Specification 3.8.1

M11 A1

add specific requirements identified in SR 3.8.1.9, SR 3.8.1.10, SR 3.8.1.12

add Required Action B.2 L7

3.9 (cont'd)

[ACTION B] 3. From and after the time that one of the Emergency Diesel Generator Systems is made or found to be inoperable, continued reactor operation is permissible for a period not to exceed 14 days provided that the two incoming power sources are available and that the remaining Diesel Generator System is operable. At the end of the 14 day period, the reactor shall be placed in a cold condition within 24 hours, unless the affected diesel generator system is made operable sooner.

[RA B.4] [ACTION D] [ACTION G] [ACTION E] L1 A6

[ACTION F]

36 L2

in MODE 3 in 12 hours and M3

[ACTION E] 4. When both Emergency Diesel Generator Systems are made or found to be inoperable restore at least one system to operable status within two hours or place the reactor in the cold condition within the following 24 hours.

ACTION F]

5. Deleted

MODE 3 in 12 hours and in M3 L2 36

and 21 days from discovery of failure to meet LCO M4

4.9 (cont'd)

LA2

3. The emergency diesel generator system instrumentation shall be checked during the monthly generator test.

A8 add Note to SR 3.8.1.9, SR 3.8.1.10, and SR 3.8.1.12

verified by an actual or L4

A2 M9

4. Once every 24 months, the conditions under which the Emergency Diesel Generator System is required will be simulated to demonstrate that the pair of diesel generators will start, accelerate, force parallel, and accept the emergency loads in the prescribed sequence.

[SR 3.8.1.9] [SR 3.8.1.10] [SR 3.8.1.12]

5. While the reactor is being operated in accordance with Specification 3.9.B.3, the availability of the operable Emergency Diesel Generator System shall be demonstrated once within 8 hours by manual starting and force paralleling. The operability of the remaining Emergency Diesel Generator System need not be demonstrated if the affected Emergency Diesel Generator System became inoperable due to:

[RA B.3.1]

24 L5

a. Preplanned preventive maintenance or testing.

b. An inoperable support system with no potential common mode failure for the remaining diesel generators, or

c. An independently testable component with no potential common mode failure for the remaining diesel generators.

[RA B.3.1]

A7

(5) RAI 3.8.1-12

C

add Action G A6

Specification 3.8.1 (A1)

add SR 3.8.1.1 Frequency (M12)

3.9 Continued

4.9 Continued

[RA A.1]
[RA B.1]

6. Once within one hour and at least once per eight hours thereafter, while the reactor is being operated in accordance with Specifications 3.9.B.1, 3.9.B.3 and 3.9.B.4, the availability of off-site power shall be assured by verifying correct breaker alignment and by verifying that the associated off-site electrical line is energized. (C)

(M12)

[SR 3.8.1.1]

C. Diesel Fuel

There will be a minimum of 64,000 gal. of diesel fuel on site for each operable pair of diesel generators.

1. From and after the time that fuel oil storage tank level instrumentation is made or found to be inoperable for any reason continued reactor operation is permissible indefinitely, provided that the level in the affected storage tank is manually measured at least once/day.

See ITS 3.8.3

C. Diesel Fuel

Perform SR 3.8.1.1 for Operable circuit

Once a month the quantity of diesel fuel available in each storage tank shall be manually measured and compared to the reading of the local level indicators to ensure the proper operation thereof.

1. Once a month a sample of the diesel fuel in each storage tank shall be checked for quality as per the following:

Flash Point - °F	125°F min.
Pour Point - °F	10°F max.
Water & Sediment	0.05% max.
Ash	0.01% max.
Distillation 90% Point	540 min.
Viscosity (SSU) at 100°F	40 max.
Sulfur	1% max.
Copper Strip Corrosion	No. 3 max.
Cetane #	35 min.

Specification 3.8.3

A1

3.9 Continued

C. Diesel Fuel

[LCO 3.8.3] There will be a minimum of 64,000 gal. of diesel fuel on site for each operable pair of diesel generators.

M1

32,000

[Applicability]

1. From and after the time that fuel oil storage tank level instrumentation is made or found to be inoperable for any reason continued reactor operation is permissible indefinitely, provided that the level in the affected storage tank is manually measured at least once/day.

L1

A2 add ACTIONS NOTE, separate conditions entry

M2 add proposed LCO, ACTION B and F for lube oil inventory

M3 add proposed LCO, ACTION C, D and F for fuel oil properties

4.9 Continued

See 3.8.1

6. Once within one hour and at least once per eight hours thereafter, while the reactor is being operated in accordance with Specifications 3.9.B.1, 3.9.B.3 and 3.9.B.4, the availability of off-site power shall be assured by verifying correct breaker alignment and by verifying that the associated off-site electrical line is energized.

C. Diesel Fuel

LA1

Once a month the quantity of diesel fuel available in each storage tank shall be manually measured and compared to the reading of the local level indicators to ensure the proper operation thereof.

[SR 3.8.3.1]

1. Once a month a sample of the diesel fuel in each storage tank shall be checked for quality as per the following:

[SR3.8.3.5]

Flash Point - °F	125°F min.
Pour Point - °F	10°F max.
Water & Sediment	0.05% max.
Ash	0.01% max.
Distillation 90% Point	540 min.
Viscosity (SSU) at 100°F	40 max.
Sulfur	1% max.
Copper Strip Corrosion	No. 3 max.
Cetane #	35 min.

LA2

M2 add proposed SR 3.8.3.2

M3 add proposed SR 3.8.3.3

Specification 3.8.1

A13

3.9 Continued

A5

2. The Diesel Fuel Oil Transfer System shall be operable whenever the diesel generator it supplies is required to be operable, except as specified below:

a. From and after the time that one fuel oil transfer pump per Diesel Generator System is made or found to be inoperable for any reason, continued reactor operation is permissible for a period not to exceed 60 days; provided that the remaining fuel oil transfer pumps are demonstrated to be operable immediately and weekly thereafter.

L6

b. From and after the time that only two fuel oil transfer pumps per Diesel Generator System are operable, continued reactor opertion is permissible for a period not to exceed 30 days total per pair of diesels, provided that the remaining fuel oil transfer pumps are demonstrated to be operable and daily thereafter.

M13

4.9 Continued

2. Once per month demonstrate each EDG the fuel oil transfer system operates to transfer fuel from its storage tank systems to the associated fuel oil day tank automatically.

[SR3.8.1.6]

G RAI 3.8.1-05

M13

219

Rev G

Specification 3.8.1

A1

3.9 (cont'd)

c. ~~From and after the time that only one fuel oil transfer pump in a Diesel Generator System is found to be operable, that Diesel Generator System shall be considered inoperable and continued reactor operation shall be in accordance with Specification 3.9.B.3 above.~~

M13

3. Whenever the diesel fuel on site for each operable pair of Diesel Generators decreases to less than 64,000 gallons as a result of operation of the Diesel Generators "to meet Technical Specification requirements," Specification 3.0.C does not apply. 48 hours are allowed to restore fuel oil storage tank quantity to a minimum of 64,000 gallons.

See ITS 3.8.3

D. <u>AC Power Operability During Cold Shutdown or Refueling Modes</u>

Whenever the reactor is in the cold shutdown or refueling mode, a minimum of one offsite power source and one Emergency Diesel Generator System, capable of supporting required emergency equipment, shall be operable whenever any work is being done which has the potential for draining the vessel, secondary containment is required, or a core or containment cooling system is required. When this condition is not met, initiate actions to suspend all work that could cause draining of the vessel, suspend core alterations and handling of irradiated fuel assemblies in the secondary containment, declare required core or containment cooling systems inoperable and immediately initiate actions to restore required AC power sources.

4.9 (cont'd)

D. <u>Not Applicable</u>

See ITS 3.8.2

C

Specification 3.8.2

3.9 (cont'd)

4.9 (cont'd)

A1

c. From and after the time that only one fuel oil transfer pump in a Diesel Generator System is found to be operable, that Diesel Generator System shall be considered inoperable and continued reactor operation shall be in accordance with Specification 3.9.B.3 above.

See ITS 3.8.1

3. Whenever the diesel fuel on site for each operable pair of Diesel Generators decreases to less than 64,000 gallons as a result of operation of the Diesel Generators "to meet Technical Specification requirements," Specification 3.0.C does not apply. 48 hours are allowed to restore fuel oil storage tank quantity to a minimum of 64,000 gallons.

See ITS 3.8.3

AC Sources-Shutdown

[LCO 3.8.2]

D. AC Power Operability During Cold Shutdown or Refueling Modes

D. Not Applicable

M3

[Applicability]

MODE 4

MODE 5

Whenever the reactor is in the cold shutdown or refueling mode, a minimum of one offsite power source and one Emergency Diesel Generator System, capable of supporting required emergency equipment, shall be operable whenever any work is being done which has the potential for draining the vessel, secondary containment is required, or a core or containment cooling system is required. When this condition is not met, initiate actions to suspend all work that could cause draining of the vessel, suspend core alterations and handling of irradiated fuel assemblies in the secondary containment, declare required core or containment cooling systems inoperable and immediately initiate actions to restore required AC power sources.

during movement of irradiated fuel assemblies in secondary containment

M1

M2

[LCO]

A2

M4

add ACTIONS Note: LCO 3.0.3 Not Applicable

C

[ACTION A & B]

OR

L1

add ACTION A Note: Enter LCO 3.8.8

A3

M5

add SR 3.8.2.1

Specification 3.8.3

ITS 3.9 (cont'd) | 4.9 (cont'd)

A1

c. From and after the time that only one fuel oil transfer pump in a Diesel Generator System is found to be operable, that Diesel Generator System shall be considered inoperable and continued reactor operation shall be in accordance with Specification 3.9.B.3 above.

See 3.8.1

[ACTION A] 3. Whenever the diesel fuel on site for each operable ~~pair of~~ Diesel Generators decreases to less than ~~64,000~~ 32,000 gallons and > 28,000 gallons ~~as a result of operation of the Diesel Generators "to meet Technical Specification requirements," Specification 3.0.C does not apply.~~ L2 48 hours are allowed to restore fuel oil storage tank quantity to a minimum of ~~64,000~~ 32,000 gallons. add ACTION F

M1

D. ~~AC Power Operability During Cold Shutdown or Refueling Modes~~

Whenever the reactor is in the cold shutdown or refueling mode, a minimum of one offsite power source and one Emergency Diesel Generator System, capable of supporting required emergency equipment, shall be operable whenever any work is being done which has the potential for draining the vessel, secondary containment is required, or a core or containment cooling system is required. When this condition is not met, initiate actions to suspend all work that could cause draining of the vessel, suspend core alterations and handling of irradiated fuel assemblius in the secondary containment, declare required core or containment cooling systems inoperable and immediately initiate actions to restore required AC power sources.

D. Not Applicable

See 3.8.2

Specification 3.8.4

DC Sources - Operating

3.9 (cont'd)

E. Station Batteries

[LCO 3.8.4] 1. The reactor shall not be made critical unless both station batteries and associated chargers and DC buses are in service, except as specified below.

Applicability MODES 1, 2 and 3 — M1

a. Division 1 and Division 2 125 VDC electrical power subsystems — LA1

OPERABLE — A5

See ITS: 3.8.7

[ACTION A] 2. During power operation, if one battery becomes unavailable, repairs shall be made immediately, and continued reactor operation is permissible for a period not to exceed 7 days total/calendar-month provided that:

inoperable

8 hrs — M2

125 VDC electrical power subsystem — L4

[RA A.1]

add ACTION B — M6 — L7

A2 — a. The other battery including its battery charger and distribution system is operable. — L3

b. Pilot cell voltage, specific gravity, and overall voltage and temperature is measured immediately and daily thereafter for the operable battery. — L1

c. The availability of the unaffected Emergency Diesel Generator System shall be demonstrated in accordance with Specification 4.9.B.5.

DC Sources - Operating

4.9 (cont'd)

E. Station Batteries

[SR 3.8.4.1] 1. Every week the specific gravity, voltage and temperature of the pilot cell and overall battery voltage shall be measured.

See ITS: 3.8.6

add limits — M3

2. Every three months the following measurements shall be made: a) voltage of each cell to nearest 0.01 v, b) specific gravity of each cell, c) temperature of every fifth cell.

See ITS: 3.8.6

A1 — G — RAI 3.8.4-2

[LCO 3.8.6] A2 Battery Cell Parameters

3.9 (cont'd)

B. Station Batteries

See ITS: 3.8.4

M1 APPLICABILITY

1. The reactor shall not be made critical unless both station batteries and associated chargers and d-c buses are in service, except as specified below.

See ITS: 3.8.7

2. During power operation, if one battery becomes unavailable, repairs shall be made immediately and continued reactor operation is permissible for a period not to exceed 7 days total/calendar-month provided that:

See ITS: 3.8.4

a. The other battery including its battery charger, and distribution system is operable.

b. Pilot cell voltage, specific gravity, and overall voltage and temperature is measured immediately and daily thereafter for the operable battery.

c. The availability of the unaffected Emergency Diesel Generator System shall be demonstrated in accordance with Specification 4.9.B.5.

A1

4.9 (cont'd)

B. Station Batteries

M4

RAI 3.8.6-1

M5 [SR 3.8.6.1]

1. Every week the specific gravity, voltage and temperature of the pilot cell and overall battery voltage shall be measured.

float

electrolyte level

L2

See ITS: 3.8.4

within Table 3.8.6-1 Category A limits

M5 [SR 3.8.6.2]

2. Every three months the following measurements shall be made: a) voltage of each cell to nearest 0.01 v, b) specific gravity of each cell, c) temperature of every fifth cell.

within Table 3.8.6-1 Category B limits

float

A3

M4

electrolyte level of each cell

M6 [SR 3.8.6.3]

L A1

Average

representative cells

L3

M5 add Table 3.8.6-1

A4 add Note to ACTIONS

M2 add proposed ACTION A

M3 add proposed ACTION B

Revising

Specification 3.8.7 (A1)

Distribution Systems - Operating

~~3.9 (cont'd)~~

[3.8.7] E. Station Batteries

[Applicability] [L to 3.8.7]

1. The reactor shall not be made critical unless both station batteries and associated chargers and d-c buses are in service except as specified below.

(M1) MODES 1,2 and 3

See ITS: 3.8.4

OPERABLE (A2)

(LA) Division 1 and 2 125 VDC electrical power distribution subsystems

2. During power operation, if one battery becomes unavailable, repairs shall be made immediately, and continued reactor operation is permissible for a period not to exceed 7 days total/calendar-month provided that:

a. The other battery including its battery charger, and distribution systems is operable.

b. Pilot cell voltage, specific gravity, and overall voltage and temperature is measured immediately and daily thereafter for the operable battery.

c. The availability of the unaffected Emergency Diesel Generator System shall be demonstrated in accordance with Specification 4.9.B.5.

(L1) add ACTION B

add ACTION C (L2) (M3)

(A3) add ACTION D for loss of Function

See ITS: 3.8.4

~~4.9~~ (cont'd)

[3.8.7] ~~E.~~ Station Batteries

see ITS: 3.8.4 3.8.6

1. Every week the specific gravity, voltage and temperature of the pilot cell and overall battery voltage shall be measured.

2. Every three months the following measurements shall be made: a) voltage of each cell to nearest 0.01 v, b) specific gravity of each cell, c) temperature of every fifth cell.

See ITS: 3.8.6

add SR 3.8.7.1 for DC buses (M2)

Page 2 of 2

Specification 3.8.4

A1

L8 [NOTE to SR 3.8.4.3]

G

TSTF-283

A4

3.9 (cont'd)

3. From and after the time that both batteries are made or found to be inoperable for any reason, the reactor shall be in a cold condition within 24 hrs.

4.9 (cont'd)

[SR 3.8.4.3] 3. Once every 24 months, during shutdown, each station battery shall be subjected to a service (duty cycle) test.[1]

[NOTE to SR 3.8.4.4] L8

G

[SR 3.8.4.4] 4. Once every 60 months, during shutdown, each battery shall be subjected to a performance test (or modified performance test). This test shall verify that the battery capacity is at least 80% of the manufacturer's rating.

[SR 3.8.4.4] 5. Accelerated performance testing (or modified performance test) shall be conducted on any battery:

[SR 3.8.4.4 2nd Surveillance]

L A2

Shows degradation

a) Annually if capacity drops more than 10% from its previous performance test (or modified performance test).

b) Annually if capacity is below 90% of manufacturer's rating.

c) Annually if it has reached 85% of its service life with capacity < 100% of manufacturer's rating.

[SR 3.8.4.4 3rd Frequency]

d) Once every 24 months if it has reached 85% of its service life with capacity ≥ 100% of the manufacturer's rating.

L A3

[SR 3.8.4.2]

6. Each battery charger shall be visually inspected weekly and a performance test conducted once every 24 months.

7. Once/month: open the battery charger output breakers one at a time and observe performance for proper operation.

L5

[SR 3.8.4.3]

M4

add load, voltage and time duration

1 A modified performance test may be performed in lieu of the battery service test.

Revision 6

Specification 3.5.1

3.9 (cont'd)

F. LPCI MOV Independent Power Supplies

4.9 (cont'd)

See ITS: 3.8.4 3.8.6

F. LPCI MOV Independent Power Supplies

1. Every week the specific gravity, voltage and temperature of each pilot cell, and overall battery voltage shall be measured and chargers and inverters shall be visually inspected.

LAB

2. Every three months the following measurements shall be made:

   a. Voltage of each cell to the nearest of 0.01v;

   b. Specific gravity of each cell;

   c. Temperature of every fifth cell.

see ITS: 3.8.6

3. Once every 24 months, each battery shall be subjected to a service (duty cycle) test.[1]

4. Once every 60 months, each battery shall be subjected to a performance test (or modified performance test). This test shall verify that the battery capacity is at least 80% of the manufacturer's rating.

see ITS: 3.8.4

see ITS: 3.8.4

1. A modified performance test may be performed in lieu of the battery service test.

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Specification 3.8.4

A1

RAI 3.8.4-2

~~3.9 (cont'd)~~

DC Sources - Operating

~~E.~~ LPCI MOV Independent Power Supplies

4.9 (cont'd)

F. LPCI MOV Independent Power Supplies

1. Every week the specific gravity, voltage and temperature of each pilot cell, and overall battery voltage shall be measured, and chargers and inverters shall be visually inspected within limits. [SR 3.8.4.1] LA3 M3 see ITS 3.5.1

2. Every three months the following measurements shall be made: see ITS 3.8.6

   a. Voltage of each cell to the nearest of 0.01v;

   b. Specific gravity of each cell;

   c. Temperature of every fifth cell.

3. Once every 24 months, each battery shall be subjected to a service (duty cycle) test.[1] [SR 3.8.4.3]

4. Once every 60 months, each battery shall be subjected to a performance test (or modified performance test). This test shall verify that the battery capacity is at least 80% of the manufacturer's rating. [SR 3.8.4.4]

[SR 3.8.4.3]

1. A modified performance test may be performed in lieu of the battery service test.

Specification 3.8.6

A1

3.9 (cont'd)

F. LPCI MOV Independent Power Supplies

4.9 (cont'd)

Battery Cell Parameters

F. LPCI MOV Independent Power Supplies

[SR 3.8.6.1]

L2

1. Every week the specific gravity, voltage and temperature of each pilot cell, and overall battery voltage shall be measured, and chargers and inverters shall be visually inspected.

M4

electrolyte level

See ITS: 3.8.4

G

RAI 3.8.6-1

M5

within Table 3.8.6-1 Category A limits

2. Every three months the following measurements shall be made:

[SR 3.8.6.2]

M5

within Table 3.8.6-1 Category B limits

float

a. Voltage of each cell to the nearest of 0.01 v;

A3

M4

L3

b. Specific gravity of each cell;

electrolyte level of each cell

average

M6

[SR 3.8.6.3]

c. Temperature of every fifth cell

LA1

3. Once every 24 months, each battery shall be subjected to a service (duty cycle) test.[1]

representative cells

L3

4. Once every 60 months, each battery shall be subjected to a performance test (or modified performance test). This test shall verify that the battery capacity is at least 80% of the manufacturer's rating.

See ITS: 3.8.4

See ITS: 3.8.4

1. A modified performance test may be performed in lieu of the battery service test.

Specification 3.5.1

[add MODE 3 Applicability] M8

3.5 ECCS and RCIC
3.5.1 ECCS - Operating

3.9 (con'd)

F. LPCI MOV Independent Power Supplies

A2

[SR 3.5.1.5]

1. Reactor shall not be made critical unless both independent power supplies, including the batteries, inverters and chargers and their associated buses (MCC-155 and MCC-168) are in service, except as specified below. L9

[ACTION A]

2. During power operation, if one independent power supply becomes unavailable, repairs shall be made immediately and continued reactor operation is permissible for a period not to exceed 7 days unless the unavailable train is made operable sooner. From and after the date one of the independent power supplies is made or found to be inoperable for any reason, the following would apply:

a. The other independent power supply including its charger, inverter, battery and associated bus is operable. A12

b. Pilot cell voltage, specific gravity and temperature and overall battery voltage are measured immediately and weekly thereafter for the operable independent power supply battery.

c. The inoperable independent power supply shall be isolated from its associated LPCI MOV bus, and this bus will be manually switched to its alternate power source. L10

L11

See JTS: 3.8.4
3.8.6

JAFNPP

4.9 (con'd)

See JTS: 3.8.4
JTS 3.8.6

F. LPCI MOV Independent Power Supplies

5. Accelerated performance testing (or modified performance test) shall be conducted on any battery:

a) Annually if capacity drops more than 10% from its previous performance test (or modified performance test).

b) Annually if capacity is below 90% of manufacturer's rating.

c) Annually if it has reached 85% of its service life with capacity < 100% of manufacturer's rating.

d) Once every 24 months if it has reached 85% of its service life with capacity ≥ 100% of the manufacturer's rating.

SR 3.5.1.12 6. Each battery charger and inverter shall be visually inspected weekly and a performance test conducted once every 24 months. LA8

[SR 3.5.1.5] 7. Once/month: open the battery charger A-C input breakers one at a time and observe performance for proper operation.

add inverter output voltage M7

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Specification 3.8.4

A1

DC Sources - Operating

M1

Applicability Modes 1, 2 and 3

JAFNPP

DC Sources - Operating

3.9 (con'd)

F. LPCI MOV Independent Power Supplies

See ITS: 3.5.1

1. Reactor shall not be made critical unless both independent power supplies, including the batteries, inverters and chargers and their associated buses (MCC-155 and MCC-165) are in service, except as specified below.

[LCO 3.8.4]

LA1

OPERABLE

A6

2. During power operation, if one independent power supply becomes unavailable, repairs shall be made immediately and continued reactor operation is permissible for a period not to exceed 7 days unless the unavailable train is made operable sooner. From and after the date one of the independent power supplies is made or found to be inoperable for any reason, the following would apply:

[ACTION C]

A3

A1

Declare the associated LPCI subsystem inoperable

L7

a. The other independent power supply including its charger, inverter, battery and associated bus is operable.

L3

L1

b. Pilot cell voltage, specific gravity and temperature and overall battery voltage are measured immediately and weekly thereafter for the operable independent power supply battery.

c. The inoperable independent power supply shall be isolated from its associated LPCI MOV bus, and this bus will be manually switched to its alternate power source.

L6

4.9 (con'd)

F. LPCI MOV Independent Power Supplies

5. Accelerated performance testing (or modified performance test) shall be conducted on any battery:

[SR 3.8.4.4]

LA2

Shows degration

a) Annually if capacity drops more than 10% from its previous performance test (or modified performance test).

b) Annually if capacity is below 90% of manufacturer's rating.

LA2

c) Annually if it has reached 85% of its service life with capacity < 100% of manufacturer's rating.

[SR 3.8.4.4 2nd Surveillance]

G RAI 3.8.4-4

d) Once every 24 months if it has reached 85% of its service life with capacity ≥ 100% of the manufacturer's rating.

[SR 3.8.4.4 3rd Frequency]

See ITS: 3.5.1

LA3

6. Each battery charger and inverter shall be visually inspected weekly and a performance test conducted once every 24 months.

LA4

7. Once/month: open the battery charger A-C input breakers one at a time and observe performance for proper operation.

See ITS: 3.5.1

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Revision 6

Specification 3.8.6 (A1)

[LCO 3.8.6] (A2)

3.9 (con'd)

Battery Cell Parameters

6. LPCI MOV Independent Power Supplies

See ITS: 3.8.4

(M1) (APPLICABILITY)

1. Reactor shall not be made critical unless both independent power supplies, including the batteries, inverters and chargers and their associated buses (MCC-155 and MCC-165) are in service, except as specified below.

2. During power operation, if one independent power supply becomes unavailable, repairs shall be made immediately and continued reactor operation is permissible for a period not to exceed 7 days unless the unavailable train is made operable sooner. From and after the date one of the independent power supplies is made or found to be inoperable for any reason, the following would apply:

See ITS: 3.8.4

a. The other independent power supply including its charger, inverter, battery and associated bus is operable.

b. Pilot cell voltage, specific gravity and temperature and overall battery voltage are measured immediately and weekly thereafter for the operable independent power supply battery.

c. The inoperable independent power supply shall be isolated from its associated LPCI MOV bus, and this bus will be manually switched to its alternate power source.

4.9 (con'd)

F. LPCI MOV Independent Power Supplies

5. Accelerated performance testing (or modified performance test) shall be conducted on any battery:

a) Annually if capacity drops more than 10% from its previous performance test (or modified performance test).

b) Annually if capacity is below 90% of manufacturer's rating.

c) Annually if it has reached 85% of its service life with capacity < 100% of manufacturer's rating.

d) Once every 24 months if it has reached 85% of its service life with capacity ≥ 100% of the manufacturer's rating.

6. Each battery charger and inverter shall be visually inspected weekly and a performance test conducted once every 24 months.

7. Once/month: open the battery charger A-C input breakers one at a time and observe performance for proper operation.

add proposed ACTION A (M2)

Specification 3.3.8.2

see ITS: 3.8.4

3.9 (cont'd)

3. From and after the time both power supplies are made or found inoperable the reactor shall be brought to cold condition within 24 hours.

