# Final Submittal
(Blue Paper)

# FINAL RO/SRO WRITTEN

# EXAMINATION REFERENCES

BRUNSWICK

**BRUNSWICK NOVEMBER 2008 EXAM**

**05000325/2008302 & 05000324/2008302**

List of Reference Material for RO NRC Exam

Steam Tables

0EOP-01-UG, Attachment 5, Figure 5, Core Spray NPSH Limit

0EOP-01-UG, Attachment 5, Figure 6, RHR NPSH Limit

0EOP-01-UG, Attachment 6, Figure 18, Unit 1 Reactor Water Level at LL-4

2OP-27, Figure 1, Estimated Capability Curves

2OI-03.2, Attachment 1, SJAE Off-Gas Radiation Monitors Channel Check Calculation

EHC Logic Diagram

Applicant
References for
RO/SRO exams

List of Reference Material for SRO NRC Exam

Steam Tables

0EOP-01-UG, Attachment 5, Figure 5, Core Spray NPSH Limit

0EOP-01-UG, Attachment 5, Figure 6, RHR NPSH Limit

0EOP-01-UG, Attachment 6, Figure 18, Unit 1 Reactor Water Level at LL-4

2OP-27, Figure 1, Estimated Capability Curves

2OI-03.2, Attachment 1, SJAE Off-Gas Radiation Monitors Channel Check Calculation

EHC Logic Diagram

0PEP-02.1, Attachment 1, Emergency Action Levels

0EOP-01-UG, Attachment 6, Figure 19, Unit 1 Reactor Water Level at LL-5

0EOP-01-UG, Attachment 5, Figure 3, Heat Capacity Temperature Limit

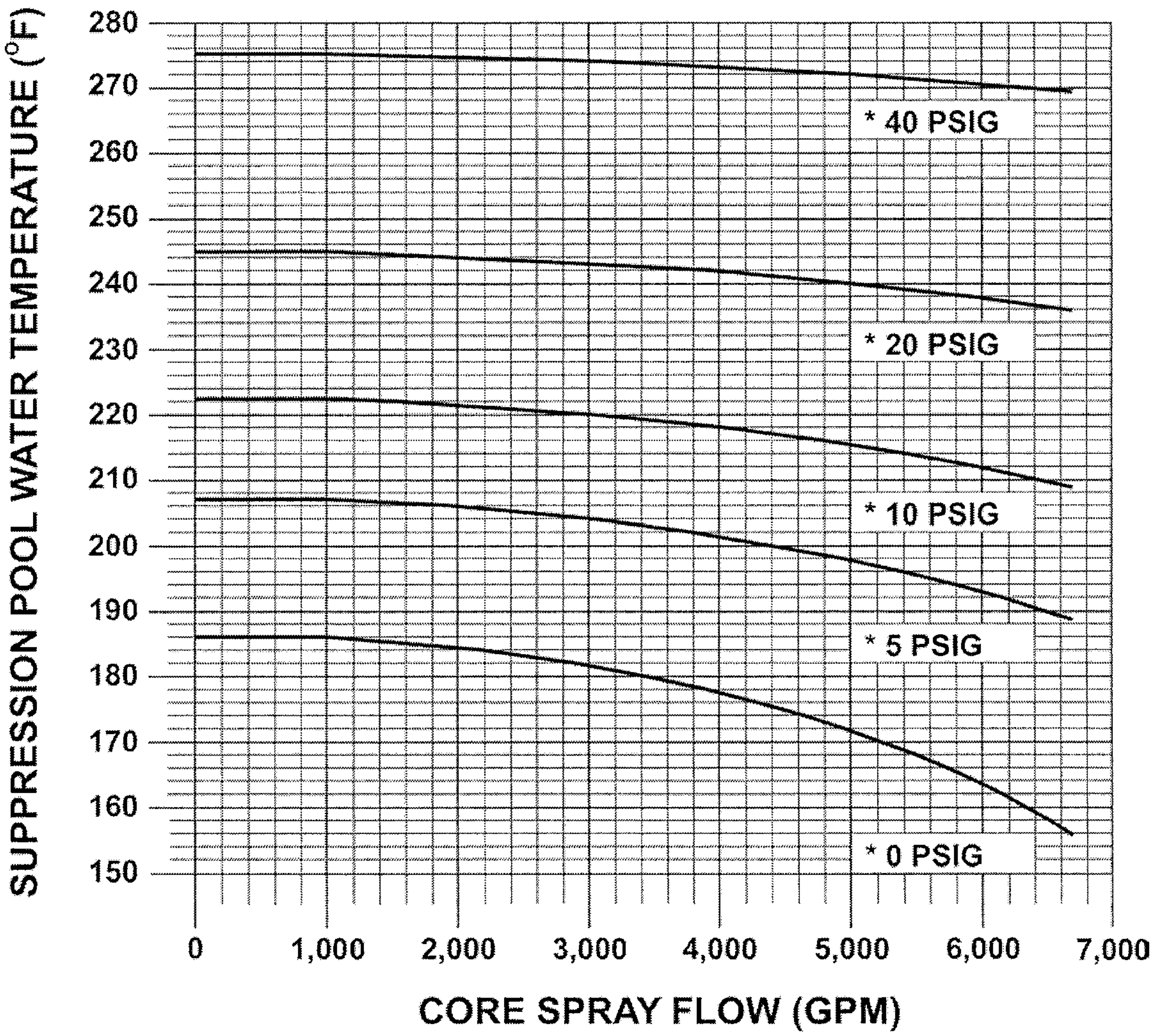
0MST-PCIS21Q, Acceptance Criteria

0OI-18, B21-LT-N024A-1, B-1; B21-LT-N025A-1, B-1

TS 3.3.6.1, Primary Containment Isolation Instrumentation

0OP-06.4, Recirculation and Sampling of Saltwater Release Tank #1

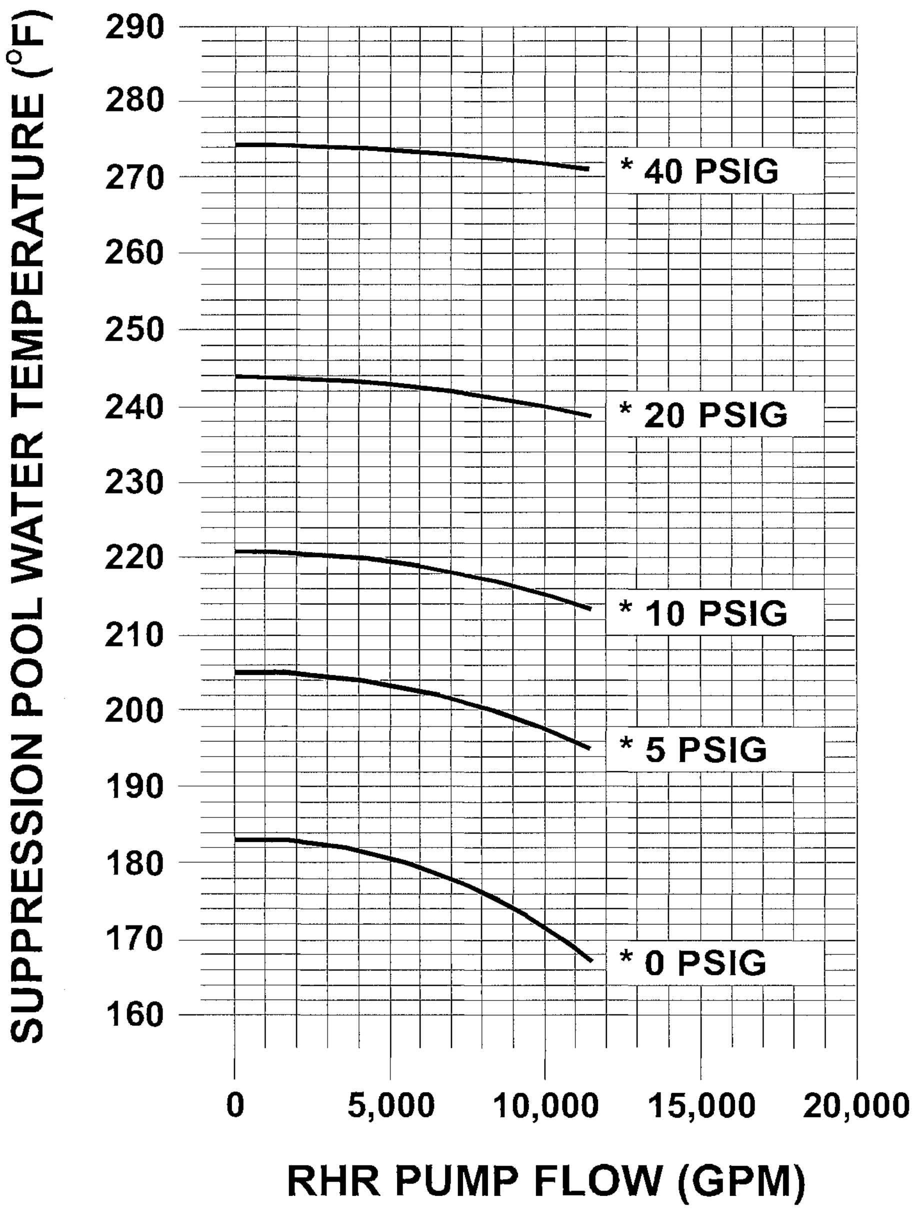
ATTACHMENT 5
Page 21 of 28
FIGURE 5
**Core Spray NPSH Limit**




SUBTRACT 0.5 PSIG FROM INDICATED SUPPRESSION CHAMBER PRESSURE FOR EACH FOOT OF WATER LEVEL BELOW A SUPPRESSION POOL WATER LEVEL OF -31 INCHES (-2.6 FEET).

