50-373
50-374

PROC.	NO.	TITLE	REV.	REV DATE	DISKETT
LGA	000-01	LOSS OF COOLANT (FAST LEAK)	~~00~~ 01	~~3/79~~ 10/79SL	01
LGA	000-02	LOSS OF COOLANT (SLOW LEAK)	~~00~~ 01	~~3/79~~ 12/79SL	01
LGA	000-03	MAJOR STEAM LEAKS (OUTSIDE THE DRYWELL)	00	3/79	01
LGA	000-04	HIGH REACTOR WATER LEVEL	00	3/79	01
LGA	000-05	LOW REACTOR WATER LEVEL	~~00~~ 01	~~4/79~~ 1180	01
LGA	000-06	REACTOR HIGH PRESSURE	00	3/79	01
LGA	000-07	INADVERTANT REACTIVITY ADDITION	~~00~~ 01	~~3/79~~ 12/79SL	01
LGA	000-08	LOSS OF RECIRCULATION FLOW ~~ONE~~ single PUMP	~~00~~ 01	~~3/79~~ 12/79SL	01
LGA	000-09	LOSS OF RECIRCULATION FLOW-BOTH LOOPS	00	3/79	01
LGA	000-10	LOSS OF TURBINE GENERATOR LOAD GREATER THAN 25%	~~00~~ 01	~~3/79~~ 12/79SL	01
LGA	000-11	LOSS OF TURBINE GENERATOR LOAD LESS THAN 25%	00	3/79	01
LGA	000-12	LOSS OF AUXILIARY ELECTRICAL POWER	00	3/79	01
LGA	000-13	AREA HIGH RADIATION	00	0/00	01
LGA	000-14	HIGH AIRBORNE ACTIVITY	00	4/79	01
LGA	000-15	RADIOACTIVE SURFACE CONTAMINATION	00	3/79	01
LGA	000-16	FUEL ELEMENT FAILURE	00	3/79	01
LGA	000-17	BOMB THREAT RESPONSE	~~00~~ 01	~~3/79~~ 6/79SL	01
LGA	000-18	Transient With Failure To Scram	00	10/79SL	

LOSS OF COOLANT (FAST LEAK)

A. SYMPTOMS

1. Control Room Panel 1H13-P601 (2H13-P601) Alarms:

   a. RHR DRYWELL CH B/D PRESSURE HI.

   b. RHR REACTOR CH B/D LEVEL LO.

   c. RHR SYTEM 1 ACTUATED.

   d. RHR SYSTEM 2 ACTUATED.

   e. HPCS SYSTEM ACTUATES.

   f. LPCS SYSTEM ACTUATED.

   g. LEAK DETECTION PRI CONT TEMP HI.

   h. DRYWELL PRESSURE HI.

   i. REACTOR WATER LEVEL 1 LO.

   j. REACTOR WATER LEVEL 2 LO.

   k. ADS LOGIC A ACTUATED/B ACTUATED/C ACTUATED/D ACTUATED.

   l. REACTOR VESSEL LOW WATER LEVEL 3 CONFIRMED.

2. Control Room Panel 1H13-P603 (2H13-P603) Alarms:

   a. DIV 1 RX VESSEL WATER LEVEL 2 LO.

   b. DIV 2 RX VESSEL WATER LEVEL 2 LO.

   c. CH A PRIM CONT PRESS HI.

   d. CH B PRI CONT PRESS HI.

   e. CH A RX VESSEL LO LEVEL 3.

   f. CH B RX VESSEL LO LEVEL 3.

3. Control Room Panel 1PM13J (2PM13J) Alarms:

a. DRYWELL AIR HUMIDITY HI.

b. DRYWELL EQUIPMENT DRAIN SUMP LEVEL HI-HI.

c. DRYWELL EQUIPMENT DRAIN SUMP TEMPERATURE HI.

d. FLOW TO DRYWELL FLOOR DRAIN SUMP LEVEL HI.

e. DRYWELL FLOOR DRAIN SUMP LEVEL HI-HI.

f. DRYWELL SUMP FILL-UP RATE.

g. DRYWELL SUPPRESSION POOL CHAMBER TEMPERATURE HI.

4. Control Room Panel 1PM065 (2PM065) Containment Mimic Indicates Isolation.

B. AUTOMATIC ACTIONS

1. Level 3, Reactor Vessel Water Level 12.5".

a. Reactor SCRAM.

b. TRIP Reactor Recirculation Pumps to 15 Hz.

c. ADS Permissive.

d. ISOLATE Reactor Vessel and Containment Groups 4, 5, and 7.

e. Automatic Programmed Vessel Level DOWNSHIFT to 18".

2. Level 2, Reactor Vessel Water Level - 50".

a. AUTOSTART RCIC.

b. AUTOSTART HPCS and HPCS Diesel Generator 1B(2B).

c. TRIP Reactor Recirculation Pumps to OFF.

d. ISOLATE Reactor Vessel and Containment Groups 1, 2, 3, and 5.

3. Level 1, Reactor Vessel Water Level - 129".

b. AUTOSTART RHR LPCI Mode.

b. AUTOSTART LPCS.

c. AUTOSTART Diesel Generators 0 and 1A(2A).

d. ADS Permissive.

4. 1.69# Containment High Pressure.

a. Reactor SCRAM.

b. ISOLATE Reactor Vessel and Containment Groups 2, 4, 6, and 7.

c. ISOLATE Reactor Vessel and Containment Group 9 if 57# Reactor Pressure also present.

C. IMMEDIATE OPERATOR ACTIONS

1. VERIFY if reached:

a. Level 3 Actions.

b. Level 2 Actions.

c. Level 1 Actions.

d. 1.69# Containment Actions.

2. MAINTAIN Feedwater Flow if possible.

3. MONITOR:

a. Vessel Level and Pressure.

b. Containment Pressure, Temperature, Hydrogen Concentration, and Radiation.

4. As necessary,

a. NOTIFY and EVACUATE Personnel.

b. RESTRICT Access.

c. NOTIFY Shift Engineer or GSEP Station Director to classify the event and Initiate GSEP if required.

d. NOTIFY Rad/Chem to survey and sample.

D. SUSEQUENT OPERATOR ACTIONS

1. INSERT IRM and SRM Detectors and continue to MONITOR Reactor Power.

2. Continue to MONITOR ECCS performance to assure that Core Cooling and/or Vessel Depressurization are maintained.

3. VERIFY Post LOCA Monitor Initiation. MONITOR Containment Pressure, Temperature and Hydrogen Concentration.

CAUTION

Containment Spray should not be activated within 10 minutes of the LOCA occurence in order to prevent rapid Drywell Depressurization and diversion of water from the Reactor.

4. Approximately 10 minutes after the LOCA, RHR Service Water should be introduced to the RHR Heat Exchangers, in accordance with LOP-RH-05, in preparation for long term cooling of the core and containment.

5. INITIATE Suppression Pool Cooling in accordance with LOP-RH-13 and Containment Spray if the core remains covered when LPCI A/B Injection Valve 1E12-F042A/B (2E12F042A/B) is closed. If the Suppression Pool Cooling System cannot be actuated, DIRECT LPCI A/B Flow to the A/B RHR Heat Exchanger by CLOSING MO-1E12-F048A/B (MO-2E12-F048A/B).

CAUTION

If off-site power is lost during the LOCA, only one (1) operable RHR Subsystem need be operated in the Suppression Pool Cooling Mode.

6. If Drywell Pressure exceeds 30 psig or Suppression Chamber Pressure exceeds 25 psig, a Containment Leakage Bypass Path may exist. INITIATE Containment Spray to prevent exceeding Primary Containment Design Pressure.

CAUTION

If Containment Spray is not available, and a Bypass Leakage Path exists, DEPRESSURIZE the Reactor by OPENING Safety-Relief Valves.

7. CLOSE MO-1821-F065A and B (MO-2821-F065A and B) and MO-1G33-F040 (MO-2G33-F040) Feedwater and Reactor Water Cleanup Vessel Return Valves, if these systems are unavailable or unnecessary.

CAUTION

MO-1821-F065A and B (MO-2821-F065A and B) and MO-1G33-F040 (MO-2G33-F040) Feedwater and Reactor Water Cleanup Vessel Return Valves should be closed within 20 mins. of the initiation of the LOCA if a total loss of Feedwater Flow occurs during this time.

8. START the MSIV Leakage Control System in accordance with LOP-MS-02 if the following conditions are met:

   a. Twenty (20) minutes elapsed since start of LOCA.

   b. Main Steam Line Pressure is below Reactor Pressure.

   c. Main Steam Line Pressure is less than Drywell Pressure.

9. START the Hydrogen Recombiner System in accordance with LOP-HG-01, LOP-HG-02, LOP-HG-03, or LOP-HG-04 prior to four (4) hours after start of LOCA and prior to Primary Containment Hydrogen Concentration exceeding 3.3 percent by volume.

CAUTION

Bypassed steam resulting from operation of the Hydrogen Recombiner System could lead to high suppression chamber pressures. A portion of the RHR System flow should be diverted to the Suppression Chamber spray headers to quench the bypassed steam if this action will not uncover the core.

10. SHED unncessary loads from the Diesel Generators.

11. CHECK Lake Screen House Traveling Screens for blockage. If all screens are blocked, OPEN 0E12-F300, Screen Bypass Valve.