[3.3.8.2] 2.B. REACTOR PROTECTION SYSTEM (RPS) ELECTRICAL PROTECTION ASSEMBLIES

[LCO 3.3.8.2] Two RPS electrical protection assemblies for each inservice RPS MG set and inservice alternate source shall be operable except as specified below. add Applicability L1

[ACTION A] 1. With one RPS electrical protection assembly for an inservice RPS MG set or an inservice alternate power supply inoperable, restore the inoperable channel to operable status within 72 hours or remove the associated RPS MG set or alternate power supply from service. Remove A2

[ACTION B] 2. With two RPS electrical protection assemblies for an inservice RPS MG set or an inservice alternate power supply inoperable, restore at least one to operable status within 30 minutes or remove the associated RPS MG set or alternate power supply from service. 1 hour L2

L3 M2

add proposed ACTION C

add proposed ACTION D M1

4.9 (cont'd)

POWER MONITORING

A1

[3.3.8.2] B. REACTOR PROTECTION SYSTEM ELECTRICAL PROTECTION ASSEMBLIES

The RPS electrical protection assemblies instrumentation shall be determined operable by:

[SR 3.3.8.2.1 Note]

1. Performing a channel functional test [SR 3.3.8.2.1] each time the plant is in cold shutdown for a period of more than 24 hours, unless performed in the previous 6 months. 184 days L5

2. Once per 24 months, demonstrating the operability of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a channel calibration [SR 3.3.8.2.3] including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers [SR 3.3.8.2.4] and verifying the following setpoints: actual L4

Allowable Values A3

[SR 3.3.8.2.2]

RPS MG SET SOURCE	
OVER-VOLTAGE	≤132V ≤4 second Time Delay
UNDER-VOLTAGE	≥112.5V for "A"Channel ≥113.9V for "B" Channel ≤4 second Time Delay
UNDER-FREQUENCY	≥57Hz ≤4 second Time Delay

(continued on page 222d)

Page 1 of 2

Specification 3.5.1

A1

RA1 3.5.1-BS1

3.9 (cont'd)

A8

M4

MODE 3 with 12 hours

4.9 (cont'd)

[ACTION B] From and after the time both power supplies are made or found inoperable the reactor shall be brought to cold condition within 24 hours. 36 L4

add ACTION H A8

G. REACTOR PROTECTION SYSTEM ELECTRICAL PROTECTION ASSEMBLIES

Two RPS electrical protection assemblies for each inservice RPS MG set and inservice alternate source shall be operable except as specified below:

1. With one RPS electrical protection assembly for an inservice RPS MG set or an inservice alternate power supply inoperable, restore the inoperable channel to operable status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

2. With two RPS electrical protection assemblies for an inservice RPS MG set or an inservice alternate power supply inoperable, restore at least one to operable status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

See ITS: 3.3.8.2

G. REACTOR PROTECTION SYSTEM ELECTRICAL PROTECTION ASSEMBLIES

The RPS electrical protection assemblies instrumentation shall be determined operable by:

1. Performing 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. Once per 24 months, demonstrating the operability of over-voltage, under-voltage and under-frequency 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:

	RPS MG SET SOURCE
OVER-VOLTAGE	≤132V ≤4 second Time Delay
UNDER-VOLTAGE	≥112.5V for "A"Channel ≥113.9V for "B" Channel ≤4 second Time Delay
UNDER-FREQUENCY	≥57Hz ≤4 second Time Delay

(continued on page 222d)

Specification 3.8.4

A1

~~3.9 (cont'd)~~

~~4.9 (cont'd)~~

3. From and after the time both power supplies are made or found inoperable the reactor shall be brought to cold condition within 24 hours. L7

G RAI 3.8.4-4

G. REACTOR PROTECTION SYSTEM ELECTRICAL PROTECTION ASSEMBLIES

Two RPS electrical protection assemblies for each inservice RPS MG set and inservice alternate source shall be operable except as specified below:

1. With one RPS electrical protection assembly for an inservice RPS MG set or an inservice alternate power supply inoperable, restore the inoperable channel to operable status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

2. With two RPS electrical protection assemblies for an inservice RPS MG set or an inservice alternate power supply inoperable, restore at least one to operable status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

see ITS: 3.3.8.2

G. REACTOR PROTECTION SYSTEM ELECTRICAL PROTECTION ASSEMBLIES

The RPS electrical protection assemblies instrumentation shall be determined operable by:

1. Performing 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. Once per 24 months, demonstrating the operability of over-voltage, under-voltage and under-frequency 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:

	RPS MG SET SOURCE
OVER-VOLTAGE	≤132V ≤4 second Time Delay
UNDER-VOLTAGE	≥112.5V for "A"Channel ≥113.9V for "B" Channel ≤4 second Time Delay
UNDER-FREQUENCY	≥57Hz ≤4 second Time Delay

Amendment No. ~~41, 76, 189, 233~~ 245

(continued on page 222d)

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Revision 6

Specification 3.3.8.2

~~3.9.G (cont'd)~~

~~3. With the reactor in the RUN mode, at least one (1) RPS division shall be powered from the MG set except as specified below:~~

~~With both RPS divisions powered from the alternate sources, at least one division power source shall be restored to a MG set with operable electrical protection assemblies within seven (7) days or the reactor shall be brought to the cold condition within the subsequent 24 hours.~~

L4

4.9.G (cont'd)

	ALTERNATE SOURCE
OVER-VOLTAGE	≤132V ≤4s Time Delay
UNDER-VOLTAGE	≥~~100~~V ≤4s Time Delay
UNDER-FREQUENCY	≥57HZ ≤4s Time Delay

[SR 3.3.8.2.3]

A1

109.9

M3

Page 2 of 2

Specification 3.9.1

A1

3.10 LIMITING CONDITIONS FOR OPERATION

3.10 CORE ALTERATIONS

Applicability:

Applies to fuel handling and core reactivity limitations.

Objective:

To assure that core reactivity is within the capability of the control rods and to prevent criticality during refueling.

Specification:

[3.9.1] [LOO 3.9.1]

A. Refueling Interlocks

Equipment A2

Applicability: During in-vessel fuel movement with equipment associated with interlocks when the reactor mode switch is in the refuel position L2

1. The Reactor Mode Switch shall be locked in the Refuel position during core alterations and the refueling interlocks shall be operable except as permitted by Specifications 3.10.A.5, 3.10.A.6, 3.10.A.7 and 3.10.D. A3

L2

See ITS 3.9.2

See ITS 3.9.3

2. Fuel shall not be loaded into the reactor core unless all control rods are fully inserted except as permitted by Specification 3.10.A.7.

See ITS 3.10.6

equipment A2

add proposal ACTION A L2

4.10 SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS

Applicability:

Applies to the periodic testing of those interlocks and instruments used during refueling and core alterations.

Objective:

To verify the operability of instrumentation and interlocks used in refueling and core alterations.

Specification:

[3.9.1]

A. Refueling Interlocks

Equipment

L3 A2

1. Prior to any fuel handling, with the head off the reactor vessel, the refueling interlocks shall be functionally tested. They shall also be tested at weekly intervals thereafter until no longer required and following any repair work associated with the interlocks. [SR 3.9.1.1] L1

equipment

2. Whenever the reactor mode switch is in the Refuel position and refueling interlocks are bypassed, one licensed operator and a member of the reactor analyst department shall verify that the control cell contains no fuel before the corresponding control rod is withdrawn.

See ITS 3.10.6

add SR 3.9.1.1 List of Refueling Interlocks M1

Page 1 of 2

Specification 3.9.2 (A1)

3.10 LIMITING CONDITIONS FOR OPERATION

3.10 CORE ALTERATIONS

Applicability:

Applies to fuel handling and core reactivity limitations.

Objective:

To assure that core reactivity is within the capability of the control rods and to prevent criticality during refueling.

Specification:

3.9.2 Refuel Position One-Rod-Out Interlock

A. Refueling Interlocks

[SR 3.9.2.1] [LCO 3.9.2]

1. The Reactor Mode Switch shall be locked in the Refuel position during core alterations and the refueling interlocks shall be operable except as permitted by Specifications 3.10.A.5, 3.10.A.6, 3.10.A.7 and 3.10.D. (A3)

M1 — every 12 hours — verified — one rod-out (M2)

2. Fuel shall not be loaded into the reactor core unless all control rods are fully inserted except as permitted by Specification 3.10.A.7.

See ITS: 3.9.3.

See ITS: 3.10.6

Applicability: MODES with the reactor mode switch in the refuel position and any control rod withdrawn (L1)

add Proposal Action A

4.10 SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS

Applicability:

Applies to the periodic testing of those interlocks and instruments used during refueling and core alterations.

Objective:

To verify the operability of instrumentation and interlocks used in refueling and core alterations.

Specification:

A. Refueling Interlocks

[SR 3.9.2.2]

Note: Not required to be performed until 1 hour after any control rod is withdrawn (L2)

1. Prior to any fuel handling, with the head off the reactor vessel, the refueling interlocks shall be functionally tested. They shall also be tested at weekly intervals thereafter until no longer required and following any repair work associated with the interlocks. (L4) (A2) one-rod-out (L3)

2. Whenever the reactor mode switch is in the Refuel position and refueling interlocks are bypassed, one licensed operator and a member of the reactor analyst department shall verify that the control cell contains no fuel before the corresponding control rod is withdrawn.

See ITS: 3.10.6

Page 1 of 1

Specification 3.9.3

(A1)

3.10 LIMITING CONDITIONS FOR OPERATION

3.10 CORE ALTERATIONS

Applicability:

Applies to fuel handling and core reactivity limitations.

Objective:

To assure that core reactivity is within the capability of the control rods and to prevent criticality during refueling.

Specification:

[3.9.3] ~~A. Refueling Interlocks~~ Control Rod Position

1. The Reactor Mode Switch shall be locked in the Refuel position during core alterations and the refueling interlocks shall be operable except as permitted by Specifications 3.10.A.5, 3.10.A.6, 3.10.A.7 and 3.10.D.

[LCO 3.9.3] 2. ~~Fuel shall not be loaded into the reactor core unless all control rods are fully inserted except as permitted by Specification 3.10.A.7.~~

See ITS: 3.10.6

[Applicability] When loading fuel assemblies into the core.

add proposed ACTION A (A2)

4.10 SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS

Applicability:

Applies to the periodic testing of those interlocks and instruments used during refueling and core alterations.

Objective:

To verify the operability of instrumentation and interlocks used in refueling and core alterations.

Specification:

[3.9.3] A. ~~Refueling Interlocks~~

1. Prior to any fuel handling, with the head off the reactor vessel, the refueling interlocks shall be functionally tested. They shall also be tested at weekly intervals thereafter until no longer required and following any repair work associated with the interlocks.

See ITS: 3.9.1

2. Whenever the reactor mode switch is in the Refuel position and refueling interlocks are bypassed, one licensed operator and a member of the reactor analyst department shall verify that the control cell contains no fuel before the corresponding control rod is withdrawn.

See ITS: 3.10.6

add SR 3.9.3.1 (M1)

Page 1 of 1

Specification 3.10.6 (A1)

3.10 LIMITING CONDITIONS FOR OPERATION

3.10 CORE ALTERATIONS

Applicability:

Applies to fuel handling and core reactivity limitations.

Objective:

To assure that core reactivity is within the capability of the control rods and to prevent criticality during refueling.

Specification:

A. Refueling Interlocks (See ITS 3.9.1)

1. The Reactor Mode Switch shall be locked in the Refuel position during core alterations and the refueling interlocks shall be operable except as permitted by Specifications 3.10.A.5, 3.10.A.6, 3.10.A.7 and 3.10.D.

(See ITS 3.9.3)

2. Fuel shall not be loaded into the reactor core unless all control rods are fully inserted except as permitted by Specification 3.10.A.7. [Spiral reload sequence]

[LCO 3.10.6.c]

4.10 SURVEILLANCE REQUIREMENTS

(A1)

4.10 CORE ALTERATIONS

Applicability:

Applies to the periodic testing of those interlocks and instruments used during refueling and core alterations.

Objective:

To verify the operability of instrumentation and interlocks used in refueling and core alterations.

Specification:

A. Refueling Interlocks

1. Prior to any fuel handling, with the head off the reactor vessel, the refueling interlocks shall be functionally tested. They shall also be tested at weekly intervals thereafter until no longer required and following any repair work associated with the interlocks.

See ITS: 3.9.1

2. Whenever the reactor mode switch is in the Refuel position and refueling interlocks are bypassed, one licensed operator and a member of the reactor analyst department shall verify that the control cell contains no fuel before the corresponding control rod is withdrawn.

(LA2)

[SR 3.10.6.1]

[LCO 3.10.6.a]

(A5)

(24 hours)

(M1)

Page 3 of 3

Specification 3.9.1

A1

3.10 (cont'd)

3. The fuel grapple hoist load switch shall be set at less than or equal to 650 lbs.

LA1

4. If the frame-mounted auxiliary hoist, the mono-rail-mounted auxiliary hoist, or the service platform hoist is to be used for handling fuel with the head off the reactor vessel, the hoist load switch on the hoist to be used shall be set at less than or equal to 400 lbs.

A4

5. Any number of control rods may be withdrawn or removed from the reactor core provided:

   a. The reactor mode switch is locked in the "Refuel" position; and

   b. The fuel assemblies situated in the control cell of the control rod to be withdrawn have been removed; and

   c. Refueling interlocks associated with all control cells containing fuel are operable. Refueling interlocks associated with a specific control rod may be bypassed after the fuel assemblies in the control cell have been removed; and

   d. Fuel on-loading operations shall be suspended until Specification 3.10.A.2 is satisfied.

See ITS: 3.10.6

Page 2 of 2

Specification 3.10.6

A1

A1

JAFNPP

3.10.6 Multiple Control Rod Withdrawal – Refueling

3.10 (cont'd)

3. The fuel grapple hoist load switch shall be set at less than or equal to 650 lbs.

4. If the frame-mounted auxiliary hoist, the mono-rail-mounted auxiliary hoist, or the service platform hoist is to be used for handling fuel with the head off the reactor vessel, the hoist load switch on the hoist to be used shall be set at less than or equal to 400 lbs.

see ITS: 3.9.1

A6

add Applicability

[LCO 3.10.6]

5. Any number of control rods may be withdrawn or removed from the reactor core provided:

M2

add SRs 3.10.6.2 3.10.6.3

L1

a. ~~The reactor mode switch is locked in the "Refuel" position; and~~

[LCO 3.10.6.a]

b. The fuel assemblies situated in the control cell of the control rod to be withdrawn have been removed; and

L1

c. ~~Refueling interlocks associated with all control cells containing fuel are operable;~~

[LCO 3.10.6.b]

[LCO 3.10.6.a]

Refueling interlocks associated with a specific control rod may be bypassed after the fuel assemblies in the control cell have been removed; and

A4

add LCO 3.10.6.b

[LCO 3.10.6.c]

d. ~~Fuel on-loading operations shall be suspended until Specification 3.10.A.2 is satisfied.~~

A3

add ACTION A

A2

Amendment No. 115

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Specification 3.10.6

AT

3.10 (cont'd)

[LCO 3.10.6]

6. A spiral off-load may be conducted provided:

a. Refueling interlocks are operable for any control cell which contains fuel; and

LI

add Applicability

A6

b. Refueling interlocks are bypassed only for those control cells which contain no fuel; and

[LCO 3.10.6.a]

c. Fuel is removed from a control cell before its control rod is withdrawn.

A2

add ACTION A

[LCO 3.10.6.3]

7. A spiral onload may be conducted provided:

add Applicability

A6

[LCO 3.10.6]

a. Refueling interlocks may be bypassed only for those control cells which contain no fuel; and

[LCO 3.10.6.c]

b. The spiral onload may commence at either the core center around a "dunking type detector" or, around one of the source range monitors. (Placement of the "dunking type detector" in the core center does not violate the intent of the spiral onloading pattern. Fuel may be loaded into this bundle location when the dunking detector has been removed.); and

Fuel assemblies shall only be loaded in compliance with an approved

LA1

[LCO 3.10.6.b]

c. Before loading fuel into an empty control cell, its control rod is fully inserted, and

[LCO 3.10.6]

the refueling interlocks for that control rod are operable; and

A2

d. Refueling interlocks are operable for any control cell which contains fuel.

add ACTION A

LI

Amendment No. ~~54~~, ~~87~~, 115

Page 2 of 3

Source Range Monitor (SRM) Instrumentation

Specification 3.3.1.2

A1

add proposed footnote (b) to Table 3.3.1.2-1

L2

JAFNPP

~~3.10 (cont'd)~~

4.10 (cont'd)

~~B. Core Monitoring~~

Note 1 to SR 3.3.1.2.2

During core alterations two SRM's shall be operable, one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant. For an SRM to be considered operable, the following conditions shall be satisfied:

SR 3.3.1.2.2

1. The SRM shall be inserted to the normal operating level. (Use of special movable, dunking type detectors during initial fuel loading and major core alterations in place of normal detectors is permissable as long as the detector is connected into normal SRM circuit).

Table 3.3.1.2-1 footnote (c)

LA1

2. The SRM shall have a minimum of 3 counts/sec with all rods fully inserted in the core except as noted in 3 and 4 below.

SR 3.3.1.2.4

3. Prior to spiral unloading, the SRM's shall have an initial count rate of 3 CPS. During spiral unloading, the count rate of the SRM's may drop below 3 CPS.

L2

M8

add proposed ACTION E

add every 12 hours

M3

SR 3.3.1.2.5

Prior to making alterations to the core the SRM's shall be functionally tested and checked for neutron response. Fuel may be on-loaded as described in Specification 3.10.B.4 prior to this functional test. Thereafter, the SRM's will be checked daily for response, except as specified in Specification 3.10.B.3 and 4.

SR 3.3.1.2.1

M11

M10

every 7 days

M10

M11

every 12 hours

M6

add proposed SR 3.3.1.2.2 Frequency

add proposed SR 3.3.1.2.2.a and Note 2 to SR 3.3.1.2.2

L2

add proposed signal to noise ratio

M7

RAI 3.3.1.2-1

add proposed SR 3.3.1.2.7 for MODE 5

M9

Page 2 of 3

Revision F

Specification 3.3.1.2

A1

L3

3.10 (cont'd)

Note to SR 3.3.1.2.4

4. During Spiral reload, SRM operability will be verified by using a portable external source every 12 hours until enough fuel is loaded to maintain 3 CPS. Alternatively, a maximum of four fuel assemblies will be loaded in different cells containing control blades around each SRM to obtain the required 3 CPS. Until these assemblies have been loaded in a given quadrant, it is not necessary for the SRM in that quadrant to indicate the minimum count rate of 3 CPS. The loading of fuel near the SRM's does not violate the intent of the spiral reloading pattern.

Specification 3.7.7 (A1)

3.10 (cont'd)

[3.7.7] Spent Fuel Storage Pool Water Level

[Applicability] (L1)

Whenever irradiated fuel is stored in the spent fuel storage pool, the pool water level shall be maintained at a minimum level of ~~18 ft.~~ (M1) (LA1)

[L (03.7.7)]

D. Control Rod and Control Rod Drive Maintenance

1. Two control rods may be withdrawn from the reactor core to perform maintenance provided:

   a. The Reactor Mode Switch is locked in the Refuel position and all refueling interlocks are operable except for those necessary to perform the demonstration and maintenance described in Specification 4.10.D.1.

   b. Control rods immediately face and diagonally adjacent to the control rods to be withdrawn are fully inserted, electrically disarmed and sufficient margin to criticality demonstrated.

   c. Control rods to be withdrawn are separated by three or more cells in any direction. (This specification does not apply to the control rods used to perform the demonstration required by Specification 3.10.D.1.b.)

See ITS: 3.10.5

4.10 (cont'd)

[3.7.7] Spent Fuel Storage Pool Water Level (L1)

Whenever irradiated fuel is stored in the spent fuel storage pool, the pool water level shall be recorded ~~daily~~. every 7 days (L2)

[SR 3.7.7.1]

D. Control Rod and Control Rod Drive Maintenance

1. When two control rods are withdrawn from the reactor core for maintenance, the following surveillance shall be performed:

   a. If the reactor vessel head is removed, specification 4.10.A.1 shall be satisfied.

   b. Demonstrate that the reactor core can be maintained subcritical with a margin of 0.38 percent Δk at any time during the maintenance with the analytically determined strongest worth operable control rod fully withdrawn. This margin shall be demonstrated after Specification 3.10.D.1 has been satisfied.

See ITS: 3.10.5

add proposed Action A (M2)

add ACTION A Note (L3)

Page 1 of 1

Specification 3.10.5 (A1)

see IRS: 3.7.7

[3.10.5]

3.10 (cont'd)

C. Spent Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the spent fuel storage pool, the pool water level shall be maintained at a minimum level of 33 ft.

(L1) add proposed LCOs LCO 3.10.5.d

D. Control Rod and Control Rod Drive Maintenance (Single CRD Removal-Refueling)

1. ~~Two~~ one (M1) control rods may be withdrawn from the reactor core to perform maintenance provided: [LCO 3.10.5]

a. The Reactor Mode Switch is locked in the Refuel position and all refueling interlocks are operable except for those necessary to perform the demonstration and maintenance described in Specification 4.10.D.1. [LCO 3.10.5]

b. Control rods ~~immediately face and diagonally~~ adjacent to the control rods to be withdrawn are fully inserted, electrically disarmed and sufficient margin to criticality demonstrated. [LCO 3.10.5.b] (L2)

(M2) five by five array centered on the withdrawn control rod

c. ~~Control rods to be withdrawn are separated by three or more cells in any direction. (This specification does not apply to the control rods used to perform the demonstration required by Specification 3.10.D.1.b.)~~

(M1) add LCO 3.10.5.a

4.10 (cont'd)

C. Spent Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the spent fuel storage pool, the pool water level shall be recorded daily.

D. Control Rod and Control Rod Drive Maintenance

1. When ~~two~~ one (M1) control rods are withdrawn from the reactor core for maintenance, the ~~following surveillance shall be performed:~~ (A1)

a. ~~If the reactor vessel head is removed, specification 4.10.A.1 shall be satisfied.~~ (L1)

b. Demonstrate that the reactor core can be maintained subcritical with a margin of 0.38 percent Δk at any time during the maintenance with the analytically determined strongest worth operable control rod fully withdrawn. This margin shall be demonstrated after Specification 3.10.D.1 has been satisfied. [SR 3.10.5.4] (L2)

Perform SR 3.1.1.1

add SR 3.10.5.1

(M2) add SR 3.10.5.2

add SR 3.10.5.3 SR 3.10.5.5 (M3)

(A2) add LCO 3.10.5.c

add LCO allowance to suspend LCO 3.3.1.1 LCO 3.3.1.2 LCO 3.9.5 (L3)

Amendment No. ~~54~~, 115

add Applicability (A4)

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Specification 3.10.5

A1

~~3.10 (cont'd)~~

2. More than two control rods may be withdrawn from the reactor core to perform maintenance provided:

   a. Specification 3.10.A.5 is satisfied

A3

4.10 (cont'd)

2. When more than two control rods are withdrawn from the reactor core for maintenance, the following surveillance shall be performed:

   a. Specifications 4.10.A.1 and 4.10.A.2 shall be satisfied.

add proposed ACTION A

M4

Specification 3.3.7.1 (A1)

3.11 LIMITING CONDITIONS FOR OPERATION

3.11 ADDITIONAL SAFETY RELATED PLANT CAPABILITIES

Applicability:

Applies to the operating status of the main control and relay rooms, and battery room ventilation and cooling. Applies to emergency service water system and intake deicing heaters.

Objective:

To assure the availability of the main control and relay room, and battery room ventilation systems, to assure the availability of the emergency service water system and intake deicing heaters, under the conditions for which the capability is an essential response to plant abnormalities.

A. Main Control Room Ventilation

1. The reactor shall not have a coolant temperature greater than 212 °F and fuel may not be handled unless both of the control room emergency ventilation air supply fans and fresh air filter trains are available for normal operation except that one emergency

[Applicability] (M1)

See ITS: 3.7.3

4.11 SURVEILLANCE REQUIREMENTS

4.11 ADDITIONAL SAFETY RELATED PLANT CAPABILITIES

Applicability:

Applies to the surveillance requirements for the main control and relay room, battery room ventilation systems, emergency service water and intake deicing heaters.

Objective:

To verify the operability or availability under conditions for which these capabilities are an essential response to plant abnormalities.

A. Main Control Room Ventilation

1. Each of the control room emergency ventilation* air supply fans and dampers shall be tested for operability every 3 months.

   The fresh air filter trains shall be tested once every 6 months as follows:

   a. Pressure drop test across each filter and the filter system.

See ITS: 3.7.3
ITS Section: 5.5

Control Room Emergency Ventilation Air Supply (CREVAS) System Instrumentation (A1)

Specification 3.7.3 A1

3.11 LIMITING CONDITIONS FOR OPERATION

3.11 ADDITIONAL SAFETY RELATED PLANT CAPABILITIES

Applicability:

Applies to the operating status of the main control and relay rooms, and battery room ventilation and cooling. Applies to emergency service water system and intake deicing heaters.

Objective:

To assure the availability of the main control and relay room, and battery room ventilation systems, to assure the availability of the emergency service water system and intake deicing heaters, under the conditions for which the capability is an essential response to plant abnormalities.

4.11 SURVEILLANCE REQUIREMENTS

4.11 ADDITIONAL SAFETY RELATED PLANT CAPABILITIES

Applicability:

Applies to the surveillance requirements for the main control and relay room, battery room ventilation systems, emergency service water and intake deicing heaters.

Objective:

To verify the operability or availability under conditions for which these capabilities are an essential response to plant abnormalities.

[3.7.3] A. Main Control Room Emergency Ventilation Air Supply (CREVAS) System

1. The reactor shall not have a coolant temperature greater than 212 °F and fuel may not be handled unless both of the control room emergency ventilation air supply fans and fresh air filter trains are available for normal operation except that one emergency LA1

[LCO 3.7.3]

[APPLICABILITY]

Two CREVAS Subsystems shall be OPERABLE.

add: MODE 2, during CORE ALTERATIONS, and during operations with a potential to drain the reactor vessel (OPDRVs) M1

ADD LCO NOTE M4

[3.7.3] B. Main Control Room Emergency Ventilation Air Supply System (CREVAS)

1. Each of the control room emergency ventilation* air supply fans and dampers shall be tested for operability every 3 months. M2

The fresh air filter trains shall be tested once every 6 months as follows:

a. Pressure drop test across each filter and the filter system.

[SR 3.7.3.2] A2

See ITS: 5.5.8

Operate each CREVAS subsystem 92 days for ≥ 15 minutes

[SR 3.7 3.1]

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Section 5.5

A1

3.11 LIMITING CONDITIONS FOR OPERATION

3.11 ADDITIONAL SAFETY RELATED PLANT CAPABILITIES

Applicability:

Applies to the operating status of the main control and relay rooms, and battery room ventilation and cooling. Applies to emergency service water system and intake deicing heaters.

Objective:

To assure the availability of the main control and relay room, and battery room ventilation systems, to assure the availability of the emergency service water system and intake deicing heaters, under the conditions for which the capability is an essential response to plant abnormalities.

A. Main Control Room Ventilation

1. The reactor shall not have a coolant temperature greater than 212 °F and fuel may not be handled unless both of the control room emergency ventilation air supply fans and fresh air filter trains are available for normal operation except that one emergency

See ITS! 3.7.3

4.11 SURVEILLANCE REQUIREMENTS

4.11 ADDITIONAL SAFETY RELATED PLANT CAPABILITIES

Applicability:

Applies to the surveillance requirements for the main control and relay room, battery room ventilation systems, emergency service water and intake deicing heaters.

Objective:

To verify the operability or availability under conditions for which these capabilities are an essential response to plant abnormalities.

See ITS! 3.7.3

A. Main Control Room Ventilation

1. Each of the control room emergency ventilation air supply fans and dampers shall be tested for operability every 3 months.

The fresh air filter trains shall be tested once every ~~6~~ 24 months as follows: L2

[5.5.B.d] a. Pressure drop test across ~~each filter and~~ the filter system. L7

is less than 5.8 inches of water at a flowrate of 900 to 1100 scfm

M2

A12 Add SR 3.0.2 and SR 3.0.3 appl. capability

CTS Amend 269

Specification 3.3.7.1

A1

3.11 (cont'd)

ventilation air supply fan and/or filter may be out of service for 14 days.

See ITS 3.7.3

[LCO 3.3.7.]

2. The main control room air radiation monitor shall be operable whenever the control room emergency ventilation air supply fans and filter trains are required to be operable by 3.11.A.1 or filtration of the control room ventilation intake air must be initiated.

M3

[ACTION A]

[APPLICABILITY]

M1

4.11 (cont'd)

b. Di-octylphtalate (DOP) test for particulate filter efficiency greater than 99% for particulate greater than 0.3 micron size.

c. Freon-112 test for charcoal filter bypass as a measure of filter efficiency of at least 99.5% for halogen removal.

2. At least once per 24 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release, that could adversely affect the ability of the charcoal to perform its intended function, in any ventilation zone communicating with the system, verify:

(1) Within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration to be less than or equal to 5 percent when tested in accordance with ASTM D3803-1989 at a temperature of 30 degrees C (86 degrees F), and a relative humidity of at least 95 percent.

(2) Within 31 days of completing 720 hours of charcoal adsorber operation, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration to be less than or equal to 5 percent when tested in accordance with ASTM D3803-1989 at a temperature of 30 degrees C (86 degrees F), and a relative humidity of at least 95 percent.

See ITS 3.7.3 & ITS Section 5.5

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FEB-06-2001 11:04 P.10/23

Specification 3.7.3

A1

3.11 (cont'd)

LA1

[ACTION A] ventilation air supply fan and/or filter may be out of service for 14 days.

M4 7 M3

add ACTION B

add ACTIONS Cand D

M3

2. The main control room air radiation monitor shall be operable whenever the control room emergency ventilation air supply fans and filter trains are required to be operable by 3.11.A.1 or filtration of the control room ventilation intake air must be initiated.

See ITS 3.3.7.1

4.11 (cont'd)

See ITS 5.5.8

A2

[SR 3.7.3.2]

b. Di-octylphtalate (DOP) test for particulate filter efficiency greater than 99% for particulate greater than 0.3 micron size.

c. Freon-112 test for charcoal filter bypass as a measure of filter efficiency of at least 99.5% for halogen removal.

TSTF-287

2. At least once per 24 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release, that could adversely affect the ability of the charcoal to perform its intended function, in any ventilation zone communicating with the system, verify:

(1) Within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration to be less than or equal to 5 percent when tested in accordance with ASTM D3803-1989 at a temperature of 30 degrees C [86 degrees F], and a relative humidity of at least 95 percent.

(2) Within 31 days of completing 720 hours of charcoal adsorber operation, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration to be less than or equal to 5 percent when tested in accordance with ASTM D3803-1989 at a temperature of 30 degrees C [86 degrees F], and a relative humidity of at least 95 percent.