*SUPPRESSION CHAMBER PRESSURE (CAC-PI-1257-2A OR CAC-PI-1257-2B)

FIGURE 6
**RHR NPSH Limit**

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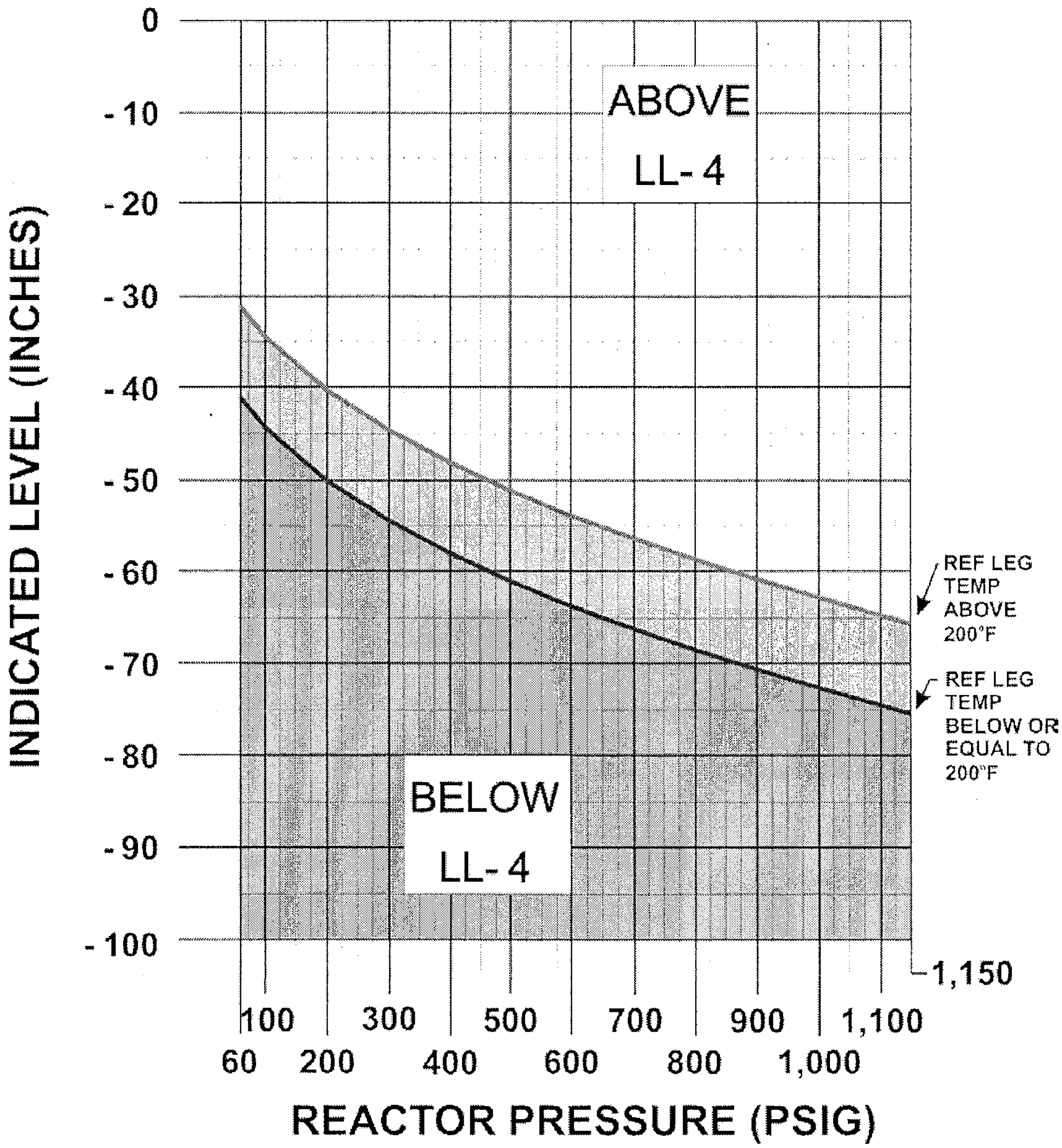
*SUPPRESSION CHAMBER PRESSURE (CAC-PI-1257-2A OR CAC-PI-1257-2B)

FIGURE 18

**Unit 1 Reactor Water Level at LL-4**
**(Minimum Steam Cooling Level)**




WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
LL-4 IS -30.0 INCHES.

FIGURE 1
Page 1 of 1
**Estimated Capability Curves**

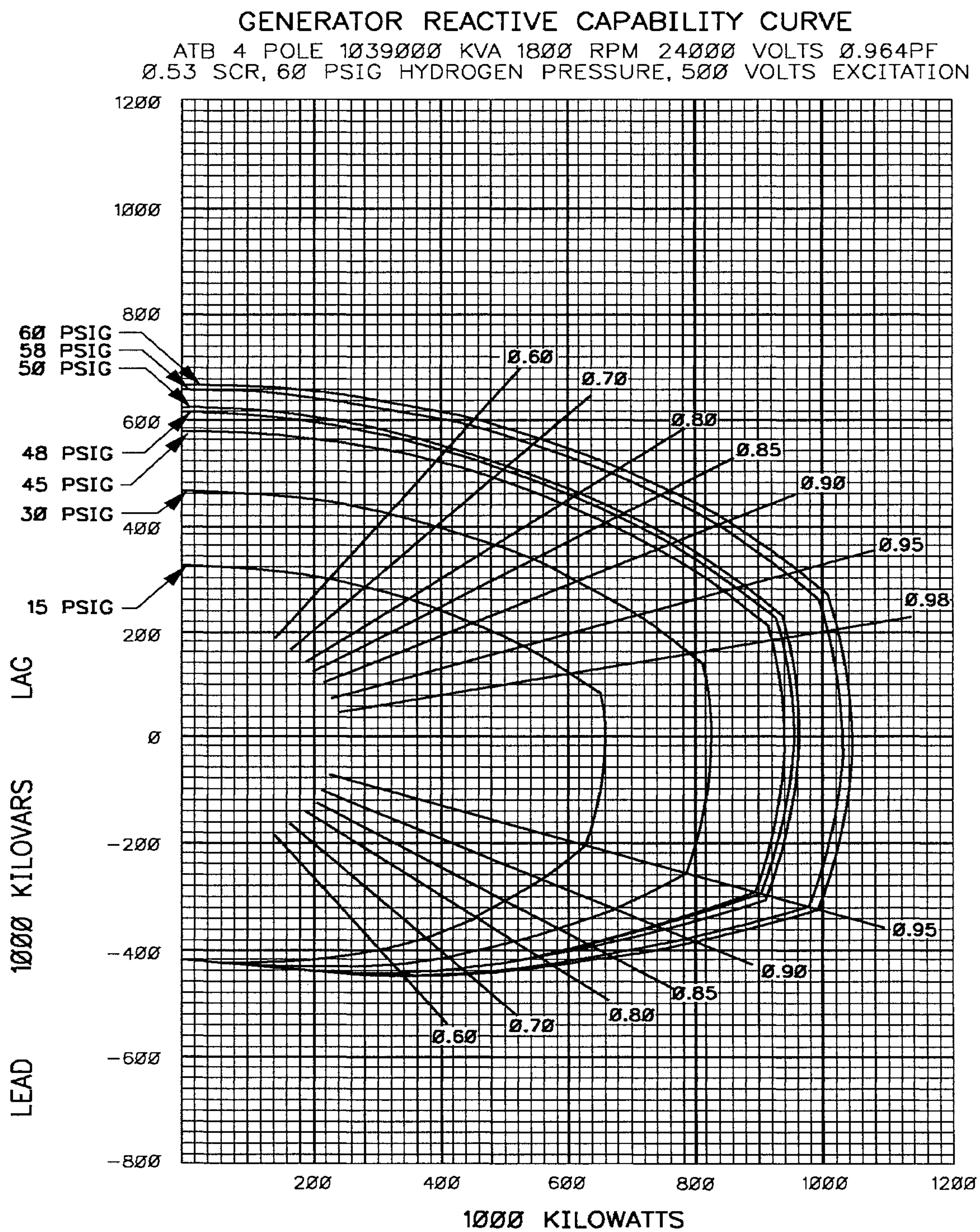


Table 1
SJAE OFF-GAS RADIATION MONITORS
CHANNEL CHECK CALCULATION

| **RECORD** SJAE Off-Gas Radiation Monitor readings from Item 55 & 56. | SAT | SUN | MON | TUE | WED | THUR | FRI |
|---|---|---|---|---|---|---|---|
| *D12-RM-K601A* | | | | | | | |
| *D12-RM-K601B* | | | | | | | |

