: Complete

Enclosure 1 is our input for the Safety Evaluation Report (SER) on the SNUPPS application. It was prepared by the Containment Systems Branch (CSB) and its contractors, after having reviewed the applicable portions of the FSAR. The bases used in the review are contained in SRP Sections 6.2.1, 6.2.2, 6.2.4, 6.2.5 and 6.2.6. Portions of the SER were prepared by our consultant, the Lawrence Livermore National Laboratory, through its subcontractor, Energy Incorporated, under the direction of the CSB. We have reviewed their input and concur in their findings.

The enclosed SER is applicable to both the Callaway and Wolf Creek nuclear generating stations, and can be made plant-specific by inserting Callaway or Wolf Creek where reference is made to SNUPPS.

In addition, because the applicant has not yet submitted its proposed Technical Specifications (T.S.), the requirements of SRP Section 6.2 for reviewing pertinent T.S. have not been met and hence have not been specifically addressed in our SER. We request that the CSB be given the opportunity to review the applicant's proposed T.S. to ensure the areas related to containment systems are appropriately covered.

Although we do not have any open items in the enclosed SER, we have two contingencies; one appears in Section 6.2.1, regarding the limited pipe break sizes for subcompartment analysis and the other appears in Section 6.2.4, regarding operability of the purge valves. The review responsibility for these areas rests with the MEB and EQB, respectively. The CSB should be informed if their review findings are not consistent with those in the enclosed SER.

Contact: Y. Huang, CSB:DSI
24204

Furthermore, with reference to the containment purge system, we require in the enclosed SER that the safety related function of the system and the need for continuous operation be demonstrated by plant operating experiences. The applicant must demonstrate during the first year of plant operation that operation of the mini-purge system for more than 90 hours per year is required. This demonstration, to be documented in a report to the NRC not later than 15 months following start of commercial operations, must justify, based on operational data, the minimum number of hours per year that mini-purge system operation is needed to allow reasonable containment access (planned and unplanned) and necessary containment venting. The result of the NRC review of this report will be either approval of continuous mini-purge system operation or a limit in the plant T.S. defining the maximum number of hours per year during which the mini-purge system may be operated. *

Enclosure 2 provides the names of those individuals who have participated in the review of the SNUPPS application in the area of containment systems.

L S Rubenstein

Lester S. Rubenstein, Assistant Director
for Core and Containment Systems
Division of Systems Integration

Enclosures:
As stated

cc: R. Mattson
D. Eisenhut
W. Butler
B. J. Youngblood
G. Edison (5)
J. Shapaker
L. Kripps (EI)
Y. Huang
P. Triplett

CONTAINMENT SYSTEMS BRANCH

INPUT FOR SAFETY EVALUATION REPORT

CALLAWAY PLANT, UNITS 1 AND 2

WOLF CREEK, UNIT 1

DOCKET NOS. 50-483, -486 AND -482

ENCLOSURE 1

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introduction that "there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified." For SNUPPS, although GDC 57 states that a simple check valve is not an acceptable automatic isolation valve when located outside containment, the check valve is judged to be acceptable for containment isolation in light of other considerations. These considerations include the high quality design of the secondary system inside containment, which is the first isolation barrier, and the availability of other power operated valves to provide ba.kup isolation. We find that these containment isolation provisions are acceptable and that the requirements of GDC 57 have been met.

The containment isolation system meets the provisions of Regulatory Guides 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," 1.29, "Seismic Design Classification," and 1.141, "Containment Isolation Provisions for Fluid Systems".

Contingent on the acceptability of the valve operability assurance program, the containment isolation provisions of the containment atmosphere purge system conform to the provisions of BTP CSB 6-4, "Containment Purging During Normal Plant Operation," with one exception. The exception is that the size of the containment on-line purge (minipurge) system lines is 18 inches in diameter instead of 8 inches or less. The applicant has provided information detailing the minipurge system design requirements for an 18-inch line versus a smaller line and justifying, on a theoretical basis, the need for continuous minipurge system operation. We find that the 18-inch minipurge system design is acceptable. However, the safety related function of the system and the need for continuous operation must be demonstrated by the plant operating experience. Furthermore, as a result of staff study of valve leakage due to seal deterioration, leakage integrity tests must be conducted periodically. Testing frequency for the purge valves will be included in the plant technical specifications.

We conclude that the containment isolation system meets the requirements of GDCs 54, 55, 56, and 57, satisfies the provisions of Regulatory Guide 1.141, and conforms to all staff positions and industry codes and standards, and is therefore acceptable.

ATTACH 1

FINAL SER NUREG-0830

Compliance with GDC 57

The containment isolation system meets the explicit requirements of GDC 57, except in the case where simple check valves are used as automatic isolation valves outside containment in the auxiliary feedwater pump discharge lines. GDC 57 does not contain the statement that permits alternate containment isolation provisions on an "other defined basis." However, Appendix A to 10 CFR Part 50 states, in part, in the introduction that "...there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified." Although GDC 57 states that a simple check valve is not an acceptable automatic isolation valve when it is located outside containment, for Callaway the check valve is judged to be acceptable for containment isolation in light of other considerations. These considerations include the high quality design of the secondary system inside containment, which is the first isolation barrier, and the availability of other power-operated valves to provide backup isolation. The staff finds that these containment isolation provisions are acceptable and that the requirements of GDC 57 have been met.

The containment isolation system meets the provisions of Regulatory Guides 1.26, 1.29, and 1.141.

Contingent on the acceptability of the valve operability assurance program, the containment isolation provisions of the containment atmosphere purge system conform to the provisions of BTP CSB 6-4, "Containment Purging During Normal Plant Operation," with one exception. The exception is that the size of the containment online purge (minipurge) system lines is 18 in. in diameter instead of 8 in. or less. The applicant has provided information detailing the minipurge system design requirements for an 18-in. line versus a smaller line and justifying, on a theoretical basis, the need for continuous minipurge system operation. The staff finds that the 18-in. minipurge system design is acceptable. However, the safety-related function of the system and the need for continuous operation must be demonstrated by plant operating experience. Furthermore, as a result of staff study of valve leakage due to seal deterioration, leakage integrity tests must be conducted periodically. Testing frequency for the purge valves will be included in the plant Technical Specifications.

The staff concludes that the containment isolation system meets the requirements of GDCs 54, 55, 56, and 57, satisfies the provisions of Regulatory Guide 1.141; and conforms to all staff positions and industry codes and standards. It is, therefore, acceptable.

6.2.4 Combustible Gas Control System

After a LOCA, hydrogen may accumulate within containment as a result of (1) metal-water reaction between the zirconium fuel cladding and the reactor coolant, (2) radiolytic decomposition of the water in the reactor core and the containment sump, and (3) corrosion of metals by emergency core cooling and contaiment spray solutions. The applicant has provided a hydrogen control system (HCS) to monitor and control the hydrogen concentration and ensure uniform mixing of the hydrogen in containment after a LOCA. The HCS consists of redundant hydrogen recombiners, a redundant hydrogen monitoring system, a redundant hydrogen mixing system, and a hydrogen purge system.

B

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 Containment Spray System capable of taking suction from the RWST and transferring suction to the containment sump.

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

ACTION:

With one Containment Spray System inoperable, restore the inoperable Containment Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Containment 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 that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 265 psig when tested pursuant to Specification 4.0.5;

c. 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 Pressure-High (CSAS) test signal, and

   2) Verifying that each spray pump starts automatically on a Containment Pressure-High (CSAS) test signal.

~~d. 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.~~

B - NEW 4-27-83

Specification 4.6.2.1

Justification:

The deletion of this surveillance requirement is requested because the SNUPPS design features and system design precludes the need for verifying that the spray nozzles are clear of obstructions.

As described in FSAR Section 6.2.2.1.4, Tests and Inspections, the spray nozzles will be verified to be clear of obstruction during preoperational testing. Following this test there is no identified means for debris of any significant size to enter the nozzles. The following considerations support this statement.

1. The system periodic test required by surveillance requirement 4.6.2.1.b does not admit water to the portion of the system inside the containment.

2. The spray header piping inside containment is drained of water during power operation.

3. The Whirljet nozzles have an opening of 7/16" which is sufficiently large to prevent clogging by any foreseable objects which could be in the system from the RWST to the containment.

4. Water will not enter the spray nozzles unless an accident condition causes the system to be actuated automatically or manually. Inadvertent operation is precluded by design as described in FSAR Section 6.2.1.1.3.j.

5. Even if the spray system were operated with suction from the containment sump the debris which can enter the system is limited to 1/8" (Refer to FSAR Table 6.2.2-1). Since the nozzle openings are 7/16 inch, they would not be clogged.

Based on the above considerations clogging of the spray nozzles is precluded by design of the SNUPPS system and this surveillance requirement is considered to be unnecessary for ensuring the proper operation of the system following an accident condition which would require its use.

CONTAINMENT SYSTEMS

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE with:

a. A spray additive tank containing a volume of between 4340 and 4540 gallons of between ______ and ______ percent by weight NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a containment spray system pump flow.

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

ACTION:

With the Spray Additive 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 spray additive 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 Spray Additive 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 NaOH solution by chemical analysis.

c. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Hi-3 (CSAS) signal; and

d. At least once per 5 years by verifying ~~each solution flow rate (to be determined during pre-operational tests) from the following drain connections in the Spray Additive System:~~

   ~~1) (Drain line location) ______ ± ______ gpm~~

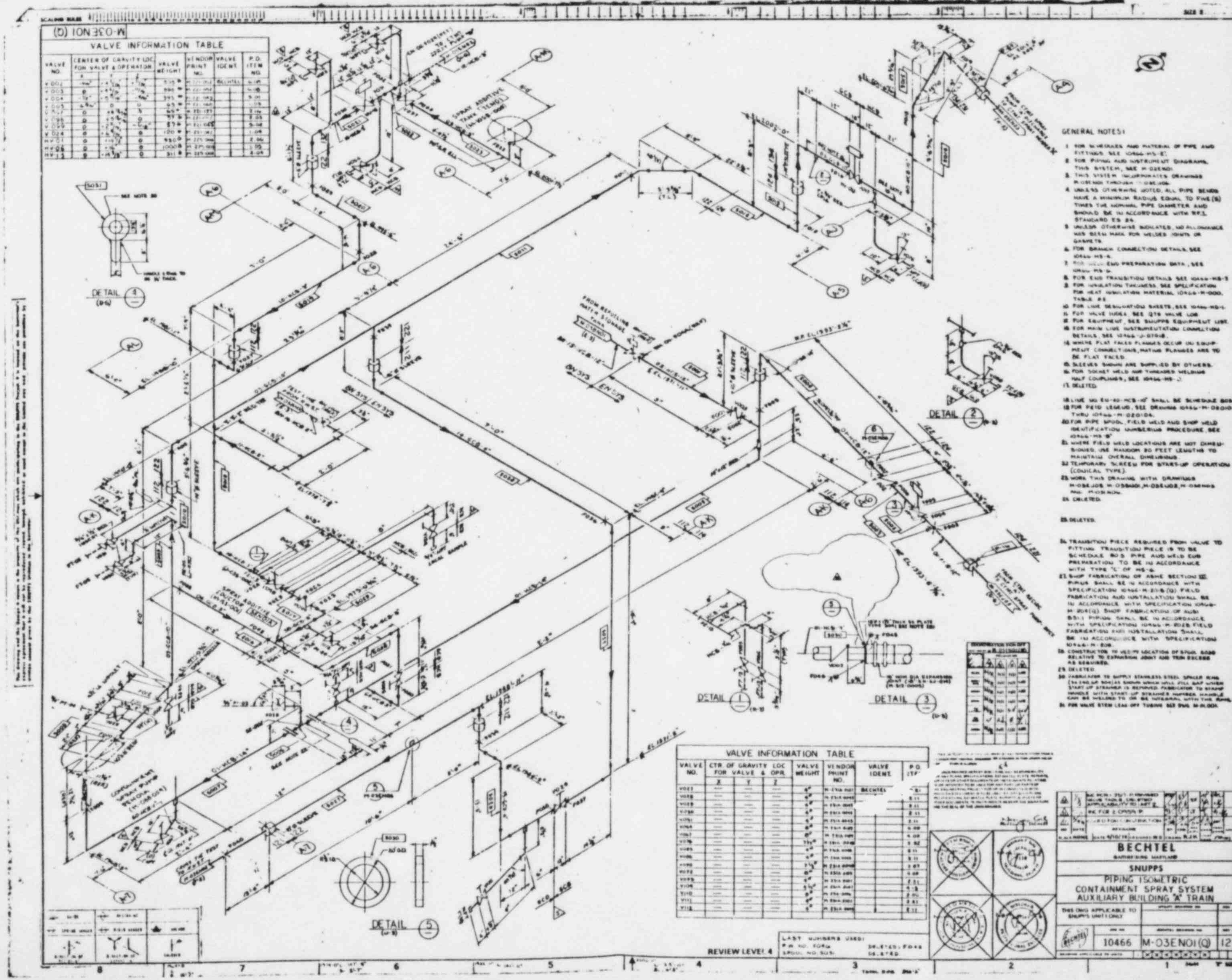
   ~~2) (Drain line location) ______ ± ______ gpm~~

   1) Each eductor flow rate is greater than or equal to 52 gpm using the RWST as the test source throttled to 17 psig at eductor inlet, and

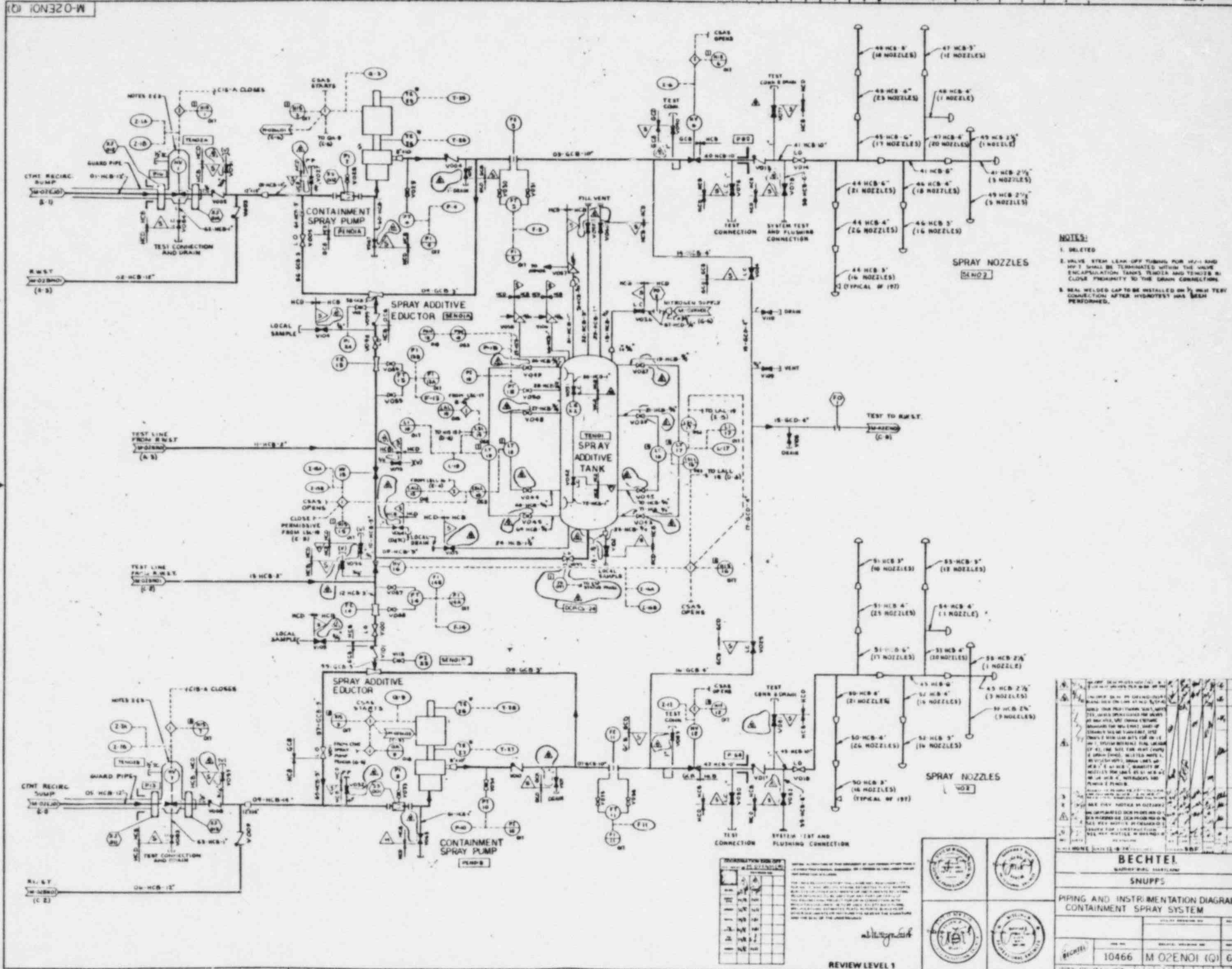
   2) The lines between the spray additive tank and the eductors are not blocked by verifying flow.

Specification: 4.6.2.2.d

Justification:
A new 4.6.2.2.d was added which will allow SNUPPS to prove the capability of injecting NaOH from the spray additive tank to the eductor without risking contamination of the RWST water with NaOH solution. See Bechtal drawings: M-02EN01, M-03EN01, and M-03EN02.



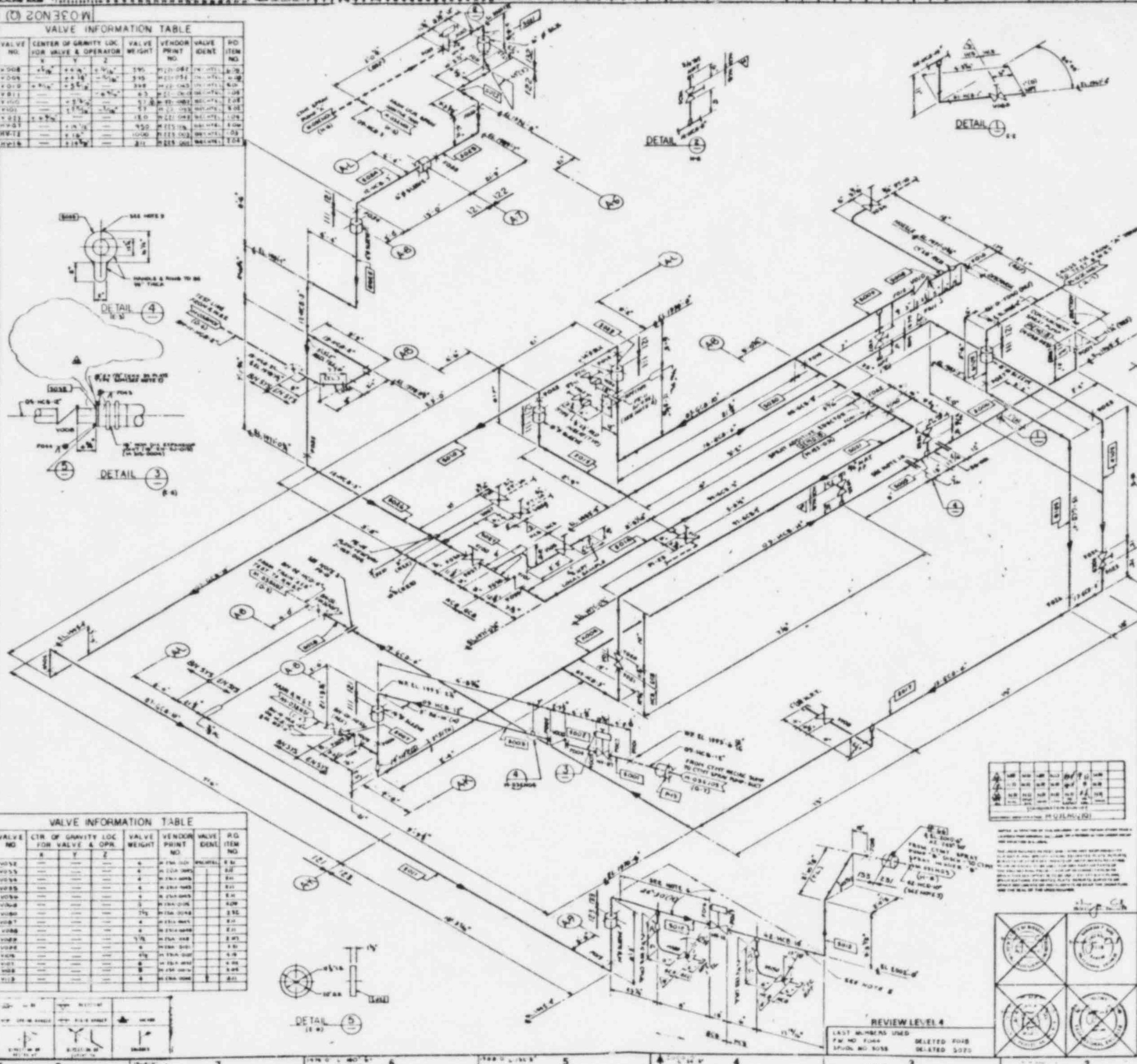
M-03EN01 (Q)
VALVE INFORMATION TABLE
DETAIL 1
DETAIL 2
DETAIL 3
DETAIL 4
DETAIL 5
SPRAY ADDITIVE TANK (TEN01)
GENERAL NOTES:
VALVE INFORMATION TABLE
REVIEW LEVEL 4
BECHTEL
GAITHERSBURG, MARYLAND
SNUPPS
PIPING ISOMETRIC
CONTAINMENT SPRAY SYSTEM
AUXILIARY BUILDING "A" TRAIN
THIS DWG APPLICABLE TO SNUPPS UNIT ONLY
10466
M-03EN01(Q)
12


M-02EN01 (Q)
CONTAINMENT SPRAY PUMP
SPRAY ADDITIVE EDUCTOR
SPRAY ADDITIVE TANK
SPRAY NOZZLES
CTMT RECIRC. SUMP
GUARD PIPE
TEST CONNECTION AND DRAIN
RWST
TEST LINE FROM R.W.S.T.
LOCAL SAMPLE
FILL VENT
NITROGEN SUPPLY
TEST TO R.W.S.T.
TEST CONNECTION
SYSTEM TEST AND FLUSHING CONNECTION
CSAS OPENS
CSAS STARTS
NOTES:
1. DELETED
2. VALVE STEM LEAK-OFF TUBING FOR HV-1 AND HV-7 SHALL BE TERMINATED WITHIN THE VALVE ENCAPSULATION TANKS TEN01A AND TEN01B IN CLOSE PROXIMITY TO THE DRAIN CONNECTIONS.
3. SEAL WELDED CAP TO BE INSTALLED ON 3/4" HIGH TEST CONNECTION AFTER HYDROTEST HAS BEEN PERFORMED.
BECHTEL
SNUPPS
PIPING AND INSTRUMENTATION DIAGRAM CONTAINMENT SPRAY SYSTEM
10466
M-02EN01 (Q) 6
REVIEW LEVEL 1

M-03EN02 (Q)

VALVE INFORMATION TABLE

VALVE NO.	CENTER OF GRAVITY LOC FOR VALVE & OPERATOR X	Y	Z	VALVE WEIGHT	VENDOR PRINT NO.	VALVE IDENT	PO ITEM NO.
V008	[illegible]	[illegible]	[illegible]	395	[illegible]	[illegible]	[illegible]
V009	—	[illegible]	[illegible]	395	[illegible]	[illegible]	[illegible]
V010	[illegible]	[illegible]	—	398	[illegible]	[illegible]	[illegible]
V011	—	—	[illegible]	63	[illegible]	[illegible]	[illegible]
[illegible]	—	[illegible]	—	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	—	[illegible]	[illegible]	57	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	—	—	150	[illegible]	[illegible]	[illegible]
[illegible]	—	[illegible]	—	950	[illegible]	[illegible]	[illegible]
HV-12	—	[illegible]	—	1000	[illegible]	[illegible]	[illegible]
HV-14	—	[illegible]	—	211	[illegible]	[illegible]	T04



NOTES:

1. FOR GENERAL NOTES & REFERENCES SEE DWG. M-03EN01.
2. WORK THIS DRAWING WITH DRAWINGS M-03EN01, M-03EN01, M-03EN03, M-03EL05, M-03EN02, [illegible]
3. FO-D: 300# RF/WN INSTRUMENT TAPS ON NORTH SIDE.
4. FE-11 300# RF/WN INSTRUMENT TAPS ON WEST SIDE.
5. EN-42-HCB-10" SHALL BE SCHEDULE 80 S.
6. TRANSITION PIECE TO BE SCHEDULE 80S PIPE WELD END PREP TO BE IN ACCORDANCE WITH TYPE "C" OF PS-4.
7. CONSTRUCTOR TO VERIFY LOCATION OF SPOOL 5032 RELATIVE TO EXPANSION JOINT AND TRIM EXCESS AS REQUIRED.
8. DELETED
9. FABRICATOR TO SUPPLY 1/2 SPACER RING [illegible] WHICH WILL FILL GAP WHEN START-UP STRAINER IS REMOVED. FABRICATOR TO STAMP HANDLE WITH START-UP STRAINER NO. HANDLE CAN BE WELDED TO OR INTEGRAL WITH THE RING.
10. TEMPORARY SCREEN FOR START-UP OPERATION (CONICAL TYPE)

VALVE INFORMATION TABLE

VALVE NO	CTR OF GRAVITY LOC FOR VALVE & OPR. X	Y	Z	VALVE WEIGHT	VENDOR PRINT NO	VALVE IDENT	P.O. ITEM NO
V052	—	—	—	4	[illegible]	[illegible]	[illegible]
V053	—	—	—	4	[illegible]	[illegible]	[illegible]
V054	—	—	—	4	[illegible]	[illegible]	[illegible]
V055	—	—	—	4	[illegible]	[illegible]	[illegible]
V056	—	—	—	4	[illegible]	[illegible]	[illegible]
[illegible]	—	—	—	5	[illegible]	[illegible]	[illegible]
[illegible]	—	—	—	7½	[illegible]	[illegible]	[illegible]
V087	—	—	—	4	[illegible]	[illegible]	[illegible]
V088	—	—	—	4	[illegible]	[illegible]	[illegible]
[illegible]	—	—	—	7½	[illegible]	[illegible]	[illegible]
[illegible]	—	—	—	4	[illegible]	[illegible]	[illegible]
[illegible]	—	—	—	4½	[illegible]	[illegible]	[illegible]
V107	—	—	—	8	[illegible]	[illegible]	[illegible]
V108	—	—	—	8	[illegible]	[illegible]	[illegible]
V112	—	—	—	4	[illegible]	[illegible]	[illegible]

REVIEW LEVEL 4

LAST NUMBERS USED
FW NO. F044 | DELETED F048
SPOOL NO. 5058 | DELETED 5025

BECHTEL

SNUPPS

PIPING ISOMETRIC
CONTAINMENT SPRAY SYSTEM
AUXILIARY BUILDING - "B" TRAIN

THIS DWG APPLICABLE TO SNUPPS UNIT 1 ONLY

10466 | M-03EN02 (Q) | 12

K

CONTAINMENT COOLING SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.6.2.3 Two independent groups of containment cooling fans shall be OPERABLE with two fan systems to each group.

