LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION PAGE

1.0 DEFINITIONS

MINIMUM CRITICAL POWER RATIO (MCPR)

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Specification 6.8 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and ~~Semi-annual~~ Radioactive Effluent Release Reports required by Specification 6.9.

Annual

OPERABLE - OPERABILITY

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

OPERATIONAL MODE

An OPERATIONAL MODE, i.e., MODE, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1-2.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the UFSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

TABLE 4.1.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	Applicable OPERATIONAL MODES	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL[a] CALIBRATION
1. Intermediate Range Monitor:				
a. Neutron Flux - High	2 3, 4, 5	S[b] S	S/U[c], W[o] W[o]	E[o,d] E[o,f]
b. Inoperative	2, 3, 4, 5	NA	W[o]	NA
2. Average Power Range Monitor[f]:				
a. Setdown Neutron Flux - High	2 3, 5[m]	S[b] S	S/U[c], W[o] W	SA[o] SA
b. Flow Biased Neutron Flux - High	1	S, D[g]	W	W[d,e], SA
c. Fixed Neutron Flux - High	1	S	W	W[d], SA
d. Inoperative	1, 2, 3, 5[m]	NA	W	NA
3. Reactor Vessel Steam Dome Pressure - High	1, 2[l]	NA	M	Q
4. Reactor Vessel Water Level - Low	1, 2	D	M	E[h]
5. Main Steam Line Isolation Valve - Closure	1, 2[p]	NA	M	E
6. Main Steam Line Radiation - High	1, 2[l]	S	M	E[q]
7. Drywell Pressure - High	1, 2[n]	NA	M	Q

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least (½) decades during each startup after entering OPERATIONAL MODE 2 and the IRM and APRM channels shall be determined to overlap for at least (½) decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days. The weekly CHANNEL FUNCTIONAL TEST may be used to fulfill this requirement.

(d) This calibration shall consist of the adjustment of the APRM CHANNEL to conform, within 2% of RATED THERMAL POWER, to the power values calculated by a heat balance during OPERATIONAL MODE 1 when THERMAL POWER is ≥25% of RATED THERMAL POWER. This adjustment must be accomplished: a) within 2 hours if the APRM CHANNEL is indicating lower power values than the heat balance, or b) within 12 hours if the APRM CHANNEL is indicating higher power values than the heat balance. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

Any APRM CHANNEL gain adjustment made in compliance with Specification 3.11.B shall not be included in determining the above difference. This calibration is not required when THERMAL POWER is <25% of RATED THERMAL POWER. The provisions of Specification 4.0.D are not applicable.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

DELETED

(g) (f) The LPRMs ~~shall be~~ calibrated at least once per 2000 effective full power hours (EFPH).

~~(g) Verify measured recirculation loop flow to be greater than or equal to established recirculation loop flow at the existing pump speed.~~

(h) Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

(i) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.12.A.

(j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(k) This function may be bypassed, provided a control rod block is actuated, for reactor protection system reset in Refuel and Shutdown positions of the reactor mode switch.

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

(l) This function not required to be OPERABLE when THERMAL POWER is less than 45% of RATED THERMAL POWER.

(m) Required to be OPERABLE only prior to and during required SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(n) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(o) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for a period of 24 hours after entering OPERATIONAL MODE 2 or 3 when shutting down from OPERATIONAL MODE 1.

(p) This function is not required to be OPERABLE when reactor pressure is less than 600 psig.

(q) A current source provides an instrument channel alignment every 3 months.

(r) The CHANNEL CALIBRATION surveillance requirements shall be performed if not performed within the previous seven days.

TABLE 4.2.B-1

ECCS ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. CORE SPRAY (CS) SYSTEM				
a. Reactor Vessel Water Level - Low Low	S	M	Q(f)	1, 2, 3, 4[b], 5[b]
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Reactor Vessel Pressure - Low (Permissive)	NA	M	Q	1, 2, 3, 4[b], 5[b]
d. CS Pump Discharge Flow - Low (Bypass)	NA	~~M~~ Q	Q[e]	1, 2, 3, 4[b], 5[b]
2. LOW PRESSURE COOLANT INJECTION (LPCI) SUBSYSTEM				
a. Reactor Vessel Water Level - Low Low	S	M	Q(f)	1, 2, 3, 4[b], 5[b]
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Reactor Vessel Pressure - Low (Permissive)	NA	M	Q	1, 2, 3, 4[b], 5[b]
d. LPCI Pump Discharge Flow - Low (Bypass)	NA	~~M~~ Q	Q[e]	1, 2, 3, 4[b], 5[b]
3. HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM[a]				
a. Reactor Vessel Water Level - Low Low	S	M	Q(f)	1, 2, 3
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Condensate Storage Tank Level - Low	NA	M	NA	1, 2, 3
d. Suppression Chamber Water Level - High	NA	M	NA	1, 2, 3
e. Reactor Vessel Water Level - High (Trip)	NA	M	~~E~~ Q(f)	1, 2, 3
f. HPCI Pump Discharge Flow - Low (Bypass)	NA	~~M~~ Q	Q	1, 2, 3
g. Manual Initiation	NA	E	NA	1, 2, 3

TABLE 4.[illegible]-1 (Continued)

ECCS ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
4. AUTOMATIC DEPRESSURIZATION SYSTEM[a]				
a. Reactor Vessel Water Level - Low Low	S	M	Q (f)	1, 2, 3
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Initiation Timer	NA	E	E	1, 2, 3
d. Low Low Level Timer	NA	E	E	1, 2, 3
e. CS Pump Discharge Pressure - High (Permissive)	NA	M	Q	1, 2, 3
f. LPCI Pump Discharge Pressure - High (Permissive)	NA	M	Q	1, 2, 3
5. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	NA	E	E	1, 2, 3, 4[c], 5[c]
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	NA	E	E	1, 2, 3, 4[c], 5[c]

TABLE 4.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Not required to be OPERABLE when reactor steam dome pressure is ≤150 psig.

(b) When the system is required to be OPERABLE per Specification 3.5.B.

(c) Required when the associated diesel generator is required to be OPERABLE per Specification 3.9.B.

(d) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(e) Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

(f) Unit 2 transmitters are calibrated once per 18 months. Unit 2 trip units and Unit 3 level switches are calibrated at the frequency identified in the table.

TABLE 4.2.C-1

ATWS - RPT INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Water Level - Low Low	S	~~M~~ Q	E[a]
2. Reactor Vessel Pressure - High	S	~~M~~ Q	E[a]

a Trip units are calibrated at least once per ~~31~~ 92 days and transmitters are calibrated at the frequency identified in the table.

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function[i]	Applicable OPERATIONAL MODE(s)	ACTION
3. SOURCE RANGE MONITORS				
a. Detector not full in[b]	NA	3 2	2 5	51 51
b. Upscale[c]	$\leq 1 \times 10^5$ cps	3 2	2 5	51 51
c. Inoperative[c]	NA	3 2	2 5	51 51
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in[d]	NA	6	2, 5	51
b. Upscale	$\leq$108/125 of full scale	6	2, 5	51
c. Inoperative	NA	6	2, 5	51
d. Downscale[d]	$\geq$5/125 of full scale	6	2, 5	51

TABLE 4.2.E-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
1. ROD BLOCK MONITORS				
a. Upscale	NA	S/U[b,c], M[c]	Q	1[d]
b. Inoperative	NA	S/U[b,c], M[c]	NA	1[d]
c. Downscale	NA	S/U[b,c], M[c]	Q	1[d]
2. AVERAGE POWER RANGE MONITORS				
a. Flow Biased Neutron Flux - High				
1. Dual Recirculation Loop Operation	NA	S/U[b], M	SA	1
2. Single Recirculation Loop Operation	NA	S/U[b], M	SA	1
b. Inoperative	NA	S/U[b], M	NA	1, 2, 5[i]
c. Downscale	NA	S/U[b], M	Q	1
d. Startup Neutron Flux - High	NA	S/U[b], M	SA	2, 5[i]
3. SOURCE RANGE MONITORS				
a. Detector not full in[f]	NA	S/U[b], W	E	2,[i] 5
b. Upscale[g]	NA	S/U[b], W	E	2,[i] 5
c. Inoperative[g]	NA	S/U[b], W	NA	2,[i] 5

TABLE 4.[illegible]-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[e]	Applicable OPERATIONAL MODE(s)
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in[h]	NA	S/U[b], W	E	2[i], 5
b. Upscale	NA	S/U[b], W	E[k]	2[i], 5
c. Inoperative	NA	S/U[b], W	NA	2[i], 5
d. Downscale[h]	NA	S/U[b], W	E[k]	2[i], 5
5. SCRAM DISCHARGE VOLUME (SDV)				
a. Water Level - High	NA	Q	NA	1, 2, 5[e]
b. SDV Switch in Bypass	NA	E	NA	5[e]

TABLE 4.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) Within 7 days prior to startup.

(c) Includes reactor manual control "relay select matrix" system input.

(d) With THERMAL POWER ≥30% of RATED THERMAL POWER.

(e) With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(f) This function shall be automatically bypassed if detector count rate is >100 cps or the IRM channels are on range 3 or higher.

(g) This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

(h) This function shall be automatically bypassed when the IRM channels are on range 1.

(i) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for entry into the applicable OPERATIONAL MODE(s) from OPERATIONAL MODE 1 provided the surveillances are performed within 12 hours after such entry

(j) Required to be OPERABLE only during SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

~~(k) The CHANNEL CALIBRATION surveillance requirements shall be performed within 12 hours upon each entry into any OPERATIONAL MODE(s) from OPERATIONAL MODE 1 if not performed within the previous seven days.~~

TABLE 3.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION

ACTION 60 - a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 61- With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:

a. Either restore the inoperable CHANNEL(s) to OPERABLE status within 7 days of the event, or

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.B within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 62- a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) one less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

~~b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1; and provided the high radiation sampling system (HRSS) combustible gas monitoring capability for the drywell is OPERABLE; restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.~~

b. ~~c.~~ With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1~~; and the HRSS combustible gas monitoring capability for the drywell inoperable~~; restore at least one inoperable CHANNEL to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

TABLE 4.2.F-1

ACCIDENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. Reactor Vessel Pressure	M	SA	1, 2
2. Reactor Vessel Water Level	M	SA (d)	1, 2
3 Torus Water Level	M	A	1, 2
4. Torus Water Temperature	M	A	1, 2
5. Drywell Pressure - Wide Range	M	E	1, 2
6. Drywell Pressure - Narrow Range	M	Q	1, 2
7. Drywell Air Temperature	M	E	1, 2
8. Drywell ~~Hydrogen/~~Oxygen Concentration - Analyzer and Monitor	M	Q	1, 2
9. Drywell Hydrogen Concentration - Analyzer and Monitor	M	Q	1, 2
10. ~~9.~~ Safety/Relief Valve Position Indicators - Acoustic & Temperature	M[c]	E	1, 2
11. ~~10.~~ (Source Range) Neutron Monitors	M	Q[b]	1, 2, 3
12. ~~11.~~ Drywell Radiation Monitors	M	E[a]	1, 2
13. ~~12.~~ Torus Pressure	M	Q	1, 2

TABLE 4.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) CHANNEL CALIBRATION shall consist of an electronic calibration of the CHANNEL, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

(b) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

(c) CHANNEL CHECK of the Acoustic Monitors shall consist of verifying the instrument threshold levels.

(d) Analog transmitters are calibrated every 18 months. The control Room indication for the analog transmitter is calibrated at the frequency identified in the table.

BASES

3/4.2.E Control Rod Block Actuation Instrumentation

The control rod block functions are provided to prevent excessive control rod withdrawal so that the MINIMUM CRITICAL POWER RATIO (MCPR) does not go below the MCPR fuel cladding integrity Safety Limit. During shutdown conditions, control rod block instrumentation initiates withdrawal blocks to ensure that all control rods remain inserted to prevent inadvertent criticality.

The trip logic for this function is one-out-of-n; e.g., any trip of one of the six average power range monitors (APRMs), eight intermediate range monitors (IRMs), or four source range monitors (SRMs), will result in a rod block. The minimum instrument CHANNEL requirements assure sufficient instrumentation to assure that the single failure criterion is met. The minimum instrument CHANNEL requirements for the rod block monitor may be reduced by one for a short period of time to allow for maintenance, testing, or calibration.

The APRM rod block function is flow-biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection, i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

In the REFUEL and STARTUP/HOT STANDBY OPERATIONAL MODE(s), the APRM rod block function setpoint is significantly reduced to provide the same type of protection in the REFUEL and STARTUP/HOT STANDBY OPERATIONAL MODE(s) as the APRM flow-biased rod block does in the RUN OPERATIONAL MODE, i.e., prevents control rod withdrawal before a scram is reached.

The rod block monitor (RBM) function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error. The trip setting is flow-biased. At low power, the worst-case withdrawal of a single control rod without rod block action will not violate the fuel cladding integrity Safety Limit. Thus the RBM rod block function is not required below the specified power level. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the fuel cladding integrity Safety Limit.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of ten above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity Safety Limit.

A downscale indication on an APRM is an indication that the instrument has failed or is not sufficiently sensitive. In either case, the instrument will not respond to changes in control rod motion, and the control rod motion is thus prevented.

The SRM rod blocks of low count rate and the detector not fully inserted assure that the SRMs are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram discharge volume, ~~and~~ a high water level rod block provide~~s~~ annunciation for operator action. The alarm setpoint has been selected to provide adequate time to allow for the determination of the cause for the level increase and corrective action prior to automatic scram initiation.

3.3 - LIMITING CONDITIONS FOR OPERATION

D. Maximum Scram Insertion Times

The maximum scram insertion time of each control rod from the fully withdrawn position to 90% insertion, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

1. Declare the control rod(s) exceeding the above maximum scram insertion time inoperable, and
2. When operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of power operation. ALL CAPS

With the provisions of the ACTION(s) above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

D. Maximum Scram Insertion Times

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators:

1. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER:
   a. following CORE ALTERATION(s), or
   b. after a reactor shutdown that is greater than 120 days,
2. For specifically affected individual control rods[a] following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and
3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of power operation.

ALL CAPS

a The provisions of Specification 4.0.D are not applicable provided this surveillance is conducted prior to exceeding 40% of RATED THERMAL POWER.

3.3 - LIMITING CONDITIONS FOR OPERATION

N. Economic Generation Control (EGC) System

The economic generation control (EGC) system may be in operation with automatic flow control provided:

(1.) ~~a)~~ Core flow is within 65% to 100% of rated core flow, and

(2.) ~~b)~~ THERMAL POWER is ≥20% of RATED THERMAL POWER.

APPLICABILITY

OPERATIONAL MODE 1.

ACTION:

With core flow less than 65% or greater than 100% of rated core flow, or THERMAL POWER less than 20% of RATED THERMAL POWER, restore operation to within the limits within one hour. Otherwise, immediately remove the plant from EGC operation.

4.3 - SURVEILLANCE REQUIREMENTS

N. Economic Generation Control (EGC) System

The economic generation control system shall be demonstrated OPERABLE by verifying that core flow is within 65% to 100% of rated core flow and THERMAL POWER is ≥20% of RATED THERMAL POWER:

(1.) ~~a)~~ Prior to entry into EGC operation, and

(2.) ~~b)~~ At least once per 12 hours while operating in EGC.

BASES

3/4.3.A SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The SHUTDOWN MARGIN limitation is a restriction to be applied principally to a new refueling pattern. Satisfaction of the limitation must be determined at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. This determination is provided by core design calculations and administrative control of fuel loading patterns. These procedures include restrictions to allow only those intermediate fuel assembly configurations that have been shown to provide the required SHUTDOWN MARGIN.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed during xenon free conditions and adjusted to 68°F to accommodate the current moderator temperature. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.38% Δk/k or R + 0.28% Δk/k, as appropriate, with the strongest control rod fully withdrawn and all others fully inserted. Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of determination of the highest control rod worth, either analytically or by test. This is due to the reduced uncertainty in the SHUTDOWN MARGIN test when the highest worth control rod is determined by demonstration. When SHUTDOWN MARGIN is determined by calculations not associated with a test, additional margin must be added to the specified SHUTDOWN MARGIN limit to account for uncertainties in the calculation.

The value of R in units of % Δk/k is the difference between the calculated beginning-of-life core reactivity, at the beginning of the operating cycle, and the calculated value of maximum core reactivity at any time later in the operating cycle, where it would be greater than at the beginning. The value of R shall include the potential SHUTDOWN MARGIN loss assuming full B_4C settling in all inverted poison tubes present in the core. R must be a positive quantity or zero and a new value of R must be determined for each new fuel cycle.

Δk/k

The value of % Δk/k in the above expression is provided as a finite, demonstrable, subcriticality margin. This margin is verified using an in-sequence control rod withdrawal at the beginning-of-life fuel cycle conditions. This assures subcriticality with not only the strongest fully withdrawn but at least an R + 0.28% Δk (or 0.38% Δk) margin beyond this condition. This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion following a scram signal. Any control rod that is immovable as a result of excessive friction or mechanical interference, or is known to be unscrammable, per Specification 3.3.C, is considered to be incapable of insertion following a scram signal. It is important to note that a control rod can be electrically immovable, but scrammable, and no increase in SHUTDOWN MARGIN is required for these control rods.

3.5 - LIMITING CONDITIONS FOR OPERATION

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[b] and 3[b].

ACTION:

1. For the core spray system:
   a. With one CS subsystem inoperable, provided that the LPCI subsystem is OPERABLE, restore the inoperable CS subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
   b. With both CS subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
2. For the LPCI subsystem:
   a. With one LPCI pump inoperable[d], provided that both CS subsystems are OPERABLE, restore the inoperable LPCI pump to OPERABLE status within 30 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

b. Three LPCI pumps together develop a flow of at least 14,500 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥20 psig.

c. The HPCI pump develops a flow of at least 5000 gpm against a system head corresponding to reactor vessel pressure, when steam is being supplied to the turbine between 920 and 1005 psig[c].

3. At least once per 18 months:
   a. For the CS system, the LPCI subsystem, and the HPCI system, verify each system/subsystem actuates on an actual or simulated automatic initiation signal. Actual injection of coolant into the reactor vessel may be excluded from this test.
   b. For the HPCI system, verifying that:
      1) The system develops a flow of ≥5000 gpm against a system head corresponding to reactor vessel pressure, when steam is being supplied to the turbine between 150 and 350 psig[c].

Otherwise, enter Specification 3.5.A, Action 2.c.

---

b The HPCI system and ADS are not required to be OPERABLE when reactor steam dome pressure is ≤150 psig.

d The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, both LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3.5 - LIMITING CONDITIONS FOR OPERATION

b. With the LPCI subsystem otherwise inoperable[d], provided that both CS subsystems are OPERABLE, restore the LPCI subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the LPCI subsystem and one or both CS subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

3. With the HPCI system inoperable, provided both CS subsystems, the LPCI subsystem, the ADS and the Isolation Condenser (IC) system are OPERABLE, restore the HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ 150 psig within the following 24 hours.

4. For the ADS:

a. With one of the above required ADS valves inoperable, provided the HPCI system, both CS subsystems and three LPCI pumps are OPERABLE, restore the inoperable ADS valve to OPERABLE

4.5 - SURVEILLANCE REQUIREMENTS

2) The pump suction is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber water level - high signal.

c. Performing a CHANNEL CALIBRATION of the CS and LPCI system discharge line "keep filled" alarm instrumentation.

d. Deleted.

4. At least once per 18 months for the ADS:

a. Verify the ADS actuates on an actual or simulated automatic initiation signal. Actual valve actuation may be excluded from this test.

b. Manually opening each ADS valve when the reactor steam dome pressure is ≥ 150 psig[c] and observing that either:

1) The turbine control valve or turbine bypass valve position responds accordingly, or

2) There is a corresponding change in the measured steam flow.

Otherwise, enter Specification 3.5.A, Action 2.c.

---

d The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, both LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3.6 - LIMITING CONDITIONS FOR OPERATION

J. Specific Activity

The specific activity of the reactor coolant shall be limited to ≤0.2 μCi/gram DOSE EQUIVALENT I-131.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3, with any main steam line not isolated.