12. MANUALLY BACKWASH Diesel Generator Cooling water Strainers if high differential pressure occurs.

13. MONITOR Stack Gas Activity to determine GSEP category. INITIATE GSEP.

E. DISCUSSION

This procedure assumes the rupture of a Reactor connected system and a resultant loss of Reactor Coolant. It is further assumed that the leak occurs inside the Drywell and may be either steam or water. The operator must use all available information to identify the source and location of the leak. The temperature and pressure transients resulting from this accident maintain cladding integrity. Thus, activity released to the Drywell is a result of intrinsic coolant activity and activity released as a consequence of Reactor scram and depressurization. Subsequent to the LOCA, fission product decay heat will result in a continuing energy dump to the Suppression Pool. Unless this energy is removed from the Containment, it will result in unnacceptable Suppression Pool Temperature. After ten (10) minutes an RHR Heat Exchanger must be activated with cooling water flow.

Due to the uninerted Drywell, special attention must be given to monitoring Containment Hydrogen Concentration to prevent exceeding a four (4) percent volume concentration.

As a final last resort method of adding water to the reactor, capability exists to add lake water to the Reactor via B RHR Injection Line.

LOSS OF COOLANT (SLOW LEAK)

A. SYMPTOMS

1. If leak is within the Primary Containment:

   a. Control Room Panel 1PM13J (2PM13J) Alarms:

      1) DRYWELL AIR HUMIDITY HI.

      2) DRYWELL SUP POOL CHAMBER TEMP HI.

      3) DRYWELL SUPPRESSION CHAMBER RADIOACT HI.

      4) DRYWELL EQPT DRN SUMP TEMP HI.

      5) DRYWELL FLOOR DRN SUMP LEVEL HI-HI.

      6) DRYWELL EQUIP DRN SUMP LEVEL HI-HI.

      7) DRYWELL SUMP FILL-UP RATE HI-LO.

      8) DRYWELL EQUIP DRN SUMP FILL-UP RATE HI-LO.

      9) FLOW TO DRYWELL FLOOR DRN SUMP LEVEL HI.

   b. Control Room Panel 1PM13J (2PM13J) Indications:

      1) Drywell Equipment Drain Sump Leakage Volume and Pump Discharge and Flow Recorder Increasing.

      2) Drywell Floor Drain Sump Fill up Rate and Pump Discharge Flow Recorder Increasing.

      3) Drywell Containment Narrow Range Pressure Recorder Increasing.

      4) Drywell Air Particulate, Noble Gas and Iodine Recorder Increasing.

      5) Drywell Cooler Condensate Flow 1FI-RF001 (2FI-RF001) Increasing above 2 gpm.

      6) Drywell Equipment Drain Sump Temperature Indicator Increasing.

7) Drywell Relative Humidity Indicator Increasing.

c. Control Room Panel 1H13-P601 (2H13-P601) Alarms:

1) LEAK DETECTION PRIMARY CONT TEMP HI.

d. Other Symptoms:

1) High Drywell Cooler Differential Cooling Water Temperature.

2) High Drywell Cooler Differential Air Temperature.

2. If leakage exists outside the Primary Containment in the Reactor Building:

a. Control Room Panel 1PM13J (2PM13J) Alarms:

1) RX BLDG EQP DRAIN TK TEMP HI.

2) RX BLDG EQP DRN TANK FILL-UP RATE HI.

b. Control Room Panel 1PM13J (2PM13J) Indications:

1) Reactor Building Equipment Drain Tank Level Recorder Increasing.

2) Reactor Building Equipment Drain Tank Temperature Indicator Increasing.

c. Control Room Panel 1H13P601 (2H13P601) Alarms:

1) REACTOR BLDG RADIATION HI.

2) RHR Pump B/C CUBICLE TEMP HI.

3) RHR PUMP A CUBICLE TEMP HI.

4) LPCS/RCIC PUMP CUBICLE TEMP HI.

5) RCIC PIPE RTE EQUIP AREA TEMP HI.

6) LEAK DETECTION RX WATER CLNUP AMB TEMP HI.

7) LEAK DETECTION RX WATER CLNUP DIFF TEMP HI.

8) LEAK DETECTION RWC ROOMS AMB TEMP HI.

9) LEAK DETECTION RWC ROOMS DIFF TEMP HI.

10) LEAK DETECTION STM PIPE TUNNEL AMB TEMP HI.

11) LEAK DETECTION STM PIPE TUNNEL DIFF TEMP HI.

12) RHR EQUIP AREA DIFF TEMP OR AMB TEMP HI.

13) LD PIPE TUNNEL MS LINE AREA TEMP HI.

d. Other Symptoms:

1) Excess Flow Check Valve position (PM10J).

2) Area Radiation Monitor Alarms.

3) Increasing Indication on the following meters:

a) Stack Gas Radiation Increase.

b) Reactor Building Exhaust Plenum Radiation.

c) RBCCW Radiation Monitor.

d) RHR Cooling Water Effluent Radiation.

4. The following symptoms indicate an Instrument Line Break:

a. Instrument readings among instruments monitoring the same parameters differs or is erratic.

b. Annunciation of alarms associated with a broken Instrument Line.

c. Possible half scram associated with an RPS Instrument Line.

5. Standby Gas Treatment Auto Starts.

B. AUTOMATIC ACTIONS

1. Possible Reactor Vessel and Containment ISOLATIONS from Leak Detection or Process Radiation Instrumentation.

   a. Group 1 (Main Steam).

   b. Group 4 (Reactor Bldg. Ventilation, SBGTS INITIATION).

   c. Group 5 (Reactor Water Cleanup).

   d. Group 6 (Residual Heat Removal).

   e. Group 8 (Reactor Core Isolation Cooling).

C. IMMEDIATE OPERATOR ACTIONS

1. ATTEMPT to locate leak, EVALUATE effects of isolating leak, and ISOLATE leak if prudent.

2. As necessary:

   a. NOTIFY and EVACUATE Personnel.

   b. RESTRICT access.

   c. NOTIFY Shift Engineer or GSEP Station Director to classify the event and Initiate GSEP if required.

   d. NOTIFY Rad/Chem to survey and sample.

D. SUBSEQUENT OPERATOR ACTIONS

1. REVIEW Technical Specification Requirements.

   a. If the unisolatable leak is part of the Reactor Pressure Boundary in the Reactor Building:

      1) INITIATE a SCRAM and COMMENCE a Normal Reactor Cooldown so that the Reactor is in Cold Shutdown within six (6) hours.

      2) STARTUP Standby Gas Treatment and ISOLATE Reactor Building Ventilation.

b. If the unisolatable leak is inside the Primary Containment and greater than 5 gpm unidentified, or 25 gpm total leakage averaged over any twenty-four (24) hour period, or is determined to be Reactor Pressure Boundary leakage:

1) COMMENCE a Normal Reactor Shutdown and Cooldown so that the Reactor is in Hot Shutdown within twelve (12) hours, and Cold Shutdown within the next twenty-four (24) hours (T.S.).

2) START Suppression Pool Cooling in accordance with LOP-RH-13 if Suppression Pool Temperature approaches 100°F (T.S.).

2. REVIEW the following procedures:

a. LGA-01; Loss of Coolant (Fast Leak).

b. LGA-13; High Area Radiation.

c. LGA-14; High Airborne Activity.

d. LGA-15; Radiation Surface Contamination.

3. RESTORE any Out-of-Service Emergency Core Cooling Systems to service.

4. CONSULT with the Rad/Chem Supervisor and INCREASE the frequency of routine station and environmental monitoring if necessary. OBTAIN a sample of Reactor Coolant for analysis.

5. MONITOR the leak rate closely. If Primary Containment Pressure approaches the Scram setpoint or if the leak endangers personnel or equipment, MANUALLY SCRAM the Reactor and PERFORM a controlled cooldown.

E. DISCUSSION

This procedure is written to specify actions in the event of a Small Leak in the Reactor or connected systems which does not result in immediate Primary Containment and Reactor Vessel Isolation.

A slow leak will be observed as a result of high temperature, humidity, or radioactivity in the vicinity of the leak rather than changes in plant process variables.

such as steam flow, feed flow, reactor pressure, or reactor level.

Since automatic protective action does not occur immediately, the decision must be made to continue operation, place the plant in a more reliable mode or to perform a plant shutdown and cooldown. Variables that must be considered include the assumption that a leak due to a small crack can propagate into a Fast Leak, as well as the radiological consequences of the escape of the coolant.

Due to the various leak detection systems associated with Reactor connected systems, isolation of these systems may occur as an automatic action. Thus, the operator must determine the effect on plant operation of the loss of the isolated system.

Any leak inside the Primary Containment that cannot be treated as a Slow Leak must be treated operationally as a Fast Leak.

MAJOR STEAM LEAKS (OUTSIDE THE DRYWELL)

A. SYMPTOMS

1. Control Room Panel 1H13-P601 (2H13-P601) Alarms:

   a. LO PIPE TUNNEL MS LINE AREA TEMP HI.

   b. MAIN STEAM LINE FLOW HI.

2. Control Room Panel 1H13-P603 (2H13-P602) Alarms:

   a. MAIN STM ISOL VLV NOT FULL OPEN.

   b. CHAN A MSIV NOT FULL OPEN.

   c. CHAN B MSIV NOT FULL OPEN.

   d. MN STEAM LINE FLOW HI.

3. Other Symptoms:

   a. High Reactor Water Level.

   b. Increase in steam flow, feedwater flow and reactor power with turbine load unchanged.

   c. Area Temperature and Radiation Monitors reading high.

   d. Noise and water accumulation.

B. AUTOMATIC ACTIONS

1. ISOLATE Reactor Vessel and Containment.

   a. Group 1.