**DIVIDE** the highest reading monitor by 2.

**CONFIRM** the lower reading monitor is greater than this value.

**IF** the lower reading monitor indication is > ½ the higher reading monitor then the channel check is satisfactory.

**IF** the lower reading monitor indication is ≤ ½ the higher reading monitor contact E&RC Health Physics to obtain a local reading with an appropriate survey instrument for alternate criteria.

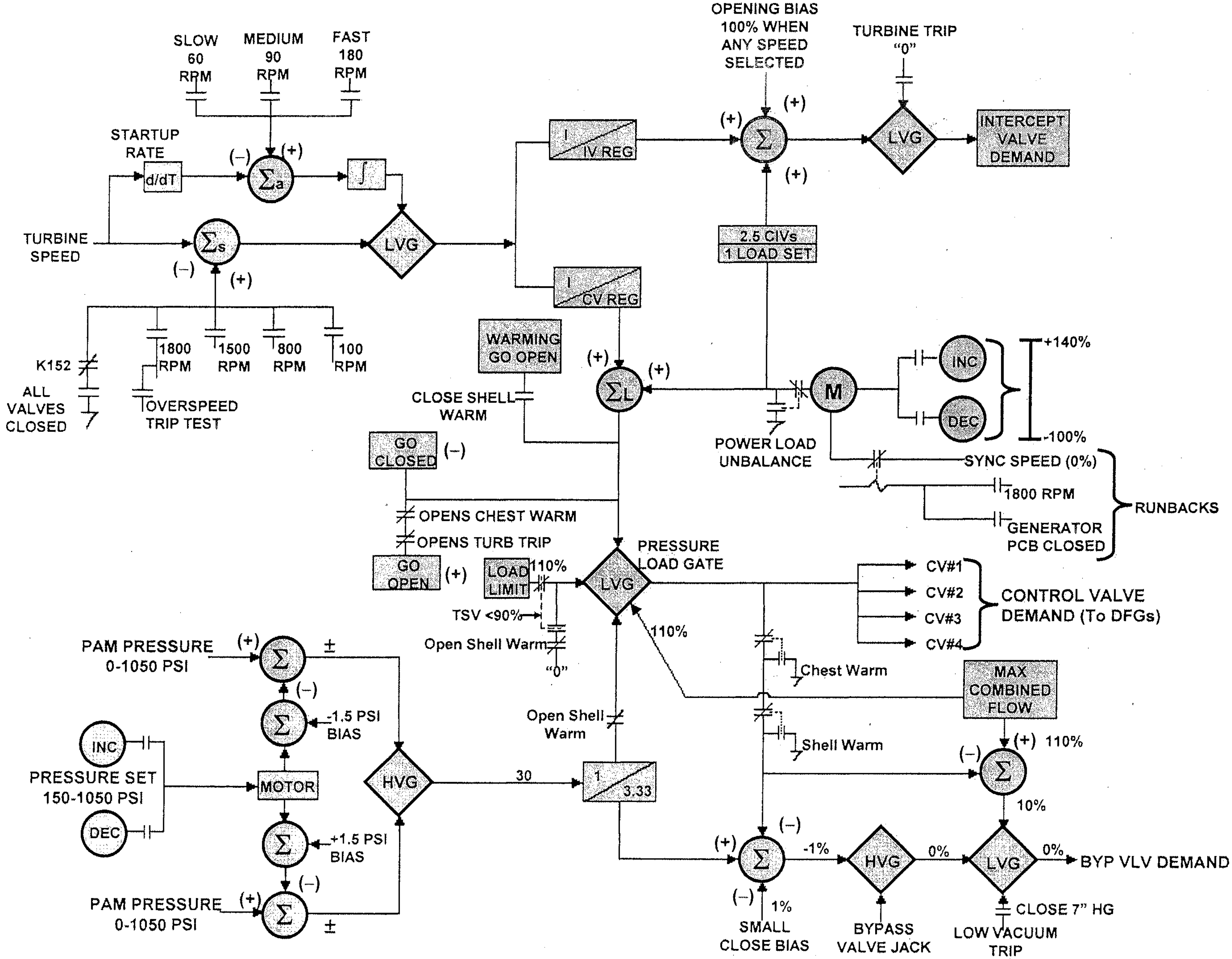
> **NOTE:** The survey instrument should be positioned near the operator aid [alternate channel check survey point] located on the sample chamber.

| **IF UTILIZED, RECORD** local survey instrument reading | SAT | SUN | MON | TUE | WED | THUR | FRI |
|---|---|---|---|---|---|---|---|
| | | | | | | | |

**MULTIPLY** the local instrument reading by .75.

**IF** the lower reading monitor indication is >.75 of the local survey instrument, the deviation is conservative and the channel check is satisfactory. Initiate a W/R to evaluate the deviation between the A and B monitors.

**IF** the lower reading monitor indication is ≤ .75 the local survey instrument, the deviation is non-conservative and the instrument should be declared inoperable.



SLOW 60 RPM
MEDIUM 90 RPM
FAST 180 RPM
STARTUP RATE
d/dT
TURBINE SPEED
LVG
IV REG
CV REG
OPENING BIAS 100% WHEN ANY SPEED SELECTED
TURBINE TRIP "0"
INTERCEPT VALVE DEMAND
2.5 CIVs
1 LOAD SET
K152
ALL VALVES CLOSED
1800 RPM
1500 RPM
800 RPM
100 RPM
OVERSPEED TRIP TEST
WARMING GO OPEN
CLOSE SHELL WARM
POWER LOAD UNBALANCE
INC
DEC
+140%
-100%
SYNC SPEED (0%)
1800 RPM
GENERATOR PCB CLOSED
RUNBACKS
GO CLOSED
OPENS CHEST WARM
OPENS TURB TRIP
GO OPEN
LOAD LIMIT
110%
PRESSURE LOAD GATE
TSV <90%
Open Shell Warm
"0"
CV#1
CV#2
CV#3
CV#4
CONTROL VALVE DEMAND (To DFGs)
PAM PRESSURE 0-1050 PSI
-1.5 PSI BIAS
PRESSURE SET 150-1050 PSI
MOTOR
+1.5 PSI BIAS
HVG
30
3.33
Chest Warm
Shell Warm
MAX COMBINED FLOW
10%
-1%
0%
BYP VLV DEMAND
SMALL CLOSE BIAS
1%
BYPASS VALVE JACK
CLOSE 7" HG
LOW VACUUM TRIP

**Emergency Action Levels**

| Section | Event Category | Page No. |
|---|---|---|
| 1.0 | Abnormal Primary Leak Rate | 11 |
| 2.0 | Steam Line Break or Safety/Relief Valve Failure | 13 |
| 3.0 | Abnormal Core Conditions and Core Damage | 14 |
| 4.0 | Abnormal Radiological Effluent or Radiation Levels | 16 |
| 5.0 | Loss of Shutdown Functions: Decay Heat and Reactivity | 18 |
| 6.0 | Electrical or Power Failures | 20 |
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## 1.0 Abnormal Primary Leak Rate

### 1.1 Notification of Unusual Event

01.01.01 Reactor Coolant System total leakage greater than 25 gpm averaged over the previous 24-hour period using the sum of drywell equipment drain integrator (G16-FQ-K603) and drywell floor drain integrator (G16-FQ-K601), and the leakage rate has not been reduced to less than 25 gpm within eight hours, or plant shutdown is not achieved within required time period.

**OR**

01.01.02 Unidentified Reactor Coolant System leakage greater than 5 gpm averaged over the previous 24-hour period using the drywell floor drain integrator (G16-FQ-K601), and the leakage rate has not been reduced to less than 5 gpm within eight hours, or plant shutdown is not achieved within required time period.

