2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

APPLICABILITY

Applies to the limiting combination of thermal power, Reactor Coolant System pressure and coolant temperature during operation.

OBJECTIVE

To maintain the integrity of the fuel cladding.

SPECIFICATION

The combination of rated power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure TS 2.1-1. The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

BASIS - Safety Limits, Reactor Core (TS 2.1)

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters of rated power, reactor coolant temperature and pressure have been related to DNB through the W-3 & "L" Grid DNB correlations. The "L" Grid DNB correlation has been developed to predict the DNB flux and the location of the DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. This minimum DNBR corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.[1]

The curves of Figure TS 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the DNB ratio is equal to 1.3 or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNB ratio reaches 1.3 and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is assured is below these lines.

The curves are based on the following nuclear hot channel factors:

$F_{\Delta H}^{N} = 1.55$ $\qquad$ $F_{Q}^{N} = 2.51$

and include an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:

$F_{\Delta H}^{N} = 1.55\ [1 + 0.2\ (1 - P)]$ where P is the fraction of rated power

[1] USAR Section 3.3.3

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are given in TS 3.10.d. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits dictated by Figure TS 3.10-3 insure that the DNBR is always greater at partial power than at full power.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions that would result in a DNBR of $\leq$ 1.30.

<u>REFERENCES</u>

(1) WCAP 8092

2.2 SAFETY LIMIT, REACTOR COOLANT SYSTEM PRESSURE

APPLICABILITY

Applies to the maximum limit on Reactor Coolant System pressure.

OBJECTIVE

To maintain the integrity of the Reactor Coolant System.

SPECIFICATION

The Reactor Coolant System pressure shall not exceed 2735 psig with fuel assemblies installed in the reactor vessel.

Basis - Safety Limit, Reactor Coolant System Pressure (TS 2.2)

The Reactor Coolant System[(1)] serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the Reactor Coolant System is the primary barrier against the release of fission products. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System is assured. The maximum transient pressure allowable in the reactor pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the Reactor Coolant System piping, valves and fittings under USASI B.31.1.0 is 120% of design pressure. Thus, the safety limit of 2735 psig (110% of design pressure, 2485 psig) has been established.[(2)]

The nominal settings of the power-operated relief valves (2335 psig), the reactor high pressure trip (2385 psig) and the safety valves (2485 psig) have been established to prevent exceeding the safety limit of 2735 psig. The initial hydrostatic test was conducted at 3107 psig to assure the integrity of the Reactor Coolant System.

[(1)] USAR Section 4

[(2)] USAR Section 4.3

Leakage of Reactor Coolant (TS 3.1.d)[18]

TS (TS 3.1.d.1)

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage would not exceed the limits of 10 CFR Part 20. This is shown as follows:

If the reactor coolant activity is $91/\bar{E}\,\mu$ Ci/cc ($\bar{E}$ = average beta plus gamma energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, or through the Component Cooling System vent line, the yearly whole body dose resulting from this activity at the site boundary, using an annual average $X/Q = 2.0 \times 10^{-6}$ sec/m^3, is 0.09 rem/yr, compared with the 10 CFR Part 20 limits of 0.1 rem/yr.

With the limiting reactor coolant activity and assuming initiation of a 1 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet header would annunciate in the control room. Operators would then investigate the source of the leak and take actions necessary to isolate it. Should the leak result in a continuous discharge to the atmosphere via the component cooling surge tank and waste holdup tank, the resultant dose rate at the site boundary would be 0.09 rem/yr as given above.

Leakage directly into the containment indicates the possibility of a breach in the coolant envelope. The limitation of 1 gpm for an unidentified source of leakage is sufficiently above the minimum detectable leak rate to provide a reliable indication of leakage, and is well below the capacity of one charging pump (60 gpm).

Twelve (12) hours of operation before placing the reactor in the HOT SHUTDOWN condition are required to provide adequate time for determining whether the leak is into the containment or into one of the closed systems and to identify the leakage source.