2. Indirect Reactor SCRAM.

C. IMMEDIATE OPERATOR ACTIONS

1. VERIFY:

   a. SCRAM, rods in, power decaying.

b. Group 1 ISOLATION.

2. As necessary.

a. NOTIFY and EVACUATE Personnel.

b. RESTRICT access.

c. NOTIFY Shift Engineer or GSEP Station Director to classify the event and Initiate GSEP if required.

d. NOTIFY Rad/Chem to survey and sample.

D. SUBSEQUENT OPERATOR ACTIONS

1. CARRY OUT procedure LGP 3-2 using the Motor Driven Reactor Feed Pump to maintain Reactor Water Level.

2. CONTINUE to control Reactor Pressure and Water Level.

3. VERIFY Reactor Mode Switch is in SHUTDOWN.

4. MAINTAIN Condenser Vacuum if possible.

5. After the Main Steam Lines have depressurized, LOCATE and ISOLATE the leak if possible.

6. If the leak is isolated and Condenser Vacuum is greater than 7" Hg:

a. RESET the Isolation Signal.

b. OPEN the Main Steam Isolation Valves in accordance with LOP-MS-01.

c. PERFORM a Normal Reactor Cooldown in accordance with LGP 2-1; Normal Unit Shutdown.

7. If all of the conditions in step D.6 cannot be met, PERFORM a Normal Reactor Cooldown in accordance with LOP-RH-09; Steam Condensing Startup and Operation.

8. REVIEW the following procedures:

a. LGA-13; High Area Radiation.

b. LGA-14; High Airborne Activity.

c. LGA-15; Radioactive Surface Contamination.

9. CHECK Stack Release Rate.

E. DISCUSSION

A Main Steam Line failure of sufficient severity will initiate closure of the Main Steam Isolation Valves. Since initial automatic protection is initiated, verification of automatic action ensures that MSIV Isolation occurs, the Reactor Scrams, Reactor water Level is maintained and Reactor Pressure remains under control.

After monitoring Reactor process parameters, action is directed toward ensuring adequate shutdown cooling via the Feedwater and Condensate Systems or the Steam Condensing and Shutdown Cooling Modes of RHR.

Since large amounts of energy and radioactivity may have been released, the extent of equipment damage should be ascertained as soon as possible.

HIGH REACTOR WATER LEVEL

A. SYMPTOMS

1. Control Room Panel 1H13-P603(2H13-P603) Alarms:
   a. FW CONTROL RX WATER LEVEL 7 HI.
   b. FW CONTROL RX WATER LEVEL 8 TRIP.
   c. Possible FW VALVE CONTROL SIGNAL FAILURE.
   d. Possible TURBINE RFP CONTROL SIGNAL FAILURE.
   e. Possible RX AUTO SCRAM.
   f. Possible RX VESSEL PRESSURE HI.
   g. TSV NOT FULL OPEN.
   h. TURB STOP VLV CLOSURE TRIP.
2. Control Room Panel 1H13-P603(2H13-P603) Indications:
   a. Reactor Water Level Instrumentation Level Increasing.
   b. Turbine Driven or Motor Driven Reactor Pump increasing flow or Feedwater Header increasing flow with constant Steam Flow.
3. Control Room Panel 1PM03J(2PM03J) Alarms:
   a. RX FEED WATER PUMP TRIP.
   b. Possible CNDS HOTWELL LEVEL HI-LO.
4. Control Room Panel 1PM02J(2PM02J) Alarms:
   a. TURB MN STEAM BYPASS VALVE NOT CLOSED.
   b. Possible FW PUMPS TURB VIBRATION HI.
   c. Possible TURBINE TRIP VIBRATION HI.

5. Control Room Panel 1H13-P601(2H13-P601) Alarms:

   a. REACTOR VESSEL WATER LEVEL 8 HI.

   b. Possible RCIC TURBINE TRIP.

6. Other Indications:

   a. Increasing Level Indication on POST LOCA Recorder.

   b. Loss of Main Turbine Generator Output.

B. AUTOMATIC ACTIONS

1. Level 8, Reactor Vessel Water Level 54.5".

   a. TRIP Main Turbine.

   b. TRIP RCIC Turbine.

   c. TRIP Reactor Feed Pumps.

   d. CLOSE HPCS Injection Valve.

2. Reactor Scram from Turbine Stop Valve Closure above Main Turbine First Stage Pressure corresponding to 30% Reactor Power.

3. Recirculation Pump Trip (RPT) from Turbine Stop Valve Closure above Main Turbine First Stage Pressure corresponding to 30% Reactor Power.

4. Possible Lockout of the Turbine Driven Reactor Feed Pumps or Feedwater Regulation Valve (FRV).

C. IMMEDIATE OPERATOR ACTIONS

1. If the Automatic Feedwater Control System has malfunctioned or A or B Out of Service Light is illuminated:

   a. VERIFY Manual/Auto signals are balanced.

   b. PLACE Manual/Auto Transfer Station in MANUAL.

c. RESET Lockout, if one exists.

d. Manually CONTROL Reactor Level.

e. If TDRFP Lockout cannot be cleared, CONTROL TDRFP with Motor Speed Changer and Jack.

2. If Level 8 is reached, VERIFY Actions. Also,

a. If Reactor Power is greater than 30%:

1) SCRAM, rods in, power decaying.

2) RPT to 15 Hz.

3) Bypass and Safety/Relief valves control pressure.

b. As level decreases,

1) Place Feedwater Reg. M/A Station to Manual and CLOSE valve.

2) START MDRFP.

3) CONTROL Level using Feedwater Reg. Valve.

4) START RCIC if necessary to maintain level above Level 4.

D. SUBSEQUENT OPERATOR ACTIONS

1. If control of Reactor Water Level has been re-established prior to automatic protective action:

a. STATION an additional operator in the Control Room to monitor and control Reactor Vessel Water Level. He should be assigned no other duties until the Feedwater System is returned to normal.

b. MONITOR the Reactor Water Level and Turbine Supervisory Instrumentation. NOTIFY the Shift Supervisor of any abnormal indications.

c. DETERMINE the specific cause for the High Reactor Water Level and INITIATE appropriate corrective action as required.

2. Subsequent Action if Reactor Water Level 8 was exceeded:

a. CONTINUE to control Reactor Water Level and Pressure.

1) OPERATE Bypass Valves as necessary if the Main Condenser is available. If the Main Steam Isolation Valves were closed due to a low Reactor Water Level Isolation Signal, RESET the Isolation Signal and OPEN the MSIV's in accordance with LOP-MS-01.

2) If the Main Condenser is not available, STARTUP the Steam Condensing Mode of RHR in accordance with LOP-RH-09, if required, to control Reactor Pressure.

b. VERIFY that the Main Generator Trips, OCBs 9-10 and 10-11 (OCBs 2-3 and 3-4) OPEN, and the Transfer of Auxiliary Power to the System Auxiliary Transformer (SAT).

c. CHECK the Main Turbine and the Turbine Driven Reactor Feed Pumps vibration during coastdown. MINIMIZE Water Induction as follows:

1) OPEN the following Drain Valves in order to minimize the effects of possible Turbine Water Induction:

a) Main Turbine Lead Drain Valves MO-1821-CV2 and 4 (MO-2821-CV2 and 4) and AO-1821-F421A and C (AO-1821-F421A and C).

b) "A" Reactor Feed Pump Turbine.

(1) Below Seat Drain Valves MO-1821-F425A and F427A(MO-2821-F425A and F427A).

(2) First Stage Drain Valve, MO-1821-F432A(MO-2821-F432B).

c) "B" Reactor Feed Pump Turbine.

(1) Below Seat Drain Valves MO-1821-F425B and F427B (MO-2821-F425B and F427B).

(2) First Stage Drain Valve, MO-1821-F432B(MO-2821-F432B).

d. VERIFY that the Turning Gear Oil Pump Starts.

e. START Suppression Pool Cooling in accordance with LOP-RH-13, if Suppression Pool temperature approaches 100°F (T.S.).

f. REFER to LGP 3-2, Reactor Scram Procedure. VERIFY that Reactor Safety Limits were not exceeded during the transient prior to restart of the Unit.

E. DISCUSSION

High Reactor Vessel Water Level during steady state power operation can occur as a direct result of erroneous Feedwater Control System increased demand, or a symptom of other transients, such as a Pressure Regulator failure calling for decreased Reactor Pressure.

Action is directed towards restoring the level to normal by taking control of the Feedwater Control System. However, if Level EIGHT is exceeded, action is directed toward ensuring automatic actions to limit Reactor Water Level rise, and prevent Turbine damage. In addition, protective features to mitigate the associated pressure transient are verified. Finally, control of Reactor Water Level and Pressure is established.

It is important to monitor Turbine machinery to minimize water induction.