ACTION:

1. ~~In OPERATIONAL MODE(s) 1, 2 or 3~~ With the specific activity of the reactor coolant >0.2 μCi/gram DOSE EQUIVALENT I-131 but ≤4.0 μCi/gram DOSE EQUIVALENT I-131~~, for more than 48 hours during one continuous time interval or >4.0 μCi/gram DOSE EQUIVALENT I-131, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.~~, determine DOSE EQUIVALENT I-131 once per 4 hours and restore DOSE EQUIVALENT I-131 to within limits within 48 hours[a].

2. ~~In OPERATIONAL MODE(s) 1, 2 or 3,~~ With the specific activity of the reactor coolant >0.2 μCi/gram DOSE EQUIVALENT I-131~~, perform the sampling and analysis requirements of Item 3.a of Table 4.6.J-1 until the specific activity of the reactor coolant is restored to within its limit.~~ for greater than 48 hours, or with the specific activity of the reactor coolant > 4.0 μCi/gram DOSE EQUIVALENT I-131, determine DOSE EQUIVALENT I-131 once per 4 hours, and isolate all main steam lines within 12 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

~~3. In OPERATIONAL MODE(s) 1 or 2, with:~~

4.6 - SURVEILLANCE REQUIREMENTS

J. Specific Activity

In OPERATIONAL MODE 1, the specific activity of the reactor coolant shall be ~~demonstrated to be within the limits by performance of the sampling and analysis program of Table 4.6.J-1.~~ verified to be ≤ 0.2 μCi/gram DOSE EQUIVALENT I-131 once per 7 days.

---

a. The provisions of Specification 3.0.D are not applicable.

3.6 - LIMITING CONDITIONS FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENTS

~~a. THERMAL POWER changed by more than 20% of RATED THERMAL POWER in 1 hour, or~~

~~b. The offgas level, prior to the holdup line, increased by >25,000 μCi/second in one hour during steady state operation at release rates <100,000 μCi/second, or~~

~~c. The offgas level, prior to the holdup line, increased by >15% in one hour during steady state operation at release rates >100,000 μCi/second,~~

~~Perform the sampling and analysis requirements of Item 3.b of Table 4.6.J-1 until the specific activity of the reactor coolant is restored to within its limit.~~

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TABLE 4.6.J-1

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

Type of Measurement and Analysis	Sample and Analysis Frequency	OPERATIONAL MODE(s) in Which Sample and Analysis Required
1. Gross Beta and Gamma Activity Determination	At least once per 72 hours	1, 2, 3
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per 31 days	1
3. Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION 2.	1[a], 2[a], 3[a]
	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION 3.	1[a], 2[a]
4. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for Xe-133, Xe-135 and Kr-88	At least once per 31 days	1

a Until the specific activity of the reactor coolant system is restored to within its limits.

3.6 - LIMITING CONDITIONS FOR OPERATION

O. Shutdown Cooling - HOT SHUTDOWN

Two[a] shutdown cooling (SDC) loops shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling loop shall be in operation[b][c], with each loop consisting of at least:

1. One OPERABLE SDC pump, and

2. One OPERABLE SDC heat exchanger.

APPLICABILITY:

OPERATIONAL MODE 3, with reactor vessel coolant temperature less than the SDC cut-in permissive setpoint.

ACTION:

1. With less than the above required SDC loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible. Within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable SDC loop. Be in at least COLD SHUTDOWN within 24 hours[d].

4.6 - SURVEILLANCE REQUIREMENTS

O. Shutdown Cooling - HOT SHUTDOWN

At least one SDC loop, one recirculation pump or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

---

a One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and in operation.

b A shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period provided the other loop is OPERABLE.

c The shutdown cooling loop may be removed from operation during hydrostatic testing.

d Whenever two or more SDC ~~subsystems~~ loops are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

3.7 - LIMITING CONDITIONS FOR OPERATION

2. With one or more reactor instrumentation line excess flow check valves inoperable, operation may continue and the provisions of Specification 3.0.C are not applicable, provided that within 4 hours either:

   a. The inoperable valve is restored to OPERABLE status, or

   b. The instrument line is isolated and the associated instrument is declared inoperable.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing at least one explosive squib from an ~~each~~ explosive valve such that each explosive squib ~~in~~ ~~each explosive valve~~ will be tested at least once per ~~36~~ 90 months, and initiating the removed explosive squib(s). The replacement charge for the exploded squib(s) shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable.

6. At the frequency specified by the Primary Containment Leakage Rate Testing Program, verify leakage for any one main steam line isolation valve when tested at P_t (25 psig) is ≤11.5 scfh.

3.7 - LIMITING CONDITIONS FOR OPERATION

. Standby Gas Treatment System

Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

ACTION:

1. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:
   a. In OPERATIONAL MODE(s) 1,2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
   b. In OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

~~2. Deleted.~~

4.7 - SURVEILLANCE REQUIREMENTS

P. Standby Gas Treatment System

Each standby gas treatment subsystem shall be demonstrated OPERABLE:

1. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters operating.
2. 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 subsystem by:
   a. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of $<1\%$ 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 4000 cfm $\pm 10\%$.
   b. 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 ASTM-D-3803-89, for a methyl iodide penetration of $<10\%$, when tested at 30°C and 70% relative humidity; and

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

~~3. Deleted.~~

4.7 - SURVEILLANCE REQUIREMENTS

c. Verifying a subsystem flow rate of 4000 cfm ±10% during system operation when tested in accordance with ANSI N510-1980.

720

3. After every ~~1440~~ 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 ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity.

4. At least once per 18 months by:

   a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 4000 cfm ±10%.

   b. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

      1) Manual initiation from the control room, and

      2) Simulated automatic initiation signal.

   c. Verifying that the heaters dissipate 30 ±3 kw when tested in accordance with ANSI N510-1989. This reading shall include the appropriate correction for variations from 480 volts at the bus.

3.7 - LIMITING CONDITIONS FOR OPERATION

2.

4. With both standby gas treatment subsystems otherwise inoperable in OPERATIONAL MODE(s) 1,2 or 3, restore at least one subsystem to OPERABLE status within one hour, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

5.

5. With both standby gas treatment subsystems inoperable in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

Move to previous pages

4.7 - SURVEILLANCE REQUIREMENTS

5. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of < 1% in accordance with ANSI N510-1980 while operating the system at a flow rate of 4000 cfm ± 10%.

6. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of < 1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ± 10%.

Move to previous pages

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION

2. In OPERATIONAL MODE *, with the control room emergency filtration system or the RCU inoperable, immediately suspend CORE ALTERATION(s), handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

3. The provisions of Specification 3.0.C are not applicable in OPERATIONAL MODE *.

4.8 - SURVEILLANCE REQUIREMENTS

b. 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 ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity; and

c. Verifying a system flow rate of 2000 scfm ±10% during system operation when tested in accordance with ANSI N510-1980.

4. After every 720 ~~1440~~ 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 ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity.

5. At least once per 18 months by:

a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 2000 scfm ±10%.

b. Verifying that the filter train starts and isolation dampers close on manual initiation from the control room.

* when handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION

4.8 - SURVEILLANCE REQUIREMENTS

c. Verifying that during the pressurization mode of operation, control room positive pressure is maintained at ≥1/8 inch water gauge relative to adjacent areas during system operation at a flow rate ≤2000 scfm.

d. Verifying that the heaters dissipate 12 ±1.2 kw when tested in accordance with ANSI N510-~~1980.~~ 1989 This reading shall include the appropriate correction for variations from 480 volts at the bus.

6. After each complete or partial replacement of an HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 2000 scfm ±10%.

7. After each complete or partial replacement of an charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at flow rate of 2000 scfm ±10%.

BASES

3/4.8.G Sealed Source Contamination

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources, including startup sources and fission detectors, 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.

3/4.8.H Explosive Gas Mixture

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the offgas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10CFR Part 50.

3/4.8.I Main Condenser Offgas Activity

Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50. The release rates are determined at a more expeditious frequency following the determination of an increase of greater than 50%, as indicated by the air ejector noble gas monitor, after factoring out increases due to changes in THERMAL POWER level and off-gas flow in the nominal steady-state fission gas release from the primary coolant.

3/4.8.J Liquid Holdup Tanks

2

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10CFR Part 20, Appendix B, Table ~~II~~, Column 2, ~~at the nearest potable water supply and the nearest surface water supply~~ in ~~an~~ unrestricted area. Recirculation of the tank contents for the purpose of reducing the radioactive content is not considered to be an addition of radioactive material to the tank.

3.9 - LIMITING CONDITIONS FOR OPERATION

E. Distribution - Operating

The following power distribution systems shall be energized:

1. A.C. power distribution, consisting of:

   a. Both Unit engineered safety features 4160 volt buses:

      1) For Unit 2, Nos. 23-1 and 24-1,

      2) For Unit 3, Nos. 33-1 and 34-1.

   b. Both Unit engineered safety features 480 volt buses:

      1) For Unit 2, Nos. 28 and 29,

      2) For Unit 3, Nos. 38 and 39.

   c. The Unit 120 volt Essential Service Bus and Instrument Bus.

2. 250 volt D.C. power distribution, consisting of:

   a. RB MCC Nos. 2 and 3, and

   b. TB MCC Nos. 2 and 3.

   a. For Unit 2, TB MCC 2 and RB MCC 2

   b. For Unit 3, TB MCC 3 and RB MCC 3.

3. For Unit 2, 125 volt D.C. power distribution, consisting of:

   a. TB Main Bus Nos. 2A-1and 3A,

   b. TB Res. Bus Nos. 2B and 2B-1,

   c. Reserve Bus No. 2, and

   d. RB Distribution Panel No. 2.

4.9 - SURVEILLANCE REQUIREMENTS

E. Distribution - Operating

Each of the required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

3.9 - LIMITING CONDITIONS FOR OPERATION

G. RPS Power Monitoring

Two Reactor Protection System (RPS) electric power monitoring CHANNEL(s) for each inservice RPS Motor Generator (MG) set or alternate power supply shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4[a] and 5.(a)

ACTION:

1. With one RPS electric power monitoring CHANNEL for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring CHANNEL to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.
2. With both RPS electric power monitoring CHANNEL(s) for an inservice RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring CHANNEL to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

4.9 - SURVEILLANCE REQUIREMENTS

G. RPS Power Monitoring

The specified RPS electric power monitoring CHANNEL(s) shall be determined OPERABLE:

1. By performance of a CHANNEL FUNCTIONAL TEST(b) each time the plant is in COLD SHUTDOWN for a period of more than 24 hours, unless performed in the previous 6 months.
2. At least once per 18 months by demonstrating the OPERABILITY of overvoltage, undervoltage, and underfrequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic, and output circuit breakers, and verifying the following setpoints:
   a. Overvoltage ≤129.6 volts AC
   b. Undervoltage ≥105.3 volts AC
   c. Underfrequency ≥55.4 Hz

b Only required to be performed prior to entering MODE 2 or 3 from MODE 4.

a With any control rod withdrawn.

BASES

redundancy in components and features not available, the plant must be placed in a condition for which the Limiting Condition for Operation does not apply.

The term verify as used toward A.C. electrical power sources means to administratively check by examining logs or other information to determine if certain components are out-of-service for pre-planned preventative maintenance, testing, or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

With one offsite circuit and one diesel generator inoperable, individual redundancy is lost in both the offsite and onsite electrical power system. Therefore, the allowable outage time is more limited. The time limit takes into account the capacity and capability of the remaining sources, reasonable time for repairs, and the low probability of a design basis event occurring during this period.

With both of the required offsite circuits inoperable, sufficient onsite A.C. sources are available to maintain the unit in a safe shutdown condition in the event of a design basis transient or accident. In fact, a simultaneous loss of offsite A.C. sources, a loss-of-coolant accident, and a worst-case single failure were postulated as a part of the design basis in the safety analysis. Thus, the allowable outage time provides a period of time to effect restoration of all or all but one of the offsite circuits commensurate with the importance of maintaining an A.C. electrical power system capable of meeting its design intent.

With two diesel generators inoperable there are no remaining standby A.C. sources. Thus, with an assumed loss of offsite electrical power, insufficient standby A.C. sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of A.C. power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown, which could result in grid instability and possibly a loss of total A.C. power. The allowable time to repair is severely restricted during this condition. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

~~Reporting requirements are included for a "problem emergency diesel generator," as recommended in Regulatory Guide 1.9, draft Revision 3. The required report should include a description of the failures, the underlying causes, and the corrective actions taken.~~

Surveillance Requirements are provided which assure proper circuit continuity for the offsite A.C. electrical power supply to the onsite distribution network and availability of offsite A.C. electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure distribution buses and loads are connected to their preferred power source. The frequency is adequate since breaker position is not likely to change without the operator being aware of it and because status is displayed in the control room. Should the action provisions of this specification require an increase in frequency, this Surveillance Requirement assures proper circuit continuity for the available offsite A.C. sources during periods of degradation and potential information on common cause failures that would otherwise go undiscovered.

BASES

Each battery of the D.C. electrical power systems is sized to start and carry the normal D.C. loads plus all D.C. loads required for safe shutdown on one unit and operations required to limit the consequences of a design basis event on the other unit for a period of 4 hours following loss of all A.C. sources. The battery chargers are sized to restore the battery to full charge under normal (non-emergency) load conditions. A normally disconnected alternate 125 volt battery is also provided as a backup for each normal battery. If both units are operating , the normal 125 volt battery must be returned to service within the specified time frame since the design configuration of the alternate battery circuit is susceptible to single failure and, hence, is not as reliable as the normal station circuit. During times when the other unit is in a Cold Shutdown or Refuel condition, an alternate 125 volt battery is available to replace a normal station 125 volt battery on a continuous basis to provide a second available power source. With the alternate 125 volt battery in service, ~~the normally open breaker~~ on the DC Reserve Bus is placed in the open position and posted, i.e., "tagged out."

With one of the required D.C. electrical power subsystems inoperable the remaining system has the capacity to support a safe shutdown and to mitigate an accident condition. However, a subsequent worst-case single failure would result in complete loss of ESF functions. Therefore, an allowed outage time is provided based on a reasonable time to assess plant status as a function of the inoperable D.C. electrical power subsystem and, if the D.C. electrical power subsystem is not restored to OPERABLE status, prepare to effect an orderly and safe plant shutdown.

Inoperable chargers do not necessarily indicate that the D.C. systems are not capable of performing their post-accident functions as long as the batteries are within their specified parameter limits. With both the required charger inoperable and the battery degraded, prompt action is required to assure an adequate D.C. power supply.

ACTION(s) are provided to delineate the measurements and time frames needed to continue to assure OPERABILITY of the Station batteries when battery parameters are outside their identified limits.

Battery surveillance requirements are based on the defined battery cell parameter values. Category A defines the normal parameter limit for each designated pilot cell in each battery. The pilot cells are the average cells in the battery based on previous test results. These cells are monitored closely as an indication of battery performance. Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out because of a degraded condition or for any other reason. Category B also defines allowable values for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category B allowable value, the assurance of sufficient capacity as described above no longer exists and the battery must be declared inoperable.

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations.

3.10 - LIMITING CONDITIONS FOR OPERATION

D. Decay Time

~~The reactor shall be subcritical for at least 24 hours.~~

~~APPLICABILITY:~~

~~OPERATIONAL MODE 5, during movement of irradiated fuel in the reactor pressure vessel.~~

~~ACTION:~~

~~With the reactor subcritical for less than 24 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.~~

4.10 - SURVEILLANCE REQUIREMENTS

D. ~~Decay Time~~

~~The reactor shall be determined to have been subcritical for at least 24 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.~~

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3.10 - LIMITING CONDITIONS FOR OPERATION

H. Water Level - Spent Fuel Storage Pool

The pool water level shall be maintained at a level of ≥ 33 feet.

APPLICABILITY:

Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.C are not applicable.

4.10 - SURVEILLANCE REQUIREMENTS

H. Water Level - Spent Fuel Storage Pool

The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

BASES

3/4.10.C Control Rod Position

The requirement that all control rods be inserted during other CORE ALTERATION(s) ensures that fuel will not be loaded into a cell without an inserted control rod.

3/4.10.D ~~Decay Time~~ Deleted

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.10.E Communications

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status regarding core reactivity conditions during movement of fuel within the reactor pressure vessel.

3/4.10.F DELETED

3/4.10.G Water Level - Reactor Vessel

3/4.10.H Water Level - Spent Fuel Storage Pool

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

ADMINISTRATIVE CONTROLS

6.8 PROCEDURES AND PROGRAMS

6.8.A Written procedures shall be established, implemented, and maintained covering the activities referenced below:

1. The applicable procedures recommended in Appendix A, of Regulatory Guide 1.33, Revision 2, February 1978,

2. The Emergency Operating Procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33,

3. Station Security Plan implementation,

4. Generating Station Emergency Response Plan implementation,

5. PROCESS CONTROL PROGRAM (PCP) implementation,

6. OFFSITE DOSE CALCULATION MANUAL (ODCM) implementation, and

7. Fire Protection Program implementation.

6.8.B Deleted.

6.8.C Deleted

(post-accident sampling of reactor coolant and containment atmosphere)

6.8.D The following programs shall be established, implemented, and maintained:

1. Reactor Coolant Sources Outside Primary Containment

This program provides controls to minimize leakage from those portions of systems outside primary containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include CS, HPCI, LPCI, IC, process sampling, containment monitoring, and standby gas treatment systems. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements, and

b. Leak test requirements for each system at a frequency of at least once per operating cycle.

4. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by station procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

no caps

no caps

a. Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

no caps

b. Limitations on the instantaneous concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to ten (10) times the concentration values in 10 CFR Part 20, Appendix B, Table 2, Column 2 to 10 CFR Part 20.1001 - 20.2402,

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

no caps

d. Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each Unit conforming to Appendix I to 10 CFR Part 50,

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

---

a A MEMBER OF THE PUBLIC shall be an individual in a CONTROLLED or UNRESTRICTED AREA. An individual is not a MEMBER OF THE PUBLIC during any period in which the individual receives an occupational dose.

b The CONTROLLED AREA shall be an area, outside of a RESTRICTED AREA but inside the SITE BOUNDARY, access to which can be limited by the licensee for any reason.

c An UNRESTRICTED AREA shall be any area, access to which is neither limited nor controlled by the licensee.

d RESTRICTED AREA shall be an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. RESTRICTED AREA(s) do not include areas used as residential quarters, but separate rooms in a residential building may be set apart as a RESTRICTED AREA.

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

f. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose conforming to Appendix I to 10 CFR Part 50,

g. Limitations on the dose rate resulting from radioactive materials released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY shall be limited to the following: no caps

a) For noble gases: less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

b) For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to a dose rate of 1500 mrem/yr to any organ.

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each Unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50, no caps no caps

i. Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with halflives greater than 8 days in gaseous effluents released from each Unit conforming to Appendix I to 10 CFR Part 50, no caps

j. Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190. no caps no caps

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Regional Administrator of the appropriate Regional Office of the NRC unless otherwise noted.

6.9.A. Routine Reports

1. Deleted

2. Annual Report

Annual reports covering the activities of the Unit for the previous calendar year, as described in this section shall be submitted prior to ~~March 1~~ May 1 of each year.

The reports required shall include:

a. Tabulation of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/year and their associated person rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter or TLD. Small exposures totaling 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.

b. The results of specific activity analysis in which the reactor coolant exceeded the limits of Specification 3.6.J. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the 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.

3. Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the Unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

4. Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the facility during the previous calendar year shall be submitted prior to April 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

5. Monthly Operating Report

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to safety valves or safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

6. CORE OPERATING LIMITS REPORT

   a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

      (1) The Control Rod Withdrawal Block Instrumentation for Table 3.2.E-1 of Specification 3.2.E.

      (2) The Average Planar Linear Heat Generation Rate (APLHGR) Limit for Specification 3.11.A.

      (3) The ~~Local~~ Steady State Linear Heat Generation Rate (LHGR) for Specification 3.11.D.