TS 3.1.d.2

The 150 gpd leakage limit through any one steam generator is specified to ensure tube integrity is maintained in the event of a main steam line break or under loss-of-coolant accident conditions. This reduced operational leakage rate is applicable in conjunction with the tube support plate voltage-based plugging criteria as specified in TS 4.2.b.5.

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[18] USAR Sections 6.5, 11.2.3, 14.2.4

3.7 AUXILIARY ELECTRICAL SYSTEMS

APPLICABILITY

Applies to the availability of electrical power for the operation of plant auxiliaries.

OBJECTIVE

To define those conditions of electrical power availability necessary to provide 1) safe reactor operation and 2) continuing availability of engineered safety features.

SPECIFICATION

a. The reactor shall not be made critical unless all of the following requirements are satisfied:

1. The reserve auxiliary transformer is fully operational and energized to supply power to the 4160-V buses.

2. A second external source of power is fully operational and energized to supply power to emergency buses 1-5 and 1-6.

3. The 4160-V buses 1-5 and 1-6 are both energized.

4. The 480-V buses 1-52 and 1-62 and their MCC's are both energized from their respective station service transformers.

5. The 480-V buses 1-51 and 1-61 are both energized from their respective station service transformers.

6. Both station batteries and both DC systems are OPERABLE, except during testing and surveillance as described in TS 4.6.b.

7. Both diesel generators are OPERABLE. The two underground storage tanks combine to supply at least 35,000 gallons of fuel oil for either diesel generator and the day tanks for each diesel generator contain at least 1,000 gallons of fuel oil.

8. At least one pair of physically independent transmission lines serving the substation is OPERABLE. The three pairs of physically independent transmission lines are:

   A. R-304 and Q-303
   B. F-84 and Y-51
   C. R-304 and Y-51

b. During power operation or recovery from inadvertent trip, any of the following conditions of inoperability may exist during the time intervals specified. If OPERABILITY is not restored within the time specified, then within 1 hour action shall be initiated to achieve HOT STANDBY within the next 6 hours.

1. Either auxiliary transformer may be out of service for a period not exceeding 7 days provided the other auxiliary transformer and both diesel generators are OPERABLE.

2. One diesel generator may be inoperable for a period not exceeding 7 days provided the other diesel generator is tested daily to ensure OPERABILITY and the engineered safety features associated with this diesel generator are OPERABLE.

3. One battery may be inoperable for a period not exceeding 24 hours provided the other battery and two battery chargers remain OPERABLE with one charger carrying the d-c supply system.

4. If the conditions in TS 3.7.a.8 cannot be met, power operation may continue for up to 7 days provided at least two transmission lines serving the substation are OPERABLE.

5. Three off-site power supply transmission lines may be out of service for a period of 7 days provided reactor power is reduced to 50% of rated power and the two diesel generators shall be tested daily for OPERABILITY.

6. One 4160-V or 480-V engineered safety features bus may be out of service for 24 hours provided the redundant bus and its loads remain OPERABLE.

c. When its normal or emergency power source is inoperable, a system, train or component may be considered OPERABLE for the purpose of satisfying the requirements of its applicable LIMITING CONDITION FOR OPERATION, provided:

1. Its corresponding normal or emergency power source is OPERABLE; and

2. Its redundant system, train, or component is OPERABLE.

BASIS

The intent of this TS is to provide assurance that at least one external source and one standby source of electrical power is always available to accomplish safe shutdown and containment isolation and to operate required engineered safety features equipment following an accident.

Plant auxiliary power is normally supplied by two separate external power sources which have multiple off-site network connections[1]: the reserve auxiliary transformer from the 138-Kv portion of the plant substation, and a tertiary winding on the substation auto transformer. Either source is sufficient to supply all necessary accident and post-accident load requirements from any one of four available transmission lines.

Each diesel generator is connected to one 4160-V safety features bus and has sufficient capacity to start sequentially and operate the engineered safety features equipment supplied by that bus. The set of safety features equipment items supplied by each bus is, alone, sufficient to maintain adequate cooling of the fuel and to maintain containment pressure within the design value in the event of a loss-of-coolant accident.