### 1.2 Alert

01.02.01 Small break LOCA with primary system leakage greater than 50 gpm. A LOCA is indicated by a significant loss of reactor inventory to the drywell resulting in increased drywell pressure, temperature, and/or sump pump usage indicated by:

- Low or falling Reactor Coolant System pressure with rising drywell pressure and temperature (C32-R608, CAC-PI-2685-1, CAC-TR-4426-1A, CAC-TR-4426-1B, CAC-TR-4426-2A and CAC-TR-4426-2B).

### 1.3 Site Area Emergency

01.03.01 Loss of coolant accident requiring the initiation of Low Pressure Coolant Injection, Core Spray, or the Automatic Depressurization System, **AND REQUIRED FOR ADEQUATE CORE COOLING**.

**Emergency Action Levels**

## 1.0 Abnormal Primary Leak Rate (Continued)

### 1.4 General Emergency

01.04.01 Loss of coolant accident requiring the initiation of Low Pressure Coolant Injection, Core Spray, or Automatic Depressurization System, **AND REQUIRED FOR ADEQUATE CORE COOLING**;

**AND**

Inability to provide makeup water to the Reactor Coolant System (i.e, failure of HPCI, Core Spray A and B, RHR Loops A and B, RCIC, condensate, and feedwater) as indicated by falling or low reactor vessel level with attempts to inject water not successful.

## 2.0 Steam Line Break or Safety/Relief Valve Failure

### 2.1 Notification of Unusual Event

02.01.01 Reactor Coolant System pressure ≥ 1250 psig.

**OR**

02.01.02 Inability to close an SRV with Reactor Coolant System pressure ≤ 900 psig.

### 2.2 Alert

02.02.01 Main Steam, HPCI or RCIC steam line break inside the primary containment without (full) line isolation valve closure.

### 2.3 Site Area Emergency

02.03.01 Main Steam, HPCI or RCIC steam line break outside primary containment and line isolation valve(s) fail to close indicated by valid area radiation and/or temperature alarms.

### 2.4 General Emergency

02.04.01 N/A

## 3.0 Abnormal Core Conditions and Core Damage

### 3.1 Notification of Unusual Event

03.01.01 Liquid

A. Reactor Coolant System (RCS) activity greater than 4.0 μCi/gm I-131 dose equivalent.

B. RCS activity greater than 0.2 μCi/gm I-131 dose equivalent but less than limit above for more than 48 hours.

03.01.02 Gaseous

A. Steam jet air ejector off-gas radiation monitor (D12-RM-K601A and B) reading of greater than $1.2 \times 10^4$ mR/hr.

B. Steam jet air ejector off-gas radiation monitor (D12-RM-K601A and B) increase of greater than $2.4 \times 10^3$ mR/hr in 30 minutes.

### 3.2 Alert

03.02.01 Liquid

Reactor coolant activity greater than 300 μCi/gm I-131 dose equivalent.

03.02.02 Gaseous

Steam jet air ejector off-gas radiation monitor (D12-RM-K601A and B) reading of greater than $1.2 \times 10^5$ mR/hr.

### 3.3 Site Area Emergency

03.03.01 Reactor Coolant System activity is greater than 4000 μCi/gm I-131 dose equivalent.

## 3.0 Abnormal Core Conditions and Core Damage (Continued)

### 3.4 General Emergency

03.04.01 Any two functional high range drywell radiation monitors (D22-RI-4195, 4196, 4197, and 4198) reading greater than 5000 R/hr.

## 4.0 Abnormal Radiological Effluent or Radiation Levels

### 4.1 Notification of Unusual Event

04.01.01 Liquid Release

Any unplanned release from the liquid waste system resulting in activity levels in the discharge canal greater than those in 10CFR20, Appendix B, Table II, Column 2.

04.01.02 Gaseous Release

Any gaseous release which exceeds the dose limit specified in ODCM 7.3.7 (i.e., exceeding the noble gas instantaneous dose rate limit as evaluated by 0E&RC-2020).

04.01.03 Any building evacuation based on confirmed radiological conditions (i.e., greater than 10 dac airborne [except precautionary evacuations]).

### 4.2 Alert

04.02.01 Liquid Release

Any liquid release resulting in activity concentration levels in the discharge canal that are greater than 10 times those given in 10CFR20, Appendix B, Table II, Column 2 (10 times the concentration listed in Unusual Event).

04.02.02 Gaseous Release

Any gaseous release which exceeds 10 times the dose rate limit specified in ODCM 7.3.7 (i.e., exceeding 10 times the noble gas instantaneous dose rate limit as evaluated by 0E&RC-2020).

04.02.03 In-Plant Leak or Spill

Unplanned, valid direct area radiation (gamma and/or neutron) reading(s) increase by a factor of 1000 over normal levels.

## 4.0 Abnormal Radiological Effluent or Radiation Levels (Continued)

### 4.3 Site Area Emergency

04.03.01 Projected dose exceeding 50 mRem Whole body (TEDE) **OR** exceeding 250 mRem Thyroid (CDE) at site boundary.

04.03.02 Measured dose rate exceeding 100 mR/hr at site boundary.

04.03.03 Measured I-131 dose equivalent concentration exceeds 3.9E-7 μCi/cc at the site boundary.

### 4.4 General Emergency

04.04.01 Offsite release resulting in a dose exceeding one (1) Rem Whole Body (TEDE) **OR** five (5) Rem Thyroid (CDE) at the Site Boundary as indicated by dose projection or field data.

04.04.02 Measured I-131 Dose Equivalent concentration exceeding 3.9E-6 μCi/cc at the site boundary.

## 5.0 Loss of Shutdown Functions: Decay Heat and Reactivity

### 5.1 Notification of Unusual Event

05.01.01 N/A

**Alert**

05.02.01 Complete loss of ability to maintain plant in cold shutdown:

1. Loss of essential service water loops, or Loss of RHR Loops A and B.

**AND**

2. Loss of Condenser Condensate System.

**AND**

3. Either:

   a. Coolant temperature exceeds 212°F,

   **OR**

   b. Uncontrolled temperature rise approaching 212°F.

05.02.02 Failure of the Reactor Protection System to initiate and complete a scram, indicated on Panel A-5, which brings the reactor to a subcritical condition as indicated by full core display panel P603 and neutron monitoring instruments (APRM and IRM).

## 5.0 Loss of Shutdown Functions: Decay Heat and Reactivity (Continued)

### 5.3 Site Area Emergency

05.03.01 Failure of the Reactor Protection System to initiate and complete a scram as indicated by Section 05.02.02 above.

**AND**

Failure of standby liquid control to bring the reactor to a subcritical condition.

05.03.02 Complete loss of reactor heat removal capability indicated by inability to maintain Suppression Pool below Heat Capacity Temperature Limit curve.

### 5.4 General Emergency

05.04.01 Site Area Emergency as indicated in Section 05.03.01 above lasting greater than 30 minutes.

**AND**

Loss of main condenser heat removal capability indicated by MSIVs shut or loss of vacuum on condenser vacuum indicator.

**AND EITHER**

1. Failure of all low pressure coolant injection trains indicated on panel P601.

**OR**

2. Failure of all service water trains necessary for decay heat removal indicated on panel P601 (RHR Service Water) and Panel XU2 (Nuclear and Conventional Service Water).

05.04.02 Containment pressure approaching Primary Containment Pressure Limit (PCPL), and containment venting will be required within the next six (6) hours.

**Emergency Action Levels**

## 6.0 Electrical or Power Failures

### 6.1 Notification of Unusual Event

06.01.01 Inability to power either 4 kV E Bus from off-site power.

**OR**

06.01.02 Loss of all on-site AC power capability indicated by failure of diesel generators to start or synchronize.

### 6.2 Alert

06.02.01 Loss of all vital DC power.

**OR**

06.02.02 Inability to power either 4 kV E Bus from off-site power.

**AND**

Loss of all on-site AC power capability indicated by failure of diesel generators to start or synchronize.

### 6.3 Site Area Emergency

06.03.01 Either Alert condition in Section 06.02.01 or 06.02.02 listed above **AND** lasting longer than 15 minutes.

### 6.4 General Emergency

06.04.01 N/A

**Emergency Action Levels**

## 7.0 Fire

### 7.1 Notification of Unusual Event

07.01.01 Fire located in or adjacent to the areas listed below **NOT** extinguished within 15 minutes of alarm verification or Control Room notification.

Areas:

Emergency Diesel Generator Building
Control Building
Central Alarm Station/Secondary Alarm Station
Reactor Building
Turbine Building
Unit Intake Structures
Service Water Building

### 7.2 Alert

07.02.01 Fire which could potentially affect vital safety-related equipment.

### 7.3 Site Area Emergency

07.03.01 Any fire that impairs the operability of any vital equipment which, in the opinion of the Site Emergency Coordinator, is essential to maintain the plant in a safe condition.