LOW REACTOR WATER LEVEL

A. SYMPTOMS

1. Control Room Panel 1H13-P603 (2H13-P603) Alarms:
   a. FW CONTROL RX WATER LEVEL 4 LO.
   b. FW VALVE CONTROL SIGNAL FAILURE.
   c. TURBINE RFP CONTROL SIGNAL FAILURE.
   d. Possible RX VESSEL WTR LEVEL 3 LO/TRIP.
   e. Possible RX AUTO SCRAM.
   f. Possible RX VESSEL WTR LEVEL 2 LO.
2. Control Room Panel 1H13-P603 (2H13-P603) Indications:
   a. Turbine Driven Feed Pump Decreasing Flow.
   b. Motor Driven Feed Pump Decreasing Flow.
   c. Feedwater Heater Flow Decreasing.
   d. LOCKOUT light(s) illuminated.
   e. LEVEL A or LEVEL B OUT OF SERVICE Light illuminated.
   f. Abnormal Demand or Output Signal on a Feedwater System Controller.
   g. Reactor Water Level Decreasing.
3. Control Room Panel 1PM03J (2PM03J) Alarms:
   a. RX FEEDWATER PUMP TRIP.
   b. FW PUMP DISCH PRESS LO.
   c. RX FW PUMP TURBINE HYD OIL PRESS LO.
   d. FW PUMPS SUCTION PRESS LO.

e. CNDS and BOOSTER PUMP AUTO-TRIP.

f. CNDS BSTR PMPS SUCTION HDR PRESS LO.

g. CNDS PUMPS DISCH HDR PRESS LO.

h. HEATER DRAIN PUMP AUTO TRIP.

4. Control Room Panel 1PM03J (2PM03J) Indications:

a. Feedwater Pump(s) Flow Decreasing.

b. Turbine Driven Reactor Feed Pump Speed Decreasing.

c. Low Pressure or High Pressure Heater String Isolation.

d. Motor Driven Feed Pump AC Amps Decreasing.

e. Heater Drain Pump Decreasing Amps.

f. Feedwater Pump Suction Header Pressure Changes.

g. Feedwater Pump Discharge Header Pressure Changes.

h. Feedwater Turbine Control Oil Pressure Decreasing, Control or Stop Valves Closing.

i. Condensate/Condensate Booster Pump Suction or Discharge Pressure Changes.

j. Heater Drain Tank High or Low Level.

5. Other Indications:

a. Possible Recirc. Flow Control Valve Runback.

b. Possible Decreasing Reactor Power, Recirc. Flow, or Jetpump Flow.

c. Feedwater Isolation Valves Closed Indication.

d. Auto Start of Auxiliary Oil Pump.

e. Possible Decreasing Turbine Generator Output and Bypass Valve Steam Flow.

f. Decreasing Level indication on Control Room Panels 1H13-P601 (2H13-P601) and 1H13-P602 (2H13-P602).

g. Possible RHR F006A(B) - F054A(B) OPEN alarm.

B. AUTOMATIC ACTIONS

1. Level 4, Reactor Vessel Water Level 31.5" actions.

a. Low Level Alarm.

b. Recirc Flow Control Valve Runback permissive.

2. Level 3, Reactor Vessel Water Level 12.5" actions.

a. Reactor Scram.

b. TRIP Reactor Recirculation Pumps to 15 HZ.

c. ADS Permissive.

d. ISOLATE Vessel and Containment Groups 4, 5, and 7.

e. SETBACK Programmed Level Control to 18".

C. IMMEDIATE OPERATOR ACTIONS

1. If Level 3 is reached, verify actions, DO NOT continue this procedure, and go to LGP 3-2.

2. If Level 4, is reached, verify actions.

3. Determine cause and correct if possible.

4. Restore level if possible by:

a. Increasing Feedwater Flow.

b. Reducing Reactor Power to match Steam Flow and Feedwater Flow.

D. SUBSEQUENT OPERATOR ACTIONS

1. If the Reactor has scrammed, complete performance of LGP 3-2, Reactor Scram.

2. If control of Reactor Water Level has been re-established prior to automatic protective action:

   a. STATION an additional operator in the Control Room to monitor and control Reactor Water Level. He should be assigned no other duties until the Feedwater System is returned to normal.

   b. MONITOR all Reactor Water Level instrumentation. NOTIFY the Shift Supervisor of any abnormal indications.

   c. DETERMINE the Specific Cause for the Low Reactor Water Level and INITIATE corrective action as required.

E. DISCUSSION

A decreasing Reactor Water Level can occur at any power level and in any mode of the Feedwater Level Control System. Probable causes are loss of Feedwater or failure of the Feedwater Level Control System.

Automatic Reactor protection is ultimately provided by a Reactor Scram due to Low Reactor Water Level 3. However, inherent designs in the Feedwater System and Recirculation System may prevent a scram especially if supplemented by operator action. The basic features are:

1. Auto Start of the Motor Driven Reactor Feed Pump, if both Turbine Driven Reactor Feed Pumps Trip.

2. Runback of the Recirculation Flow Control Valve to limit Reactor Steam flow to the capacity of one Turbine Driven Reactor Feed Pump.

3. Lockout of the Turbine Driven Reactor Feed Pumps and Feedwater Regulation Valve, if an abnormal control signal is detected.

Action is directed toward maintaining sufficient Feedwater flow. However, should a Reactor Scram occur, then scram recovery is initiated while paying particular attention to core coverage and cooling.

REACTOR HIGH PRESSURE

A. SYMPTOMS

1. Control Room Panel 1H13-P603 (2H13-P603) Alarms:

   a. RX VESSEL PRESSURE HI.

2. Control Room Panel 1H13-P603 (2H13-P603) Indications:

   a. Reactor Narrow Range Pressure Recorder High.

   b. Reactor Wide Range Pressure Indicator High.

3. Control Room Panel 1H13-P602 (2H13-P602) Alarms:

   a. RR PUMP 1A TRIP - ATWS INITIATED.

   b. RR PUMP 1B TRIP - ATWS INITIATED.

4. Control Room Panel 1H13-P601 (2H13-P601) Alarms:

   a. SAFETY/RELIEF VALVE FULLY OPEN.

5. Control Room Panel 1H13-P601 (2H13-P601) Indications:

   a. Post Accident Monitor Recorder A Pressure High.

   b. Post Accident Monitor Recorder B Pressure High.

B. AUTOMATIC ACTIONS

1. 1043# Reactor Pressure Scrams Reactor.

2. 1076# Reactor Pressure and above. SAFETY/RELIEF VALVES ACTUATE at setpoints. Also, possible LOW LEVEL SETBACK (LLS) if 2 or more valves actuated.

3. 1120# Reactor Pressure TRIPS Reactor Recirculation Pumps to OFF (ATWS).

C. IMMEDIATE OPERATOR ACTIONS

1. If 1043# is reached, verify actions.

2. If 1076# is reached verify S/R valve operation as appropriate and LLS if 2 or more valves open.

3. If 1120# is reached, verify RPT.

4. Control level with Feedwater, RCIC, and HPCS, as necessary.

D. SUBSEQUENT OPERATOR ACTION

1. Continue to CONTROL Reactor Pressure, Level and Suppression Pool Temperature.

2. CORRECT the cause of the High Reactor Pressure Condition.

3. When recovery can be commenced, REFER to Procedure LGP 1-1; Normal Unit Startup.

E. DISCUSSION

High Reactor Pressure is always an indication of another initiating event. For example, a Turbine Trip, Reactivity addition, Pressure Regulator malfunction (double failure), MSIV closure or Feed System failure will cause transients which increase Reactor Pressure. During power operation an APRM Scram should always preceed a Reactor High Pressure Scram except for the unlikely event of a Turbine Trip with Bypass Valve failure below 30% Power. Should failure of an increase in Neutron Flux to scram the Reactor occur, Reactor High Pressure will initiate the scram. In the event that this backup protection fails, Anticipated Transient Without Scram (ATWS) will trip the Reactor Recirculation Pumps.

Action is directed toward ensuring the scram occurs and that Reactor Pressure is relieved to prevent over pressurization. In addition, Reactor Water Level and core cooling must be maintained.

INADVERTENT REACTIVITY ADDITION

A. SYMPTOMS

1. Control Room Panel 1H13-P603 (2H13-P603) Alarms:

   a. SRM HI/INOP.

   b. SRM SHORT PERIOD.

   c. RBM HI/INOP.

   d. IRM HI.

   e. LPRM HI.

   f. APRM HI.

   g. Possible ROD OUT BLOCK.

   h. Possible CRD DRIFT.

   i. Possible FW FLOW CONTROL SIGNAL FAILURE.

   j. Possible FLUX CONTROLLER OUTPUT SIGNAL ABNORMAL.

   k. Possible MASTER CONTLR OUTPUT SIGNAL ABNORMAL.

   l. Possible RX AUTO SCRAM.

   m. Other alarms signfying increasing power.

2. Control Room Panel 1H13-P603(2H13-P603) Indications:

   a. Positive Reactor Period.

   b. Increasing Reactor Power.

   c. A transient in Reactor Pressure.

3. Other Symptoms:

   a. Possible Increasing Turbine Generator Output or Bypass Valve Flow.

   b. Possible Increase in Feedwater Flow.

c. Possible Increase in Recirculation Flow.

B. AUTOMATIC ACTIONS

1. Reactor Scram if Limiting Safety System Settings are exceeded.

2. If in Master Automatic Flux Control, Recirculation Flow decreases unless the cause of the reactivity addition was a Recirculation Control System Failure.

C. IMMEDIATE OPERATOR ACTIONS

1. If LSSS is exceeded and Reactor Scrams, DO NOT continue this procedure and go to LGP 3-2.

2. If LSSS is exceeded and Reactor DOES NOT Scram, DO NOT continue this procedure and go to LGA-18 (ATWS).

3. If LSSS is NOT exceeded, determine cause of reactivity addition, and initiate corrective action to terminate cause.

4. Maintain Reactor Vessel Pressure and Level.

D. SUBSEQUENT OPERATOR ACTIONS

1. CHECK Stack Gas Activity and Off-Gas Activity for any increase that would indicate core damage or abnormal release of radioactivity.