      (4) The Minimum Critical Power Operating Limit (including 20% scram insertion time) for Specification 3.11.C. This includes rated and off-rated flow conditions.

ADMINISTRATIVE CONTROLS

6.12 HIGH RADIATION AREA

6.12.A Pursuant to 10 CFR 20.1601(c), in lieu of the requirements of paragraph 20.1601 of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr at 30 cm (12 in.) 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)[a] (or equivalent document). 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 rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

3. An individual 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 in the RWP (or equivalent document).

(a)

(a) Health Physics personnel or personnel escorted by health physics personnel shall be exempt from the RWP issuance requirements during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

ADMINISTRATIVE CONTROLS

6.12.B In addition to the requirements of 6.12.A, areas accessible to personnel with radiation levels greater than 1000 mrem/hr at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall require the following:

1. Doors shall be locked to prevent unauthorized entry and shall not prevent individuals from leaving the area. In place of locking the door, direct or electronic surveillance that is capable of preventing unauthorized entry may be used. The keys shall be maintained under the administrative control of the Shift Manager on duty and/or health physics supervision.

2. Personnel access and exposure control requirements of activities being performed within these areas shall be specified by an approved RWP(or equivalent document).

3. Each person entering the area shall be provided with an alarming radiation monitoring device that continuously integrates the radiation dose rate (such as an electronic dosimeter.) Surveillance and radiation monitoring by a Radiation Protection Technician may be substituted for an alarming dosimeter.

4. ~~During emergency situations which involve personnel injury or actions taken to prevent major equipment damage, surveillance and radiation monitoring of the work area by a qualified individual may be substituted for the routine RWP (or equivalent document).~~

5. For individual HIGH RADIATION AREAS accessible to personnel with radiation levels of greater than 1000 mrem/h at 30 cm (12 in.) that are located within large areas where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual areas, then such individual areas shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

Deleted.

ADMINISTRATIVE CONTROLS

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.A Changes to the PCP:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

    a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

    b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

by

2. Shall become effective after approval of the Station Manager.

review and acceptance, including

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.A Changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

   b. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall become effective after approval ~~of~~ by the Station Manager .

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Report for the period of the report in which any change to the ODCM was made effective. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

review and acceptance, including

ATTACHMENT B
(continued)

MARKED-UP TSUP PAGES
QUAD CITIES NUCLEAR POWER STATION
LICENSE NOS. DPR-29 AND DPR-30

DEFINITIONS

SECTION PAGE

Section 1 DEFINITIONS

MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) - - - - 1-3

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION PAGE

1.0 DEFINITIONS

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Specification 6.8 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and ~~Semi-~~annual Radioactive Effluent Release Reports required by Specification 6.9.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE

An OPERATIONAL MODE, i.e., MODE, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1-2.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the UFSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

TABLE 4.1.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	Applicable OPERATIONAL MODES	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL[a] CALIBRATION
1. Intermediate Range Monitor:				
a. Neutron Flux - High	2 3, 4, 5	S[b] S	S/U[c], W[o] W[o]	E[o,g] E[o,g]
b. Inoperative	2, 3, 4, 5	NA	W[o]	NA
2. Average Power Range Monitor[f]:				
a. Setdown Neutron Flux - High	2 3, 5[m]	S[b] S	S/U[c], W[o] W	SA[o] SA
b. Flow Biased Neutron Flux - High	1	S, D[g]	W	W[d,e], SA
c. Fixed Neutron Flux - High	1	S	W	W[d], SA
d. Inoperative	1, 2, 3, 5[m]	NA	W	NA
3. Reactor Vessel Steam Dome Pressure - High	1, 2[l]	NA	M	Q
4. Reactor Vessel Water Level - Low	1, 2	D	M	E[h]
5. Main Steam Line Isolation Valve - Closure	1	NA	M	E
6. Main Steam Line Radiation - High	1, 2[l]	S	M	E[p]
7. Drywell Pressure - High	1, 2[n]	NA	M	Q

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least (½) decades during each startup after entering OPERATIONAL MODE 2 and the IRM and APRM channels shall be determined to overlap for at least (½) decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days. The weekly CHANNEL FUNCTIONAL TEST may be used to fulfill this requirement.

(d) This calibration shall consist of the adjustment of the APRM CHANNEL to conform, within 2% of RATED THERMAL POWER, to the power values calculated by a heat balance during OPERATIONAL MODE 1 when THERMAL POWER is ≥25% of RATED THERMAL POWER. This adjustment must be accomplished: a) within 2 hours if the APRM CHANNEL is indicating lower power values than the heat balance, or b) within 12 hours if the APRM CHANNEL is indicating higher power values than the heat balance. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

Any APRM CHANNEL gain adjustment made in compliance with Specification 3.11.B shall not be included in determining the above difference. This calibration is not required when THERMAL POWER is <25% of RATED THERMAL POWER. The provisions of Specification 4.0.D are not applicable.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 2000 effective full power hours (EFPH).

(g) ~~Verify measured recirculation loop flow to be greater than or equal to established recirculation loop flow at the existing pump speed.~~ Deleted.

(h) Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

(i) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.12.A.

(j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(k) This function may be bypassed, provided a control rod block is actuated, for reactor protection system reset in Refuel and Shutdown positions of the reactor mode switch.

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

(l) This function not required to be OPERABLE when THERMAL POWER is less than 45% of RATED THERMAL POWER.

(m) Required to be OPERABLE only prior to and during required SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(n) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(o) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for a period of 24 hours after entering OPERATIONAL MODE 2 or 3 when shutting down from OPERATIONAL MODE 1.

(p) A current source provides an instrument channel alignment every 3 months.

~~(q) The CHANNEL CALIBRATION surveillance requirements shall be performed if not performed within the previous seven days.~~

TABLE 3.2.A-1

ISOLATION ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[l]	Minimum CHANNEL(s) per TRIP SYSTEM[a]	Applicable OPERATIONAL MODE(s)	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low	≥144 inches	2	1, 2, 3	20
b. Drywell Pressure - High[d]	≤2.5 psig	2	1, 2, 3	20
c. Drywell Radiation - High	≤100 R/hr	1	1, 2, 3	23
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low[c,k]	≥144 inches	2	1, 2, 3 & *	24
b. Drywell Pressure - High[c,d,k]	≤2.5 psig	2	1, 2, 3	24
c. Reactor Building Ventilation Exhaust Radiation - High[c,k]	≤~~8~~ 5 mR/hr	2	1, 2, 3 & **	24
d. Refueling Floor Radiation - High[c,k]	≤100 mR/hr	2	1, 2, 3 & **	24
3. MAIN STEAM LINE (MSL) ISOLATION				
a. Reactor Vessel Water Level - Low Low	≥84 inches	2	1, 2, 3	21
b. MSL Tunnel Radiation - High[b]	≤15[h] x normal background	2	1, 2, 3	21
c. MSL Pressure - Low	≥825 psig	2	1	22
d. MSL Flow - High[k]	≤140% of rated	2/line	1, 2, 3	21
e. MSL Tunnel Temperature - High	≤200°F	2 of 4 in each of 2 sets	1, 2, 3	21

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function[i]	Applicable OPERATIONAL MODE(s)	ACTION
3. SOURCE RANGE MONITORS				
a. Detector not full in[b]	NA	3 2	2 5	51 51
b. Upscale[c]	$\leq 1 \times 10^5$ cps	3 2	2 5	51 51
c. Inoperative[c]	NA	3 2	2 5	51 51
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in[d]	NA	6	2, 5	51
b. Upscale	≤108/125 of full scale	6	2, 5	51
c. Inoperative	NA	6	2, 5	51
d. Downscale[d]	≥3/125 of full scale	6	2, 5	51

TABLE 3/4.2.E-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
1. ROD BLOCK MONITORS				
a. Upscale	NA	S/U[b,c], M[c]	Q	1[d]
b. Inoperative	NA	S/U[b,c], M[c]	NA	1[d]
c. Downscale	NA	S/U[b,c], M[c]	Q	1[d]
2. AVERAGE POWER RANGE MONITORS				
a. Flow Biased Neutron Flux - High				
1. Dual Recirculation Loop Operation	NA	S/U[b], M	SA	1
2. Single Recirculation Loop Operation	NA	S/U[b], M	SA	1
b. Inoperative	NA	S/U[b], M	NA	1, 2, 5[j]
c. Downscale	NA	S/U[b], M	SA	1
d. Startup Neutron Flux - High	NA	S/U[b], M	SA[k]	2, 5[j]
3. SOURCE RANGE MONITORS				
a. Detector not full in[f]	NA	S/U[b], W	E	2[i], 5
b. Upscale[g]	NA	S/U[b], W	E	2[i], 5
c. Inoperative[g]	NA	S/U[b], W	NA	2[i], 5

TABLE 4.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in~~[h]~~	NA	S/U[b], W	E	2[i], 5
b. Upscale	NA	S/U[b], W	E~~[k]~~	2[i], 5
c. Inoperative	NA	S/U[b], W	NA	2[i], 5
d. Downscale[h]	NA	S/U[b], W	E~~[k]~~	2[i], 5
5. SCRAM DISCHARGE VOLUME (SDV)				
a. Water Level - High	NA	Q	NA	1, 2, 5[e]
b. SDV Switch in Bypass	NA	E	NA	5[e]

TABLE 4.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) Within 7 days prior to startup.

(c) Includes reactor manual control "relay select matrix" system input.

(d) With THERMAL POWER ≥30% of RATED THERMAL POWER.

(e) With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(f) This function shall be automatically bypassed if detector count rate is >100 cps or the IRM channels are on range 3 or higher.

(g) This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

(h) This function shall be automatically bypassed when the IRM channels are on range 1.

(i) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for entry into the applicable OPERATIONAL MODE(s) from OPERATIONAL MODE 1 provided the surveillances are performed within 12 hours after such entry.

(j) Required to be OPERABLE only during SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

~~(k) The CHANNEL CALIBRATION surveillance requirements shall be performed within 12 hours upon each entry into any OPERATIONAL MODE(s) from OPERATIONAL MODE 1 if not performed within the previous seven days.~~

TABLE 3.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION

ACTION 60 - a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 61- With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:

a. Either restore the inoperable CHANNEL(s) to OPERABLE status within 7 days of the event, or

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.B within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 62- a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) one less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

~~b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1; and provided the high radiation sampling system (HRSS) combustible gas monitoring capability for the drywell is OPERABLE; restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.~~

b. ~~c.~~ With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1~~; and the HRSS combustible gas monitoring capability for the drywell inoperable;~~ restore at least one inoperable CHANNEL to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

TABLE 3/4.2.F-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. Reactor Vessel Pressure	M	E	1, 2
2. Reactor Vessel Water Level	M	E	1, 2
3 Torus Water Level	M	E	1, 2
4. Torus Water Temperature	M	E	1, 2
5. Drywell Pressure - Wide Range	M	E	1, 2
6. Drywell Pressure - Narrow Range	M	E	1, 2
7. Drywell Air Temperature	M	E	1, 2
8. Drywell ~~Hydrogen/~~Oxygen Concentration - Analyzer and Monitor	M	Q	1, 2
9. Drywell Hydrogen Concentration -Analyzer and Monitor	M	Q	1, 2
10. ~~9.~~ Safety & Relief Valve Position Indicators - Acoustic & Temperature	M	E	1, 2
11. ~~10.~~ (Source Range) Neutron Monitors	M	E[b]	1, 2
12. ~~11.~~ Drywell Radiation Monitors	M	E[a]	1, 2, 3
13. ~~12.~~ Torus Air Temperature	M	E	1, 2
14. ~~13.~~ Torus Pressure	M	E	1, 2

BASES

3/4.2.E Control Rod Block Actuation Instrumentation

The control rod block functions are provided to prevent excessive control rod withdrawal so that the MINIMUM CRITICAL POWER RATIO (MCPR) does not go below the MCPR fuel cladding integrity Safety Limit. During shutdown conditions, control rod block instrumentation initiates withdrawal blocks to ensure that all control rods remain inserted to prevent inadvertent criticality.

The trip logic for this function is one-out-of-n; e.g., any trip on one of the six average power range monitors (APRMs), eight intermediate range monitors (IRMs), or four source range monitors (SRMs), will result in a rod block. The minimum instrument CHANNEL requirements assure sufficient instrumentation to assure that the single failure criterion is met. The minimum instrument CHANNEL requirements for the rod block monitor may be reduced by one for a short period of time to allow for maintenance, testing, or calibration.

The APRM rod block function is flow-biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection, i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

In the REFUEL MODE during SHUTDOWN MARGIN demonstrations and the STARTUP/HOT STANDBY OPERATIONAL MODE, the APRM rod block function setpoint is significantly reduced to provide the same type of protection in the REFUEL and STARTUP/HOT STANDBY OPERATIONAL MODE(s) as the APRM flow-biased rod block does in the RUN OPERATIONAL MODE, i.e., prevents control rod withdrawal before a scram is reached.

The rod block monitor (RBM) function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error. The trip setting is flow-biased. At low power, the worst-case withdrawal of a single control rod without rod block action will not violate the fuel cladding integrity Safety Limit. Thus the RBM rod block function is not required below the specified power level. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the fuel cladding integrity Safety Limit.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of ten above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity Safety Limit.

A downscale indication on an APRM is an indication that the instrument has failed or is not sufficiently sensitive ~~enough~~. In either case, the instrument will not respond to changes in control rod motion, and the control rod motion is thus prevented.

The SRM rod blocks of low count rate and the detector not fully inserted assure that the SRMs are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram

3.3 - LIMITING CONDITIONS FOR OPERATION

D. Maximum Scram Insertion Times

The maximum scram insertion time of each control rod from the fully withdrawn position to 90% insertion, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

1. Declare the control rod(s) exceeding the above maximum scram insertion time inoperable, and

2. When operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of power operation — ALL CAPS

With the provisions of the ACTION above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

D. Maximum Scram Insertion Times

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators:

1. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER:

   a. following CORE ALTERATION(s), or

   b. after a reactor shutdown that is greater than 120 days,

2. For specifically affected individual control rods[a] following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of power operation

ALL CAPS

(a) The provisions of Specification 4.0.D are not applicable provided this surveillance is conducted prior to exceeding 40% of RATED THERMAL POWER.

3.3 - LIMITING CONDITIONS FOR OPERATION

N. Economic Generation Control (EGC) System

The economic generation control (EGC) system may be in operation with automatic flow control provided:

(1.) ~~a.~~ Core flow is within 65% to 100% of rated core flow, and

(2.) ~~b.~~ THERMAL POWER is ≥20% of RATED THERMAL POWER.

APPLICABILITY

OPERATIONAL MODE 1.

ACTION:

With core flow less than 65% or greater than 100% of rated core flow, or THERMAL POWER less than 20% of RATED THERMAL POWER, restore operation to within the limits within one hour. Otherwise, immediately remove the plant from EGC operation.

4.3 - SURVEILLANCE REQUIREMENTS

N. Economic Generation Control (EGC) System

The economic generation control system shall be demonstrated OPERABLE by verifying that core flow is within 65% to 100% of rated core flow and THERMAL POWER is ≥20% of RATED THERMAL POWER:

(1.) ~~a.~~ Prior to entry into EGC operation, and

(2.) ~~b.~~ At least once per 12 hours while operating in EGC.

BASES

3/4.3.A SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The SHUTDOWN MARGIN limitation is a restriction to be applied principally to a new refueling pattern. Satisfaction of the limitation must be determined at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. This determination is provided by core design calculations and administrative control of fuel loading patterns. These procedures include restrictions to allow only those intermediate fuel assembly configurations that have been shown to provide the required SHUTDOWN MARGIN.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed during xenon free conditions and adjusted to 68°F to accommodate the current moderator temperature. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.38% Δk/k or R + 0.28% Δk/k, as appropriate, with the strongest control rod fully withdrawn and all others fully inserted. Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of determination of the highest control rod worth, either analytically or by test. This is due to the reduced uncertainty in the SHUTDOWN MARGIN test when the highest worth control rod is determined by demonstration. When SHUTDOWN MARGIN is determined by calculations not associated with a test, additional margin must be added to the specified SHUTDOWN MARGIN limit to account for uncertainties in the calculation.

The value of R in units of % Δk/k is the difference between the calculated beginning-of-life core reactivity, at the beginning of the operating cycle, and the calculated value of maximum core reactivity at any time later in the operating cycle, where it would be greater than at the beginning. The value of R shall include the potential SHUTDOWN MARGIN loss assuming full B_4C settling in all inverted poison tubes present in the core. R must be a positive quantity or zero and a new value of R must be determined for each new fuel cycle.

Δk/k

The value of % Δk/k in the above expression is provided as a finite, demonstrable, subcriticality margin. This margin is verified using an in-sequence control rod withdrawal at the beginning-of-life fuel cycle conditions. This assures subcriticality with not only the strongest fully withdrawn but at least an R + 0.28% Δk (or 0.38% Δk) margin beyond this condition. This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion following a scram signal. Any control rod that is immovable as a result of excessive friction or mechanical interference, or is known to be unscrammable, per Specification 3.3.C, is considered to be incapable of insertion following a scram signal. It is important to note that a control rod can be electrically immovable, but scrammable, and no increase in SHUTDOWN MARGIN is required for these control rods.

3.5 - LIMITING CONDITIONS FOR OPERATION

4. The automatic depressurization system (ADS) with at least 5 OPERABLE ADS valves.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[b] and 3[b].

ACTION:

1. For the core spray system:

   a. With one CS subsystem inoperable, provided that the LPCI subsystem is OPERABLE, restore the inoperable CS subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. With both CS subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. For the LPCI subsystem:

   a. With one LPCI pump inoperable[f], provided that both CS subsystems are OPERABLE, restore the inoperable LPCI pump to OPERABLE status within 30 days,

4.5 - SURVEILLANCE REQUIREMENTS

2. Verifying that, when tested pursuant to Specification 4.0.E:

   a. The CS pump in each subsystem develop a flow of at least 4500 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥90 psig.

   b. Two LPCI pumps together develop a flow of at least 9,000 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥20 psig.

   c. The HPCI pump develops a flow of at least 5000 gpm against a system head corresponding to reactor vessel pressure, when steam is being supplied to the turbine between 920 and 1005 psig[c].

3. At least once per 18 months:

   a. For the CS system, the LPCI subsystem, and the HPCI system, verify each system/subsystem actuates on an actual or simulated automatic initiation signal. Actual injection of coolant into the reactor vessel may be excluded from this test.

Otherwise, enter Specification 3.5.A, Action 2.c.

---

b The HPCI system and ADS are not required to be OPERABLE when reactor steam dome pressure is ≤150 psig.

f The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, both LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3.5 - LIMITING CONDITIONS FOR OPERATION

or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the LPCI subsystem otherwise inoperable[f], provided that both CS subsystems are OPERABLE, restore the LPCI subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours[g]. e (d)

c. With the LPCI subsystem and one or both CS subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours[g].

3. With the HPCI system inoperable, provided both CS subsystems, the LPCI subsystem, the ADS and the Reactor Core Isolation Cooling (RCIC) system are OPERABLE, restore the HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

b. For the HPCI system, verifying that:

1) The system develops a flow of ≥5000 gpm against a system head corresponding to reactor vessel pressure, when steam is being supplied to the turbine between 150 and 180 psig[c].

2) The pump suction is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber water level - high signal.

c. Performing a CHANNEL CALIBRATION of the ECCS discharge line "keep filled" alarm instrumentation.

d. Deleted.

Otherwise, enter Specification 3.5.A, Action 2.c.

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f The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, <u>both</u> LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE.

g e Whenever the two required RHR SDC mode subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3.6 - LIMITING CONDITIONS FOR OPERATION

J. Specific Activity

The specific activity of the reactor coolant shall be limited to ≤0.2 μCi/gram DOSE EQUIVALENT I-131.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3, with any main steam line not isolated.