### 7.4 General Emergency

07.04.01 Any fire which in the opinion of the Site Emergency Coordinator could cause massive common damage to plant systems.

## 8.0 Control Room Evacuation

### 8.1 Notification of Unusual Event

08.01.01 N/A

### 8.2 Alert

08.02.01 Evacuation of Control Room anticipated or required with control of shutdown established from local stations.

### 8.3 Site Area Emergency

08.03.01 Evacuation of Control Room **AND** local control of shutdown is not established in 15 minutes.

### 8.4 General Emergency

08.04.01 N/A

## 9.0 Loss of Monitors or Alarms or Communication Capability

### 9.1 Notification of Unusual Event

09.01.01 Site communications capability impaired as determined by loss of all of the following:

1. Both site Private Branch Exchanges (PBX's)
2. All private phone lines (not routed through Plant Branch Exchange; Control Room, Security, Site Vice President Office)
3. Selective Signaling
4. Decision Line
5. State and Local emergency management radio system
6. Cellular phone system access
7. Satellite telephone

09.01.02 Unplanned loss of most or all annunciators on Panels P601, P603, XU-1, XU-2, XU-3, XU-51, and XU-80 for > 15 minutes with the affected unit in Mode 1, 2, or 3;

**AND**

Compensatory (non-alarming) indications are available.

### 9.2 Alert

09.02.01 Unplanned loss of most or all annunciators on Panels P601, P603, XU-1, XU-2, XU-3, XU-51, and XU-80 for > 15 minutes with the affected unit in Mode 1, 2, or 3;

**AND**

Either;

- Compensatory (non-alarming) indications are **NOT** available.

**OR**

- A plant transient is in progress.

## 9.0 Loss of Monitors or Alarms or Communication Capability (Continued)

### 9.3 Site Area Emergency

09.03.01 Unplanned loss of most or all annunciators on Panels P601, P603, XU-1, XU-2, XU-3, XU-51, and XU-80 with the affected unit in Operational Condition 1, 2, or 3;

**AND**

- Compensatory (non-alarming) indications are NOT available.

**AND**

- A plant transient is in progress.

**AND**

- Plant safety function indications (reactor power, reactor level, reactor pressure, containment parameters) are **NOT** available.

### 9.4 General Emergency

09.04.01 N/A

**Emergency Action Levels**

## 10.0 Fuel Handling Accident

### 10.1 Notification of Unusual Event

10.01.01 N/A

### 10.2 Alert

10.02.01 Fuel handling accident involving damage to new or spent fuel indicated by:

A. Observation/report **AND** alarm on:

1. Process Reactor Building ventilation RAD monitor D12-K609A, B or D12-RR-R605.

**OR**

2. Reactor Building roof ventilation monitor CAC-AIQ-1264-3.

**OR**

3. Refuel floor area monitor ARM channel 1-28 or 2-28.

### 10.3 Site Area Emergency

10.03.01 Major damage to spent fuel indicated by:

1. Observation of substantial damage to multiple fuel assemblies, or observation that water level has dropped below the top of the fuel.

**AND**

2. Indications or alarms listed in Attachment 1, Section 10.02.01.A above.

### 10.4 General Emergency

10.04.01 N/A

## 11.0 Security Threats

### 11.1 Notification of Unusual Event

11.01.01 Security threat or attempted entry (PA) or attempted sabotage.

11.01.02 A credible site specific security threat notification.

11.01.03 A validated notification from NRC providing information of an aircraft threat.

### 11.2 Alert

11.02.01 Ongoing security compromise (as determined by security).

11.02.02 A validated notification from NRC of an airliner attack threat less than 30 minutes away.

11.02.03 A notification from the site security force of an armed attack, explosive attack, airliner impact, or other HOSTILE ACTION within the OCA.

### 11.3 Site Area Emergency

11.03.01 Imminent loss of physical control of the plant.

11.03.02 A notification from the site security force that an armed attack, explosive attack, airliner impact, or other HOSTILE ACTION is occurring or has occurred within the protected area.

### 11.4 General Emergency

11.04.01 A HOSTILE FORCE has taken control of plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions.

**Emergency Action Levels**

## 12.0 Fission Product Barriers and Specific LCOs

### 12.1 Notification of Unusual Event

12.01.01 Loss of containment operability requiring shutdown by Technical Specifications and shutdown is not achieved within required time period.

12.01.02 Loss of engineered safety feature requiring shutdown by Technical Specifications and shutdown is not achieved within required time period.

### 12.2 Alert

12.02.01 Loss of either Fuel Clad or the Reactor Coolant Boundary.

### 12.3 Site Area Emergency

12.03.01 Loss of two-out-of-three fission product barriers.

### 12.4 General Emergency

12.04.01 Loss of any two-out-of-three fission product barriers with a potential loss of the third barrier.

## 13.0 Hazards to Plant Operations

### 13.1 Notification of Unusual Event

13.01.01 Non-hostile Aircraft crash within site boundaries with the potential to endanger safety-related equipment.

13.01.02 Unplanned non-hostile explosion within the site boundaries with the potential to endanger safety-related equipment.

13.01.03 Release of toxic or flammable gas that could endanger personnel.

13.01.04 Turbine rotating component failure causing rapid plant shutdown.

### 13.2 Alert

13.02.01 Non-hostile explosion, aircraft crash, or missile resulting in major damage to structures housing safety-related systems.

13.02.02 Unplanned and uncontrolled entry of toxic or flammable gases into vital areas in sufficient quantities to endanger personnel or the operability of safety-related equipment.

13.02.03 Turbine failure causing penetration of its outer casing.

### 13.3 Site Area Emergency

13.03.01 Non-hostile explosion, aircraft crash, or missile resulting in major damage to safe shutdown equipment with plant not in cold shutdown.

13.03.02 Uncontrolled entry of flammable or toxic gases into vital areas where lack of access constitutes a safety problem with plant not in cold shutdown.

### 13.4 General Emergency

13.04.01 Any major internal or external event substantially beyond design basis which could cause massive common damage to plant systems.

**Emergency Action Levels**

## 14.0 Natural Events

### 14.1 Notification of Unusual Event

14.01.01 Alarm on seismic monitor AND confirmation of earthquake.

14.01.02 Hurricane warning issued.

14.01.03 Tornado on site.

### 14.2 Alert

14.02.01 Earthquake registering greater than 0.08g on seismic instrumentation.

14.02.02 Any adverse weather conditions that causes a loss of function of two or more safety trains.

14.02.03 Tornado striking inside protected area resulting in major damage to structures housing safety-related systems.

14.02.04 Hurricane winds on site estimated:

1. $\geq$ 130 mph at 30 ft above ground level
2. $\geq$ 180 mph at 300 ft above ground level

### 14.3 Site Area Emergency

14.03.01 Earthquake registering greater than 0.16g on seismic instrumentation with plant not in cold shutdown.

14.03.02 Flood, low water, or hurricane surge greater than design levels or failure to protect vital equipment at lower levels and plant not in cold shutdown.

14.03.03 Plant not in cold shutdown with hurricane winds on site estimated:

1. $\geq$ 130 mph at 30 ft above ground level
2. $\geq$ 180 mph at 300 ft above ground level

## 14.0 Natural Events (Continued)

### 14.4 General Emergency

14.04.01 Any major natural event substantially beyond design basis which could cause massive common damage to plant systems.

**Emergency Action Levels**

## 15.0 Shift Superintendent/Site Emergency Coordinator Judgments

When any condition exists which indicates a necessity for an increased level of awareness or readiness above previous plant conditions, the Shift Superintendent/Site Emergency Coordinator should use his judgment to declare the appropriate emergency status for the plant.

### 15.1 Notification of Unusual Event

15.01.01 Plant conditions exist that warrant increased awareness by plant staff such as exceeding any Technical Specification safety limit.

### 15.2 Alert

15.02.01 Plant conditions exist that reflect a significant degradation in the safety of the reactor, but releases from this event would be small.