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

ACTION:

a. With one group of the above required containment cooling fans inoperable and both Containment Spray Systems OPERABLE, restore the inoperable group of cooling fans 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 two groups of the above required containment cooling fans inoperable and both Containment Spray Systems OPERABLE, restore at least one group of cooling fans 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. Restore both above required groups of cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one group of the above required containment cooling fans inoperable and 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 and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable group of containment cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of containment cooling fans shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1) Starting each non-running fan group from the control room, and verifying that each fan group operates for at least 15 minutes, and

   2) Verifying a cooling water flow rate of greater than or equal to 2200 gpm to each cooler group. Δ

b. At least once per 18 months by verifying that on a Safety Injection test signal, the fans start in slow speed or, if operating, shift to slow speed and the cooling water flow rate increases to 4000 gpm to each cooler group.

Specification: 4.6.2.3.a.1,2

Justification:

Some containment cooling fans will be running to maintain satisfactory containment temperature. The only ones which should have to be started are those not already on.

The SNUPPS design provides flow monitoring to a group of fans (2 fans per group). Thus SNUPPS cannot monitor 2200 gpm per cooler, but rather 2200 gpm to a cooler group.

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:

With one or more of the containment 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:

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 each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, ~~or~~ OTHERWISE

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. THE PROVISIONS OF SPECIFICATION 3.0.4 ARE NOT APPLICABLE.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The containment 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 by performance of a cycling test, and verification of isolation time.

Technical Specification 3.6.3 Containment Isolation Valves
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Justification -

Excepting the provision of specification 3.0.4 allow power operations to be reestablished and operation to continue once the effected penetration has been isolated and deactivated. Operations with some penetrations isolated can be accomplished safely by utilizing alternate flow paths or by simply not utilizing the effected penetrations function. The isolated penetration represents no additional safety concerns as it remains in its' safeguard or isolated position until maintenance can be performed and the affected penetration declared operable.

Specification Table 3.6-1, Pages 3/4 6-18 to 23

Justification -

The changes and additions are being made to this table to clarify our commitments for Type C testing of containment isolation valves and to update the table. It is also being reorganized in alphanumeric order according to valve numbers within each section of the table to provide easy location of the valve(s).

TABLE 3.6-1
CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (SECONDS)
1. Phase "A" Isolation (active)				
P-62	BB HV-8026	PRT Nitrogen Iso Valve	C	10
P-62	BB HV-8027	PRT Nitrogen Iso Valve	C	10
P-24	BG HV-8100	Seal Water Return CTMT Iso Valve	C	10
P-24	BG HV-8112	Seal Water Return CTMT Iso Valve	C	10
P-23	BG HV-8152	Letdown System CTMT Iso Valve	C	10
P-23	BG HV-8160	Letdown System CTMT Iso Valve	C	10
P-25	BL HV-8047	Reactor Makeup Water CTMT Iso Valve	C	10
P-21	EJ HCV-8825	RHR to SI Test Line Iso Valve	A	10
P-82	EJ HCV-8890A	RHR A to SI Pumps Test Line Iso Valve	A	13
P-27	EJ HCV-8890B	RHR B to SI Pumps Test Line Iso Valve	A	13
~~P-15~~	~~EJ HV-21~~	~~CTMT Recirc Sump to PASS~~		
~~P-14~~	~~EJ HV-22~~	~~CTMT Recirc Sump to PASS~~		
P-49	EM HV-8823	SI/Accumulator Injection Test Line Iso Valve	A	10
P-48	EM HV-8824	Safety Injection Pump B Test Line Iso Valve	A	10

~~*May be opened on an intermittent basis under administrative control.~~

*These valves were assumed to be closed during the accident analysis, and are normally closed but may be opened on an intermittant basis under administrative control.

TABLE 3.6-1 (Continued)
CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (SECONDS)
P-88	EM HV-8843	Boron Injection Upstream Test Line Iso	A	10
P-92	EM HV-8871	SI Test Line to RWST Iso Valve	C	10
P-87	EM HV-8881	Safety Injection Pump A Test Line Iso Valve	A	10
P-92	EM HV-8964	SI Test Line System Outside CTMT Iso	C	10
P-99	GS HV-3	Hydrogen Analyzer B Inlet Iso	A,C	5
P-99	GS HV-4	Hydrogen Analyzer B Inlet Iso	A,C	5
P-99	GS HV-5	Hydrogen Analyzer B Inlet Iso	A,C	5
P-56	GS HV-8	Hydrogen Analyzer B Disch Iso	A,C	5
P-56	GS HV-9	Hydrogen Analyzer B Disch Iso	A,C	5
P-101	GS HV-12	Hydrogen Analyzer A Inlet Iso	A,C	5
P-101	GS HV-13	Hydrogen Analyzer A Inlet Iso	A,C	5
P-101	GS HV-14	Hydrogen Analyzer A Inlet Iso	A,C	5
P-97	GS HV-17	Hydrogen Analyzer A Disch Iso	A,C	5

~~*May be opened on an intermittent basic under administrative control.~~

*These valves were assumed to be closed during the accident analysis, and are normally closed but may be opened on an intermittant basis under administrative control.

TABLE 3.6-1 (Continued)
CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (SECONDS)
P-97	GS HV-18	Hydrogen Analyzer A Disch Iso	A,C	5
P-101	GS HV-31	Sample Line to CTMT Atmos Monitor	A,C	5
P-101	GS HV-32	Sample Line to CTMT Atmos Monitor	A,C	5
P-97	GS HV-33	Hydrogen Sample Return From PASS	A,C	5
P-97	GS HV-34	Hydrogen Sample Return From PASS	A,C	5
P-99	GS HV-36	Sample Line to CTMT Atmos Monitor	A,C	5
P-99	GS HV-37	Sample Line to CTMT Atmos Monitor	A,C	5
P-56	GS HV-38	Sample Return CTMT Atmos Monitor	A,C	5
P-56	GS HV-39	Sample Return CTMT Atmos Monitor	A,C	5
P-44	HB HV-7126	RCDT Vent Inside CTMT	C	10
P-26	HB HV-7136	RCDT Pumps Disch Hdr Outside CTMT Iso	C	10
P-44	HB HV-7150	RCDT Vent Outside CTMT	C	10
P-26	HB HV-7176	RCDT Pumps Disch Hdr Inside CTMT Iso	C	10
P-30	KA FV-29	Reactor Bldg Instr Air Supply Outside CTMT Iso	C	5

~~*May be opened on an intermittent basis under administrative control.~~

*These valves were assumed to be closed during the accident analysis, and are normally closed but may be opened on an intermittant basis under administrative control.

TABLE 3.6-1 (Continued)
CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (SECONDS)
P-32	LF FV-95	CTMT Normal Sumps to Floor Drain Tank Inside CTMT Iso	C	30
P-32	LF FV-96	CTMT Normal Sumps to Floor Drain Tank Outside CTMT Iso	C	4
P-93	SJ HV-5	PZR/RCS Liquid Sample Inner CTMT Iso	C	5
P-93	SJ HV-6	PZR/RCS Liquid Sample Outer CTMT Iso	C	5
P-69	SJ HV-12	PZR Vapor Sample Inner CTMT Iso	C	5
P-69	SJ HV-13	PZR Vapor Sample Outer CTMT Iso	C	5
P-95	SJ HV-18	Accumulator Sample Inner CTMT Iso	C	5
P-95	SJ HV-19	Accumulator Sample Outer CTMT Iso	C	5
P-64	SJ HV-128	PZR/RCS Liquid Sample Inner CTMT Iso	A,C	5
P-64	SJ HV-129	PZR/RCS Liquid Sample Outer CTMT Iso	A,C	5
P-64	SJ HV-130	PZR/RCS Liquid Sample Outer CTMT Iso Valve	A,C	5
P-57	SJ HV-131	PASS Discharge to RCDT	A,C	5
P-57	SJ HV-132	PASS Discharge to RCDT	A,C	5

~~*May be opened on an intermittent basic under administrative control.~~

*These valves were assumed to be closed during the accident analysis, and are normally closed but may be opened on an intermittant basis under administrative control.

TABLE 3.6-1 (Continued)
CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (SECONDS)
2. Phase "A" Isolation (passive)*				
P-58	EM HV-8888	Accumulator Tank Fill Line Iso Valve	C	NA
P-16	EN HV-01	CTMT Recirc Sump to CTMT Spray Pump A Iso	A	NA
P-13	EN HV-07	CTMT Recirc Sump to CTMT Spray Pump B Iso	A	NA
P-45	EP HV-8880	CTMT Nitrogen Supply Iso Valve	C	NA
P-65	GS HV-20	Hydrogen Purge Inner CTMT Iso	C	NA
P-65	GS HV-21	Hydrogen Purge Outer CTMT Iso	C	NA
P-67	KC HV-253	Fire Protection System Hdr Outer CTMT Iso	C	NA
3. Phase "B" Isolation (active)				
P-74	EG HV-58	CCW to RCS Iso	C	30
P-75	EG HV-59	CCW Return From RCS Iso	C	30
P-75	EG HV-60	CCW Return From RCS Iso	C	30
P-76	EG HV-61	CCW Return From RCS Iso	C	30
P-76	EG HV-62	CCW Return From RCS Iso	C	30
4. Containment Purge Isolation (active)				
V-161	GT HZ-4	CTMT Mini-Purge Supply Outside CTMT Iso	C	3

~~*May be opened on an intermittent basis under administrative control.~~

*These valves were assumed to be closed during the accident analysis, and are normally closed but may be opened on an intermittant basis under administrative control.

TABLE 3.6-1 (Continued)
CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (SECONDS)
V-161	GT HZ-5	CTMT Mini-Purge Supply Inside CTMT Iso	C	3
V-160	GT HZ-11	CTMT Mini-Purge Exh Inside CTMT Iso	C	3
V-160	GT HZ-12	CTMT Mini-Purge Exh Outside CTMT Iso	C	3
5. Containment Purge Isolation (passive)				
V-161	GT HZ-6	CTMT S/D Purge Supply Outside CTMT Iso	C	NA
V-161	GT HZ-7	CTMT S/D Purge Supply Inside CTMT Iso	C	NA
V-160	GT HZ-8	CTMT S/D Purge Exh Inside CTMT Iso	C	NA
V-160	GT HZ-9	CTMT S/D Purge Exh Outside CTMT Iso	C	NA
6. Remote Manual				
P-41	BB HV-8351A	RCP A Seal Water Supply	C	NA
P-22	BB HV-8351B	RCP B Seal Water Supply	C	NA
P-39	BB HV-8351C	RCP C Seal Water Supply	C	NA
P-40	BB HV-8351D	RCP D Seal Water Supply	C	NA
P-79	BB PV-8702A	RCS Hot Leg 1 to RHR Pump A Suction	A	NA
P-52	BB PV-8702B	RCS Hot Leg 4 to RHR Pump B Suction	A	NA

See insert (next pg) for additional values in this section

~~*May be opened on an intermittent basic under administrative control.~~

*These valves were assumed to be closed during the accident analysis, and are normally closed but may be opened on an intermittant basis under administrative control.

Insert for Table 3.6-1 section 6

6. Remote Manual (cont.)

P-15	EJ HV-23	PASS Sump Sample Ctmt Iso	C	NA
P-15	EJ HV-25	PASS Sump Sample Ctmt Iso	C	NA
P-14	EJ HV-24	PASS Sump Sample Ctmt Iso	C	NA
P-14	EJ HV-26	PASS Sump Sample Ctmt Iso	C	NA

TABLE 3.6-1 (Continued)
CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (SECONDS)
P-71	EF HV-31	ESW Supply To Containment Coolers	C	NA
P-28	EF HV-32	ESW Supply to Containment Coolers	C	NA
P-71	EF HV-33	ESW Supply to Containment Coolers	C	NA
P-28	EF HV-34	ESW Supply to Containment Coolers	C	NA
P-73	EF HV-45	ESW Return From Containment Coolers	C	NA
P-29	EF HV-46	ESW Return From Containment Coolers	C	NA
P-73	EF HV-47	ESW Return From Containment Coolers	C	NA
P-29	EF HV-48	ESW Return From Containment Coolers	C	NA
P-73	EF HV-49	ESW Return From Containment Coolers	C	NA
P-29	EF HV-50	ESW Return From Containment Coolers	C	NA
P-74	EG HV-127*	CCW Supply to RCP	C	NA
P-75	EG HV-130*	CCW Return From RCP	C	NA
P-75	EG HV-131*	CCW Return From RCP	C	NA
P-76	EG HV-132*	CCW From RCP Thermal Barriers	C	NA

~~*May be opened on an intermittent basis under administrative control.~~

*These valves were assumed to be closed during the accident analysis, and are normally closed but may be opened on an intermittant basis under administrative control.

TABLE 3.6-1 (Continued)
CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (SECONDS)
P-76	EG HV-133*	CCW From RCP Thermal Barrier	C	NA
P-79	EJ HV-8701A	RCS Hot Leg 1 to RHR Pump A Suction	A	NA
P-52	EJ HV-8701B	RCS Hot Leg 4 to RHR Pump B Suction	A	NA
P-82	EJ HV-8809A	RHR Pump A Cold Leg Injection Iso Valve	A	NA
P-27	EJ HV-8809B	RHR Pump B Cold Leg Injection Iso Valve	A	NA
P-15	EJ HV-8811A	CTMT Recirc Sump to RHR Pump A Suction	A	NA
P-14	EJ HV-8811B	CTMT Recirc Sump to RHR Pump B Suction	A	NA
P-21	EJ HV-8840	RHR Hot Leg Recirc Iso Valve	A	NA
P-87	EM HV-8802A*	SI Pump A Disch Hot Leg Iso Valve	A	NA
P-48	EM HV-8802B*	SI Pump B Disch to Hot Leg Iso Valve	A	NA
P-49	EM HV-8835	SI Pumps Disch to Cold Leg Iso Valve	A	NA
P-89	EN HV-6	CTMT Spray Pump A Disch Iso Valve	A	NA
P-66	EN HV-12	CTMT Spray Pump B Discharge Iso Valve	A	NA

~~*May be opened on an intermittent basis under administrative control.~~

*These valves were assumed to be closed during the accident analysis, and are normally closed but may be opened on an intermittant basis under administrative control.

TABLE 3.6-1 (Continued)
CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (SECONDS)
7. Active for SIS				
P-80	BG HV-8105	CVCS Charging Line	C	NA
P-88	EM HV-8801A	Boron Injection to RCS Cold Legs	A	NA
P-88	EM HV-8801B	Boron Injection to RCS Cold Legs	A	NA
8. Hand-Operated and Check Valves				
P-41	BB V-118	RCP A Seal Water Supply	C	NA
P-22	BB V-148	RCP B Seal Water Supply	C	NA
P-39	BB V-178	RCP C Seal Water Supply	C	NA
P-40	BB V-208	RCP D Seal Water Supply	C	NA
P-24	BG V-135	RCP Seal Water Return	C	NA
P-80	BG 8381	CVCS Charging Line	A	NA
P-25	BL 8046	Reactor Makeup Water Supply	C	NA
P-78	BM V-045	Steam Generator Drain Line Iso Valve	C	NA
P-78	BM V-046	Steam Generator Drain Line Iso Valve	C	NA
P-53	EC V-083	Refueling Pool Supply From Fuel Pool Cleanup	C	NA
P-53	EC V-084	Refueling Pool Supply From Fuel Pool Cleanup	C	NA

~~*May be opened on an intermittent basis under administrative control.~~

*These valves were assumed to be closed during the accident analysis, and are normally closed but may be opened on an intermittant basis under administrative control.

TABLE 3.6-1 (Continued)
CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (SECONDS)
P-54	EC V-087	Refueling Pool Return to Fuel Pool Cooling	C	NA
P-54	EC V-088	Refueling Pool Return to Fuel Pool Cooling	C	NA
P-55	EC V-095	Refueling Pool Skimmers To Fuel Pool Cooling Loop	C	NA
P-55	EC V-096	Refueling Pool Skimmers to Fuel Pool Cooling Loop	C	NA
P-74	EG V-204	CCW Supply to RCP	C	NA
P-82	EJ 8818A	RHR Pump to Cold Leg 1 Injection	A	NA
P-82	EJ 8818B	RHR Pump to Cold Leg 2 Injection	A	NA
P-27	EJ 8818C	RHR Pump to Cold Leg 3 Injection	A	NA
P-27	EJ 8818D	RHR Pump to Cold Leg4 Injection	A	NA
P-21	EJ 8841A	RHR Pump Disch to RCS Hot Leg 2	A	NA
P-21	EJ 8841B	RHR Pump Disch to RCS Hot Leg 3	A	NA
P-87	EM V-001	SI Pump Hot Leg 1 Injection	A	NA
P-87	EM V-002	SI Pump Hot Leg 2 Injection	A	NA
P-48	EM V-003	SI Pump Hot Leg 3 Injection	A	NA

~~*May be opened on an intermittent basis under administrative control.~~

*These valves were assumed to be closed during the accident analysis, and are normally closed but may be opened on an intermittant basis under administrative control.

TABLE 3.6-1 (Continued)
CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (SECONDS)
P-48	EM V-004	SI Pump Hot Leg 4 Injection	A	NA
P-58	EM V-006	Accumulator Fill Line From SI Pumps	C	NA
P-49	EM V-010	SI Pump Disch to Cold Leg 1	A	NA
P-49	EM V-020	SI Pump Disch to Cold Leg 2	A	NA
P-49	EM V-030	SI Pump Disch to Cold Leg 3	A	NA
P-49	EM V-040	SI Pump Disch to Cold Leg 4	A	NA
P-88	EM 8815	BIT to RCS Cold Leg Injection	A	NA
P-89	EN V-013	CTMT Spray Pump A to CTMT Spray Nozzles	A	NA
P-66	EN V-017	CTMT Spray Pump B to CTMT Spray Nozzles	A	NA
P-45	EP V-046	Accumulator Nitrogen Supply Line	C	NA
P-43	HD V-016	Auxiliary Steam to Decon System	C	NA
P-43	HD V-017	Auxiliary Steam to Decon System	C	NA
P-63	KA V-039	Rx Bldg Service Air Supply	C	NA

~~*May be opened on an intermittent basis under administrative control.~~

*These valves were assumed to be closed during the accident analysis, and are normally closed but may be opened on an intermittant basis under administrative control.

TABLE 3.6-1 (Continued)
CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (SECONDS)
P-63	KA V-118	Rx Bldg Service Air Supply	C	NA
P-30	KA V-204	Rx Bldg Instrument Air Supply	C	NA
P-67	KC V-478	Fire Protection Supply to Rx Bldg	C	NA
P-57	SJ V-111	Liquid Sample from PASS to RCDT	A,C	NA

~~*May be opened on an intermittent basic under administrative control.~~

*These valves were assumed to be closed during the accident analysis, and are normally closed but may be opened on an intermittant basis under administrative control.

CONTAINMENT SYSTEMS

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen ~~monitors~~ analyzers shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one containment hydrogen ~~monitor~~ analyzer inoperable, restore the inoperable containment hydrogen ~~monitor~~ analyzer to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

~~4.6.4.1 Each containment hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, an ANALOG CHANNEL OPERATIONAL 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 gas containing:~~

~~a. One volume percent hydrogen, balance nitrogen, and~~

~~b. Four volume percent hydrogen, balance nitrogen.~~

4.6.4.1 Each hydrogen analyzer shall be demonstrated OPERABLE by the performance of:

a. An ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and

b. A CHANNEL CALIBRATION using a span gas containing ten percent hydrogen, balance nitrogen, at least once per 92 days on a STAGGERED TEST BASIS.

Specification 3.6.4.1

Justification:

SNUPPS has 2 hydrogen analyzers which can be brought on line for hydrogen analysis in an accident situation. It is SNUPPS intent to have these analyzers in standby status. In this manner SNUPPS can ensure the ability to obtain a containment hydrogen sample within 30 minutes following an accident.

The Channel Check was deleted because in order to conduct the channel check it is necessary to compare the indications of both hydrogen analyzer channels as a qualitative assessment of channel performance. This is accomplished by sampling, "analyze", containment atmosphere with both analyzers since the containment atmosphere is the only common point between the two systems. Since the expected hydrogen concentration in the containment atmosphere is zero, no meaningful comparison can be made.

Ten percent hydrogen gas is used to allow calibration throughout the full range of the meter scale.

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W

CONTAINMENT SYSTEMS

HYDROGEN CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.4.2 A Containment Hydrogen Control System shall be OPERABLE with two independent Hydrogen Recombiner Systems or a Containment Hydrogen Purge Subsystem and one of the two independent Hydrogen Recombiner Systems.

APPLICABILITY: MODES 1 and 2

ACTION:

With one of the two independent Hydrogen Recombiner Systems and the Containment Hydrogen Purge Subsystem inoperable, restore the inoperable Hydrogen Recombiner System or the Containment Hydrogen Purge Subsystem to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2.1 Each Containment Hydrogen Recombiner System shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying, during a Containment Hydrogen Recombiner System functional test, that the heater air temperature increases to greater than or equal to 1150°F within ~~120 minutes,~~ 5 hours and

b. At least once per 18 months by:

1) Performing a CHANNEL CALIBRATION of all Containment Hydrogen Recombiner System instrumentation and control circuits,

2) Verifying through a visual examination that there is no evidence of abnormal conditions within the Containment Hydrogen Recombiner System enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and

3) Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

4.6.4.2.2 The hydrogen purge subsystem shall be demonstrated OPERABLE by cycling valves GS-HV20, GS-HV21 and KA-HV30 at least once per 31 days.

Technical Specification 4.6.4.2.1

Justification:

The Hydrogen Recombiner system is not required for over 24 hours following a LOCA and the Westinghouse Tech. Manual shows a Heatup Time of approximately 5 hours. Therefore, a functional test to verify operability need not be done until 5 hours of operation have elapsed. The revised Tech. Spec. is attached.

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SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific radioiodine activity of the Secondary Coolant System shall be less than or equal to 0.10 microCurie~~s~~/gram DOSE EQUIVALENT I-131.

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

ACTION:

With the specific radioiodine activity of the Secondary Coolant System greater than 0.10 microCurie~~s~~/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 radioiodine 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 Specific Activity Determination~~	At least once per 72 hours.
1. ~~2.~~ Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	~~a) Once per 31 days, whenever the gross specific activity determination indicates ~~radionuclide~~ concentrations greater than 10% of the allowable limit for radioiodine.~~ ~~b) Once per 6 months, whenever the gross specific activity, determination indicates ~~radionuclide~~ concentrations below 10% of the allowable limit for radioiodine.~~

Specification 3.7.1.4
Surveillance 4.7.1.4
Table 4.7-1

JUSTIFICATION

These changes are requested so that the analysis performed will allow the determination of the parameter governing the limit. (Dose Equivalent Iodine)

PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent component cooling water loops shall be OPERABLE with a minimum of one component cooling water pump per loop OPERABLE.

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

ACTION:

With only one component 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 component 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. In addition, an ANALOG CHANNEL OPERATIONAL TEST of the surge tank level and flow instrumentation which provide automatic isolation of the non-nuclear safety-related portion of the system shall be performed once per 31 days; and

b. At least once per 18 months during shutdown:

1) Verify that each automatic valve servicing safety-related equipment or isolating the non-nuclear safety-related portion of the system actuates to its correct position on a Loss of Power or Safety Injection test signal and on a simulated High Flow and Low Surge Tank Level test signal;

2) Perform a CHANNEL CALIBRATION of the surge tank level and flow instrumentation which provide automatic isolation of the non-nuclear safety-related portion of the system; and

3) Verify that each required Component Cooling Water System pump starts automatically on a ~~Safety Injection~~ test signal. and Loss of Site Power

X

Specification: 3.7.3, 4.7.3

Justification:

SNUPPS has 2 CCW pumps per loop, one of which is an installed spare. One in each loop could be out-of-service and each loop would still be OPERABLE. Hence, 3.7.3 was changed to reflect this. On a LOCA or LOSP, only one CCW pump in each train is automatically sequenced with the other one sequenced if the first pump fails to start. Hence, on a LOCA or LOSP, only A or C and B or D start. SNUPPS, therefore, added "required" to 4.7.3.b.3.

W.C. ONLY

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink (UHS) shall be OPERABLE with:

See insert A

~~a. A minimum water level at or above elevation 1068.7 Mean Sea Level, USGS datum, and~~

~~b. An average water temperature of less than or equal to 93°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 the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIRMENTS

See insert B

~~4.7.5 The UHS shall be determined OPERABLE at least once per 24 hours by verifying the average water temperature and water level to be within their limits.~~

Insert A

a. The crest of the ultimate heat sink dam and corresponding water level at or above elevation 1069.5 Mean Sea Level, USGS datum, and

b. The Lake outlet (plant inlet) water temperature of less than or equal to 90°F.

Insert B

4.7.5 The ultimate heat sink shall be demonstrated OPERABLE:

a. At least once per 12 months by verifying that the top of the lowest of the ultimate heat sink settlement monuments (nine total) is at or above elevation 1069.5 Mean Sea Level, USGS datum, and

b. At least once per 24 hours by verifying the water temperature to be below or equal to 90°F.

Specification 3.7.5

Justification -

The value provided for minimum water level is incorrect for the ultimate heat sink to be considered operable. The water must be at the top of the dam, Elevation 1070 to 1069.5 Mean Sea Level (MSL). If it was at 1068.7 MSL, the ultimate heat sink (UHS) would be at its lower limit of operation. Therefore, the water level must be at or above 1069.5 MSL.

Ninety-five degrees Farenheit is the maximum permissible temperature of water supplied to the plant. The correct temperature, 90°F, is the temperature at which the Architect/Engineer commenced the evaluation of the UHS with 2-unit operation and 2-unit LOCA. Ninety degrees, as an initial plant inlet temperature, will yield a maximum plant inlet temperature of 94.8°F during the accident. This is less than the design base plant inlet temperature (95°F).

Specification 4.7.5

Justification -

This specification has been revised to address the UHS dam monitoring program discussed in Section 2.5.6 of NUREG-0881, Supplement No. 1.

The existing standard Technical Specification requirement is not applicable for a plant with a submerged UHS. The lake is the limiting factor for plant operation (see FSAR Site Addendum, Pg. 2.4-47). The volume of water behind the UHS dam is the limiting condition for UHS operation. A volume reduction calculation indicates that the crest of the UHS dam can be lowered by 6 inches, without structural damage to the embankment, to Elevation 1069.5 MSL and still provide adequate, less than 95°F, plant inlet water temperature.

CAL ONLY

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink (UHS) shall be OPERABLE with:

a. A minimum water level at or above 10.5 feet from the bottom of the UHS,

b. An average water temperature of less than or equal to 95°F, and

c. Two UHS cooling towers.

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

ACTION:

~~With one UHS cooling tower inoperable, restore at least two UHS cooling towers 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.~~

Later

SURVEILLANCE REQUIRMENTS

4.7.5.1 The UHS shall be determined OPERABLE at least once per 24 hours by verifying the average water temperature and water level to be within their limits.

~~4.7.5.2 The UHS cooling towers shall be demonstrated OPERABLE at least once per 31 days by verifying that each cooling tower fan operates for at least 15 minutes in both the slow and fast mode and at least once per 18 months by visually inspecting and verifying no breakage or abnormal degradation of the fill materials.~~

4.7.5.3 The UHS shall be determined OPERABLE at least once per 31 days by visually inspecting the UHS riprap for any abnormal degradation which might lead to blockage of the ESW pump suction.