AMD 269

when tested in accordance with Section C.5.a and C.5.c of Regulatory Guide 1.52, Rev 2, at a flowrate of 900 to 1100 scfm

M2

JAFNPP

Section 5.5

A1

M2

CTS Insert 238-1

M2

3.11 (cont'd) See ITS 3.7.3

M2

ventilation air supply fan and/or filter may be out of service for 14 days.

when tested in accordance with Section C.5.a and C.5.d of Regulatory Guide 1.52, Rev 2, at a flowrate of 900 to 1100 scfm

2. The main control room air radiation monitor shall be operable whenever the control room emergency ventilation air supply fans and filter trains are required to be operable by 3.11.A.1 or filtration of the control room ventilation intake air must be initiated.

See ITS 3.3.7.1

that could affect the filter system efficiency

4.11 (cont'd)

[5.5.8.a] b. Di-octylphtalate (DOP) test for particulate filter efficiency greater than 99% for particulate greater than 0.3 micron size.

CTS Insert 238-3

[5.5.8.b] c. Freon-112 test for charcoal filter bypass as a measure of filter efficiency of at least 99.5% for halogen removal.

[5.5.8.c] 2. At least once per 24 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release that could adversely affect the ability of the charcoal to perform its intended function, in any ventilation zone communicating with the system, verify:

A13

(1) Within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration to be less than or equal to 5 percent when tested in accordance with ASTM D3803-1989 at a temperature of 30 degrees C (86 degrees F), and a relative humidity of at least 95 percent.

LA4

(2) Within 31 days of completing after 720 hours of charcoal adsorber operation, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration to be less than or equal to 5 percent when tested in accordance with ASTM D3803-1989 at a temperature of 30 degrees C (86 degrees F), and a relative humidity of at least 95 percent.

LA4

RAI 5.5-3

RAI 5.5-2

CTS Amend 269

Specification 3.3.7.1

3.11 (cont'd)

3. The control room emergency ventilation system shall not be out of service for a period exceeding 3 days during normal reactor operation or refueling operations. In the event that the system is not returned to service within 3 days, the reactor shall be in cold shutdown within 24 hours and any handling of irradiated fuel, core alterations, and operations with a potential for draining the reactor vessel shall be suspended as soon as practicable

see ITS 3.7.3

~~4. Not Used~~

~~B. DELETED~~

C. Battery Room Ventilation

Battery room ventilation shall be operable on a continuous basis whenever specification 3.9.E is required to be satisfied.

1. From and after the date that one of the battery room ventilation systems is made or found to be inoperable, its associated battery shall be considered to be inoperable for purposes of specification 3.9.E.

4.11 (cont'd)

[SR 3.3.7.1.2]

3. Operability of the main control room air intake radiation monitor shall be tested once/3 months.

see ITS 3.7.3

4. Temperature transmitters and differential pressure switches shall be calibrated once per 24 months.

5. Main control room emergency ventilation air supply system capacity shall be tested once every 18 months to assure that it is ±10% of the design value of 1000 cfm.

~~B. DELETED~~

C. Battery Room Ventilation

Battery room ventilation equipment shall be demonstrated operable once/week.

1. When it is determined that one battery room ventilation system is inoperable, the remaining ventilation system shall be verified operable and daily thereafter.

2. Temperature transmitters and differential pressure switches shall be calibrated once per 24 months.

see CTS 3/4.11.C

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CTS 3/4.11.C

3.11 (cont'd)

B. DELETED

C. Battery Room Ventilation

Battery room ventilation shall be operable on a continuous basis whenever specification 3.9.E is required to be satisfied.

1. From and after the date that one of the battery room ventilation systems is made or found to be inoperable, its associated battery shall be considered to be inoperable for purposes of specification 3.9.E.

4.11 (cont'd)

B. DELETED

C. Battery Room Ventilation

Battery room ventilation equipment shall be demonstrated operable once/week.

1. When it is determined that one battery room ventilation system is inoperable, the remaining ventilation system shall be verified operable and daily thereafter.

2. Temperature transmitters and differential pressure switches shall be calibrated once per 24 months.

LA1

Page 1 of 1

Specification 3.7.3

A1

3.11 (cont'd)

3. The control room emergency ventilation system shall not be out of service for a period exceeding 3 days during normal reactor operation or refueling operations. In the event that the system is not returned to service within 3 days, the reactor shall be in cold shutdown within 24 hours and any handling of irradiated fuel, core alterations, and operations with a potential for draining the reactor vessel shall be suspended as soon as practicable

[ACTION E]

Enter LCO 3.0.3

M4

[ACTION F]

add ACTION F Note

M6

4. Not Used

4.11 (cont'd)

See ITS 3.3.7.1

3. Operability of the main control room air intake radiation monitor shall be tested once/3 months

LB1

4. Temperature transmitters and differential pressure switches shall be calibrated once per 24 months.

[SR 3.7.3.3]

5. Main control room emergency ventilation air supply system capacity shall be tested once every 18 months to assure that it is ± 10% of the design value of 1000 cfm.

TSTF 287

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L1

add STAGGERED TEST BASIS

M5

Verify each CREVAS subsystem can maintain a positive pressure of ≥ 0.125 in. water gauge relative to the atmosphere and turbine building during the isolation mode of operation at a flow rate ≤ 1100 cfm.

B. DELETED

C. Battery Room Ventilation

Battery room ventilation shall be operable on a continuous basis whenever specification 3.9.E is required to be satisfied.

1. From and after the date that one of the battery room ventilation systems is made or found to be inoperable, its associated battery shall be considered to be inoperable for purposes of specification 3.9.E.

B. DELETED

C. Battery Room Ventilation

Battery room ventilation equipment shall be demonstrated operable once/week.

1. When it is determined that one battery room ventilation system is inoperable, the remaining ventilation system shall be verified operable and daily thereafter.

2. Temperature transmitters and differential pressure switches shall be calibrated once per 24 months.

See CTS 3/4.11.C in this Section

Page 3 of 3

Specification 3.3.7.3

3.11 (cont'd)

(ESW) Instrumentation

D. Emergency Service Water System

LA1

[LCO 3.3.7.3] 1 To ensure adequate equipment and area cooling, both ESW systems shall be operable when the requirements of specification 3.5.A and 3.5.B must be satisfied, ~~except as specified below in specification 3.11.D.2.~~

Four channels of ESW pressure instrumentation shall be Operable

A2

[Applicability]

MODES 1, 2 and 3

4.11 (Cont'd)

(ESW) Instrumentation A1

D. Emergency Service Water System

1. Surveillance of the ESW system shall be performed as follows: A1

	Item	Frequency
a.	Simulated Automatic Actuation Test	Once every 24 months
b.	Flow Rate Test - Each ESW pump shall deliver at least 1500 gpm to its respective loop. The loop. The pump total developed head shall be greater than or equal to the corresponding point on the pump curve, reduced by a maximum of 7%, for the measured flow.	In Accordance with the Inservice Testing Program
c.	Verify that each valve (manual, power operated, or automatic) in the system flowpath that is not locked, sealed or otherwise secured in position, is in the correct position.	Once per 31 Days
d.	Motor Operated Valves	In Accordance with the Inservice Testing Program

See ITS: 5.7.2

Page 1 of 2

Specification 3.7.2

A1

3.11 (cont'd)

[3.7.2] D. Emergency Service Water System

[L(0 3.7.2] 1 To ensure adequate equipment and area cooling, both ESW systems shall be operable when the requirements of specification 3.5.A and 3.5.B must be satisfied, except as specified below in specification 3.11.D.2.

sub

[Applicability]

MODES 1, 2, and 3

and UHS

M2

and Ultimate Heat Sink

two

4.11 (Cont'd)

[3.7.2] D. Emergency Service Water System

1. Surveillance of the ESW system shall be performed as follows:

"actual or"

L1

Item	Frequency
a. Simulated Automatic Actuation Test [SR 3.7.2.7]	Once every 24 months
b. Flow Rate Test - Each ESW pump shall deliver at least 1500 gpm to its respective loop. The loop. The pump total developed head shall be greater than or equal to the corresponding point on the pump curve, reduced by a maximum of 7%, for the measured flow.	In Accordance with the Inservice Testing Program
g. Verify that each valve (manual, power operated, or automatic) in the system flowpath that is not locked, sealed or otherwise secured in position, is in the correct position. [SR 3.7.2.4]	Once per 31 Days
d. Motor Operated Valves	In Accordance with the Inservice Testing Program

LA1

LA1

A7

add Note

Page 1 of 3

Specification 3.3.7.3

A3

add ACTION Note

JAFNPP

add SR Note

L2

A1

3.11 (cont'd)

~~4.11 (cont'd)~~

add ACTIONS A, B and C

L3

	e.	ESW instrumentation check	Once/day
[SR 3.3.7.3.1]		ESW instrument channel calibration	Once/3 months
[SR 3.3.7.3.2]		Logic System Functional Test	Once every 24 months

add Allowable Value

M1

L1

2. From and after the time that one Emergency Service Water System is made or found to be inoperable for any reason continued reactor operation is permissible for a period not to exceed 7 days, provided that:

- the operable Emergency Diesel Generator System is demonstrated to be operable immediately and daily thereafter; and,

L4

- all Emergency Diesel Generator System emergency loads are verified operable immediately and daily thereafter.

L5

3. If specification 3.11.D.2 cannot be met the reactor shall be placed in the cold condition within 24 hours.

L1

2. ESW will not be supplied to RBCLC system during testing.

See ITS: 3.7.2

3. Not Used

Page 2 of 2

Specification 3.7.2 (A1)

3.11 (cont'd)

add Required Action A.1 Note (A2)

[ACTION A] 2. From and after the time that one Emergency Service Water System is made or found to be inoperable for any reason continued reactor operation is permissible for a period not to exceed 7 days, provided that:

- the operable Emergency Diesel Generator System is demonstrated to be operable immediately and daily thereafter; and, (L4)
- all Emergency Diesel Generator System emergency loads are verified operable immediately and daily thereafter. (L2)

[ACTION C] 3. If specification 3.11.D.2 cannot be met the reactor shall be placed in the cold condition within ~~24~~ 36 hours. (L3)

be in MODE 3 within 12 hours (M1)

(L3) (M1) add ACTION C second part

add ACTION C third part (M2)

4.11 (cont'd)

(A3) moved to ITS 3.3.7.3

e.	ESW instrumentation check	Once/day
	ESW instrument channel calibration	Once/3 months
f.	Logic System Functional Test	Once every 24 months

2. ESW will not be supplied to RBCLC system during testing. (LA2)

add SR 3.7.2.1
SR 3.7.2.2 (M2)

3. Not Used

Page 2 of 3

Specification 3.7.2

A1

3.11 (cont'd)

A4

add Applicability

A5

E. Intake Deicing Heaters

Note to SRs 3.7.2.4 3.7.2.5 3.7.2.6

Intake heaters are required to be operable when intake water temperature is less than or equal to 37°F. A minimum of 18 out of 88 heaters are required to be operable to maintain the required flow for the ESW and RHRSW Systems.

LA3

1. If specification 3.11.E above cannot be met the reactor shall be placed in a cold condition within 24 hours.

[ACTION C]

MODE 3 in 12 hours

M1

36

L3

M3

add ACTION B

4.11 (cont'd)

A4

E. Intake Deicing Heaters

A6

required

Note to SR 3.7.2.3

1. [SR 3.7.2.3] The six heater feeder ammeters shall be checked weekly whenever the intake water temperature is less than or equal to 37°F.

required

A6

2. [SR 3.7.2.5] The individual heaters shall be monitored once/6 months for rated heater current or as required by large deviations in the feeder checks in 4.11.E.1 above.

L5

[SR 3.7.2.6] 3. Resistance to ground shall be checked once/year.

The required

A6

power

A8

Page 3 of 3

Specification 3.10.1

A1

3.12 LIMITING CONDITIONS FOR OPERATION

3.12 SPECIAL OPERATIONS

Applicability:

Applies to the status of systems during special operations.

Objective:

To allow performance of special operations.

Specification:

4.12 SURVEILLANCE REQUIREMENTS

4.12 SPECIAL OPERATIONS

Applicability:

Applies to periodic testing of systems during special operations.

Objective:

To verify operability of required systems during special operations.

Specification:

[3.10.1] A. Inservice Leak and Hydrostatic Testing Operation

A4 [3.10.1] A. Inservice Leak and Hydrostatic Testing Operation

[Applicability] A4 [LCO 3.10.1] The reactor may be considered to be in COLD SHUTDOWN with reactor coolant system (RCS) temperature between 212°F and 200°F the reactor vessel not vented, and LCO requirements normally applicable when RCS temperature is greater than 212°F are not required, to allow performance of inservice leak or hydrostatic testing provided the following LCOs are met:

[SR 3.10.1.1] Perform the applicable surveillance requirements for the required LCOs.

add LCO 3.4.8 requirements suspension

A3

1. LCO 3.5.F, "ECCS-Cold Condition," a minimum of two low pressure subsystems shall be operable;

A2

[LCO 3.10.1.d] 2. LCO ~~3.7.B~~ 3.6.4.3, "Standby Gas Treatment System";

[LCO 3.10.1.a] 3. Secondary Containment isolation and Standby Gas Treatment initiation instrumentation:

a. LCO 3.2.A, "Primary Containment Isolation Functions", Table 3.2-1, Trip Function "Reactor Low Water Level (Notes 4 and 7)";

[Function 1] A1

A1

LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," Functions 1, 3, 4

Page 1 of 2

Specification 3.10.1

A1

3.12 LIMITING CONDITIONS FOR OPERATION

3.12 SPECIAL OPERATIONS (cont.)

[LCO 3.10.1.a Functions 3,4]

b. Radiological Effluent TS LCO 3.8, "Standby Gas Treatment System", Radiological Effluent TS Table 3.10-1, Trip Functions "Refuel Area Exhaust Monitor" and "Reactor Building Area Exhaust Monitors".

A1

LCO 3.3.6.2 "Secondary Containment Isolation Instrumentation," Function 3,4

[LCO 3.10.1.b and LCO 3.10.1.c]

4. LCO 3.7.C, "Secondary Containment"; (including the maintenance of Secondary Containment Integrity as defined by Definition T.0.S); and

A1

LCO 3.6.4.1 "Secondary Containment"

LCO 3.6.4.2 "Secondary Containment Isolation Valves"

5. LCO 3.9, "Auxiliary Electrical Systems," the necessary systems shall be operable to support equipment required to be operable.

A2

add proposed ACTION NOTE

[RA A.2.1 A.2.2]

6. With the above requirements not met, immediately suspend activities that could increase reactor coolant temperature or pressure and reduce reactor coolant temperature to less than 212°F within 24 hours.

add RA A.1

L1

≤

L2

Page 2 of 2

Chapter 4.0

A1

[4.0] ~~5.0~~ DESIGN FEATURES

[4.1] ~~5.1~~ SITE

LA1

5.1.1 The James A. FitzPatrick Nuclear Power Plant is located on the Entergy Nuclear FitzPatrick, LLC portion the Nine Mile Point site, approximately 3,000 ft. east of the Nine Mile Point Nuclear Station, Unit 1. The NPP-JAF site is on Lake Ontario in Oswego County, New York, approximately 7 miles northeast of Oswego. The plant is located at coordinates north 4,819,545.012m, east 386,968,945 m, on the Universal Transverse Mercator System.

5.1.2 The nearest point on the property line from the reactor building and any points of potential gaseous effluents, with the exception of the lake shoreline, is located at the northeast corner of the property. This distance is approximately

[4.1.1] 3,200 ft. and is the radius of the exclusion areas as defined in 10 CFR 100.3.

M1

[4.2] ~~5.2~~ REACTOR CORE

or ZIRLO — L2

4 ~~5.2.1~~ The reactor core consists of not more than 560 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy clad fuel rods with an initial composition of slightly enriched uranium dioxide (UO_2) as fuel material. Fuel assemblies shall be limited to those fuel designs approved by the NRC staff for use in BWRs.

INSERT 245-1

A3

L1

INSERT 245-2

L2

4 ~~5.2.2~~ The reactor core contains 137 cruciform-shaped control rods as described in Section 3.4 of the FSAR.

INSERT 245-3

A4

A4

5.3 REACTOR PRESSURE VESSEL

The reactor pressure vessel is as described in Table 4.2-1 and 4.2.2 of the FSAR. The applicable design codes are described in Section 4.2 of the FSAR.

5.4 CONTAINMENT

5.4.1 The principal design parameters and characteristics for the primary containment are given in Table 5.2-1 of the FSAR.

5.4.2 The secondary containment as described in Section 5.3 and the applicable codes are as described in Section 12.4 of the FSAR.

5.4.3 Penetrations of the primary containment and piping passing through such penetrations are designed in accordance with standards set forth in Section 5.2 of the FSAR.

Amendment No. ~~30, 42, 49, 64, 66, 74, 109, 117, 162, 256~~, 268

LA1

AMEND # 256, 268

<ADD ITS 4.1.2> — M2

Page 2 of 7

[4.3] ~~5.5~~ FUEL STORAGE

[4.3.1] ~~5.5.1~~ Criticality

[4.3.1.1] ~~5.5.1.1~~ The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum k_∞ of 1.32 in the normal reactor core configuration at cold conditions (20°C);

b. k_{eff} < 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.3 of the FSAR; and

4

c. A nominal 6.625 inch center to center distance between fuel assemblies placed in the aluminum high density storage racks ~~as described in Section 9.3 of the FSAR~~ INSERT 246-1 A6

[4.3.1.2] ~~5.5.1.2~~ The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum k_∞ of 1.31 in the normal reactor core configuration at cold conditions (20°C);

b. k_{eff} ≤ 0.90 if dry;

c. k_{eff} ≤ 0.95 if fully flooded with unborated water; and

d. A nominal 6.625 inch center to center distance between fuel assemblies placed in storage racks.

[4.3.3] ~~5.5.2~~ Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3239 fuel assemblies.

~~5.6 SEISMIC DESIGN~~ LA1

~~The reactor building and all engineered safeguards are designed on a basis of dynamic analysis using acceleration response spectrum curves which are normalized to a ground motion of 0.08 g for the Operating Basis Earthquake and 0.15 g for the Design Basis Earthquake.~~

<ADD ITS 4.3.2> M3

Section 5.1 (A.1)

ITS

[5.0] 6.0 ADMINISTRATIVE CONTROLS (A2)

~~Administrative Controls are the means by which plant operations are subject to management control. Measures specified in this section provide for the assignment of responsibilities, plant organization, staffing qualifications and related requirements, review and audit mechanisms, procedural controls and reporting requirements. Each of these measures are necessary to ensure safe and efficient facility operation~~

[5.1] 6.1 RESPONSIBILITY (A3)

[5.1.1] The Site Executive Officer (Plant manager) is responsible for safe (overall) operation of the plant and shall delegate in writing the succession to this responsibility during his absence. (A4) (M2)

CTS INSERT add 5.1.1

CTS INSERT add 5.1.2

6.2 ORGANIZATION

6.2.1 Facility Management and Technical Support (M3)

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities that affect the safety of the nuclear power plant.

1. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of department responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Updated FSAR.

2. The Site Executive Officer shall be responsible for overall plant operation, and shall have control over those onsite activities that are necessary for safe operation and maintenance of the plant.

3. The Chief Nuclear Officer shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

4. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 Plant Staff

The plant staff organization shall be as follows:

1. Each shift crew shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

See ITS Section 5.2

CTS Amend 254

Section 5.2 (A1)

ITS

See ITS Section 5.1

6.0 ADMINISTRATIVE CONTROLS

Administrative Controls are the means by which plant operations are subject to management control. Measures specified in this section provide for the assignment of responsibilities, plant organization, staffing qualifications and related requirements, review and audit mechanisms, procedural controls and reporting requirements. Each of these measures are necessary to ensure safe and efficient facility operation.

6.1 RESPONSIBILITY

The Site Executive Officer is responsible for safe operation of the plant and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

[5.2.1] 6.2.1 Facility Management and Technical Support

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities that affect the safety of the nuclear power plant.

[5.2.1.a] 1. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of department responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Updated FSAR/Quality Assurance Program. (A4) INSERT 247-2 (A2)

[5.2.1.b] 2. The Site Executive Officer (Plant manager) shall be responsible for (safe) overall plant operation, and shall have control over those onsite activities that are necessary for safe operation and maintenance of the plant. Shall have corporate responsibility for overall plant nuclear safety and (A3) (A2)

[5.2.1.c] 3. The Chief Nuclear Officer shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

[5.2.1.d] 4. The individuals who train the operating staff and those who carry out health physics (radiation protection) and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures. (A2)

[5.2.2] 6.2.2 Plant Staff

The plant staff organization shall be as follows:

1. Each shift crew shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

CTS Amend 254

A1

CTS Amend 270

A6

2. An SRO or an SRO with a license limited to fuel handling shall directly supervise all Core Alterations. This person shall have no other duties during this time.

3. DELETED

Provided immediate action is taken to restore shift composition to minimum requirements

[5.2.2.b] [5.2.2.c] 4. In the event of illness or unexpected absence, up to two (2) hours is allowed to restore the shift crew to the minimum complement.

A7 A2

[5.2.2.e] 5. The Operations Manager or Assistant Operations Manager, Shift Manager and Control Room Supervisor shall hold an SRO license and the Senior Nuclear Operator and the Nuclear Control Operator shall hold an RO license or an SRO license.

TSTF-258, R4

[5.2.2.d] 6. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; (e.g., senior reactor operators, health physicists, auxiliary operators, and maintenance personnel who are working on safety-related systems.

personnel licensed (SROs, licensed Reactor Operators (RO)) radiation protection technicians key

A2 LA2

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8 to 12 hours a day, nominal 40-hour week, while the plant is operating.

However, in the event that unforeseen problems require substantial amounts of overtime to be used or during extended periods of shutdown for refueling, major maintenance or major modifications, on a temporary basis, the following guidelines shall be followed:

LA2

a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 168-hour period, all excluding shift turnover time.

c. A break of at least eight hours should be allowed between work periods, shift turnover time can be included in the breaktime.

d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

INSERT 247a-1

working hour in advance

Any deviation from the above guidelines shall be authorized by the Site Executive Officer or the General Manager - Operations, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Site Executive Officer or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

LA2

TSTF-258, R4

be shall approved

INSERT 247a-2

working hour

A2

Plant manager or Plant manager's designee

CTS Amend 270

Section 5.2

An individual shall provide advisory technical support to the SS in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual

A1

JAFNPP

A8

See ITS Section 5.3

TSTF-258, R4

6.3 PLANT STAFF QUALIFICATIONS

6.3.1 The minimum qualifications with regard to educational background and experience for plant staff positions shown in FSAR Figure 13.2-7 shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions; except for the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.3.2 ~~The Shift Technical Advisor (STA) shall meet or exceed the minimum requirements of either Option 1 (Combined SRO/STA Position) or Option 2 (Continued use of STA Position), as defined in the Commission Policy Statement on Engineering Expertise on Shift, published in the October 28, 1985 Federal Register (50 FR 43621). When invoking Option 1, the STA role may be filled by the Shift Manager or Control Room Supervisor. (1)~~

[5.2.2.f]

A9

6.3.3 ~~Any deviations will be justified to the NRC prior to an individual's filling of one of these positions.~~

A10

NOTE:

(1) The 13 individuals who hold SRO licenses, and have completed the FitzPatrick Advanced Technical Training Program prior to the issuance of License Amendment 111, shall be considered qualified as dual-role SRO/STAs.

LA3

6.4 RETRAINING AND REPLACEMENT TRAINING

A training program shall be maintained under the direction of the Training Manager to assure overall proficiency of the plant staff organization. It shall consist of both retraining and replacement training and shall meet or exceed the minimum requirements of Section 5.5 of ANSI N18.1-1971.

The retraining program shall not exceed periods two years in length with a curriculum designed to meet or exceed the requalification requirements of 10 CFR 55.59.

6.5 REVIEW AND AUDIT

See CTS Chapter 6.0

Review requirements are completed by using designated technical reviewers/qualified safety reviewer and two separate review committees. The Plant Operating Review Committee (PORC) is an onsite review group; the Safety Review Committee (SRC) is an independent offsite review and audit group.

6.5.0 REVIEW AND APPROVAL OF PROGRAMS AND PROCEDURES

6.5.0.1 The procedure review and approval process shall be controlled and implemented by administrative procedure(s).

6.5.0.2 Each program and procedure required by Specification 6.8 and other procedures that affect nuclear safety, and changes thereto, shall be reviewed by a minimum of two designated technical reviewers who are knowledgeable in the affected functional area.

CTS Amend 270

Section 5.3 (A1)

[5.3]

[5.3.1] 6.3 PLANT STAFF QUALIFICATIONS (A3)

6.3.1 The minimum qualifications with regard to educational background and experience for plant staff positions shown in FSAR Figure 13.2-7 shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions; except for the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

CTS Amend 270

CTS INSERT 248-1

(A4)

6.3.2 The Shift Technical Advisor (STA) shall meet or exceed the minimum requirements of either Option 1 (Combined SRO/STA Position) or Option 2 (Continued use of STA Position), as defined in the Commission Policy Statement on Engineering Expertise on Shift, published in the October 28, 1985 Federal Register (50 FR 43621). When invoking Option 1, the STA role may be filled by the Shift Manager or Control Room Supervisor. (1)

TSTF-258, R4

6.3.3 Any deviations will be justified to the NRC prior to an individual's filling of one of these positions.

NOTE:

(1) The 13 individuals who hold SRO licenses, and have completed the FitzPatrick Advanced Technical Training Program prior to the issuance of License Amendment 111, shall be considered qualified as dual-role SRO/STAs.

See ITS Section 5.2

6.4 RETRAINING AND REPLACEMENT TRAINING

A training program shall be maintained under the direction of the Training Manager to assure overall proficiency of the plant staff organization. It shall consist of both retraining and replacement training and shall meet or exceed the minimum requirements of Section 5.5 of ANSI N18.1-1971.

The retraining program shall not exceed periods two years in length with a curriculum designed to meet or exceed the requalification requirements of 10 CFR 55.59

6.5 REVIEW AND AUDIT

Review requirements are completed by using designated technical reviewers/qualified safety reviewer and two separate review committees. The Plant Operating Review Committee (PORC) is an onsite review group; the Safety Review Committee (SRC) is an independent offsite review and audit group.

6.5.0 REVIEW AND APPROVAL OF PROGRAMS AND PROCEDURES

6.5.0.1 The procedure review and approval process shall be controlled and implemented by administrative procedure(s).

6.5.0.2 Each program and procedure required by Specification 6.8 and other procedures that affect nuclear safety, and changes thereto, shall be reviewed by a minimum of two designated technical reviewers who are knowledgeable in the affected functional area.

See ITS Section 6.0

CTS Amend 270

CTS Chapter 5.0

see ITS: Section 5.3

6.3 PLANT STAFF QUALIFICATIONS

6.3.1 The minimum qualifications with regard to educational background and experience for plant staff positions shown in FSAR Figure 13.2-7 shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions; except for the Radiological and Environmental Services Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

see ITS: Section 5.2

6.3.2 The Shift Technical Advisor (STA) shall meet or exceed the minimum requirements of either Option 1 (Combined SRO/STA Position) or Option 2 (Continued use of STA Position), as defined in the Commission Policy Statement on Engineering Expertise on Shift, published in the October 28, 1985 Federal Register (50 FR 43621). When invoking Option 1, the STA role may be filled by the Shift Manager or Control Room Supervisor. (1)

6.3.3 Any deviations will be justified to the NRC prior to an individual's filling of one of these positions.

NOTE:

(1) The 13 individuals who hold SRO licenses, and have completed the FitzPatrick Advanced Technical Training Program prior to the issuance of License Amendment 111, shall be considered qualified as dual-role SRO/STAs.

6.4 RETRAINING AND REPLACEMENT TRAINING

A training program shall be maintained under the direction of the Training Manager to assure overall proficiency of the plant staff organization. It shall consist of both retraining and replacement training and shall meet or exceed the minimum requirements of Section 5.5 of ANSI N18.1-1971.

LA1

The retraining program shall not exceed periods two years in length with a curriculum designed to meet or exceed the requalification requirements of 10 CFR 55.59

6.5 REVIEW AND AUDIT

Review requirements are completed by using designated technical reviewers/qualified safety reviewer and two separate review committees. The Plant Operating Review Committee (PORC) is an onsite review group; the Safety Review Committee (SRC) is an independent offsite review and audit group.