2. OBTAIN a sample of Reactor Coolant. If indications of fission product release are present, PERFORM LGA-16, Fuel Element Failure.

3. CHECK that Reactor Power and Flow have not exceeded Reactor Operational or Thermal Limits for Single and Two Pump Recirculation Flow. CHECK that Reactor Coolant System Temperature Cooldown Rate Limits were not exceeded (T.S.).

4. If any Limiting Condition for Operation has been exceeded, EVALUATE the necessity of Normal Shutdown in accordance with LGP 2-1 in order to be in Hot Shutdown within 12 hours and Cold Shutdown within the following 24 hours (T.S.).

5. If corrective action succeeded in correcting the Reactivity Addition, ADJUST Control Rods as necessary to correspond to the new Power level.

6. RE-ESTABLISH the appropriate Reactor Power Level in accordance with LGP 1-1, Normal Unit Startup and LGP 3-1, Power Changes.

E. DISCUSSION

Inadvertent Reactivity Addition is an indication of another initiating cause. The following are possible causes: Continuous Withdrawal of a High Worth Rod (Multiple Failure), Rod Drop Accident, Improper Startup of an idle Recirculation Pump (Operator error), Recirculation Flow Control Valve Failure in the open direction, a Xenon transient or the injection of cold water into the core from sources such as RHR Shutdown Cooling (Operator error) or loss of Feedwater Heaters.

Action is immediately directed towards terminating the cause of the Reactivity Addition if the cause is known and time is available. If the Reactivity Addition is rapid, action is directed toward verifying automatic action occurs and controlling Reactor Pressure and Level. In both cases Operational, Thermal Hydraulic and Radiation Release Limits are evaluated prior to continuing recovery.

LOSS OF RECIRCULATION FLOW - SINGLE PUMP

A. SYMPTOMS

1. Control Room Panel 1H13-P603 (2H13-P603) Alarms:

   a. ROD OUT BLOCK.

   b. APRM HI.

2. Control Room Panel 1H13-P603 (2H13-P603) Indications:

   a. Total Core Flow and Pressure Recorder Decreasing.

   b. Reactor Power Decreasing.

   c. Steam Flow and Feedwater Flow Decreasing.

   d. Reactor Feedpump Flow Indicators Decreasing.

   e. Reactor Water Level Increases then Decreases to Normal.

3. Control Room Panel 1H13-P602 (2H13-P602) Alarms:

   a. RX RECIRC PUMP 1A(2A) or 1B(2B) AUTO TRIP.

4. Control Room Panel 1H13-P602 (2H13-P602) Indications:

   a. Reactor Recirculation Pump Flow Decreasing in one (1) Recirculation Loop.

   b. Jet Pump Flow associated with the applicable Recirculation Loop Decreasing.

   c. Jet Pump Flow associated with the other Recirculation Loop Increasing.

   d. Reactor Recirculation Pump Differential Pressure A or B Decreasing.

   e. Reactor Recirculation Pump Motor A or B Current Decreasing.

5. Other Symptoms:

a. Decreasing Main Generator Amps and watts.

b. MSR Outlet Pressures Decreasing.

B. AUTOMATIC ACTIONS

1. 1A or 1B (2A or 2B) Reactor Recirculation Pump TRIPS.

C. IMMEDIATE OPERATOR ACTIONS

1. Verify M/A Station transferred to Manual.

2. Place tripped recirc pump Bkr 3 in Pull-to-Lock (PTL).

3. Decrease Reactor Power to less than (later) by decreasing recirc. flow.

4. Maintain Reactor Pressure and Level.

D. SUBSEQUENT OPERATOR ACTIONS

1. VERIFY that the Reactor Recirculation Pump Discharge and Suction Block Valves are OPEN in the tripped Recirculation Loop.

2. VERIFY Seal Staging Flow Stop Valve SO-1B33-F079A/B (SO-2B33-F079A/B) is CLOSED as indicated on 1H13-P602 (2H13-P602).

3. DETERMINE the cause of the Reactor Recirculation Pump Failure and INITIATE appropriate corrective action as required.

4. After the cause has been found and corrected and it is VERIFIED that no Operational Limits have been exceeded. RESTART the tripped Reactor Recirculation Pump in accordance with LOP-RR-06.

5. If any Limiting Condition for Operation has been exceeded. EVALUATE the necessity of Normal Shutdown in accordance with LGP 2-1 in order to be in Hot Shutdown within twelve (12) hours and Cold Shutdown within the following twenty-four (24) hours (T.S.).

6. If two (2) Loop Reactor Recirculation cannot be restored. RESTRICT Reactor Power to less than or equal to 75% of Rated Thermal Power. See Technical Specification 3/4.4.1.

7. CHECK Stack Gas Activity and OBTAIN a sample of Reactor Coolant.

E. DISCUSSION

A loss of one (1) Reactor Recirculation Pump with normal coastdown will not normally result in a Reactor Scram during operation. Should a Reactor Scram occur during this transient, the possibility of instrument malfunction or the unlikely event of Reactor Recirculation Pump seizure may exist. Therefore, procedure LGP 3-2, Reactor Scram, should be followed if a Reactor Scram occurs.

Action is directed toward ensuring that the Reactor Recirculation Pump trip proceeds to completion (e.g. that the Reactor Recirc. MG Set does not energize the pump in Slow Speed) and that Core Thermo-Hydraulic Limits are not exceeded.

The trip of one (1) Reactor Recirculation Pump to Slow Speed may be indicative of a failure of both Reactor Recirculation Pumps to transfer to Slow Speed. This situation would occur following the failure of a cavitation protection trip to transfer both Reactor Recirculation Pumps to Slow Speed.

The following are likely causes for a single Reactor Recirculation Pump Trip:

1. Suction or Discharge Block Valve less than 90% open.

2. Pump Electrical Malfunction.

3. Low Frequency MG Set Electrical Malfunction.

LOSS OF RECIRCULATION FLOW - BOTH LOOPS

A. SYMPTOMS

1. Control Room Panel 1H13-P603 (2H13-P603) Alarms:

   a. ROD OUT BLOCK.

   b. APRM HI.

2. Control Room Panel 1H13-P603 (2H13-P603) Indications:

   a. Total Core Flow and Pressure Recorder Decreasing.

   b. Reactor Power Decreasing.

   c. Steam Flow and Feedwater Flow Decreasing.

   d. Feedwater Pump Flow Indicators Decreasing.

   e. Reactor Water Level Increases then Decreases.

3. Control Room Panel 1H13-P602 (2H13-P602) Alarms:

   a. RX RECIRC PUMP 1A(2A) AUTO TRIP.

   b. RX RECIRC PUMP 1B(2B) AUTO TRIP.

4. Control Room Panel 1H13-P602 (2H13-P602) Indications:

   a. Reactor Recirculation Pump Flow Decreasing in both Recirculation Loops.

   b. Jet Pump Flow Decreasing.

   c. Reactor Recirculation Pump A and B Differential Pressure Decreasing.

   d. Reactor Recirculation Pump Motor A and B Current Decreasing.

5. Other Symptoms:

   a. Main Generator output decreasing.

   b. MSR Outlet Pressures Decreasing.

B. AUTOMATIC ACTIONS

1. None.

C. IMMEDIATE OPERATOR ACTIONS

1. PLACE the mode switch in Shutdown (T.S.) and carry out LGP 3-2.

D. SUBSEQUENT OPERATOR ACTIONS

1. VERIFY that the Reactor Recirculation Pump Discharge and Suction Block Valves are open in A and B Recirculation Loops.

2. VERIFY Seal Staging Flow Stop Valves SD-1833-F079 A and B (SD-2833-F079 A & B) are closed.

3. DETERMINE the cause of the Reactor Recirculation Pump Failure, and INITIATE appropriate corrective action as required.

4. After the cause has been found and corrected and it is VERIFIED that no Operational Limits or Core Thermal Limits have been exceeded, RESTART both Reactor Recirculation Pumps in accordance with LOP-RR-06.

5. CHECK Stack Gas ACtivity. OBTAIN a sample of Reactor Coolant.

6. CHECK Turbine for excessive vibration.

E. DISCUSSION

The complete loss of two (2) Reactor Recirculation Pumps is a highly unlikely event signified by the loss of four (4) power supplies. Should both Reactor Recirculation Pumps trip, ENSURE that Anticipated Transient Without Scram - ATWS (Reactor Vessel Pressure 1135 psig or Low Reactor water Level 2) was not the cause. SCRAM the Reactor in accordance with LGP 3-2, Reactor Scram, if ATWS was the cause.

In addition, a reactor Recirc. Flow Control Valve (FCV) Runback due to a loss of Feedwater Flow or failure of the Reactor Recirc. Flow Control Valves (FCV) in a closed direction will result in a similar transient response.

A tion is directed toward ensuring that Core Thermo-hydraulic Limits are not exceeded and toward ensuring a timely restoration of Reactor Recirculation Flow if possible.