Each diesel generator starts automatically upon low voltage on its associated bus, and both diesel generators start in the event of a safety injection signal. A minimum of 7 days fuel supply for one diesel generator is maintained by requiring 36,000 gallons of fuel oil, thus assuring adequate time to restore off-site power or to replenish fuel. The diesel fuel oil storage capacity requirements are consistent with those specified in ANSI N195-1976/ANS-59.51, Sections 5.2, 5.4, and 6.1.

The plant 125-V d-c power is normally supplied by two batteries each of which will have a battery charger in service to maintain full charge and to assure adequate power for starting the diesel generators and supplying other emergency loads. A third charger is available to supply either battery.

The arrangement of the auxiliary power sources and equipment and this TS ensure that no single fault condition will deactivate more than one redundant set of safety features equipment items and will therefore not result in failure of the plant protection systems to respond adequately to a loss-of-coolant accident.

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[1] USAR Figure 8.2-2

m. Reactor Coolant Flow

1. During steady-state power operation, reactor coolant flow rate shall be $\geq$ 89,000 gallons per minute average per loop. If reactor coolant flow rate is < 89,000 gallons per minute per loop, action shall be taken in accordance with TS 3.10.n.

2. Compliance with this flow requirement shall be demonstrated by verifying the reactor coolant flow during initial power escalation following each REFUELING, between 70% and 95% power with plant parameters as constant as practical.

n. If, during power operation any of the conditions of TS 3.10.k, TS 3.10.l, or TS 3.10.m.1 are not met, restore the parameter in 2 hours or less to within limits or reduce power to < 5% of thermal rated power within an additional 6 hours. Following analysis, thermal power may be raised not to exceed a level analyzed to maintain a minimum DNBR of 1.30.

3.12 CONTROL ROOM POST-ACCIDENT RECIRCULATION SYSTEM

APPLICABILITY

Applies to the OPERABILITY of the Control Room Post-Accident Recirculation System.

OBJECTIVE

To specify OPERABILITY requirements for the Control Room Post-Accident Recirculation System.

SPECIFICATION

a. The reactor shall not be made critical unless both trains of the Control Room Post-Accident Recirculation System are OPERABLE.

b. Both trains of the Control Room Post-Accident Recirculation System, including filters, shall be OPERABLE or the reactor shall be shut down within 12 hours, except that when one of the two trains of the Control Room Post-Accident Recirculation System is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days.

c. During testing the system shall meet the following performance requirements:

1. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filter and charcoal adsorber banks shall show $\geq$ 99% DOP removal and $\geq$ 99% halogenated hydrocarbon removal.

2. The results of the laboratory carbon sample analysis from the Control Room Post-Accident Recirculation System carbon shall show $\geq$ 90% radioactive methyl iodide removal at conditions of 66°C, and 95% RH.

3. Fans shall operate within $\pm$ 10% of design flow when tested.

BASIS - Control Room Post-Accident Recirculation System (TS 3.12)

The Control Room Post-Accident Recirculation System is designed to filter the Control Room atmosphere during Control Room isolation conditions. The Control Room Post-Accident Recirculation System is designed to automatically start upon SIS or high radiation signal at inlet of unit.

If the system is found to be inoperable, there is no immediate threat to the Control Room and reactor operation may continue for a limited period of time while repairs are being made. If the system cannot be repaired within 7 days, the reactor is placed in HOT STANDBY until the repairs are made.

3.14 SHOCK SUPPRESSORS (SNUBBERS)

APPLICABILITY

Applies to the OPERABILITY of shock suppressors which are related to plant safety.

OBJECTIVE

To ensure that shock suppressors, which are used to restrain safety-related piping under dynamic load conditions, are functional during reactor operation.

