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(continued)

1.1 Definitions

DOSE EQUIVALENT I-131
(continued)

conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites;" Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977; or ICRP 30, Supplement to Part 1, pages 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or
2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE; and

d. Pressure Boundary LEAKAGE

LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

(continued)

1.1 Definitions (continued)

LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components required for OPERABILITY of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, division, 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 and seal water, lubrication, and other auxiliary equipment that

(continued)

1.1 Definitions

OPERABLE — OPERABILITY (continued)	are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2957 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from the opening of the sensor contact until the opening of the trip actuator. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	<p>SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:</p> <ol style="list-style-type: none">The reactor is xenon free;The moderator temperature is 68°F; andAll control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. <p>With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.</p>
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1 Reduce THERMAL POWER to < 38.5% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Be in MODE 2.	8 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1 Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

- NOTES-----
1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.
 2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.
-

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	<p>-----NOTE-----</p> <p>Not required to be performed until 12 hours after THERMAL POWER \geq 25% RTP.</p> <p>-----</p> <p>Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is \leq 2% RTP.</p>	7 days
SR 3.3.1.1.3	Adjust the channel to conform to a calibrated flow signal.	7 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.1.1.11 Calibrate the trip units.	92 days
SR 3.3.1.1.12 Perform CHANNEL CALIBRATION.	92 days
SR 3.3.1.1.13 Verify Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is \geq 38.5% RTP.	92 days
SR 3.3.1.1.14 -----NOTES----- 1. Neutron detectors are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 24 hours after entering MODE 2. 3. For Function 2.b, not required for the flow portion of the channels. ----- Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.15 Perform CHANNEL FUNCTIONAL TEST.	24 months

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 121/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 121/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.17	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.17	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, Setdown	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.14 SR 3.3.1.1.17	≤ 17.1% RTP
b. Flow Biased Neutron Flux - High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.16 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 0.56 W + 67.4% RTP and ≤ 122% RTP(b)
(continued)					

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) 0.56 W + 63.2% and ≤ 118.4% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Fixed Neutron Flux - High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 122% RTP
d. Inop	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.17	NA
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.11 SR 3.3.1.1.16 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 1050 psig
4. Reactor Vessel Water Level - Low	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.11 SR 3.3.1.1.16 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 3.8 inches
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.16 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 9.8% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 2.43 psig

(continued)

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Volume Water Level - High					
a. Thermal Switch	1.2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 38.9 gallons
	5(a)	2	H	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 38.9 gallons
b. Differential Pressure Switch	1.2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 32.3 gallons
	5(a)	2	H	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 32.3 gallons
8. Turbine Stop Valve - Closure	≥ 38.5% RTP	4	E	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 9.7% closed
9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 38.5% RTP	2	E	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 475 psig
10. Turbine Condenser Vacuum - Low	1	2	F	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 21.6 inches Hg vacuum
11. Reactor Mode Switch - Shutdown Position	1.2	1	G	SR 3.3.1.1.15 SR 3.3.1.1.17	NA
	5(a)	1	H	SR 3.3.1.1.15 SR 3.3.1.1.17	NA
12. Manual Scram	1.2	1	G	SR 3.3.1.1.8 SR 3.3.1.1.17	NA
	5(a)	1	H	SR 3.3.1.1.8 SR 3.3.1.1.17	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Primary Containment Isolation Instrumentation
3.3.6.1

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	\geq -55.2 inches
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	\geq 831 psig
c. Main Steam Line Pressure - Timer	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	\leq 0.331 seconds
d. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	\leq 254.3 psid
e. Main Steam Line Tunnel Temperature - High	1,2,3	2 per trip string	D	SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	\leq 198°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level - Low	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	\geq 3.8 inches
b. Drywell Pressure - High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	\leq 2.43 psig
c. Drywell Radiation - High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	\leq 70 R/hr

(continued)

CREV System Isolation Instrumentation
3.3.7.1

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Ventilation (CREV) System Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low	1,2,3, (a)	2	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.5 SR 3.3.7.1.6	≥ 3.8 inches
2. Drywell Pressure - High	1,2,3	2	C	SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.6	≤ 2.43 psig
3. Main Steam Line Flow - High	1,2,3	2 per MSL	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 254.3 psid
4. Refueling Floor Radiation - High	1,2,3, (a),(b)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.6	≤ 100 mR/hr
5. Reactor Building Ventilation Exhaust Radiation - High	1,2,3, (a),(b)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.6	≤ 9 mR/hr

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in the secondary containment.

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

- b. A loss of safety function exists when, assuming no concurrent single failure, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:
 - 1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or
 - 2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or
 - 3. A required system redundant to support system(s) for the supported systems described in b.1 and b.2 above is also inoperable.
- c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Primary Containment Leakage Rate Testing Program

- a. This program shall establish the leakage 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.
- b. The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a , is 43.9 psig.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3. The LHGR for Specification 3.2.3.
4. Control Rod Block Instrumentation Setpoint for the Rod Block Monitor—Upscale Function Allowable Value for Specification 3.3.2.1.
- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
 1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel."
 2. Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods."
 3. Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A).
 4. Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-85-67(P)(A).
 5. Qualification of Exxon Nuclear Fuel for Extended Burnup, XN-NF-82-06(P)(A).
 6. Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A).
 7. Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A).
 8. Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A).
 9. ANFB Critical Power Correlation, ANF-1125(P)(A).

(continued)