4.7.5.2 The UHS cooling towers shall be demonstrated OPERABLE;

a. At least once per 31 days by verifying that each cooling tower fan operates for at least 15 minutes in both the slow and fast mode;

b. At least once per 18 months by visually inspecting and evaluating breakage or degradation of the fill material.

Technical Specification: 4.7.5.2 Page 3/4 7-13

Justification:

Recommended change allows reference to separate surveillance requirements for monthly surveillance and refueling interval surveillance. It also assists in establishing the matrix for surveillance schedules and identifies the diverse requirements as unique inspection requirements.

The phrase "no breakage" is an absolute term used in an unrealistic application. Fill breakage of some degree is considered as normal wear. This wear should be evaluated for degree but cannot be eliminated by decree.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 Two independent control room emergency ventilation systems shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION:

MODES 1, 2, 3 and 4:

With one control room emergency ventilation 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 emergency ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE control room emergency ventilation system in the recirculation mode.

b. With both control room emergency ventilation systems inoperable, or with the OPERABLE control room emergency ventilation system, required to be in the recirculation mode 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

4.7.6 Each control room emergency ventilation system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 104°F.

b. 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 of both the filtration and pressurization systems and verifying that the pressurization system operates for at least 10 hours ~~with the heaters operating~~ → Delete

Delete 10 hours and change to 15 minutes

SURVEILLANCE REQUIREMENTS (Continued)

c. 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 Control Room Emergency Ventilation System satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance in 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 2000 cfm ± 10% for the Filtration System and 2000 cfm ± 10% for the Pressurization System with 500 cfm ± 10% going through the pressurization filter adsorber unit ~~.~~ when tested in accordance with ANSI N510-1975

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, for a methyl iodide penetration of less than 1%; and

~~3) Verifying a system flow rate of 2000 cfm ± 10% for the Filtration System and 2000 cfm ± 10% for the Pressurization System with 500 cfm ± 10% going through the pressurization filter adsorber unit during system operation when tested in accordance with ANSI N510-1975.~~

d. 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, for a methyl iodide penetration of less than 1%;

e. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 5.4 inches Water Gauge while operating the system at a flow rate of 2000 cfm ± 10% for the Filtration System and 500 cfm ± 10% for the pressurization filter adsorber unit;

2) Verifying that on a Control Room Ventilation Isolation test signal, the system automatically switches into a recirculation 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 1/4 inch W.G. relative to the outside atmosphere during system operation; and

4) Verifying that the pressurization filter adsorber unit heaters dissipate 15 ± 2 kW in the Pressurization System when tested in accordance with ANSI N510-1975.

Specifications 4.7.6b, 4.7.7a, 4.9.13a   Pages 3/4 7-14, 3/4 7-17
3/4 9-16

Justification:

Running time of ten hours is unnecessary to establish the operability of these systems and needlessly reduces the operating lifetime of the filter train components.

Requirement that heaters be operating during test is unnecessary in that heater operation is tested every 18 months in accordance with ANSI N510-1975 and in that heater operation is automatic and output is regulated by the relative humidity of the incoming air into the filter train.

Tech Spec bases suggest that the filter train fans and heaters could be operated once every 31 days for ten continuous hours to keep the charcoal filters dry. During normal operations, however, the filter trains are isolated from the air flow paths and would be exposed to limited moisture condensation. The vendor, Mine Safety Appliances, agrees that in the described arrangement moisture condensation would be limited.

During testing and emergency operations the heaters are automatically turned on with the start of their emergency exhaust fan and heater power output is regulated by a signal from a relative humidity sensor and transmitter located downstream of the heater. The heaters' output is controlled by the relative humidity of air entering the filter train units in order to keep the relative humidity of the air at a level that will limit moisture condensation.

A review of the following documentation established no set run time required for determining system operability:

ANSI/ASME N510-1975	Testing of Nuclear Air Cleaning Systems
ANSI/ASME N510-1980	Testing of Nuclear Air Cleaning Systems
ANSI/ASME N509-1980	Nuclear Power Plants Air Cleaning Units and Components

NRC Regulatory Guide 1.52 - March 1978

Technical Specification 4.7.6.C.3 Page 3/4 7-15

Justification -

Delete 4.7.6.C.3 Surveillance requirement and revise 4.7.6.C.1 as indicated on attached. System flow rate verification is accomplished by 4.7.6.C.1, therefore 4.7.6.C.3 is redundant and not needed.

3/4.7.7 EMERGENCY EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent Emergency Exhaust Systems shall be OPERABLE.

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

ACTION:

With one Emergency Exhaust System inoperable, restore the inoperable Emergency Exhaust 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

4.7.7 Each Emergency Exhaust 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 ~~10 hours~~ 15 minutes with the heaters operating;

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 Emergency Exhaust System satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance in 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 9000 cfm ± 10% WHEN TESTED IN ACCORDANCE WITH ANSI N510-1975.

Specifications 4.7.6.b, 4.7.7a, 4.9.13a   Pages 3/4 7-14, 3/4 7-17
3/4 9-16

Justification:

Running time of ten hours is unnecessary to establish the operability of these systems and needlessly reduces the operating lifetime of the filter train components.

Requirement that heaters be operating during test is unnecessary in that heater operation is tested every 18 months in accordance with ANSI N510-1975 and in that heater operation is automatic and output is regulated by the relative humidity of the incoming air into the filter train.

Tech Spec bases suggest that the filter train fans and heaters could be operated once every 31 days for ten continuous hours to keep the charcoal filters dry. During normal operations, however, the filter trains are isolated from the air flow paths and would be exposed to limited moisture condensation. The vendor, Mine Safety Appliances, agrees that in the described arrangement moisture condensation would be limited.

During testing and emergency operations the heaters are automatically turned on with the start of their emergency exhaust fan and heater power output is regulated by a signal from a relative humidity sensor and transmitter located downstream of the heater. The heaters' output is controlled by the relative humidity of air entering the filter train units in order to keep the relative humidity of the air at a level that will limit moisture condensation.

A review of the following documentation established no set run time required for determining system operability:

ANSI/ASME N510-1975	Testing of Nuclear Air Cleaning Systems
ANSI/ASME N510-1980	Testing of Nuclear Air Cleaning Systems
ANSI/ASME N509-1980	Nuclear Power Plants Air Cleaning Units and Components

NRC Regulatory Guide 1.52 - March 1978

NRC

SURVEILLANCE REQUIREMENTS (Continued)

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, for a methyl iodide penetration of less than 1%; and

3) ~~Verifying a system flow rate of 9000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.~~

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, for a methyl iodide penetration of less than 1%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks of less than 5.4 inches Water Gauge while operating the system at a flow rate of 9000 cfm ± 10%,

2) Verifying that the system starts on a Safety Injection test signal, and

3) Verifying that the heaters dissipate 37 ± 3 kw when tested in accordance with ANSI N510-1975.

4) Verifying that the heaters will activate when subjected to relative humidities in excess of 70%.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter bank~~s~~ satisfies the in-place penetration and bypass leakage testing criteria of less than 1% ~~remove greater than or equal to 99% of the DOP when they are tested in-place~~ in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 9000 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber~~s~~ satisfies the in-place penetration and bypass leakage testing criteria of less than 1% ~~remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place~~ in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 9000 cfm ± 10%.

Technical Specification 4.7.7.b.3 Page 3/4 7-17

Justification -

Delete surveillance 4.7.7.b.3 as noted on attached. Technical specification 4.7.7.6.1 requires verification of flow rates referenced in 4.7.7.6.3. This surveillance is therefore redundant and should be deleted.

Specification: 4.7.7.d.4

Justification:

The capability of impregnated charcoal to remove elemental iodine and organic iodines and the performance of prefilters and HEPA filters is a function of air stream relative humidity. Heaters are provided to limit the relative humidity of the downstream air to levels which will not adversely affect prefilters, charcoal absorber or HEPA filter efficiency. Thus, surveillance of the heater controls is required to assure filtration unit efficiency.

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All snubbers listed in Tables 3.7-4a and 3.7-4b shall be OPERABLE.

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, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8 Each snubber shall be demonstrated OPERABLE by performance of the following inservice inspection program.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity. Snubbers will be grouped by type, application, and environmental conditions.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation and may be treated independently. The accessibility of each snubber shall be determined based upon the then existing radiation levels in each snubber location and the expected time to perform the visual inspection.

The first inservice visual examination of snubbers shall be performed at the first refueling outage and shall include all snubbers listed in Tables 3.7-4a and 3.7-4b. Subsequent visual examinations shall be performed in accordance with the following schedule.

No. Inoperable Snubbers of Each Group per Inspection Period	Subsequent Visual Inspection Period*#
	18 months ± 25%
	12 months ± 25%
	6 months ± 25%
	124 days ± 25%
	62 days ± 25%
	31 days ± 25%

* The inspection interval for each group of snubbers 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 group are found.

The provisions of Specification 4.0.2 are not applicable.

c. Refueling Outage Inspections

At least once per 18 months an inspection shall be performed of all the snubbers listed in Tables 3.7-4a and 3.7-4b attached to sections of Safety Systems piping that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems. In addition 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; (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

d. Visual Inspection Acceptance Criteria

Visual inspections shall verify: (1) that there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure 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 that may be generically susceptible; or (2) the affected snubber is functionally tested in the as found condition and determined to be OPERABLE per Specifications 4.7.8f. When a fluid port of a hydraulic snubber is found to be uncovered the snubber shall be declared inoperable and shall not 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. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

e. Functional Testing

For the purpose of functional testing of safety-related snubbers, the sample plan and resample plan should provide a confidence level of 95% that 90% to 100% of the snubbers are operable. Several sample plans are known to meet this criteria. These sample plans are listed, as options, to allow latitude for continued investigation leading to the implementation of the most effective sample plan.

During the first refueling shutdown and at least once per refueling thereafter, a representative sample of snubbers shall be tested using one of the following sample plans.

The NRC Regional Administrator shall be notified in writing of the sample plan selected for implementation prior to the initiation of the functional testing program. If notice is not given prior to the initiation of the functional testing program, that sample plan implemented in the most recent refueling outage shall be implemented.

1) A representative sample of 10% of all snubbers listed in tables 3.7-4a and 3.7-4b shall be functionally tested either in-place or in a bench test.

For the first sample tested, a sample which is representative of the snubber designs and installations shall be selected. For each snubber failing the functional test acceptance criteria, an additional sample lot of 1/2 the size of the initial lot, within the group represented by the failed snubber, shall be tested. Testing shall continue within the representative group until no failures are found in subsequent sample lots or all units in the representative groups have been tested.

At subsequent testing intervals, each representative sample shall consist of previously untested snubbers.

2) A representative sample of at least 37 snubbers listed in Tables 3.7-4a and 3.7-4b shall be functionally tested. Additional testing shall be in accordance with Figure 4.7-1 which includes acceptance and rejection criteria. "C" is the cumulative total number of snubbers found not meeting the functional test acceptance criteria. The cumulative number of snubbers tested is denoted by "N". At the end of each testing day, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on the Figure 4.7-1. If at any time the point plotted falls in the "Reject" region, all snubbers of that group shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of that group of snubbers may be terminated. When the point plotted falls in the "Continue Testing" region, additional snubbers shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that group have been tested.

SURVEILLANCE REQUIREMENTS (Continued)

e. Functional Tests (Continued)

3) A representative sample of 55 snubbers listed in Tables 3.7-4a and 3.7-4b shall be functionally tested. For each snubber failing the functional test acceptance criteria another sample of at least 1/2 the initial lot 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 failing the functional test acceptance criteria. Another sample of at least 1/2 of the initial test lot shall be tested for each subsequent snubber determined to fail the functional test acceptance criteria.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the resampling.

For the purpose of functional testing, the hydraulic and mechanical snubbers will be considered as independent groups with sampling and testing criteria applied accordingly.

If it is determined by an engineering analysis that a series of unit failures is unique to a certain group or model, then the sample quantities need not exceed the total number of snubbers in that group.

Snubbers identified in Tables 3.7-4a and 3.7-4b as "Especially Difficult To Remove" or in "High Radiation Zones During Shutdown" shall be included in the representative sample.

Inservice operability testing may be accomplished with the snubber installed in its permanent location by utilizing owner-approved test methods and equipment.

If it is extremely difficult to utilize the conventional test methods, due to the physical size of the snubber or inaccessibility of location, the snubber subcomponents shall be examined and tested in accordance with approved procedures. Reassembly of individual components must be in accordance with approved procedures.

The snubbers of parallel and multiple installation locations shall be identified and counted individually.

Snubbers shall not receive prior maintenance specifically for the purpose of meeting an operability test requirement.

SURVEILLANCE REQUIREMENTS (Continued)

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 rate for hydraulic snubbers 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 not great enough to overstress the attached piping or component during thermal movement, or to indicate impending failure of the snubber.

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. Service Life Monitoring Program

The service life of a snubber is evaluated 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 in Station Procedures 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.

The seal service life of hydraulic snubbers shall be monitored to ensure that the seals do not fail between surveillance examinations. The maximum expected service life for the various seals, seal materials, and applications shall be estimated based on engineering information, and the seals shall be replaced so that the maximum expected service life does not expire during a period when the snubber is required to be OPERABLE. The seal replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

Mechanical snubber drag force increases greater than 50% of previously measured values shall be evaluated as an indication of impending failure of the snubber. These evaluations, and any associated corrective action such as repair or replacement of the snubbers, shall be documented, and the documentation shall be retained in accordance with Station Procedures.

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. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

TABLE 3.7-4a

SAFETY-RELATED HYDRAULIC SNUBBERS*

(MANUFACTURER)

SYSTEM	SIZE (KIPS)		
	Small	Medium	Large
	( ) ( )	( ) ( )	( ) ( )
Subtotal-1			
Subtotal-2			
TOTAL			

*Snubbers may be added to safety-related systems without prior License Amendment to Table 3.7-4a provided that a revision to Table 3.7-4a is included with the next License Amendment request. Snubbers may be removed from safety-related systems without prior License Amendment if their deletion can be justified by engineering evaluation or system design change, provided that the justification for their removal and revision to Table 3.7-4a is included with the next License Amendment request.

TABLE 3.7-4b

SAFETY-RELATED MECHANICAL SNUBBERS*

(MANUFACTURER)

SYSTEM	SIZE (KIPS) Small ( ) ( )	Medium ( ) ( )	Large ( ) ( )
Subtotal-1			
Subtotal-2			
TOTAL			

*Snubbers may be added to safety-related systems without prior License Amendment to Table 3.7-4b provided that a revision to Table 3.7-4b is included with the next License Amendment request. Snubbers may be removed from safety-related systems without prior License Amendment if their deletion can be justified by engineering evaluation or system design change, provided that the justification for their removal and a revision to Table 3.7-4b is included with the next License Amendment request.

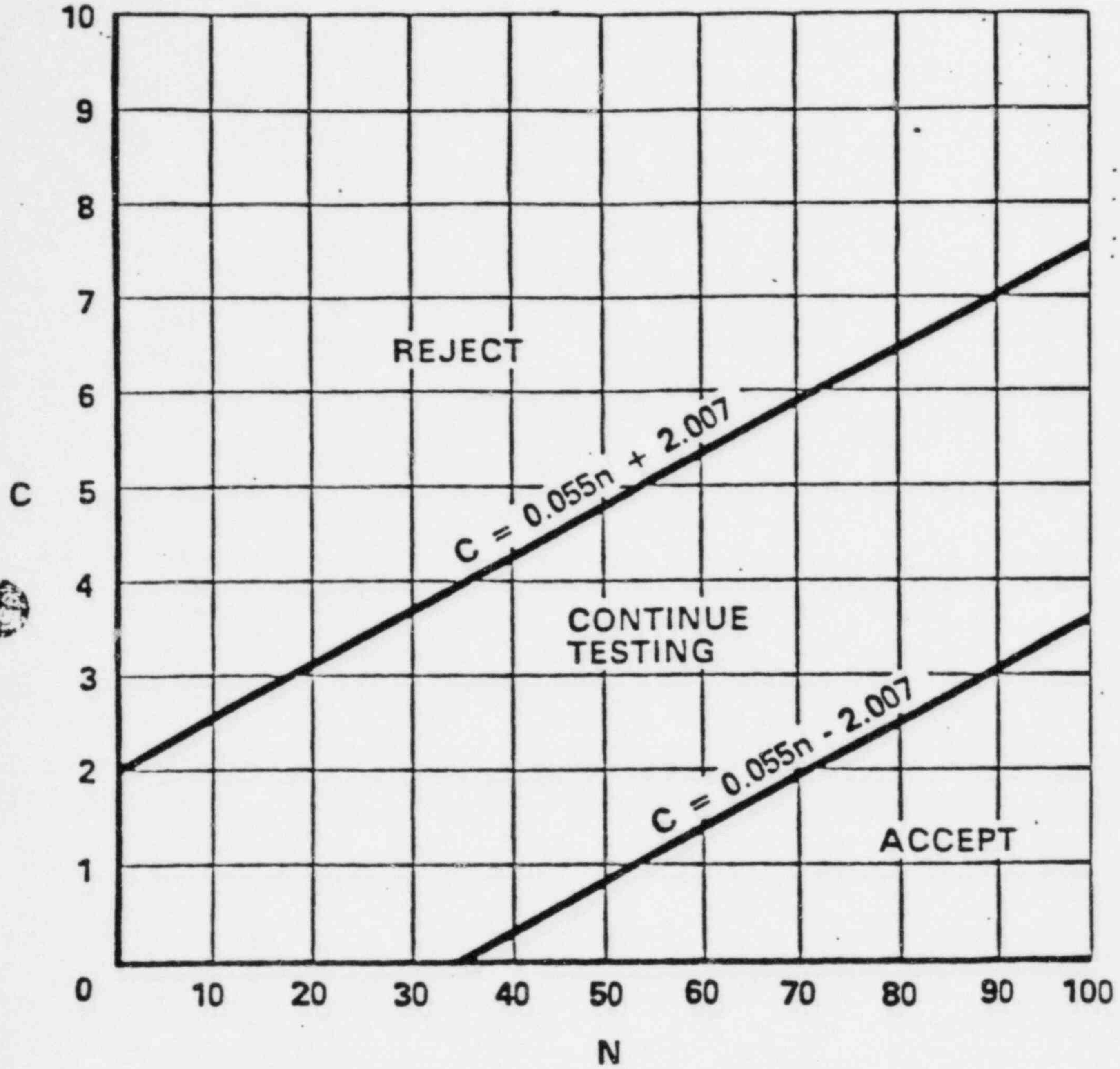

FIGURE 4.7-1
SAMPLING PLAN 2) FOR SNUBBER FUNCTIONAL TEST

3/4.7.8 SNUBBERS

SECTION	JUSTIFICATION
4.7.8	The reference to Spec. 4.0.5 is deleted because of the contradictions between this Tech. Spec. and Section XI. IWF. We anticipate asking for relief from IWF. Section XI based upon our approved Technical Specification.
4.7.8.a	Our initial grouping will be only by type but as operational experience is gained, we plan to further refine the groups. The concept of grouping snubbers with similar environmental conditions and application should give a higher confidence factor that the installed snubbers are operable. When failures occur, if the resample consist of snubbers which are subject to the same conditions and thus the same failure modes, the chances of finding a generic problem or proving the failure was an isolated case are increased. The grouping concept is also consistent with O&M 4.
4.7.8.b	The Station Health Physicists will be involved in determining the radiation levels during power operation which will lead to the "accessible and inaccessible list." But we feel that the Technical Specifications is not the appropriate place to dictate which persons shall accomplish the given functions. The SNUPPS utilities have already committed to ALARA and the guidelines of Reg. Guide 8.8 and 8.10 in Section 12 of the SNUPPS FSAR and in the site addenda. Again, we feel that the Technical Specification is an inappropriate document to make a commitment of this nature.

SECTION	JUSTIFICATION
4.7.8.b Paragraph 2	The SNUPPS utilities believe this paragraph is overly restrictive because of our pre-operational commitments and plans to assure snubber operability.

The utilities' plans include:

1. Having the snubbers presently removed from the plant for protection from construction damage.

2. Performing a functional (Activation and Drag) test on 100% of the mechanical snubbers prior to installation.

3. Installing snubbers after the majority of the construction activities are complete in an area except where snubbers are necessary for initial system testing. Operability will be assured just prior to Hot Functional Testing (HFT).

4. The utilities have committed to a rigorous preservice examination and pre-operational testing program for snubbers as specified in their Safety Evaluation Reports (NuReg 0830 and NuReg 0881).

   Preservice examinations, which are made after snubber installation but not more than six months prior to HFT, verify the following:

   1. There are no visible signs of damage or impaired operability as a result of storage, handling or installation.

   2. The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifications.

   3. Snubbers are not seized, frozen or jammed.

   4. Adequate swing clearance is provided to allow snubber movement.

SECTION	JUSTIFICATION
4.7.8.b Paragraph 2	4. 5. If applicable, fluid is to the recommended level and is not leaking from the snubber system. 6. Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire, and cotter pins are installed correctly. During pre-operational testing (HFT), snubber thermal movements for systems whose operating temperatures exceed 250°F will be verified as follows: a. During initial system heatup and cooldown at specified temperature intervals for any system which attains operating temperature, verify the snubber expected thermal movement. b. For those systems which do not attain operating temperature, verify via observation and/or calculation that the snubber will accommodate the projected thermal movement. c. Verify the snubber using clearance at specified heatup and cooldown intervals. Any discrepancies or inconsistencies shall be evaluated for cause and corrected prior to proceeding to the next specified interval. The above program assures the SNUPPS utilities a high confidence level in the operability of their snubbers. The program is far more comprehensive than what other plants have done in the past. The utilities believe that because of this high confidence level, the requirement for performing the first visual inspection within 4 to 10 months of commencing Power Operation is unnecessary and contradictory to the ALARA concept.
Table Heading and Footnote	See Section 4.7.8.a
4.7.8.b Examination Period Table	The table in its present form does not take into consideration the difference in the number of snubbers installed in various plants. For plants such as SNUPPS, this table is overly

SECTION	JUSTIFICATION
	restrictive due to the large number of snubbers installed. This table should be changed to reflect a % rate of failure. Hence the SNUPPS utilities propose a "later" entry into this table until the NRC can complete its study in progress.
4.7.8d	The SNUPPS utilities believe that if the cause of rejection is clearly established and remedied it should not be used as a consideration for establishing the subsequent visual examination interval. An example of this would be an "isolated failure" as defined in O&M 4: "Isolated Failures - failures resulting from damage caused during installation or maintenance (i.e., standing on the units, dropping equipment or tools on the unit, etc.), where the nature of the failure does not lend other units to be suspect." The snubbers in the example above could easily be damaged to the extent that they could not pass a functional test. But it would not be reasonable to count those snubbers as failed units for the purpose of establishing the subsequent examination. The action required by O&M 4 for isolated failures is as follows, "All unacceptable units in this group shall be repaired or replaced and recategorized as acceptable for the purpose of establishing the time to subsequent examinations." Also (1) and (2) are contradictory in that if a snubber is found to "appear inoperable" as the result of a visual examination but is functionally tested and found to be "operable" than there is no reason for the cause of the rejection to be remedied since the snubber has not been rejected.
4.7.8.f.4	This statement is not a verification but more of an explanation as to the type of testing that is acceptable.
Table 3.7-4a and 3.7-4b Footnote	It is proposed that utilities have the ability to delete snubber(s) from Tables 3/7-4a and 3.7-4b if, based upon an engineering evaluation, a design change is made which justifies the removal of the snubber(s).

SECTION	JUSTIFICATION
Table 3.7-4a and 3.7-4b Footnote (Cont.)	All snubbers listed in tables 3.7-4a and 3.7-4b require visual inspections during a refueling outage. If a design change has removed a particular snubber(s), compliance to the visual inspection requirements cannot be accomplished until the respective snubber(s) have been removed from the tables. The request to remove the affected snubber would be included in the next appropriate Technical Specification change.

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9 Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or ~~5~~ 10 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 limits, 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

4.7.9.1 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.

4.7.9.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

   1) With a half-life greater than 30 days (excluding Hydrogen 3), and

   2) In any form other than gas.

BASES

SNUBBERS (Continued)

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. The requirements for the maintenance of records and the snubber service life review is not intended to affect plant operation.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitation 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 shielded mechanism.

3/4.7.10 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The fire suppression system consists of the water system, spray, and/or sprinklers, Halon, and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire-fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire-fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

Technical Specification 3/4.7.9

Justification:

A value of 10 microcuries is justified because of compatibility and consistency with the existing sealed source leak test requirements as specified by the Callaway Plant by-product license #24-02020-06 and the Wolf Creek Radioactive Materials License #26-B401-01 (both attached).

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Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974 (Public Law 438), and Title 10, Code of Federal Regulations, Chapter 1, Parts 30, 31, 32, 33, 34, 35, 36, 40 and 70, and in reliance statements and representations heretofore made by the licensee, a license is hereby issued authorizing the licensee receive, acquire, possess, and transfer byproduct, source, and special nuclear material designated below; to use su material for the purpose(s) and at the place(s) designated below; to deliver or transfer such material to persons authori: to receive it in accordance with the regulations of the applicable Part(s); and to import such byproduct and sou material. This license shall be deemed to contain the conditions specified in Section 183 of the Atomic Energy Act of 19 as amended, and is subject to all applicable rules, regulations and orders of the Nuclear Regulatory Commission now hereafter in effect and to any conditions specified below.

Licensee	
1. Union Electric Company	3. License number 24-02020-06
2. P. O. Box 149 St. Louis, Missouri 63166	4. Expiration date January 31, 1984
	5. Docket or Reference No.

6. Byproduct, source, and/or special nuclear material	7. Chemical and/or physical form	8. Maximum amount that licen may possess at any one tim under this license
A. Strontium 90	A. Sealed sources	A. No single source to exceed 0.5 microcuri
B. Strontium 90	B. Sealed sources	B. No single source to exceed 1 microcurie
C. Cesium 137	C. Sealed sources (New England Nuclear Model NER-401 H)	C. No single source to exceed 30 millicurie

9. Authorized use

A., B. and C. Receipt and storage.

CONDITIONS

10. Licensed material shall be used only at Union Electric Company, Daniel Internati Corporation, Callaway Plant, Highway CC - Three miles North of Highway 94, Portl Missouri.

11. The licensee shall comply with the provisions of Title 10, Chapter 1, Code of Federal Regulations, Part 19, "Notices, Instructions and Reports to Workers; Inspections" and Part 20, "Standards for Protection Against Radiation."

12. Licensed material shall be used by, or under the supervision of, Neel G. Slaten.

ATTACHMENT 1 of 2

(continued)

13. Sealed sources containing licensed material shall not be opened.

14. A. (1) Each sealed source containing licensed material, other than Hydrogen 3 with a half-life greater than thirty days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months. In the absence of a certificate from a transfer(indicating that a test has been made within six months prior to the transfer, a sealed source received from another person shall not be put into use until tested.

(2) Notwithstanding the periodic leak test required by this condition, any licensed sealed source is exempt from such leak tests when the source contains 100 microcuries or less of beta and/or gamma emitting materia or 10 microcuries or less of alpha emitting material.

(3) The periodic leak test required by this condition does not apply to sealed sources that are stored and not being used. The sources except from this test shall be tested for leakage prior to any use or transfe to another person unless they have been leak tested within six months prior to the date of use or transfer.

B. The test shall be capable of detecting the presence of 0.005 microcurie of radioactive material on the test sample. The test sample shall be taken from the sealed source or from the surfaces of the device in which the sealed source is permanently mounted or stored on which one might expect contamination to accumulate. Records of leak test results shall be kept in units of microcuries and maintained for inspection by the Commission.