SPECIFICATION

a. The reactor shall not be made critical unless all safety-related shock suppressors are OPERABLE except as noted in 3.14.b.

b. During power operation or recovery from inadvertent trip, if any safety-related shock suppressor is found inoperable one of the following actions shall be taken within 72 hours:

   1. The inoperable shock suppressor shall be restored to an OPERABLE condition or replaced with a spare shock suppressor of similar specifications; or

   2. The fluid line restrained by the inoperable shock suppressor shall, if feasible, be isolated from other safety-related systems if otherwise permitted by the TS and thereafter operation may continue subject to any limitations by the TS for that fluid line; or

   3. Actions shall be initiated to shut down the reactor and the reactor shall be in a HOT SHUTDOWN condition within 36 hours.

BASIS - Shock Suppressors (Snubbers) (TS 3.14)

Shock suppressors (snubbers) are designed to prevent unrestrained pipe motion under dynamic loads, as might occur during seismic activity or severe plant transients, while allowing normal thermal motion during startup or shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic event or other events initiating dynamic loads. It is therefore required that all snubbers designed to protect the reactor coolant and other safety-related systems or components be operable during reactor operation. The intent of this TS is to prohibit startup or continued operation with defective safety-related shock suppressors.

Because the protection afforded by snubbers is required only during low probability events, TS 3.14.b allows a period of 72 hours for repairs or feasible alternative action before reactor shutdown is required.

TABLE TS 3.1-2

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

SYSTEM	VALVE NO.	MAXIMUM[1][2] ALLOWABLE LEAKAGE BASED ON NORMAL OPERATING PRESSURE
Reactor Vessel, Core Flooding Line (Upper Plenum Injection)	SI-304A	≤ 5.0 gallons per minute
	SI-303A	≤ 5.0 gallons per minute
	SI-304B	≤ 5.0 gallons per minute
	SI-303B	≤ 5.0 gallons per minute
Loop B 12" Accumulator Discharge Line	SI-22B	≤ 5.0 gallons per minute

[1]Leakage rates ≤ 1.0 gpm are considered acceptable.

Leakage rates > 1.0 gpm but ≤ 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

Leakage rates > 1.0 gpm but ≤ 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

Leakage rates greater than 5.0 gpm are considered unacceptable.

[2]Minimum test differential pressure shall not be < 150 psid.

BASIS

The Auxiliary Feedwater System (AFW) mitigates the consequences of any event that causes a loss of normal feedwater. The design basis of the AFW System is to remove decay and residual heat by delivering the minimum required flow to at least one steam generator until the Reactor Coolant System (RCS) is cooled to the point of placing the Residual Heat Removal System into operation.

In accordance with ASME Code Section XI, Subsection IWP, an in-service test of each auxiliary feedwater pump shall be run nominally every 3 months (quarterly) during normal plant operation. It is recommended that this test frequency be maintained during shutdown periods if this can be reasonably accomplished, although this is not mandatory. If the normally scheduled test is not performed during a plant shutdown, then the motor-driven pumps shall be demonstrated OPERABLE within 1 week exceeding 350°F; and the turbine-driven pump shall be demonstrated OPERABLE within 72 hours of exceeding 350°.

Quarterly testing of the AFW pumps is used to detect degradation of the component. This type of testing may be accomplished by measuring the pump's developed head at one point of the pump characteristic curve. This verifies that the measured performance is within an acceptable tolerance of the original pump baseline performance.

TS 3.4.b requires all three AFW pumps be OPERABLE prior to heating the RCS average temperature > 350°F. It is acceptable to heat the RCS to > 350°F with the turbine-driven pump inoperable for a limited time period of 72 hours. The wording of TS 3.4.b.2.B and TS 4.8.b allows delaying the testing until the steam flow is consistent with the conditions under which the performance acceptance criteria were generated.

The discharge valves of the two motor-operated pumps are normally open, as are the suction valves from the condensate storage tanks and the two valves on a cross tie line that directs the turbine-driven pump discharge to either or both steam generators. The only valve required to function upon initiation of auxiliary feedwater flow is the steam admission valve on the turbine-driven pump. Proper opening of the steam admission valve will be demonstrated each time the turbine-driven pump is tested.

4.9 REACTIVITY ANOMALIES

APPLICABILITY

Applies to potential reactivity anomalies.