LA2

6.5.0 REVIEW AND APPROVAL OF PROGRAMS AND PROCEDURES

6.5.0.1 The procedure review and approval process shall be controlled and implemented by administrative procedure(s).

6.5.0.2 Each program and procedure required by Specification 6.8 and other procedures that affect nuclear safety, and changes thereto, shall be reviewed by a minimum of two designated technical reviewers who are knowledgeable in the affected functional area.

CTS Chapter 6.0

6.5.0.3 Designated technical reviewer(s) shall meet or exceed the qualifications described in Section 4 of ANSI N18.1-1971 for applicable positions, with the exclusion of the positions identified in Sections 4.3.2 and 4.5. Individuals whose positions are described in Sections 4.3.2 and 4.5 may qualify as designated technical reviewers provided they meet the qualifications described in other portions of Section 4.

6.5.0.4 The designated technical reviewer shall determine the need for cross-disciplinary reviews. Individuals performing cross-disciplinary reviews shall meet or exceed the qualifications described in Section 4 of ANSI N18.1-1971 for applicable positions.

6.5.0.5 Each program and procedure required by Specification 6.8 and other procedures that affect nuclear safety, and changes thereto, shall be reviewed from a safety perspective by a qualified safety reviewer. Safety and/or environmental impact evaluations, when required, shall be reviewed by PORC per Specification 6.5.1.6.a.

6.5.0.6 Nuclear safety related procedures and procedure changes shall be reviewed and approved, prior to implementation, by the appropriate member(s) of management.

6.5.1 PLANT OPERATING REVIEW COMMITTEE (PORC)

6.5.1.1 Function

PORC shall function to advise the Site Executive Officer on matters related to nuclear safety and environmental impact.

6.5.1.2 Membership

The Plant Operating Review Committee shall be composed of a Chairman, Vice-Chairmen and members designated in writing by the Site Executive Officer. Members shall collectively have responsibility in at least the areas of Operations, Maintenance, Radiological and Environmental Services, Engineering, Reactor Engineering, Instrumentation and Controls, Nuclear Licensing, and Quality Assurance.

CTS Chapter 6.0

LA 2

6.5.1.3 Alternates

Alternative members shall be appointed in writing by the PORC Chairman to serve on a temporary basis.

6.5.1.4 Meeting Frequency

Meetings will be called by the Chairman as the occasions for review or investigation arise. Meetings will be no less frequent than once a month.

6.5.1.5 Quorum

A quorum of the PORC shall consist of the Chairman or a Vice-Chairman and five members including designated alternates. Vice-Chairmen may act as members when not acting as Chairman. A quorum shall contain no more than two alternates.

6.5.1.6 Responsibilities

The PORC shall be responsible for the:

a. Review of 10 CFR 50.59 safety and environmental impact evaluations associated with procedures and programs required by Specification 6.8, and changes thereto.

b. Review of proposed tests and experiments that affect nuclear safety.

c. Review of proposed changes to the Operating License and Technical Specifications.

d. Review of proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of violations of the Technical Specifications. The PORC shall prepare and present a report covering the evaluations and recommendations to prevent recurrence to the Site Executive Officer, who will then forward the report to the Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, and to the Chairman of the Safety Review Committee.

f. Review of plant operations to detect potential safety hazards.

g. Performance of special reviews and/or investigations at the request of the Site Executive Officer.

h. Review of all reportable events.

i. Review of the Process Control Program and the Offsite Dose Calculation Manual (ODCM) and changes thereto.

B

CTS Chapter 6.0

6.5.1.3 Alternates

Alternative members shall be appointed in writing by the PORC Chairman to serve on a temporary basis.

6.5.1.4 Meeting Frequency

Meetings will be called by the Chairman as the occasions for review or investigation arise. Meetings will be no less frequent than once a month.

6.5.1.5 Quorum

A quorum of the PORC shall consist of the Chairman or a Vice-Chairman and five members including designated alternates. Vice-Chairmen may act as members when not acting as Chairman. A quorum shall contain no more than two alternates.

6.5.1.6 Responsibilities

The PORC shall be responsible for the:

a. Review of 10 CFR 50.59 safety and environmental impact evaluations associated with procedures and programs required by Specification 6.8, and changes thereto.

b. Review of proposed tests and experiments that affect nuclear safety.

c. Review of proposed changes to the Operating License and Technical Specifications.

d. Review of proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of violations of the Technical Specifications. The PORC shall prepare and present a report covering the evaluations and recommendations to prevent recurrence to the Site Executive Officer, who will then forward the report to the Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, and to the Chairman of the Safety Review Committee.

f. Review of plant operations to detect potential safety hazards.

g. Performance of special reviews and/or investigations at the request of the Site Executive Officer.

h. Review of all reportable events.

i. Review of the Process Control Program and the Offsite Dose Calculation Manual (ODCM) and changes thereto.

j. Review the FitzPatrick Fire Protection Program and implementing procedures and changes thereto.

see JPTS-95-005

CTS Chapter 6.0

-92

6.5.1.7- Authority

The PORC shall:

a. Recommend to the Site Executive Officer, approval or disapproval of those items reviewed under Specifications 6.5.1.6.a through 6.5.1.6.d.

b. Render determinations with regard to whether or not items considered under Specification 6.5.1.6.a through 6.5.1.6.e constitute an Unreviewed Safety Question as defined in 10 CFR 50.59.

c. In the event of a disagreement between the PORC and Site Executive Officer, notify the Chief Nuclear Officer and the SRC Chairman, or their designated alternates, within 24 hours and provide written notification by the next business day. The Site Executive Officer shall have responsibility for resolution of such disagreement pursuant to Section 6.1.

6.5.1.8 Records

Minutes of all meetings of the PORC shall be recorded and numbered. Copies will be retained in file. Copies will be forwarded to the Chairman of the SRC and the Chief Nuclear Officer.

6.5.1.9 Procedures

Conduct of the PORC and the mechanism for implementation of its responsibilities and authority are defined in the pertinent Administrative Procedures.

6.5.2 SAFETY REVIEW COMMITTEE (SRC)

FUNCTION

6.5.2.1 The SRC shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations

b. Nuclear engineering

c. Chemistry and radiochemistry

d. Metallurgy

e. Instrumentation and control

CTS Charter 6.0

f. Radiological safety
g. Mechanical engineering
h. Electrical engineering
i. Administrative controls and quality assurance practices
j. Environment
k. Civil/Structural Engineering
l. Nuclear Licensing
m. Emergency Planning
n. Other appropriate fields associated with the unique characteristics of a nuclear power plant

LA2

CHARTER

6.5.2.2 The conduct of the SRC will be in accordance with a charter approved by the Chief Nuclear Officer. The charter will define the SRC's authority and establish the mechanism for carrying out its responsibilities.

MEMBERSHIP

6.5.2.3 The SRC shall be composed of at least six individuals including a Chairman and a Vice Chairman. Members shall be appointed by the Director Regulatory Affairs and Special Projects and approved by the Chief Nuclear Officer. SRC members and alternates shall have an academic degree in engineering or a physical science, or the equivalent, and shall have a minimum of five years technical experience in one or more areas listed in Section 6.5.2.1.

ALTERNATES

6.5.2.4 Alternates for the Chairman, Vice Chairman and members may be appointed in writing by the Director Regulatory Affairs and Special Projects and approved by the Chief Nuclear Officer.

CONSULTANTS

6.5.2.5 Consultants may be used as determined by the SRC Chairman and as provided for in the charter.

MEETING FREQUENCY

6.5.2.6 The SRC shall meet at least once per six months.

CTS Chapter 6.0

QUORUM

6.5.2.7 A quorum shall consist of at least a majority of the appointed individuals (or their alternates) and the Chairman (or the designated alternate). No more than two alternates may participate as SRC voting members at any one time. No more than a minority of the quorum shall have direct line responsibility for the operation of the plant.

REVIEW

6.5.2.8 The SRC shall review facility activities in accordance with the Quality Assurance Program, as described in Chapter 17 of the JAF FSAR.

AUDIT

6.5.2.9 Audits of facility activities shall be performed under the cognizance of the SRC and in accordance with the Quality Assurance Program, as described in Chapter 17 of the JAF FSAR.

LA2

CTS Chapter 6.0

AUTHORITY

6.5.2.10 The SRC shall advise the Chief Nuclear Officer on those areas of responsibility specified in Section 6.5.2.8 and 6.5.2.9.

LAZ

CTS Chapter 6.0

RECORDS

6.5.2.11 Records will be maintained in accordance with ANSI 18.7-1972. The following shall be prepared and distributed as indicated below:

a. Minutes of each SRC meeting shall be prepared and forwarded to the Chief Nuclear Officer within 30 days after the date of the meeting.

b. Reports of reviews encompassed by Section 6.5.2.8 above shall be processed in accordance with the Quality Assurance Program, as described in Chapter 17 of the JAF FSAR.

c. Audit reports encompassed by Section 6.5.2.9 above, shall be processed in accordance with the Quality Assurance Program, as described in Chapter 17 of the JAF FSAR.

LAZ

Chapter 2.0 A1

see CTS Chapter 6.0

6.6 REPORTABLE EVENT ACTION

The following actions shall be taken for Reportable Events:

(A) The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

(B) Each Reportable Event shall be reviewed by the PORC, and the results of this review shall be submitted to the Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, and the Chairman of the SRC.

[2.2] 6.7 SAFETY LIMIT VIOLATION

within 2 hours M2

A2

[2.2] [2.2.1] [2.2.2]

(A) If a safety limit is exceeded, the reactor shall be shut down and reactor operation shall only be resumed in accordance with the provisions of 10 CFR 50.36 (c)(1)(i).

L3

(B) An immediate report of each safety limit violation shall be made to the NRC by the Site Executive Officer. The Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, and the Chairman of the SRC will be notified within 24 hours.

(C) The PORC shall prepare a complete investigative report of each safety limit violation and include appropriate analysis and evaluation of: (1) applicable circumstances preceding the occurrence, (2) effects of the occurrence upon facility component systems or structures and (3) corrective action required to prevent recurrence. The Site Executive Officer shall forward this report to the Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, the Chairman of the SRC, and the NRC.

L A1

6.8 PROCEDURES

(A) Written procedures and administrative policies shall be established, implemented, and maintained that:

1. meet or exceed the requirements and recommendations of Section 5 of ANSI 18.7-1972 "Facility Administrative Policies and Procedures."

2. are recommended in Appendix A of Regulatory Guide 1.33, November 1972.

3. implement the Fire Protection Program.

4. include programs specified in Appendix B of the Radiological Effluent Technical Specifications, Section 7.2.

(B) Each procedure of Specification 6.8.(A), and changes thereto, shall be approved prior to implementation by the appropriate responsible member of management as specified in Specification 6.5.0.

See ITS 5.4

[2.2.1] Within 2 hours, restore compliance with all SLs M2

Section 5.4 (A1)

See CTS Chapter 6.0

6.6 REPORTABLE EVENT ACTION

The following actions shall be taken for Reportable Events:

(A) The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

(B) Each Reportable Event shall be reviewed by the PORC, and the results of this review shall be submitted to the Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, and the Chairman of the SRC.

See ITS: Chapter 2.0

6.7 SAFETY LIMIT VIOLATION

(A) If a safety limit is exceeded, the reactor shall be shut down and reactor operation shall only be resumed in accordance with the provisions of 10 CFR 50.36, (c)(1)(i).

(B) An immediate report of each safety limit violation shall be made to the NRC by the Site Executive Officer. The Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, and the Chairman of the SRC will be notified within 24 hours.

(C) The PORC shall prepare a complete investigative report of each safety limit violation and include appropriate analysis and evaluation of: (1) applicable circumstances preceding the occurrence, (2) effects of the occurrence upon facility component systems or structures and (3) corrective action required to prevent recurrence. The Site Executive Officer shall forward this report to the Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, the Chairman of the SRC, and the NRC.

[5.4] 6.8 PROCEDURES

[5.4.1] (A) Written procedures ~~and administrative policies~~ shall be established, implemented, and maintained ~~that~~:

1. meet or exceed the requirements and recommendations of Section 5 of ANSI 18.7-1972 "Facility Administrative Policies and Procedures." (LA1)

2. ~~a.~~ ~~b.~~ recommended in Appendix A of Regulatory Guide 1.33, November 1972.

3. ~~c.~~ Implement the Fire Protection Program. The applicable procedures

4. Include programs specified in Appendix B of the Radiological Effluent Technical Specifications, Section 7.2. (A2)

(B) Each procedure of Specification 6.8.(A), and changes thereto, shall be approved prior to implementation by the appropriate responsible member of management as specified in Specification 6.5.0. (LA1)

b. The EOPs required to implement the requirements of NUREG 0737 and NUREG 0737 Supp. 1, as stated in GL 82-33;

e. All programs specified in Specification 5.5.

(M1)

CTS Amend 254

CTS Chapter 6.0

A1 LA3

6.6 REPORTABLE EVENT ACTION

The following actions shall be taken for Reportable Events:

A1

(A) The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

LA3

(B) Each Reportable Event shall be reviewed by the PORC, and the results of this review shall be submitted to the Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, and the Chairman of the SRC.

See ITS: Chapter 2.0

6.7 SAFETY LIMIT VIOLATION

(A) If a safety limit is exceeded, the reactor shall be shut down and reactor operation shall only be resumed in accordance with the provisions of 10 CFR 50.36, (c)(1)(i).

(B) An immediate report of each safety limit violation shall be made to the NRC by the Site Executive Officer. The Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, and the Chairman of the SRC will be notified within 24 hours.

(C) The PORC shall prepare a complete investigative report of each safety limit violation and include appropriate analysis and evaluation of: (1) applicable circumstances preceding the occurrence, (2) effects of the occurrence upon facility component systems or structures and (3) corrective action required to prevent recurrence. The Site Executive Officer shall forward this report to the Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, the Chairman of the SRC, and the NRC.

See ITS: Section 5.4

6.8 PROCEDURES

(A) Written procedures and administrative policies shall be established, implemented, and maintained that:

1. meet or exceed the requirements and recommendations of Section 5 of ANSI 18.7-1972 "Facility Administrative Policies and Procedures."

2. are recommended in Appendix A of Regulatory Guide 1.33, November 1972.

3. implement the Fire Protection Program.

4. include programs specified in Appendix B of the Radiological Effluent Technical Specifications, Section 7.2.

(B) Each procedure of Specification 6.8.(A), and changes thereto, shall be approved prior to implementation by the appropriate responsible member of management as specified in Specification 6.5.0.

Section 5.4
A1

(C) Temporary changes to the procedures required by Specification 6.8.(A) may be made provided:

1. the intent of the original procedure is not altered.

2. the change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's license.

3. the change is documented, reviewed and approved by the appropriate member of plant management as required by Specification 6.5.0 within 14 days of implementation.

LA1

6.9 REPORTING REQUIREMENTS

(A) ROUTINE REPORTS

The following reports shall be submitted in accordance with 10 CFR 50 unless otherwise noted.

1. STARTUP REPORT

a. A summary report of plant startup and power escalation testing shall be submitted following (1) amendment to the license involving a planned increase in power level, (2) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (3) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The Startup Report for the initial fuel cycle shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report. Startup Reports for subsequent fuel cycles shall address startup tests that are necessary to demonstrate the acceptability of changes and modifications.

See ITS Section 5.6

CTS Amend 254

Section 5.6

A1

(C) Temporary changes to the procedures required by Specification 6.8.(A) may be made provided:

1. the intent of the original procedure is not altered.

2. the change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's license.

3. the change is documented, reviewed and approved by the appropriate member of plant management as required by Specification 6.5.0 within 14 days of implementation.

see ITS: Section 5.4

[5.6] 6.9 REPORTING REQUIREMENTS

(A) ROUTINE REPORTS

50.4

A2

The following reports shall be submitted in accordance with 10 CFR 50 unless otherwise noted.

1. STARTUP REPORT

a. A summary report of plant startup and power escalation testing shall be submitted following (1) amendment to the license involving a planned increase in power level, (2) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (3) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The Startup Report for the initial fuel cycle shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report. Startup Reports for subsequent fuel cycles shall address startup tests that are necessary to demonstrate the acceptability of changes and modifications.

LA1

CTS Amend 254

Amendment No. ~~7, 22, 33, 110, 213,~~ 222

Page 1 of 9

REVISION H

(A) ROUTINE REPORTS (Continued)

1. STARTUP REPORT (Continued)

b. Startup Reports shall be submitted within (1) 90 days following completion of the startup test program, or (2) 90 days following resumption or commencement of commercial power operation, or whichever is earliest. If the Start-up Report does not cover both events, i.e., completion of startup test program and resumption or commencement of commercial power operation, supplementary reports shall be submitted at least every three months until both events are completed.

2. ANNUAL REPORTS

[5.6.1] a. ~~Annual~~ Radiation Occupational Exposure Report ~~Tabulation~~

A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) ~~receiving exposures greater than 100 mrem/yr and their associated man rem exposure~~ according to work and job functions, ~~1/~~ e.g., reactor operations and surveillance, in-service inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket ~~dosimeter, TLD~~, or film badge measurements. 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 ~~shall~~ should be assigned to specific major work functions.

INSERT 254b-1 A3

INSERT 254b-2

INSERT 254b-3 L1

INSERT 254b-4

b. ~~Annual Report of SRV Failures and Challenges~~ M3

~~An annual report of safety/relief valve failures and challenges will be submitted prior to March 1 of each year.~~ M1 A3

[5.6.4] 3. MONTHLY OPERATING REPORT

~~A~~ Routine report ~~providing a narrative summary of facility operating~~ statistics and shutdown experience, ~~major safety-related maintenance, and other pertinent information~~ ~~should~~ shall be submitted no later than the 15th of each month following the calendar month covered to the USNRC Director, Office of Management Information and Program Control. M2

1/ This tabulation supplements the requirements of ~~§ 20.407 of 10 CFR Part 20~~ 10 CFR 20.2206

CTS Amend 254

(A) ROUTINE REPORTS (Continued)

4. CORE OPERATING LIMITS REPORT

[5.6.5.a] a. Core operating limits shall be established prior to startup from each reload cycle, or prior to any remaining portion of a reload cycle for the following:

- The Average Planar Linear Heat Generation Rates (APLHGR) of Specification 3.5.H. 3.2.1 (A7)
- The Minimum Critical Power Ratio (MCPR) and MCPR low flow adjustment factor, K_f, of Specifications 3.1.B and 4.1.E 3.2.2
- The Linear Heat Generation Rate (LHGR) of Specification 3.5.J. 3.2.3
- The Reactor Protection System (RPS) APRM flow biased trip settings of Table 3.1.3; 3.3.1.1-1 Neutron Flux-High Allowable Value
- (A8) The flow biased APRM and Rod Block Monitor (RBM) rod block settings of Table 3.2.3 and 3.3.2.1-1 Upscale
- The Power/Flow Exclusion Region of Specification 3.8.J. 3.4.1

and shall be documented in the Core Operating Limits Report (COLR).

[5.6.5.b] b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC as described in:

(A11)

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, 14 June 2000.
2. "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis," NEDC-31317P, Revision 2, April 1993.
3. "BWR Owners' Group Long-term Stability Solutions Licensing Methodology," NEDO-31960-A, June 1993.
4. "BWR Owners' Group Long-term Stability Solutions Licensing Methodology," NEDO-31960-A, Supplement 1, March 1992.

The COLR will contain the complete identification of each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

TSTF-363, R0

CTS Amend 266

page 7 of 9

Section 5.6

A1

ITS [5.6.5.c]

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

[5.6.5.d]

d. The COLR, including any mid-cycle revisions or supplements thereto, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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RAI 5.6-1

CTS Chapter 6.0
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Section 5.6

A1

(B) SPECIAL REPORTS

A6

1. Fifteen copies of the Evaluation Report of the results of the first five years of performance of the non-destructive inspection listed in Table 4.6-1 of Technical Specifications 4.6.F, Structural Integrity, relating to the FitzPatrick in-service inspection program shall be submitted to the NRC, Director of Operating Reactors, within three months of the completion of the fifth year of the program.

2. DELETED

See CTS Chapter 6.0

6.10 RECORD RETENTION

(A) The following records shall be retained for at least five years:

1. Records and logs of facility operation covering time intervals at each power level.
2. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.
3. All Reportable Events.
4. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.
5. Records of reactor tests and experiments.
6. Records of changes made to Operating Procedures.
7. Records of radioactive shipments.
8. Records of sealed source leak tests and results.
9. Records of annual physical inventory of all source material of record.

A9 - Add proposed Section 5.6.6, PAM Report

CTS Amend 254

CTS Chapter 6.0

(B) SPECIAL REPORTS

1. Fifteen copies of the Evaluation Report of the results of the first five years of performance of the non-destructive inspection listed in Table 4.6-1 of Technical Specifications 4.6.F, Structural Integrity, relating to the FitzPatrick in-service inspection program shall be submitted to the NRC, Director of Operating Reactors, within three months of the completion of the fifth year of the program.

2. DELETED

See ITS: Section 5.6

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(A) The following records shall be retained for at least five years:

1. Records and logs of facility operation covering time intervals at each power level.

2. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

LA4

3. All Reportable Events.

4. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

5. Records of reactor tests and experiments.

6. Records of changes made to Operating Procedures.

7. Records of radioactive shipments.

8. Records of sealed source leak tests and results.

9. Records of annual physical inventory of all source material of record.

Component Cyclic or Transient Limit

see CTS: Chapter 6.0

(M) The following records shall be retained for the duration of the Facility Operating License:

1. Records of any drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

2. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

3. Records of facility radiation and contamination surveys.

4. Records of radiation exposure for all individuals entering radiation control areas.

5. Records of gaseous and liquid radioactive material released to the environs.

[5.5.5] 6. ~~Records of~~ This program provides controls to track transient or operational cycles for those facility components identified in ~~Table 4.1-2~~. INSERT 255-1

7. Records of training and qualification for current members of the plant staff.

8. Records of in-service inspections performed pursuant to these Technical Specifications.

9. Records of Quality Assurance activities required by the Quality Assurance Manual.

10. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

11. Records of meetings of the PORC and the SRC.

12. Records for Environmental Qualification which are covered under the provisions of paragraph 6.15.

13. Records of the service life of all hydraulic and mechanical snubbers, whose failure could adversely affect any safety-related system, including the date at which the service life commences and associated installation and maintenance records as of the effective date of this amendment.

see CTS: Chapter 6.0

5.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared and adhered to for all plant operations. These procedures shall be formulated to maintain radiation exposures received during operation and maintenance as far below the limits specified in 10 CFR 20 as practicable. The procedures shall include planning, preparation, and training for operation and maintenance activities. They shall also include exposure allocation, radiation and contamination control techniques, and final debriefing.

see ITS: Section 5.7

CTS Amend 261

Amendment No. 28, 34, 92
Order dated October 24, 1980

Specification 5.7

A1

See ITS Chapter 6.0

(B) The following records shall be retained for the duration of the Facility Operating License:

1. Records of any drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

2. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

3. Records of facility radiation and contamination surveys.

4. Records of radiation exposure for all individuals entering radiation control areas.

5. Records of gaseous and liquid radioactive material released to the environs.

6. Records of transient or operational cycles for those facility components identified in Table 6.10-1.

See ITS Section 5.5

7. Records of training and qualification for current members of the plant staff.

8. Records of in-service inspections performed pursuant to these Technical Specifications.

9. Records of Quality Assurance activities required by the Quality Assurance Manual.

10. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

11. Records of meetings of the PORC and the SRC.

12. Records for Environmental Qualification which are covered under the provisions of paragraph 6.15.

13. Records of the service life of all hydraulic and mechanical snubbers, whose failure could adversely affect any safety-related system, including the date at which the service life commences and associated installation and maintenance records as of the effective date of this amendment.

See CTS Chapter 6.0

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LA1

Page 1 of 2

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1. Records of any drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

2. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

3. Records of facility radiation and contamination surveys.

4. Records of radiation exposure for all individuals entering radiation control areas.

5. Records of gaseous and liquid radioactive material released to the environs.

L4

6. Records of transient or operational cycles for those facility components identified in Table 6.10-1.

See ITS: Section 5.5

7. Records of training and qualification for current members of the plant staff.

8. Records of in-service inspections performed pursuant to these Technical Specifications.

9. Records of Quality Assurance activities required by the Quality Assurance Manual.

10. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

11. Records of meetings of the PORC and the SRC.

12. Records for Environmental Qualification which are covered under the provisions of paragraph 6.15.

13. Records of the service life of all hydraulic and mechanical snubbers, whose failure could adversely affect any safety-related system, including the date at which the service life commences and associated installation and maintenance records as of the effective date of this amendment.

L4

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared and adhered to for all plant operations. These procedures shall be formulated to maintain radiation exposures received during operation and maintenance as far below the limits specified in 10 CFR 20 as practicable. The procedures shall include planning, preparation, and training for operation and maintenance activities. They shall also include exposure allocation, radiation and contamination control techniques, and final debriefing.

See ITS: Section 5.7

Amendment No. 22, 31, 92
Order dated October 24, 1980

Section 5.7 (A.1)

add proposed ITS Section 5.7 (L1)

[5.7] 6.11 (A) High Radiation Area

1. In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each High Radiation Area (i.e., ≥100 mrem/hr) in which the intensity of radiation is 1000 mrem/hr or less 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).* Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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

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

   c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility health physicist in the Radiation Work Permit.

2. The requirements of 6.11.A.1 above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or the radiation protection manager.

*Radiation protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

ITS Amend 270

27

CTS Chapter 6.0
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CTS Chapter 6.0

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6.12 INDUSTRIAL SECURITY PROGRAM

An industrial security program shall be maintained throughout the life of the plant in accordance with the provisions of the Plant Security Plan.

6.13 EMERGENCY PLAN

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6.14 FIRE PROTECTION PROGRAM

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6.15 ENVIRONMENTAL QUALIFICATION

A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License DPR-59 dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

A3

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see CTS Chapter 6.0

6.16 PROCESS CONTROL PROGRAM (PCP)

A. The PCP shall be a manual containing operational information concerning the solidification of radioactive wastes from liquid systems.

B. The PCP shall be maintained at the plant consistent with these Technical Specifications and with approved plant procedures.

C. Revisions of the PCP:

1. shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the revisions were made effective. This submittal shall contain:

a. sufficiently detailed information to support the rationale for the revisions without benefit of additional information;

b. a determination that the revision did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

c. documentation that the revision has been reviewed and found acceptable by the PORC.

2. shall become effective upon issue following review and acceptance by the PORC.

[5.5.1] 6.17 OFFSITE DOSE CALCULATION MANUAL (ODCM)

[5.5.1.a] A. The ODCM shall describe the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and effluents monitoring instrumentation alarm/trip setpoints consistent with the applicable LCOs contained in these Technical Specifications.

B. The ODCM shall be maintained at the plant and shall reflect accepted methodologies and calculational procedures.

A2

INSERT 258b-1

M6 ADD 5.5.1.b

CTS Amend 268

6.16 PROCESS CONTROL PROGRAM (PCP)

A. The PCP shall be a manual containing operational information concerning the solidification of radioactive wastes from liquid systems.

B. The PCP shall be maintained at the plant consistent with these Technical Specifications and with approved plant procedures.

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b. a determination that the revision did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

c. documentation that the revision has been reviewed and found acceptable by the PORC.

2. shall become effective upon issue following review and acceptance by the PORC.

LA5

6.17 OFFSITE DOSE CALCULATION MANUAL (ODCM)

A. The ODCM shall describe the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and effluents monitoring instrumentation alarm/trip setpoints consistent with the applicable LCOs contained in these Technical Specifications.

B. The ODCM shall be maintained at the plant and shall reflect accepted methodologies and calculational procedures.

See ITS: Section 5.5

Section 5.5

A1

[5.5.1.c] c. Licensee initiated Revisions of the ODCM: M6 in the form of a complete legible copy of the entire ODCM as part of or concurrent with

[5.5.1.c.3] 1. shall be submitted to the Commission ~~or~~ the ~~Semiannual~~ Radioactive Effluent Release Report for the period in which the revisions were made effective. ~~This submittal ash~~ This document shall contain:

editorial

[5.5.1.c.1.(a)] INSERT 258c-1

a. sufficiently detailed information to support the rationale for the revisions ~~without benefit of additional information (information submitted shall consist of revised pages of the ODCM, with each page numbered and provided with an approval and date box,~~ together with appropriate evaluations analyses or justifying the revisions);

M6

[5.5.1.c.1.(b)] INSERT 258c-2

b. a determination that the revisions ~~will not reduce~~ the effluent accuracy or reliability of dose calculation or setpoint determinations; and M6 records of reviews performed shall be retained.