C. If the test reveals the presence of 0.005 microcurie or more of removable contamination, the licensee shall immediately withdraw the sealed source from use and shall cause it to be decontaminated and repaired or to be disposed of in accordance with Commission regulations. A report shall be filed within 5 days of the test with the U. S. Nuclear Regulatory Commissio Region III, Office of Inspection and Enforcement, 799 Roosevelt Road, Glen Ellyn, Illinois 60137, describing the equipment involved, the test results and the corrective action taken.

D. Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an Agreeme State to perform such services.

STATE OF KANSAS
RADIOACTIVE MATERIALS LICENSE

Pursuant to the Nuclear Development and Radiation Control Act (L. 1963, ch. 290) and the Radiation Protection Regulations, Part 3, and in reliance on statements and representations heretofore made by the licensee designated below, a license is hereby issued authorizing such licensee to transfer, receive, possess and use the radioactive material(s) designated below; and to use such radioactive materials for the purpose(s) and at the place(s) designated below. This license is subject to all applicable rules, regulations, and orders now or hereafter in effect of the State Department of Health and to any conditions specified below.

Amendment No. 4

Licensee	
1. Name Kansas Gas and Electric Company Nuclear Department 2. Address P.O. Box 208 Wichita, Kansas 67201	3. License number 26-B401-01 amended in its entirety 4. Expiration date June 30, 1983 (F 83) 5. Reference number

6. Radioactive materials (element and mass number)	7. Chemical and/or physical form	8. Maximum quantity licensee may possess at any one time
See Page # 2	See Page # 2	See Page # 2

CONDITIONS

9. Authorized use. (Unless otherwise specified, the authorized place of use is the licensee's address stated in Item 2 above.)

A. and B. To be used for fixed and portable instrument calibration and radioanalytical techniques.

C. and D. To be used for instrument calibrations.

E. and F. To be contained in a J.L. Shepard Model Number 89 Calibrator and to be used for instrument calibration.

G. To be used for instrument calibrations.

H. To be used as a Westinghouse Solid S.S. Capsule Boronometer Source.

I. To be used in the Science Applications, Inc. designed Boron Monitors.

J. and K. To be used in Westinghouse capsules sealed in steel blocks.

L. To be encapsulated in 304L stainless steel welded capsules and used as primary sources contained in Primary Source Rod Assembly as per Westinghouse Drawing Number 8765D59.

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STATE OF KANSAS
RADIOACTIVE MATERIALS LICENSE

Supplementary Sheet

License Number 26-B401-01

Amendment No. 4

6. Radioactive materials (element and mass number)	7. Chemical and/or physical form	8. Maximum quantity licensee may possess at any one time
A. Any radioactive material between Atomic No. 3 and 83.	A. Any.	A. No single source to exceed 1.0 millicurie, the total not to exceed 50 millicuries.
B. Any radioactive material between Atomic No. 84 and 98 (except special nuclear material).	B. Any.	B. No single source to exceed 1.0 millicurie, the total not to exceed 10 millicuries.
C. Cesium 137.	C. Sealed source (New England Nuclear Model NER-401H).	C. 30 millicuries.
D. Cesium 137.	D. Sealed source (J.L. Shepard and Associates Model Number 7810).	D. 120 millicuries.
E. Cesium 137.	E. Sealed source (J.L. Shepard Model Number 78-2M).	E. 400 curies.
F. Cesium 137.	F. Sealed source (J.L. Shepard Model Number 78-2M).	F. 130 millicuries.
G. Cesium 137.	G. Sealed source (General Atomic Company Model RT-11).	G. 100 millicuries.
H. Americium 241/ Beryllium.	H. Sealed source (Westinghouse Solid S.S. Capsule Boronometer source).	H. 1.0 curie.
I. Americium 241/ Beryllium.	I. Sealed sources (Amersham Model X.2 Code AMN·18).	I. Four (4) sources no single source to exceed 300 millicuries each.

STATE OF KANSAS

RADIOACTIVE MATERIALS LICENSE

Supplementary Sheet

License Number 26-B401-01

Amendment No. 4

6. Radioactive materials (element and mass number)	7. Chemical and/or physical form	8. Maximum quantity licensee may possess at any one time
J. Uranium 238.	J. Powder as U_3O_8 in capsule.	J. Six (6) capsules, 12 mg per capsule.
K. Neptunium 237.	K. Powder as NpO_2 in capsule.	K. Six (6) capsules, 20 mg per capsules; 12.1 microcuries each.
L. Californium 252.	L. Sources rods (Monsanto Research Corporation Model Number 2765-AA00).	L. Two (2) sources not to exceed 182 millicuries each.
M. Antimony/ Beryllium.	M. Sources rods (Westinghouse Drawing Number 271C813G02).	M. Eight (8) sources not to exceed 550 grams each.
N. Uranium 235.	N. Fission chambers.	N. Fifteen (15) sources, not to exceed 5.2 mg nominal Uranium Oxide (U_3O_8) and 4.1 mg nominal U-235 each.
O. Thorium 228.	O. Sealed sources.	O. Two (2) sources, not to exceed 5.0 micro-curies each.
P. Uranium 235.	P. Lined fission counter.	P. 1.6 grams of U-235.

M. To be contained in 304 stainless steel tubing as per Westinghouse Source Assembly Drawing Number 1449E70G02 and used as secondary sources.

N. To be used as Westinghouse Fission Chambers.

O. To be used as calibration/check sources.

P. To be used in a Reuter-Stokes Fission Counter.

STATE OF KANSAS
RADIOACTIVE MATERIALS LICENSE

Supplementary Sheet

License Number 26-B401-01

Amendment No. 4

10. Radioactive material shall be used at Wolf Creek Generating Station, Burlington, Kansas 66839. Radioactive material described in Items 6, 7, and 8, Subitems A and B may also be used at Kansas State University, Manhattan, Kansas and/or at the University of Kansas, Lawrence, Kansas. Radioactive material described in Items 6, 7, and 8, Subitem D may also be used at Kansas State University, Manhattan, Kansas.

11. The licensee shall comply with the provisions of Part 4, Kansas Radiation Protection Regulations, "Standards for Protection Against Radiation" and Part 10, Kansas Radiation Protection Regulations, "Notices, Instructions and Reports to Workers; Inspections".

12. Radioactive material shall be used by, or under the supervision of, Ray Lewis and/or Mike Nichols.

13. A. (1) Each sealed source acquired from another person and containing radioactive material, other than Hydrogen 3, with a half-life greater than thirty (30) days and in any form other than gas shall be tested for contamination and/or leakage prior to use. In the absence of a certificate from a transferor indicating that a test has been made within six (6) months prior to the transfer, a sealed source received from another person shall not be put into use until tested.

(2) Notwithstanding the periodic leak test required by this condition, any radioactive sealed source is exempt from such leak tests when the source contains 100 microcuries or less of beta and/or gamma emitting material or 10 microcuries or less of alpha emitting material.

(3) Except for alpha sources, the periodic leak test required by this condition does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another person unless they have been leak tested within six (6) months prior to the date of use or transfer.

B. Each sealed source fabricated by the licensee shall be inspected and tested for construction defects, leakage, and contamination prior to use or transfer as a sealed source. If the inspection or test reveals any construction defects or 0.005 microcurie or greater of contamination, the source shall not be used or transferred as a sealed source until it has been repaired, decontaminated and retested.

C. Each sealed source containing radioactive material, other than Hydrogen 3, with a half-life greater than thirty (30) days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six (6) months except that each source designed for the purpose of emitting alpha particles shall be tested at intervals not to exceed three months.

RADIOACTIVE MATERIALS LICENSE

Supplementary Sheet

License Number 26-B401-01

Amendment No. 4

D. The test shall be capable of detecting the presence of 0.005 microcurie of radioactive material on the test sample. The test sample shall be taken from the sealed source or from the surfaces of the device in which the sealed source is permanently or semipermanently mounted or stored on which one might expect contamination to accumulate. Records of leak test results shall be kept in units of microcuries and maintained for inspection by the Department.

E. If the test required by Subsection A or C of this condition reveals the presence of 0.005 microcurie or more of removable contamination, the licensee shall immediately withdraw the sealed source from use and shall cause it to be decontaminated and repaired or to be disposed of in accordance with Department regulations. A report shall be filed within five (5) days of the test with the Director, Bureau of Radiation Control, State of Kansas, Department of Health and Environment, Topeka, Kansas 66620, describing the equipment involved, the test results, and the corrective action taken.

14. Except as specifically provided otherwise by this license, the licensee shall possess and use radioactive material described in Items 6, 7, and 8 of this license in accordance with statements, representations, and procedures contained in the following documents:

    (a) The application dated January 29, 1982, signed by Glenn L. Koester, with attachments.

    (b) The letter dated March 17, 1982, signed by Craig A. Swartzendruber, with attachments.

    (c) The letter dated April 19, 1982, signed by Craig A. Swartzendruber, with attachments.

    (d) The letter dated May 11, 1982, signed by Craig A. Swartzendruber, with attachments.

15. The licensee may transport radioactive material or deliver radioactive material to a carrier for transport, in accordance with the provisions of Kansas Radiation Protection Regulations 28-35-195, "Transportation of Radioactive Materials", and 28-35-196, "Preparation of Radioactive Material for Transport".

16. The licensee shall conduct a physical inventory every six (6) months to account for all sealed sources received and possessed under the license. The records of the inventories shall be maintained for inspection by the Department, and shall include the quantities and kinds of radioactive material, location of sealed sources, and the date of the inventory.

FOR THE STATE DEPARTMENT OF HEALTH & ENVIRONMENT

Date MAY [illegible] 1982

By [signature]

Gerald W. Allen, Director
Bureau of Radiation Control

WOLF CREEK ONLY

3/4.7.10 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.10.1 The Fire Suppression Water System shall be OPERABLE with:

a. Two fire suppression pumps, each with a capacity of $\geq$3300 gpm, with their discharge aligned to the fire suppression header, and

b. An OPERABLE flow path capable of taking suction from the Wolf Creek Generating Station cooling lake and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve ahead of the deluge valve on each Deluge or Spray System required to be OPERABLE per Specifications 3.7.10.2, 3.7.10.3, and 3.7.10.4.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the Fire Suppression Water System otherwise inoperable establish a backup Fire Suppression Water System within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.10.1.1 The Fire Suppression Water System shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that lake level exceeds 1075 feet ~~(______ gallons)~~,

b. At least once per 31 days on a STAGGERED TEST BASIS by starting the electric motor driven pump and operating it for at least 15 minutes on recirculation flow,

c. At least once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path is in its correct position,

W.C. ONLY

d. At least once per 6 months by performance of a yard loop and fire hydrant flush,

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel,

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

1) Verifying that each automatic valve in the flow path actuates to its correct position,

2) Verifying that each pump develops at least 3300 gpm at a system head of 277 feet,

3) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

4) Verifying that each fire suppression pump starts (sequentially) on decreasing pressure in the fire suppression header at a header pressure greater than or equal to 80 psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

4.7.10.1.2 Each fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1) The fuel storage tank Contains ~~level is~~ at least ~~3/4 tank level~~ (Later) gallons of fuel~~)~~, and

2) The diesel starts from ambient conditions and operates for at least 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel Storage ~~oil day~~ tank, obtained in accordance with ASTM-D270-1975, is within the acceptable limits specified in Table 1 of ASTM D975-1977 when checked for viscosity, water, and sediment; and

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

Specification: 4.7.10.1.1.a

Justification:

Delete the reference to gallons. The Wolf Creek Cooling Lake holds millions of gallons of water even at the low operating level of 1075 feet. Normally when fire protection storage water tanks are used, two tanks of 300,000 gallon capacity are required.

Specification: 4.7.10.1.2.a.1 and b.

Justification:

The gallonage figure which Wolf Creek will provide at a later date will reflect the fuel quantity required to provide 8 hours of diesel fire pump operation at full, rated load. This commitment should meet the intent of all known code and regulatory position statements on the minimum fuel reserve for diesel fire pump operation.

Wolf Creek has only one storage tank for diesel fuel for the diesel fire pump. It does not have a separate fuel oil storage tank and day tank.

UE ONLY

3/4.7.10 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.10.1 The Fire Suppression Water System shall be OPERABLE with:

a. Two ~~Three~~ fire suppression pumps, each with a capacity of 1500 gpm, with their discharge aligned to the fire suppression header;

b. Two separate water supply tanks, each with a minimum level of 29.5 feet (250,000 gallons); and

c. An OPERABLE flow path capable of taking suction from both fire water storage tanks and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve ahead of the deluge valve on each deluge or spray system required to be OPERABLE per Specifications 3.7.10.2, and 3.7.10.4.

APPLICABILITY: At all times.

ACTION:

a. With less than two pumps OPERABLE ~~With one pump and~~/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the Fire Suppression Water System otherwise inoperable establish a backup Fire Suppression Water System within 24 hours.

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SURVEILLANCE REQUIREMENTS

4.7.10.1.1 The Fire Suppression Water System shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the water level in each fire water storage tank exceeds 29.5 feet (______ gallons),

b. At least once per 31 days on a STAGGERED TEST BASIS by starting the electric motor-driven pump and operating it for at least 15 minutes on recirculation flow,

c. At least once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path is in its correct position,

U.E d. At least once per ~~6~~ 18 months by performance of a ~~system~~ YARD LOOP AND HYDRANT flush,

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel,

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

1) Verifying that each automatic valve in the flow path actuates to its correct position,

U.E 2) Verifying that each pump develops at least 1500 gpm at a system head of ~~150~~ 135 psig,

4 ~~3)~~ Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

5 ~~4)~~ Verifying that each fire suppression pump starts (sequentially) on decreasing pressure in the fire suppression header at a header pressure greater than or equal to 110 psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

3.) VERIFYING EACH PUMP DELIVERS AT LEAST 2250 gpm AT A SYSTEM PRESSURE OF 105 psig

CAL ONLY

SURVEILLANCE REQUIREMENTS (Continued)

4.7.10.1.2 Each fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1) The fuel oil day tank ~~level is at~~ CONTAINS AT least ~~3/4 tank level.~~ 175 gallons of fuel, and

2) The diesel starts from ambient conditions and operates for at least 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel oil day tank, obtained in accordance with ASTM-D270-1975, is within the acceptable limits specified in Table 1 of ASTM D975-1977 when checked for viscosity, water, and sediment; and

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.10.1.3 Each fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The electrolyte level of each battery is above the plates, and

2) The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery; and

c. At least once per 18 months by verifying that:

1) The batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration, and

2) The battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anti-corrosion material.

Specification 3.7.10.1 and 4.7.10.1.1.f.3 Page 3/4 7-28 & 29

Justification -

The largest sprinkler system demands in safety-related areas in terms of flow and pressure, are for the upper and lower cable spreading rooms and auxiliary building cable trays on elevations 2026', 2000' and 1974'. Hydraulic calculations are attached which determine the required flow and pressure at the interfaces between Bechtel and the site A-E in the building supply laterals that service these systems. These calculations include an additional 1000 gpm for hose streams and no credit is taken for the internal loops within buildings and the interconnections between buildings. Hose flow is assumed to be through lines that feed the sprinkler system which will yield a conservative result. Pressure losses in the yard loop are considered negligible and pressure losses between the yard loop and Bechtel/site A-E interface point is assumed to be overcome by the elevation difference of the pumps above the yard loop.

Each of Callaway's fire pumps is rated at 1500 gpm at 150 psig. Per NFPA 20, a fire pump must be capable of providing 150% of rated capacity (2250 gpm in this case) at greater than 65% of rated system head (98 psig in this case). The Technical Specification limit of 2250 gpm at 105 psig is more restrictive than NFPA 20 and ensures adequate pump performance for each of the sprinkler systems addressed. Given this, each fire pump can supply the largest safety-related water demand alone. Thus, considering only safety-related areas, Callaway has three 100% capacity pumps and only two of the three must remain operable.

Technical Specification 4.7.10.1.1.d

Justification

The water supply at Callaway is drawn from covered storage tanks, not a river or open reservoir. As a result, the accumulation of sediment within system piping is minimal. No National Fire Protection Association standards require a system flush beyond the original post-construction flush unless problems due to sedimentation or blockage become evident. A refueling interval flush is sufficient to remove sedimentation from Callaway yard loop fire protection system piping.

The design of the Fire Protection water supply feed into safety related buildings is such that water can only flow in to the building, not out, due to check valves. Thus a system flush into the buildings would require a means of disposing of a considerable quantity of water within the safety related buildings. Surveillance requirements specified for the sprinkler and hose systems are sufficient to ensure an adequate water supply to these systems. Since the source of water to these systems is from loops which are flushed and assured to be clean, flushing into these buildings is unnecessary and would involve a considerable expenditure of time and effort to dispose of the water, it is therefore appropriate to require only the yard loop and hydrants to be flushed.

Specification 4.7.10.1.1.f.2, Page 3/4 7-29

Justification

The purpose of this surveillance is to verify fire pump performance is not reduced significantly. The previous requirement was to verify pump performance at or above the rated point. Such an acceptance criteria is appropriate for a pre-operational test wherein one is ensuring adequate performance of new equipment to a design specification, but not for a surveillance performed over plant life. The rated point of 1500 gpm at a system head of 150 psig is, by design, a point on the characteristic curve for the fire pumps. If the pumps are required to operate at this point throughout plant life, any reduction in pump performance, no matter how insignificant, would be a violation of Technical Specifications. In fact, given the tolerances allowed in the manufacturing process, degraded pump components could be replaced with new components and pump performance could still be below the rated point. Considering the quality of water being pumped and the type of service the pumps will experience, expecting no reduction in performance is unrealistic. It is more appropriate to set realistic limits which, if exceeded, would indicate degradation of pump components sufficient to merit pump repairs. The 10% reduction in pump performance specified is appropriate because it represents enough of a reduction in pump performance to indicate the need for pump maintenance, but is not sufficiently low to impair the pumps from performing their safety-related function. The capability of each pump to perform its safety-related function is verified in new surveillance 4.7.10.1.1.f.3.

Technical Specification 4.7.10.1.2.a

Justification:

Add 175 gallons of fuel to day tank contained volume. Delete reference to 3/4 tank level. We will add required calibrated limits for level to assure necessary contained volume in appropriate surveillance procedure.

CALCULATIONS OF REQUIRED PRESSURE/FLOW FOR WORST-CASE SAFETY-RELATED SPRINKLERS

Assumptions:

1) Pressure drop in 14 inch yard loop at the flows specified is negligible

$$p[\frac{psi}{ft}] = \frac{4.52\ (2160)^{1.85}}{(120)^{1.85}(14)^{4.87}} = \frac{4.52\ (1,474,821)}{(7022)\ (381,631)} = 0.0025\ \frac{psi}{ft}$$

2) Elevation of pump discharge line is sufficiently above the elevation of the feeder lines into the buildings to overcome any friction losses in the feeder lines between the yard loop and the interface point between Bechtel and S & P.

Pump Disch Elev.	= 2001 ft.
High Interface Elev.	= 1991 ft.
Δ Elevation	= 10 ft. = 4.34 psi

The length of pipe that would yield this op is:

$$l = \frac{(4.34)(120)^{1.85}(8)^{4.87}}{(4.52)(2160)^{1.85}} = \frac{(4.34)(7022)(25,006)}{(4.52)(1,474,821)} = 114\ ft.$$

3) c = 120

4) Add 1000 gpm to required flow for hose streams and hose flow is through lines that feed sprinklers.

5) Only one flow path is addressed and no credit is taken for internal loops within buildings and interconnection between buildings.

References:

M-650A-003
M-650A-013
M-650A-023
FP 95211
M-03KC17
M-03KC19
M-650A-004
M-650A-015
NFPA 13
M-03KC04
M-03KC18
8600-X-88938

UPPER CABLE SPREADING ROOM

Flow and pressure required at Bechtel/Vendor interface = 963 gpm @ 50 psi
Flow path from Bechtel/S & P interface:

- 191-KB[illegible]"
- KC-V390
- KC-V389
- 477-KBF-8"
- 553-KBF-6"
- 556-KBF-6"
- KC-V442

KC-V442 Elevation = 2049' 10 ½"
Interface Elevation = 1976' 6 ½"
Δ Elevation = 73'4" = 73.34 ft.
Elevation Δ p = (73.34 ft) (.4335 $\frac{psi}{ft}$) = 31.79 psi

8" Pipe

Component	Equivalent Length (ft)
18'	18
T	35
3' 6"	3.5
90°	18
16"	1.33
Chk Vlv	45
Gte Vlv	4
28"	2.33
90°	18
3' 2 1/2"	3.21
	148.37 ft

6" Pipe

Component	Equivalent Length (ft)
T	30
22' 4 3/8"	22.36
T	30
29' 3"	29.25
45°	7
17"	1.42
45°	7
2' 9"	2.75
45°	7
17"	1.42
45°	7
35' 3"	35.25
Gte Vlv	3
	183.45 ft

$$8"\ \Delta p = \frac{4.52\ (1963)^{1.85}(148.37)}{(120)^{1.85}\ (8)^{4.87}} = \frac{4.52(1{,}235{,}670)(148.37)}{(7022)\ (25{,}006)} = 4.72 \text{ psi}$$

$$6"\ \Delta p = \frac{4.52\ (1963)^{1.85}(183.45)}{(120)^{1.85}\ (6)^{4.87}} = \frac{4.52(1{,}235{,}670)(183.45)}{(7022)\ (6160)} = 23.69 \text{ psi}$$

Total p required = 31.79 + 4.72 + 23.69 + 50 = 110.2 psi

Required Pump Performance = 1963 gpm @ 111 psi

LOWER CABLE SPREADING ROOM

Flow and pressure required at Bechtel/Vendor interface - 998 gpm @ 60 psi
Flow path from Bechtel/S & P interface:

- 191-KBF-8"
- KC-V390
- KC-V389
- 477-KBF-8"
- 553-KBF-6"
- 555-KBF-6"
- KC-V441

KC-V441 Elevation = 2018'
Interface Elevation = 1976' 6 1/2"
Δ Elevation = 41' 5 1/2" = 41.46'
Elevation Δp = (41.46 ft) (.4335 $\frac{psi}{ft}$) = 17.97 psi

6" Pipe

Component	Equivalent Length (ft)
T	30
22' 4 3/8"	22.36
T	30
29' 3"	29.25
45°	7
17"	1.42
45°	7
2' 9"	2.75
45°	7
17"	1.42
45°	7
4' 3"	4.25
T	30
9' 1/8"	.76
Gte Vlv	3
	183.21 ft

8" Pipe

Same as Upper Cable Spreading Room = 148.37 ft

$$8" \; \Delta p = \frac{4.52(1998)^{1.85}(148.37)}{(120)^{1.85}\;(8)^{4.87}}$$

$$= \frac{4.52(1,276,738)(148.37)}{(7022)(25,006)} = 4.88 \text{ psi}$$

$$6" \; \Delta p = \frac{4.52(1998)^{1.85}\;(183.21)}{(120)^{1.85}\;(6)^{4.87}}$$

$$= \frac{4.52(1,276,738)(183.21)}{(7022)\;(6160)} = 24.44 \text{ psi}$$

Total p required = 17.97 + 4.88 + 24.44 + 60 = 107.29 psi

Required pump performance = 1998 gpm @ 108 psi

U-1

Flow and pressure required at Bechtel/Vendor interface = 902 gpm @ 50 psi
Flow path from Bechtel/S & P interface:

447-KBF-8"
KC-V376
KC-V377
473-KBF-8"
551-KBF-6"
557-KBF-6"
KC-V445

KC-V445 Elevation = 2028'
Interface elevation = 1991'
Δ Elevation - 37 ft.
Elevation Δp = (37 ft.)(.4335 $\frac{psi}{ft}$) = 16.04 psi

Component	8" Pipe Equivalent Length (ft)	Component	8" Pipe Equivalent Length (ft)
19' 1 1/8"	19.09	2'	2
45°	9	90°	18
54' 11"	54.92	9' 2"	9.17
90°	18	90°	18
5' 9"	5.75	5' 5 3/16"	5.43
90°	18	90°	18
6'9"	6.75	14' 6"	14.5
90°	18	90°	18
27'	27	2'	2
45°	9	90°	18
2' 1/16"	2.01	5' 5"	5.42
45°	9		486.71 ft
4' 5"	4.42		
90°	18		
3' 11 1/2"	3.96		
90°	18		
7' 4"	7.33		
T	35		
1' 7"	1.58		
90°	18		
6' 4 1/2"	6.38		
Chk Vlv	45		
Gte Vlv	4		

U-1

Component	6" Pipe Equivalent Length (ft)
T	30
7' 3 11/16"	7.31
T	30
9 6/16"	.78
45°	7
7 1/2"	.63
45°	7
42' 5/16"	42.03
90°	14
12 1/2"	1.04
Gte Vlv	3
	142.79 ft

$$8" \ \Delta p = \frac{4.52\ (1902)^{1.85}\ (486.71)}{(120)^{1.85}\ (8)^{4.37}} = \frac{4.52\ (1,165,573)\ (486.71)}{(7022)\ (25,006)} = 14.6 \text{ psi}$$

$$6" \ \Delta p = \frac{4.52\ (1902)^{1.85}\ (142.79)}{(120)^{1.85}\ (6)^{4.87}} = \frac{4.52(1,165,573)\ (142.79)}{(7022)\ (6160)} = 17.39 \text{ psi}$$

Total p required = 16.04 + 14.6 + 17.39 + 50 = 98.03 psi

Required pump performance = 1902 gpm @ 99 psi

AUX BLDG CABLE TRAYS 2000'

Flow and pressure required at Bechtel/Vendor interface = 1047 gpm @ 55 psi
Flow path from Bechtel/S & P interface:

447-KBF-8"
KC-V376
KC-V377
473-KBF-8"
551-KBF-6"
559-KBF-6"
KC-V444

KC-V444 Elevation = 2002'
Interface Elevation = 1991'
Δ Elevation = 11 ft.
Elevation $\Delta p = (11 \text{ ft.})(.4335 \frac{\text{psi}}{\text{ft}}) = 4.77$ psi

Component	6" Pipe Equivalent Length (ft.)
T	30
7' 3 11/16"	7.31
T	30
9 6/16"	.78
45°	7
7 1/2"	.63
45°	7
16' 11 5/16"	16.94
T	30
9 1/8"	.76
Gte Vlv	3
	133.42 ft

8" Pipe

Same as Aux Bldg Cable Trays 2026' = 486.71 ft

$$8" \ \Delta p = \frac{4.52(2047)^{1.85}\ (486.71)}{(120)^{1.85}\ (8)^{4.87}}$$

$$= \frac{4.52(1,335,267)(486.71)}{(7022)\ (25,006)} = 16.73 \text{ psi}$$

$$6" \ \Delta p = \frac{4.52(2047)^{1.85}\ (133.42)}{(120)^{1.85}\ (6)^{4.87}}$$

$$= \frac{4.52(1,335,267)\ (133.42)}{(7022)\ (6160)} = 18.62 \text{ psi}$$

Total p required = 4.77 + 16.73 + 18.62 + 55 = 95.12 psi

Required pump performance = 2047 gpm @ 96 psi

AUX BLDG CABLE TRAYS 1974'

Flow and pressure required at Bechtel/Vendor interface = 1160 gpm @ 70 psi
Flow path from Bechtel/S & P interface:

- 447-KBF-8"
- KC-V376
- KC-V377
- 473-KBF-8"
- 551-KBF-6"
- 558-KBF-6"
- KC-V443

KC-V443 Elevation = 1976'
Interface Elevation = 1991'
Δ Elevation - 15 ft.
Elevation $\Delta p = (-15 \text{ ft.}) (.4335 \frac{\text{psi}}{\text{ft}}) = -6.5$ psi

6" Pipe

Component	Equivalent Length (ft)
T	30
7' 3 11/16"	7.31
T	30
7' 10"	7.83
90°	14
12 1/2"	1.04
Gte Vlv	3
	93.18 ft

8" Pipe

Same as Aux Bldg Cable Trays 2026' = 486.71 ft.

$$8" \ \Delta p = \frac{4.52\ (2160)^{1.85}\ (486.71)}{(120)^{1.85}\ (8)^{4.87}}$$

$$= \frac{4.52(1,474,821)(486.71)}{(7022)(25,006)} = 18.48 \text{ psi}$$

$$6" \ \Delta p = \frac{4.52\ (2160)^{1.85}\ (93.18)}{(120)^{1.85}\ (6)^{4.87}}$$

$$= \frac{4.52(1,474,821)(93.18)}{(7022)\ (6160)} = 14.36 \text{ psi}$$

Total p required = -6.5 + 18.48 + 14.36 + 70 = 96.34 psi

Required pump performance = 2160 gpm @ 97 psi

PUMP HEAD AT FLOW

(from certified pump curves)

Pump	1900 gpm	1965 gpm	2000 gpm	2050 gpm	2160 gpm	2250 gpm
PKC1001A	139 psi	136 psi	136 psi	134 psi	130 psi	125 psi
PKC1002A	139 psi	136 psi	135 psi	133 psi	128 psi	124 psi
PKC1002B	134 psi	130 psi	128 psi	125 psi	119 psi	113 psi

If pump performance degrades to 2250 gpm at 105 psi, assuming pump curve is decreased the same percentage over the section in question.