OBJECTIVE

To require evaluation of reactivity anomalies within the reactor.

SPECIFICATION

Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of 1% in reactivity, an evaluation as to the cause of the discrepancy shall be made and reported to the Commission within 30 days.

BASIS - REACTIVITY ANOMALIES[1]

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burn-up and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burn-up. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safe limit since a shutdown margin of at least 1% with the most reactive rod in the fully withdrawn position is always maintained.

[1] USAR Section 3.2

4.12 SPENT FUEL POOL SWEEP SYSTEM

APPLICABILITY

Applies to testing and surveillance requirements for the spent fuel pool sweep system in TS 3.8.a.9.

OBJECTIVE

To verify the performance capability of the spent fuel pool sweep system.

SPECIFICATION

a. At least once per operating cycle or once every 18 months, whichever occurs first, the following conditions shall be demonstrated:

   1. Pressure drop across the combined HEPA filters and charcoal adsorber banks is $\leq$ 10 inches of water and the pressure drop across any HEPA bank is $\leq$ 4 inches of water at the system design flow rate ($\pm$ 10%).

   2. Automatic initiation of each train of the system.

b. 1. The in-place DOP test for HEPA filters shall be performed (1) at least once per 18 months and (2) after each complete or partial replacement of a HEPA filter bank or after any maintenance on the system that could affect the HEPA bank bypass leakage.

   2. The laboratory tests for activated carbon in the charcoal filters shall be performed (1) at least once per 18 months for filters in a standby status or after 720 hours of filter operation, and (2) following painting, fire, or chemical release in any ventilation zone communicating with the system.

   3. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any maintenance on the system that could affect the charcoal adsorber bank bypass leakage.

c. Perform an air distribution test on the HEPA filter bank after any maintenance or testing that could affect the air distribution within the system. The test shall be performed at design flow rate (± 10%). The results of the test shall show the air distribution is uniform within ± 20%[1].

[1] In WPS letter of August 25, 1976 to Mr. Al Schwencer (NRC) from Mr. E. W. James, we relayed test results for flow distribution for tests performed in accordance with ANSI N510-1975. This standard refers to flow distribution tests performed upstream of filter assemblies. Since the test results upstream of filters were inconclusive due to high degree of turbulence, tests for flow distribution were performed downstream of filter assemblies with acceptable results (within 20%). The safety evaluation attached to Amendment 12 references our letter of August 25, 1976 and acknowledges acceptance of the test results.

BASIS

Pressure drop across the combined HEPA filters and charcoal adsorbers of $\leq$ 10 inches of water and 4 inches across any HEPA filter bank at the system design flow rate ($\pm$ 10%) will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A test frequency of once per operating cycle establishes system performance capability. This pressure drop is approximately 6 inches of water when filters are clean.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52 dated June 1973. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly, and obtaining at least two samples. Each sample should be at least 2 inches in diameter and a length equal to the thickness of the bed. The use of multi-sample assemblies for test samples is an acceptable alternate to mixing one bed for a sample. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system should be replaced. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52 (Rev. 1) dated June 1976.

If painting, fire, or chemical release occurs such that the charcoal adsorbers become contaminated from the fumes, chemicals, or foreign materials, the same tests and sample analysis should be performed as required for operational use.

Degradation of the HEPA filters due to painting, fire or chemical release in a communicating ventilation zone would be detected by an increased pressure drop across the filters. Should the filters become contaminated, engineering judgment would be used to determine if further leakage and/or efficiency testing was required.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline only.

4.13 RADIOACTIVE MATERIALS SOURCES

APPLICABILITY

Applies to the possession, leak test, and record requirements for radioactive material sources required for operation of the facility.

OBJECTIVE

To ensure that radioactive material sources which are beneficial to facility operation are available to the facility and these sources are verified to be free from leakage.