ACTION:

1. ~~In OPERATIONAL MODE(s) 1, 2 or 3~~ With the specific activity of the reactor coolant >0.2 μCi/gram DOSE EQUIVALENT I-131 but ≤4.0 μCi/gram DOSE EQUIVALENT I-131 ~~for more than 48 hours during one continuous time interval or >4.0 μCi/gram DOSE EQUIVALENT I-131, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours~~, determine DOSE EQUIVALENT I-131 once per 4 hours and restore DOSE EQUIVALENT I-131 to within limits within 48 hours [a].

2. ~~In OPERATIONAL MODE(s) 1, 2 or 3,~~ With the specific activity of the reactor coolant >0.2 μCi/gram DOSE EQUIVALENT I-131~~, perform the sampling and analysis requirements of Item 3.a of Table 4.6.J-1 until the specific activity of the reactor coolant is restored to within its limit.~~ for greater than 48 hours, or with the specific activity of the reactor coolant > 4.0 μCi/gram DOSE EQUIVALENT I-131, determine DOSE EQUIVALENT I-131 once per 4 hours, and isolate all main steam lines within 12 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

~~3. In OPERATIONAL MODE(s) 1 or 2, with:~~

4.6 - SURVEILLANCE REQUIREMENTS

J. Specific Activity

In OPERATIONAL MODE 1, The specific activity of the reactor coolant shall be ~~demonstrated to be within the limits by performance of the sampling and analysis program of Table 4.6.J-1~~ verified to be ≤ 0.2 μCi/gram DOSE EQUIVALENT I-131 once per 7 days.

---

a The provisions of Specification 3.0.D are not applicable.

3.6 - LIMITING CONDITIONS FOR OPERATION

a. THERMAL POWER changed by more than 20% of RATED THERMAL POWER in 1 hour, or

b. The offgas level, prior to the holdup line, increased by > 25,000 μCi/second in one hour during steady state operation at release rates ≤ 100,000 μCi/second, or

c. The offgas level, prior to the holdup line, increased by > 15% in one hour during steady state operation at release rates > 100,000 μCi/second,

Perform the sampling and analysis requirements of Item 3.b of Table 4.6.J-1 until the specific activity of the reactor coolant is restored to within its limit.

e

4.6 - SURVEILLANCE REQUIREMENTS

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TABLE 4.6.J-1

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

Type of Measurement and Analysis	Sample and Analysis Frequency	OPERATIONAL MODE(s) in Which Sample and Analysis Required
1. Gross Beta and Gamma Activity Determination	At least once per 72 hours	1, 2, 3
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per 31 days	1
3. Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION 2.	1[a], 2[a], 3[a]
	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION 3.	1[a], 2[a]
4. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for Xe-133, Xe-135 and Kr-88	At least once per 31 days	1

a Until the specific activity of the reactor coolant system is restored to within its limits.

3.7 - LIMITING CONDITIONS FOR OPERATION

2. With one or more reactor instrumentation line excess flow check valves inoperable, operation may continue and the provisions of Specification 3.0.C are not applicable, provided that within 4 hours either:

   a. The inoperable valve is restored to OPERABLE status, or

   b. The instrument line is isolated and the associated instrument is declared inoperable.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing at least one explosive squib from ~~each~~ explosive valve such that each explosive squib or ~~each explosive valve~~ will be tested at least once per ~~36~~ months, and initiating the removed explosive squib(s). The replacement charge for the exploded squib(s) shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable.

24

90

6. At the frequency specified by the Primary Containment Leakage Rate Testing Program, verify leakage for any one main steam line isolation valve when tested at P_t (25 psig) is ≤11.5 scfh.

3.7 - LIMITING CONDITIONS FOR OPERATION

P. Standby Gas Treatment System

Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

ACTION:

1. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

   a. In OPERATIONAL MODE(s) 1,2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. In OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

~~2. Deleted.~~ 2

4.7 - SURVEILLANCE REQUIREMENTS

P. Standby Gas Treatment System

Each standby gas treatment subsystem shall be demonstrated OPERABLE:

1. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters operating.

2. 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 subsystem by:

   a. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of $<1\%$ 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 4000 cfm $\pm 10\%$.

   b. 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 ASTM-D-3803-89, for a methyl iodide penetration of $<10\%$, when tested at 30°C and 70% relative humidity; and

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION	4.7 - SURVEILLANCE REQUIREMENTS

~~3. Deleted.~~

c. Verifying a subsystem flow rate of 4000 cfm ±10% during system operation when tested in accordance with ANSI N510-1980.

3. After every 1440 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 ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity.

4. At least once per 18 months by:

   a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 4000 cfm ±10%.

   b. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

      1) Manual initiation from the control room, and

      2) Simulated automatic initiation signal.

   c. Verifying that the heaters dissipate 30 ±3 kw when tested in accordance with ANSI N510-1989. This reading shall include the appropriate correction for variations from 480 volts at the bus.

3.7 - LIMITING CONDITIONS FOR OPERATION

2. 4. e. With both standby gas treatment subsystems ~~otherwise~~ inoperable in OPERATIONAL MODE(s) 1,2 or 3, restore at least one subsystem to OPERABLE status within one hour, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. 5. e. With both standby gas treatment subsystems inoperable in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

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4.7 - SURVEILLANCE REQUIREMENTS

5. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of <1% in accordance with ANSI N510-1980 while operating the system at a flow rate of 4000 cfm ±10%.

6. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of <1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ±10%.

move up to previous pages

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION

2. In OPERATIONAL MODE *, with the control room emergency filtration system or the RCU inoperable, immediately suspend CORE ALTERATION(s), handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

3. The provisions of Specification 3.0.C are not applicable in OPERATIONAL MODE *.

4.8 - SURVEILLANCE REQUIREMENTS

b. 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 ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity; and

c. Verifying a system flow rate of 2000 scfm ±10% during system operation when tested in accordance with ANSI N510-1980.

4. After every ~~1440~~ 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 ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity.

5. At least once per 18 months by:

a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 2000 scfm ±10%.

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION

4.8 - SURVEILLANCE REQUIREMENTS

b. Verifying that the isolation dampers close on each of the following signals:

   1) Manual initiation from the control room, and

   2) Simulated automatic isolation signal.

c. Verifying that during the pressurization mode of operation, control room positive pressure is maintained at ≥1/8 inch water gauge relative to adjacent areas during system operation at a flow rate ≤2000 scfm.

d. Verifying that the heaters dissipate 12 ±1.2 kw when tested in accordance with ANSI N510-1980. This reading shall include the appropriate correction for variations from 480 volts at the bus. 1989

6. After each complete or partial replacement of an HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 2000 scfm ±10%.

7. After each complete or partial replacement of an charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at flow rate of 2000 scfm ±10%.

3.9 - LIMITING CONDITIONS FOR OPERATION

E. Distribution - Operating

The following power distribution systems shall be energized:

1. A.C. power distribution, consisting of:

   a. Both Unit engineered safety features 4160 volt buses:

      1) For Unit 1, Nos. 13-1 and 14-1,

      2) For Unit 2, Nos. 23-1 and 24-1.

   b. Both Unit engineered safety features 480 volt buses:

      1) For Unit 1, Nos. 18 and 19,

      2) For Unit 2, Nos. 28 and 29, and

   c. The Unit 120 volt Essential Service Bus and Instrument Bus.

2. 250 volt D.C. power distribution, consisting of:

   a. ~~TB MCC Nos. 1 and 2, and~~

   1) For Unit 1, TB MCC No. 1,
   2) For Unit 2, TB MCC No. 2.

   b. 1) For Unit 1, RB MCC Nos. 1A and 1B,

      2) For Unit 2, RB MCC Nos. 2A and 2B.

3. For Unit 1, 125 volt D.C. power distribution, consisting of:

   a. TB Main Bus Nos. 1A, 1A-1 and 2A,

   b. TB Reserve Bus Nos. 1B and 1B-1, and

   c. RB Distribution Panel No. 1.

4.9 - SURVEILLANCE REQUIREMENTS

E. Distribution - Operating

Each of the required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

3.9 - LIMITING CONDITIONS FOR OPERATION

G. RPS Power Monitoring

Two Reactor Protection System (RPS) electric power monitoring CHANNEL(s) for each inservice RPS Motor Generator (MG) set or alternate power supply shall be OPERABLE.

(a)

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4[a] and 5.

ACTION:

1. With one RPS electric power monitoring CHANNEL for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring CHANNEL to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.
2. With both RPS electric power monitoring CHANNEL(s) for an inservice RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring CHANNEL to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

4.9 - SURVEILLANCE REQUIREMENTS

G. RPS Power Monitoring

The specified RPS electric power monitoring CHANNEL(s) shall be determined OPERABLE:

(b)

1. By performance of a CHANNEL FUNCTIONAL TEST each time the plant is in COLD SHUTDOWN for a period of more than 24 hours, unless performed in the previous 6 months.
2. At least once per 18 months by demonstrating the OPERABILITY of overvoltage, undervoltage, and underfrequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic, and output circuit breakers, and verifying the following setpoints:
   a. Overvoltage ≤129.6 volts AC
   b. Undervoltage ≥105.3 volts AC
   c. Underfrequency ≥55.4 Hz

b Only required to be performed prior to entering Mode 2 or 3 from MODE 4.

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a With any control rod withdrawn.

BASES

With suitable redundancy in components and features not available, the plant must be placed in a condition for which the Limiting Condition for Operation does not apply.

The term verify as used toward A.C. electrical power sources means to administratively check by examining logs or other information to determine if certain components are out-of-service for pre-planned preventative maintenance, testing, or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

With one offsite circuit and one diesel generator inoperable, individual redundancy is lost in both the offsite and onsite electrical power system. Therefore, the allowable outage time is more limited. The time limit takes into account the capacity and capability of the remaining sources, reasonable time for repairs, and the low probability of a design basis event occurring during this period.

With both of the required offsite circuits inoperable, sufficient onsite A.C. sources are available to maintain the unit in a safe shutdown condition in the event of a design basis transient or accident. In fact, a simultaneous loss of offsite A.C. sources, a loss-of-coolant accident, and a worst-case single failure were postulated as a part of the design basis in the safety analysis. Thus, the allowable outage time provides a period of time to effect restoration of all or all but one of the offsite circuits commensurate with the importance of maintaining an A.C. electrical power system capable of meeting its design intent.

With two diesel generators inoperable there are no remaining standby A.C. sources. Thus, with an assumed loss of offsite electrical power, insufficient standby A.C. sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of A.C. power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown, which could result in grid instability and possibly a loss of total A.C. power. The allowable time to repair is severely restricted during this condition. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

~~Reporting requirements are included for a "problem emergency diesel generator," as recommended in Regulatory Guide 1.9, draft Revision 3. The required report should include a description of the failures, the underlying causes, and the corrective actions taken.~~

Surveillance Requirements are provided which assure proper circuit continuity for the offsite A.C. electrical power supply to the onsite distribution network and availability of offsite A.C. electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure distribution buses and loads are connected to their preferred power source. The frequency is adequate since breaker position is not likely to change without the operator being aware of it and because status is displayed in the control room. Should the action provisions of this specification require an increase in frequency, this Surveillance Requirement assures proper circuit continuity for the available offsite A.C. sources during periods of degradation and potential information on common cause failures that would otherwise go undiscovered.

BASES

Each battery of the D.C. electrical power systems is sized to start and carry the normal D.C. loads plus all D.C. loads required for safe shutdown on one unit and operations required to limit the consequences of a design basis event on the other unit for a period of 4 hours following loss of all A.C. sources. The battery chargers are sized to restore the battery to full charge under normal (non-emergency) load conditions. A normally disconnected alternate 125 volt battery is also provided as a backup for each normal battery. If both units are operating , the normal 125 volt battery must be returned to service within the specified time frame since the design configuration of the alternate battery circuit is susceptible to single failure and, hence, is not as reliable as the normal station circuit. During times when the other unit is in a Cold Shutdown or Refuel condition, an alternate 125 volt battery is available to replace a normal station 125 volt battery on a continuous basis to provide a second available power source. ~~With the alternate 125 volt battery in service, the normally open breaker on the DC Reserve Bus is placed in the open position and posted, i.e., "tagged out."~~

With one of the required D.C. electrical power subsystems inoperable the remaining system has the capacity to support a safe shutdown and to mitigate an accident condition. However, a subsequent worst-case single failure would result in complete loss of ESF functions. Therefore, an allowed outage time is provided based on a reasonable time to assess plant status as a function of the inoperable D.C. electrical power subsystem and, if the D.C. electrical power subsystem is not restored to OPERABLE status, prepare to effect an orderly and safe plant shutdown.

Inoperable chargers do not necessarily indicate that the D.C. systems are not capable of performing their post-accident functions as long as the batteries are within their specified parameter limits. With both the required charger inoperable and the battery degraded, prompt action is required to assure an adequate D.C. power supply.

ACTION(s) are provided to delineate the measurements and time frames needed to continue to assure OPERABILITY of the Station batteries when battery parameters are outside their identified limits.

Battery surveillance requirements are based on the defined battery cell parameter values. Category A defines the normal parameter limit for each designated pilot cell in each battery. The pilot cells are the average cells in the battery based on previous test results. These cells are monitored closely as an indication of battery performance. Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out because of a degraded condition or for any other reason. Category B also defines allowable values for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category B allowable value, the assurance of sufficient capacity as described above no longer exists and the battery must be declared inoperable.

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations.

3.10 - LIMITING CONDITIONS FOR OPERATION

D. ~~Decay Time~~

The reactor shall be subcritical for at least 24 hours.

APPLICABILITY:

OPERATIONAL MODE 5, during movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 24 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

4.10 - SURVEILLANCE REQUIREMENTS

D. ~~Decay Time~~

The reactor shall be determined to have been subcritical for at least 24 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

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3.10 - LIMITING CONDITIONS FOR OPERATION

H. Water Level - Spent Fuel Storage Pool

The pool water level shall be maintained at a level of 33 feet.

≥

APPLICABILITY:

Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.C are not applicable.

4.10 - SURVEILLANCE REQUIREMENTS

H. Water Level - Spent Fuel Storage Pool

The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

3.10 - LIMITING CONDITIONS FOR OPERATION

L. Residual Heat Removal and Coolant Circulation - Low Water Level

Two shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE, with each loop consisting of at least:

1. One OPERABLE RHR pump, and
2. One OPERABLE RHR heat exchanger.

APPLICABILITY:

OPERATIONAL MODE 5, when irradiated fuel is in the reactor vessel and the water level is <23 feet above the top of the reactor pressure vessel flange.

ACTION:

With less than the above required shutdown cooling mode loops of the RHR system OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the OPERABILITY of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

4.10 - SURVEILLANCE REQUIREMENTS

L. Residual Heat Removal and Coolant Circulation - Low Water Level

1. At least one shutdown cooling mode loop of the RHR system shall be verified to be capable of circulating reactor coolant at least once per 12 hours.
2. Monitor the reactor coolant temperature at least once per hour.

BASES

3/4.10.C Control Rod Position

The requirement that all control rods be inserted during other CORE ALTERATION(s) ensures that fuel will not be loaded into a cell without an inserted control rod.

3/4.10.D ~~Decay Time~~ DELETED

~~The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.~~

3/4.10.E Communications

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status regarding core reactivity conditions during movement of fuel within the reactor pressure vessel.

3/4.10.F DELETED

3/4.10.G Water Level - Reactor Vessel

3/4.10.H Water Level - Spent Fuel Storage Pool

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

6.8 PROCEDURES AND PROGRAMS

6.8.A Written procedures shall be established, implemented, and maintained covering the activities referenced below:

1. The applicable procedures recommended in Appendix A, of Regulatory Guide 1.33, Revision 2, February 1978,
2. The Emergency Operating Procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33,
3. Station Security Plan implementation,
4. Generating Station Emergency Response Plan implementation,
5. PROCESS CONTROL PROGRAM (PCP) implementation,
6. OFFSITE DOSE CALCULATION MANUAL (ODCM) implementation, and
7. Fire Protection Program implementation.

6.8.B Deleted

6.8.C Deleted

(post-accident sampling of reactor coolant and containment atmosphere)

6.8.D The following programs shall be established, implemented, and maintained:

1. Reactor Coolant Sources Outside Primary Containment

   This program provides controls to minimize leakage from those portions of systems outside primary containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include CS, HPCI, LPCI, RCIC, process sampling, containment monitoring, and standby gas treatment systems. The program shall include the following:

   a. Preventive maintenance and periodic visual inspection requirements, and

   b. Leak test requirements for each system at a frequency of at least once per operating cycle.

4. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC ~~(a)(b)(c)(d)(e)~~ from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by station procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements: no caps

no caps

a. Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

no caps

b. Limitations on the instantaneous concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to ten (10) times the concentration values in 10 CFR Part 20, Appendix B, Table 2, Column 2 to 10 CFR Part 20.1001 - 20.2402,

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM, no caps

d. Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each Unit conforming to Appendix I to 10 CFR Part 50,

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

---

a A MEMBER OF THE PUBLIC shall be an individual in a CONTROLLED or UNRESTRICTED AREA. An individual is not a MEMBER OF THE PUBLIC during any period in which the individual receives an occupational dose.

b The CONTROLLED AREA shall be an area, outside of a RESTRICTED AREA but inside the SITE BOUNDARY, access to which can be limited by the licensee for any reason.

c An UNRESTRICTED AREA shall be any area, access to which is neither limited nor controlled by the licensee.

d RESTRICTED AREA shall be an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. RESTRICTED AREA(s) do not include areas used as residential quarters, but separate rooms in a residential building may be set apart as a RESTRICTED AREA.

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

f. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose conforming to Appendix I to 10 CFR Part 50,

g. Limitations on the dose rate resulting from radioactive materials released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY shall be limited to the following: no caps

a) For noble gases: less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

b) For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to a dose rate of 1500 mrem/yr to any organ.

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each Unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50, no caps no caps

i. Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with halflives greater than 8 days in gaseous effluents released from each Unit conforming to Appendix I to 10 CFR Part 50, no caps

j. Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190. no caps no caps

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Regional Administrator of the appropriate Regional Office of the NRC unless otherwise noted.

6.9.A. Routine Reports

1. Deleted

2. Annual Report

   Annual reports covering the activities of the Unit for the previous calendar year, as described in this section shall be submitted prior to ~~March 1,~~ of each year.

   May 1

ADMINISTRATIVE CONTROLS

6.12 HIGH RADIATION AREA

6.12.A Pursuant to 10 CFR 20.1601(c), in lieu of the requirements of paragraph 20.1601 of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr at 30 cm (12 in.) 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)(a) (or equivalent document). 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 rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

3. An individual 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 in the RWP (or equivalent document).

6.12.B In addition to the requirements of 6.12.A, above, areas accessible to personnel with radiation levels greater than 1000 mrem/hr at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall require the following:

1. Doors shall be locked to prevent unauthorized entry and shall not prevent individuals from leaving the area. In place of locking the door, direct or electronic surveillance that is capable of preventing unauthorized entry may be used. The keys shall be maintained under the administrative control of the Shift Engineer on duty and/or health physics supervision.

2. Personnel access and exposure control requirements of activities being performed within these areas shall be specified by an approved RWP (or equivalent document).

(a) Health Physics personnel or personnel escorted by health physics personnel shall be exempt from the RWP issuance requirements during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

ADMINISTRATIVE CONTROLS

3. Each person entering the area shall be provided with an alarming radiation monitoring device that continuously integrates the radiation dose rate (such as an electronic dosimeter.) Surveillance and radiation monitoring by a Radiation Protection Technician may be substituted for an alarming dosimeter.

4. ~~During emergency situations which involve personnel injury or actions taken to prevent major equipment damage, surveillance and radiation monitoring of the work area by a qualified individual may be substituted for the routine RWP procedure (or equivalent document).~~

5. For individual HIGH RADIATION AREAS accessible to personnel with radiation levels of greater than 1000 mrem/h at 30 cm (12 in.) that are located within large areas where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual areas, then such individual areas shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

Deleted.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.A Changes to the PCP:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

   b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

by

2. Shall become effective after approval ~~of~~ the Station Manager.

review and acceptance, including

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.A Changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

   b. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall become effective after approval ~~of~~ by the Station Manager.