Pump	1900 gpm	1965 gpm	2000 gpm	2050 gpm	2160 gpm	2250 gpm
PKC1001A	116 psi	114 psi	114 psi	112 psi	109 psi	105 psi
PKC1002A	117 psi	115 psi	114 psi	112 psi	108 psi	105 psi
PKC1002B	124 psi	120 psi	118 psi	116 psi	110 psi	105 psi
Required*	99 psi	111 psi	108 psi	96 psi	97 psi	---

* from previous calculations

Mose restrictive case is Upper Cable Spreading Room which requires 1963 gpm at 111 psi. If individual pump performance is reduced to 2250 gpm at 105 psi, each pump will still exceed this requirement.

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.10.2 The following Spray and/or Sprinkler Systems shall be OPERABLE:

a. Wet Pipe Sprinkler Systems:

Building	Elevation	Area Protected
Auxiliary	2000	North Electric Cable Chase
Auxiliary	1988/2000	South Electric Cable Chase
Control	1974 - 2073	Vertical Electrical Chases
~~Control~~	~~1984~~	~~Access Control Area Below Ceiling~~
Control	1992	Cable Area Above Access Control
Control	1974	Pipe Space and tank room

b. Pre-Action Sprinkler Systems

Building	Elevation	Area Protected
Auxiliary	1974	Cable Trays *
Auxiliary	2000	Cable Trays *
Auxiliary	2026	Cable Trays *
Control	2032	Lower Cable Srpeading Room
Control	2073	Upper Cable Penetration Area
Reactor	2026	North Cable Penetration Area
Reactor	2026	South Cable Penetration Area
Diesel Gen. (E)	2000	East Diesel Generator Room
Diesel Gen. (W)	2000	West Diesel Generator Room
~~Fuel Building~~	~~2000~~	~~Railroad Bay Area~~

c. Water Sprays Systems

Building	Elevation	Area Protected
Auxiliary	2000	Auxiliary Feedwater Pump Turbine

APPLICABILITY: Whenever equipment protected by the Spray/Sprinkler System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Spray and/or Sprinkler Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish a hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.10.2 Each of the above required Spray and/or Sprinkler Systems shall be demonstrated OPERABLE:

* AREAS CONTAIN REDUNDANT SYSTEMS OR COMPONENTS WHICH COULD BE DAMAGED.

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 31 days, by verifying that each valve (manual, power operated, or automatic) in the flow path is in its correct position,

b. At least once per 12 months, by cycling each testable valve in the flow path through at least one complete cycle of full travel,

c. At least once per 18 months:

1) By performing a system functional test which includes simulated automatic actuation of the system, and:

a) Verifying that the automatic valves in the flow path actuate to their correct positions on a Simulated Fire test signal, and

b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2) By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity, and

3) By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

d. At least once per 3 years by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

The term "simulated fire" test signal is interpeted to mean actuation of an automatic Fire Protection system by any of the release mechanisms provided, e.g. fire detectors, hand pull stations, fusible link/mechanical, manual, hydro/mechanical, etc.

Specification: 3.7.10.2.a

Justification:

A wet pipe sprinkler system has been added to the SNUPPS design in order to comply with 10CFR50 Appendix R Section IIIG.2. This system provides protection on the 1974 elevation of the Control Building.

Specification: 4.7.10.2.c.1.a)

Justification:

The note was added to clarify what is meant by a "simulated fire" test signal.

Technical Specification: 3.7.10.2 Page 3/4 7-31

Justification:

a) Delete Wet Pipe Sprinkler System in the Control Building, elevation 1984, access control below ceiling.

b) There are no safety related equipment or systems be.~w the ceiling. The area is protected above access control by the sprinkler system on elevation 1992.

c) Delete pre-action sprinkler system in the Fuel Building, elevation 2000, Railroad Bay Area. There is no equipment in this area which would prevent a safe plant shutdown.

d) Proposed change to tables of Systems adds asterisks to areas which are protected by spray or sprinkler systems and contain redundant Safety Systems or components associated with safe shutdown. This change helps Operations to determine which part of action statement (a) applies if parts of the Systems are determined to be inoperable.

PLANT SYSTEMS

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.10.3 The following Halon Systems shall be OPERABLE.

	Building →	Elevation	Area Protected
a.	Auxiliary	2026	North Electrical Penetration Room
b.	Auxiliary	2026	South Electrical Penetration Room
c.	Auxiliary	2026	Load Center and M. G. Sets Room *
d.	Control	2000	ESF Switchgear Rooms *
e.	Control	2016	Switchgear Rooms
f.	Control	2047	Control Room Cable Trenches and Chases

APPLICABILITY: Whenever equipment protected by the Halon System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.10.3 Each of the above required Halon Systems 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 in its correct position;~~

b. At least once per 6 months by verifying Halon storage tank weight (or level) to be at least 95% of full charge weight and pressure to be at least 90% of full charge pressure; and

c. At least once per 18 months by:

1) Verifying the system, including associated ventilation dampers and fire door release mechanisms, actuates manually and automatically, upon receipt of a simulated actuation signal, and

2) Performance of a flow test through headers and nozzles to assure no blockage.

* Areas contain redundant systems or components which could be damaged.

Specification 4.7 10.3.a, Page 3/4 7-33

Justification

Neither ANI nor NFPA #12A recommend such an inspection. SNUPPS design has solenoid valves which do not lend themselves to an external visual inspection. Do not feel it is necessary to tear down solenoid valves monthly. Know of no operating plants doing this weekly test to verify valve position (i.e., halon cylinders have not discharged).

Specification 4.7.10.3.b, Page 3/4 7-33

Justification

There are acceptable methods available to determine the level in the Halon cylinders which can be equated to weight. Addition of the use of level will remove the necessity to remove each cylinder and weigh it.

Technical Specification 3.7.10.3 Page 3/4 7-33

Justification:

Proposed change adds asterisks to areas which are protected by Halon Systems and contain redundant Safety Systems or components associated with safe shutdown. This clarification is needed by operations to determine which part of action (a) applies if parts of the Halon System are determined to be inoperable.

PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.10.4 The fire hose stations shown in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

route an equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station,

a. With one or more of the fire hose stations shown in Table 3.7-5 inoperable, ~~provide gated wye (s) on the nearest OPERABLE hose station (s). One outlet of the wye shall be connected to the standard length of hose provided for the hose station. The second outlet of the wye shall be connected to a length of hose sufficient to provide coverage for the area left unprotected by the inoperable hose station. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to operating technicians, plant equipment, or the hose itself, the fire hose shall be stored in a roll at the outlet of the OPERABLE hose station. Signs shall be mounted above the gated wye (s) to identify the proper hose to use.~~ The above ACTION shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.10.4 Each of the fire hose stations shown in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days by a visual inspection of the fire hose stations accessible during plant operations to assure all required equipment is at the station;

b. At least once per 18 months by:

   1) Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station,

   2) Removing the hose for inspection and re-racking, and

   3) Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years by:

   1) Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage, and

   2) Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

Specification 3.7.10.4 Page 3/4 7-34

Justification -

1. Hose stations are not designed to accommodate gated wye connections. Areas have access to stand pipe systems which may provide a preferred alternate source over adjacent hose stations. Hose routed, where practicable, from adjacent stations provide adequate protection in case of fire for either the affected area or area from which the hose has been routed.

TABLE 3.7-5

FIRE HOSE STATIONS

BUILDING	ELEVATION	AREA	HOSE RACK
Auxiliary	1974	1122	KC-HR-051
Auxiliary	1974	1122	KC-HR-047
Auxiliary	1974	1120	KC-HR-031
Auxiliary	1974	1120	KC-HR-025#
Auxiliary	1974	1101	KC-HR-023#
Auxiliary	1974	1101	KC-HR-040
Auxiliary	1974	1101	KC-HR-042
Auxiliary	1988	1201	KC-HR-024
Auxiliary	2000	1329	KC-HR-111
Auxiliary	2000	1320	KC-HR-048
Auxiliary	2000	1320	KC-HR-046#
Auxiliary	2000	1314	KC-HR-030
Auxiliary	2000	1321	KC-HR-029#
Auxiliary	2000	1301	KC-HR-035#
Auxiliary	2000	1301	KC-HR-039
Auxiliary	2000	1301	KC-HR-041#
Auxiliary	2026	1408	KC-HR-049
Auxiliary	2026	1408	KC-HR-044
Auxiliary	2026	1408	KC-HR-032#
Auxiliary	2026	1408	KC-HR-026#
Auxiliary	2026	1401	KC-HR-034
Auxiliary	2026	1403	KC-HR-037#
Auxiliary	2047	1506	KC-HR-050
Auxiliary	2047	1513	KC-HR-043
Auxiliary	2047	1506	KC-HR-045
Auxiliary	2047	1501	KC-HR-038
Auxiliary	2047	1504	KC-HR-033
Auxiliary	2047	1502	KC-HR-027
Auxiliary	2064	1119	KC-HR-028#
Control	1974	3101	KC-HR-002#
Control	1974	3101	KC-HR-014#
Control	1984	3204	KC-HR-015#
Control	1984	3221	KC-HR-001#
Control	2000	3301	KC-HR-004#
Control	2000	3301	KC-HR-017#
Control	2000	3302	KC-HR-016#
Control	2016	3401	KC-HR-005
Control	2016	3401	KC-HR-019
Control	2016	3401	KC-HR-018

TABLE 3.7-5 (Continued)

FIRE HOSE STATIONS

BUILDING	ELEVATION	AREA	HOSE RACK
Control	2032	3501	KC-HR-006#
Control	2032	3501	KC-HR-020#
Control	2047	3604	KC-HR-007
Control	2047	3616	KC-HR-021
Control	2073	3801	KC-HR-008#
Control	2073	3801	KC-HR-022#
Reactor	2000	2201	KC-HR-120##
Reactor	2000	2201	KC-HR-131##
Reactor	2000	2201	KC-HR-124##
Reactor	2000	2201	KC-HR-129##
Reactor	2026	N.A.	KC-HR-121##
Reactor	2026	N.A.	KC-HR-132##; #
Reactor	2026	N.A.	KC-HR-125##
Reactor	2026	N.A.	KC-HR-130##
Reactor	2047	N.A.	KC-HR-128##
Reactor	2047	N.A.	KC-HR-122##
Reactor	2047	N.A.	KC-HR-126##
Reactor	2068	N.A.	KC-HR-123##
Reactor	2068	N.A.	KC-HR-127##
Fuel	2000	6102	KC-HR-142#
Fuel	2000	6102	KC-HR-054#
Fuel	2000	6102	KC-HR-143
Fuel	2000	6104	KC-HR-057
Fuel	2026	6201	KC-HR-133
Fuel	2026	6203	KC-HR-052
Fuel	2047	6301	KC-HR-055#
Fuel	2047	6302	KC-HR-056#
Fuel	2047	6301	KC-HR-053#
ESW	2000	N.A.	KC-HR-140
ESW	2000	N.A.	KC-HR-141

Secondary means of fire suppression to a Primary Water Spray/Deluge or Halon System.

Fire hose for station to be stored external to Reactor Building.

Table 3.7-5

Justification -

\# note. Added reference allows easy reference to secondary stations which clarifies time frame in which action statement must be satisfied.

\## note. Hose stations inside containment require 975 feet of fire hose. This results in exposing hose to excessive environmental conditions and contamination. Surveillance such as visual examinations and hydro tests will be complicated by contamination. This will also result in the production of additional non-compactible wastes when hose is replaced. U.E. will store approximately 200 feet of hose, external to containment and accessible to the Fire Brigade response team if needed to combat fire emergencies.

3/4.7.11 FIRE BARRIER PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.7.11 All fire barrier penetrations (walls, floor/ceilings, cable tray enclosures and other fire barriers) separating safety related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable, piping and ventilation duct penetration seals) shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

Insert A.

a. ~~With one or more of the above required fire barrier penetrations inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected penetration or verify the OPERABILITY of fire detectors on at least one side of the inoperable fire barrier and establish an hourly fire watch patrol.~~

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

SURVEILLANCE REQUIREMENTS

4.7.11.1 At least once per 18 months the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE by performing a visual inspection of:

a. The exposed surfaces of each fire rated assembly,

b. Each fire window/fire damper/and associated hardware, and

c. At least 10 ~~percent~~ % of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10 ~~percent~~ % of each type of sealed penetration shall be made. This inspection process shall continue until a 10 ~~percent~~ % sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected every 15 years.

4.7.11.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. The OPERABILITY of the Fire Door Supervision System for each electrically supervised fire door by performing a TRIP ACTUATING DEVICE OPERATIONAL TEST at least once per 31 days,

b. At least once per 7 days that each locked closed fire door is in the closed position ~~That each locked closed fire door is closed at least once per 7 days~~,

c. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours and performing a functional test at least once per 18 months, and

d. At least once per 24 hours that each unlocked fire door without electrical supervision is in the closed ~~at least once per 24 hours.~~ position,

INSERT A FOR 3.7.11, ACTION

ACTION:

a. With one or more of the required fire rated assemblies and/or sealing devices inoperable, within 1 hour either:

   1) establish a continuous fire watch on at least one side of the affected assembly, or

   2) verify the operability of fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol, or

   3) install a temporary sealant material of at least an equal fire resistance rating to the inoperable assembly and/or sealing device and establish an hourly fire watch patrol.

Specification: 3.7.11, ACTION

Justification:

The above change more clearly defines the 3 remedial options available to the licensee.

Option 3 is a new alternative method for achieving the desired level of temporary fire safety. Products such as BIO Fire Protection Systems, "Fire Protection Pillows", would be used to provide the temporary level of protection. These units have met the U.L. 3 hour fire tests for both flame and heat propagation.

Specification 4.7.11.2b and d Page 3/4 7-37

Justification -

The changes to these surveillances clearly defines the intent of the surveillances.

3/4.7.12 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.12 The temperature of each area shown in Table 3.7-6 shall be maintained within the limits indicated in Table 3.7-6.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION: INSERT

~~With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6:~~

a. ~~For more than 8 hours, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.~~

b. ~~By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to within its temperature limit or declare the equipment in the affected area inoperable.~~

SURVEILLANCE REQUIREMENTS

4.7.13 The temperature in each of the areas shown in Table 3.7-6 shall be determined to be within its limit at least once per 12 hours.

Specification 3.7.12 ACTION Page 3/4 7-38

Action -

a. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6 for more than eight hours, in lieu of any other reporting requirement, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a report of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

b. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6 by more than 30 degrees F, in addition to the Special Report required above, within four hours either restore the area to within its temperature limit or declare the equipment in the affected area inoperable.

Justification -

The NRC Draft 2 version of this specification would require an LER each time the temperature in the areas exceeded the limit by any amount for any duration. Since the equipment housed in the specified areas is under separate surveillance requirements the probability of a threat to the health and safety of the public from an entry into the Action a does not warrant an LER nor should it preclude a return to power while in the Action Statement.

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 between the offsite transmission network and the Onsite Class 1E Distribution System, and

b. Two separate and independent diesel generators, each with:

1) A separate day tank containing a minimum volume of 390 gallons of fuel,

2) A separate Fuel Oil Storage System containing a minimum volume of 85,300 gallons of fuel,

3) A separate fuel transfer pump,

~~4) Lubricating oil storage containing a minimum total volume of ( ) gallons of lubricating oil; and~~

~~5) Capability to transfer lubricating oil from storage to the diesel generator unit.~~

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

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specifications 4.8.1.1.1a. and 4.8.1.1.2a.4) within 1 hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators 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 offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specifications 4.8.1.1.1a. and 4.8.1.1.2a.4) within 1 hour and at least once per 8 hours thereafter; restore at least 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 at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one diesel generator inoperable in addition to ACTION a. or b. above, verify that:

1. All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

See insert

INSERT FOR 3.8.1.1, ACTION C

The term "verify" as used in this specification means to administratively check by use of logs, control room indication, or other means that the required components are in service. It does not mean to perform surveillances for these components as spelled out in the specifications for those components.

SURVEILLANCE REQUIREMENTS (Continued)

penetration

Verifying that the Emergency Exhaust System satisfies the in-place and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance in 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 9000 cfm ± 10%;

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, for a methyl iodide penetration of less than 1%; and

Verifying a system flow rate of 9000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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, for a methyl iodide penetration of less than 1%;

b. At least once per 18 months by:

~~1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 5.4 inches Water Gauge while operating the system at a flow rate of 9000 cfm ± 10%,~~

1) Verifying that on a Spent Fuel Pool High Radioactivity test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks and isolates the normal fuel building exhaust flow to the auxiliary/fuel building exhaust fan;

Technical Specification: 3.8.1.1.b.4,5 Page 3/4 8-1
4.8.1.1.2.a.4 Page 3/4 8-3
3.8.1.2.b.4,5 Page 3/4 8-8

Justification - (reference Attachment 1)

Eliminate reference to lube oil storage and transfer system in LCO for diesel operability. Lube oil useage and contained volume of oil in the crankcase provide sufficient volumes to allow operation of system for greater than 7 days. The system does represent a convenient method to add oil and maintain a level in the crankcase. It should not be used to determine diesel operability.

Assuming the diesel is in a stand-by mode and is at the add oil mark on the dip stick, a contained volume of 948 gallons of lube oil remains in the sump. Since the diesel can be run and will not fail at 300 gallons of lube oil in the crankcase, an allowable consumed volume of 648 gallons of lube oil exists. This 648 gallons represents a 10 day reserve supply since the diesel uses 449.67 gallons of lube oil in a 7 day period. If credit were taken for the lube oil control system, between 1063 and 1143 gallons of oil would exist. At the lower end of the control level, an additional volume of 115 gallons would exist. This would result in a reserve supply of 11.8 days of operation.

Prior to controlled start-ups of the diesel and after surveillance on shutdown, verification of crankcase levels as well as other critical temperature and equipment status data are obtained.

ATTACHMENT 1

Crankcase Level*		Crankcase Volume
-22.2"	Hi alarm	1215 gallons
-22.575"	Dip stick-full mark	1200 gallons
-24.0"	Close make up valve	1143 gallons
-26.0"	Open make up valve	1063 gallons
-28.5"	Low alarm	963 gallons
-28.875"	Dip stick-add oil mark	948 gallons
	Empty	300 gallons

Crankcase in area of dip stick and level control represents .40 gallons/inch.

* 0" reference at crank shaft centerline.

Specification: 3.8.1.1.c

Justification:

Use of this note is essential to clarify to the operator what is meant by "verifying" the listed items. Revision 2, Technicial Specifications for Westinghouse Pressurized Water Reactors did have such a note present. SNUPPS viewed this as a good practice which made it clear to the operator exactly what requirements had to be satisfied.

ACTION: (Continued)

2. When in MODE 1, 2, or 3, the steam-driven auxiliary feedwater pump is OPERABLE.

If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Specification 4.8.1.1.2a.4) within 1 hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least 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. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required 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 alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring manually unit 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,

SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying the fuel level in the fuel storage tank,

3) Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day tank,

~~4) Verifying the lubricating oil inventory in storage,~~

5) Verifying the diesel starts from ambient condition and accelerates to at least 471 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 $\pm$ 420 volts and 60 $\pm$ 1.2 Hz within 10 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Simulated loss of offsite power by itself, or

c) Safety Injection Test Signal ~~Simulated loss of offsite power in conjunction with an ESF actuation test signal, or~~

~~d) An ESF actuation test signal by itself.~~

6) Verifying the generator is synchronized, loaded to greater than or equal to 6201 kW ~~in less than or equal to 60 seconds~~, and operates ~~with a load greater than or equal to 6201 kW~~ for at least 60 minutes, and

7) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tanks;

Replace with next page to be submitted later

c. ~~At least once per 92 days and from new fuel oil prior to addition to the storage tanks by verifying that a sample obtained in accordance with ASTM-D270-1975 has a water and sediment content of less than or equal to .05 volume percent and a kinematic viscosity @ 40°C of greater than or equal to 1.3 but less than or equal to 2.4 when tested in accordance with ASTM-D975-77, and an impurity level of less than 2 mg. of insolubles per 100 ml. when tested in accordance with ASTM-D2274-70;~~

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;

To be submitted Later

c. ~~At least once per 92 days and from new fuel prior to addition to the storage tanks, by obtaining a sample of fuel oil in accordance with ASTM-D270-1975, and by verifying that the sample meets the following minimum requirements and is tested within the specified time limits:~~

   1) ~~As soon as sample is taken (or prior to adding new fuel to the storage tank) verify in accordance with the tests specified in ASTM-D975-77 that the sample has:~~

      a) ~~A water and sediment content of less than or equal to 0.05 volume percent,~~

      b) ~~A kinematic viscosity @ 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, and~~

      c) ~~A specific gravity as specified by the manufacturer @ 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity @ 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees.~~

   2) ~~Within 1 week after obtaining the sample, verify an impurity level of less than 2 mg of insolubles per 100 ml when tested in accordance with ASTM-D2274-70; and~~

   3) ~~Within 2 weeks of obtaining the sample verify that the other properties specified in Table 1 of ASTM-D975-77 and Regulatory Guide 1.137 Position 2.a. are met when tested in accordance with ASTM-D975-77.~~

Specification: 4.8.1.1.2.a.5)

Justification:

SNUPPS does not need item c. as written in the NRC version since all this test does is test diesel start. The diesel starts on 1 of 3 signals; manual, SIS, UV on 2/4 on NB01 or NB02. If SNUPPS were testing the sequencer, item c. would be important because it must be shown that the LOCA sequencer dominates the S/D sequencer -- but that is not being tested here.

Technical Specification: 4.8.1.1.2.a.6

Justification:

Delete reference to 60 second or less loading time. To achieve the continuous rated load of 6201 KW requires manual loading of the diesel. To require loading in less than 60 seconds results in unnecessary transients on the system and the diesel generator. The ability of the diesel to pick up loads under transient conditions is proven during the refueling outage surveillance requirements. Good practices dictate controlled loading of large generating units where practicable. Manual loading of the diesel will be accomplished at a maximum practical rate.

Technical Specification: 4.8.1.1.2.c TO BE SUBMITTED LATER

2) Verifying the diesel generator capability to reject a load of greater than or equal to 1352 kW (ESW Pump) while maintaining voltage at 4160 $\pm$ 420 volts and frequency at 60 $\pm$ 6.5 Hz;

3) Verifying the diesel generator capability to reject a load of 6201 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection;

4) Simulating a loss of offsite power by itself, and:

   a) Verifying de-energization of the emergency busses and load shedding from the emergency busses, and

   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 shutdown sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 $\pm$ 420 volts and 60 $\pm$ 6.5 Hz during this test.

~~5) Verifying that on a Safety Injection test signal without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 $\pm$ 420 volts and 60 $\pm$ 6.5 Hz within 10 seconds after the auto-start signal; the generator steady-state generator voltage and frequency shall be maintained within these limits during this test;~~

~~6) Verifying that on a simulated loss of the diesel generator, with offsite power not available, the loads are shed from the emergency busses and that subsequent loading of the diesel generator is in accordance with design requirements;~~

7) Simulating a loss of offsite power in conjunction with a Safety Injection test signal, and

   a) Verifying de-energization of the emergency busses and load shedding from the emergency busses;

Technical Specification 4.8.1.1.2.d.5

Justification:

Omit surveillance requirement which is redundant to existing Tech Spec surveillance requirements. Surveillance 4.3.2.2 Response Time Testing assures that upon an initiating signal from SI, the start signal is received to initiate a diesel start. Monthly surveillance 4.8.1.1.2.5 provides for verifying diesel starts and operation with voltage and frequency being maintained within required limits.

B
REV. 1

Specification: 4.8.1.1.2.d.6 Page 3/4 8-4

Justification:

This spec is being deleted because it does not reflect actual system design. As written, this spec assumes that for any simulated loss of the diesel generator, it can be regained and subsequently reloaded through operation of the load shedder/ sequencer. In the SNUPPS design, there are trips that will trip the diesel and lock it out, i.e. the diesel generator cannot be restarted automatically and subsequently reloaded. Two examples of such trips are 1) differential and 2) overspeed trips.

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 LOCA sequencer and operators for greater than or equal to 5 minutes while its generator is loaded with emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 6.5 Hz during this test; and

c) Verifying that all automatic diesel generator trips, except ~~engine overspeed and generator differential~~, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal. — See attached

8) 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 greater than or equal to 6821 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 6201 kW. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 6.5 Hz within 10 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, perform Specification 4.8.1.1.2d.7)b);

9) Verifying that the auto-connected loads to each diesel generator do not exceed 6635 kW;

10) 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) Be restored to its standby status.

11) Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizing the emergency loads with offsite power;

Technical Specification: 4.8.1.1.2.d.7.b

Justification:

Omit requirement. Logic to prove load shedding and subsequent start signal to diesel may be proven without starting diesel generator. Diesel generator loading requirements have been met by 4.8.1.2.d.4.b.