LOSS OF TURBINE GENERATOR LOAD GREATER THAN 25%

A. SYMPTOMS

1. Control Room Panel 1PM02J (2PM02J) Alarms:

   a. Possible TURBINE TRIP THRUST BEARING FAILURE.

   b. Possible TURBINE TRIP MSR LEVEL HI.

   c. Possible TURBINE TRIP VIBRATION HI.

   d. Possible TURBINE TRIP LOSS OF STATOR COOLANT.

   e. Possible TURBINE TRIP EXHAUST HOOD TEMP HI.

   f. Possible TURBINE TRIP EHC HYD PRESS LO.

   g. Possible TURBINE TRIP EHC MANUAL.

   h. Possible TURBINE TRIP EHC MASTER.

   i. Possible TURBINE TRIP EHC COMPUTER TRIP.

   j. Possible TURBINE TRIP VACUUM LO.

   k. Possible TURBINE TRIP SPEED SIGNALS LOST.

   l. Possible TURBINE TRIP SHAFT PUMPS DIS PRESS LO.

   m. Possible TURBINE TRIP LOSS OF PS 100 A/B.

   n. Possible TURBINE TRIP OVERSPEED.

   o. Possible TURBINE TRIP LOSS OF 24V DC.

   p. Possible TURBINE TRIP BACKUP OVERSPEED.

   q. Possible TURBINE TRIP NON-EHC.

2. Control Room Panel 1PM01J(2PM01J) Alarms:

   a. Possible GEN 1(2) PROT RELAY TRIP.

b. Possible GEN 1(2) TRIP SYS 1 LKO TRIP.

c. Possible GEN 1 TRIP SYS 2 LKO TRIP.

d. Possible Alarms Indicating Main Transformer Trouble.

3. Other Indications:

a. Decreasing Turbine Generator Output.

b. Extraction Dump Valves Open.

c. MSR Outlet Pressure Decreasing.

d. Bypass Valves Open.

e. Safety-Relief Valves Open.

f. Possible Loss or Transfer of Auxiliary Power.

g. Reactor Scram.

h. Recirculation Pumps Trip to Slow Speed (RPT).

i. Possible High Reactor Water Level Indications.

B. AUTOMATIC ACTIONS

1. Turbine Stop Valves (MSV's) Close.

2. Direct Reactor Scram from Turbine Stop Valve Closure.

3. Recirculation Pump Trip to slow speed (RPT) from Turbine Stop Valve Closure.

4. Bypass Valves Open to control Reactor Pressure.

5. Safety-Relief Valves Open as necessary.

6. Main Generator direct or indirect trip.

C. IMMEDIATE OPERATOR ACTIONS

1. VERIFY that the Reactor Scrams, all Control Rods fully insert and Reactor Power decreases with normal neutron decay.

2. VERIFY that the Recirculation Pumps Trip (RPT) to Slow Speed.

3. VERIFY that the Bypass Valves and Safety-Relief Valves Open as necessary to control Reactor Pressure.

4. VERIFY that Main Turbine Speed is decreasing. If Main Turbine Speed is not decreasing, CHECK that the Turbine Stop Valves (MSV's) have closed.

   a. If MSV's are not closed, CONSIDER closing the MSIV's.

5. VERIFY that the Main Generator Trips, OCBs 9-10 and 10-11 (OCBs 2-3 and 3-4) OPEN, and the Transfer of Auxiliary Power to the System Auxiliary Transformer (SAT).

6. CHECK Reactor Water Level. MAINTAIN Reactor Water Level above Level 4, using the Reactor Feed System if possible.

7. START RCIC if necessary to maintain Reactor Water Level above Level 4.

D. SUBSEQUENT OPERATOR ACTIONS

1. CONTINUE to control Reactor Water Level and Pressure:

   a. OPERATE Bypass Valves as necessary if the Main Condenser is available. If the Main Steam Lines are closed, VERIFY Isolation Signal is RESET, and OPEN MSIV's in accordance with LOP-MS-01.

   b. If the Main Condenser is not available, STARTUP Steam Condensing in accordance with LOP-RH-09 if required to control Reactor Pressure.

2. VERIFY that the Turning Gear Oil Pump Starts.

3. VERIFY that the Turning Gear Engages.

4. START Suppression Pool Cooling in accordance with LOP-RH-13 if Suppression Pool Temperature approaches 100°F (T.S.).

5. DETERMINE the cause of the Main Turbine Generator Trip and INITIATE appropriate corrective action as required.

6. REFER to LGP 3-2, Reactor Scram. VERIFY that Reactor Safety Limits were not exceeded during the transient prior to restart of the unit.

E. DISCUSSION

Loss of the Main Turbine Generator at Reactor Power Levels above 30% results in a direct Reactor Scram. Safety-Relief Valves may operate to mitigate the Reactor Pressure Transient. The initial Reactor Power Level directly affects the number of Safety-Relief Valves required to operate. A Turbine Trip from high Reactor Power Levels with loss of bypass valve capability requires all Safety-Relief Valves to operate.

Action is directed toward verifying Reactor Scram, RPT and proper Bypass and Safety-Relief Valve operation. Once the integrity of the Reactor Coolant Pressure Boundary is verified, Turbine Generator Protective Actions are verified. Subsequently, Reactor Level is controlled to ensure adequate core cooling, and a timely recovery.

At high Reactor Power Levels, transient response is very rapid. Thus, the verification of Automatic Action is required prior to initiating manual recovery.

LOSS OF TURBINE GENERATOR LOAD LESS THAN 25%

A. SYMPTOMS

1. Control Room Panel 1PM02J(2PM02J) Alarms:

   a. Possible TURBINE TRIP THRUST BEARING FAILURE.

   b. Possible TURBINE TRIP MSR LEVEL HI.

   c. Possible TURBINE TRIP VIBRATION HI.

   d. Possible TURBINE TRIP LOSS OF STATOR COOLANT.

   e. Possible TURBINE TRIP EXHAUST HOOD TEMP HI.

   f. Possible TURBINE TRIP EHC HYD PRESS LO.

   g. Possible TURBINE TRIP EHC MANUAL.

   h. Possible TURBINE TRIP EHC MASTER.

   i. POssible TURBINE TRIP EHC COMPUTER TRIP.

   j. Possible TURBINE TRIP VACUUM LO.

   k. Possible TURBINE TRIP SPEED SIGNALS LOST.

   l. Possible TURBINE TRIP SHAFT PUMPS DIS PRESS LO.

   m. Possible TURBINE TRIP LOSS OF PS 100 A/B.

   n. Possible TURBINE TRIP OVERSPEED.

   o. Possible TURBINE TRIP LOSS OF 24VDC.

   p. Possible TURBINE TRIP BACKUP OVERSPEED.

   q. Possible TURBINE TRIP NON-EHC.

2. Control Room Panel 1PM01J(2PM01J) Alarms:

   a. Possible GEN 1(2) PROT RELAY TRIP.

b. Possible GEN 1(2) TRIP SYS 1 LKO TRIP.

c. Possible GEN 1 TRIP SYS 2 LKO TRIP.

d. Possible Alarms Indicating Main Transformer Trouble.

3. Other indications:

a. Decreasing Turbine Generator Output.

b. Extraction Dump Valves Open.

c. MSR Outlet Pressure Decreasing.

d. Bypass Valves Open.

e. Possible Safety-Relief Valves Open.

f. Possible Loss or Transfer of Auxiliary Power.

g. Reactor Power below 30%.

h. Possible High Reactor Water Level Indications.

B. AUTOMATIC ACTIONS

1. Turbine Stop Valves (MSV's) Close.

2. Bypass Valves Open to Control Reactor Pressure.

3. Main Generator direct or indirect trip.

C. IMMEDIATE OPERATOR ACTIONS

1. CHECK that Reactor Power is below 30% and NOT increasing.

2. If Reactor Power is increasing, MANUALLY SCRAM the Reactor. REFER to LGA-10, Loss of Turbine Generator Load Greater than 25%.

3. VERIFY that the Bypass Valves are controlling Reactor Pressure via the Pressure Regulator.

4. VERIFY that Main Turbine Speed is decreasing. If Main Turbine Speed is not decreasing, CHECK that the Turbine Stop Valves (MSV's) have closed.

a. If the MSV's are not closed, CONSIDER Scramming the Reactor and Closing the Main Steam Isolation Valves (MSIV's). VERIFY that the Safety-Relief Valves open as necessary to control Reactor Pressure.

5. VERIFY that the Main Generator Trips, OCBs 9-10 and 10-11 (OCBs 2-3 and 3-4) OPEN, and the Transfer of Auxiliary Power to the System Auxiliary Transformer (SAT).

6. CHECK Reactor Water Level, MAINTAIN Reactor Water Level above Level 4, using the Reactor Feed System if possible.

D. SUBSEQUENT OPERATOR ACTIONS

1. If the Reactor Scrammed, REFER to LGA-10, Loss of Turbine Generator Load Greater than 25%.

2. If the Reactor has not Scrammed:

a. VERIFY that the Turning Gear Oil Pump Starts.

b. DECREASE Reactor Power to limit Bypass Valve Steam Flow.

c. MAINTAIN Main Condenser Vacuum.

d. VERIFY that the Turning Gear Engages.

e. DETERMINE the cause of the Main Turbine Generator Trip and INITIATE appropriate corrective action as required.

f. CONTINUE Normal Unit Startup or Normal Unit Shutdown as applicable.

E. DISCUSSION

Loss of the Main Turbine Generator at Reactor Power Levels below 30 percent should not result in a direct Reactor Scram, since this Reactor Power Level is within the capacity of the Bypass Valves and Auxiliary Steam loads. Should the Bypass Valves fail to respond to accomodate Reactor Output, a Reactor Power or Reactor Pressure Scram will be initiated.

Action is directed toward verifying Reactor Power Level and proper Bypass Valve operation. Once these criteria are established, Turbine Generator Protective Actions are verified. Subsequently, depending upon plant response, Reactor Power, Pressure and Level are controlled to prepare for a Unit Restart or Controlled Shutdown.