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Report for the period of the report in which any change to the ODCM was made effective. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

review and acceptance, including

ATTACHMENT C

REVISED TSUP PAGES FOR
DRESDEN NUCLEAR POWER STATION
LICENSE NOS. DPR-19 AND DPR-25

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION PAGE

1.0 DEFINITIONS

MINIMUM CRITICAL POWER RATIO (MCPR)

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Specification 6.8 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specification 6.9.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE

An OPERATIONAL MODE, i.e., MODE, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1-2.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the UFSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

TABLE 4.1.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	Applicable OPERATIONAL MODES	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL[a] CALIBRATION
1. Intermediate Range Monitor:				
a. Neutron Flux - High	2	S[b]	S/U[c], W[o]	E[o]
	3, 4, 5	S	W[o]	E[o]
b. Inoperative	2, 3, 4, 5	NA	W[o]	NA
2. Average Power Range Monitor[f]:				
a. Setdown Neutron Flux - High	2	S[b]	S/U[c], W[o]	SA[o]
	3, 5[m]	S	W[o]	SA[o]
b. Flow Biased Neutron Flux - High	1	S, D	W	W[d,e], SA
c. Fixed Neutron Flux - High	1	S	W	W[d], SA
d. Inoperative	1, 2, 3, 5[m]	NA	W	NA
3. Reactor Vessel Steam Dome Pressure - High	1, 2[i]	NA	M	Q
4. Reactor Vessel Water Level - Low	1, 2	D	M	E[h]
5. Main Steam Line Isolation Valve - Closure	1, 2[p]	NA	M	E
6. Main Steam Line Radiation - High	1, 2[i]	S	M	E[q]
7. Drywell Pressure - High	1, 2[n]	NA	M	Q

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least (½) decades during each startup after entering OPERATIONAL MODE 2 and the IRM and APRM channels shall be determined to overlap for at least (½) decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days. The weekly CHANNEL FUNCTIONAL TEST may be used to fulfill this requirement.

(d) This calibration shall consist of the adjustment of the APRM CHANNEL to conform, within 2% of RATED THERMAL POWER, to the power values calculated by a heat balance during OPERATIONAL MODE 1 when THERMAL POWER is ≥25% of RATED THERMAL POWER. This adjustment must be accomplished: a) within 2 hours if the APRM CHANNEL is indicating lower power values than the heat balance, or b) within 12 hours if the APRM CHANNEL is indicating higher power values than the heat balance. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

Any APRM CHANNEL gain adjustment made in compliance with Specification 3.11.B shall not be included in determining the above difference. This calibration is not required when THERMAL POWER is <25% of RATED THERMAL POWER. The provisions of Specification 4.0.D are not applicable.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 2000 effective full power hours (EFPH).

(g) Deleted.

(h) Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

(i) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.12.A.

(j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(k) This function may be bypassed, provided a control rod block is actuated, for reactor protection system reset in Refuel and Shutdown positions of the reactor mode switch.

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

(l) This function not required to be OPERABLE when THERMAL POWER is less than 45% of RATED THERMAL POWER.

(m) Required to be OPERABLE only prior to and during required SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(n) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(o) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for a period of 24 hours after entering OPERATIONAL MODE 2 or 3 when shutting down from OPERATIONAL MODE 1.

(p) This function is not required to be OPERABLE when reactor pressure is less than 600 psig.

(q) A current source provides an instrument channel alignment every 3 months.

TABLE 4.2.B-1

ECCS ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. CORE SPRAY (CS) SYSTEM				
a. Reactor Vessel Water Level - Low Low	S	M	Q[f]	1, 2, 3, 4[b], 5[b]
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Reactor Vessel Pressure - Low (Permissive)	NA	M	Q	1, 2, 3, 4[b], 5[b]
d. CS Pump Discharge Flow - Low (Bypass)	NA	Q	Q[e]	1, 2, 3, 4[b], 5[b]
2. LOW PRESSURE COOLANT INJECTION (LPCI) SUBSYSTEM				
a. Reactor Vessel Water Level - Low Low	S	M	Q[f]	1, 2, 3, 4[b], 5[b]
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Reactor Vessel Pressure - Low (Permissive)	NA	M	Q	1, 2, 3, 4[b], 5[b]
d. LPCI Pump Discharge Flow - Low (Bypass)	NA	Q	Q[e]	1, 2, 3, 4[b], 5[b]
3. HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM[a]				
a. Reactor Vessel Water Level - Low Low	S	M	Q[f]	1, 2, 3
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Condensate Storage Tank Level - Low	NA	M	NA	1, 2, 3
d. Suppression Chamber Water Level - High	NA	M	NA	1, 2, 3
e. Reactor Vessel Water Level - High (Trip)	NA	M	Q[f]	1, 2, 3
f. HPCI Pump Discharge Flow - Low (Bypass)	NA	Q	Q	1, 2, 3
g. Manual Initiation	NA	E	NA	1, 2, 3

TABLE 4.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
4. AUTOMATIC DEPRESSURIZATION SYSTEM[a]				
a. Reactor Vessel Water Level - Low Low	S	M	Q[f]	1, 2, 3
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Initiation Timer	NA	E	E	1, 2, 3
d. Low Low Level Timer	NA	E	E	1, 2, 3
e. CS Pump Discharge Pressure - High (Permissive)	NA	M	Q	1, 2, 3
f. LPCI Pump Discharge Pressure - High (Permissive)	NA	M	Q	1, 2, 3
5. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	NA	E	E	1, 2, 3, 4[c], 5[c]
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	NA	E	E	1, 2, 3, 4[c], 5[c]

TABLE 4.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Not required to be OPERABLE when reactor steam dome pressure is ≤150 psig.

(b) When the system is required to be OPERABLE per Specification 3.5.B.

(c) Required when the associated diesel generator is required to be OPERABLE per Specification 3.9.B.

(d) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(e) Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

(f) Unit 2 transmitters are calibrated once per 18 months. Unit 2 trip units and Unit 3 level switches are calibrated at the frequency identified in the table.

TABLE 4.2.C-1

ATWS - RPT INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Water Level - Low Low	S	Q	E[a]
2. Reactor Vessel Pressure - High	S	Q	E[a]

---

a Trip units are calibrated at least once per 92 days and transmitters are calibrated at the frequency identified in the table.

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function[i]	Applicable OPERATIONAL MODE(s)	ACTION
3. SOURCE RANGE MONITORS				
a. Detector not full in[b]	NA	3 2	2 5	51 51
b. Upscale[c]	$\leq 1 \times 10^5$ cps	3 2	2 5	51 51
c. Inoperative[c]	NA	3 2	2 5	51 51
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	6	2, 5	51
b. Upscale	≤108/125 of full scale	6	2, 5	51
c. Inoperative	NA	6	2, 5	51
d. Downscale[d]	≥5/125 of full scale	6	2, 5	51

TABLE 4.2.E-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
1. ROD BLOCK MONITORS				
a. Upscale	NA	S/U[b,c], M[c]	Q	1[d]
b. Inoperative	NA	S/U[b,c], M[c]	NA	1[d]
c. Downscale	NA	S/U[b,c], M[c]	Q	1[d]
2. AVERAGE POWER RANGE MONITORS				
a. Flow Biased Neutron Flux - High				
1. Dual Recirculation Loop Operation	NA	S/U[b], M	SA	1
2. Single Recirculation Loop Operation	NA	S/U[b], M	SA	1
b. Inoperative	NA	S/U[b], M	NA	1, 2, 5[j]
c. Downscale	NA	S/U[b], M	Q	1
d. Startup Neutron Flux - High	NA	S/U[b], M	SA	2, 5[j]
3. SOURCE RANGE MONITORS				
a. Detector not full in[f]	NA	S/U[b], W	E	2,[i] 5
b. Upscale[g]	NA	S/U[b], W	E[k]	2,[i] 5
c. Inoperative[g]	NA	S/U[b], W	NA	2,[i] 5

TABLE 4.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U[b], W	E	2[i], 5
b. Upscale	NA	S/U[b], W	E	2[i], 5
c. Inoperative	NA	S/U[b], W	NA	2[i], 5
d. Downscale[h]	NA	S/U[b], W	E	2[i], 5
5. SCRAM DISCHARGE VOLUME (SDV)				
a. Water Level - High	NA	Q	NA	1, 2, 5[e]
b. SDV Switch in Bypass	NA	E	NA	5[e]

TABLE 4.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) Within 7 days prior to startup.

(c) Includes reactor manual control "relay select matrix" system input.

(d) With THERMAL POWER ≥30% of RATED THERMAL POWER.

(e) With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(f) This function shall be automatically bypassed if detector count rate is >100 cps or the IRM channels are on range 3 or higher.

(g) This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

(h) This function shall be automatically bypassed when the IRM channels are on range 1.

(i) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for entry into the applicable OPERATIONAL MODE(s) from OPERATIONAL MODE 1 provided the surveillances are performed within 12 hours after such entry

(j) Required to be OPERABLE only during SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

TABLE 3.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION

ACTION 60 -

a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 61- With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:

a. Either restore the inoperable CHANNEL(s) to OPERABLE status within 7 days of the event, or

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.B within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 62-

a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) one less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1; restore at least one inoperable CHANNEL to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

TABLE 4.2.F-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. Reactor Vessel Pressure	M	SA	1, 2
2. Reactor Vessel Water Level	M	SA [d]	1, 2
3 Torus Water Level	M	A	1, 2
4. Torus Water Temperature	M	A	1, 2
5. Drywell Pressure - Wide Range	M	E	1, 2
6. Drywell Pressure - Narrow Range	M	Q	1, 2
7. Drywell Air Temperature	M	E	1, 2
8. Drywell Oxygen Concentration - Analyzer and Monitor	M	Q	1, 2
9. Drywell Hydrogen Concentration - Analyzer and Monitor	M	Q	1, 2
10. Safety/Relief Valve Position Indicators - Acoustic & Temperature	M[c]	E	1, 2
11. (Source Range) Neutron Monitors	M	Q[b]	1, 2, 3
12. Drywell Radiation Monitors	M	E[a]	1, 2
13. Torus Pressure	M	Q	1, 2

TABLE 4.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) CHANNEL CALIBRATION shall consist of an electronic calibration of the CHANNEL, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

(b) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

(c) CHANNEL CHECK of the Acoustic Monitors shall consist of verifying the instrument threshold levels.

(d) Analog transmitters are calibrated every 18 months. The control room indicator for the analog transmitter is calibrated at the frequency identified in the table.

BASES

3/4.2 INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram (Sections 2.2 and 3/4.1), protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or which terminates operator errors before they result in serious consequences. The objectives of these specifications are to assure the effectiveness of the protective instrumentation when required and to prescribe the trip settings required to assure adequate performance. As indicated, one CHANNEL may be required to be made inoperable for brief intervals to conduct required surveillance. Some of the settings have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations. Surveillance requirements for the instrumentation are selected in order to demonstrate proper function and OPERABILITY. Additional instrumentation for REFUELING operations is identified in Sections 3/4.10.B.

3/4.2.A Isolation Actuation Instrumentation

The isolation actuation instrumentation automatically initiates closure of appropriate isolation valves and/or dampers, which are necessary to prevent or limit the release of fission products from the reactor coolant system, the primary containment and the secondary containment in the event of a loss-of-coolant accident or other reactor coolant pressure boundary (RCPB) leak. The parameters which result in isolation of the secondary containment also actuate the standby gas treatment system. The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary and secondary containment and RCPB system isolation. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. Redundant sensor input signals for each parameter are provided for initiation of isolation (one exception is standby liquid control system initiation).

The reactor low level instrumentation is set to trip at greater than or equal to 144 inches above the top of active fuel (which is defined to be 360 inches above vessel zero). Retrofit 8x8 fuel has an active fuel length 1.24 inches longer than earlier fuel designs. However, present trip setpoints were used in the loss-of-coolant accident (LOCA) analysis for Dresden Units 2 & 3. This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps. For this trip setting and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs, even for the maximum break.

BASES

3/4.2.B Emergency Core Cooling System Actuation Instrumentation

The emergency core cooling system (ECCS) instrumentation generates signals to automatically actuate those safety systems which provide adequate core cooling in the event of a design basis transient or accident. The instrumentation which actuates the ECCS is generally arranged in a one-out-of-two taken twice logic circuit. The logic circuit is composed of the four CHANNEL(s) and each CHANNEL contains the logic from the functional unit sensor up to and including all relays which actuate upon a signal from that sensor. For core spray and low pressure coolant injection, the divisionally powered actuation logic is duplicated and the redundant components are powered from the other division's power supply. The single-failure criterion is met through provisions for redundant core cooling functions, e.g., sprays and automatic blowdown and high pressure coolant injection. Although the instruments are listed by system, in some cases the same instrument is used to send the actuation signal to more than one system at the same time.

For effective emergency core cooling during small pipe breaks, the high pressure coolant injection (HPCI) system must function since reactor pressure does not decrease rapidly enough to allow either core spray or the low pressure coolant injection (LPCI) system to operate in time. The automatic pressure relief function is provided as a backup to the HPCI, in the event HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing or calibration and also minimizes the risk of inadvertent operation, i.e., only one instrument CHANNEL out-of-service.

3/4.2.C ATWS - RPT Instrumentation

The anticipated transient without scram (ATWS) recirculation pump trip (RPT) provides a means of limiting the consequences of the unlikely occurrence of a failure to scram concurrent with the associated anticipated transient. The response of the plant to this postulated event falls within the bounds of events studied in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO24222, dated December 1979. Tripping the recirculation pumps adds negative reactivity from the increase in steam voiding in the core area as core flow decreases.

3/4.2.D Isolation Condenser Actuation Instrumentation

The isolation condenser system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

BASES

3/4.2.E Control Rod Block Actuation Instrumentation

The control rod block functions are provided to prevent excessive control rod withdrawal so that the MINIMUM CRITICAL POWER RATIO (MCPR) does not go below the MCPR fuel cladding integrity Safety Limit. During shutdown conditions, control rod block instrumentation initiates withdrawal blocks to ensure that all control rods remain inserted to prevent inadvertent criticality.

The trip logic for this function is one-out-of-n; e.g., any trip of one of the six average power range monitors (APRMs), eight intermediate range monitors (IRMs), or four source range monitors (SRMs), will result in a rod block. The minimum instrument CHANNEL requirements assure sufficient instrumentation to assure that the single failure criterion is met. The minimum instrument CHANNEL requirements for the rod block monitor may be reduced by one for a short period of time to allow for maintenance, testing, or calibration.

The APRM rod block function is flow-biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection, i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

In the REFUEL and STARTUP/HOT STANDBY OPERATIONAL MODE(s), the APRM rod block function setpoint is significantly reduced to provide the same type of protection in the REFUEL and STARTUP/HOT STANDBY OPERATIONAL MODE(s) as the APRM flow-biased rod block does in the RUN OPERATIONAL MODE, i.e., prevents control rod withdrawal before a scram is reached.

The rod block monitor (RBM) function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error. The trip setting is flow-biased. At low power, the worst-case withdrawal of a single control rod without rod block action will not violate the fuel cladding integrity Safety Limit. Thus the RBM rod block function is not required below the specified power level. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the fuel cladding integrity Safety Limit.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of ten above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity Safety Limit.

A downscale indication on an APRM is an indication that the instrument has failed or is not sufficiently sensitive. In either case, the instrument will not respond to changes in control rod motion, and the control rod motion is thus prevented.

The SRM rod blocks of low count rate and the detector not fully inserted assure that the SRMs are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram discharge volume, high water level rod block provides annunciation for operator action. The alarm setpoint has been selected to provide adequate time to allow for the determination of the cause for the level increase and corrective action prior to automatic scram initiation.

3.3 - LIMITING CONDITIONS FOR OPERATION

D. Maximum Scram Insertion Times

The maximum scram insertion time of each control rod from the fully withdrawn position to 90% insertion, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

1. Declare the control rod(s) exceeding the above maximum scram insertion time inoperable, and
2. When operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION.

With the provisions of the ACTION(s) above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

D. Maximum Scram Insertion Times

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators:

1. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER:
   a. following CORE ALTERATION(s), or
   b. after a reactor shutdown that is greater than 120 days,
2. For specifically affected individual control rods[a] following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and
3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

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a The provisions of Specification 4.0.D are not applicable provided this surveillance is conducted prior to exceeding 40% of RATED THERMAL POWER.

3.3 - LIMITING CONDITIONS FOR OPERATION

N. Economic Generation Control (EGC) System

The economic generation control (EGC) system may be in operation with automatic flow control provided:

1. Core flow is within 65% to 100% of rated core flow, and
2. THERMAL POWER is ≥20% of RATED THERMAL POWER.

APPLICABILITY

OPERATIONAL MODE 1.

ACTION:

With core flow less than 65% or greater than 100% of rated core flow, or THERMAL POWER less than 20% of RATED THERMAL POWER, restore operation to within the limits within one hour. Otherwise, immediately remove the plant from EGC operation.

4.3 - SURVEILLANCE REQUIREMENTS

N. Economic Generation Control (EGC) System

The economic generation control system shall be demonstrated OPERABLE by verifying that core flow is within 65% to 100% of rated core flow and THERMAL POWER is ≥20% of RATED THERMAL POWER:

1. Prior to entry into EGC operation, and
2. At least once per 12 hours while operating in EGC.

BASES

3/4.3.A SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The SHUTDOWN MARGIN limitation is a restriction to be applied principally to a new refueling pattern. Satisfaction of the limitation must be determined at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. This determination is provided by core design calculations and administrative control of fuel loading patterns. These procedures include restrictions to allow only those intermediate fuel assembly configurations that have been shown to provide the required SHUTDOWN MARGIN.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed during xenon free conditions and adjusted to 68°F to accommodate the current moderator temperature. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.38% Δk/k or R + 0.28% Δk/k, as appropriate, with the strongest control rod fully withdrawn and all others fully inserted. Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of determination of the highest control rod worth, either analytically or by test. This is due to the reduced uncertainty in the SHUTDOWN MARGIN test when the highest worth control rod is determined by demonstration. When SHUTDOWN MARGIN is determined by calculations not associated with a test, additional margin must be added to the specified SHUTDOWN MARGIN limit to account for uncertainties in the calculation.

The value of R in units of % Δk/k is the difference between the calculated beginning-of-life core reactivity, at the beginning of the operating cycle, and the calculated value of maximum core reactivity at any time later in the operating cycle, where it would be greater than at the beginning. The value of R shall include the potential SHUTDOWN MARGIN loss assuming full B_4C settling in all inverted poison tubes present in the core. R must be a positive quantity or zero and a new value of R must be determined for each new fuel cycle.

The value of % Δk/k in the above expression is provided as a finite, demonstrable, subcriticality margin. This margin is verified using an in-sequence control rod withdrawal at the beginning-of-life fuel cycle conditions. This assures subcriticality with not only the strongest fully withdrawn but at least an R + 0.28% Δk/k (or 0.38% Δk/k) margin beyond this condition. This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion following a scram signal. Any control rod that is immovable as a result of excessive friction or mechanical interference, or is known to be unscrammable, per Specification 3.3.C, is considered to be incapable of insertion following a scram signal. It is important to note that a control rod can be electrically immovable, but scrammable, and no increase in SHUTDOWN MARGIN is required for these control rods.

3.5 - LIMITING CONDITIONS FOR OPERATION

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[b] and 3[b].

ACTION:

1. For the core spray system:

   a. With one CS subsystem inoperable, provided that the LPCI subsystem is OPERABLE, restore the inoperable CS subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. With both CS subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. For the LPCI subsystem:

   a. With one LPCI pump inoperable[d], provided that both CS subsystems are OPERABLE, restore the inoperable LPCI pump to OPERABLE status within 30 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

b. Three LPCI pumps together develop a flow of at least 14,500 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥20 psig.

c. The HPCI pump develops a flow of at least 5000 gpm against a system head corresponding to reactor vessel pressure, when steam is being supplied to the turbine between 920 and 1005 psig[c].