(E)
rev. 1

Insert to 4.8.1.1.2.d.7.c

Verifying that all automatic diesel generator trips, except the following, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a safety injection signal:

1. start failure relay
2. engine overspeed
3. high jacket coolant temperature
4. low lube oil pressure
5. high crankcase pressure
6. generator differential

B
REV.1

Specification: 4.8.1.1.2.d.7.c Page 3/4 8-5

Justification:

The description of this spec. is being revised to more accurately reflect system design. Section 8.3.1.1.g of the Callaway Safety Evaluation Report, pages 8-8 and 8-9, describe the Power Systems Branch (PSB) review of the SNUPPS design. Subsequent discussions and meetings with SNUPPS & PSB have found the SNUPPS design to be consistent with the applicable criteria and acceptable. Engine overspeed and generator differential diesel trips are not required to be bypassed on loss of voltage on the emergency bus concurrent with a safety injection signal because they are provided to prevent massive damage to the engines. PSB determined that for the start failure trip, if the diesel does not start within the predetermined time period, it is indicative of some fundamental malfunction of the diesel generator. No matter how many start attempts are made, the diesel will most likely not start. Thus, it is preferable and acceptable to conserve air versus bypassing the trip on an accident signal and running the diesel start system until the air is depleted. Therefore, the start failure trip need not be bypassed for an accident signal. High jacket coolant temperature, low lube oil pressure and high crankcase pressure diesel trips are provided in two-out-of-three logic schemes and, therefore, do not require bypassing on an accident signal. All other automatic diesel trips are bypassed for an accident signal and only cause a trip during tests when the diesel generator is operating in parallel with the preferred power system.

B
REV. 1

Specification: 4.8.1.1.2.d.11 Page 3/4 8-5

Justification:

This spec. is being deleted because it does not reflect actual system design. On SNUPPS, if the diesel generator is operating in parallel with offsite power for testing and an SIS occurs, the required emergency loads are sequenced onto the bus. However, the diesel generator is not disconnected from the bus by the SIS signal. Both the diesel generator and offsite power remain connected to the bus. (However, in a sense, the diesel generator is placed in its emergency mode as the SIS signal will bypass the appropriate protective trip signals and return the diesel generator to the isochronous mode).

12) Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross-connection lines;

13) Verifying that the automatic LOCA and shutdown sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval;

14) Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

a) Master Transfer switch in LOCAL/MAN position, and

b) Barring Device engaged.

15) Verifying that with all diesel generator air start receivers pressurized to less than or equal to ( ) psig and the compressors secured, the diesel generator starts at least ( ) times from ambient conditions and accelerates to at least (900) rpm in less than or equal to (10) seconds.

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 at least 471 rpm in less than or equal to 10 seconds; and

f. At least once per 10 years by:

1) Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution, and

2) Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code at a test pressure equal to 110% of the system design pressure.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.1. ~~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.~~

See Insert B for remainder

Technical Specification: 4.8.1.1.2.d.14 a, b

Justification:

Delete surveillance requirement. The intent of both the barring device and the master transfer switch are for personnel safety. These features are normally utilized for maintenance activities at the diesel or generator.

The master transfer switch is provided in the diesel room for automatic and local-manual control selection. The switch is normally in the automatic position, whereby the engineered safety features system senses an accident or loss of preferred power and starts the diesel. The master transfer switch is placed in the local-manual position to allow manual operation of the diesel locally when it is out for maintenance. Equipment is provided locally at each diesel generator for manual starting in case of a control room evacuation. The local emergency start functions to start the diesel generator, regardless of the position of the master transfer switch.

Both the barring device and the master transfer switch are indicated locally and in the control room. Actual operability of these devices are proven by post maintenance testing to prove operability and surveillance 4.8.1.1.2.a.5 and 4.3.2.2.

B - NEW 4-21-83

Specification 4.8.1.1.2.d.15

Justification:

The deletion of this newly proposed surveillance requirement is requested because it is not necessary to periodically confirm the capacity of the air start receivers. The capacity of the receivers was verified by factory testing which was required in the purchase specification, refer to FSAR Section 9.5.6.3.

The frequent testing of the diesel generator in accordance with Table 4.8-1 will ensure that the air start system is not degraded or inoperable. The capacity of the air receivers will not change with time since they are fixed volume pressure vessels, refer to FSAR Table 9.5.6.1. The air receivers are normally pressurized to the required pressure by a non-safety-related compressor system. Proper operation of the system only requires that the isolation valves open to admit air to the cylinder banks. This function is verified during every start attempt.

In summary, the number of successful starts (less than 10 seconds) available from the air receivers was determined during factory testing. Since the capacity of the receivers will not change with time this surveillance requirement is considered to be meaningless and unnecessary.

K

Insert B

4.8.1.1.3 REPORTS

A VALID FAILURE IS ANY OF THE FOLLOWING:

a. All start attempts (automatic, including those from bona fide signals, or manual) that result in a failure to start. This does not include unsuccessful start and load attempts that can be definitely attributed to operating error, to spurious operation of a trip that is bypassed in the emergency operating mode, or to equipment malfunction that is not operative in the emergency operating mode (e.g. synchronizing ciruitry) or is not part of the defined diesel generator unit design.

b. A successful start followed by an unsuccessful loading attempt except as attributable to the causes in 4.8.1.1.3 a. above.

c. Tests which are terminated intentionally before completion because of an alarmed abnormal condition that would ultimately have resulted in diesel generator damage or failure.

d. Cranking and venting procedures that lead to the discovery of conditions (e.g. excessive water or oil in a cylinder) that would have resulted in the failure of the diesel generator unit during test or during response to a bona fide signal.

Specification: 4.8.1.1.3

Justification:

The referenced Regulatory Guide was excerpted to make the specification clearer. An operator needs to know what constitutes valid test, valid failure, etc. Although this adds more words to the specification, it makes 4.8.1.1.3 more meaningful.

Table 4.8-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN LAST 20 VALID TESTS*	TEST FREQUENCY
$\leq$ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
$\geq$ 4	At least once per 3 days

* Criteria for determining a valid failure shall be as given in 4.8.1.1.3. A valid test is any of the following:

a. Any of the items in 4.8.1.1.3 which results in a valid failure
b. Successful starts, including those initiated by bona fide signals, followed by successful loading (sequential or manual) to at least 3101 kw and continued operation for at least one hour.
c. Tests performed to verify correction of a problem.

---

*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, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. For the purposes of this test schedule, only valid tests conducted after the OL issuance date shall be included in the computation of the "last 100 valid tests." Entry into this test schedule shall be made at the 31 day test frequency.

W

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) A day tank containing a minimum volume of 390 gallons of fuel,

2) A fuel storage system containing a minimum volume of 85,300 gallons of fuel, [redacted]

3) A fuel transfer pump~~;~~

~~4) Lubricating oil storage containing a minimum total volume of ( ) gallons of lubricating oil, and~~

~~5) Capability to transfer lubricating oil from storage to the diesel generator unit.~~

APPLICABILITY: MODES ~~5~~ 6 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes, ~~movement of irradiated fuel, or crane operation with loads over the spent fuel pool, and within 8 hours, depressurize and vent the Reactor Coolant System through at least a 2 square inch vent. 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.~~ UNTIL THE MINIMUM REQUIRED A.C SOURCES ARE RESTORED TO OPERABLE STATUS ~~RCS vent or vents which are capable of relieving a combined total of at least 460 gpm of primary coolant at 560 psig RCS pressure.~~

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 requirements of Specifications 4.8.1.1.1, 4.8.1.1.2 (except for 4.8.1.1.2a.~~5~~6), and 4.8.1.1.3.

Technical Specification:

3.8.1.2.b.4,5 Page 3/4 8-8

Justification - (reference Attachment 1)

Eliminate reference to lube oil storage and transfer system in LCO for diesel operability. Lube oil useage and contained volume of oil in the crankcase provide sufficient volumes to allow operation of system for greater than 7 days. The system does represent a convenient method to add oil and maintain a level in the crankcase. It should not be used to determine diesel operability.

Assuming the diesel is in a stand-by mode and is at the add oil mark on the dip stick, a contained volume of 948 gallons of lube oil remains in the sump. Since the diesel can be ran and will not fail at 300 gallons of lube oil in the crankcase, an allowable consumed volume of 648 gallons of lube oil exists. This 648 gallons represents a 10 day reserve supply since the diesel uses 449.67 gallons of lube oil in a 7 day period. If credit were taken for the lube oil control system, between 1063 and 1143 gallons of oil would exist. At the lower end of the control level, an additional volume of 115 gallons would exist. This would result in a reserve supply of 11.8 days of operation.

Prior to controlled start-ups of the diesel and post surveillance on shutdown includes verification of crankcase levels as well as other critical temperature and equipment status data.

ATTACHMENT 1

Crankcase Level*		Crankcase Volume
-22.2"	Hi alarm	1215 gallons
-22.575"	Dip stick-full mark	1200 gallons
-24.0"	Close make up valve	1143 gallons
-26.0"	Open make up valve	1063 gallons
-28.5"	Low alarm	963 gallons
-28.875"	Dip stick-add oil mark	948 gallons
	Empty	300 gallons

Crankcase in area of dip stick and level control represents .40 gallons/inch.

* 0" reference at crank shaft centerline.

Technical Specifications - 3.8.1.2, 3.8.2.2

Justification

Technical Specifications governing Crane Travel, Residual Heat Removal, over pressure protection, vessel and storage pool water levels and ventilation, exist and dictate limiting conditions for operation and action statement requirements for same. These controls coupled with the minimum A.C. and D.C. power sources and distribution systems ensure the facility can be maintained in a shutdown or refueling condition for extended periods of time. It also assures sufficient instrumentation and control capability is available to monitor and maintain unit status.

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following DC electrical sources ~~one 125-volt battery bank and its associated full capacity charger~~ shall be OPERABLE.
(Insert A)

APPLICABILITY: MODES 5 and 6.

ACTION:

a. ~~With the 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 battery bank to OPERABLE status as soon as possible, and within 8 hours, depressurize and vent the Reactor Coolant System through at least a 2 square inch vent.~~

b. ~~With the required full capacity charger inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Specification 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.~~

With less than the above compliment of DC sources OPERABLE and energized, establish containment integrity within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt battery banks and associated chargers shall be demonstrated OPERABLE per Specification 4.8.2.1.

(INSERT A)

a. 125-Volt Battery Bank NK11 and NK13, and its associated full capacity chargers NK21 and NK23 or,

b. 125-Volt Battery Bank NK12 and NK14, and its associated full capacity chargers NK22 and NK24.

Technical Specifications - 3.8.1.2, 3.8.2.2,

Justification

Technical Specifications governing Crane Travel, Residual Heat Removal, over pressure protection, vessel and storage pool water levels and ventilation, exist and dictate limiting conditions for operation and action statement requirements for same. These controls coupled with the minimum A.C. and D.C. power sources and distribution systems ensure the facility can be maintained in a shutdown or refueling condition for extended periods of time. It also assures sufficient instrumentation and control capability is available to monitor and maintain unit status.

3/4.8.3 ONSITE POWER DISTRIBUTION

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. Division #1 A.C. Emergency Busses consisting of:

1) 4160-Volt Emergency Bus #NB01, and
2) 480-Volt Emergency Busses #NG01, NG03 and NG05E

b. Division #2 A.C. Emergency Busses consisting of:

1) 4160-Volt Emergency Bus #NB02, and
2) 480-Volt Emergency Busses #NG02, NG04 and NG06E

c. 120-Volt A.C. Vital Busses #NN01 and NN03 energized from their associated inverter connected to D.C. Busses #NN11 and NN13*,

d. 120-Volt A.C. Vital Busses #NN02 and NN04 energized from their associated inverter connected to D.C. Busses #NN12 and NN14*,

e. 125-Volt D.C. Busses NK01 and NK03 energized from Batteries NK11 and NK13, and

f. 125-Volt D.C. Busses NK02 and NK04 energized from Batteries NK12 and NK14.

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

ACTION:

a. With one of the required divisions of A.C. emergency busses not fully energized, re-energize 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) re-energize 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) re-energize 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.~~ SEE ATTACHED

c. With one D.C. bus not energized from its associated battery bank, re-energize 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.

*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.

Attachment

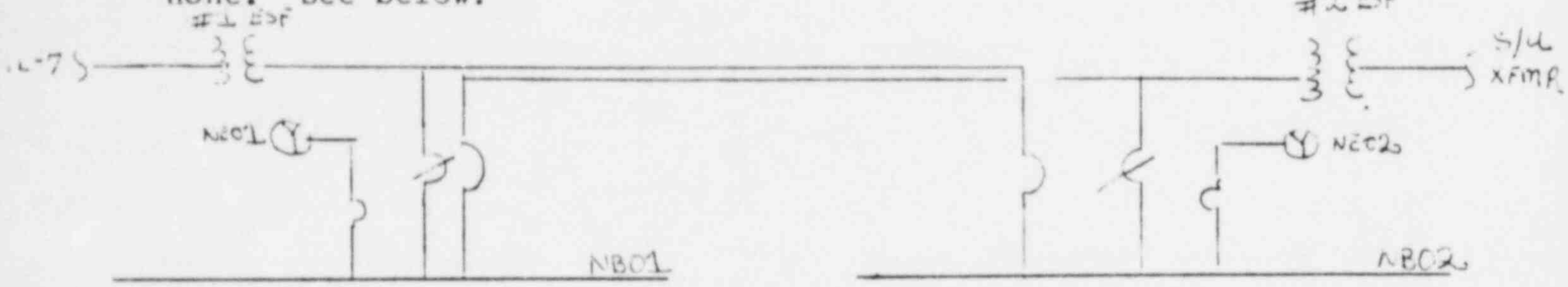
ACTION:

b. (2) With transformers, supplied from a single ES Group, powering one or two vital busses, operation may continue and may be initiated provided the remaining vital busses are powered from inverters.

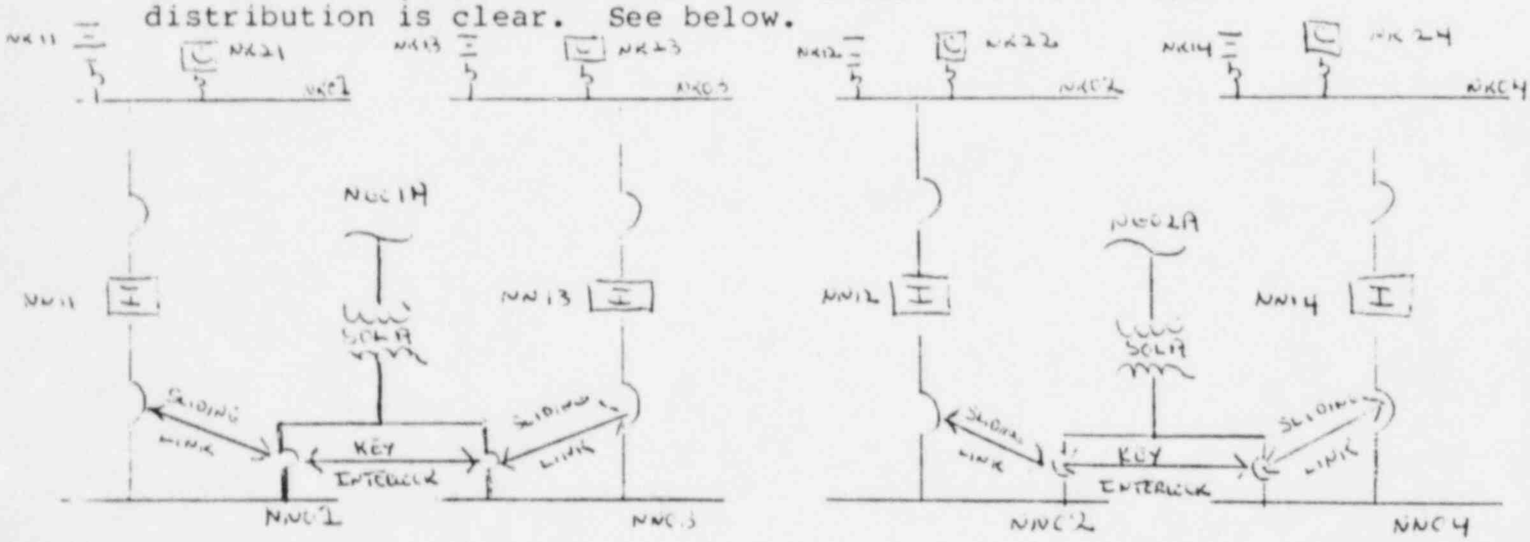
Specification: 3.8.3.1

Justification:

SNUPPS has no problem with the layout of the LCO, but a reference to tie breakers is not necessary because the SNUPPS design has none. See below:



Also, SNUPPS will accept 2 inverters in the note vice the one submitted, but SNUPPS wants to ensure our electrical distribution is clear. See below.

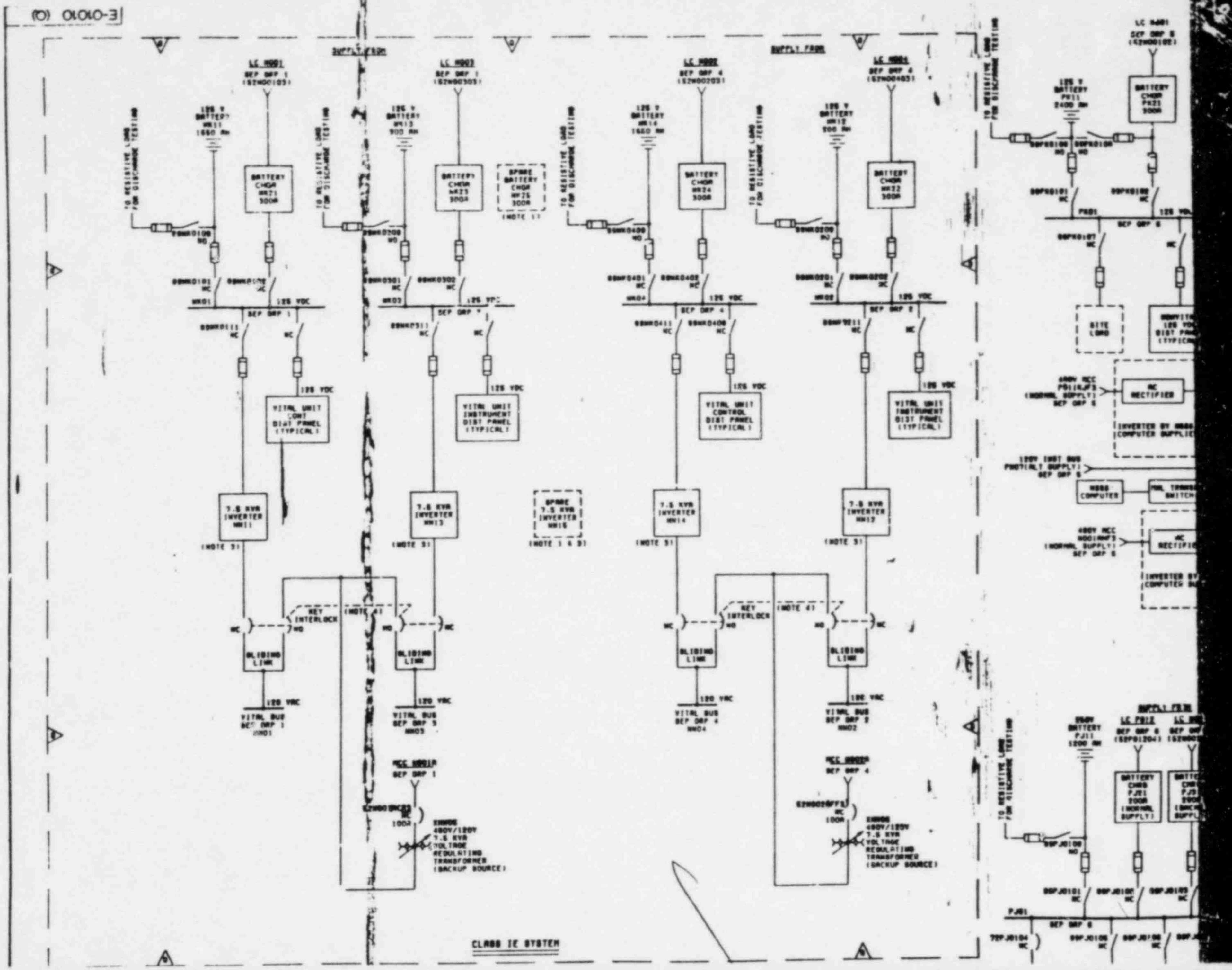


Thus, SNUPPS can have, at the most in this system, 2 inverters out of service and still have the respective NN busses energized.

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.

4.8.3.2 THE TRANSFORMERS CAPABLE OF POWERING VITAL BUSSES SHALL BE DEMONSTRATED OPERABLE AT LEAST ONCE PER 18 MONTHS & WITHIN 24 HOURS AFTER ENTERING ACTION b(2) ABOVE.



SUPPLY FROM
LC H001
SEP GRP 1
LC H003
LC H002
SEP GRP 4
LC H004
SEP GRP 2
125 V BATTERY
BATTERY CHGR
300A
SPARE BATTERY CHGR
(NOTE 1)
TO RESISTIVE LOAD FOR DISCHARGE TESTING
125 VDC
VITAL UNIT CONT DIST PANEL (TYPICAL)
VITAL UNIT INSTRUMENT DIST PANEL (TYPICAL)
VITAL UNIT CONTROL DIST PANEL (TYPICAL)
7.5 KVA INVERTER
(NOTE 3)
SPARE 7.5 KVA INVERTER
(NOTE 1 & 3)
KEY INTERLOCK
(NOTE 4)
SLIDING LINK
120 VAC
VITAL BUS
MCC H001A
MCC H002A
480V/120V 7.5 KVA VOLTAGE REGULATING TRANSFORMER (BACKUP SOURCE)
CLASS IE SYSTEM
SITE LOAD
AC RECTIFIER
INVERTER BY H880 COMPUTER SUPPLIER
H880 COMPUTER

Technical Specification: 3.8.3.1 Action b.2

Justification:

The change to the action statement is proposed to allow use of a backup power supply which is available, reliable and compatible with the system it would serve as backup. The independence and redundance is still maintained with respect to the other load group. The ability to utilize the A.C. transformer is commensurate with FSAR section 8.3.1.1.5, Reg. Guide 1.32, and IEEE Standard 308-1974. This revision allows time to repair the inoperable inverter and allows the plant to select the optimum time to place the inverter back in service. This selection would give consideration to the impact of the dead bus transfer.

Technical Specification: 4.8.3.2 Addition

Justification:

This is to prove the operability of the A.C. transformers in being available and reliable in acting as a backup to the vital bus due to the 3.8.3.1.b (2) change.

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One division of A.C. emergency busses consisting of one 4160-volt and one 480-volt A.C. emergency bus,

b. Two 120-volt A.C. vital busses energized from their associated inverters connected to their respective D.C. busses, and

c. One 125-volt D.C. bus energized from its associated battery bank.

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, and within 8 hours depressurize and vent the RCS through at least a 2 square inch vent.~~

SEE ATTACHED

SURVEILLANCE REQUIREMENTS

4.8.3.2

a. ~~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.

b. THE TRANSFORMER PROVIDING POWER TO THE VITAL BUS SHALL BE DEMONSTRATED OPERABLE WITHIN 24 HOURS AFTER ENTERING ACTION b. ABOVE.

Attachment to Specification: 3.8.3.2

ACTION:

(a) With less than the above complement of electrical busses operable and energized, establish containment integrity within 8 hours.

(b) With a transformer, supplied from a single ES Group, powering one vital bus, operation may continue provided the remaining vital bus is powered from an inverter.

Technical Specification: 3.8.3.2 Action Statement

Justification:

(a) Overpressure protection is provided and covered in Tech. Spec. 3.4.9.3 and establishment of containment integrity should allow for the ability to perform core alterations or movement of irradiated fuel.

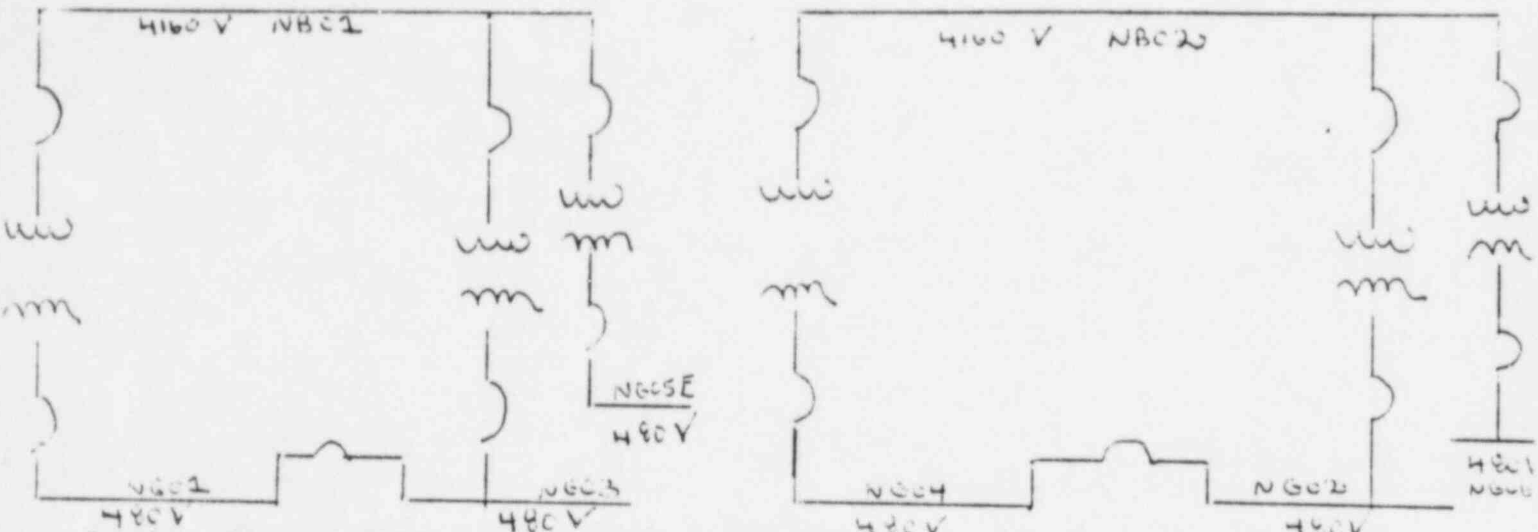
(b) The change to the action statement is proposed to allow use of a backup power supply which is available, reliable and compatible with the system it would serve as backup. The independence and redundance is still maintained with respect to the other load group. The ability to utilize the A.C. transformer is commensurate with FSAR section 8.3.1.1.5, Reg. Guide 1.32, and IEEE Standard 308-1974. This revision allows time to repair the inoperable inverter and allows the plant to select the optimum time to place the inverter back in service. This selection would give consideration to the impact of the dead bus transfer.

The addition is to provide surveillance to prove the operability of the A.C. transformers in being available and reliable in acting as a backup to the vital bus due to the 3.8.3.2 actions statement change.

Specification: 3.8.3.2

Clarification:

SNUPPS has 2 480 volt busses off each 4160 volt bus plus a 480 volt MCC off each 4160 volt bus. See below.



Again, SNUPPS will accept only 1 480 volt bus out of 2 operable on any one side, but SNUPPS wants to make sure our electrical distribution is clear.

K

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection and bypass devices, integral with the motor starter of each valve listed in Table 3.8-2, shall be OPERABLE.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With one or more of the thermal overload protection and/or bypass devices inoperable, declare the affected valve(s) inoperable and apply the appropriate ACTION statement(s) for the affected valve(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2 The above required thermal overload protection and bypass devices shall be demonstrated OPERABLE:

a. At least once per 18 months, by the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST of the bypass circuitry for those thermal overload devices which are either:

   1) Continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, or

   2) Normally in force during plant operation and bypassed under accident conditions.