M6

[5.5.1.c.1] c. ~~documentation that the revisions have been reviewed and accepted by the PORC~~ M1

[5.5.1.c.2] 2. shall become effective after approval of the plant manager upon issue following review and acceptance by the PORC. see CTS Chapter 6.0

TSTF-76, R1

6.18 MAJOR MODIFICATIONS TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATEMENT SYSTEMS*

A. Major modifications to radioactive waste systems (liquid, gaseous and solid):

1. shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the modifications is completed and made operational. The discussion of each modification shall contain:

a. a summary of the evaluation that led to the determination that the modification could made in accordance with 10 CFR 50.59;

b. sufficient information to support the reason for the modification without benefit of additional or supplemental information; and

c. a description of the equipment, components and processes involved and the interfaces with other plant systems.

* Entergy Nuclear Operations, Inc. may elect to submit the information called for in this Specification as part of the annual 10 CFR 50.59 Safety Evaluation Report.

CTS Amend 268

CTS Chapter 6.0

See ITS: Section 5.5

C. Revisions of the ODCM:

1. shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the revisions were made effective. This submittal shall contain:

   a. sufficiently detailed information to support the rationale for the revisions without benefit of additional information (information submitted shall consist of revised pages of the ODCM, with each page numbered and provided with an approval and date box, together with appropriate evaluations justifying the revisions);

   b. a determination that the revisions will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. documentation that the revisions have been reviewed and found acceptable by the PORC.

2. shall become effective upon issue following review and acceptance by the PORC.

LA7

6.18 MAJOR MODIFICATIONS TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS*

A. Major modifications to radioactive waste systems (liquid, gaseous and solid):

1. shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the modification is completed and made operational. The discussion of each modification shall contain:

   a. a summary of the evaluation that led to the determination that the modification could be made in accordance with 10 CFR 50.59;

   b. sufficient information to support the reason for the modification without benefit of additional or supplemental information; and

   c. a description of the equipment, components and processes involved and the interfaces with other plant systems.

LA6

*The Authority may elect to submit the information called for in this Specification as part of the annual 10 CFR 50.59 Safety Evaluation Report.

2. The following evaluations shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report, where such evaluations are required to be performed in order to assure compliance with the requirements of 10 CFR 50.59:

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

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

   c. a comparison of the predicted release of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the modifications are to be made.

Section 5.5

A1

This program provides control that

[5.5.3] 6.19 POSTACCIDENT SAMPLING PROGRAM

A program shall be established, implemented, and maintained which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

A) Training of personnel,

B) Procedures for sampling and analysis,

C) Provisions for maintenance of sampling and analysis

[5.5.6] 6.20 PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM

This 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 exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995, as modified by the exception that Type C testing of valves not isolable from the containment free air space may be accomplished by pressurization in the reverse direction provided that testing in this manner provides equivalent or more conservative results than testing in the accident direction. If potential atmospheric leakage paths (e.g., valve stem packing) are not subjected to test pressure, the portions of the valve not exposed to test pressure shall be subjected to leakage rate measurement during regularly scheduled Type A testing. A list of these valves, the leakage rate measurement method, and the acceptance criteria, shall be contained in the Program.

a. The peak Primary Containment internal pressure for the design basis loss of coolant accident (P_a), is 45 psig.

b. The maximum allowable Primary Containment leakage rate (L_a), at P_a, shall be 1.5% of primary containment air weight per day.

c. The leakage rate acceptance criteria are:

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

2. Airlock testing acceptance criteria are:

a. Overall airlock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

b. For each door seal, leakage rate is $\leq$ 120 scfd when tested at $\geq P_a$.

3. MSIV leakage rate acceptance criteria is $\leq$ 11.5 scfh for each MSIV when tested at $\geq$ 25 psig.

A5

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

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

SR 3.0.3

A6

add ITS 5.5.4 Radioactive Effluent Controls Program

A5

e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

Amendment No. ~~130, 234~~, 261

TSTF-52, R3/ RAI 5.5-1

Section 5.5 A1

[5.5.13] ~~5.21~~ CONFIGURATION RISK MANAGEMENT PROGRAM (CRMP)

The ~~Configuration Risk Management Program (CRMP)~~ provides a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The program applies to technical specification structures, systems, or components for which a risk-informed allowed outage time has been granted. The program is to include the following:

a. Provisions for the control and implementation of a Level 1 at-power internal events PRA-informed methodology. The assessment is to be capable of evaluating the applicable plant configuration.

b. Provisions for performing an assessment prior to entering the plant configuration described by the Limiting Conditions for Operation (LCO) ~~Action Statement~~ Condition(s) for preplanned activities. A11

c. Provisions for performing an assessment after entering the plant configuration described by the LCO ~~Action Statement~~ Condition(s) for unplanned entry into the LCO ~~Action Statement~~ Condition(s).

d. Provisions for assessing the need for additional actions after the discovery of additional equipment-out-of-service conditions while in the plant configuration described by the LCO ~~Action Statement~~ Condition(s).

e. Provisions for considering other applicable risk-significant contributors such as Level 2 issues and external events, qualitatively or quantitatively.

edit

[5.5.9] ~~5.22~~ ~~OFFGAS TREATMENT SYSTEM~~ EXPLOSIVE GAS and Storage Tank Radioactivity MONITORING PROGRAM

The program provides controls for potentially explosive gas mixtures contained in the Main Condenser Offgas Treatment System.

[5.5.9.a] The program shall include the limits for concentration of hydrogen and oxygen in the Main Condenser Offgas Treatment System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion).

The provisions of ~~Specification 3.0.B and 4.0.C~~ SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring program surveillance frequencies.

and quantity of radioactivity contained in unprotected outside liquid storage tanks

CTS Amend 270

M3 add 5.5.10 Diesel Fuel Oil Testing Program

M4 add 5.5.11 Technical Specification (TS) Bases Control Program

M5 add 5.5.12, Safety Function Determination Program (SFDP)

CTS Amend 270

CTS Chapter 6.3
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Amendment No. 60, 76, 105, 121

Section 5.2

A1

Table 6.2-1

ENTERGY NUCLEAR OPERATIONS, INC.
JAMES A. FITZPATRICK NUCLEAR POWER PLANT

MINIMUM SHIFT MANNING REQUIREMENTS

A1

Hot

		Refuel & Cold Shutdown (fuel in reactor)	Start-up, Shutdown or Run
LA4	SRO	1 on site	2 (1 in C. R.)
[5.2.2.f]	STA	None	1 on site* A5
LA4	RO	1 in C. R.	2 (1 in C. R.) **
[5.2.2.a]	Non-Licensed Operator	1 on site	2 on site
[5.2.2.c]	~~Individual Qualified in Radiation Protection Procedures~~ radiation protection technician	1 on site A2	1 on site A5

TSTF-258, R4

Note: * The STA position may be combined with one of the SRO positions and fulfilled by any individual meeting the dual-role SRO/STA qualification in accordance with Section 6.3.2.

** During startup or planned shutdown; both in Control Room. LA1

(SRO) - Licensed Senior Operator
(STA) - Shift Technical Advisor
(RO) - Licensed Reactor Operator
(C.R.) - Control Room

CTS Amend 268

Amendment No. ~~111~~, 268

ITS

[5.5.5]

TABLE 6.10-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	TRANSIENT CONDITION	OCCURRENCES
Reactor Pressure Vessel	Normal Startup (100°F/hr)	120
	50% Power Operation	14,600
	Rod Worth Tests	400
	Loss of Feedwater Heaters	
	Turbine Trip at 25% Power	10
	Feedwater Heater Bypass	70
	Loss of Feedwater Pumps	10
	Turbine Generator Trip	40
	Reactor Overpressure	1
	Safety Valve Blowdown	2
	All Other Scrams	147
	Improper Start of Cold Recirc. Loop	5
	Sudden Start of Cold Recirc. Loop	5
	Normal Shutdown	
	100°F/hr Cooldown (540-375°F)	
	Shutdown Flooding (375-330°F)	
	100°F/hr Cooldown (330-100°F)	118

LA2

Section 5.5

A1

CTS Amend 261

Chapter No. 3
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NOTE: Pages 262 through 264 were deleted by Amendment No. 37

The next page is Page 285

Chapter 1.0

A1

~~RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATION~~

[1.1] ~~1.0 DEFINITIONS~~

A. Dose Equivalent I-131

shall be that

~~The~~ Dose Equivalent that I-131 ~~is the~~ concentration of I-131 (microcuries/gram) ~~which~~ alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in International Commission on Radiological Protection Publication 30 (ICRP-30), "Limits for Intake by Workers" or in NRC Regulatory Guide 1.109, Revision 1, October 1977.

RAI 1.0-02

B. ~~Instrument Channel Calibration~~
~~See Appendix A Technical Specifications.~~

C. ~~Instrument Channel Functional Test~~
~~See Appendix A Technical Specifications.~~

D. ~~Instrument Check~~
~~See Appendix A Technical Specifications.~~

A6

E. ~~Logic System Function Test~~
~~See Apppendix A Technical Specifications.~~

F. ~~Members(s) of the Public~~
~~Member(s) of the Public includes all persons who are not occupationally associated with the facilities on the Entergy Nuclear FitzPatrick, LLC (ENF)/(NMPC) Niagara Mohawk Power Corporation site. This category does not include employees of the companies, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plants.~~

G. ~~Offgas Treatment System~~
~~The Offgas Treatment System is the system designed and installed to: reduce radioactive gaseous effluents by collecting primary coolant system offgases from the main condenser; and, providing for delay of the offgas for the purpose of reducing the total radioactivity prior to release to the environment.~~

H. Offsite Dose Calculation Manual (ODCM)
The ODCM describes the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluents monitoring instrumentation alarm/trip set points and in the conduct of the environmental monitoring program.

See ITS 5.5

I. ~~Operable~~
~~See Appendix A Technical Specifications.~~

A6

AND # 261, 268

Section 5.5

A1

RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATION

1.0 DEFINITIONS

See ITS Chapter 1.0

A. Dose Equivalent I-131

The Dose Equivalent I-131 is the concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in International Commission on Radiological Protection Publication 30 (ICRP-30), "Limits for Intake by Workers" or in NRC Regulatory Guide 1.109, Revision 1, October 1977.

B. Instrument Channel Calibration
See Appendix A Technical Specifications.

C. Instrument Channel Functional Test
See Appendix A Technical Specifications.

D. Instrument Check
See Appendix A Technical Specifications.

E. Logic System Function Test
See Apppendix A Technical Specifications.

F. Members(s) of the Public
Member(s) of the Public includes all persons who are not occupationally associated with the facilities on the Entergy Nuclear FitzPatrick, LLC (ENF)/(NMPC) Niagara Mohawk Power Corporation site. This category does not include employees of the companies, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plants.

G. Offgas Treatment System
The Offgas Treatment System is the system designed and installed to: reduce radioactive gaseous effluents by collecting primary coolant system offgases from the main condenser; and, providing for delay of the offgas for the purpose of reducing the total radioactivity prior to release to the environment.

[5.5.1.a] H. Offsite Dose Calculation Manual (ODCM)
The ODCM describes the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluents monitoring instrumentation alarm/trip set points and in the conduct of the radiological environmental monitoring program.

I. Operable
See Appendix A Technical Specifications.

See ITS Chapter 1.0

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[RETS]

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REVISION H

Chapter 1.0

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J. Process Control Program (PCP)
The PCP is a document which identifies the current formulas, sampling methods, analyses, tests, and determinations used to control the processing and packaging of solid radioactive wastes. The PCP controls these activities in such a way as to assure compliance with 10 CFR 20, 10 CFR 61, 10 CFR 71 and other applicable regulatory requirements governing the disposal of the radioactive waste.

A6

K. Rated Thermal Power
See Rated Power, Appendix A Technical Specifications.

L. Site Boundary
The Site Boundary is that line beyond which the land is not owned, leased, or otherwise controlled by ENF, ENO or NMPC. Refer to Figure 5.1-1 for the map of the site boundary with regard to liquid and gaseous releases.

See ITS Chapter 4.0

M. Solidification
Solidification is the conversion of wet wastes into a form that meets shipping and burial ground requirements.

N. Source Check
A Source Check is the qualification assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

O. Treatment
Any process which effectively reduces the concentration of radioactive material per unit measure released to the environment. This includes such processes as filtration, evaporation/condensation, settling/decanting, and solidification.

A6

P. Unrestricted Area
An unrestricted area shall be any area at or beyond the site boundary access to which is not controlled by ENF or ENO for purposes of protection of individuals from exposure to radiation and radioactive material, or any area within the site boundary used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

The definition of unrestricted area used in implementing the Radiological Effluent Technical Specifications has been expanded over that in 10 CFR 20.3(a)(17). The unrestricted area boundary may coincide with the exclusion (fenced) area boundary, as defined in 10 CFR 100.3(a), but the unrestricted area does not include areas over water bodies. The concept of unrestricted areas, established at or beyond the site boundary, is utilized in the Limiting Conditions for Operation to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10 CFR 50.36a.

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Chapter 4.0

See ITS: Chapter 1.0

A1 -

J. Process Control Program (PCP)
The PCP is a document which identifies the current formulas, sampling methods, analyses, tests, and determinations used to control the processing and packaging of solid radioactive wastes. The PCP controls these activities in such a way as to assure compliance with 10 CFR 20, 10 CFR 61, 10 CFR 71 and other applicable regulatory requirements governing the disposal of the radioactive waste.

K. Rated Thermal Power
See Rated Power, Appendix A Technical Specifications.

[4.1.1] ~~L~~. Site Boundary and Exclusion Area (4) LA1
The Site Boundary is that line beyond which the land is not owned, leased, or otherwise controlled by ENF, ENO or NMPC. Refer to Figure ~~5~~ 11-1 for the map of the site boundary ~~with regard to liquid and gaseous releases~~ add proposed 4.1.1 M1 LA1

M. Solidification
Solidification is the conversion of wet wastes into a form that meets shipping and burial ground requirements.

N. Source Check
A Source Check is the qualification assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

O. Treatment
Any process which effectively reduces the concentration of radioactive material per unit measure released to the environment. This includes such processes as filtration, evaporation/condensation, settling/decanting, and solidification.

P. Unrestricted Area
An unrestricted area shall be any area at or beyond the site boundary access to which is not controlled by ENF or ENO for purposes of protection of individuals from exposure to radiation and radioactive material, or any area within the site boundary used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

The definition of unrestricted area used in implementing the Radiological Effluent Technical Specifications has been expanded over that in 10 CFR 20.3(a)(17). The unrestricted area boundary may coincide with the exclusion (fenced) area boundary, as defined in 10 CFR 100.3(a), but the unrestricted area does not include areas over water bodies. The concept of unrestricted areas, established at or beyond the site boundary, is utilized in the Limiting Conditions for Operation to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10 CFR 50.36a.

AMEND #268

CTS RETS 2.1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
2.0 LIQUID EFFLUENTS	
2.1 LIQUID EFFLUENT MONITORS	**2.1 LIQUID EFFLUENT MONITORS**
Applicability	**Applicability**
Applies to instrumentation required for monitoring radioactive liquid effluent discharges to the environment as specified in Table 2.1-1.	Applies to instrumentation for monitoring radioactive liquid effluent discharges.
Objective	**Objective**
To assure that radioactive liquid effluent discharges are properly monitored and recorded during release.	To ensure that instrumentation required for radioactive liquid effluent discharges are maintained and calibrated.
Specifications	**Specifications**
a. The limiting conditions for operation of the instruments that monitor radioactive liquid effluents are given in Table 2.1-1. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip set point less conservative than required by the ODCM, without delay suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable, or change the set point so it is acceptably conservative.	a. The alarm/trip set points of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculation Manual (ODCM).
b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels operable, take the action shown in Table 2.1-1. Take corrective actions to return the instruments to operable	b. The surveillance requirements for the radioactive liquid effluent monitoring instrumentation is shown on Table 3.10-2.

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CTS RETS 2.1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner. NOTE: This reporting requirement does not apply to instruments which would not have been required to be operable during the 30 day period.	

CTS RETS 2.1

Table 2.1-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Action
Gross radioactivity monitors providing alarm and automatic termination of release		
Liquid radwaste effluent line	1	(a)
Gross beta or gamma radioactivity monitors providing alarm but not providing automatic termination of release		
Service water system effluent line	1	(b)
Flow rate measurement devices		
Liquid radwaste effluent line	1	(c)

NOTES FOR TABLE 2.1-1

(a) With the number of operable channels less than the required minimum number, effluent releases may continue provided that prior to initiating a release:

a. Two independent samples are analyzed;

b. Two technically qualified members of the facility staff verify the discharge line valving;

Otherwise, suspend release of radioactive effluents via this pathway.

(b) With the number operable of channels less than the required minimum number, effluent releases in this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for principal gamma emitters at a limit of detection of at least $5x10^{-7}$ microcuries/ml. The principal gamma emitters for which the LLD specification applies exclusively are described in Note (c) to Table 2.3-1.

(c) With the number of operable channels less than the required minimum number, effluent releases via this pathway may continue provided the flow rate is estimated at least once per four hours during actual releases. Pump curves or tank level decreases generated in situ may be used to estimate flow.

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CTS RETS 2.2

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
2.2 CONCENTRATION OF LIQUID EFFLUENTS	**2.2 CONCENTRATION OF LIQUID EFFLUENTS**
Applicability	**Applicability**
Applies to the concentration of radioactive materials in liquid effluents.	Applies to the analysis of radioactive liquid wastes from the plant through a liquid pathway to an unrestricted area.
Objective	**Objective**
To ensure that the concentrations of radioactive materials in liquid effluents are kept to acceptable levels.	To ensure that analyses are performed and concentration determined for radioactive liquid releases.
Specifications	**Specifications**
a. The concentration of radioactive materials released to the unrestricted areas shall not exceed the values specified in 10 CFR 20, Appendix B, Table II, Column 2. For dissolved or entrained noble gases the concentration shall be limited to 2×10^{-4} µCi/ml.	a. Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analyses program of Table 2.2-1.
b. With the concentration of radioactive material released from the plant to unrestricted areas exceeding the above limits, restore the concentration to within the above limits or terminate the release.	b. The results of the radioactivity analyses shall be used in accordance with the methods in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 2.2.a.

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CTS BETS 2.2

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TABLE 2.2-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (uCi/ml)
Batch Waste Release Tanks[b]	Prior to each batch	Each batch	Principal gamma emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
	One batch per month	Monthly	Dissolved and entrained gases (gamma emitters)	1×10^{-5}
	Prior to each batch	Quarterly composite[d]	H-3	1×10^{-5}
			Gross alpha	1×10^{-7}
	Prior to each batch	Quarterly composite[d]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-5}
Continuous Releases[e]	Continuous[f]	Weekly composite[f]	Principal gamma emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
	Monthly grab sample	Monthly	Dissolved and entrained gases (gamma emitters)	1×10^{-5}
	Continuous[f]	Monthly composite[f]	H-3	1×10^{-5}
			Gross alpha	1×10^{-7}
	Continuous[f]	Quarterly composite[f]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-5}

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R1

NOTES FOR TABLE 2.2-1

(a) The LLD (Lower Limit of Detection) is defined, for purpose of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count (above system background) that will be detected with 95% probability and with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the a priori lower limit of detection, as defined above (in microcuries per unit mass or volume);

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample, as appropriate (in counts per minute);

E is the counting efficiency (in counts per disintegration);

V is the sample size (in units of mass or volume);

2.22×10^6 is the number of disintegrations per minute per microcurie;

Y is the fractional radiochemical yield (when applicable);

λ is the radioactive decay constant for the particular radionuclide; and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

(b) A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then mixed to assure representative sampling.

(c) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, and Cs-137. This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measurable and identifiable, together with the above nuclides, shall also be identified and reported in the Semiannual Radioactive Effluent Release Report. The LLD for Mo-99, Ce-141, and Ce-144 is 5×10^{-5}.

R1

NOTES FOR TABLE 2.2-1 (continued)

(d) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

(e) A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

(f) To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

CTS RETS: 2.3

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
2.3 DOSE FROM LIQUID EFFLUENTS	**2.3 DOSE FROM LIQUID EFFLUENTS**
Applicability	**Applicability**
Applies to radiation doses from liquid effluents containing radioactive materials.	Applies to the calculation of the radiation dose from liquid effluents containing radioactive materials.
Objective	**Objective**
To assure that the dose limitations of 10 CFR 50, Appendix I for liquids are met.	To ensure that the radiation dose from radioactive liquid effluents is determined.
Specifications	**Specifications**
a. The dose to a member of the public from radioactive materials released from the plant in liquid effluents to unrestricted areas shall be limited as follows: 1. During any calendar quarter, limited to less than or equal to 1.5 mrem to the whole body and to less than or equal to 5 mrem to any organ; and, 2. During any calendar year, limited to less than or equal to 3 mrem to the whole body and to less than or equal to 10 mrem to any organ.	a. Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM at least monthly for the current calendar quarter and the current calendar year.
b. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, the following shall be done:	

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CTS RETS 2, 3

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
1. Identify the causes for such release rates; 2. Define and initiate a program of corrective action; and 3. Prepare and submit a report to the NRC within 30 days.	

CTS RBTS 2.4

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
2.4 LIQUID RADIOACTIVE WASTE TREATMENT SYSTEM OPERATIONS	2.4 LIQUID RADIOACTIVE WASTE TREATMENT SYSTEM OPERATIONS
Applicability Applies to the operability of radioactive liquid processing equipment.	**Applicability** Dose projections apply to liquid effluents released to unrestricted areas.
Objective To ensure liquid radwaste treatment system(s) are operated to prevent exceeding the dose limits of Specification 2.3.	**Objective** To ensure that action levels to require operation of waste treatment systems are determined.
Specifications a. The liquid radioactive waste treatment system shall be used when the projected dose from untreated liquid releases, over a 31 day period, to a member of the public would exceed: 1. 0.06 mrem to the whole body; or, 2. 0.2 mrem to any organ. b. With radioactive liquid waste being discharged in excess of the above limits, prepare and submit to the Commission within 30 days a report that includes the following information: 1. Explanation if liquid radwaste was being discharged without treatment; and if so:	**Specifications** a. Doses to individuals in unrestricted areas due to liquid releases shall be projected at least monthly in accordance with ODCM.

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CTS RETS 2.4

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
2. Identification of any inoperable equipment or subsystem and the reason for the inoperability;	
3. Action(s) taken to restore the inoperable equipment to operable status; and	
4. Summary description of action(s) taken to prevent a recurrence.	

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
2.5 MAXIMUM ACTIVITY IN OUTSIDE TANKS	**2.5 MAXIMUM ACTIVITY IN OUTSIDE TANKS**
Applicability	**Applicability**
[5.5.9.b] Applies to tanks located outdoors that do not have catch basins that drain back to the building.	Applies to outdoor tanks that do not have catch basins that drain back to the building.
Objective	**Objective**
To ensure that in the event of an uncontrolled release of the tank's contents, the resulting concentrations would be less than the limits of 10 CFR 20, Appendix B, Table II, Column 2, at the nearest surface water supply in an unrestricted area. (A10)	To ensure that the radioactivity contained in outdoor tanks is kept within applicable limits.
Specifications	[5.5.9.b] **Specifications**
[5.5.9.b] a. The quantity of liquid radioactive material contained in a condensate storage tank or any outside temporary tank shall be limited to 10 curies, excluding Tritium and dissolved or entrained noble gases.	a. The quantity of radioactive material contained in a condensate storage tank or any outside temporary tank shall be determined by analyzing a liquid sample of the tank's contents weekly when radioactive liquid is being added to the tank.
b. With the quantity of liquid radioactive material in a tank above this limit, reduce the tank's radioactive contents to within the limit within 48 hours; and	(LA3)
c. Describe the events leading to this condition in the next Semiannual Effluent Release Report.	

(L6) The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies

(RAI 5.5-7)

CTS RETS 3.1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

RI

3.0 GASEOUS EFFLUENTS

3.1 GASEOUS EFFLUENT MONITORS

Applicability

These requirements apply to instruments which monitor the gaseous radioactivity effluent pathways to the environment.

Objective

To assure that radioactive gaseous effluent discharges are properly monitored and recorded during release.

Specifications

a. Radioactive gaseous wastes released to the environment via the below listed pathways shall be monitored and recorded during release from the respective pathway.

1. Main stack exhaust
2. Refuel floor exhaust
3. Reactor building exhaust
4. Turbine building exhaust
5. Radwaste building exhaust

3.1 GASEOUS EFFLUENT MONITORS

Applicability

Applies to instrumentation listed in Specification 3.1.a and analyses of gaseous effluent releases.

Objective

To ensure that instrumentation required for gaseous effluent releases is maintained and calibrated and the radioactivity of gaseous releases is determined.

Specifications

a. Operation of the gaseous effluent monitors and recorders listed in Specification 3.1.a shall be verified by performing instrument surveillance as specified on Table 3.10-2.

CTS RETS 3.1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
b. Each pathway listed in a. above, shall also be sampled for iodine and particulate radioactivity on a continuous basis during release from the respective pathway.	b. The iodine cartridge and the particulate filter for each pathway listed in Specification 3.1.a shall be changed out at least weekly.
c. If Specifications 3.1.a and b., above, cannot be met, effluent releases may continue via the respective pathway provided gaseous grab samples are collected in the case of a monitor out of service and auxiliary samplers are used in case a particulate or iodine sampler is out of service; 1. Return the instrument to operable status within 30 days; or 2. Provide an explanation in the next Semi-annual Radioactive Release Report as to why the inoperability was not corrected within 30 days.	c. Grab samples, when required, shall be collected at least once per 12 hours and analyzed within 24 hours of collection. Auxiliary samplers shall run continuously and be changed out at least weekly.
d. Alarm/trip set points shall be determined in accordance with the ODCM and set to ensure that the limits of Specification 3.2 are not exceeded. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip set point less conservative than required by the above specification: 1. Without delay, suspend the release of radioactive gaseous effluents monitored by the affected channel; or 2. Declare the channel inoperable; or 3. Change the set point so it is acceptably conservative.	

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CTS RETS 3.2

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.2 GASEOUS DOSE RATES	**3.2 GASEOUS DOSE RATES**
Applicability	**Applicability**
Applies to the radiation dose from radioactive material in gaseous effluents from the plant.	Applies to the calculation of the dose rates from radioactive materials in gaseous effluents from the plant.
Objective	**Objective**
To ensure that the dose rates at or beyond the site boundary from gaseous effluents do not exceed the annual dose limits of 10 CFR 20 for unrestricted areas.	To ensure that appropriate calculations are performed to determine the dose rates from gaseous effluents from the plant.
Specifications	**Specifications**
a. The dose rate at or beyond the site boundary due to radioactive materials released from the plant in gaseous effluents shall be limited as follows: 1. ≤500 mrem/year to the whole body and ≤3000 mrem/year to the skin from noble gases; and, 2. ≤1500 mrem/year to any organ from Iodine-131, Iodine-133, Tritium and for radioactive materials in particulate form with half-lives greater than 8 days (inhalation pathway only).	a. The dose rate due to noble gases in gaseous effluents shall be determined to be continuously within the limits of Specification 3.2.a, in accordance with the methods and procedures of the ODCM.
b. With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limits.	b. The dose rate due to Iodine-131, Iodine-133, Tritium and to radionuclides in particulate form with half-lives greater than 8 days in

R1

CTS RETS 3.2

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	gaseous effluents, shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM. This will be done by obtaining representative samples and performing analyses in accordance with the sampling and analyses program specified in Table 3.2-1.