### 15.3 Site Area Emergency

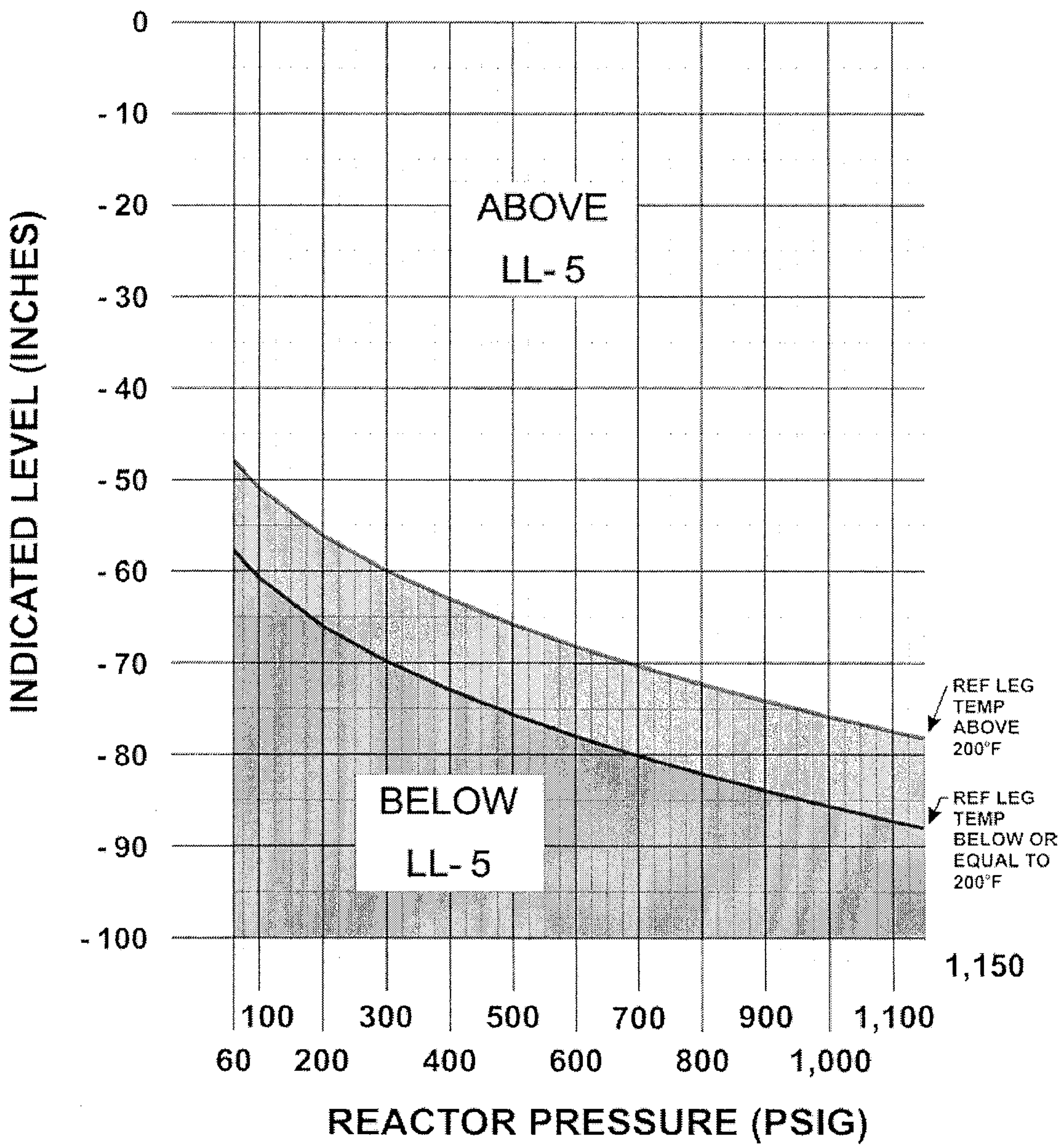
15.03.01 Plant conditions exist that involve major failures of equipment and that will lead to core damage. Unless corrective action is taken, significant radiation releases may occur.

### 15.4 General Emergency

15.04.01 Plant conditions exist that make a release of a large amount of radioactivity in a short time possible; any core melt situation.

FIGURE 19

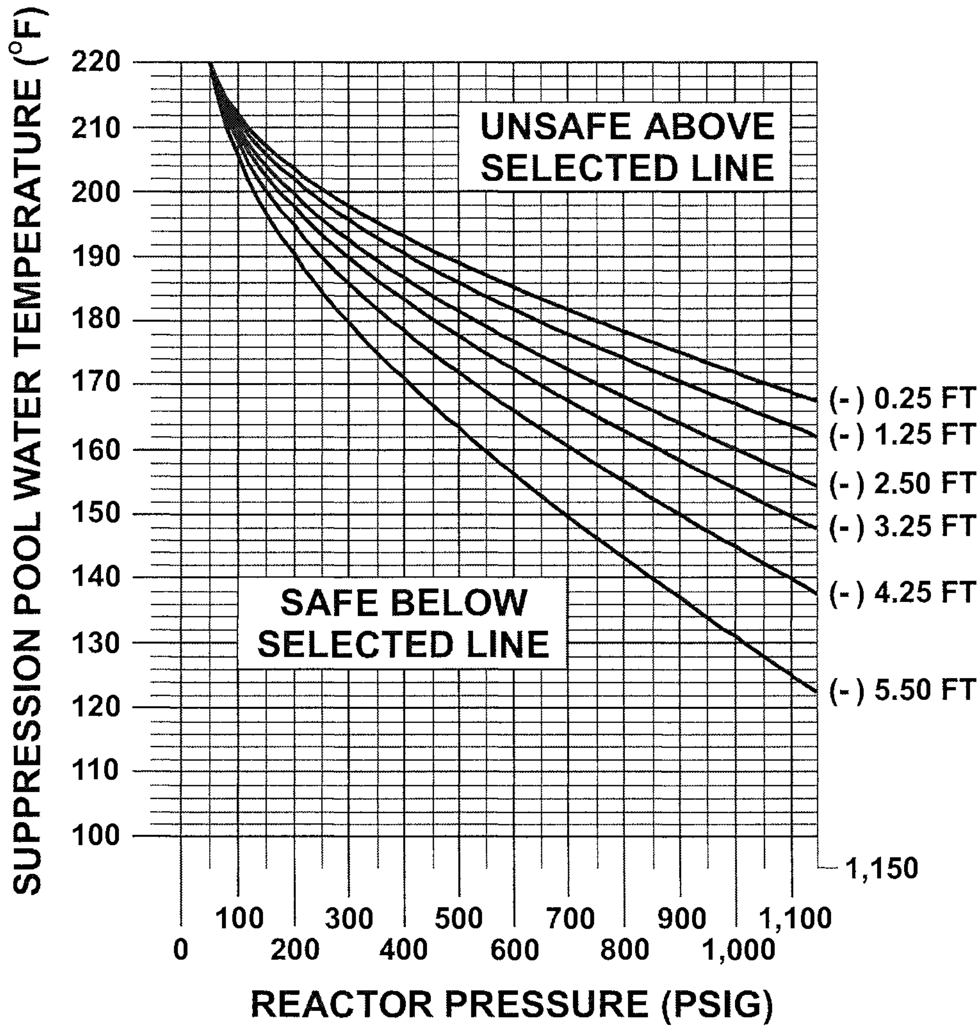
**Unit 1 Reactor Water Level at LL-5**
**(Minimum Zero Injection Level)**




WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
LL-5 IS -47.5 INCHES.

FIGURE 3

**Heat Capacity Temperature Limit**




SUPPRESSION POOL WATER TEMPERATURE IS DETERMINED BY:

- CAC-TR-4426-1A, POINT WTR AVG **OR**
- CAC-TR-4426-2A, POINT WTR AVG **OR**
- COMPUTER POINT G050 **OR**
- COMPUTER POINT G051 **OR**
- CAC-TY-4426-1 **OR**
- CAC-TY-4426-2

SELECT GRAPH LINE IMMEDIATELY BELOW SUPPRESSION POOL WATER LEVEL AS THE LIMIT.

## 4.0 PRECAUTIONS, LIMITATIONS AND NOTES (Continued)

4.5.3 Trip point adjustment potentiometer on front of Master Trip Unit should not be rotated beyond its end points; otherwise, damage will occur to trip unit.

4.5.4 False gross failure alarms may be caused by such actions as removing trip units from card file or pulling out the center knob of the calibration unit and should be reset as they occur.

4.5.5 If readout assembly has warmed up for ten minutes and is immediately moved from a card file to another card file, no further warm up is required.

## 5.0 SPECIAL TOOLS AND EQUIPMENT

5.1 510 DU readout assembly or equivalent

5.2 510 DU calibration unit

5.3 510 DU extender board

5.4 Jumper

## 6.0 ACCEPTANCE CRITERIA

6.1 Criteria stated in Step 6.1 must be met to satisfy Technical Specifications requirements:

| |
|---|
| **NOTE:** Technical Specifications allowable value for this LL2 trip is greater than or equal to 101 inches of water which relates to 11.70 mAdc. |

6.1.1 When a calibration current of 12.00 (11.96 to 12.04) mAdc decreasing is applied to Master Trip Units B21-LTM-N024A-1-1 or B21-LTM-N025A-1-1, Contacts 5-6 and 7-8 of the associated relay (Relay A71B-K1A or A71B-K1C) in Panel H12-P609 open.