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of:

   1) All thermal overload devices which are not bypassed, such that each non-bypassed device is calibrated at least once per 6 years, and

   2) All thermal overload devices which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, and thermal overload devices normally in force and bypassed under accident conditions such that each thermal overload is calibrated and each valve is cycled through at least one complete cycle of full travel with the motor-operator when the thermal overload is OPERABLE and not bypassed, at least once per 6 years.

K

TABLE 3.8-2

MOTOR OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)

Specification: 3.8.4.2, 4.8.4.2

Justification:

SNUPPS desires to delete these specifications because the SNUPPS thermal overload and bypass devices will be "jumpered" out prior to fuel load per the FSAR. The specification would have no substance for SNUPPS because the surveillances could not be met. (e.g., SNUPPS could not perform 4.8.4.2.a.1) because this calls for a TRIP ACTUATING DEVICE OPERATIONAL TEST of the bypass circuitry for those thermal overload devices continuously bypassed <u>and</u> temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing. SNUPPS will not temporarily place in force these devices for testing.)

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

(installed or portable) operating

3.9.2 As a minimum, two Source Range Neutron Flux Monitors shall be ~~OPERABLE~~ each with continuous visual indication in the control room and one with audible indication in the containment and control room. X

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 Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of: X

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

Specification: 3.9.2

Justification:

The SNUPPS wording is designed to allow the flexibility of having a source range detector which is not part of the "as built" plant so that fuel could still be moved if one of the installed detectors was undergoing maintenance or repair.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor vessel flange is less tnan 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective ACTION to return the required RHR loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR 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 RHR 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 RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm at least once per 12 hours.

~~*Prior to initial criticality, the RHR 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 vessel hot legs.~~

* Prior to initial criticality, the requirement for OPERABLE RHR loops is reduced to one and the operating pump may be removed from service for any 1 hour in a two hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

Technical Specification: 3.9.8.2

Justification:

Before initial criticality, RHR flow is only required to prevent boron stratification in the Reactor Coolant System. Based on the reduced water volume and the high flow rate of the RHR pump, 1 hour out of 2 hours is adequate to assure no stratification.

K

REFUELING OPERATIONS

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the reactor vessel.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.10 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 or control rods.

See following two pages

Specification: 3.9.10

Justification:

SNUPPS desired this Specification be divided into two separate Specifications as was done at Farley-Unit 2. (See attached Specifications). This will enable adjustment of water level to facilitate easier uncoupling of control rod drive shafts.

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 containment when the reactor pressure vessel contains irradiated fuel assemblies.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies within the pressure vessel. The provisions of Specification 3.0.3 are not applicable.

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 irradiated fuel assemblies within containment.

K

REFUELING OPERATIONS

WATER LEVEL - REACTOR VESSEL

CONTROL RODS

LIMITING CONDITION FOR OPERATION

3.9.10.2 At least 23 feet of water shall be maintained over the top of the irradiated fuel assemblies within the reactor pressure vessel.

APPLICABILITY: During movement of control rods within the reactor pressure vessel while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of control rods within the pressure vessel. The provisions of Specification 3.0.3 are not applicable.

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 control rods within the reactor pressure vessel.

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 fuel storage pool.

ACTION:

a. With the requirements of the above 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.

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

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the fuel 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.

Specification 3.9.11 Page 3/4 9-13

Justification -

The way the specification is written, startup or mode ascension could not continue once the action statement is entered. The spent fuel pool Action Statement adequately controls concerns for dropping or damaging a spent fuel assembly. This action should not preclude plant startup or power operations as no new safety concern associated with power operation is generated. There is, therefore no basis to prevent a plant restart based on fuel pool level.

REFUELING OPERATIONS

3/4.9.12 SPENT FUEL ASSEMBLY STORAGE

LIMITING CONDITION FOR OPERATION

3.9.12 Spent fuel assemblies stored in Region 2 shall be subject to the following conditions:

a. The combination of initial enrichment and cumulative exposure shall be within the acceptable domain of Figure 3.9-1, and

b. No spent fuel assemblies shall be placed in Region 2, nor shall any storage location be changed in designation from being in Region 1 to being in Region 2, while refueling operations are in progress.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all other movement of fuel assemblies and crane operations with loads in the fuel storage areas and move the non-complying fuel assemblies to Region 1. Until these requirements of the above specification are satisfied, boron concentration of the spent fuel pool shall be verified to be greater than or equal to 2000 ppm at least once per 8 hours.

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

SURVEILLANCE REQUIREMENTS

4.9.12 The burnup of each spent fuel assembly stored in Region 2 shall be ascertained by careful analysis of its burnup history, prior to storage in Region 2. A complete record of such analysis shall be kept for the time period that the spent fuel assembly remains in Region 2 of the spent fuel pool.

Specification 3.9.12 Page 3/4 9-14

Action -

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

Justification -

The way the specification is written, startup or mode ascension could not continue once the action statement is entered. The spent fuel pool action statement adequately controls concerns for dropping or damaging a spent fuel pool assembly. This action should not preclude plant start-up or power operations as no new safety concern associated with power operations is generated. There is, therefore no basis to prevent a plant restart based on fuel pool level.

REFUELING OPERATIONS

3/4.9.13 EMERGENCY EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.~~12~~ 13 Two independent Emergency Exhaust Systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the fuel storage pool.

ACTION:

a. With one Emergency Exhaust System inoperable, fuel movement within the fuel storage areas or crane operation with loads over the fuel storage areas may proceed provided the OPERABLE Emergency Exhaust System ~~is capable of being powered from an OPERABLE emergency power source and~~ is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no Emergency Exhaust System OPERABLE, suspend all operations involving movement of fuel within the fuel storage areas or crane operation with loads over the fuel storage areas until at least one Emergency Exhaust System is restored to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.9.~~12~~ 13 The above required Emergency Exhaust 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 10 hours with the heaters operating;~~

~~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:~~

a. As required in Section 4.7.7

2) ~~3)~~ Verifying that the system maintains the fuel building at a negative pressure of greater than or equal to 1/4 inch Water Gauge relative to the outside atmosphere during system operation; and

~~4) Verifying that the heaters dissipate 37 ± 3 kW when tested in accordance with ANSI N510-1975.~~

X

c. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter banks satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% ~~remove greater than or equal to 99% of the DOP when they are tested in-place~~ in accordance with ANSI N510-1975 while operating the system at a flow rate of 9000 cfm ± 10%; and for a DOP test aerosol

X After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal adsorbers satisfies the in-place penetration and bypass leakage testing criteria of less than 1% ~~remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place~~ in accordance with ANSI N510-1975 while operating the system at a flow rate of 9000 cfm ± 10%.

for a halogenated hydrocarbon refrigerant test gas

Technical Specification 3.9.13 Page 3/4 9-16

Justification:

The design of the plant has the normal power supply for the Emergency Exhaust Systems as a Safety Related bus. Therefore, the normal supply to each system meets the criteria of an Emergency Power source and the reference in the ACTION statement has no meaning. Continued operation of the plant if the minimum requirement of the LCO or Action statement cannot be met poses no threat to health and safety of the public, therefore the provisions of specification 3.0.3 should be excepted.

Technical Specification 4.9.12 Page 3/4 9-16, 3/4 9-17, 3/4 9-18

Justification:

The Surveillance Requirements for the Emergency Exhaust System that are listed in Sections 4.7.7 and 4.9.13 are nearly identical. There is only one Emergency Exhaust System; though the surveillance requirements as written would imply that there are two. By combining or referencing one section with the other, the surveillance requirements can be met while still providing the flexibility for additional surveillance required by the applicable plant conditions. Duplication of surveillance testing would also be eliminated.

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~~ 3.10.3 may be suspended for measurement of control rod worth and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any ~~full-length~~ control rod 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 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification ~~3.1.1.1~~ 3.10.3 is restored.

b. With all ~~full-length~~ control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification ~~3.1.1.1~~ 3.10.3 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each ~~full-length~~ rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each ~~full-length~~ rod 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.~~ 3.10.3

Specification: 3/4.10.1

Justification:

This specification relates to the shutdown margin/rod insertion limits and is modified for the same reason as in definition 1.28.

These specifications have been revised due to elimination of Modes 1 and 2 critical from the shutdown margin specification. These changes are necessary to ensure shutdown margin requirements are satisfied when excepting rod insertion limit specifications for physics testing.

Related changes occur in specifications: 1.28, 3/4.1.1.1, 3/4.1.1.2, 3/4.1.3.1, 3/4.1.3.6, 3/4.10.2, 3/4.10.3.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

W

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

c. A. SHUTDOWN MARGIN of at least 1.3% delta K/K is maintained.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The ~~surveillance~~ Requirements of the below listed specfications shall be performed at least once per 12 hours during PHYSICS TESTS:

X

a. Specifications 4.2.2.2 and 4.2.2.3, and

b. Specification 4.2.3.2.

c. SHUTDOWN MARGIN will be verified to be greater than or equal to 1.3% delta K/K.

W

Specification: 3/4.10.2

Justification:

This specification relates to shutdown margin/rod insertion limits and is modified for the same reason as in definition 1.28 and specification 3/4.10.1.

Related changes occur in specifications: 1.28, 3/4.1.1.1, 3/4.1.1.2, 3/4.1.3.1, 3/4.1.3.6, 3/4.10.1, 3/4.10.3.

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6, may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Neutron Flux channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 541°F. X

d. A SHUTDOWN MARGIN of at least 1.3% delta K/K is maintained

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes. X

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 PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range Neutron Flux channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F at least once per 30 minutes during PHYSICS TESTS. X

4.10.3.4 SHUTDOWN MARGIN shall be verified to be greater than an equal to 1.3% delta K/K at least once per 12 hours.

Specification: 3/4.10.3

Justification:

This specification relates to the shutdown margin/rod insertion limits and is modified for the same reason as in definition 1.28 and specification 3/4.10.1.

Related changes occur in specifications: 1.28, 3/4.1.1.1, 3/4.1.1.2, 3/4.1.3.1, 3/4.1.3.6, 3/4.10.1, 3/4.10.2.

K

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual ~~full-length~~ (shutdown and control) rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the Position Indication System inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required Rod Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication Systems agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

initial

*This requirement is not applicable during the ~~initial~~ calibration of the Rod Position Indication System provided: (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

Specification: 3.10.5.b

Justification:

The periodic (18 month) surveillance testing to determine each rod position indication operable is best accomplished during rod drop testing. SNUPPS interprets the asterisked (*) note to apply each time the reactor vessel head is installed. Thus, after each refueling SNUPPS would do rod position indication operability checks in conjuction with rod drop testing.

3/4.2.4 QUADRANT POWER TILT RATIO

The Quadrant Power Tilt Ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective ACTION is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. ~~A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted.~~ A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2 hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such ACTION does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

Specification: B 3/4.2.4

Justification:

W has no basis by which to justify this sentence. Experience has shown that a tilt of greater than 2.5% can be tolerated before any F_Q margin is depleted. Therefore, it is not correct to state that the uncertainty in F_Q is depleated if the tilt is greater than 1.025. The revised specification is attached.

3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

Protection

The OPERABILITY of the Reactor ~~Trip~~ System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated 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)~~ and 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. X

Consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation

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 accident analyses. 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.

INSERT → (B) [

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A setpoint is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy. X

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the setpoints have been specified in Table 3.3-4. Operation with setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 3.3-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the trip setpoint and the value used in the analysis for the actuation. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip

INSERT (B) pg B3/4 3-1

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP 10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System" and supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Emergency Safeguards Features instrumentation.

Table: 3.1-1 and 4.3-1

Specification: 3/4.3.1

Justification:

These sections were modified based on the specified surveillance intervals and surveillance and maintenance outage times recommended in WCAP-10271 "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System" and supplements to that report. The following are identified difficulties with the current testing and maintenance practices:

1. A significant manpower expenditure to accomplish and document testing.
2. Distraction of the operator and shift supervisor away from normal plant control.
3. A large percentage of time spent with portions of the RPS partially inoperable.
4. A large percentage of time spent in a partial trip condition.
5. Subjecting the operator to frequent false alarms and indication.

These changes to RPS surveillance requirements are beneficial to plant safety for the following reasons:

1. A reduction in the number of unnecessary plant transients and challanges to the protection systems.

2. A potential increase in equipment reliability with an associated decrease in equipment random failure probabilities which results in a factor of 3 to 5 increase in RPS availability.

3. Performance of testing and maintenance in a bypass condition for shorter total time which contributes to the increase in RPS availability.

4. An improvement in plant availability.

5. A potential decrease in testing and maintenance errors.

6. More effective use of the operating staff with the capability to redirect significant amounts of manpower to non-surveillance matters.

Changes in these sections are consistent with the modifications to the Westinghouse Standard Tech Specs, Rev. 4 provided in Appendix A of WCAP-10271.

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3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above 1.30 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 RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR 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 RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump with one or more RCS cold legs less than or equal to 310°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant 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 ~~either: (1) restricting the water volume in the pressurizer and thereby providing a volume for the reactor coolant to expand into, or (2) by~~ restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than ~~10°F~~ 50°F above each of the RCS cold leg temperatures.

Specification: B3/4.4.1

Justification:

The LCO is modified for the same reason as in specification 3.4.1.3.

The requirement to maintain the pressurizer water volume below a limiting maximum value is not necessary with COMS.

Related changes occur in specifications: 3.4.1.3, 3.4.1.4.1, 3.4.9.3, Figure 3.4-4, 3.8.1.2, 3.8.2.2, 3.8.3.2, B3/4.4.9.

STEAM GENERATORS (Continued)

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 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.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 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. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 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 of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross 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 allowed limit.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance 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 CONTROLLED LEAKAGE limitation restricts operation when the total flow from ~~supplied to~~ the reactor coolant pump seals exceeds ~~40~~ 8 gpm ~~with the modulating~~ per pump ~~valve in the supply line fully open~~ at a nominal RCS pressure of 2235 psig. This limitation ensures ~~that in the event of a LOCA, the Safety Injection flow will not be less than assumed in the accident analyses.~~ adequate performance of the Reactor Coolant Pump Seals.

The total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The (500 gpd) 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.

Technical Specification: Controlled Leakage
1.8 Definition
3.4.6.2.e
4.4.6.2.1.c
Bases 3/4.4.6.2

Justification:

Revise the above referenced sections of Technical Specifications to reflect changes as noted. The bases for the changes is to define controlled leakage as leakage from the seal, which is consistent from a human engineering point of view with the balance of defined RCS leakage terms. This also provides a limit and surveillance which will monitor actual seal performance. Seal injection does not measure performance and seal degradation.

The charging pump discharge supply to the RCP seal water injection is controlled by locked and throttled angle stop check valves BGV-198, 199, 200, 201. Controlled leak off return is limited by a locked and throttled angle stop check valve BGV-202. HCV-182 allows flow to be balanced between the normal charging flow path to the reactor coolant loop 1 cold leg and the reactor coolant pump seals. Controlled leak off from the #1 seal is isolated by building isolation valves BGHV-8100 and 8112 on a Phase A containment isolation signal. Technical Specification 3.5.2 for ECCS subsystems establishes operability for the centrifugal charging pumps and safety injection pumps by verifying proper line ups, pump capacities, and balanced flows to assure minimum flow rates to the reactor coolant system normal injection flow paths.

Since Technical Specification 3.5.2 assures ECCS system operability, the revision of surveillance requirements for specification 4.4.6.2.1 will allow consideration for leakage and correct the assumptions made on the design and bases of the pump seal control system.

Eight gpm controlled leakage per pump was choosen to be consistent with the requirement to isolate the #1 pump seal as recommended by Westinghouse.

This change is submitted in conjunction with the proposed change to ECCS subsystems 4.5.2.g.2.

OPERATIONAL LEAKAGE (Continued)

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.7 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.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the site boundary will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values 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. 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 Callaway site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

SPECIFIC ACTIVITY (Continued)

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the Reactor Coolant's specific activity greater than 1.0 microCuries/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. Operation with specific activity levels exceeding 1.0 microcuries/gram DOSE EQUIVALENT I-131, but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10% of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the SITE BOUNDARY by a factor of up to 20 following a postulated Steam Generator tube rupture. The reporting of cumulative operating time over 500 hours in any 6-month consecutive period with greater than 1.0 microCuries/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800-hour limit.

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than 10 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The ~~selection~~ choice of ~~a half life of less than~~ 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinct window for determination of the radionuclides above and below a half-life of 10 minutes. For these reasons the ~~shorter lived~~ radionuclides excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-absorption characteristics.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes, ~~and allowing some flexibility in the determination of the gross specific activity and~~ $\bar{E}$. The determination of the contributors to the $\bar{E}$ results should be based upon those energy peaks identified with a 95% confidence level. ~~Although the initial determination of both the gross~~ The radiochemical determination of the nuclides should be

The counter should be reset to reproducible efficiency versus energy. It is not necessary to identify specific nuclides.

SPECIFIC ACTIVITY (Continued)

Based on multiple counting of the samples with typical counting times following sampling ~~specific activity and the analyses should be completed within 2 hours after the sample is taken. Additional counting periods should be considered to more clearly define the applicable gross specific activity and $\bar{E}$ values.~~ of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor 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 reactor 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 phenomenon. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon.

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. ~~a~~ These limit lines shall be calculated periodically using methods provided below.

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F. 4. The pressurizer heatup and cooldown ~~rates shall not exceed 100°F/hr~~ and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F. 5. System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI. 6. The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the 1972 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code.

Technical Specification: Bases 3/4 4.8

Justification:

It is requested that this material be deleted due to its lack of relevance to describing the Bases for the Limiting Condition for Operation on Specific Activity i.e., $\overline{E}$. This is a bases for an analytical requirement not an LCO.

The deleted section appears to describe an analytical method at times. This type of description is not found in any other section of the Technical Specifications. The method is also difficult to understand since it at times is referencing a gross specific activity determination which would not be nuclide specific, but at other times is referencing a nuclide specific method when talking about an energy vs. efficiency curve such as is used in isotopic analysis by germanium gamma ray spectroscopy. It should also be noted that the half-life cutoff should be 15 minutes with the possible exception of Xe-138. Reasons for this are given in the justification for change of Item 3 of Table 4.4-4.

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PRESSURE/TEMPERATURE LIMITS (Continued)

7

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of ~~16~~ 7 effective full power years of service life. The ~~16~~ 7 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

X

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. Therefore, a adjusted reference temperature, based upon the fluence and copper content and phosphate content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by either Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of ~~16~~ 7 EFPY ~~(as well as adjustments for possible errors in the pressure and temperature sensing instruments)~~.

X

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict the future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current

Specification: B 3/4.4.9

Justification:

This specification is modified to reflect the service life period of 7EFPY for Callaway. This corresponds to the applicability of the heatup curves specified in the Callaway Tech Specs (Figure 3.4-2).

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

Component	Comp Code	ASME Material Type	CU (%)	P (%)	T_{NDT} (°F)	50 FT-LB 35 Mil Temp (°F)	RT_{NDT} (°F)	Avg. Upper Shelf NMWD (FT-LB)	Avg. Upper Shelf MWD (FT-LB)
Closure Head Dome	R2516-1	A533B, CL.1	0.12	0.010	-40	60	0	112	----
Closure Head Torus	R2515-1	A533B, CL.1	0.11	0.009	-20	<40	-20	119	----
Closure Head Flange	R2504-1	A508, CL.2	----	0.013	20	<80	20	139	----
Vessel Flange	R2501-1	A508, CL.2	----	0.012	20	<80	20	102	----
Inlet Nozzle	R2502-1	A508, CL.2	----	0.010	-20	<40	-20	147	----
Inlet Nozzle	R2502-2	A508, CL.2	----	0.009	-20	<40	-20	137	----
Inlet Nozzle	R2502-3	A508, CL.2	0.11	0.010	-20	<40	-20	156	----
Inlet Nozzle	R2502-4	A508, CL.2	0.11	0.010	-30	<30	-30	156	----
Outlet Nozzle	R2503-1	A508, CL.2	----	0.006	-10	<50	-10	126	----
Outlet Nozzle	R2503-2	A508, CL.2	----	0.009	0	<60	0	129	----
Outlet Nozzle	R2503-3	A508, CL.2	----	0.007	0	<60	0	136	----
Outlet Nozzle	R2503-4	A508, CL.2	----	0.007	0	<60	0	114	----
Nozzle Shell	R2004-1	A533B, CL.1	0.05	0.010	-40	70	10	118	----
Nozzle Shell	R2004-2	A533B, CL.1	0.04	0.011	-40	80	20	121	----
Nozzle Shell	R2004-3	A533B, CL.1	0.04	0.008	-50	60	0	133	----
Inter. Shell	R2005-1	A533B, CL.1	0.04	0.008	-20	<40	-20	127	156
Inter. Shell	R2005-2	A533B, CL.1	0.04	0.007	-30	40	-20	127	143
Inter. Shell	R2005-3	A533B, CL.1	0.05	0.007	-30	40	-20	135	164
Lower Shell	R2508-1	A533B, CL.1	0.09	0.009	-40	60	0	87	118
Lower Shell	R2508-2	A533B, CL.1	0.06	0.008	-10	70	10	100	127
Lower Shell	R2508-3	A533B, CL.1	0.07	0.008	-20	100	40	86	127
Bottom Head Torus	R2517-1	A533B, CL.1	0.11	0.010	-80	30	-30	92	----
Bottom Head Dome	R2518-1	A533B, CL.1	0.12	0.009	-60	0	-60	154	----
Inter. & Lower Shell Long. Weld Seams	G2.06	SAW	0.04	0.006	-50	<10	-50	150	----
Inter. to Lower Shell Girth Weld Seams	E3.16	SAW	0.05	0.007	-50	10	-50	98	----

NMWD - Normal to Major Working Directions

MWD - Major Working Directions

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

Component	Comp Code	ASME Material Type	CU (%)	P (%)	T_{NDT} (°F)	50 FT-LB 35 Mil Temp (°F)	RT_{NDT} (°F)	Avg. Upper Shelf NMWD (FT-LB)	Avg. Upper Shelf MWD (FT-LB)
Closure Head Dome	R2713-1	A533B, CL.1	0.13	0.008	-30	<30	-30	113	----
Closure Head Torus	R2712-1	A533B, CL.1	0.12	0.014	-20	40	-20	117	----
Closure Head Flange	R2704-1	A508, CL.2	----	0.011	30	<90	30	130	----
Vessel Flange	R2701-1	A508, CL.2	----	0.010	40	<100	40	123	----
Inlet Nozzle	R2702-1	A508, CL.2	----	0.013	10	<70	10	138	----
Inlet Nozzle	R2702-2	A508, CL.2	----	0.011	10	<70	10	141	----
Inlet Nozzle	R2702-3	A508, CL.2	----	0.009	-10	<50	-10	139	----
Inlet Nozzle	R2702-4	A508, CL.2	----	0.010	-10	<50	-10	134	----
Outlet Nozzle	R2703-1	A508, CL.2	----	0.010	-10	<50	-10	130	----
Outlet Nozzle	R2703-2	A508, CL.2	----	0.009	10	<70	10	108	----
Outlet Nozzle	R2703-3	A508, CL.2	----	0.004	10	<70	10	126	----
Outlet Nozzle	R2703-4	A508, CL.2	----	0.006	0	<60	0	122	----
Nozzle Shell	R2706-1	A533B, CL.1	0.05	0.010	10	80	20	103	----
Nozzle Shell	R2706-2	A533B, CL.1	0.06	0.009	0	90	30	88	----
Nozzle Shell	R2706-3	A533B, CL.1	0.08	0.011	0	90	30	101	----
Inter. Shell	R2707-1	A533B, CL.1	0.04	0.008	-40	100	40	78	99
Inter. Shell	R2707-2	A533B, CL.1	0.05	0.008	-50	70	10	100	121
Inter. Shell	R2707-3	A533B, CL.1	0.06	0.010	-40	50	-10	99	122
Lower Shell	R2708-1	A533B, CL.1	0.07	0.006	0	110	50	82	95
Lower Shell	R2708-2	A533B, CL.1	0.05	0.007	-30	70	10	105	130
Lower Shell	R2708-3	A533B, CL.1	0.07	0.006	-10	80	20	101	122
Bottom Head Torus	R2714-1	A533B, CL.1	0.15	0.010	-20	40	-20	139	----
Bottom Head Dome	R2715-1	A533B, CL.1	0.17	0.011	-40	20	-40	152	----
Inter. & Lower Shell Long. Weld Seams	G2.03	SAW	0.04	0.008	-60	<0	-60	143	----
Inter. to Lower Shell Girth Weld Seams	E3.14	SAW	0.04	0.006	-60	<0	-60	112	----

NMWD - Normal to Major Working Directions

MWD - Major Working Directions

Table: B 3/4.4-1

Justification:

The results of the reactor vessel materials tests which are used to determine the initial RT_{NDT} are shown in Table B 3/4.4-1. This table should be incorporated into the Tech Specs.

PRESSURE/TEMPERATURE LIMITS (Continued)

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs or an RCS vent ~~opening of at least 2 square~~ OR VENTS WHICH ARE CAPABLE OF RELIEVING A COMBINED TOTAL OF AT LEAST 460 GPM OF PRIMARY COOLANT AT 560 PSIG RCS PRESSURE, inches 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 310°F. Either PORV has adequate 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 50°F above the RCS cold leg temperatures, or (2) the start of a HPSI pump and its injection into a water solid RCS.

3/4.4.10 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 Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 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.

Specification: B3/4.4.9

Justification:

The LCO is modified for the same reason as in specification 3.4.1.3.

Vents are more appropriately specified by a flow rate requirement than a vent area.

Related changes occur in specifications: 3.4.1.3, 3.4.1.4.1, Figure 3.4-4, 3.8.1.2, 3.8.2.2, B3/4.4.1.

ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump and one Safety Injection pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and Safety Injection pumps except the required OPERABLE charging pump to be inoperable below 310°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

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 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 Surveillance Requirements for leakage testing of ECCS check valves ensure that a failure of one valve will not cause an intersystem LOCA.

3/4.5.4 BORON INJECTION SYSTEM

~~The OPERABILITY of the Boron Injection System as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident, or a steam line rupture.~~

The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the Steam Line break analysis are met. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. The limit on boron concentration represents that which is normally contained in the High Pressure Injection System.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) 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 RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reacter will remain subcritical in the cold condition following mixing of the RWST and the RCS water

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 line isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the accident analyses.

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 of 70°F and 200 psig are based on a steam generator RT_{NDT} of 60°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component 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 accident analyses. *Each independent Component Cooling Water loop contains two 100% capacity pumps, therefore the failure of one pump does not affect the OPERABILITY of that loop.*

3/4.7.4 ESSENTIAL SERVICE WATER SYSTEM

The OPERABILITY of the Essential Service 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 accident conditions within acceptable limits.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available either to: (1) provide normal cooldown of the facility, or (2) mitigate the effects of accident conditions within acceptable limits.

U-1

Specification B 3/4.7.3 Page B 3/4 7-3

Justification -

The plant design has two, 100% capacity component cooling water pumps in each redundant loop. The additional wording in the bases will assist the operator in determining the OPERABILITY of the Component Cooling Water System if a single pump is inoperable.

BASES

FIRE SUPPRESSION SYSTEMS (Continued)

The Surveillance Requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying either the weight or the level of the tanks. ~~Level measurements are made by either a U.L., or an F.M. approved method.~~

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

3/4.7.11 FIRE BARRIER PENETRATIONS

The functional integrity of the fire barrier penetrations ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection of and the extinguishing of the fire. The fire barrier penetrations are a passive element in the facility fire protection program and are subject to periodic inspections.