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TABLE 3.2-1

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

R1

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)(a) (μCi/ml)
Main Stack and Refuel Floor Vent and Reactor Building Vent and Turbine Building Vent and Radwaste Building Vent	Monthly Grab Sample(d)	Monthly Noble Gases(b)	Principal Gamma Emitters(b)	1×10^{-4}
	Quarterly Grab Sample	Quarterly	H-3	1×10^{-6}
	Continuous(c)	Weekly Charcoal Sample(e)	I-131	1×10^{-12}
			I-133	None
	Continuous(c)	Weekly Particulate Sample(e)	Principal Gamma Emitters(b)	1×10^{-11}
			(I-131, I-133, others)	None
	Continuous(c)	1 Wk/Mo Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous(c)	4 Wk/Qr Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous(c)	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-5}
Incinerated Oil(f)	Prior to Each Batch(g)	Each Batch(g)	Principal Gamma Emitters(b)	5×10^{-7}
			I-131	1×10^{-6}

R1

NOTES FOR TABLE 3.2-1

(a) The LLD is defined, for purpose of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count (above system background) that will be detected with 95% probability and with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the a priori lower limit of detection, as defined above (in microcuries per unit mass or volume);

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample, as appropriate (in counts per minute);

E is the counting efficiency (in counts per disintegration);

V is the sample size (in units of mass or volume);

2.22×10^6 is the number of disintegrations per minute per microcurie;

Y is the fractional radiochemical yield (when applicable);

λ is the radioactive decay constant for the particular radionuclide; and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

(b) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, Xe-135m, and Xe-138 for gaseous emissions; and, Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, and Cs-137 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measurable and identifiable, together with the above nuclides, shall also be identified and reported in the Semiannual Radioactive Effluent Release Report. The LLD for Mo-99, Ce-141, and Ce-144 is 5×10^{-11}. For oil samples, Table 2.2-1, Note (c) applies.

(c) The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications. This determination shall be made using design flow rates if flow meters are not provided or are inoperable.

CTS RETS 3.2

R1

NOTES FOR TABLE 3.2-1 (continued)

(d) Main stack gaseous sampling and analysis shall also be performed following shutdown, startup, or a thermal power change exceeding 20% of rated thermal power in one hour.

1. This requirement applies only if:

- Analysis shows that the dose equivalent I-131 concentration in the primary coolant has increased more than a factor of 3; and
- The noble gas monitor shows that effluent activity has increased more than a factor of 3; and
- Correction for increases due to changes in thermal power level have been made in both cases.

(e) Main stack iodine and particulate sampling shall also be performed daily following each shutdown, startup or thermal power change exceeding 20% of rated thermal power in one hour.

1. Daily sampling is not required for thermal power changes if the off gas charcoal filters are in service.
2. In addition, this requirement applies only if:
   - Analysis shows that the dose equivalent I-131 concentration in the primary coolant has increased more than a factor of 3; and
   - The noble gas monitor shows that effluent activity has increased more than a factor of 3; and
   - Corrections for increases due to changes in thermal power level have been made in both cases.
3. Daily sampling shall be performed until two consecutive samples show no increase in concentration but not to exceed 7 consecutive days.
4. LLDs may be increased by a factor of 10 for analysis of daily samples.
5. Analysis of daily and weekly samples shall be completed within 48 hours of changing.

(f) Incinerated oil may be discharged via points other than the main stack and building vents (i.e., auxiliary boiler). Release shall be accounted for based on pre-release grab sample data.

(g) Samples of incinerated oil releases shall be collected from and representative of filtered oil in liquid form. Whenever oil samples cannot be filtered such as No. 6 bunker fuel oil, raw oil samples shall be collected and analyzed.

CTS RETS: 53

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

R1

3.3 AIR DOSE, NOBLE GASES

Applicability

Applies to the air dose due to noble gases released from the plant.

Objective

To assure that the noble gas dose limitations of 10 CFR 50, Appendix I, are met.

Specifications

a. The air dose to areas at or beyond the site boundary from noble gases released from the plant in gaseous effluents shall be limited:

   1. During any calendar quarter, to less than or equal to 5 mrad from gamma radiation, and less than or equal to 10 mrad from beta radiation; and,

   2. During any calendar year, to less than or equal to 10 mrad from gamma radiation and less than or equal to 20 mrad from beta radiation.

b. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission, within 30 days, a report that:

3.3 AIR DOSE, NOBLE GASES

Applicability

Applies to the calculation of the air dose due to noble gas effluent.

Objective

To ensure that appropriate calculations are performed to determine the air dose from noble gas effluents.

Specifications

a. Cumulative air dose contributions for noble gases shall be calculated at least monthly in accordance with the ODCM for the current calendar quarter and the current calendar year.

CTS RETS: 3.3

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
1. Identifies the cause(s) for exceeding the limit(s); and 2. Defines the corrective actions that have been taken to reduce the releases; and 3. Identifies the proposed corrective actions to be taken to assure than subsequent releases will be in compliance with the above limits.	

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CTS RETS: 3.4

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.4 DOSE DUE TO IODINE-131, IODINE-133, TRITIUM AND RADIONUCLIDES IN PARTICULATE FORM

Applicability

Applies to the cumulative dose from Iodine-131, Iodine-133, Tritium, and radionuclides in particulate form in gaseous effluents.

Objective

To assure that the dose limitations of 10 CFR 50, Appendix I, are met.

Specifications

a. The dose to a member of the public at or beyond the site boundary from Iodine-131, Iodine-133, Tritium, and radionuclides in particulate form with half-lives greater than 8 days released from the plant in gaseous effluents shall be limited:

1. During any calendar quarter to less than or equal to 7.5 mrem to any organ; and,

2. During any calendar year to less than or equal to 15 mrem to any organ.

3. Less than 0.1% of the limits of Specification 3.4.a.1 and 3.4.a.2 as a result of burning contaminated oil.

3.4 DOSE DUE TO IODINE-131, IODINE-133, TRITIUM AND RADIONUCLIDES IN PARTICULATE FORM

R1

Applicability

Applies to the calculation of the dose due to Iodine-131, Iodine-133, Tritium, and radionuclides in particulate form in gaseous effluents.

Objective

To ensure that appropriate calculations are performed to determine the dose from Iodine-131, Iodine-133, Tritium, and radionuclides in particulate form.

Specifications

a. Cumulative dose contributions shall be calculated at least monthly in accordance with the ODCM for the current calendar quarter and the current calendar year.

CTS RETS: 3.4

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
b. With the calculated dose from the release of Iodine-131, Iodine-133, Tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days a report that: 1. Identifies the cause(s) for exceeding the limit; and 2. Defines the corrective actions that have been taken to reduce the releases; and 3. Identifies the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.	

R1

Specification 3.7.5.

A1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
[3.7.5] ~~3.5~~ MAIN CONDENSER STEAM JET AIR EJECTOR (SJAE)	3.5 MAIN CONDENSER STEAM JET AIR EJECTOR (SJAE)
[Applicability] **Applicability** Applies to main condenser offgas discharge rate for noble gases when the reactor is in the run, startup/hot standby or hot shutdown mode of operation and the SJAE is in service.	**Applicability** Applies to the point of discharge at the SJAE when the reactor is in the run, startup/hot standby or hot shutdown mode of operation and the SJAE is in service.
Objective To ensure that the SJAE release rates are maintained at a level compatible for further treatment and release.	**Objective** To ensure that the SJAE release rates are properly monitored.
Specifications	**Specifications**
[LCO 3.7.5] a. The gross radioactivity (beta and/or gamma) rate of noble gases measured at the SJAE is given on ~~Table 3.10-1~~.	a. The gross radioactivity (beta and/or gamma) rate of noble gases from the SJAE shall be determined to be within the limits of Specification 3.5.a by performing an isotopic analysis of a representative sample of gases taken at the discharge (prior to dilution and/or discharge) of the SJAE, or at the recombiner discharge (prior to delay of the offgas to reduce the total radioactivity) as follows: [SR 3.7.5.1]
	1. At least monthly. [SR 3.7.5.1 Frequency]
	2. With the SJAE Monitor reading at 5,000 µCi/sec or greater, within 4 hours following an increase of greater than 50% (after factoring out increases due to changes in thermal power level) in the nominal steady state fission gas release from the primary coolant. [SR 3.7.5.1 Frequency] M1 or equal to ...

A2

A2

add Note to SR 3.7.5.1 L3

≤ 600,000 µCi/sec L2

LA1

Specification 3.7.5

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
b. Except as specified in 1. and 2. below, both SJAE system radiation monitors shall be operable during reactor power operation. The trip time delay setting for closure of the SJAE isolation valve shall not exceed 15 min. 1. A channel may be placed in an inoperable status for up to six hours during periods of required surveillance without placing the Trip System in the tripped condition provided the other OPERABLE channel is monitoring that Trip Function. Otherwise, in the event that one of the two SJAE radiation monitors is made or found to be inoperable, continued reactor power operation is permissible provided that the inoperable monitor is tripped. 2. Upon the loss of both SJAE system radiation monitors, either temporarily monitor radiation levels at the SJAE or initiate an orderly shutdown and have the main steam isolation valves closed within 8 hours.	b. Operation of the SJAE radioactive offgas monitors shall be verified by performing instrument surveillance as specified on Table 3.10-2.

R1

CTS RETS 3.6

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6 **OFFGAS TREATMENT SYSTEM**	3.6 **OFFGAS TREATMENT SYSTEM**
Applicability Applies to the system installed for reduction of radioactive materials in gaseous waste prior to discharge.	**Applicability** Applies to the calculation of the radiation dose from gaseous effluents containing radioactive materials.
Objective To minimize concentration of radioactive materials released from the site.	**Objective** To ensure that treatment of gaseous wastes by the offgas system is implemented when required.
Specifications a. The offgas treatment system shall be used to reduce the concentration of radioactive materials in gaseous effluents prior to release from the plant within 24 hours after the start-up of the second turbine driven feedwater pump.	**Specifications** a. If the charcoal beds are not in service when the offgas treatment system is required, doses due to gaseous releases from the site shall be projected at least monthly in accordance with the ODCM.

R1

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CTS RETS 3.6

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
b. The offgas charcoal beds shall be used, when offgas treatment system operation is required and the projected doses over a 31 day period due to gaseous effluent releases to a member of the public would exceed: 1. 0.2 mrad for gamma radiation 2. 0.4 mrad for beta radiation; or 3. 0.3 mrem to any organ c. With gaseous effluent from the main condenser being discharged without use of the charcoal beds for greater than seven days when treatment is required, and projected doses are in excess of the above limits, prepare and submit to the Commission, within 30-days, a Special Report that includes the following information: 1. Explanation of why gaseous effluent is being discharged without charcoal bed treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability, 2. Action(s) taken to restore the inoperable equipment to operable status; and 3. Summary description of action(s) taken to prevent a recurrence.	

R1

Specification 3.3.6.2 (A1)

Secondary Containment Isolation Instrumentation

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
[3.3.6.2] 3.7 STANDBY GAS TREATMENT SYSTEM (SBGTS)	[3.3.6.2] 4.7 STANDBY GAS TREATMENT SYSTEM (SBGTS)
Applicability	**Applicability**
Applies to the SBGTS instrumentation.	Applies to the surveillance requirement of the instrumentation which activates the SBGTS.
Objective	**Objective**
To assure that the SBGTS is actuated with the proper signal.	To specify the instrument surveillance type and frequency.
Specifications	**Specifications**
[LCO 3.3.6.2] The limiting conditions for operation are given on Table 3.10-1.	The instrument surveillance requirements are given on Table 3.10-2. [SR Table Note 1]

add Applicability for Functions 3 and 4 (A10)

Amd 257

Specification 3.3.7.2

A1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9 MECHANICAL VACUUM PUMP ISOLATION	**3.9 MECHANICAL VACUUM PUMP ISOLATION**
Applicability	**Applicability**
Applies to the mechanical pump isolation instrumentation.	Applies to the surveillance requirement which isolates the mechanical vacuum pump.
Objective	**Objective**
To assure operability of the trip circuitry.	To specify the instrument surveillance type and frequency.
[3.3.7.2] **Specifications**	**Specifications**
[LCO 3.3.7.2] a. The mechanical vacuum (air removal) pump shall be capable of being automatically isolated and secured by a signal of high radiation in the main steam line tunnel [APPLICABILITY] whenever the main steam isolation valves are open.	The instrument surveillance requirements are given on Table 3.10-2. A13
b. If the limits of Table 3.10-1 are not met, the vacuum pump shall be isolated. A2	

L1 Modes 1 and 2 with any air removal pumps in service

Condenser Air Removal Pump Isolation Instrumentation

Specification 3.3.7.1

Control Room Emergency Ventilation Air Supply (CREVAS) System Instrumentation

A1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10 MAIN CONTROL ROOM VENTILATION RADIATION MONITOR	3.10 MAIN CONTROL ROOM VENTILATION RADIATION MONITOR
Applicability	**Applicability**
Applies to the emergency air supply treatment system to the main control room.	Applies to the instrumentation which monitors the control room supply air.
Objective	**Objective**
To assure operability of the in-line radiation monitor and annunciator.	To specify the instrument surveillance type and frequency.
Specifications	**Specifications**
[LCO 3.3.7.1] The limiting conditions for operation are given on Table 3.10-1. AB	The instrument surveillance requirements are given on Table 3.10-2. AB [SR 3.3.7.1.1] [SR 3.3.7.1.2]

add Applicability

M1

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Revision E

A1

Table 3.3.6.1 Primary Containment Isolation Instrumentation

Table 3.10-1

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

add ACTIONS NOTE 2 (A3)

L18 add ACTION NOTE 1

A16 Allowable Value

Minimum No. of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design	Action	
1(a)	Refuel Area Exhaust Monitor	(b)	2	(c) or (d)	See ITS: 3.3.6.2
1(a)	Reactor Building Area Exhaust Monitors	LA10 [illegible] ← 24,800 cpm M3	2	(d) M5	
(j)	SJAE Radiation Monitors	≤ 500,000 µCi/sec	2	(e)	See ITS: 3.7.5
1(a)	Turbine Building Exhaust Monitors	(b)	2	(f)	See CTS RETS:
1(a)	Radwaste Building Exhaust Monitors	(b)	2	(f)	
(k)	Main Control Room Ventilation	≤4 x 10^3 cpm[illegible]	1	(g)	See ITS: 3.3.7.1
(h)	Mechanical Vacuum Pump Isolation	≤3 x Normal Full Power Background	4	(h)	See ITS: 3.3.7.2

M5 [2.d] [2.i]

TSTF 206 R2

NOTES FOR TABLE 3.10-1

(a) A channel may be placed in an inoperable status for up to six hours during periods of required surveillance without placing the Trip System in the tripped condition provided the other OPERABLE channel is monitoring that Trip Function, that is, trip capability is maintained.

[SR Table Note 2]

[ACTION A] An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to operable status within 24 hours, or the indicated action shall be taken. LA2

(b) Trip level setting is in accordance with the methods and procedures of the ODCM. LA10

(c) Cease operation of the refueling equipment. See ITS: 3.3.6.2

(d) Isolate secondary containment and start the SBGTS. add ACTION B, C and F M5

(e) Bring the SJAE release rate below the trip level within 72 hours or isolate either the SJAE or all main steam lines within the next 12 hours. See ITS 3.7.5

add footnote (b) M5

Specification 3.3.6.2

A1

Table 3.3.6.2-1 Secondary Containment Isolation Instrumentation

Table 3.10-1

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

Required Minimum No. of Operable Instrument Channels per Trip System	add ACTIONS Note A2 Trip Function	Allowable Value Trip Level Setting	Total Number of Instrument Channels Provided by Design LA1	Action
[(3)] 1(a)	Refuel Area Exhaust Monitor	M2 add Allowable Value (b) LA3	2	A1 (c) or (d) [a]
[(4)] 1(a)	Reactor Building Area Exhaust Monitors	(b)	2	(d)
(j)	SJAE Radiation Monitors	≤ 500,000 μCi/sec	2	(e) See ITS:3.7.5
1(a)	Turbine Building Exhaust Monitors	(b)	2	(f) See CTS RETS:3.1
1(a)	Radwaste Building Exhaust Monitors	(b)	2	(f) See ITS:3.3.7.1
(k)	Main Control Room Ventilation	≤4 x 10^3 cpm[a]	1	(g) See ITS:3.3.7.2
(h)	Mechanical Vacuum Pump Isolation	≤3 x Normal Full Power Background	4	(h)

NOTES FOR TABLE 3.10-1 — add ACTIONS LB

(a) A channel may be placed in an inoperable status for up to six hours during periods of required surveillance without placing the Trip System in the tripped condition provided the other OPERABLE channel is monitoring that Trip Function, that is, trip capability is maintained. LA2

[SR Table Note 2] [ACTION A] An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to operable status within 24 hours, or the indicated action shall be taken.

[RA C.1.1] (b) Trip level setting is in accordance with the methods and procedures of the ODCM. LA3

(c) Cease operation of the refueling equipment. [RA C.2.1] A1

[ACTION C] (d) Isolate secondary containment and start the SBGTS. in one hour M3

add proposed Required Actions C.1.2 and C.2.2 L4

(e) Bring the SJAE release rate below the trip level within 72 hours or isolate either the SJAE or all main steam lines within the next 12 hours. See ITS:3.7.5

add Table 3.3.6.2-1 Footnote (a) Applicability M1

Am 257

Specification 3.3.7.1

A1

Table 3.10-1 A8

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

~~Minimum No. of Operable Instrument Channels per Trip System~~ A8	~~Trip Function~~	A2 ~~Trip Level Setting~~ Allowable Value [SR 3.3.7.1.2]	A8 ~~Total Number of Instrument Channels Provided by Design~~	~~Action~~
1(a)	Refuel Area Exhaust Monitor	(b)	2	(c) or (d)
1(a)	Reactor Building Area Exhaust Monitors	(b)	2	(d)
(j)	SJAE Radiation Monitors	≤ 500,000 μCi/sec	2	(e)
1(a)	Turbine Building Exhaust Monitors	(b)	2	(f)
1(a)	Radwaste Building Exhaust Monitors	(b)	2	(f)
(k)	Main Control Room Ventilation	≤4 x 10^3 cpm[a]	[LCO 3.3.7.1] 1	(g) [A] M3
(h)	Mechanical Vacuum Pump Isolation	≤3 x Normal Full Power Background	4	(h)

[LCO 3.3.7.1]

See ITS: 3.3.6.2

See ITS: 3.7.5

See CTS RETS: 3.1

See ITS: 3.3.7.2

NOTES FOR TABLE 3.10-1

(a) A channel may be placed in an inoperable status for up to six hours during periods of required surveillance without placing the Trip System in the tripped condition provided the other OPERABLE channel is monitoring that Trip Function, that is, trip capability is maintained.
An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to operable status within 24 hours, or the indicated action shall be taken.

(b) Trip level setting is in accordance with the methods and procedures of the ODCM. → See ITS: 3.3.6.2 CTS RETS: 3.1

(c) Cease operation of the refueling equipment. → See ITS: 3.3.6.2

(d) Isolate secondary containment and start the SBGTS.

(e) Bring the SJAE release rate below the trip level within 72 hours or isolate either the SJAE or all main steam lines within the next 12 hours. → See ITS: 3.7.5

add Note to ACTIONS Table (A5)

(A1)

Table 3.10-1

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

Minimum No. of Operable Instrument Channels per Trip System (LA1)	Trip Function	Allowable Value (A14) / Trip Level Setting	Total Number of Instrument Channels Provided by Design (A3)	Action	
1(a)	Refuel Area Exhaust Monitor	(b)	2	(c) or (d)	See ITS: 3.3.6.2
1(a)	Reactor Building Area Exhaust Monitors	(b)	2	(d)	See ITS: 3.7.5
(j)	SJAE Radiation Monitors	≤ 500,000 μCi/sec	2	(e)	See CTS RETS: 3.1
1(a)	Turbine Building Exhaust Monitors	(b)	2	(f)	
1(a)	Radwaste Building Exhaust Monitors	(b)	2	(f)	
(k)	Main Control Room Ventilation	≤4 x 10^3 cpm*	1	(g)	See ITS: 3.3.7.1
[LCO 3.3.7.2] (A4)	Mechanical Vacuum Pump Isolation	[SR 3.3.7.2.2] ≤3 x Normal Full Power Background	[LCO 3.3.7.2] (A3)	(A4)	

NOTES FOR TABLE 3.10-1

(a) A channel may be placed in an inoperable status for up to six hours during periods of required surveillance without placing the Trip System in the tripped condition provided the other OPERABLE channel is monitoring that Trip Function, that is, trip capability is maintained.
An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to operable status within 24 hours, or the indicated action shall be taken.

(b) Trip level setting is in accordance with the methods and procedures of the ODCM. (See ITS: 3.3.6.2 CTS RETS: 3.1) (See ITS: 3.3.6.2 CTS RETS: 3.1)

(c) Cease operation of the refueling equipment. (See ITS: 3.3.6.2)

(d) Isolate secondary containment and start the SBGTS. (See ITS: 3.7.5)

(e) Bring the SJAE release rate below the trip level within 72 hours or isolate either the SJAE or all main steam lines within the next 12 hours.

Specification 3.7.5

A1

R1

Table 3.10-1

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

Minimum No. of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design	Action
1(a)	Refuel Area Exhaust Monitor	(b)	2	(c) or (d)
1(a)	Reactor Building Area Exhaust Monitors	(b)	2	(d)
[LCO 3.7.5] (j)	SJAE Radiation Monitors [LCO 3.7.5]	≤ 500,000 μCi/sec L2	2	(e)
1(a)	Turbine Building Exhaust Monitors	(b)	2	(f)
1(a)	Radwaste Building Exhaust Monitors	(b)	2	(f)
(k)	Main Control Room Ventilation	≤4 x 10^3 cpm[a]	1	(g)
(h)	Mechanical Vacuum Pump Isolation	≤3 x Normal Full Power Background	4	(h)

See ITS: 3.3.6.2

R1

See CTS RETS 3.1

See ITS 3.3.7

See ITS: 3.3.7.2

NOTES FOR TABLE 3.10-1

(a) A channel may be placed in an inoperable status for up to six hours during periods of required surveillance without placing the Trip System in the tripped condition provided the other OPERABLE channel is monitoring that Trip Function, that is, trip capability is maintained.
An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to operable status within 24 hours, or the indicated action shall be taken.

See ITS: 3.3.6.2
CTS RETS: 3.1

(b) Trip level setting is in accordance with the methods and procedures of the ODCM.

(c) Cease operation of the refueling equipment. See ITS: 3.3.6.2

(d) Isolate secondary containment and start the SBGTS.

[ACTION B]

[ACTION A] (e) Bring the SJAE release rate below the trip level within 72 hours or isolate either the SJAE or all main steam lines within the next 12 hours.

add RA B.3.1 and B.3.2 L1

B

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37
[RETS]

CTS RETS 3.1

Table 3.10-1

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

Minimum No. of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design	Action
1(a)	Refuel Area Exhaust Monitor	(b)	2	(c) or (d)
1(a)	Reactor Building Area Exhaust Monitors	(b)	2	(d)
(j)	SJAE Radiation Monitors	≤ 500,000 μCi/sec	2	(e)
1(a)	Turbine Building Exhaust Monitors	(b)	2	(f)
1(a)	Radwaste Building Exhaust Monitors	(b)	2	(f)
(k)	Main Control Room Ventilation	≤4 x 10^3 cpm*	1	(g)
(h)	Mechanical Vacuum Pump Isolation	≤3 x Normal Full Power Background	4	(h)

R1

See ITS: 3.3.6.2

See ITS: 3.7.5

R1

See ITS: 3.3.7.1

See ITS: 3.3.2.2

NOTES FOR TABLE 3.10-1

(a) A channel may be placed in an inoperable status for up to six hours during periods of required surveillance without placing the Trip System in the tripped condition provided the other OPERABLE channel is monitoring that Trip Function, that is, trip capability is maintained.
An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to operable status within 24 hours, or the indicated action shall be taken.

R1

(b) Trip level setting is in accordance with the methods and procedures of the ODCM.

R1

(c) Cease operation of the refueling equipment.

(d) Isolate secondary containment and start the SBGTS.

See ITS: 3.3.6.2

(e) Bring the SJAE release rate below the trip level within 72 hours or isolate either the SJAE or all main steam lines within the next 12 hours.

See ITS: 3.7.5

Specification 3.3.7.1

A1

Table 3.10-1

A8

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

NOTES FOR TABLE 3.10-1 (cont'd.)

(f) Refer to Appendix B LCO 3.1.c and LCO 3.1.d. — See CTS RETS! 3.1

[ACTION A] (g) Control room isolation is manually initiated. — M3

(h) Uses same sensors as primary containment isolation on high main steam line radiation. Refer to Appendix A Table 3.2-1 for minimum number of operable instrument channels and action required. — See ITS: 3.3.7.2

(i) Conversion factor is 8.15 x 10⁷ cpm - 1μCi/cc. — A7

(j) See RETS LCO 3.5.b.1 and 3.5.b.2. — See ITS: 3.7.5

[SR Table Note] (k) A channel may be placed in an inoperable status for up to six hours during periods of required surveillance without placing the Trip Function in the tripped condition, or the indicated action shall be taken.

add Required Action A.1 Completion Time and Required Action A.2 — M3

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Specification 3.3.7.2

Table 3.10-1

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

NOTES FOR TABLE 3.10-1 (cont'd.)

(f) Refer to Appendix B LCO 3.1.c and LCO 3.1.d. See CTS RETS: 3.1

(g) Control room isolation is manually initiated. See ITS: 3.3.7.1

(h) Uses same sensors as primary containment isolation on high main steam line radiation. Refer to Appendix A Table 3.2-1 for minimum number of operable instrument channels and action required. A4

(i) Conversion factor is 8.15 x 10^7 cpm - 1μCi/cc. See ITS: 3.3.7.1

(j) See RETS LCO 3.5.b.1 and 3.5.b.2. See ITS 3.7.5

(k) A channel may be placed in an inoperable status for up to six hours during periods of required surveillance without placing the Trip Function in the tripped condition, or the indicated action shall be taken. See ITS: 3.3.7.1

B

Specification 3.7.5

Table 3.10-1

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

NOTES FOR TABLE 3.10-1 (cont'd.)

(f) Refer to Appendix B LCO 3.1.c and LCO 3.1.d. See CTS RETS 3.1

(g) Control room isolation is manually initiated. See ITS: 3.3.7.1

(h) Uses same sensors as primary containment isolation on high main steam line radiation. Refer to Appendix A Table 3.2-1 for minimum number of operable instrument channels and action required. See ITS: 3.3.2.2

(i) Conversion factor is 8.15 x 10⁷ cpm - 1μCi/cc. See ITS: 3.3.7.1

(j) See RETS LCO 3.5.b.1 and 3.5.b.2. R1

(k) A channel may be placed in an inoperable status for up to six hours during periods of required surveillance without placing the Trip Function in the tripped condition, or the indicated action shall be taken. See ITS: 3.3.7.1

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CTS RETS 3.1

Table 3.10-1

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

NOTES FOR TABLE 3.10-1 (cont'd.)

R1

(f) Refer to Appendix B LCO 3.1.c and LCO 3.1.d.