## 6.0 ACCEPTANCE CRITERIA (Continued)

| | |
|---|---|
| **NOTE:** | Technical Specifications allowable value for this LL3 trip is greater than or equal to 13 inches of water which relates to 4.99 mAdc |

6.1.2 When a calibration current of 7.43 (7.39 to 7.47) mAdc decreasing is applied to Slave Trip Units B21-LTS-N024A-1-2 or B21-LTS-N025A-1-2,.Contacts 1-2 and 3-4 of the associated Relay (Relay A71B-K1E or A71B-K1G) in Panel H12-P609 open.

6.2 Technical Specifications requirements may be satisfied without meeting criteria stated in Step 6.2.

6.2.1 When a calibration current of 12.00 (11.96 to 12.04) mAdc decreasing is applied to Master Trip Units B21-LTM-N024A-1-1 or B21-LTM-N025A-1-1, Annunciator *REACTOR VESS LO-LO WATER LEVEL SYS A* (A-06 1-6) is in alarm.

6.2.2 The reset differential value of Master Trip Units B21-LTM-N024A-1-1 and -N025A-1-1 and Slave Trip Units B21-LTS-N024A-1-2 and -N025A-1-2 is within calibration specifications for pressures applied during test.

INSTRUMENT NUMBER: B21-LT-N024A-1, B-1; B21-LT-N025A-1, B-1

INSTRUMENT NAME: Reactor Vessel Water Level Low

TS REFERENCE: 3.3.6.1, 3.3.6.2; and 3.3.7.1, TRM Tables 3.3.6.1-1.1a, -1.5g; 3.3.6.2-1.1

TRIP CHANNEL: A1-N024A-1 A2-N025A-1
B1-N024B-1 B2-N025B-1

TRIP SYSTEM: A-A1 and A2
B-B1 and B2

TRIP LOGIC: A1 or A2 and B1 or B2 = Closes all MSIVs
A1 and B1 = Closes B32-F019, B21-F016, or G31-F001, RB Vent, starts SBGT and CREV
A2 and B2 = Closes B32-F020, B21-F019, or G31-F004, RB Vent, starts SBGT and CREV

Place channel in tripped condition by: Pull fuse

| CHANNEL | INSTRUMENT NUMBER | TRIP UNIT | ACTION | PANEL | FUNCTION |
|---|---|---|---|---|---|
| A1 | B21-LT-N024A-1 | B21-LTM-N024A-1-1<br>B21-LTS-N024A-1-2 | A71B-F1A<br>A71B-F61A | H12-P609<br>H12-P609 | Closes G31-F001, RB Vent, starts SBGT and CREV<br>Closes MSIVs, B32-F019, B21-F016 |
| B1 | B21-LT-N024B-1 | B21-LTM-N024B-1-1<br>B21-LTS-N024B-1-2 | A71B-F1B<br>A71B-F61B | H12-P611<br>H12-P611 | Same as A1 master<br>Same as A1 slave |
| A2 | B21-LT-N025A-1 | B21-LTM-N025A-1-1<br>B21-LTS-N025A-1-2 | A71B-F1C<br>A71B-F61C | H12-P609<br>H12-P609 | Closes G31-F004, RB Vent, starts SBGT and CREV<br>Closes MSIVs, B32-F020, B21-F019 |
| B2 | B21-LT-N025B-1 | B21-LTM-N025B-1-1<br>B21-LTS-N025B-1-2 | A71B-F1D<br>A71B-F61D | H12-P611<br>H12-P611 | Same as A2 master<br>Same as A2 slave |

COMMENTS: If both channels in a trip system are inop, both channels must be tripped to assure all required functions occur.

REFERENCE DRAWINGS: 1-FP-55109, 2-FP-50056

## 3.3 INSTRUMENTATION

### 3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

------------------------------------------------------NOTE------------------------------------------------------
Separate Condition entry is allowed for each channel.
-------------------------------------------------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more required channels inoperable. | A.1 Place channel in trip. | 12 hours for Functions 2.a, 2.b, and 6.b<br><br><u>AND</u><br><br>24 hours for Functions other than Functions 2.a, 2.b, and 6.b |
| B. One or more Functions with isolation capability not maintained. | B.1 Restore isolation capability. | 1 hour |
| C. Required Action and associated Completion Time of Condition A or B not met. | C.1 Enter the Condition referenced in Table 3.3.6.1-1 for the channel. | Immediately |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| D. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. | D.1 | Isolate associated main steam line (MSL). | 12 hours |
| | OR | | |
| | D.2.1 | Be in MODE 3. | 12 hours |
| | AND | | |
| | D.2.2 | Be in MODE 4. | 36 hours |
| E. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. | E.1 | Be in MODE 2. | 6 hours |
| F. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. | F.1 | Isolate the affected penetration flow path(s). | 1 hour |
| G. Required Action and associated Completion Time for Condition F not met.<br><br>OR<br><br>As required by Required Action C.1 and referenced in Table 3.3.6.1-1. | G.1 | Be in MODE 3. | 12 hours |
| | AND | | |
| | G.2 | Be in MODE 4. | 36 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| H. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. | H.1 | Declare associated standby liquid control subsystem (SLC) inoperable. | 1 hour |
| | OR | | |
| | H.2 | Isolate the Reactor Water Cleanup (RWCU) System. | 1 hour |
| I. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. | I.1 | Initiate action to restore channel to OPERABLE status. | Immediately |
| | OR | | |
| | I.2 | Initiate action to isolate the Residual Heat Removal (RHR) Shutdown Cooling (SDC) System. | Immediately |

SURVEILLANCE REQUIREMENTS

------------------------------------------------------ NOTES ------------------------------------------------------

1. Refer to Table 3.3.6.1-1 to determine which SRs apply for each Primary Containment Isolation Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed as follows: (a) for up to 2 hours for Functions 2.c, 2.d, 3.a, 3.b, 3.e, 3.f, 3.g, 3.h, 4.a, 4.b, 4.e, 4.f, 4.g, 4.h, 4.i, 4.k, 5.a, 5.b, 5.e, 5.f, and 6.a; and (b) for up to 6 hours for all other Functions provided the associated Function maintains isolation capability.

-------------------------------------------------------------------------------------------------------------------

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.6.1.1 | Perform CHANNEL CHECK. | 24 hours |
| SR 3.3.6.1.2 | Perform CHANNEL FUNCTIONAL TEST. | 92 days |
| SR 3.3.6.1.3 | Calibrate the trip unit. | 92 days |
| SR 3.3.6.1.4 | Perform CHANNEL CALIBRATION. | 92 days |
| SR 3.3.6.1.5 | Perform CHANNEL FUNCTIONAL TEST. | 184 days |
| SR 3.3.6.1.6 | Perform CHANNEL CALIBRATION. | 24 months |
| SR 3.3.6.1.7 | Perform LOGIC SYSTEM FUNCTIONAL TEST. | 24 months |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| | SURVEILLANCE | FREQUENCY |
|---|---|---|
| SR 3.3.6.1.8 | -----NOTES----- 1. Radiation detectors are excluded. 2. The sensor response time for Functions 1.a, 1.c, and 1.f may be assumed to be the design sensor response time. ----- Verify the ISOLATION INSTRUMENTATION RESPONSE TIME is within limits. | 24 months on a STAGGERED TEST BASIS |
| SR 3.3.6.1.9 | Perform CHANNEL FUNCTIONAL TEST. | 24 months |

Table 3.3.6.1-1 (page 1 of 5)
Primary Containment Isolation Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS PER TRIP SYSTEM | CONDITIONS REFERENCED FROM REQUIRED ACTION C.1 | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 1. Main Steam Line Isolation | | | | | |
| a. Reactor Vessel Water Level—Low Level 3 | 1,2,3 | 2 | D | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7<br>SR 3.3.6.1.8 | |
| b. Main Steam Line Pressure—Low | 1 | 2 | E | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| c. Main Steam Line Flow—High | 1,2,3 | 2 per MSL | D | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7<br>SR 3.3.6.1.8 | |
| d. Condenser Vacuum—Low | 1, 2[(a)], 3[(a)] | 2 | D | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| e. Main Steam Isolation Valve Pit Temperature—High | 1,2,3 | 2 | D | SR 3.3.6.1.2<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| f. Main Steam Line Flow—High (Not in Run) | 2,3 | 2 | D | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7<br>SR 3.3.6.1.8 | |
| 2. Primary Containment Isolation | | | | | |
| a. Reactor Vessel Water Level—Low Level 1 | 1,2,3 | 2 | G | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| b. Drywell Pressure—High | 1,2,3 | 2 | G | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |

(continued)

(a) With any turbine stop valve not closed.