Fire barrier penetrations, including cable penetration barriers, fire doors and dampers are considered functional when the visually observed condition is the same as the as-designed condition. For those fire barrier penetrations that are not in the as-designed condition, an evaluation shall be performed to show that the modification has not degraded the fire rating of the fire barrier penetration.

During periods of time when a barrier is not functional, either: (1) a continuous fire watch is required to be maintained in the vicinity of the affected barrier, or (2) the fire detectors on at least one side of the affected barrier must be verified OPERABLE and an hourly fire watch patrol established, until the barrier is restored to functional status.

3/4.7.12 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of ±3°F.

Bases: 3.7.10 Page B 3/4 7-7

Justification:

SNUPPS intends to use ultrasonics to verify level in the Halon Storage Tanks. An ultrasonic probe will be used to detect the level at ambient temperature. This reading will be corrected by graph to a reference level which corresponds to the required Halon level at 70°F. The method used to calibrate and check the ultrasonic heat equipment utilizes standards traceable to the National Bureau of Standards.

JHR/7a21

W

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 143.7 inches and contain a maximum total weight of 1748 grams uranium. The initial core loading shall have a maximum enrichment of 3.10 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.50 weight percent U-235.

1735

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length and no part length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. All control rods shall be hafnium, clad with stainless steel tubing.

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 to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,135 ± ____ cubic feet at a nominal T_{avg} of 557°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

Specification: 5.3.1

Justification:

Each fuel rod in a standard 17 x 17 fuel assembly contains 1735 grams of uranium. Both the Wolfcreek and Callaway Technical Specifications should be changed.

SECTION 6.0 CHANGES ARE APPLICABLE

TO WOLF CREEK ONLY

The title changes in this section are the result of recent organizational changes at KGE. However, the responsibilities and reporting relationships of these positions have not changed.

W.C. ONLY

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6.1 RESPONSIBILITY

6.1.1 The Plant ~~Superintendent~~ 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 Supervising Operator, under the Shift Supervisor, shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President-Nuclear 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 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 Operator shall be in the control room when fuel is in the reactor. In addition, while the Unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Reactor Operator shall be in the control room;

c. A Health Physics Technician # shall be on site when fuel is in the reactor;

d. ALL CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or licensed Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. A site fire brigade of at least 5 members shall be maintained onsite at all times.# The fire brigade shall not include 3 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; and

#The Health Physics Technician and fire brigade 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.

W.C. ONLY

K

f. Administrative procedures shall be developed and implemented to limit the working hours of ~~Unit Staff~~ shift operating personnel who perform safety-related functions; e.g., Senior Reactor Operators, Reactor Operators, Health Physicists, Auxiliary operators, and key maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have shift operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforseen 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:

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 seven day period, all excluding shift turnover time;

3) A break of at least eight hours should be allowed between work periods, including shift turnover time; and

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.

Any deviation from the above guidelines shall be authorized by the Plant ~~Superintendent~~ Manager or ~~his deputy~~ the Call Superintendent, 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 shall be reviewed monthly by the Plant ~~Superintendent~~ Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

Section 6.2.2
(items b, d and f)

Justification -

The wording reflected in the NRC Draft WCGS Technical Specifications does not reflect the terminology specified in Rev. 4 of NUREG-0452, Standardized Technical Specifications for Westinghouse Pressurized Water Reactors. It is KG&E's desire to maintain standardized wording wherever this can be reasonably achieved. KG&E, therefore, recommends the wording specified in NUREG-0452 (Rev. 4) item 6.2.2.b., 6.2.2.d. and 6.2.2.f. be used as it is less ambiguous and is standardized.

Section 6.2.2.f.

Justification -

The wording proposed by the NRC is acceptable provided some minor modifications are made. This requirement should apply to "shift" operating personnel. Because of the common use of the term "Operations" in application to all of a Nuclear Power Plant Staff, the present wording would be ambiguous resulting in confusion by individuals dealing with Technical Specifications.

In the case of WCGS, the organization does not include a "deputy" to the Plant Manager ~~Superintendent~~. This responsibility would be assigned to the individual referred to as the "Call Superintendent."

W.C. ONLY

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Table 6.2-1

MINIMUM SHIFT CREW COMPOSITION
SINGLE UNIT FACILITY

X

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SS	1	1
SRO	1	None
RO	2	1
SO	4	1
STA	1*	None
CHM	1	None

SS - Shift Supervisor with a senior operator's license on Unit 1 X
SRO - Individual with a Senior Operator license on Unit 1
RO - Individual with an Operator license on Unit 1
SO - ~~Equipment~~ Station Operator
STA - Shift Technical Advisor
CHM - Chemistry Personnel

~~Except for the Shift Supervisor~~, 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 MODES 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) 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 Operator license (other than the Shift Technical Advisor) shall be designated to assume the control room command function.

See Insert A

*The STA position may be eliminated ~~when an SRO on shift meets the NRC educational requirements for an STA and the control room design has been upgraded to NRC requirements for eliminating the STA position~~.

Insert A

Shift Technical Advisor is required anytime the Shift Supervisor on duty does not meet the educational requirements of a STA as defined in the October 31, 1980 letter from Darrell G. Eisenhut, Division of Licensing, to all licensees and applicants.

Table 6.2-1

Justification -

WCGS has recently completed a college training program which meets or exceeds the requirements recommended in Section 6 of the INPO document "Nuclear Power Plant Shift Technical Advisor - Recommendations for Position Description, Qualification, Education and Training." This document is endorsed as guidance for STA training by NUREG-0737. In addition, five of the seven WCGS Shift Supervisor candidates have degrees in engineering fields or physics. Furthermore, the requalification requirements that are set out for STAs will be included in the SRO requalifications program.

Per WCGS SER item 13.1.3.4 "In the event of unexpected absence through sickness or vacation of the SRO who has been upgraded to the educational requirements of the STA, a similarly qualified SRO replacement or a Duty/Call Technical Advisor will be furnished for that shift. The applicant has committed to adding this requirement to the administrative controls section of the proposed Technical Specifications.

A detailed Control Room Design review will be completed prior to licensing. Many of the findings of this review have been corrected or will be prior to licensing. The remaining findings will be corrected by a schedule agreed upon by the NRC Project Manager.

WCGS, therefore, recommends the note should read "*Shift Technical Advisor is required anytime the Shift Supervisor on duty does not meet the educational requirements of an STA as defined in the October 31, 1980 letter from Darrell G. Eisenhut, Division of Licensing, to all licensees and applicants."

At Wolf Creek, the title "Station Operator" is used rather than "Equipment Operator" for the position identified in Rev. 4 of NUREG-0452 as "auxiliary operator." Since the term "equipment operator" is undefined in the WCGS organization structure and procedure and is not a commonly accepted term at WCGS, its use would lead to confusion and misinterpretation by personnel using the Technical Specifications. Our recommendation is to hereafter use the term "Station Operator" in all Technical Specification entries.

The intention of the first paragraph on page 6-5 would appear to ensure immediate action is taken to ensure a minimum crew composition is restored following an unexpected absence of on-duty shift crew members. It is inappropriate to assume that the Shift Supervisor is immune from unforeseen causes of absence. Sudden illnesses, for example, are completely out of any control of KG&E. Therefore, inclusion of the term "except for the Shift Supervisor" would not realistically ensure that the Shift Supervisor is present but would only result in generation of a Technical Specification violation. As such, it is an unrealistic paperwork requirement which will not add to the safety of the public or the facility.

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~~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 2 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 and that human errors are reduced as much as practical.~~

~~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 Manager, Nuclear Safety and Emergency Preparedness and the Manager, Callaway Plant.~~

6.2.4 SHIFT TECHNICAL ADVISOR

The Shift Technical Advisor, when required by Table 6.2-1, shall provide 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.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the Unit Staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978, except for the ~~Radiation Protection Manager~~ Site Health Physicist who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

*Not responsible for sign-off function.

Specification 6.2.3

Justification -

In order to simplify the Technical Specifications, KG&E has limited this section just to the important safety review committees.

Section 6.2.4

Justification -

The wording specified by the NRC is acceptable provided the caveat in regards to the note to Table 6.2-1 is used. For further information, see the change justification for Table 6.2-1. The recommended wording would be "A Shift Technical Advisor, when required by Table 6.2-1, shall provide..."

Section 6.3.1

Justification -

The title of Site Health Physicist is defined in the Radiation Protection Manual. KG&E has identified the Site Health Physicist as the RPM and this position has the responsibility and authority as outlined in Regulatory Guide 3.8, NUREG-0654 and other applicable regulations and guides.

K

W.C. ONLY

6.4 TRAINING

administration 6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the ~~direction~~ of the ~~Superintendent of~~ Training and shall meet or exceed the requirements and recommendations of Section 5 of ANSI/ANS 3.1-1978 and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in Section 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. ~~identified by the ISEG~~. Supervisor

6.5 REVIEW AND AUDIT

Insert

6.5.1 PLANT SAFETY REVIEW COMMITTEE (PSRC)

FUNCTION

6.5.1.1 The Plant Safety Review Committee shall function to advise the Plant Superintendent on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Plant Safety Review Committee shall be composed of the:

Chairman:	Plant Manager ~~Superintendent~~
Member:	~~Operations Supervisor~~ Superintendent of Operations
Member:	~~Technical Support Supervisor~~ Superintendent of Technical Support
Member:	~~Maintenance Supervisor~~ Superintendent of Maintenance
Member:	Instrument and Control Supervisor
Member:	Reactor Engineering Supervisor
Member:	Health Physicist
Member:	Chemist
Member:	Results Engineering Supervisor

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PSRC Chairman to serve on a temporary basis; however no more than two alternates shall participate as voting members in PSRC activities at any one time.

Member: Superintendent of Plant Support
Member: Superintendent of Regulatory, Quality, and Administrative Services

Section 6.4.1

Justification -

In general the NRC proposed wording is similar to that proposed by WCGS. The less restrictive use of the words "direction of" can be used in place of our proposed wording "administration of" although KG&E prefers the latter. In regards to the title, Rev. 4 of NUREG-0452, does not specifically incorporate the standardized title "Superintendent of Training" but allows the applicant to insert the position title. The position title at WCGS is Training Supervisor. The review of relevant industry operational experience at WCGS is not the responsibility of the ISEG. This function falls under the administration of the WCGS Results Engineering Group and is a total plant staff responsibility. KG&E, therefore, recommends "identified by ISEG" be dropped from the last sentence.

Insert

This section discusses the safety review groups for Wolf Creek. Other licensing documents describe other groups within the nuclear organization with their respective review and/or audit responsibilities.

K

Specification 6.5

Justification -

These words were added to clarify the fact that all safety review groups for Wolf Creek are not included in the Technical Specifications.

MEETING FREQUENCY

6.5.1.4 The PSRC shall meet at least once per calendar month and as convened by the PSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the PSRC necessary for the performance of the PSRC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The Plant Safety Review Committee shall be responsible for:

a. Review of: (1) all procedures required by Specification 6.8 and changes thereto, (2) all programs required by Specification 6.8 and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the Plant ~~Superintendent~~ Manager to affect nuclear safety;

b. Review of all proposed changes, tests and experiments which may involve an unreviewed safety question as defined in Section 50.59, 10 CFR;

c. Review of all proposed changes to Technical Specifications or the Operating License;

d. Review of all safety evaluations performed under the provision of Section 50.59(a)(1), 10 CFR, for changes, tests and experiments;

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Director, Nuclear Operations, and to the Nuclear Safety Review Committee (NSRC);

f. Review of events requiring 24-hour written notification to the Commission;

g. Review of reports of operating abnormalities, deviations from expected performance of plant equipment and of unanticipated deficiencies in the design or operation of structures, systems or components that affect nuclear safety;

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman, NSRC;

i. Review of the plant Security Plan ~~and implementing procedures~~ and shall submit recommended changes to the NSRC;

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the NSRC;

Section 6.5.1.6.i.

Justification -

The review of Security Plan implementing procedures by the Plant Safety Review Committee would unnecessarily disseminate the procedures to an unacceptable level of personnel. This could potentially minimize the value of the procedures, depending on the identity of an "insider" involved in a security incident. It is the KG&E opinion that review of the Security Plan and not the implementing procedures constitutes a sufficient involvement by the Plant Safety Review Committee.

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W.C. ONLY

k. Review of changes to the PROCESS CONTROL PROGRAM, and the OFFSITE DOSE CALCULATION MANUAL; and

l. ~~Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective ACTION to prevent recurrence and the forwarding of these reports to the Manager, Callaway Plant and to the Nuclear Safety Review Board.~~

See insert B

AUTHORITY

6.5.1.7 The Plant Safety Review Committee shall:

a. Recommend in writing to the Plant ~~Superintendent~~ Manager approval or disapproval of items considered under Specification 6.5.1.6a. through d. above,

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6a. through e. above constitutes an unreviewed safety question, and

c. Provide written notification within 24 hours to the Director Nuclear Operations and the Nuclear Safety Review Committee of disagreement between the PSRC and the Plant ~~Superintendent~~ Manager; however, the Plant ~~Superintendent~~ Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1 above.

RECORDS

6.5.1.8 The Plant Safety Review Committee shall maintain written minutes of each PSRC meeting that, at a minimum, document the results of all PSRC activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to the Director, Nuclear Operations, and the Nuclear Safety Review Committee.

6.5.2 NUCLEAR SAFETY REVIEW COMMITTEE (NSRC)

FUNCTION

6.5.2.1 The Nuclear Safety Review Committee shall function to provide independent review and 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,

Insert B

Review of every unplanned onsite release of radioactive material to the environs, including the preparation and forwarding of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence to the Plant Superintendent and to the Nuclear Safety Review Committee.

Section 6.5.1.6.1.

Justification -

The words proposed by the NRC are not included in Rev. 4 of NUREG-0452 and appear to have come from the Callaway Technical Specifications. The wording proposed by KG&E are the appropriate words for the WCGS Technical Specifications.

g. Mechanical and electrical engineering, and

h. Quality assurance practices.

COMPOSITION

6.5.2.2 The NSRC shall be composed of the:

Chairman:	Manager Nuclear Services, KG&E
Member: ~~Vice Chairman:~~	Manager Nuclear Plant Engineering, KG&E
Member:	Quality Assurance Coordinator, KG&E
Member	Director Nuclear Operations, KG&E
Member:	Manager Licensing, KG&E
Member:	Vice President-Engineering, KG&E
Member:	Manager Nuclear Safety ~~Engineering~~

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the NSRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in NSRC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NSRC Chairman to provide expert advice to the NSRC.

MEETING FREQUENCY

6.5.2.5 The NSRC shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter.

QUORUM

6.5.2.6 The minimum quorum of the NSRC necessary for the performance of the NSRC review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least four NSRC members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the Unit.

Specification 6.5.2.2

Justification -

There are no standard technical specification requirements for an NSRC Vice-Chairman, therefore, this has been deleted from the specification.

A recent K&GE reorganization changed the title Manager Safety Engineering to Manager Nuclear Safety

K

W.C. ONLY

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months;

k. Any other area of Unit operation considered appropriate by the ~~NSRB~~ NSRC or the Vice President-Nuclear;.

e. The fire protection programmatic controls including the implementing procedures at least once per 24 months, ~~by qualified licensee QA personnel~~;

f. 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;

See Insert A

g. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months;

h. The ODCM and implementing procedures at least once per 24 months;

i. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months; ~~and~~

j. ~~The performance of activities required by the Quality Assurance Program to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974, and Regulatory Guide 4.1, Revision 1, April 1975,~~ 4.0 December/91 at ~~least once per 12 months~~; and

AUTHORITY

6.5.2.9 The NSRC shall report to and advise the Vice President-Nuclear on those areas of responsibility specified in Specifications 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of NSRC activities shall be prepared, approved and distributed as indicated below:

a. ~~Minutes of each NSRC meeting shall be prepared, approved and forwarded to the Vice President-Nuclear within 14 days following each meeting;~~

See Insert B

b. ~~Reports of reviews encompassed by Specification 6.5.2.7 above, shall be prepared, approved and forwarded to the Vice President-Nuclear within 14 days following completion of the review; and~~

c. Audit reports encompassed by Specification 6.5.2.8 above, shall be forwarded to the Vice President-Nuclear and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

Wolf Creek version
K

Insert A

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the NSRC. 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 entire 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 50, at least once per 24 months.

e. The Emergency Plan and implementing procedures at least once per 24 months.

f. The Security Plan and implementing procedures at least once per 24 months.

g. Any other area of unit operation considered appropriate by the NSRC or the Vice President-Nuclear.

h. The Fire Protection Program and implementing procedures at least once per 24 months.

i. An independent fire protection and loss prevention inspection and audit to be performed annually utilizing either qualified offsite licensee personnel or an outside fire protection firm.

j. An inspection and audit of the fire protection and loss prevention program to be performed by an outside qualified fire consultant at intervals no greater than 3 years.

Specification 6.5.2.8

Justifications -

Items 6.5.2.8.e through 6.5.2.8.k are from Callaway's submittal not Wolf Creek's submittal. Please use the attached Wolf Creek submittal.

The words "by qualified licensee, QA personnel" were deleted from item e (NRC version), since at Wolf Creek the audit may be performed by ISEG or QA personnel.

Item j (NRC version) is deleted since it is covered in item k (NRC version).

Item k (NRC version), "NSRB" should be "NSRC" in the Wolf Creek Tech. Specs.

Insert B

a. Minutes of each NSRC meeting shall be prepared, reviewed by participating members and forwarded to the Vice President - Nuclear within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, reviewed by participating members and forwarded to the Vice President - Nuclear within 14 days following completion of the review.

Specification 6.5.2.10

Justification -

The approval mechanism for NSRC records is such that approval is provided at the next regularly scheduled NSRC meeting. The words in the NRC draft version would require a second NSRC meeting to be held within 14 days to approve these records. To avoid this, use the insert words provided by KG&E.

W.C. Only

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9, and

b. Each REPORTABLE OCCURRENCE requiring 24-hour notification to the Commission shall be reviewed by the PSRC and submitted to the NSRC and the Vice President-Nuclear.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following ACTION 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 and the NSRC shall be notified within 24 hours;

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSRC. 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 NSRC and the Vice President-Nuclear within 14 days of the violation; and

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,~~

~~b. Refueling operations,~~

See insert C

~~c. Surveillance and test activities of safety-related equipment,~~

~~d. Security Plan implementation,~~

~~e. Emergency Plan implementation,~~

k

W.C. ONLY

f. Fire Protection Program implementation,

g. Process Control Program implementation,

h. ODCM implementation, and

i. Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 4.15, December 1977.

6.8.2 Each procedure of Specification 6.8.1 above, and changes thereto, shall be reviewed prior to implementation as set forth in Specification 6.5 above.

See insert C

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 management staff, at least one of whom holds a Senior Operator license on the Wolf Creek Generating Station; and

c. The change is documented, reviewed by the PSRC, and approved by the Plant Superintendent within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

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 containment spray System and the Safety Injection System, ~~chemical and volume control, Waste Gas System, and hydrogen recombiners.~~ 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.

Insert C

6.8.1 The plant shall be operated and maintained in accordance with approved procedures. Major procedures, supported by appropriate minor procedures (such as checkoff lists, operating instructions, data sheets, alarm responses, etc.) shall be provided for the following operations where these operations involve nuclear safety of the plant:

a. The applicable procedures recommended in Appendix "A" of Regulation 1.33, Rev. 2, Feb 1978

b. Refueling operation

c. Emergency plan implementation

d. Surveillance and Testing of safety-related equipment

e. Fire Protection Program implementation

f. Radioactive - waste processing implementation

g. Offsite Dose Calculation Manual implementation

6.8.2 Approval of Procedures

a. All major procedures of the categories listed in 6.8.1 and modifications to the intent thereof shall be reviewed by the Plant Safety Review Committee and approved by the Plant ~~Superintendent~~ Manager prior to implementation and reviewed periodically as set forth in Administrative Procedures.

b. Minor procedures (checkoff lists, operating instructions, data sheets, alarm responses, chemistry and analytical procedures, technical instructions, special and routine maintenance procedures, laboratory manuals, etc.) shall, prior to initial use, be approved by the PSRC.

6.8.3 Changes to Procedures

a. Temporary changes to major procedures, of the categories listed in 6.8.1 which do not change the intent of the original or subsequent approved procedure, may be made provided such changes to operating procedures are approved by the Shift Supervisor (SRO licensed) and one of the Call Superintendents. For temporary changes to major procedures under the jurisdiction of Maintenance, Instrumentation and Control, Reactor Engineering, Chemistry, or Health Physics which do not change the intent, changes may be made upon approval of the cognizant group leader and a Call Superintendent. All temporary changes to major procedures (made by a Call Superintendent and either a cognizant group head or the Shift Supervisor) shall subsequently be reviewed by the

Plant Safety Review Committee and approved by the Plant Manager ~~Superintendent~~ within 2 weeks; except that temporary changes to major procedures made during a refueling outage may be reviewed and approved at any time prior to initial criticality of the reload core. All permanent changes to major procedures shall be made in accordance with Step. 6.8.2.a.

b. All temporary or permanent changes to minor operating procedures (checkoff lists, alarm responses, data sheets, operating instructions, etc.) shall be approved by the Shift Supervisor, and shall be subsequently reviewed and approved by the Operations PSRC Subcommittee. All temporary or permanent changes to other minor procedures under the jurisdiction of Maintenance, Instrumentation and Control, Reactor Engineering, Chemistry, or Health Physics, shall be approved by a supervisor of the cognizant group and shall be subsequently reviewed and approved by the appropriate PSRC Subcommittee.

Sections 6.8.1 through 6.8.3

Justification -

The KG&E procedures section is tailored to the concept presently approved by the NRC at the Point Beach Nuclear Plant, Units 1 and 2, Wisconsin Electric Power Co. This concept, the concept of major and minor procedures has proven over the past 10 years to be an effective, appropriate and reliable method of establishing the proper level of control over procedures. It is more restrictive than some recent approved Technical Specifications because it includes review of all safety-related procedures before the PSRC or a subcommittee thereof.

The area of approval of changes to procedures is also modeled after the approved Point Beach Technical Specifications and has been very effective. The concept of license required approval for procedures not operational in nature is an imposition on the senior licensed personnel which is not in accordance with the concepts and direction of the NRC to minimize the administrative burden on the Shift Supervisor.

Section 6.8.4.a

Justification -

The wording proposed by KG&E is appropriate to the WCGS design. For instance, the hydrogen recombiners are located inside containment with no connection external to containment. The CVCS automatically isolates, by design, in an accident condition.

K

W.C. ONLY

6.9.1.2 The STARTUP Report shall address each of the tests identified in the Final Safety Analysis Report 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 of 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 three 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 mrem/yr and their associated man-rem exposure according to work and job functions,[2] 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; and b. Documentation of ~~all~~ unplanned openings of ~~challenges to~~ the pressurizer power-operated relief valves (PORVs) and safety valves.

[2]This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

Section 6.9.1.5

Justification -

The wording used in the WCGS proposal was taken from NUREG-0452 (Rev. 4) in an effort to maintain standardized wording. WCGS would have minimal problems including this sentence on PORVs and safety valves provided it is reworded for clarity. For example, what is a "challenge" to the above valves. Perhaps the sentence should read "Documentation of unplanned opening of the pressurizer power-operated relief valves (PORVs) and safety valves.

W.C. ONLY

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.6 Routine 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 activites 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** 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.

revisions to sampling locations in accordance with Specification 3.12.2;

*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.

Specification 6.9.1.6

Justification -

As Branch Technical Positions (BTP) may change, the requirement for a given format in the technical specifications is excessive. Deletion of the requirement to use a BTP will allow Wolf Creek to provide the information in a form mutually acceptable to the NRC and Wolf Creek without adding possibly excessive work loads.

Also, to be consistent, all information about the Radiological Environmental Monitoring Program will be put in a single document; the Annual Radiological Environmental Operating Report. (Revisions to sample locations were previously to be put in a different document, the Semi-Annual Radioactive Effluent Release Report). The last sentence should read "...sampling schedule of Table 3.12-1; revisions of sampling locations in accordance with Specification 3.12.2; and discussion of all analyses..."

K

W.C. ONLY

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 Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped off site 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 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 Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM and to the ODCM, 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.

MONTHLY OPERATING REPORT

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of ~~all challenges to~~ the PORVs or 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 NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report. unplanned openings of

REPORTABLE OCCURRENCES

6.9.1.9 The REPORTABLE OCCURRENCES of Specifications 6.9.1.10 and 6.9.1.11 below, including corrective ACTIONS and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a Licensee Event Report shall be completed and reference shall be made to the original report date.

Section 6.9.1.8

Justification -

See the justification for Section 6.9.1.5.

W.C. ONLY

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the Final Safety Analysis Report or in the bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses;

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the Final Safety Analysis Report or Technical Specifications bases; or discovery during Unit life of conditions not specifically considered in the Final Safety Analysis Report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition; and

~~j. Failure of one or more pressurizer PORVs or safety valves.~~

THIRTY DAY WRITTEN REPORTS

6.9.1.~~13~~ 11 The types of events listed below shall be the subject of written reports to the Regional Administrator of the NRC Regional Office within 30 days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a Licensee Event Report form. Information provided on the Licensee Event Report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a.* Reactor Trip System or Engineered Safety Features Instrumentation Setpoints which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems;

b.* Conditions leading to operation in a degraded MODE permitted by a Limiting Condition for Operation or plant shutdown required by a Limiting Condition for Operation;

c. Observe inadequacies in the implementation of Administrative or Procedural Controls which threaten to cause reduction of degree of redundancy provided in Reactor Trip Systems or Engineered Safety Features Systems; and

d. Abnormal degradation of systems other than those specified in Specification 6.9.1.11c above, designed to contain radioactive material resulting from the fission process.

add next page

Insert D

*<u>NOTE</u>: Routine surveillance testing, instrument calibration, or preventative maintenance which requires system configurations as described in a and b above need not be reported except when test results themselves reveal a degraded mode as described above.

Section 6.9.1.10.j.

Justification -

Failure of one or more pressurizer PORVs or safety valves was not included in Section 6.9.1.10.j. because the WCGS Technical Specifications were written to comply with the standardized wording of NUREG-0452 (Rev. 4).

Section 6.9.1.11

Justification -

The inserted note references an accepted NRC practice. Its inclusion is appropriate in that it will ensure proper enforcement by required NRC inspectors and understanding by WCGS staff and NRC personnel.

W.C. ONLY

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 Final Safety Analysis Report;

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 to the environs;

e. Records of transient or operational cycles for those Unit components identified in Table 5.7-1;

f. Records of reactor tests and experiments;

g. Records of training and qualification for current members of the Unit Staff;

h. Records of in-service inspections performed pursuant to these Technical Specifications;

i. Records of Quality Assurance activities required by the QA Manual;

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 meetings of the ~~ORC~~ PSRC and the ~~NSRB~~ NSRC;

l. Records of the service lives of all snubbers listed in Tables 3.7-4a and 3.7-4b including the date at which the service life commences and associated installation and maintenance records;

m. Records of secondary water sampling and water quality; and

n. Records of analysis 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.~~

6.11 RADIATION PROTECTION PROGRAM

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.

Specification 6.10.2.n

Justification -

The "accuracy of the analysis" will be verified by NRC audit functions. If these audits show the analysis to be reliable, then only the final reports need to be retained for the duration of the Unit Operating License. Retention of other documentation would serve no useful purpose.