3. At least once per 18 months:

   a. For the CS system, the LPCI subsystem, and the HPCI system, verify each system/subsystem actuates on an actual or simulated automatic initiation signal. Actual injection of coolant into the reactor vessel may be excluded from this test.

   b. For the HPCI system, verifying that:

      1) The system develops a flow of ≥5000 gpm against a system head corresponding to reactor vessel pressure, when steam is being supplied to the turbine between 150 and 350 psig[c].

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b The HPCI system and ADS are not required to be OPERABLE when reactor steam dome pressure is ≤150 psig.

d The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, <u>both</u> LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE. Otherwise, enter Specification 3.5.A, Action 2.c.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3.5 - LIMITING CONDITIONS FOR OPERATION

b. With the LPCI subsystem otherwise inoperable[d], provided that both CS subsystems are OPERABLE, restore the LPCI subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the LPCI subsystem and one or both CS subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

3. With the HPCI system inoperable, provided both CS subsystems, the LPCI subsystem, the ADS and the Isolation Condenser (IC) system are OPERABLE, restore the HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

4. For the ADS:

a. With one of the above required ADS valves inoperable, provided the HPCI system, both CS subsystems and three LPCI pumps are OPERABLE, restore the inoperable ADS valve to OPERABLE

4.5 - SURVEILLANCE REQUIREMENTS

2) The pump suction is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber water level - high signal.

c. Performing a CHANNEL CALIBRATION of the CS and LPCI system discharge line "keep filled" alarm instrumentation.

d. Deleted.

4. At least once per 18 months for the ADS:

a. Verify the ADS actuates on an actual or simulated automatic initiation signal. Actual valve actuation may be excluded from this test.

b. Manually opening each ADS valve when the reactor steam dome pressure is ≥150 psig[c] and observing that either:

1) The turbine control valve or turbine bypass valve position responds accordingly, or

2) There is a corresponding change in the measured steam flow.

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d The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, <u>both</u> LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE. Otherwise, enter Specification 3.5.A, Action 2.c.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3.6 - LIMITING CONDITIONS FOR OPERATION

J. Specific Activity

The specific activity of the reactor coolant shall be limited to ≤0.2 μCi/gram DOSE EQUIVALENT I-131.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3, with any main steam line not isolated.

ACTION:

1. With the specific acitivity of the reactor coolant >0.2 μCi/gram DOSE EQUIVALENT I-131 but ≤4.0 μCi/gram DOSE EQUIVALENT I-131, determine DOSE EQUIVALENT I-131 once per 4 hours and restore DOSE EQUIVALENT I-131 to within limits within 48 hours[a].

2. With the specific activity of the reactor coolant >0.2 μCi/gram DOSE EQUIVALENT I-131 for greater than 48 hours, or with the specific activity of the reactor coolant >4.0 μCi/gram DOSE EQUIVALENT I-131, determine DOSE EQUIVALENT I-131 once per 4 hours, and isolate all main steam lines within 12 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.6 - SURVEILLANCE REQUIREMENTS

J. Specific Activity

In OPERATIONAL MODE 1, the specific activity of the reactor coolant shall be verified to be ≤0.2 μCi/gram DOSE EQUIVALENT I-131 once per 7 days.

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a The provisions of Specification 3.0.D are not applicable.

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3.6 - LIMITING CONDITIONS FOR OPERATION

O. Shutdown Cooling - HOT SHUTDOWN

Two[a] shutdown cooling (SDC) loops shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling loop shall be in operation[b][c], with each loop consisting of at least:

1. One OPERABLE SDC pump, and

2. One OPERABLE SDC heat exchanger.

APPLICABILITY:

OPERATIONAL MODE 3, with reactor vessel coolant temperature less than the SDC cut-in permissive setpoint.

ACTION:

1. With less than the above required SDC loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible. Within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable SDC loop. Be in at least COLD SHUTDOWN within 24 hours[d].

4.6 - SURVEILLANCE REQUIREMENTS

O. Shutdown Cooling - HOT SHUTDOWN

At least one SDC loop, one recirculation pump or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

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a One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and in operation.

b A shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period provided the other loop is OPERABLE.

c The shutdown cooling loop may be removed from operation during hydrostatic testing.

d Whenever two or more SDC loops are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

3.7 - LIMITING CONDITIONS FOR OPERATION

2. With one or more reactor instrumentation line excess flow check valves inoperable, operation may continue and the provisions of Specification 3.0.C are not applicable, provided that within 4 hours either:

   a. The inoperable valve is restored to OPERABLE status, or

   b. The instrument line is isolated and the associated instrument is declared inoperable.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing at least one explosive squib from an explosive valve such that each explosive squib will be tested at least once per 90 months, and initiating the removed explosive squib(s). The replacement charge for the exploded squib(s) shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable.

6. At the frequency specified by the Primary Containment Leakage Rate Testing Program, verify leakage for any one main steam line isolation valve when tested at P_t (25 psig) is $\leq$11.5 scfh.

3.7 - LIMITING CONDITIONS FOR OPERATION

P. Standby Gas Treatment System

Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

ACTION:

1. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:
   a. In OPERATIONAL MODE(s) 1,2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
   b. In OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.
2. With both standby gas treatment subsystems inoperable in OPERATIONAL MODE(s) 1,2 or 3, restore at least one subsystem to OPERABLE status within one hour, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

P. Standby Gas Treatment System

Each standby gas treatment subsystem shall be demonstrated OPERABLE:

1. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters operating.
2. 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 subsystem by:
   a. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of <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 4000 cfm ±10%.
   b. 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 ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity; and

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

3. With both standby gas treatment subsystems inoperable in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

4.7 - SURVEILLANCE REQUIREMENTS

c. Verifying a subsystem flow rate of 4000 cfm ±10% during system operation when tested in accordance with ANSI N510-1980.

3. 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 ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity.

4. At least once per 18 months by:

   a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 4000 cfm ±10%.

   b. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

      1) Manual initiation from the control room, and

      2) Simulated automatic initiation signal.

   c. Verifying that the heaters dissipate 30 ±3 kw when tested in accordance with ANSI N510-1989. This reading shall include the appropriate correction for variations from 480 volts at the bus.

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* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION	4.7 - SURVEILLANCE REQUIREMENTS

5. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of <1% in accordance with ANSI N510-1980 while operating the system at a flow rate of 4000 cfm ±10%.

6. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of <1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ±10%.

3.8 - LIMITING CONDITIONS FOR OPERATION

2. In OPERATIONAL MODE *, with the control room emergency filtration system or the RCU inoperable, immediately suspend CORE ALTERATION(s), handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

3. The provisions of Specification 3.0.C are not applicable in OPERATIONAL MODE *.

4.8 - SURVEILLANCE REQUIREMENTS

b. 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 ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity; and

c. Verifying a system flow rate of 2000 scfm ±10% during system operation when tested in accordance with ANSI N510-1980.

4. 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 ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity.

5. At least once per 18 months by:

   a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 2000 scfm ±10%.

   b. Verifying that the filter train starts and isolation dampers close on manual initiation from the control room.

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION

4.8 - SURVEILLANCE REQUIREMENTS

c. Verifying that during the pressurization mode of operation, control room positive pressure is maintained at ≥1/8 inch water gauge relative to adjacent areas during system operation at a flow rate ≤2000 scfm.

d. Verifying that the heaters dissipate 12 ±1.2 kw when tested in accordance with ANSI N510-1989. This reading shall include the appropriate correction for variations from 480 volts at the bus.

6. After each complete or partial replacement of an HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 2000 scfm ±10%.

7. After each complete or partial replacement of an charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at flow rate of 2000 scfm ±10%.

BASES

3/4.8.G Sealed Source Contamination

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources, including startup sources and fission detectors, 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.

3/4.8.H Explosive Gas Mixture

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the offgas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10CFR Part 50.

3/4.8.I Main Condenser Offgas Activity

Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50. The release rates are determined at a more expeditious frequency following the determination of an increase of greater than 50%, as indicated by the air ejector noble gas monitor, after factoring out increases due to changes in THERMAL POWER level and off-gas flow in the nominal steady-state fission gas release from the primary coolant.

3/4.8.J Liquid Holdup Tanks

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of Appendix B, Table 2, Column 2, in unrestricted areas. Recirculation of the tank contents for the purpose of reducing the radioactive content is not considered to be an addition of radioactive material to the tank.

3.9 - LIMITING CONDITIONS FOR OPERATION

E. Distribution - Operating

The following power distribution systems shall be energized:

1. A.C. power distribution, consisting of:

   a. Both Unit engineered safety features 4160 volt buses:

      1) For Unit 2, Nos. 23-1 and 24-1,

      2) For Unit 3, Nos. 33-1 and 34-1.

   b. Both Unit engineered safety features 480 volt buses:

      1) For Unit 2, Nos. 28 and 29,

      2) For Unit 3, Nos. 38 and 39.

   c. The Unit 120 volt Essential Service Bus and Instrument Bus.

2. 250 volt D.C. power distribution, consisting of:

   a. For Unit 2, TB MCC 2 and RB MCC 2.

   b. For Unit 3, TB MCC 3 and RB MCC 3.

3. For Unit 2, 125 volt D.C. power distribution, consisting of:

   a. TB Main Bus Nos. 2A-1and 3A,

   b. TB Res. Bus Nos. 2B and 2B-1,

   c. Reserve Bus No. 2, and

   d. RB Distribution Panel No. 2.

4.9 - SURVEILLANCE REQUIREMENTS

E. Distribution - Operating

Each of the required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

3.9 - LIMITING CONDITIONS FOR OPERATION

G. RPS Power Monitoring

Two Reactor Protection System (RPS) electric power monitoring CHANNEL(s) for each inservice RPS Motor Generator (MG) set or alternate power supply shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4[a] and 5[a].

ACTION:

1. With one RPS electric power monitoring CHANNEL for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring CHANNEL to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.
2. With both RPS electric power monitoring CHANNEL(s) for an inservice RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring CHANNEL to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

4.9 - SURVEILLANCE REQUIREMENTS

G. RPS Power Monitoring

The specified RPS electric power monitoring CHANNEL(s) shall be determined OPERABLE:

1. By performance of a CHANNEL FUNCTIONAL TEST[b] each time the plant is in COLD SHUTDOWN for a period of more than 24 hours, unless performed in the previous 6 months.
2. At least once per 18 months by demonstrating the OPERABILITY of overvoltage, undervoltage, and underfrequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic, and output circuit breakers, and verifying the following setpoints:
   a. Overvoltage ≤129.6 volts AC
   b. Undervoltage ≥105.3 volts AC
   c. Underfrequency ≥55.4 Hz

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a With any control rod withdrawn.

b Only required to be performed prior to entering MODE 2 or 3 from MODE 4.

BASES

The initial conditions of design basis transient and accident analyses assume Engineering Safety Features (ESF) systems are OPERABLE. The A.C. and D.C. electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, reactor coolant system and containment design limits are not exceeded.

The A.C. and D.C. sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function. Periodic component tests are supplemented by extensive functional testing during refueling outages under simulated accident conditions.

3/4.9.A A.C. Sources - Operating

The OPERABILITY of the A.C. electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant. This includes maintaining at least one of the onsite or offsite A.C. sources, D.C. power sources and associated distribution systems OPERABLE during accident conditions concurrent with an assumed loss of all offsite power and a worst-case single failure.

There are two sources of electrical energy available, i.e., the offsite transmission system and the onsite diesel generators. Two unit reserve auxiliary transformers are available to supply the Station class 1E distribution system. The reserve auxiliary transformer is sized to carry 100% of the auxiliary load. If this reserve auxiliary transformer (the normal circuit) is lost, auxiliary power from the other unit can be obtained for one division through a 4160 volt bus tie (the alternate circuit). Additionally, two diesel generators are available to handle an accident. The allowable outage time takes into account the capacity and capability of the remaining A.C. sources, reasonable time for repairs, and the low probability of a design basis accident occurring during this period. Surveillance is required to ensure a highly reliable power source and no common cause failure mode for the remaining required offsite A.C. source.

Upon failure of one diesel generator, performance of appropriate surveillance requirements ensures a highly reliable power supply by checking the availability of the required offsite circuits, and the remaining required diesel generator. The initial surveillance is required to be completed regardless of how long the diesel inoperability persists, since the intent is that all diesel generator inoperabilities must be investigated for common cause failures. After the initial surveillance, an additional start test is required approximately mid-way through the allowed outage time to demonstrate continued OPERABILITY of the available onsite power sources. The diesel generator surveillance is limited to the normal start testing, since for cases in which less than a full complement of A.C. sources may be available, paralleling of two of the remaining A.C. sources may compromise the A.C. source independence. Additionally, the action provisions ensure that continued plant operation is not allowed when a complete loss of a required safety function (i.e., certain required components) would occur upon a loss of offsite power. These certain components which are critical to accomplishment of the required safety functions may be identified in advance and administratively controlled and/or evaluated on a case-by-case basis. With suitable

BASES

redundancy in components and features not available, the plant must be placed in a condition for which the Limiting Condition for Operation does not apply.

The term verify as used toward A.C. electrical power sources means to administratively check by examining logs or other information to determine if certain components are out-of-service for pre-planned preventative maintenance, testing, or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

With one offsite circuit and one diesel generator inoperable, individual redundancy is lost in both the offsite and onsite electrical power system. Therefore, the allowable outage time is more limited. The time limit takes into account the capacity and capability of the remaining sources, reasonable time for repairs, and the low probability of a design basis event occurring during this period.

With both of the required offsite circuits inoperable, sufficient onsite A.C. sources are available to maintain the unit in a safe shutdown condition in the event of a design basis transient or accident. In fact, a simultaneous loss of offsite A.C. sources, a loss-of-coolant accident, and a worst-case single failure were postulated as a part of the design basis in the safety analysis. Thus, the allowable outage time provides a period of time to effect restoration of all or all but one of the offsite circuits commensurate with the importance of maintaining an A.C. electrical power system capable of meeting its design intent.

With two diesel generators inoperable there are no remaining standby A.C. sources. Thus, with an assumed loss of offsite electrical power, insufficient standby A.C. sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of A.C. power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown, which could result in grid instability and possibly a loss of total A.C. power. The allowable time to repair is severely restricted during this condition. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

Surveillance Requirements are provided which assure proper circuit continuity for the offsite A.C. electrical power supply to the onsite distribution network and availability of offsite A.C. electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure distribution buses and loads are connected to their preferred power source. The frequency is adequate since breaker position is not likely to change without the operator being aware of it and because status is displayed in the control room. Should the action provisions of this specification require an increase in frequency, this Surveillance Requirement assures proper circuit continuity for the available offsite A.C. sources during periods of degradation and potential information on common cause failures that would otherwise go undiscovered.

BASES

Each battery of the D.C. electrical power systems is sized to start and carry the normal D.C. loads plus all D.C. loads required for safe shutdown on one unit and operations required to limit the consequences of a design basis event on the other unit for a period of 4 hours following loss of all A.C. sources. The battery chargers are sized to restore the battery to full charge under normal (non-emergency) load conditions. A normally disconnected alternate 125 volt battery is also provided as a backup for each normal battery. If both units are operating , the normal 125 volt battery must be returned to service within the specified time frame since the design configuration of the alternate battery circuit is susceptible to single failure and, hence, is not as reliable as the normal station circuit. During times when the other unit is in a Cold Shutdown or Refuel condition, an alternate 125 volt battery is available to replace a normal station 125 volt battery on a continuous basis to provide a second available power source.

With one of the required D.C. electrical power subsystems inoperable the remaining system has the capacity to support a safe shutdown and to mitigate an accident condition. However, a subsequent worst-case single failure would result in complete loss of ESF functions. Therefore, an allowed outage time is provided based on a reasonable time to assess plant status as a function of the inoperable D.C. electrical power subsystem and, if the D.C. electrical power subsystem is not restored to OPERABLE status, prepare to effect an orderly and safe plant shutdown.

Inoperable chargers do not necessarily indicate that the D.C. systems are not capable of performing their post-accident functions as long as the batteries are within their specified parameter limits. With both the required charger inoperable and the battery degraded, prompt action is required to assure an adequate D.C. power supply.

ACTION(s) are provided to delineate the measurements and time frames needed to continue to assure OPERABILITY of the Station batteries when battery parameters are outside their identified limits.

Battery surveillance requirements are based on the defined battery cell parameter values. Category A defines the normal parameter limit for each designated pilot cell in each battery. The pilot cells are the average cells in the battery based on previous test results. These cells are monitored closely as an indication of battery performance. Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out because of a degraded condition or for any other reason. Category B also defines allowable values for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category B allowable value, the assurance of sufficient capacity as described above no longer exists and the battery must be declared inoperable.

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations.

3.10 - LIMITING CONDITIONS FOR OPERATION	4.10 - SURVEILLANCE REQUIREMENTS
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3.10 - LIMITING CONDITIONS FOR OPERATION

H. Water Level - Spent Fuel Storage Pool

The pool water level shall be maintained at a level of ≥33 feet.

APPLICABILITY:

Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.C are not applicable.

4.10 - SURVEILLANCE REQUIREMENTS

H. Water Level - Spent Fuel Storage Pool

The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

BASES

3/4.10.C Control Rod Position

The requirement that all control rods be inserted during other CORE ALTERATION(s) ensures that fuel will not be loaded into a cell without an inserted control rod.

3/4.10.D DELETED

3/4.10.E Communications

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status regarding core reactivity conditions during movement of fuel within the reactor pressure vessel.

3/4.10.F DELETED

3/4.10.G Water Level - Reactor Vessel

3/4.10.H Water Level - Spent Fuel Storage Pool

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

6.8 PROCEDURES AND PROGRAMS

6.8.A Written procedures shall be established, implemented, and maintained covering the activities referenced below:

1. The applicable procedures recommended in Appendix A, of Regulatory Guide 1.33, Revision 2, February 1978,

2. The Emergency Operating Procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33,

3. Station Security Plan implementation,

4. Generating Station Emergency Response Plan implementation,

5. PROCESS CONTROL PROGRAM (PCP) implementation,

6. OFFSITE DOSE CALCULATION MANUAL (ODCM) implementation, and

7. Fire Protection Program implementation.

6.8.B Deleted.

6.8.C Deleted.

6.8.D The following programs shall be established, implemented, and maintained:

1. Reactor Coolant Sources Outside Primary Containment

   This program provides controls to minimize leakage from those portions of systems outside primary containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include CS, HPCI, LPCI, IC, process sampling (post accident sampling of reactor coolant and containment atmosphere), containment monitoring, and standby gas treatment systems. The program shall include the following:

   a. Preventive maintenance and periodic visual inspection requirements, and

   b. Leak test requirements for each system at a frequency of at least once per operating cycle.

4. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by station procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

b. Limitations on the instantaneous concentrations of radioactive material released in liquid effluents to unrestricted areas conforming to ten (10) times the concentration values in 10 CFR Part 20, Appendix B, Table 2, Column 2 to 10 CFR Part 20.1001 - 20.2402,

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

d. Limitations on the annual and quarterly doses to a member of the public from radioactive materials in liquid effluents released from each Unit conforming to Appendix I to 10 CFR Part 50,

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

f. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose conforming to Appendix I to 10 CFR Part 50,

g. Limitations on the dose rate resulting from radioactive materials released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following:

   a) For noble gases: less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

   b) For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to a dose rate of 1500 mrem/yr to any organ.

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each Unit to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50,

i. Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with halflives greater than 8 days in gaseous effluents released from each Unit conforming to Appendix I to 10 CFR Part 50,

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

5. Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemption. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 48 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, is 1.6% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Primary containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria is the overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

The provisions of 4.0.B do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of 4.0.C are applicable to the Primary Containment Leakage Rate Testing Program.

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Regional Administrator of the appropriate Regional Office of the NRC unless otherwise noted.

6.9.A. Routine Reports

1. Deleted.

2. Annual Report

   Annual reports covering the activities of the Unit for the previous calendar year, as described in this section shall be submitted prior to May 1 of each year.

   The reports required shall include:

   a. Tabulation of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/year and their associated person rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter or TLD. Small exposures totaling 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.

   b. The results of specific activity analysis in which the reactor coolant exceeded the limits of Specification 3.6.J. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the 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.

3. Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the Unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

4. Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the facility during the previous calendar year shall be submitted prior to April 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

5. Monthly Operating Report

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to safety valves or safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

6. CORE OPERATING LIMITS REPORT

a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

(1) The Control Rod Withdrawal Block Instrumentation for Table 3.2.E-1 of Specification 3.2.E.

(2) The Average Planar Linear Heat Generation Rate (APLHGR) Limit for Specification 3.11.A.

(3) The Steady State Linear Heat Generation Rate (SLHGR) for Specification 3.11.D.

(4) The Minimum Critical Power Operating Limit (including 20% scram insertion time) for Specification 3.11.C. This includes rated and off-rated flow conditions.

6.12 HIGH RADIATION AREA

6.12.A Pursuant to 10 CFR 20.1601(c), in lieu of the requirements of paragraph 20.1601 of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr at 30 cm (12 in.) 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)[a] (or equivalent document). 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 rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

3. An individual 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 in the RWP (or equivalent document).

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

ADMINISTRATIVE CONTROLS

6.12.B In addition to the requirements of 6.12.A, areas accessible to personnel with radiation levels greater than 1000 mrem/hr at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall require the following:

1. Doors shall be locked to prevent unauthorized entry and shall not prevent individuals from leaving the area. In place of locking the door, direct or electronic surveillance that is capable of preventing unauthorized entry may be used. The keys shall be maintained under the administrative control of the Shift Manager on duty and/or health physics supervision.

2. Personnel access and exposure control requirements of activities being performed within these areas shall be specified by an approved RWP(or equivalent document).

3. Each person entering the area shall be provided with an alarming radiation monitoring device that continuously integrates the radiation dose rate (such as an electronic dosimeter.) Surveillance and radiation monitoring by a Radiation Protection Technician may be substituted for an alarming dosimeter.

4. Deleted.

5. For individual HIGH RADIATION AREAS accessible to personnel with radiation levels of greater than 1000 mrem/h at 30 cm (12 in.) that are located within large areas where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual areas, then such individual areas shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.A Changes to the PCP:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

    a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

    b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after review and acceptance, including approval by the Station Manager.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.A Changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

    a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

    b. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall become effective after review and acceptance, including approval by the Station Manager.

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Report for the period of the report in which any change to the ODCM was made effective. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

<u>ATTACHMENT C</u>
(continued)

REVISED TSUP PAGES FOR
QUAD CITIES NUCLEAR POWER STATION
LICENSE NOS. DPR-29 AND DPR-30

DEFINITIONS

SECTION PAGE

Section 1 DEFINITIONS

DEFINITIONS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

1.0 DEFINITIONS

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Specification 6.8 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specification 6.9.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE

An OPERATIONAL MODE, i.e., MODE, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1-2.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the UFSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

TABLE 4.1.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	Applicable OPERATIONAL MODES	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL[a] CALIBRATION
1. Intermediate Range Monitor:				
a. Neutron Flux - High	2	S[b]	S/U[c], W[o]	E[o]
	3, 4, 5	S	W[o]	E[o]
b. Inoperative	2, 3, 4, 5	NA	W[o]	NA
2. Average Power Range Monitor[f]:				
a. Setdown Neutron Flux - High	2	S[b]	S/U[c], W[o]	SA[o]
	3, 5[m]	S	W[o]	SA[o]
b. Flow Biased Neutron Flux - High	1	S, D	W	W[d,e], SA
c. Fixed Neutron Flux - High	1	S	W	W[d], SA
d. Inoperative	1, 2, 3, 5[m]	NA	W	NA
3. Reactor Vessel Steam Dome Pressure - High	1, 2[i]	NA	M	Q
4. Reactor Vessel Water Level - Low	1, 2	D	M	E[h]
5. Main Steam Line Isolation Valve - Closure	1	NA	M	E
6. Main Steam Line Radiation - High	1, 2[i]	S	M	E[p]
7. Drywell Pressure - High	1, 2[n]	NA	M	Q

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least (½) decades during each startup after entering OPERATIONAL MODE 2 and the IRM and APRM channels shall be determined to overlap for at least (½) decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days. The weekly CHANNEL FUNCTIONAL TEST may be used to fulfill this requirement.

(d) This calibration shall consist of the adjustment of the APRM CHANNEL to conform, within 2% of RATED THERMAL POWER, to the power values calculated by a heat balance during OPERATIONAL MODE 1 when THERMAL POWER is ≥25% of RATED THERMAL POWER. This adjustment must be accomplished: a) within 2 hours if the APRM CHANNEL is indicating lower power values than the heat balance, or b) within 12 hours if the APRM CHANNEL is indicating higher power values than the heat balance. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

Any APRM CHANNEL gain adjustment made in compliance with Specification 3.11.B shall not be included in determining the above difference. This calibration is not required when THERMAL POWER is <25% of RATED THERMAL POWER. The provisions of Specification 4.0.D are not applicable.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 2000 effective full power hours (EFPH).

(g) Deleted.

(h) Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

(i) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.12.A.

(j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(k) This function may be bypassed, provided a control rod block is actuated, for reactor protection system reset in Refuel and Shutdown positions of the reactor mode switch.

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

(l) This function not required to be OPERABLE when THERMAL POWER is less than 45% of RATED THERMAL POWER.

(m) Required to be OPERABLE only prior to and during required SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(n) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(o) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for a period of 24 hours after entering OPERATIONAL MODE 2 or 3 when shutting down from OPERATIONAL MODE 1.

(p) A current source provides an instrument channel alignment every 3 months.

TABLE 3.2.A-1

ISOLATION ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[j]	Minimum CHANNEL(s) per TRIP SYSTEM[a]	Applicable OPERATIONAL MODE(s)	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low	≥144 inches	2	1, 2, 3	20
b. Drywell Pressure - High[d]	≤2.5 psig	2	1, 2, 3	20
c. Drywell Radiation - High	≤100 R/hr	1	1, 2, 3	23
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low[c,k]	≥144 inches	2	1, 2, 3 & *	24
b. Drywell Pressure - High[c,d,k]	≤2.5 psig	2	1, 2, 3	24
c. Reactor Building Ventilation Exhaust Radiation - High[c,k]	≤5 mR/hr	2	1, 2, 3 & **	24
d. Refueling Floor Radiation - High[c,k]	≤100 mR/hr	2	1, 2, 3 & **	24
3. MAIN STEAM LINE (MSL) ISOLATION				
a. Reactor Vessel Water Level - Low Low	≥84 inches	2	1, 2, 3	21
b. MSL Tunnel Radiation - High[b]	≤15[h] x normal background	2	1, 2, 3	21
c. MSL Pressure - Low	≥825 psig	2	1	22
d. MSL Flow - High[k]	≤140% of rated	2/line	1, 2, 3	21
e. MSL Tunnel Temperature - High	≤200°F	2 of 4 in each of 2 sets	1, 2, 3	21

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function[i]	Applicable OPERATIONAL MODE(s)	ACTION
3. SOURCE RANGE MONITORS				
a. Detector not full in[b]	NA	3 2	2 5	51 51
b. Upscale[c]	$\leq 1 \times 10^5$ cps	3 2	2 5	51 51
c. Inoperative[c]	NA	3 2	2 5	51 51
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	6	2, 5	51
b. Upscale	≤108/125 of full scale	6	2, 5	51
c. Inoperative	NA	6	2, 5	51
d. Downscale[d]	≥3/125 of full scale	6	2, 5	51

TABLE 3/4.2.E-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
1. ROD BLOCK MONITORS				
a. Upscale	NA	S/U[b,c], M[c]	Q	1[d]
b. Inoperative	NA	S/U[b,c], M[c]	NA	1[d]
c. Downscale	NA	S/U[b,c], M[c]	Q	1[d]
2. AVERAGE POWER RANGE MONITORS				
a. Flow Biased Neutron Flux - High				
1. Dual Recirculation Loop Operation	NA	S/U[b], M	SA	1
2. Single Recirculation Loop Operation	NA	S/U[b], M	SA	1
b. Inoperative	NA	S/U[b], M	NA	1, 2, 5[j]
c. Downscale	NA	S/U[b], M	SA	1
d. Startup Neutron Flux - High	NA	S/U[b], M	SA	2, 5[j]
3. SOURCE RANGE MONITORS				
a. Detector not full in[f]	NA	S/U[b], W	E	2[i], 5
b. Upscale[g]	NA	S/U[b], W	E	2[i], 5
c. Inoperative[g]	NA	S/U[b], W	NA	2[i], 5

TABLE 4.2[illegible] (Continued)

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U[b], W	E	2[i], 5
b. Upscale	NA	S/U[b], W	E	2[i], 5
c. Inoperative	NA	S/U[b], W	NA	2[i], 5
d. Downscale[h]	NA	S/U[b], W	E	2[i], 5
5. SCRAM DISCHARGE VOLUME (SDV)				
a. Water Level - High	NA	Q	NA	1, 2, 5[e]
b. SDV Switch in Bypass	NA	E	NA	5[e]

TABLE 4.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) Within 7 days prior to startup.

(c) Includes reactor manual control "relay select matrix" system input.

(d) With THERMAL POWER ≥30% of RATED THERMAL POWER.

(e) With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(f) This function shall be automatically bypassed if detector count rate is >100 cps or the IRM channels are on range 3 or higher.

(g) This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

(h) This function shall be automatically bypassed when the IRM channels are on range 1.

(i) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for entry into the applicable OPERATIONAL MODE(s) from OPERATIONAL MODE 1 provided the surveillances are performed within 12 hours after such entry.

(j) Required to be OPERABLE only during SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

TABLE 3.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION

ACTION 60 - a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 61- With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:

a. Either restore the inoperable CHANNEL(s) to OPERABLE status within 7 days of the event, or

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.B within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 62- a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) one less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1; restore at least one inoperable CHANNEL to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

TAB[illegible].2.F-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. Reactor Vessel Pressure	M	E	1, 2
2. Reactor Vessel Water Level	M	E	1, 2
3 Torus Water Level	M	E	1, 2
4. Torus Water Temperature	M	E	1, 2
5. Drywell Pressure - Wide Range	M	E	1, 2
6. Drywell Pressure - Narrow Range	M	E	1, 2
7. Drywell Air Temperature	M	E	1, 2
8. Drywell Oxygen Concentration - Analyzer and Monitor	M	Q	1, 2
9. Drywell Hydrogen Concentration - Analyzer and Monitor	M	Q	1, 2
10. Safety & Relief Valve Position Indicators - Acoustic & Temperature	M	E	1, 2
11. (Source Range) Neutron Monitors	M	E[b]	1, 2
12. Drywell Radiation Monitors	M	E[a]	1, 2, 3
13. Torus Air Temperature	M	E	1, 2
14. Torus Pressure	M	E	1, 2

BASES

3/4.2 INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram (Sections 2.2 and 3/4.1), protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or which terminates operator errors before they result in serious consequences. The objectives of these specifications are to assure the effectiveness of the protective instrumentation when required and to prescribe the trip settings required to assure adequate performance. As indicated, one CHANNEL may be required to be made inoperable for brief intervals to conduct required surveillance. Some of the settings have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations. Surveillance requirements for the instrumentation are selected in order to demonstrate proper function and OPERABILITY. Additional instrumentation for REFUELING operations is identified in Sections 3/4.10.B.

3/4.2.A Isolation Actuation Instrumentation

The isolation actuation instrumentation automatically initiates closure of appropriate isolation valves and/or dampers, which are necessary to prevent or limit the release of fission products from the reactor coolant system, the primary containment and the secondary containment in the event of a loss-of-coolant accident or other reactor coolant pressure boundary (RCPB) leak. The parameters which result in isolation of the secondary containment also actuate the standby gas treatment system. The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary and secondary containment and RCPB system isolation. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. Redundant sensor input signals for each parameter are provided for initiation of isolation (one exception is standby liquid control system initiation).

The reactor low level instrumentation is set to trip at greater than or equal to 144 inches above the top of active fuel (which is defined to be 360 inches above vessel zero). This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps. For this trip setting and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs, even for the maximum break.

3/4.2.B Emergency Core Cooling System Actuation Instrumentation

The emergency core cooling system (ECCS) instrumentation generates signals to automatically actuate those safety systems which provide adequate core cooling in the event of a design basis transient or accident. The instrumentation which actuates the ECCS is generally arranged in a one-out-of-two taken twice logic circuit. The logic circuit is composed of four CHANNEL(s) and each CHANNEL contains the logic from the functional unit sensor up to and including all relays

BASES

which actuate upon a signal from that sensor. For core spray and low pressure coolant injection, the divisionally powered actuation logic is duplicated and the redundant components are powered from the other division's power supply. The single-failure criterion is met through provisions for redundant core cooling functions, e.g., sprays and automatic blowdown and high pressure coolant injection. Although the instruments are listed by system, in some cases the same instrument is used to send the actuation signal to more than one system at the same time.

For effective emergency core cooling during small pipe breaks, the high pressure coolant injection (HPCI) system must function since reactor pressure does not decrease rapidly enough to allow either core spray or the low pressure coolant injection (LPCI) system to operate in time. The automatic pressure relief function is provided as a backup to HPCI, in the event HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing or calibration and also minimizes the risk of inadvertent operation, i.e., only one instrument CHANNEL out-of-service.

3/4.2.C ATWS - RPT Instrumentation

The anticipated transient without scram (ATWS) recirculation pump trip (RPT) provides a means of limiting the consequences of the unlikely occurrence of a failure to scram concurrent with the associated anticipated transient. The response of this plant to this postulated event falls within the bounds of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO24222, dated December 1979. Tripping the recirculation pumps adds negative reactivity by increasing steam voiding in the core area as core flow decreases.

3/4.2.D Reactor Core Isolation Cooling Actuation Instrumentation

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

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3/4.2.E Control Rod Block Actuation Instrumentation

The control rod block functions are provided to prevent excessive control rod withdrawal so that the MINIMUM CRITICAL POWER RATIO (MCPR) does not go below the MCPR fuel cladding integrity Safety Limit. During shutdown conditions, control rod block instrumentation initiates withdrawal blocks to ensure that all control rods remain inserted to prevent inadvertent criticality.

The trip logic for this function is one-out-of-n; e.g., any trip on one of the six average power range monitors (APRMs), eight intermediate range monitors (IRMs), or four source range monitors (SRMs), will result in a rod block. The minimum instrument CHANNEL requirements assure sufficient instrumentation to assure that the single failure criterion is met. The minimum instrument CHANNEL requirements for the rod block monitor may be reduced by one for a short period of time to allow for maintenance, testing, or calibration.

The APRM rod block function is flow-biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection, i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

In the REFUEL MODE during SHUTDOWN MARGIN demonstrations and the STARTUP/HOT STANDBY OPERATIONAL MODE, the APRM rod block function setpoint is significantly reduced to provide the same type of protection in the REFUEL and STARTUP/HOT STANDBY OPERATIONAL MODE(s) as the APRM flow-biased rod block does in the RUN OPERATIONAL MODE, i.e., prevents control rod withdrawal before a scram is reached.

The rod block monitor (RBM) function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error. The trip setting is flow-biased. At low power, the worst-case withdrawal of a single control rod without rod block action will not violate the fuel cladding integrity Safety Limit. Thus the RBM rod block function is not required below the specified power level. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the fuel cladding integrity Safety Limit.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of ten above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity Safety Limit.

A downscale indication on an APRM is an indication that the instrument has failed or is not sufficiently sensitive. In either case, the instrument will not respond to changes in control rod motion, and the control rod motion is thus prevented.

The SRM rod blocks of low count rate and the detector not fully inserted assure that the SRMs are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram

BASES

discharge volume, high water level rod block provides annunciation for operator action. The alarm setpoint has been selected to provide adequate time to allow for the determination of the cause for the level increase and corrective action prior to automatic scram initiation.

3/4.2.F Accident Monitoring Instrumentation

Instrumentation is provided to monitor sufficient accident conditions to adequately assess important variables and provide operators with necessary information to complete the appropriate mitigation actions. OPERABILITY of the instrumentation listed provides adequate monitoring of the containment following a loss-of-coolant accident. Information from this instrumentation will provide the operator with a detailed knowledge of the conditions resulting from the accident; based on this information, the operator can make logical decisions regarding post accident recovery. Allowable outage times are based on diverse instrumentation availability for guiding the operator should an accident occur, and on the low probability of an instrument being out-of-service concurrent with an accident. As noted in the surveillance requirements, the instrumentation CHANNEL(s) associated with Torus Pressure provides a dual function and is shared in common with the Drywell Pressure (Narrow and Wide Ranges) instrumentation. This instrumentation is identified in response to Generic Letter 82-33 and the associated NRC Safety Evaluation Report, and some instrumentation is included in accordance with the response to Generic Letter 83-36.

3/4.2.G Source Range Monitoring Instrumentation

The source range monitors (SRM) provide the operator with the status of the neutron flux in the core at very low power levels during startup and shutdown. The consequences of reactivity accidents are functions of the initial neutron flux. Therefore, the requirements for a minimum count rate assures that any transient, should it occur, begins at or above the initial value used in the analyses of transients from cold conditions. Two OPERABLE SRM CHANNEL(s) are adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. Three OPERABLE SRMs provide an added conservatism. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.2.H Explosive Gas Monitoring Instrumentation

Instrumentation is provided to monitor the concentrations of potentially explosive mixtures in the off-gas holdup system to prevent a possible uncontrolled release via this pathway. This instrumentation is included in accordance with Generic Letter 89-01.

3.3 - LIMITING CONDITIONS FOR OPERATION

D. Maximum Scram Insertion Times

The maximum scram insertion time of each control rod from the fully withdrawn position to 90% insertion, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

1. Declare the control rod(s) exceeding the above maximum scram insertion time inoperable, and
2. When operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION.

With the provisions of the ACTION above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

D. Maximum Scram Insertion Times

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators:

1. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER:
   a. following CORE ALTERATION(s), or
   b. after a reactor shutdown that is greater than 120 days,
2. For specifically affected individual control rods[a] following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and
3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

a The provisions of Specification 4.0.D are not applicable provided this surveillance is conducted prior to exceeding 40% of RATED THERMAL POWER.

3.3 - LIMITING CONDITIONS FOR OPERATION

N. Economic Generation Control (EGC) System

The economic generation control (EGC) system may be in operation with automatic flow control provided:

1 Core flow is within 65% to 100% of rated core flow, and

2. THERMAL POWER is ≥20% of RATED THERMAL POWER.

APPLICABILITY

OPERATIONAL MODE 1.

ACTION:

With core flow less than 65% or greater than 100% of rated core flow, or THERMAL POWER less than 20% of RATED THERMAL POWER, restore operation to within the limits within one hour. Otherwise, immediately remove the plant from EGC operation.

4.3 - SURVEILLANCE REQUIREMENTS

N. Economic Generation Control (EGC) System

The economic generation control system shall be demonstrated OPERABLE by verifying that core flow is within 65% to 100% of rated core flow and THERMAL POWER is ≥20% of RATED THERMAL POWER:

1. Prior to entry into EGC operation, and

2. At least once per 12 hours while operating in EGC.

BASES

3/4.3.A SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The SHUTDOWN MARGIN limitation is a restriction to be applied principally to a new refueling pattern. Satisfaction of the limitation must be determined at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. This determination is provided by core design calculations and administrative control of fuel loading patterns. These procedures include restrictions to allow only those intermediate fuel assembly configurations that have been shown to provide the required SHUTDOWN MARGIN.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed during xenon free conditions and adjusted to 68°F to accommodate the current moderator temperature. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.38% $\Delta k/k$ or R + 0.28% $\Delta k/k$, as appropriate, with the strongest control rod fully withdrawn and all others fully inserted. Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of determination of the highest control rod worth, either analytically or by test. This is due to the reduced uncertainty in the SHUTDOWN MARGIN test when the highest worth control rod is determined by demonstration. When SHUTDOWN MARGIN is determined by calculations not associated with a test, additional margin must be added to the specified SHUTDOWN MARGIN limit to account for uncertainties in the calculation.