Table 3.3.6.1-1 (page 2 of 5)
Primary Containment Isolation Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS PER TRIP SYSTEM | CONDITIONS REFERENCED FROM REQUIRED ACTION C.1 | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 2. Primary Containment Isolation (continued) | | | | | |
| c. Main Stack Radiation—High | 1,2,3 | 1 | F | SR 3.3.6.1.2<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7<br>SR 3.3.6.1.8 | |
| d. Reactor Building Exhaust Radiation—High | 1,2,3 | 1 | G | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| 3. High Pressure Coolant Injection (HPCI) System Isolation | | | | | |
| a. HPCI Steam Line Flow—High | 1,2,3 | 1 | F | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| b. HPCI Steam Line Flow—High Time Delay Relay | 1,2,3 | 1 | F | SR 3.3.6.1.6<br>SR 3.3.6.1.7<br>SR 3.3.6.1.9 | |
| c. HPCI Steam Supply Line Pressure—Low | 1,2,3 | 2 | F | SR 3.3.6.1.2<br>SR 3.3.6.1.4<br>SR 3.3.6.1.7 | |
| d. HPCI Turbine Exhaust Diaphragm Pressure—High | 1,2,3 | 2 | F | SR 3.3.6.1.2<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| e. Drywell Pressure—High | 1,2,3 | 1 | F | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| f. HPCI Steam Line Area Temperature—High | 1,2,3 | 1 | F | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| g. HPCI Steam Line Tunnel Ambient Temperature—High | 1,2,3 | 1 | F | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| h. HPCI Steam Line Tunnel Differential Temperature—High | 1,2,3 | 1 | F | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |

(continued)

(b) Allowable Value established in accordance with the methodology in the Offsite Dose Calculation Manual.

Table 3.3.6.1-1 (page 3 of 5)
Primary Containment Isolation Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS PER TRIP SYSTEM | CONDITIONS REFERENCED FROM REQUIRED ACTION C.1 | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 3. HPCI System Isolation (continued) | | | | | |
| i. HPCI Equipment Area Temperature—High | 1,2,3 | 2 | F | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| 4. Reactor Core Isolation Cooling (RCIC) System Isolation | | | | | |
| a. RCIC Steam Line Flow—High | 1,2,3 | 1 | F | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| b. RCIC Steam Line Flow—High Time Delay Relay | 1,2,3 | 1 | F | SR 3.3.6.1.6<br>SR 3.3.6.1.7<br>SR 3.3.6.1.9 | |
| c. RCIC Steam Supply Line Pressure—Low | 1,2,3 | 2 | F | SR 3.3.6.1.2<br>SR 3.3.6.1.4<br>SR 3.3.6.1.7 | |
| d. RCIC Turbine Exhaust Diaphragm Pressure—High | 1,2,3 | 2 | F | SR 3.3.6.1.2<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| e. Drywell Pressure—High | 1,2,3 | 1 | F | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| f. RCIC Steam Line Area Temperature—High | 1,2,3 | 1 | F | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| g. RCIC Steam Line Tunnel Ambient Temperature—High | 1,2,3 | 1 | F | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| h. RCIC Steam Line Tunnel and Area Temperature—High Time Delay | 1,2,3 | 1 | F | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| i. RCIC Steam Line Tunnel Differential Temperature—High | 1,2,3 | 1 | F | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |

(continued)

Table 3.3.6.1-1 (page 4 of 5)
Primary Containment Isolation Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS PER TRIP SYSTEM | CONDITIONS REFERENCED FROM REQUIRED ACTION C.1 | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 4. RCIC System Isolation (continued) | | | | | |
| j. RCIC Equipment Area Temperature—High | 1,2,3 | 2 | F | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| k. RCIC Equipment Area Differential Temperature—High | 1,2,3 | 1 | F | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| 5. Reactor Water Cleanup (RWCU) System Isolation | | | | | |
| a. Differential Flow—High | 1,2,3 | 1 | F | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| b. Differential Flow—High Time Delay | 1,2,3 | 1 | F | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| c. Area Temperature—High | 1,2,3 | 3<br>1 per room | F | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| d. Area Ventilation Differential Temperature—High | 1,2,3 | 2 | F | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| e. Piping Outside RWCU Rooms Area Temperature—High | 1,2,3 | 1 | F | SR 3.3.6.1.5<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |
| f. SLC System Initiation | 1,2 | 1[(c)] | H | SR 3.3.6.1.7 | |
| g. Reactor Vessel Water Level—Low Level 2 | 1,2,3 | 2 | F | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |

(continued)

(c) SLC System Initiation only inputs into one trip system.

Table 3.3.6.1-1 (page 5 of 5)
Primary Containment Isolation Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS PER TRIP SYSTEM | CONDITIONS REFERENCED FROM REQUIRED ACTION C.1 | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 6. RHR Shutdown Cooling System Isolation | | | | | |
| a. Reactor Steam Dome Pressure—High | 1,2,3 | 1 | F | SR 3.3.6.1.2<br>SR 3.3.6.1.4<br>SR 3.3.6.1.7 | |
| b. Reactor Vessel Water Level— Low Level 1 | 3,4,5 | 2(d) | I | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.3<br>SR 3.3.6.1.6<br>SR 3.3.6.1.7 | |

(d) In MODES 4 and 5, provided RHR Shutdown Cooling System integrity maintained, only one channel per trip system with an isolation signal available to one RHR shutdown cooling pump suction isolation valve is required.

## 5.7 Recirculation and Sampling of Saltwater Release Tank #1

### 5.7.1 Initial Conditions

1. All applicable prerequisites as listed in Section 4.0 are met. ☐

### 5.7.2 Procedural Steps

Saltwater Release Tank ______________

1. **PERFORM** the following to ensure the Saltwater Release Tank can **NOT** receive inputs from the following sources during recirculation and sampling:

   a. **ENSURE** the Unit 1 Breezeway North end West side mop water drain tube is locked. ☐

   b. **DISCONNECT** electrical supply to pipe tunnel dike portable bilge pump. ☐

   c. **REMOVE** Turbine Building portable sump pump discharge hose from Saltwater Release Tank. ☐

   d. **DIVERT** routing of air wash water to one of the following in accordance with 1(2)OP-37.3: ☐

      - Unit 1 or Unit 2 TB Floor Drain Sumps

      **OR**

      - Unit 1 or Unit 2 TB Equipment Drain Sumps

2. **IF** the tank is being recirculated through the Saltwater Release System Filters in accordance with Section 5.8, **THEN GO TO** Step 5.7.2.6. ☐

3. **OPEN** *SALTWATER RELEASE SYSTEM RECIRCULATION VALVE, 1-SWR-V11.* ☐

### 5.7.2 Procedural Steps

| | |
|---|---|
| **NOTE:** | Saltwater Release Tank level may drop 0-4% when placed in recirculation. |

4. **WHEN** placing Saltwater Release Tank in recirculation, **THEN RECORD** tank level prior to starting pump. ☐

   ____________________
   Tank Level.

5. **START** Saltwater Release System Pump #1. ☐

6. **SAMPLE** Saltwater Release Tank #1 by completing the following:

   a. **ALLOW** Saltwater Release Tank to recirculate for 4 minutes for each percent of indicated tank volume. ☐

   b. **OPEN** *SALTWATER RELEASE SYSTEM SAMPLE STATION VALVE, 1-SWR-V17.* ☐

   c. **ALLOW** sample to run for at least 5 minutes to ensure a representative sample is obtained. ☐

   d. **OBTAIN** sample in accordance with E&RC-2009. ☐

   e. **CLOSE** *SALTWATER RELEASE SYSTEM SAMPLE VALVE, 1-SWR-V17.* ☐

7. **IF** tank activity is greater than ODCM limits, **OR** additional filtration is desired, **AND** the tank is being recirculated through the Saltwater Release System Filters in accordance with Section 5.8, **THEN CONTINUE** recirculation. ☐

8. **IF** tank activity is greater than allowed ODCM limits, **OR** additional filtration is desired, **AND** the tank is **NOT** being recirculated in accordance with Section 5.8, **THEN**:

   a. **PERFORM** Section 7.6 **AND**, ☐

   b. **PERFORM** Section 5.8 to conduct cleanup. ☐

5.7.2 **Procedural Steps**

9. **IF** tank activity is within ODCM limits **AND** it is desired to release Saltwater Release Tank #1, **THEN PERFORM** Section 5.9. ☐

10. **IF** desired, **THEN SHUT DOWN** recirculation in accordance with:

    a. Section 7.6, **OR** ☐

    b. Section 7.5. ☐