SPECIFICATION

a. Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or the State.

b. Sources which contain by-product material that exceeds the quantities listed in 10 CFR 30.71, Schedule B, and all other sources containing > 0.1 microcuries shall be leak tested in accordance with this TS.

c. Any source specified by TS 4.13.2 which is determined to be leaking shall be immediately withdrawn from use, repaired or disposed of in accordance with the Commission's regulations. Leaking is defined as the presence of .005 microcuries of the source's radioactive material on the test sample.

d. Each sealed source with a half-life > 30 days, and in any form other than gas, shall be tested for leakage at intervals not to exceed 6 months, except for:

   1. Startup sources inserted in the reactor vessel,
   2. Fission detectors following exposure to core flux,
   3. Irradiation sample sources inserted in the reactor vessel,
   4. Sources enclosed within the Eberline Model 1000 Multi-Source Gamma Calibrator,
   5. Sources enclosed within the Shepherd Model 89-400 Self-Contained Calibrator, and
   6. Hydrogen-3 sources.

e. Sources specified by TS 4.13.2 which are in storage and not being used are exempt from the testing of TS 4.13.4. Prior to use or transfer to another licensee of such a source, the leakage test of TS 4.13.4 shall be current.

f. Startup sources and fission detectors shall be leak tested prior to initial insertion into the reactor vessel or prior to being subjected to core flux.

g. A complete inventory of radioactive materials sources shall be maintained current at all times.

BASIS

Ingestion or inhalation of source material may give rise to total body or organ irradiation. This specification assures that leakage from radioactive material sources does not exceed allowable limits. In the unlikely event that those quantities of radioactive by-product materials of interest to this specification which are exempt from leakage testing are ingested or inhaled, they represent less than one maximum permissible body burden for total body irradiation. The limits for all other sources (including alpha emitters) are based upon 10 CFR 70.39(c) limits for plutonium.

The Eberline Model 1000 Multi-Source Calibrator and the J. L. Shepherd Model 89-400 are totally enclosed instrument calibrating assemblies for which leak testing of the enclosed sources is not practical. Leak testing of these sources would require disassembly of the calibration assembly shield, controls, etc., resulting in personnel exposure without corresponding benefits.

4.17 CONTROL ROOM POSTACCIDENT RECIRCULATION SYSTEM

APPLICABILITY

Applies to testing and surveillance requirements for the Control Room Postaccident Recirculation System in TS 3.12.

OBJECTIVE

To verify the performance capability of the Control Room Postaccident Recirculation System.

SPECIFICATION

a. At least once per operating cycle or once every 18 months, whichever occurs first, the following conditions shall be demonstrated:

   1. Pressure drop across the combined HEPA filters and charcoal adsorber banks is $\leq$ 6 inches of water and the pressure drop across any HEPA bank is $\leq$ 4 inches of water at the system design flow rate ($\pm$ 10%).

   2. Automatic initiation of the system on a high radiation signal at the inlet of the unit and a safety injection signal.

b. 1. The in-place DOP test for HEPA filters shall be performed (1) at least once per 18 months and (2) after each complete or partial replacement of a HEPA filter bank or after any maintenance on the system that could affect the HEPA bank bypass leakage.

   2. The laboratory tests for activated carbon in the charcoal filters shall be performed (1) at least once per 18 months for filters in a standby status or after 720 hours of filter operation, and (2) following painting, fire, or chemical release in any ventilation zone communicating with the system.

   3. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any maintenance on the system that could affect the charcoal adsorber bank bypass leakage.

   4. Each train shall be operated at least 10 hours each month.

BASIS

Control Room Post-Accident Recirculation System

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water and 4 inches across any HEPA filter bank at the system design flow rate ($\pm$ 10%) will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A filter test frequency of once per operating cycle establishes system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52, dated June 1973. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly, and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. The use of multi-sample assemblies for test samples is an acceptable alternate to mixing one bed for a sample. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system should be replaced.