The value of R in units of % $\Delta k/k$ is the difference between the calculated beginning-of-life core reactivity, at the beginning of the operating cycle, and the calculated value of maximum core reactivity at any time later in the operating cycle, where it would be greater than at the beginning. The value of R shall include the potential SHUTDOWN MARGIN loss assuming full B_4C settling in all inverted poison tubes present in the core. R must be a positive quantity or zero and a new value of R must be determined for each new fuel cycle.

The value of % $\Delta k/k$ in the above expression is provided as a finite, demonstrable, subcriticality margin. This margin is verified using an in-sequence control rod withdrawal at the beginning-of-life fuel cycle conditions. This assures subcriticality with not only the strongest fully withdrawn but at least an R + 0.28% $\Delta k/k$ (or 0.38% $\Delta k/k$) margin beyond this condition. This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion following a scram signal. Any control rod that is immovable as a result of excessive friction or mechanical interference, or is known to be unscrammable, per Specification 3.3.C, is considered to be incapable of insertion following a scram signal. It is important to note that a control rod can be electrically immovable, but scrammable, and no increase in SHUTDOWN MARGIN is required for these control rods.

3.5 - LIMITING CONDITIONS FOR OPERATION

4. The automatic depressurization system (ADS) with at least 5 OPERABLE ADS valves.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[b] and 3[b].

ACTION:

1. For the core spray system:

   a. With one CS subsystem inoperable, provided that the LPCI subsystem is OPERABLE, restore the inoperable CS subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. With both CS subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. For the LPCI subsystem:

   a. With one LPCI pump inoperable[f], provided that both CS subsystems are OPERABLE, restore the inoperable LPCI pump to OPERABLE status within 30 days,

4.5 - SURVEILLANCE REQUIREMENTS

2. Verifying that, when tested pursuant to Specification 4.0.E:

   a. The CS pump in each subsystem develop a flow of at least 4500 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥90 psig.

   b. Two LPCI pumps together develop a flow of at least 9,000 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥20 psig.

   c. The HPCI pump develops a flow of at least 5000 gpm against a system head corresponding to reactor vessel pressure, when steam is being supplied to the turbine between 920 and 1005 psig[c].

3. At least once per 18 months:

   a. For the CS system, the LPCI subsystem, and the HPCI system, verify each system/subsystem actuates on an actual or simulated automatic initiation signal. Actual injection of coolant into the reactor vessel may be excluded from this test.

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b The HPCI system and ADS are not required to be OPERABLE when reactor steam dome pressure is ≤150 psig.

f The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, both LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE. Otherwise, enter Specification 3.5.A, Action 2.c.

The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3.5 - LIMITING CONDITIONS FOR OPERATION

or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the LPCI subsystem otherwise inoperable[f], provided that both CS subsystems are OPERABLE, restore the LPCI subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours[d].

c. With the LPCI subsystem and one or both CS subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours[d].

3. With the HPCI system inoperable, provided both CS subsystems, the LPCI subsystem, the ADS and the Reactor Core Isolation Cooling (RCIC) system are OPERABLE, restore the HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

b. For the HPCI system, verifying that:

1) The system develops a flow of ≥5000 gpm against a system head corresponding to reactor vessel pressure, when steam is being supplied to the turbine between 150 and 180 psig[c].

2) The pump suction is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber water level - high signal.

c. Performing a CHANNEL CALIBRATION of the ECCS discharge line "keep filled" alarm instrumentation.

d. Deleted.

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f The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, both LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE. Otherwise, enter Specification 3.5.A, Action 2.c.

d Whenever the two required RHR SDC mode subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3.6 - LIMITING CONDITIONS FOR OPERATION

J. Specific Activity

The specific activity of the reactor coolant shall be limited to ≤0.2 μCi/gram DOSE EQUIVALENT I-131.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3, with any main steam line not isolated.

ACTION:

1. With the specific activity of the reactor coolant >0.2 μCi/gram DOSE EQUIVALENT I-131 but ≤4.0 μCi/gram DOSE EQUIVALENT I-131, determine DOSE EQUIVALENT I-131 once per 4 hours and restore DOSE EQUIVALENT I-131 to within limits within 48 hours[a].

2. With the specific activity of the reactor coolant >0.2 μCi/gram DOSE EQUIVALENT I-131 for greater than 48 hours, or with the specific activity of the reactor coolant >4.0 μCi/gram DOSE EQUIVALENT I-131, determine DOSE EQUIVALENT I-131 once per 4 hours, and isolate all main steam lines within 12 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.6 - SURVEILLANCE REQUIREMENTS

J. Specific Activity

In OPERATIONAL MODE 1, the specific activity of the reactor coolant shall be verified to be ≤0.2 μCi/gram DOSE EQUIVALENT I-131 once per 7 days.

a The provisions of Specification 3.0.D are not applicable.

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3.7 - LIMITING CONDITIONS FOR OPERATION

2. With one or more reactor instrumentation line excess flow check valves inoperable, operation may continue and the provisions of Specification 3.0.C are not applicable, provided that within 4 hours either:

   a. The inoperable valve is restored to OPERABLE status, or

   b. The instrument line is isolated and the associated instrument is declared inoperable.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing at least one explosive squib from an explosive valve such that each explosive squib will be tested at least once per 90 months, and initiating the removed explosive squib(s). The replacement charge for the exploded squib(s) shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable.

6. At the frequency specified by the Primary Containment Leakage Rate Testing Program, verify leakage for any one main steam line isolation valve when tested at P_t (25 psig) is ≤11.5 scfh.

3.7 - LIMITING CONDITIONS FOR OPERATION

P. Standby Gas Treatment System

Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

ACTION:

1. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

   a. In OPERATIONAL MODE(s) 1,2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. In OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

2. With both standby gas treatment subsystems inoperable in OPERATIONAL MODE(s) 1,2 or 3, restore at least one subsystem to OPERABLE status within one hour, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

P. Standby Gas Treatment System

Each standby gas treatment subsystem shall be demonstrated OPERABLE:

1. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters operating.

2. 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 subsystem by:

   a. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of $<1\%$ 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 4000 cfm $\pm 10\%$.

   b. 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 ASTM-D-3803-89, for a methyl iodide penetration of $<10\%$, when tested at 30°C and 70% relative humidity; and

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

3. With both standby gas treatment subsystems inoperable in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

4.7 - SURVEILLANCE REQUIREMENTS

c. Verifying a subsystem flow rate of 4000 cfm ±10% during system operation when tested in accordance with ANSI N510-1980.

3. After every 1440 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 ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity.

4. At least once per 18 months by:

   a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 4000 cfm ±10%.

   b. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

      1) Manual initiation from the control room, and

      2) Simulated automatic initiation signal.

   c. Verifying that the heaters dissipate 30 ±3 kw when tested in accordance with ANSI N510-1989. This reading shall include the appropriate correction for variations from 480 volts at the bus.

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

4.7 - SURVEILLANCE REQUIREMENTS

5. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of $<1\%$ in accordance with ANSI N510-1980 while operating the system at a flow rate of 4000 cfm $\pm 10\%$.

6. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of $<1\%$ in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm $\pm 10\%$.

3.8 - LIMITING CONDITIONS FOR OPERATION

2. In OPERATIONAL MODE *, with the control room emergency filtration system or the RCU inoperable, immediately suspend CORE ALTERATION(s), handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

3. The provisions of Specification 3.0.C are not applicable in OPERATIONAL MODE *.

4.8 - SURVEILLANCE REQUIREMENTS

b. 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 ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity; and

c. Verifying a system flow rate of 2000 scfm ±10% during system operation when tested in accordance with ANSI N510-1980.

4. 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 ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity.

5. At least once per 18 months by:

a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 2000 scfm ±10%.

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* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION

4.8 - SURVEILLANCE REQUIREMENTS

b. Verifying that the isolation dampers close on each of the following signals:

1) Manual initiation from the control room, and

2) Simulated automatic isolation signal.

c. Verifying that during the pressurization mode of operation, control room positive pressure is maintained at ≥1/8 inch water gauge relative to adjacent areas during system operation at a flow rate ≤2000 scfm.

d. Verifying that the heaters dissipate 12 ±1.2 kw when tested in accordance with ANSI N510-1989. This reading shall include the appropriate correction for variations from 480 volts at the bus.

6. After each complete or partial replacement of an HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 2000 scfm ±10%.

7. After each complete or partial replacement of an charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at flow rate of 2000 scfm ±10%.

3.9 - LIMITING CONDITIONS FOR OPERATION

E. Distribution - Operating

The following power distribution systems shall be energized:

1. A.C. power distribution, consisting of:

    a. Both Unit engineered safety features 4160 volt buses:

        1) For Unit 1, Nos. 13-1 and 14-1,

        2) For Unit 2, Nos. 23-1 and 24-1.

    b. Both Unit engineered safety features 480 volt buses:

        1) For Unit 1, Nos. 18 and 19,

        2) For Unit 2, Nos. 28 and 29, and

    c. The Unit 120 volt Essential Service Bus and Instrument Bus.

2. 250 volt D.C. power distribution, consisting of:

    a. 1) For Unit 1, TB MCC No. 1,

        2) For Unit 2, TB MCC No. 2.

    b. 1) For Unit 1, RB MCC Nos. 1A and 1B,

        2) For Unit 2, RB MCC Nos. 2A and 2B.

3. For Unit 1, 125 volt D.C. power distribution, consisting of:

    a. TB Main Bus Nos. 1A, 1A-1 and 2A,

    b. TB Reserve Bus Nos. 1B and 1B-1, and

    c. RB Distribution Panel No. 1.

4.9 - SURVEILLANCE REQUIREMENTS

E. Distribution - Operating

Each of the required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

3.9 - LIMITING CONDITIONS FOR OPERATION

G. RPS Power Monitoring

Two Reactor Protection System (RPS) electric power monitoring CHANNEL(s) for each inservice RPS Motor Generator (MG) set or alternate power supply shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4[a] and 5[a].

ACTION:

1. With one RPS electric power monitoring CHANNEL for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring CHANNEL to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

2. With both RPS electric power monitoring CHANNEL(s) for an inservice RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring CHANNEL to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

4.9 - SURVEILLANCE REQUIREMENTS

G. RPS Power Monitoring

The specified RPS electric power monitoring CHANNEL(s) shall be determined OPERABLE:

1. By performance of a CHANNEL FUNCTIONAL TEST[b] each time the plant is in COLD SHUTDOWN for a period of more than 24 hours, unless performed in the previous 6 months.

2. At least once per 18 months by demonstrating the OPERABILITY of overvoltage, undervoltage, and underfrequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic, and output circuit breakers, and verifying the following setpoints:

   a. Overvoltage ≤129.6 volts AC

   b. Undervoltage ≥105.3 volts AC

   c. Underfrequency ≥55.4 Hz

With any control rod withdrawn.

b Only required to be performed prior to entering MODE 2 or 3 from MODE 4.

BASES

With suitable redundancy in components and features not available, the plant must be placed in a condition for which the Limiting Condition for Operation does not apply.

The term verify as used toward A.C. electrical power sources means to administratively check by examining logs or other information to determine if certain components are out-of-service for pre-planned preventative maintenance, testing, or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

With one offsite circuit and one diesel generator inoperable, individual redundancy is lost in both the offsite and onsite electrical power system. Therefore, the allowable outage time is more limited. The time limit takes into account the capacity and capability of the remaining sources, reasonable time for repairs, and the low probability of a design basis event occurring during this period.

With both of the required offsite circuits inoperable, sufficient onsite A.C. sources are available to maintain the unit in a safe shutdown condition in the event of a design basis transient or accident. In fact, a simultaneous loss of offsite A.C. sources, a loss-of-coolant accident, and a worst-case single failure were postulated as a part of the design basis in the safety analysis. Thus, the allowable outage time provides a period of time to effect restoration of all or all but one of the offsite circuits commensurate with the importance of maintaining an A.C. electrical power system capable of meeting its design intent.

With two diesel generators inoperable there are no remaining standby A.C. sources. Thus, with an assumed loss of offsite electrical power, insufficient standby A.C. sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of A.C. power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown, which could result in grid instability and possibly a loss of total A.C. power. The allowable time to repair is severely restricted during this condition. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

Surveillance Requirements are provided which assure proper circuit continuity for the offsite A.C. electrical power supply to the onsite distribution network and availability of offsite A.C. electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure distribution buses and loads are connected to their preferred power source. The frequency is adequate since breaker position is not likely to change without the operator being aware of it and because status is displayed in the control room. Should the action provisions of this specification require an increase in frequency, this Surveillance Requirement assures proper circuit continuity for the available offsite A.C. sources during periods of degradation and potential information on common cause failures that would otherwise go undiscovered.

BASES

Each battery of the D.C. electrical power systems is sized to start and carry the normal D.C. loads plus all D.C. loads required for safe shutdown on one unit and operations required to limit the consequences of a design basis event on the other unit for a period of 4 hours following loss of all A.C. sources. The battery chargers are sized to restore the battery to full charge under normal (non-emergency) load conditions. A normally disconnected alternate 125 volt battery is also provided as a backup for each normal battery. If both units are operating , the normal 125 volt battery must be returned to service within the specified time frame since the design configuration of the alternate battery circuit is susceptible to single failure and, hence, is not as reliable as the normal station circuit. During times when the other unit is in a Cold Shutdown or Refuel condition, an alternate 125 volt battery is available to replace a normal station 125 volt battery on a continuous basis to provide a second available power source.

With one of the required D.C. electrical power subsystems inoperable the remaining system has the capacity to support a safe shutdown and to mitigate an accident condition. However, a subsequent worst-case single failure would result in complete loss of ESF functions. Therefore, an allowed outage time is provided based on a reasonable time to assess plant status as a function of the inoperable D.C. electrical power subsystem and, if the D.C. electrical power subsystem is not restored to OPERABLE status, prepare to effect an orderly and safe plant shutdown.

Inoperable chargers do not necessarily indicate that the D.C. systems are not capable of performing their post-accident functions as long as the batteries are within their specified parameter limits. With both the required charger inoperable and the battery degraded, prompt action is required to assure an adequate D.C. power supply.

ACTION(s) are provided to delineate the measurements and time frames needed to continue to assure OPERABILITY of the Station batteries when battery parameters are outside their identified limits.

Battery surveillance requirements are based on the defined battery cell parameter values. Category A defines the normal parameter limit for each designated pilot cell in each battery. The pilot cells are the average cells in the battery based on previous test results. These cells are monitored closely as an indication of battery performance. Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out because of a degraded condition or for any other reason. Category B also defines allowable values for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category B allowable value, the assurance of sufficient capacity as described above no longer exists and the battery must be declared inoperable.

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations.

3.10 - LIMITING CONDITIONS FOR OPERATION

D. DELETED

4.10 - SURVEILLANCE REQUIREMENTS

D. DELETED

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3.10 - LIMITING CONDITIONS FOR OPERATION

H. Water Level - Spent Fuel Storage Pool

The pool water level shall be maintained at a level of ≥33 feet.

APPLICABILITY:

Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.C are not applicable.

4.10 - SURVEILLANCE REQUIREMENTS

H. Water Level - Spent Fuel Storage Pool

The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

3.10 - LIMITING CONDITIONS FOR OPERATION

L. Residual Heat Removal and Coolant Circulation - Low Water Level

Two shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE, with each loop consisting of at least:

1. One OPERABLE RHR pump, and

2. One OPERABLE RHR heat exchanger.

APPLICABILITY:

OPERATIONAL MODE 5, when irradiated fuel is in the reactor vessel and the water level is $<$23 feet above the top of the reactor pressure vessel flange.

ACTION:

With less than the above required shutdown cooling mode loops of the RHR system OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the OPERABILITY of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

4.10 - SURVEILLANCE REQUIREMENTS

L. Residual Heat Removal and Coolant Circulation - Low Water Level

1. At least one shutdown cooling mode loop of the RHR system shall be verified to be capable of circulating reactor coolant at least once per 12 hours.

2. Monitor the reactor coolant temperature at least once per hour.

BASES

3/4.10.C Control Rod Position

The requirement that all control rods be inserted during other CORE ALTERATION(s) ensures that fuel will not be loaded into a cell without an inserted control rod.

3/4.10.D DELETED

3/4.10.E Communications

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status regarding core reactivity conditions during movement of fuel within the reactor pressure vessel.

3/4.10.F DELETED

3/4.10.G Water Level - Reactor Vessel

3/4.10.H Water Level - Spent Fuel Storage Pool

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

6.8 PROCEDURES AND PROGRAMS

6.8.A Written procedures shall be established, implemented, and maintained covering the activities referenced below:

1. The applicable procedures recommended in Appendix A, of Regulatory Guide 1.33, Revision 2, February 1978,

2. The Emergency Operating Procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33,

3. Station Security Plan implementation,

4. Generating Station Emergency Response Plan implementation,

5. PROCESS CONTROL PROGRAM (PCP) implementation,

6. OFFSITE DOSE CALCULATION MANUAL (ODCM) implementation, and

7. Fire Protection Program implementation.

6.8.B Deleted

6.8.C Deleted

6.8.D The following programs shall be established, implemented, and maintained:

1. Reactor Coolant Sources Outside Primary Containment

   This program provides controls to minimize leakage from those portions of systems outside primary containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include CS, HPCI, LPCI, RCIC, process sampling (post accident sampling of reactor coolant and containment atmosphere), containment monitoring, and standby gas treatment systems. The program shall include the following:

   a. Preventive maintenance and periodic visual inspection requirements, and

   b. Leak test requirements for each system at a frequency of at least once per operating cycle.

4. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by station procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

b. Limitations on the instantaneous concentrations of radioactive material released in liquid effluents to unrestricted areas conforming to ten (10) times the concentration values in 10 CFR Part 20, Appendix B, Table 2, Column 2 to 10 CFR Part 20.1001 - 20.2402,

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

d. Limitations on the annual and quarterly doses to a member of the public from radioactive materials in liquid effluents released from each Unit conforming to Appendix I to 10 CFR Part 50,

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

f. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose conforming to Appendix I to 10 CFR Part 50,

g. Limitations on the dose rate resulting from radioactive materials released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following:

   a) For noble gases: less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

   b) For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to a dose rate of 1500 mrem/yr to any organ.

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each Unit to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50,

i. Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with halflives greater than 8 days in gaseous effluents released from each Unit conforming to Appendix I to 10 CFR Part 50,

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Regional Administrator of the appropriate Regional Office of the NRC unless otherwise noted.

6.9.A. Routine Reports

1. Deleted

2. Annual Report

   Annual reports covering the activities of the Unit for the previous calendar year, as described in this section shall be submitted prior to May 1 of each year.

6.12 HIGH RADIATION AREA

6.12.A Pursuant to 10 CFR 20.1601(c), in lieu of the requirements of paragraph 20.1601 of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr at 30 cm (12 in.) 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)[a] (or equivalent document). 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 rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

3. An individual 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 in the RWP (or equivalent document).

6.12.B In addition to the requirements of 6.12.A, above, areas accessible to personnel with radiation levels greater than 1000 mrem/hr at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall require the following:

1. Doors shall be locked to prevent unauthorized entry and shall not prevent individuals from leaving the area. In place of locking the door, direct or electronic surveillance that is capable of preventing unauthorized entry may be used. The keys shall be maintained under the administrative control of the Shift Engineer on duty and/or health physics supervision.

2. Personnel access and exposure control requirements of activities being performed within these areas shall be specified by an approved RWP (or equivalent document).

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

3. Each person entering the area shall be provided with an alarming radiation monitoring device that continuously integrates the radiation dose rate (such as an electronic dosimeter.) Surveillance and radiation monitoring by a Radiation Protection Technician may be substituted for an alarming dosimeter.

4. Deleted.

5. For individual HIGH RADIATION AREAS accessible to personnel with radiation levels of greater than 1000 mrem/h at 30 cm (12 in.) that are located within large areas where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual areas, then such individual areas shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.A Changes to the PCP:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

    a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

    b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after review and acceptance, including approval by the Station Manager.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.A Changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

    a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

    b. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall become effective after review and acceptance, including approval by the Station Manager.

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Report for the period of the report in which any change to the ODCM was made effective. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.