(g) Control room isolation is manually initiated. See ITS: 3.3.7.1

(h) Uses same sensors as primary containment isolation on high main steam line radiation. Refer to Appendix A Table 3.2-1 for minimum number of operable instrument channels and action required. See ITS: 3.3.7.2

(i) Conversion factor is 8.15 x 10^7 cpm - 1μCi/cc. See ITS: 3.3.7.1

(j) See RETS LCO 3.5.b.1 and 3.5.b.2. See ITS: 3.7.5

(k) A channel may be placed in an inoperable status for up to six hours during periods of required surveillance without placing the Trip Function in the tripped condition, or the indicated action shall be taken. See ITS: 3.3.7.1

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Specification 3.3.6.1

A1

Table 3.3.6.1-1 Primary Containment Isolation Instrumentation

TABLE 3.10-2

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RADIATION MONITORING SYSTEMS[a]

Instrument Channels	Instrument Check[b] [SR 3.3.6.1-1] M9	Instrument Channel Functional Test[w] A10 AB	Instrument Channel Calibration [SR 3.3.6.1.3]	Logic System Function Test[w][g] SR 3.3.6.1.7 A11
Main Stack Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
Refuel Area Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
Reactor Building Area Exhaust Monitors ~~Recorders~~ ~~and Isolation~~ L14 A10	~~Daily~~ (12 hours) [-1] M9	~~Quarterly~~ AB	Quarterly-3	Once per 24 Months -7
Turbine Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
Radwaste Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
SJAE Radiation Monitors/Offgas Line Isolation	Daily	Quarterly	Quarterly	Once per 24 Months
Main Control Room Ventilation Monitor	Daily	Quarterly	Quarterly	--
Mechanical Vacuum Pump Isolation[w]	--	--	--	Once per 24 Months
Liquid Radwaste Discharge Monitor/ Isolation[c][d][e][f]	Daily When Discharging	Quarterly	Quarterly	Once per 24 Months
Liquid Radwaste Discharge Flow Rate Measuring Devices[d]	Daily	Quarterly	Once per 18 Months	--
Liquid Radwaste Discharge Radioactivity Recorder[d]	Daily	Quarterly	Once per 18 Months	--
Normal Service Water Effluent	Daily	Quarterly	Quarterly	--
SBGTS Actuation	--	--	--	Once per 24 Months

[2g][2.d]

See CTS RETS: 3.1

See ITS: 3.3.6.2

See CTS RETS: 3.1

See ITS: 3.3.7.2

See CTS RETS: 2.1

See ITS: 3.7.5

See ITS: 3.3.6.2

Specification 3.3.6.2

Table 3.3.6.2-1 Secondary Containment Isolation Instrumentation

TABLE 3.10-2

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RADIATION MONITORING SYSTEMS[a]

Function / Instrument Channels	Instrument Check	Instrument Channel Functional Test[b]	Instrument Channel Calibration [SR 3.3.6.2.3]	Logic System Function Test[b] SR 3.3.6.2.6
Main Stack Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
Refuel Area Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly - 3	--
Reactor Building Area Exhaust Monitors, Recorders, and Isolation	Daily	Quarterly	Quarterly - 3	Once per 24 Months
Turbine Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
Radwaste Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
SJAE Radiation Monitors/Offgas Line Isolation	Daily	Quarterly	Quarterly	Once per 24 Months
Main Control Room Ventilation Monitor	Daily	Quarterly	Quarterly	--
Mechanical Vacuum Pump Isolation[d]	--	--	--	Once per 24 Months
Liquid Radwaste Discharge Monitor/ Isolation[c,d,e,f]	Daily When Discharging	Quarterly	Quarterly	Once per 24 Months
Liquid Radwaste Discharge Flow Rate Measuring Devices[d]	Daily	Quarterly	Once per 18 Months	--
Liquid Radwaste Discharge Radioactivity Recorder[d]	Daily	Quarterly	Once per 18 Months	--
Normal Service Water Effluent	Daily	Quarterly	Quarterly	--
SBGTS Actuation	--	--	--	Once per 24 Months

A1 A7 A8 A6 M4 LT 12 hours A8 M4 [4] [3] LA4 [SR 3.3.6.2.6] [Functions 3, 4] see CTS RETS 3.1 see CTS RETS: 3.1 see ITS: 3.25 see ITS: 3.3.7.1 see ITS: 3.3.2.2 see CTS RETS: 2.1

add SR 3.3.6.2.6 for Function 4 M5

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(A8)

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TABLE 3.10-2

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RADIATION MONITORING SYSTEMS[a]

(A4)

See ITS: 3.3.6.2, 3.3.7.2, 3.7.5, 3.6.4.2, 3.6.4.3

Instrument Channels	Instrument Check[b] (M2) [SR 3.3.7.1.1]	Instrument Channel Functional Test[d]	Instrument Channel Calibration SR 3.3.7.1.2	Logic System Function Test[f][g][h]	
Main Stack Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--	See CTS RETS: 3.1
Refuel Area Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--	See ITS: 3.3.6.2, 3.6.4.2
Reactor Building Area Exhaust Monitors, Recorders, and Isolation	Daily	Quarterly	Quarterly	Once per 24 Months	
Turbine Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--	See CTS RETS: 3.1
Radwaste Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--	
SJAE Radiation Monitors/Offgas Line Isolation	Daily	Quarterly	Quarterly	Once per 24 Months	See ITS 3.7.5
Main Control Room Ventilation Monitor	[1]- Daily (12 hours) (M2)	Quarterly (A4)	Quarterly -[2]	--	
Mechanical Vacuum Pump Isolation[d]	--	--	--	Once per 24 Months	See ITS: 3.3.7.2
Liquid Radwaste Discharge Monitor/ Isolation[c][d][e][i]	Daily When Discharging	Quarterly	Quarterly	Once per 24 Months	
Liquid Radwaste Discharge Flow Rate Measuring Devices[d]	Daily	Quarterly	Once per 18 Months	--	See CTS RETS 2.1
Liquid Radwaste Discharge Radioactivity Recorder[d]	Daily	Quarterly	Once per 18 Months	--	
Normal Service Water Effluent	Daily	Quarterly	Quarterly	--	
SBGTS Actuation	--	--	--	Once per 24 Months	

See ITS 3.3.6.2, 3.6.4.3

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TABLE 3.10-2

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RADIATION MONITORING SYSTEMS[a]

see ITS: 3.3.7.1 3.3.6.1 CTS RETS 2.1, 3.1

Instrument Channels	Instrument Check[b]	Instrument Channel Functional Test[b]	Instrument Channel Calibration	Logic System Function Test[j][k][l] [SR 3.3.7.2.4]	
Main Stack Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--	see CTS RETS 3.1
Refuel Area Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--	see ITS: 3.3.6.2
Reactor Building Area Exhaust Monitors, Recorders, and Isolation	Daily	Quarterly	Quarterly	Once per 24 Months	
Turbine Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--	see CTS RETS 3.1
Radwaste Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--	
SJAE Radiation Monitors/Offgas Line Isolation	Daily	Quarterly	Quarterly	Once per 24 Months	see ITS: 3.7.5
Main Control Room Ventilation Monitor	Daily	Quarterly	Quarterly	--	see ITS 3.3.7.1
[SR 3.3.7.2.4] Mechanical Vacuum Pump Isolation[g]	--	--	--	Once per 24 Months [4]	
Liquid Radwaste Discharge Monitor/ Isolation[c][d][e][f]	Daily When Discharging	Quarterly	Quarterly	Once per 24 Months	see CTS RETS: 2.1
Liquid Radwaste Discharge Flow Rate Measuring Devices[d]	Daily	Quarterly	Once per 18 Months	--	
Liquid Radwaste Discharge Radioactivity Recorder[d]	Daily	Quarterly	Once per 18 Months	--	
Normal Service Water Effluent	Daily	Quarterly	Quarterly	--	
SBGTS Actuation	--	--	--	Once per 24 Months	

see ITS: 3.3.6.2 3.6.4.3

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Specification 3.6.4.2

(A1)

TABLE 3.10-2

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RADIATION MONITORING SYSTEMS[a]

[SR 3.6.4.2.3]

See ITS 3.3.6.2 also

[SR 3.6.4.2.3]

Instrument Channels	Instrument Check[b]	Instrument Channel Functional Test[b]	Instrument Channel Calibration	Logic System Function Test[b][c]
Main Stack Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
Refuel Area Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
Reactor Building Area Exhaust Monitors, Recorders, and Isolation	Daily	Quarterly	Quarterly	Once per 24 Months
Turbine Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
Radwaste Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
SJAE Radiation Monitors/Offgas Line Isolation	Daily	Quarterly	Quarterly	Once per 24 Months
Main Control Room Ventilation Monitor	Daily	Quarterly	Quarterly	--
Mechanical Vacuum Pump Isolation[g]	--	--	--	Once per 24 Months
Liquid Radwaste Discharge Monitor/ Isolation[c][d][e][f]	Daily When Discharging	Quarterly	Quarterly	Once per 24 Months
Liquid Radwaste Discharge Flow Rate Measuring Devices[d]	Daily	Quarterly	Once per 18 Months	--
Liquid Radwaste Discharge Radioactivity Recorder[d]	Daily	Quarterly	Once per 18 Months	--
Normal Service Water Effluent	Daily	Quarterly	Quarterly	--
SBGTS Actuation	--	--	--	Once per 24 Months

See CTS RETS 3.1

add SR 3.6.4.2.3

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See CTS RETS 3.1

See ITS 3.2.5

See ITS 3.3.7.1

See ITS 3.3.2.2

See CTS RETS 2.1

See ITS: 3.3.6.2
3.6.4.3

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[RETS]

See CTS RETS: 2.1, 3.1
ITS: 3.3.7.1
3.3.7.2
3.3.6.2

Specification 3.6.4.3

(A1)

SR 3.6.4.3.3

TABLE 3.10-2

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RADIATION MONITORING SYSTEMS[a]

Instrument Channels	Instrument Check[b]	Instrument Channel Functional Test[d]	Instrument Channel Calibration	Logic System Function Test[f][g]
Main Stack Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
Refuel Area Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
Reactor Building Area Exhaust Monitors, Recorders, and Isolation	Daily	Quarterly	Quarterly	Once per 24 Months
Turbine Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
Radwaste Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
SJAE Radiation Monitors/Offgas Line Isolation	Daily	Quarterly	Quarterly	Once per 24 Months
Main Control Room Ventilation Monitor	Daily	Quarterly	Quarterly	--
Mechanical Vacuum Pump Isolation[e]	--	--	--	Once per 24 Months
Liquid Radwaste Discharge Monitor/ Isolation[c][d][e][f]	Daily When Discharging	Quarterly	Quarterly	Once per 24 Months
Liquid Radwaste Discharge Flow Rate Measuring Devices[d]	Daily	Quarterly	Once per 18 Months	--
Liquid Radwaste Discharge Radioactivity Recorder[d]	Daily	Quarterly	Once per 18 Months	--
Normal Service Water Effluent	Daily	Quarterly	Quarterly	--
SBGTS Actuation	--	--	--	Once per 24 Months

See CTS RETS 3.1

See ITS: 3.3.6.2

See CTS RETS 3.1

See ITS 3.7.5

See ITS 3.3.7.1

See ITS: 3.3.7.2

[SR 3.6.4.3.3]

See CTS: RETS 2.1

See ITS: 3.3.6.2

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Specification 3.7.5

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TABLE 3.10-2

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RADIATION MONITORING SYSTEMS[a]

Instrument Channels	Instrument Check[b]	Instrument Channel Functional Test[d]	Instrument Channel Calibration	Logic System Function Test[iii][e]
Main Stack Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
Refuel Area Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
Reactor Building Area Exhaust Monitors, Recorders, and Isolation	Daily	Quarterly	Quarterly	Once per 24 Months
Turbine Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
Radwaste Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
SJAE Radiation Monitors/Offgas Line Isolation	Daily	Quarterly	Quarterly	Once per 24 Months
Main Control Room Ventilation Monitor	Daily	Quarterly	Quarterly	--
Mechanical Vacuum Pump Isolation[g]	--	--	--	Once per 24 Months
Liquid Radwaste Discharge Monitor/ Isolation[c][i][j][l][f]	Daily When Discharging	Quarterly	Quarterly	Once per 24 Months
Liquid Radwaste Discharge Flow Rate Measuring Devices[g]	Daily	Quarterly	Once per 18 Months	--
Liquid Radwaste Discharge Radioactivity Recorder [g]	Daily	Quarterly	Once per 18 Months	--
Normal Service Water Effluent	Daily	Quarterly	Quarterly	--
SBGTS Actuation	--	--	--	Once per 24 Months

See CTS RETS 3.1

see ITS: 3.3.6.2

see CTS RETS: 3.1

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See ITS: 3.3.7.1

see ITS 3.3.7.2

see CTS RETS: 2.1

See ITS: 3.6.4.2 3.6.4.3

CTS RETS 2.1

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TABLE 3.10-2

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RADIATION MONITORING SYSTEMS[a]

	Instrument Channels	Instrument Check[b]	Instrument Channel Functional Test[i]	Instrument Channel Calibration	Logic System Function Test[i][j][k]
(1)	Main Stack Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
(2)	Refuel Area Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
(3)	Reactor Building Area Exhaust Monitors, Recorders, and Isolation	Daily	Quarterly	Quarterly	Once per 24 Months
(4)	Turbine Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
(5)	Radwaste Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
(6)	SJAE Radiation Monitors/Offgas Line Isolation	Daily	Quarterly	Quarterly	Once per 24 Months
(7)	Main Control Room Ventilation Monitor	Daily	Quarterly	Quarterly	--
(8)	Mechanical Vacuum Pump Isolation[g]	--	--	--	Once per 24 Months
(9)	Liquid Radwaste Discharge Monitor/ Isolation[c][d][e][f]	Daily When Discharging	Quarterly	Quarterly	Once per 24 Months
(10)	Liquid Radwaste Discharge Flow Rate Measuring Devices[d]	Daily	Quarterly	Once per 18 Months	--
(11)	Liquid Radwaste Discharge Radioactivity Recorder[d]	Daily	Quarterly	Once per 18 Months	--
(12)	Normal Service Water Effluent	Daily	Quarterly	Quarterly	--
(13)	SBGTS Actuation	--	--	--	Once per 24 Months

See CTS RETS 3.1

See ITS: 3.3.6.2

See ITS: 3.7.5

See ITS: 3.3.7.1

See ITS: 3.3.2.2

See ITS: 3.3.6.2 3.6.4.3

R1

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CTS RETS 3.1

(R1)

TABLE 3.10-2

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RADIATION MONITORING SYSTEMS[a]

Instrument Channels	Instrument Check[b]	Instrument Channel Functional Test[i]	Instrument Channel Calibration	Logic System Function Test[i][j][k]
Main Stack Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
Refuel Area Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
Reactor Building Area Exhaust Monitors, Recorders, and Isolation	Daily	Quarterly	Quarterly	Once per 24 Months
Turbine Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
Radwaste Building Exhaust Monitors and Recorders	Daily	Quarterly	Quarterly	--
SJAE Radiation Monitors/Offgas Line Isolation	Daily	Quarterly	Quarterly	Once per 24 Months
Main Control Room Ventilation Monitor	Daily	Quarterly	Quarterly	--
Mechanical Vacuum Pump Isolation[g]	--	--	--	Once per 24 Months
Liquid Radwaste Discharge Monitor/ Isolation[c][d][e][f]	Daily When Discharging	Quarterly	Quarterly	Once per 24 Months
Liquid Radwaste Discharge Flow Rate Measuring Devices[d]	Daily	Quarterly	Once per 18 Months	--
Liquid Radwaste Discharge Radioactivity Recorder[d]	Daily	Quarterly	Once per 18 Months	--
Normal Service Water Effluent	Daily	Quarterly	Quarterly	--
SBGTS Actuation	--	--	--	Once per 24 Months

see ITS: 3.3.6.2

(R1)

see ITS: 3.7.5

see ITS: 3.3.7.2

see CTS&RETS: 2.1

see ITS: 3.3.6.2 3.6.4.3

see ITS: 3.3.7.1

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Specification 3.3.6.1

A1

NOTES FOR TABLE 3.10-2

(a) Functional tests, calibrations and instrument checks need not be performed when these instruments are not required to be operable or are tripped. A9

(b) Instrument checks shall be performed at least once per day during these periods when the instruments are required to be operable. M9

(c) A source check shall be performed prior to each release. see CTS RETS! 2.1

(d) Liquid radwaste effluent line instrumentation surveillance requirements need not be performed when the instruments are not required as the result of the discharge path not being utilized.

(e) An instrument channel calibration shall be performed with known radioactive sources standardized on plant equipment which has been calibrated with NBS traceable standards.

see ITS! 3.6.4.2 3.6.4.3 CTS RETS 2.1, 3.1

(f) Simulated automatic actuation shall be performed once per 24 months. Where possible, all logic system functional tests will be performed using the test jacks. L14

(g) Refer to Appendix A for instrument channel functional test and instrument channel calibration requirements (Table 4.2-1). These requirements are performed as part of main steam high radiation monitor surveillances. see ITS: 3.3.7.2

(h) The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems. A11

(i) This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. These instrument channels will be calibrated using simulated electrical signals once every three months. A10

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Table 3.3.6.2-1 Secondary Containment Isolation Instrumentation

Specification 3.3.6.2

A1

A5

NOTES FOR TABLE 3.10-2

(a) Functional tests, calibrations and instrument checks need not be performed when these instruments are not required to be operable or are tripped.

(b) Instrument checks shall be performed at least once per day during these periods when the instruments are required to be operable. M4

(c) A source check shall be performed prior to each release.

(d) Liquid radwaste effluent line instrumentation surveillance requirements need not be performed when the instruments are not required as the result of the discharge path not being utilized.

(e) An instrument channel calibration shall be performed with known radioactive sources standardized on plant equipment which has been calibrated with NBS traceable standards.

See CTS RETS: 2.1

See ITS: 3.6.4.2 3.6.4.3 CTS RETS 2.1, 3.1

(f) Simulated automatic actuation shall be performed once per 24 months. Where possible, all logic system functional tests will be performed using the test jacks. A13

RAI 3.3.6.2-2

(g) Refer to Appendix A for instrument channel functional test and instrument channel calibration requirements (Table 4.2-1). These requirements are performed as part of main steam high radiation monitor surveillances.

See ITS: 3.3.7.2

(h) The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems. A7

(i) This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. These instrument channels will be calibrated using simulated electrical signals once every three months.

A6

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Specification 3.3.7.1

A1

NOTES FOR TABLE 3.10-2

A5

(a) Functional tests, calibrations and instrument checks need not be performed when these instruments are not required to be operable or are tripped.

M2

(b) Instrument checks shall be performed at least once per day during these periods when the instruments are required to be operable.

(c) A source check shall be performed prior to each release.

(d) Liquid radwaste effluent line instrumentation surveillance requirements need not be performed when the instruments are not required as the result of the discharge path not being utilized.

See CTS RETS: 2.1

(e) An instrument channel calibration shall be performed with known radioactive sources standardized on plant equipment which has been calibrated with NBS traceable standards.

ITS: 3.6.4.2
3.6.4.3
CTS RETS 2.1, 3.1

(f) Simulated automatic actuation shall be performed once per 24 months. Where possible, all logic system functional tests will be performed using the test jacks.

See ITS: 3.3.7.2

(g) Refer to Appendix A for instrument channel functional test and instrument channel calibration requirements (Table 4.2-1). These requirements are performed as part of main steam high radiation monitor surveillances.

(h) The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

See ITS:
3.3.6.1
3.3.6.2
3.3.7.2
3.7.5
CTS RETS: 2.1
3.1

(i) This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. These instrument channels will be calibrated using simulated electrical signals once every three months.

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Specification 3.3.7.2.

A1

see CTS RETS 2.1, 3.1
ITS: 3.3.7.1
3.3.6.2
3.7.5

NOTES FOR TABLE 3.10-2

(a) Functional tests, calibrations and instrument checks need not be performed when these instruments are not required to be operable or are tripped.

(b) Instrument checks shall be performed at least once per day during these periods when the instruments are required to be operable.

(c) A source check shall be performed prior to each release.

(d) Liquid radwaste effluent line instrumentation surveillance requirements need not be performed when the instruments are not required as the result of the discharge path not being utilized.

see CTS RETS: 2.1

(e) An instrument channel calibration shall be performed with known radioactive sources standardized on plant equipment which has been calibrated with NBS traceable standards.

L5

actual L4

[SR 3.3.7.2.4] (f) Simulated automatic actuation shall be performed once per 24 months. Where possible, all logic system functional tests will be performed using the test jacks.

A9

(g) Refer to Appendix A for instrument channel functional test and instrument channel calibration requirements (Table 4.2-1). These requirements are performed as part of main steam high radiation monitor surveillances.

A10

(h) The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

(i) This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. These instrument channels will be calibrated using simulated electrical signals once every three months.

see ITS: 3.3.6.2
3.3.7.1
3.7.5
CTS RETS: 2.1
3.1

Specification 3.6.4.2

A1

~~NOTES FOR TABLE 3.10-2~~

See ITS 3.3.7.1 3.3.6.2 3.7.5 CTS RETS: 3.1 2.1

(a) Functional tests, calibrations and instrument checks need not be performed when these instruments are not required to be operable or are tripped.

(b) Instrument checks shall be performed at least once per day during these periods when the instruments are required to be operable.

(c) A source check shall be performed prior to each release.

(d) Liquid radwaste effluent line instrumentation surveillance requirements need not be performed when the instruments are not required as the result of the discharge path not being utilized.

See CTS RETS: 2.1

(e) An instrument channel calibration shall be performed with known radioactive sources standardized on plant equipment which has been calibrated with NBS traceable standards.

[SR 3.6.4.2.3]

(f) Simulated automatic or actual actuation shall be performed once per 24 months. Where possible, all logic system functional tests will be performed using the test jacks. L7 A6

RAI 3.6.4.2-3

(g) Refer to Appendix A for instrument channel functional test and instrument channel calibration requirements (Table 4.2-1). These requirements are performed as part of main steam high radiation monitor surveillances.

See ITS: 3.3.7.2

(h) The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

See ITS: 3.3.6.1 3.3.6.2 3.3.7.2 3.7.5 CTS RETS: 2.1, 3.1

(i) This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. These instrument channels will be calibrated using simulated electrical signals once every three months.

See ITS: 3.3.7.1 3.3.6.2 3.7.5 CTS RETS: 3.1 2.1

Specification 3.6.4.3

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~~NOTES FOR TABLE 3.10-2~~

(a) Functional tests, calibrations and instrument checks need not be performed when these instruments are not required to be operable or are tripped.

(b) Instrument checks shall be performed at least once per day during these periods when the instruments are required to be operable.

See ITS: 3.3.7.1, 3.3.6.2, 3.7.5
CTS RETS: 3.1, 2.1

(c) A source check shall be performed prior to each release.

(d) Liquid radwaste effluent line instrumentation surveillance requirements need not be performed when the instruments are not required as the result of the discharge path not being utilized.

(e) An instrument channel calibration shall be performed with known radioactive sources standardized on plant equipment which has been calibrated with NBS traceable standards.

see CTS RETS 2.1

[SR 3.6.4.3.3] (f) Simulated or actual automatic actuation shall be performed once per 24 months. Where possible, all logic system functional tests will be performed using the test jacks.

L4 A6 RAI 3.6.4.3-7

(g) Refer to Appendix A for instrument channel functional test and instrument channel calibration requirements (Table 4.2-1). These requirements are performed as part of main steam high radiation monitor surveillances.

See ITS 3.3.7.2

(h) The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

(i) This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. These instrument channels will be calibrated using simulated electrical signals once every three months.

See ITS: 3.3.7.1, 3.3.6.2, 3.7.5
CTS RETS: 3.1, 2.1

See ITS: 3.3.6.1, 3.3.6.2, 3.3.7.2, 3.7.5
CTS RETS: 2.1, 3.1

ITS 3.7.5

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NOTES FOR TABLE 3.10-2

(a) Functional tests, calibrations and instrument checks need not be performed when these instruments are not required to be operable or are tripped.

(b) Instrument checks shall be performed at least once per day during these periods when the instruments are required to be operable.

(c) A source check shall be performed prior to each release. See CTS RETS: 2.1

(d) Liquid radwaste effluent line instrumentation surveillance requirements need not be performed when the instruments are not required as the result of the discharge path not being utilized.

(e) An instrument channel calibration shall be performed with known radioactive sources standardized on plant equipment which has been calibrated with NBS traceable standards.

R1

(f) Simulated automatic actuation shall be performed once per 24 months. Where possible, all logic system functional tests will be performed using the test jacks.

See ITS: 3.3.7.2

(g) Refer to Appendix A for instrument channel functional test and instrument channel calibration requirements (Table 4.2-1). These requirements are performed as part of main steam high radiation monitor surveillances.

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(h) The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

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(i) This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. These instrument channels will be calibrated using simulated electrical signals once every three months.

CTS RETS: 2.1

NOTES FOR TABLE 3.10-2

(a) Functional tests, calibrations and instrument checks need not be performed when these instruments are not required to be operable or are tripped.

(b) Instrument checks shall be performed at least once per day during these periods when the instruments are required to be operable.

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(c) A source check shall be performed prior to each release.

(d) Liquid radwaste effluent line instrumentation surveillance requirements need not be performed when the instruments are not required as the result of the discharge path not being utilized.

(e) An instrument channel calibration shall be performed with known radioactive sources standardized on plant equipment which has been calibrated with NBS traceable standards.

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(f) Simulated automatic actuation shall be performed once per 24 months. Where possible, all logic system functional tests will be performed using the test jacks.

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(g) Refer to Appendix A for instrument channel functional test and instrument channel calibration requirements (Table 4.2-1). These requirements are performed as part of main steam high radiation monitor surveillances.

see ITS: 3.3.7.2

(h) The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

(i) This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. These instrument channels will be calibrated using simulated electrical signals once every three months.

R1

CTS RETS 3.1

NOTES FOR TABLE 3.10-2

(a) Functional tests, calibrations and instrument checks need not be performed when these instruments are not required to be operable or are tripped.

(b) Instrument checks shall be performed at least once per day during these periods when the instruments are required to be operable.

R1

(c) A source check shall be performed prior to each release.

(d) Liquid radwaste effluent line instrumentation surveillance requirements need not be performed when the instruments are not required as the result of the discharge path not being utilized.

See CTS RETS: 2.1

(e) An instrument channel calibration shall be performed with known radioactive sources standardized on plant equipment which has been calibrated with NBS traceable standards.

See ITS: 3.6.4.2 3.6.4.3 3.3.7.2

(f) Simulated automatic actuation shall be performed once per 24 months. Where possible, all logic system functional tests will be performed using the test jacks.

(g) Refer to Appendix A for instrument channel functional test and instrument channel calibration requirements (Table 4.2-1). These requirements are performed as part of main steam high radiation monitor surveillances.

See ITS: 3.3.7.2

(h) The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

R1

(i) This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. These instrument channels will be calibrated using simulated electrical signals once every three months.

R1

CTS RETS 4.0

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

4.0 SOLID RADIOACTIVE WASTE

4.1 PROCESS CONTROL PROGRAM

Applicability

Applies to radioactive solid waste packages for offsite shipment.

Objective

To ensure that solid radioactive waste meet 10 CFR 20, 10 CFR 61 and 10 CFR 71 shipping and burial ground requirements prior to shipping.

Specifications

a. The solid radwaste system shall be used in accordance with the PCP to process wet radioactive wastes to meet shipping and burial ground requirements.

4.1 PROCESS CONTROL PROGRAM

Applicability

Applies to the solidification system utilized for wet solid wastes.

Objective

To ensure that solidification of wet solid wastes is performed in accordance with the PCP.

Specifications

a. The PCP shall be used to verify the solidification of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste.

1. If any test specimen fails to verify solidification, the solidification of the batch under test shall be suspended until the following are completed:

   a) Additional test specimens can be obtained;

   b) Alternative solidification parameters can be determined in accordance with the PCP; and

K1

CTS RETS 4.0

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	c) A subsequent test verifies solidification. Solidification of the batch may then be resumed using the parameters determined by the PCP. 2. If the initial test specimen from a batch of waste fails to verify solidification, the PCP shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate solidification. The PCP shall be modified as required, to assure solidification of subsequent batches of waste.
b. With the provisions of the PCP not satisfied, suspend shipments of affected packaged solid radioactive wastes from the site.	

R1

CTS RETS 5.0

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

5.0 TOTAL DOSE

5.1 TOTAL DOSE FROM URANIUM FUEL CYCLE

Applicability

Applies to radiation dose from releases of radioactivity and radiation from uranium fuel cycle sources.

Objective

To assure that the requirements of 40 CFR 190 are met.

Specifications

a. The dose or dose commitment to any member of the public, due to releases of radioactivity and radiation, from uranium fuel cycle sources shall be limited as follows:

1. Less than or equal to 25 mrem/year to the whole body; and,

2. Less than or equal to 25 mrem/year to any organ except the thyroid which shall be limited to less than or equal to 75 mrem/year.

b. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits

5.1 TOTAL DOSE FROM URANIUM FUEL CYCLE

Applicability

Applies to the calculation of total dose due to releases of radioactivity and radiation from uranium fuel cycle sources.

Objective

To ensure that appropriate calculations are performed to determine total dose to a member of the public.