LOSS OF AUXILIARY ELECTRICAL POWER

A. SYMPTOMS

1. Control Room Panel 1PM01J (2PM01J) Alarms:

   a. TR 1E-1W (2E-2W) BACKUP DIFF TRIP.

   b. GEN 1(2) PROT RELAY TRIP.

   c. GEN 1(2) SYS 1 LKO TRIP.

   d. GEN 1(2) SYS 2 LKO TRIP.

   e. MAIN T-1W (T-2W) PROT RELAY TRIP.

   f. MAIN T-1E (T-2E) PROT RELAY TRIP.

   g. UNIT AUX T-141 (T-241) PROT REL TRIP.

   h. All 6.9 KV and 4.16 KV Feeder Breaker Trip alarms annunciate.

   i. Undervoltage alarms on Non-essential Buses annunciate.

   . SYS AUX T-142 (T-242) PROT REL TRIP.

2. Control Room Panel 1PM01J (2PM01J) Indications:

   a. Main Condenser Output Breakers, SAT and UAT Supply Breaker to 6.9 KV and 4.16 KV Buses OPEN.

   b. Voltage, Current and Power to all Buses Decreasing.

   c. Diesel Generators 0 and 1A (2A) START.

3. Control Room Panel 1H13-P603 (2H13-P603) Alarms:

   a. ROD DRIVE CONTROL SYSTEM INOP.

   b. CONTROL ROD DRIVE FEED PUMP AUTO TRIP.

c. RPS MG TROUBLE.

d. MAIN CONDENSER VACUUM LO.

e. MAIN STEAM ISOL VLV NOT FULLY OPEN.

f. Possible RX AUTO SCRAM.

g. PRI CONT PRESS HI.

4. Control Room Panel 1H13-P603 (2H13-P603) Indications:

a. CRD Drive Water and Cooling Water Decreasing.

b. CRD Pump Amps Decreasing.

c. Reactor Pressure Increases then Decreases.

d. Reactor Water Level increases then Decreases.

e. Steam Flow Decreasing.

f. Turbine Driven Reactor Feed Pump Steam Flow Decreasing.

g. Motor Driven Reactor Feed Pump Amps Decreasing.

h. Rod In Lights Come On.

i. Decreasing Power.

5. Control Room Panel 1H13-P601 (2H13-P601) Alarms:

a. 4 KV BUS 143/143-1 (243/243-1) UNDERVOLTAGE.

b. DG 1B(2B) ENGINE RUNNING.

c. Other Possible ECCS Initiation Alarms.

6. Control Room Panel 1H13-P601 (2H13-P601) Indications:

a. Post Accident Monitor Reactor Pressure Increases then Decreases.

b. Post Accident Monitor Reactor Water Level Increases then Decreases.

c. Possible Start of Low Pressure ECCS Systems.

d. Auto Start of RCIC and HPCS.

7. Other Indications:

a. Reactor Recirculation Pumps Trip.

b. Reactor Water Cleanup Pumps Trip.

c. Possible Auto Start of Standby Gas Treatment.

d. Trip of Plant Ventilation and Plant Circulating Water Systems.

e. Turbine Generator Trip.

f. Loss of Feedwater, Condensate and Condensate Booster Systems.

g. Loss of Condensate Vacuum.

h. Containment Mimic Bus indicates Isolation.

i. Drywell Pressure Increase.

B. AUTOMATIC ACTIONS

1. Direct or Indirect Reactor SCRAM.

2. AUTO START of Emergency Diesel Generators and Load Transfer to the associated essential division:

a. Diesel Generator 0 - ESS Division 1.

b. Diesel Generator 1A(2A) - ESS Division 2.

c. Diesel Generator 1B(2B) - ESS Division 3.

3. Reactor Recirculation Pumps TRIP.

4. Main Turbine, Unit Aux. Transformer (UAT) and System Aux. Transformer (SAT) TRIP.

5. Reactor Feedpumps TRIP.

6. RCIC and HPCS AUTO START on Level 2.

7. AUTO START of Low Pressure Emergency Core Cooling sequence and system realignment if Reactor Water Level decreases to Level 1.

8. AUTO START of Emergency Bearing Oil Pump as Turbine Speed decreases.

9. All Reactor and Containment Isolation Valves CLOSE on Loss of Reactor Protection Buses.

C. IMMEDIATE OPERATOR ACTIONS

1. CHECK Reactor Water Level, Pressure and Containment Pressure response to ensure that a Loss of Coolant Accident has not occurred.

2. VERIFY Reactor SCRAMS, all Control Rods fully inserted and Reactor Power decreases with normal decay.

3. VERIFY AUTO START of Diesel Generators 0, 1A(2A), 1B(2B) and that ESS Divisions are energized from their respective Diesel Generators.

4. CHECK Reactor Water Level and Pressure:

   a. OPEN Safety-Relief Valves if necessary to decrease Reactor Pressure below 1076 psig.

   b. START RCIC or HPCS if necessary to maintain Reactor Water Level.

CAUTION

Alternate Safety-Relief Valve Operation to promote uniform Suppression Pool Cooling.

5. VERIFY Reactor Recirculation Pumps TRIPPED. PLACE A and B Reactor Recirculation Pump Breakers 3 in PULL to LOCK.

6. VERIFY that Standby Gas Treatment (SGTS) AUTO STARTS and that Reactor Building Ventilation has ISOLATED.

D. SUBSEQUENT OPERATOR ACTIONS

1. After Reactor Water Level is restored and it is VERIFIED that a Loss of Coolant Accident does not exist:

   a. STOP HPCS, LPCS and RHR A/B/C Pumps.

   b. START Suppression Pool Cooling in accordance with LOP-RH-13 if Suppression Pool Temperature approaches 100°F (T.S.) or if Drywell Pressure exceeds 4 psig. If high Containment Pressure exists, START Containment Spray to decrease the pressure.

2. VERIFY proper operation of the Diesel Generators.

3. VERIFY the AUTO START of the Emergency Bearing Oil Pump and the Turning Gear Oil Pump.

4. VERIFY that the Main Turbine and Generator have TRIPPED and that Generator Output Breakers 9-10 and 10-11 (2-3 and 3-4) OPEN.

5. VERIFY that the Turbine Driven Reactor Feed Pumps have TRIPPED.

6. If possible RESTORE Auxiliary Power from SAT 142(242) as follows:

   a. RE-ENERGIZE the System Auxiliary Transformer with applicable steps of LOP-AP-01.

   b. RESTORE Bus 141Y(241Y) Power from SAT 142(242) in accordance with LOP-AP-16.

   c. RESTORE Bus 142Y(242Y) Power from SAT 142(242) in accordance with LOP-AP-17.

   d. RESTORE Bus 143(243) Power from SAT 142(242) in accordance with LOP-AP-18.

7. PLACE the following Control Switches in PULL to LOCK in order to minimize the possiblity of high currents when restoring power distribution:

   a. Circulating Water Pumps 1A, 1B and 1C (2A, 2B and 2C).

   b. Heater Drain Pumps 1A, 1B, 1C and 1D (2A, 2B, 2C and 2D).

c. Service Water Pumps 1A, 1B (2A, 2B), Service Water Jockey Pump 0WS02PA (0WS02PB) and Service Water Pump "0" if Unit 2 Bus 241X was lost.

d. Condensate and Condensate Booster Pumps 1A, 1B, 1C, 1D (2A, 2B, 2C, 2D).

e. Motor Driven Reactor Feed Pump 1C (2C).

f. Electrode Boilers 0A, 0B (0C, 0D).

g. Inerting Steam Electrode Boiler 1A (2A).

h. Primary Containment Water Chiller 1A, 1B (2A, 2B).

8. RESTORE Station Air as follows:

a. If the System Aux. Transformer is energized:

1) ENERGIZE Switchgear 151 and 152 (251 and 252) in accordance with LOP-AP-01.

2) ENERGIZE Buses 131 A and B (231 A and B) and 132A and B (232A and B).

3) START one (1) Turbine Building Closed Cooling Water Pump in accordance with LOP-WT-02.

4) START Unit 1(2) Station Air Compressor in accordance with LOP-SA-02.

b. If the System Aux. Transformer can not be energized START the opposite Unit Station Air Compressor (or Common Air Compressor).

9. OBTAIN a sample from the Drywell in preparation for venting the containment (T.S.).

10. VENT the Primary Containment to clear the High Pressure Containment Isolation Signal.

11. RESET the following Emergency Core Cooling System Initiation Signals:

a. LPCS - RHR A.

b. RHR B-C.

c. ADS.

d. HPCS.

12. PLACE Emergency Core Cooling System Pumps in NORMAL unless the Pump is performing a Core or Containment Cooling Function. STARTUP Steam Condensing mode of RHR in accordance with LOP-RH-09 if required to control Reactor Temperature and Pressure.

13. STARTUP RPS in accordance with LOP-RP-01.

14. RESET the Scram.

15. RESET Reactor and Containment Isolation Signals.

16. START Primary Containment Ventilation Fans in accordance with LOP-VP-05.

17. ENERGIZE BUS 142X(242X) and START Service Water in accordance with LOP-WS-01. Using Service Water Jockey Pump OWS02PA (OWS02PB) maximize flow to the Reactor Building Closed Cooling Water and Fuel Pool Cooling Systems.

18. ENERGIZE Buses 133 and 134X and Y (233 and 234X and Y).

19. STARTUP Reactor Building Closed Cooling Water to supply the Containment in accordance with LOP-WR-02 using one RBCCW Pump.