TABLE TS 4.1-3
MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

EQUIPMENT TESTS[1]	TEST	FREQUENCY
1. Control Rods	Rod drop times of all full length rods Partial movement of all rods not fully inserted in the core	Each REFUELING outage Every 2 weeks when at or above HOT STANDBY
1a. Reactor Trip Breakers	Independent test[2] shunt and undervoltage trip attachments	Monthly
1b. Reactor Coolant Pump Breakers- Open-Reactor Trip	OPERABILITY	Each REFUELING outage
1c. Manual Reactor Trip	Open trip reactor[3] trip and bypass breaker	Each REFUELING outage
2. Deleted		
3. Deleted		
4. Containment Isolation Trip	OPERABILITY	Each REFUELING outage
5. Refueling System Interlocks	OPERABILITY	Prior to fuel movement each REFUELING outage
6. Deleted		
7. Deleted		
8. RCS Leak Detection	OPERABILITY	Weekly[4]
9. Diesel Fuel Supply	Fuel Inventory[5]	Weekly
10. Deleted		
11. Fuel Assemblies	Visual Inspection	Each REFUELING outage
12. Guard Pipes	Visual Inspection	Each REFUELING outage
13. Pressurizer PORVs	OPERABILITY	Each REFUELING cycle
14. Pressurizer PORV Block Valves	OPERABILITY	Quarterly[6]
15. Pressurizer Heaters	OPERABILITY[7]	Each REFUELING cycle
16. Containment Purge and Vent Isolation Valves	OPERABILITY[8]	Each REFUELING cycle

[1] Following maintenance on equipment that could affect the operation of the equipment, tests should be performed to verify OPERABILITY.

[2] Verify OPERABILITY of the bypass breaker undervoltage trip attachment prior to placing breaker into service.

[3] Using the Control Room push-buttons, independently test the reactor trip breakers shunt trip and undervoltage trip attachments. The test shall also verify the undervoltage trip attachment on the reactor trip bypass breakers.

[4] When reactor is at power or in HOT SHUTDOWN condition.

[5] Inventory of fuel required in all plant modes.

[6] Not required when valve is administratively closed.

[7] Test will verify OPERABILITY of heaters and availability of an emergency power supply.

[8] This test shall demonstrate that the valve(s) close in $\leq$ 5 seconds.

B. As per applicable portions of Regulatory Guide 1.16, a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated person rem exposure according to work and job functions,[1] e.g., reactor operations and surveillance, in-service inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and REFUELING. The dose assignment to various duty functions may be estimates based on pocket dosimeter or TLD. Small exposures totaling < 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 shall be assigned to specific major work functions.

C. Challenges to and failures of the pressurizer power operated relief valves and safety valves.[2]

D. This report shall document the results of specific activity analysis in which the reactor coolant exceeded the limits of TS 3.1.c.1.A during the past year. The following information shall be included:

(1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded;

(2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analvsis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations;

(3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded;

(4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and

(5) The time duration when the specific activity of the reactor coolant exceeded the radioiodine limit.

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[1]This tabulation supplements the requirements of Section 20.2206(b) of 10 CFR Part 20.

[2]Letter from E. R. Mathews (WPSC) to D. G. Eisenhut (U.S. NRC) dated January 5, 1981.

6.13 HIGH RADIATION AREA

a. In lieu of the "control device" or "alarm signal" required by Paragraph 20.1601(a) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is > 100 mrem/hr, but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)[1]. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following.

1. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

2. A radiation monitoring device which continuously integrates the radiation dose in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

3. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the RWP.

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[1]Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

b. In addition to the requirements of 6.13.a., areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose > 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose > 1000 mrem[2] that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

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[2]Measurement made at 30 centimeters from source of radioactivity.

6.19 **MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS**[1]

Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

a. Shall be reported to the Commission in the Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PORC. The discussion of each change shall contain:

   1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.

   2. Sufficient information to support the reason for the change without benefit of additional or supplemental information;

   3. A description of the equipment, components and processes involved and the interfaces with other plant systems;

   4. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

   5. An evaluation of the change, which shows the expected maximum exposures to individuals in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

   6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

   7. An estimate of the exposure to plant operating personnel as a result of the change; and

   8. Documentation of the fact that the change was reviewed and found acceptable by the PORC.

b. Shall become effective upon review and acceptance by the PORC.

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(1) Licensees may choose to submit the information called for in this TS as part of the periodic USAR update.