Specifications

a. Dose Calculations Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 2.3.a, 3.3.a, and 3.4.a and in accordance with the ODCM.

b. Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined

R1

CTS RETS 5.0

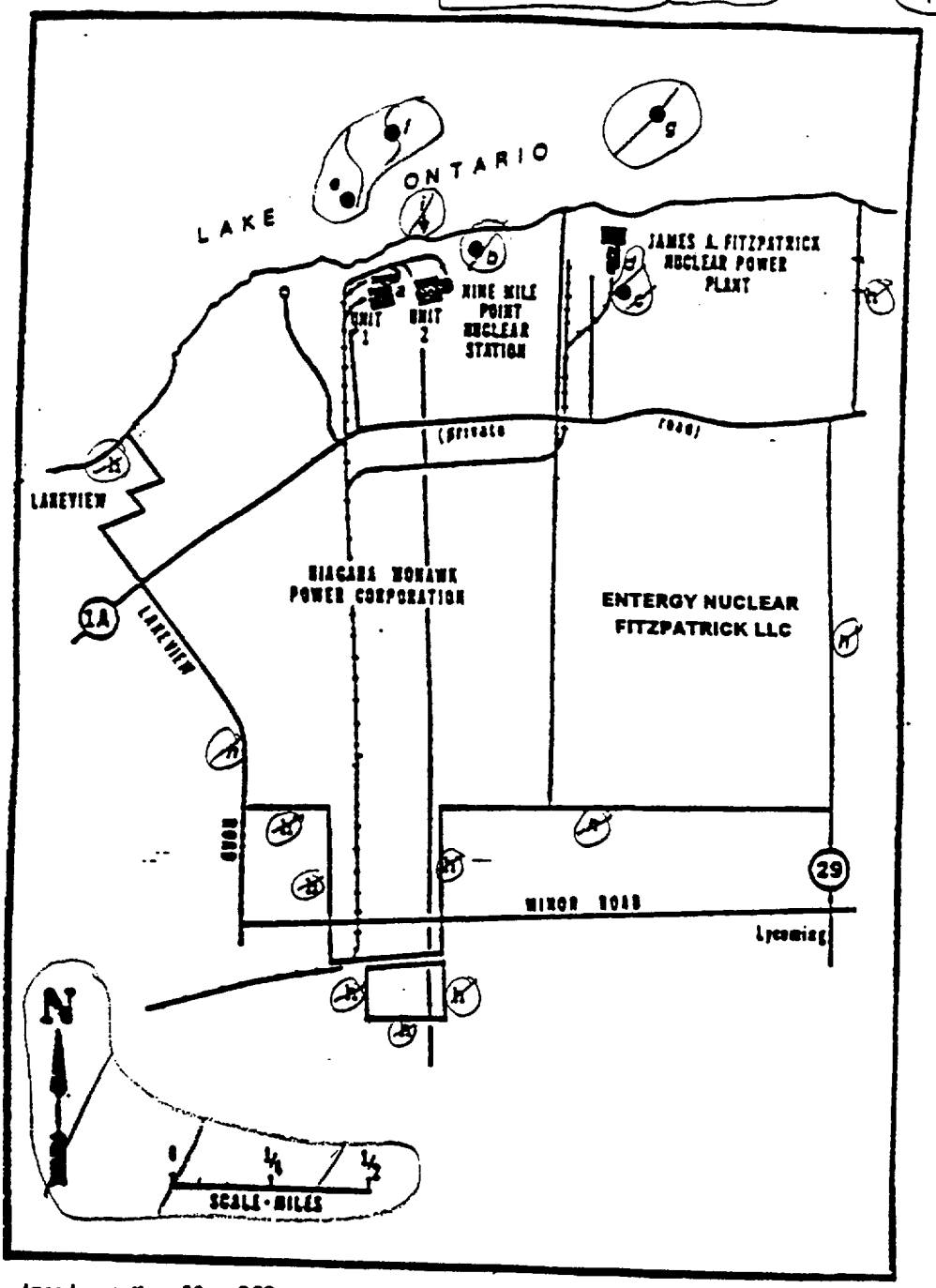
LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
of Specification 2.3.a or 3.3.a or 3.4.a, calculations shall be made including an estimate of direct radiation contributions to determine whether the limits of 5.1 have been exceeded. If this is the case, a report defining corrective actions to be taken to reduce subsequent releases to levels within limits, along with a schedule for achieving conformance, shall be prepared and submitted to the Commission within 30 days. This report, as defined in 10 CFR 20.405c, shall include estimates of the radiation exposure (dose) to a member of the public from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentration of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceed(s) the above limits, and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the report shall include a request for variance in accordance with the provisions of 40 CFR 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.	in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in Specification 5.1.b.

TR1

Chapter 4.0

A1

FIGURE 4.1-1 IES
SITE BOUNDARY and Exclusion Area A1 LA1



[RETS]

AMEND # 268

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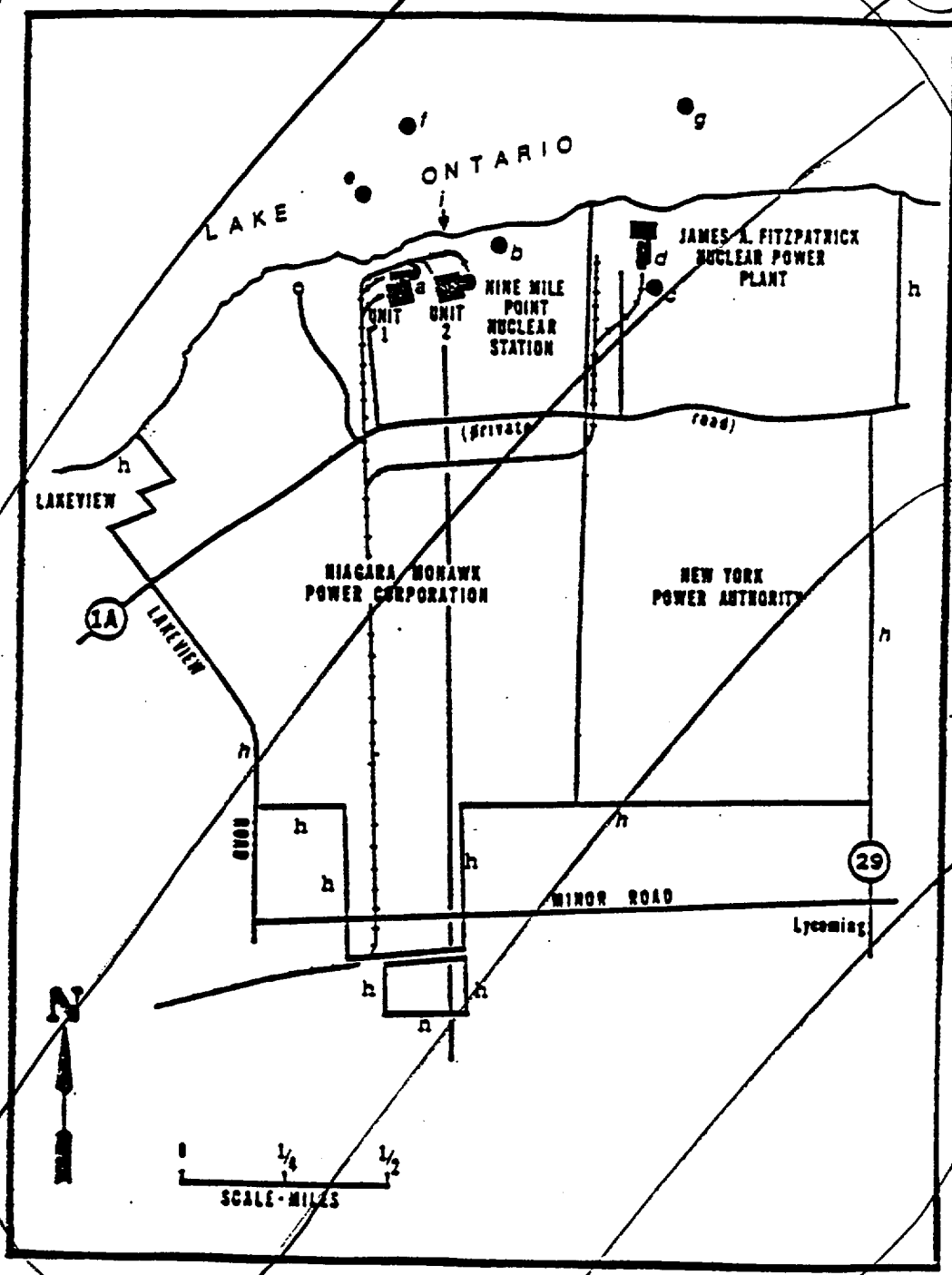
REVISION D

FIGURE 5.1-1
SITE BOUNDARY MAP



[RETS]

Chapter 4.0

(A1)

NOTES TO FIGURE 5.1-1

(a) NMP1 stack (height is 350 feet)

(b) NMP2 stack (height is 430 feet)

(c) JAFNPP stack (height is 385 feet)

(d) Building vents

(e) NMP1 radioactive liquid discharge (Lake Ontario, bottom)

(f) NMP2 radioactive liquid discharge (Lake Ontario, bottom)

(g) JAFNPP radioactive liquid discharge (Lake Ontario, bottom)

(h) Site boundary

(i) Lake Ontario shoreline

(LA1)

Additional Information:

-- NMP2 reactor building vent is located 187 feet above ground level

-- JAFNPP reactor and turbine building vents are located 173 feet above ground level

-- JAFNPP radwaste building vent is 112 feet above ground level

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NOTES TO FIGURE 5.1-1

(a) NMP1 stack (height is 350 feet)

(b) NMP2 stack (height is 430 feet)

(c) JAFNPP stack (height is 385 feet)

(d) Building vents

(e) NMP1 radioactive liquid discharge (Lake Ontario, bottom)

(f) NMP2 radioactive liquid discharge (Lake Ontario, bottom)

(g) JAFNPP radioactive liquid discharge (Lake Ontario, bottom)

(h) Site boundary

(i) Lake Ontario shoreline

R1

Additional Information:

-- NMP2 reactor building vent is located 187 feet above ground level

-- JAFNPP reactor and turbine building vents are located 173 feet above ground level

-- JAFNPP radwaste building vent is 112 feet above ground level

CTS RETS 6.1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

6.0 RADIOLOGICAL ENVIRONMENTAL MONITORING

6.1 MONITORING PROGRAM

Applicability

At all times.

Objective

To evaluate the effects of plant operation on the environs and to verify the effectiveness of the controls on radioactive material.

Specifications

a. With the radiological environmental monitoring program not being conducted as specified in Table 6.1-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

(Deviations are permitted from the required sampling schedule if samples are unobtainable due to hazardous conditions, seasonal unavailability, theft, uncooperative residents, or to malfunction of automatic sampling equipment. If the latter, efforts shall be made to complete corrective action prior to the end of the next sampling period.)

6.1 MONITORING PROGRAM

The radiological environmental monitoring samples shall be collected, pursuant to Table 6.1-1, from the locations given in the table and figure(s) in the ODCM and shall be analyzed pursuant to the requirements of Table 6.1-1, and the detection capabilities required by Table 6.1-3.

K1

CTS RETS 6.1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

b. With the level of radioactivity (as the result of plant effluents) in an environmental sampling medium at a specified location exceeding the reporting levels of Table 6.1-2 when averaged over any calendar quarter, prepare and submit to the Commission a report within thirty (30) days from the end of the affected calendar quarter or within thirty (30) days from the time it is determined that a reporting level has been exceeded. This report shall identify the cause(s) for exceeding the limit(s) and define the corrective action(s) to be taken to reduce radioactive effluents so that the calculated annual dose to a member of the public is less than the calendar year limits of Specifications 2.3, 3.3, and 3.4. When more than one of the radionuclides in Table 6.1-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 6.1-2 are detected and are the result of plant effluents, this report shall be submitted if the calculated annual dose to an individual is equal to or greater than the calendar year limits of Specification 2.3, 3.3, and 3.4.

This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

R.1

(RETS)

CTS RETS 6.1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 6.1-1, locations for obtaining replacement samples shall be identified and added to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. The cause of the unavailability of samples and the new location(s) for obtaining replacement samples shall be identified in the next Semi-annual Radioactive Effluent Release Report. Also included in the report shall be a revised figure(s) and table for the ODCM reflecting the new location(s).	

CTS RETS 6.1

TABLE 6.1-1

OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples[a] and Locations	Sampling and Collection Frequency[a]	Type and Frequency of Analysis
AIRBORNE			
Radioiodine and Particulates	Samples from 5 locations: a. 3 samples from offsite locations in different sectors of the highest calculated site average D/Q (based on all licensed site reactors). b. 1 sample from the vicinity of a community having the highest calculated site average D/Q (based on all licensed site reactors). c. 1 sample from a control location 9 to 20 miles distant and in the least prevalent wind direction[d].	Continuous sample operation with sample collection weekly or as required by dust loading, whichever is more frequent.	Radioiodine Canisters: Analyze weekly for I-131. Particulate Samples: Gross beta radioactivity[b] following filter change, composite (by location) for gamma isotopic[c] quarterly (as a minimum).
Direct Radiation[e]	32 stations with two or more dosimeters placed as follows: An inner ring of stations in the general area of the site boundary and an outer ring in the 4 to 5 mile range from the site with a station in each of the land based sectors of each ring. There are 16 land based sectors in the inner ring and 8 land based sectors in the outer ring. The balance of the stations (8) are placed in special interest areas such as population centers, nearby residences, schools, and in 2 or 3 areas to serve as control stations.	Quarterly	Gamma dose monthly or quarterly.

K1

TABLE 6.1-1 (continued)

Exposure Pathway and/or Sample	Number of Samples[a] and Locations	Sampling and Collection Frequency[a]	Type and Frequency of Analysis
WATERBORNE			
Surface[f]	a. 1 sample upstream. b. 1 sample from the site's most downstream cooling water intake[d].	Composite sample over one month period[g].	Gamma isotopic analysis monthly. Composite for Tritium analysis quarterly[c].
Sediment from Shoreline	1 sample from a downstream area with existing or potential recreational value.	Twice per year.	Gamma isotopic analysis semiannually[c].
INGESTION			
Milk	a. Samples from milch animals in 3 locations within 3.5 miles distant having the highest calculated site average D/Q. If there are none, then 1 sample from milch animals in each of 3 areas 3.5 to 5.0 miles distant having the highest calculated site average D/Q (based on all licensed site reactors)[h]. b. 1 sample from milch animals at a control location (9 to 20 miles distant and in a less prevalent wind direction)[d].	Twice per month, April through December (samples will be collected in January through March if I-131 is detected in November and December of the preceding year).	Gamma isotopic and I-131 analysis twice per month when milch animals are on pasture (April through December); monthly (January through March), if required[c].

R1

CTS RETS 6.1

TABLE 6.1-1 (continued)

Exposure Pathway and/or Sample	Number of Samples[a] and Locations	Sampling and Collection Frequency[a]	Type and Frequency of Analysis
Fish	a. 1 sample of each of 2 commercially or recreationally important species in the vicinity of a site discharge point. b. 1 sample of each of 2 species (same as in a. above or of a species with similar feeding habits) from an area at least 5 miles distant from the site[d].	Twice per year.	Gamma isotopic[c] analysis of edible portions. R1
Food Products	a. In lieu of the garden census as specified in 6.2, samples of at least 3 different kinds of broad leaf vegetation (such as vegetables) grown nearest each of two different offsite locations of highest predicted site average D/Q (Based on all licensed site Reactors). One (1) sample of each of the similar broad leaf vegetation grown at least 9.3 miles distant in a least prevalent wind direction sector[d].	Once during harvest season.	Gamma isotopic[c] analysis of edible portions. (Isotopic to include I-131.)

CTS RETS 6.1

R1

NOTES FOR TABLE 6.1-1

(a) It is recognized that, at times, it may not be possible or practical to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question. Actual locations (distance and directions) from the site shall be provided in the Annual Radiological Environmental Operating Report. Calculated site averaged D/Q values and meteorological parameters are based on historical data (specified in the ODCM) for all licensed site reactors.

(b) Particulate sample filters should be analyzed for gross beta 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air is greater than 10 times a historical yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

(c) Gamma isotopic analysis means the identification and quantification of gamma emitting radionuclides that may be attributable to the effluents from the plant.

(d) The purpose of these samples is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites which provide valid background data may be substituted.

(e) One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purpose of this table, a thermoluminescent dosimeter may be considered to be one phosphor and two or more phosphors in a pocket may be considered as two or more dosimeters. Film badges shall not be used for measuring direct radiation.

(f) The "upstream sample" shall be taken at a distance beyond significant influence of the discharge. The "downstream sample" shall be taken in an area beyond, but near, the mixing zone, if practical.

(g) Composite samples should be collected with equipment (or equivalent) which is capable of collecting an aliquoit at time intervals which are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure that a representative sample is obtained.

(h) A milk sampling location, as required in Table 6.1-1 is defined as a location having at least 10 milking cows present at a designated milk sample location. It has been found from past experience, and as a result of conferring with local farmers, that a minimum of 10 milking cows is necessary to guarantee an adequate supply of milk twice per month for analytical purposes. Locations with less than 10 milking cows are usually utilized for breeding purposes which eliminates a stable supply of milk for samples as a result of suckling calves and periods when the adult animals are dry. In the event that 3 milk sample locations cannot meet the requirement for 10 milking cows, then a sample location having less than 10 milking cows can be used if an adequate supply of milk can reasonably and reliably be obtained based on communications with the farmer.

CTS RETS 6.1

TABLE 6.1-2

REPORTING LEVEL FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES
REPORTING LEVELS

R1

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
H-3	30,000				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr/Nb-95	400				
I-131	20	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba/La-140	200			300	

CTS RETS 6.1

(R1)

TABLE 6.1-3

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a]
LOWER LIMIT OF DETECTION (LLD)[b]

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Sediment (pCi/kg.dry)
gross beta	4	0.01				
H-3	3,000					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr/Nb-95	15					
I-131	15[c]	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba/La-140	15			15		

CTS RETS 6.1

NOTES FOR TABLE 6.1-3

(a) The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability and with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation),

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the a priori lower limit of detection, as defined above (in picocurie per unit mass or volume);

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample, as appropriate (in counts per minute);

E is the counting efficiency (in counts per transformation);

V is the sample size (in units of mass or volume);

2.22 is the number of transformations per minute per picocurie;

Y is the fractional radiochemical yield (when applicable);

λ is the radioactive decay constant for the particular radionuclide;

Δt is the elapsed time between sample collection (or end of the sample collection period) and time of counting.

Typical values of E, V, Y, and Δt should be used in the calculations.

(b) It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report.

(c) No drinking water pathway exists at the Nine Mile Point Site under normal operating conditions due to the direction and distance of the nearest drinking water intake. Therefore, an LLD value of 15 pCi/liter is used.

R1

CTS RETS 6.2

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

6.2 LAND USE CENSUS PROGRAM

Applicability

At all times.

Objective

To identify locations of milch animals and gardens of greater than 50 square meters within 3 miles of the site.

Specifications

a. A land use census shall be conducted and shall identify the locations of all milch animals, the nearest residence, and all gardens* of greater than 50 square meters producing fresh leafy vegetables, in each of the 16 meteorological sectors within a distance of 5 miles from the site.

b. With a land use census identifying a milch animal in a location(s) which represents a calculated D/Q value greater than the values currently being used in calculating Surveillance Requirement 3.4, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report.

* Broad leaf vegetation sampling may be performed in lieu of the garden census as specified in Table 6.1-1.

6.2 LAND USE CENSUS PROGRAM

R1

The land use census shall be conducted during the growing season at least once per 12 months using the information that will provide the best results, such as by a door to door survey, aerial survey, or by consulting local agriculture authorities, etc. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
c. With the land use census identifying a milch animal location(s) that represents a calculated D/Q (via the same exposure pathway) 50% greater than at a location from which samples are currently being obtained in accordance with Table 6.1-1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated D/Q (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census is conducted. Identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and include the additions in the ODCM.	R1

CTS RETS 6.3

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
6.3 INTERLABORATORY COMPARISON PROGRAM **Applicability** At all times. **Objective** To provide quality control of environmental sample analyses. **Specifications** a. Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the Commission. Participation in this program shall include all media for which samples are routinely collected and for which intercomparison samples are available. b. With analyses not being performed as required in 6.3.a above, report the corrective actions taken to prevent a recurrence in the Annual Radiological Environmental Operating Report.	**6.3 INTERLABORATORY COMPARISON PROGRAM** A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report.

R1

A1

7.0 ADMINISTRATIVE CONTROLS

[5.1] 7.1 RESPONSIBILITY plant A3

[5.1.1] The [redacted] Manager shall have [redacted] responsibility for [redacted] operation of the [redacted] plant is conducted in [redacted] environment. During periods when the Resident Manager is unavailable, one of the three General Managers will assume this responsibility. In the event all four are unavailable, the Resident Manager may delegate this responsibility to other qualified supervisory personnel during his absence. M1

and shall

in writing the succession to

b. Implementation of the Radiological Effluent Technical Specifications is the responsibility of the General Manager - Operations, with the assistance of the plant staff organization.

See ITS: Section 5.2

7.2 PROCEDURES

Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Section 5 "Facility Administrative Policies and Procedures" of ANSI 18.7-1972 and Regulatory Guide 1.33, November 1972, Appendix A. In addition, procedures shall be established, implemented and maintained for the PCP, ODCM, and Quality Control Program for effluent and environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1.

See ITS: Section 5.4

7.3 REPORTING REQUIREMENTS

a. Planned Liquid and Gaseous Releases

The limits for radioactive materials contained in liquid and gaseous effluents are contained in Specifications 2.3, 3.3 and 3.4.

See ITS: Section 5.6

b. Environmental Samples Exceeding Limits of Table 6.1-2

When the limits of Table 6.1-2 are exceeded, refer to Specification 6.1.b for reporting requirements.

c. Semiannual Radioactive Effluent Release Report

Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

1. The Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid wastes released from the unit using as guidance Regulatory Guide 1.21, Revision 1, June 1974, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", with data summarized on a quarterly basis following the format of Appendix B thereof.

CTS Amend 254

Section 5.2 (A1)

7.0 ADMINISTRATIVE CONTROLS

See ITS Section 5.1

7.1 RESPONSIBILITY

a. The Site Executive Officer shall have direct responsibility for assuring the operation of the James A. FitzPatrick Plant is conducted in such a manner as to provide continuing protection to the environment and shall delegate in writing the succession to this responsibility during his absence.

LA5

b. Implementation of the Radiological Effluent Technical Specifications is the responsibility of the General Manager - Operations, with the assistance of the plant staff organization.

7.2 PROCEDURES

See ITS Section 5.4

Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Section 5 "Facility Administrative Policies and Procedures" of ANSI 18.7-1972 and Regulatory Guide 1.33, November 1972, Appendix A. In addition, procedures shall be established, implemented and maintained for the PCP, ODCM, and Quality Control Program for effluent and environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1.

7.3 REPORTING REQUIREMENTS

See ITS Section 5.6

a. Planned Liquid and Gaseous Releases

The limits for radioactive materials contained in liquid and gaseous effluents are contained in Specifications 2.3, 3.3 and 3.4.

b. Environmental Samples Exceeding Limits of Table 6.1-2

When the limits of Table 6.1-2 are exceeded, refer to Specification 6.1.b for reporting requirements.

c. Semiannual Radioactive Effluent Release Report

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CTS Amend 254

Section 5.4 (A1)

ITS

7.0 ADMINISTRATIVE CONTROLS

See ITS Section 5.1

7.1 RESPONSIBILITY

a. The Site Executive Officer shall have direct responsibility for assuring the operation of the James A. FitzPatrick Plant is conducted in such a manner as to provide continuing protection to the environment and shall delegate in writing the succession to this responsibility during his absence.

b. Implementation of the Radiological Effluent Technical Specifications is the responsibility of the General Manager - Operations, with the assistance of the plant staff organization.

See ITS Section 5.2

[5.4] ~~7.2~~ PROCEDURES

(LA1)

[5.4.1] Written procedures ~~and administrative policies~~ shall be established, implemented and maintained ~~that meet or exceed the requirements and recommendations of Section 5 "Facility Administrative Policies and Procedures" of ANSI 18.7-1972 and Regulatory Guide 1.33, November 1972, Appendix A.~~ In addition, procedures shall be established, implemented and maintained for the ~~PCP~~, ODCM, and Quality Control Program for effluent and environmental monitoring ~~using the guidance in Regulatory Guide 4.1, Revision 1.~~ (A3) (LA1)

[5.4.1.d]
[5.4.1.e]

7.3 REPORTING REQUIREMENTS

See ITS Section 5.6

a. Planned Liquid and Gaseous Releases

The limits for radioactive materials contained in liquid and gaseous effluents are contained in Specifications 2.3, 3.3 and 3.4.

b. Environmental Samples Exceeding Limits of Table 6.1-2

When the limits of Table 6.1-2 are exceeded, refer to Specification 6.1.b for reporting requirements.

c. Semiannual Radioactive Effluent Release Report

Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

1. The Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit using as guidance Regulatory Guide 1.21, Revision 1, June 1974, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", with data summarized on a quarterly basis following the format of Appendix B thereof.

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7.0 ADMINISTRATIVE CONTROLS

See ITS Section 5.4

7.1 RESPONSIBILITY

a. The Site Executive Officer shall have direct responsibility for assuring the operation of the James A. FitzPatrick Plant is conducted in such a manner as to provide continuing protection to the environment and shall delegate in writing the succession to this responsibility during his absence.

b. Implementation of the Radiological Effluent Technical Specifications is the responsibility of the General Manager - Operations, with the assistance of the plant staff organization.

7.2 PROCEDURES

Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Section 5 "Facility Administrative Policies and Procedures" of ANSI 18.7-1972 and Regulatory Guide 1.33, November 1972, Appendix A. In addition, procedures shall be established, implemented and maintained for the PCP, ODCM, and Quality Control Program for effluent and environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1.

7.3 REPORTING REQUIREMENTS

A3

a. Planned Liquid and Gaseous Releases

The limits for radioactive materials contained in liquid and gaseous effluents are contained in Specifications 2.3, 3.3 and 3.4.

b. Environmental Samples Exceeding Limits of Table 6.1-2

When the limits of Table 6.1-2 are exceeded, refer to Specification 6.1.b for reporting requirements.

c. ~~Semiannual~~ Radioactive Effluent Release Report year

Prior to May 1 of each year in accordance with 10 CFR 50.36a

[5.6.3] ~~Routine~~ Radioactive Effluent Release Reports covering the operation of the unit during the previous ~~6 months of operation~~ shall be submitted ~~within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.~~ A3 L3

1. The Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit ~~using as guidance Regulatory Guide 1.21, Revision 1, June 1974, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", with data summarized on a quarterly basis following the format of Appendix B thereof.~~ A4

Plant. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.2

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2. The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year may include an annual summary of meteorological data collected over the previous year. If the meteorological data is not included, ENO shall retain it on file and provide it to the U.S. Nuclear Regulatory Commission upon request. This same report shall include an assessment of the radiation doses* due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year to the public. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. The assessment of radiation doses shall be performed in accordance with the ODCM.

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3. The Radioactive Effluent Release Reports shall include any change to the PCP or the ODCM made during the reporting period, as well as a listing of new locations for dose calculations and/or environmental monitoring indentified by the land use census pursuant to Specification 6.2.

4. The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses* to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) during the previous calendar year, to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. This assessment of radiation doses is performed in accordance with the ODCM.

5. The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (defined by 10 CFR 61) shipped offsite during the report period:

   (a) Container volume;

   (b) Total curie quantity (specify whether determined by measurement or estimate),

   (c) Principal radionuclides (specify whether determined by measurement or estimate),

   (d) Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

   (e) Type of container (e.g., LSA, Type A, Large Quantity), and

   (f) Solidification agent or absorbent (e.g., cement, Dow media, etc.)

6. The Radioactive Effluent Release Reports shall include a list and description of unplanned releases, to unrestricted areas of radioactive materials in gaseous and liquid effluents made during the reporting period.

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* The dose assessment sections of the Semiannual Radiological Effluent Release Report shall be submitted within 90 days after January 1 of each year as an addendum to the Semiannual Radiological Effluent Release Report.

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7. The Radioactive Effluent Release Report shall contain the cause for unavailability of any environmental sample required by Table 6.1.1 and shall identify the locations for obtaining replacement samples. This shall also include a revised figure(s) and table for the ODCM reflecting the new location(s). Refer to Specification 6.1.c.

8. The Radioactive Effluent Release Report shall contain new locations identified in the land use census in accordance with Specifications 6.2.b or 6.2.c.

9. The Radioactive Effluent Release Report shall contain the events leading to the condition which resulted in exceeding 10 curies for tanks specified in the Limiting Conditions for Operation, Section 2.5.a.

[5.6.2] d. Annual Radiological Environmental Operating Report

~~Routine~~ The Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted ~~prior to May 1~~ by May 15 of each year.

L2

The material provided shall be consistent with objectives outlined in ODCM and in 10 CFR 50, App I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period. ~~The report shall include a comparison with preoperational studies, operational controls (as appropriate), and environmental surveillance reports from the previous five years, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the Land Use Census required by Specification 6.2~~

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The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all measurements taken during the period ~~pursuant to Table 6.1.1~~ at the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion in the report, the report shall note and explain the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

~~The reports shall also include the following: A summary description of the Radiological Environmental Monitoring Program; at least two legible maps* covering all sampling locations and keyed to a table giving distances and directions from the centerline of one reactor; the results of participation in the Interlaboratory Comparison Program required by Specification 6.1 (or appropriate EPA cross-check program code), and discussion of all analyses in which the LLD's required by Table 6.1-3 were not routinely achievable.~~

~~* One map shall cover stations near the site boundary; a second shall include the more distant stations.~~

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