20. STARTUP the Reactor Water Cleanup System in accordance with LOP-RT-02 in order to promote mixing in the Reactor and RESTORE reject capability.

21. STARTUP the Control Rod Drive Hydraulic System in accordance with LOP-RD-01.

22. ENERGIZE Bus 141X (241X).

23. STARTUP Primary Containment Cooling in accordance with LOP-VP-02. OPERATE the Chiller Unit that will equalize loading on Bus 141Y and 142Y. If absolutely necessary, SUPPLY Primary Containment Chill Water

with Reactor Building Closed Cooling Water in accordance with LOP-VP-09.

24. START an additional Service Water Pump if required.

25. STOP venting of the Primary Containment.

26. VERIFY that the Fuel Pool Cooling System is operating.

27. VERIFY AUTO START of the Main Turbine and Feed Pump Turbine Turning Gears.

28. TERMINATE Lake Blowdown and Radwaste Discharge.

29. If OFFSITE Power has not been restored, assess Unit Shutdown Power requirements.

    a. ENSURE an available supply of Diesel Oil is available.

    b. ENERGIZE 480 Volt AC Buses as necessary maintaining Breaker and Diesel Load Limitations.

30. CHECK Stack Gas Activity. OBTAIN a sample of Reactor Coolant.

31. COMPLETE LGP 3-2, Scram Recovery.

32. STARTUP Reactor Building Ventilation in accordance with LOP-VR-01 and SHUTDOWN Standby Gas Treatment in accordance with LOP-VG-02 after power is restored to the 6.9 KV - 480 Volt Distribution System.

33. If STARTUP electrical lineup is available and Limiting Condition for Operation will preclude a normal startup (T.S.), REFER to LGP 1-1; Normal Unit Startup or LGP 1-3; Unit Hot Standby to Power Operation as applicable.

E. DISCUSSION

Loss of the System Auxiliary Transformer with simultaneous loss of the Main Generator or a loss of all grid connections will cause a Loss of Auxiliary Electric Power. Initial automatic action is initiated by any combination of the following events: Turbine Generator Load Rejection, Loss of Condenser Vacuum and the Undervoltage trip of Reactor Protection Motor Generator Sets.

Action is directed toward ensuring that a Reactor Scram occurs, Auxiliary Power is transferred to the Diesel Generators, and that Reactor Pressure and Water Level remain under control. In addition, ensuring continuous oil flow to the Main Turbine and Reactor Feedpumps is included to prevent equipment damage.

The Steam Condensing Mode of RHR will be used to control Reactor Pressure and Water Level due to the loss of the Main Condenser.

Subsequent action is directed toward restoring power via the 4160 Volt and 6900 Volt distribution systems to reset the Isolation, depressurize the Drywell, reset the Scram, and minimize temperature stratification in the Reactor.

AREA HIGH RADIATION

A. SYMPTOMS

1. Control Room Panel 1H13-P601 (2H13-P601) Alarms:

   a. NEW FUEL STORAGE AREA RADIATION HI.

   b. REACTOR BLDG RADIATION HI.

   c. TURBINE BLDG RADIATION HI.

   d. AREA MONITORS RADIATION HI.

   e. REACTOR BLDG TIP ROOMS RADIATION HI.

   f. REFUELING FLOOR AREA RADIATION HI.

   g. FUEL POOL VENT RADIATION HI.

2. Control Room Panel 1N62-P600 (2N62-P600) Alarms:

   a. CARBON BED VAULT RAD HI.

   b. STATION VENT STACK RAD HI/HI-HI.

   c. BUILDING VENT EXHAUST RAD HI.

3. Other Symptoms.

   a. Any Area Rad Monitor Reading above Normal.

   b. Routine Survey indicates High Radiation.

   c. Radwaste Building High Radiation.

B. AUTOMATIC ACTIONS

1. None.

C. IMMEDIATE OPERATOR ACTIONS

1. As necessary.

   a. NOTIFY and EVACUATE Personnel.

b. RESTRICT access.

c. NOTIFY Shift Engineer or GSEP Station Director to classify the event and Initiate GSEP if required.

d. NOTIFY Rad/Chem to survey and sample.

D. SUBSEQUENT OPERATOR ACTIONS

1. LOCATE the source of Radiation using the following information:

a. CHECK Radiation Monitors for abnormal reading.

b. CHECK Stack Gas Release Rate.

c. CHECK Off-Gas Release Rate and Flow.

d. CHECK Area Temperatures and Leak Detection System Temperatures.

e. CHECK Continuous Air Monitors.

f. CHECK the Areas for Visible System Leakage or Loss of Shielding.

g. CHECK Areas with Portable Radiation Monitors.

2. STOP the cause of the Radiation. If the cause is a leak, ISOLATE the leak if possible by using manual Stop Valves, or by shutting down the affected system. Consider using temporary shielding if necessary.

3. Request Rad/Chem to SURVEY and ESTABLISH a Controlled Area. When a controlled area is established, personnel are no longer required to limit access.

4. ASSEMBLE PERSONNEL who may have been exposed and read their dosimeters. If necessary, restrict their further exposure and have film badges developed.

5. REFER to the following procedures if the cause of the Area Radiation is applicable:

a. Fuel Element Failure; LGA-16.

b. Fast Leak; LGA-01.

c. Slow Leak; LGA-02.

d. Major Steam Leak; LGA-03.

e. Load Reduction for Off-Gas Emergency; LOA-OG-01.

f. Operation of Control Room HVAC During Radiation, Smoke, or Chlorine Detection; LOA-VC-01.

g. Operation of Auxiliary Electric Equipment HVAC During High Radiation, Smoke or Chlorine Detection; LOA-VE-01.

h. Filling the Operating Reactor Well and Dryer Separator Pit; LOP-FC-09.

E. DISCUSSION

Abnormally high area radiation is an indication of many possible conditions. A local situation may cause high area radiation; system leaks and movement of contaminated waste are examples. High radiation in several areas is a symptom of major leaks, major equipment failure, and spreading contamination.

Action is taken to minimize personnel exposure, stop the source of the radiation, stop the spread of contamination and clean up the contamination.

HIGH AIRBORNE ACTIVITY

A. SYMPTOMS

1. Continuous Air Monitor alarm.

2. Portable Air Monitor sample above limits for an uncontrolled area.

3. Welding, flame cutting, grinding or heating of materials known to be contaminated.

4. High Area Radiation Monitor alarms.

B. AUTOMATIC ACTIONS

1. Refueling Floor Ventilation Exhaust or Reactor Building Ventilation Exhaust System High Radiation will Auto-Start the Standby Gas Treatment System and Isolate the Reactor Building Ventilation System.

C. IMMEDIATE OPERATOR ACTIONS

1. As necessary.

   a. NOTIFY and EVACUATE Personnel.

   b. RESTRICT access.

   c. NOTIFY Shift Engineer or GSEP Station Director to classify the event and initiate GSEP if required.

   d. NOTIFY Rad/Chem to survey and sample.

D. SUBSEQUENT OPERATOR ACTIONS

1. STOP the cause of the Radiation. If the cause is a leak, ISOLATE the leak if possible by using Manual Stop Valves or by SHUTTING DOWN the affected system.

2. Request Rad/Chem to SURVEY and ESTABLISH a Controlled Area. When a controlled area is established, personnel are no longer required to limit access.

3. If the Airborne Activity can be diluted or filtered without spreading, MAXIMIZE Ventilation flow to the affected area.

4. If the Stack Gas Release Rate is approaching Limits (T.S.) or if Airborne Contamination is spreading throughout a building, CONTAIN the Contamination. CONSIDER stopping Ventilation as follows:

   a. Turbine Building in accordance with LOP-VT-02.

   b. Off-Gas Building in accordance with LOP-VO-02.

   c. Radwaste Building in accordance with LOP-VW-03.

   d. Machine Shop Building in accordance with LOP-VJ-02.

   e. Aux. Building in accordance with LOP-VV-02.

5. If the contamination is in the Reactor Building, CONSIDER starting the Primary Containment Filter Train or Standby Gas Treatment System to ventilate the Reactor Building. VERIFY Reactor Building Ventilation System Isolation, if applicable.

6. If the Control Room or Aux Electric Room are Airborne, REFER to LOP-VC-05; Startup and Shutdown of Control Room HVAC Emergency Makeup Train.

E. DISCUSSION

Abnormal Airborne Activity is a symptom of Airborne Release of Particulate, Iodine, or Noble Gas Activity.

Action is taken to minimize personnel exposure, eliminate the source and spread of contamination and effect clean-up. Analysis of the sample will help to determine the source.

Since Airborne Contamination results in an internal dose, use of respirators and face masks may be necessary to minimize internal dose.

RADIOACTIVE SURFACE CONTAMINATION

A. SYMPTOMS

1. Routine survey or smear sample indicates surface contamination.

2. Radioactive water spill or leak.

3. Contaminated material inadvertently moved from a controlled area in violation of contamination control procedures.

4. Improper handling of radiological protective clothing.

5. Welding, flame cutting, grinding or heating of contaminated materials.

B. AUTOMATIC ACTIONS

1. None.

C. IMMEDIATE OPERATOR ACTIONS

1. As necessary.

   a. NOTIFY and EVACUATE Personnel.

   b. RESTRICT access.

   c. NOTIFY Shift Engineer or GSEP Station Director to classify the event and initiate GSEP if required.

   d. NOTIFY Rad/Chem to survey and sample.

D. SUBSEQUENT OPERATOR ACTIONS

1. Request Rad/Chem to SURVEY and ESTABLISH a Controlled Area. When a controlled area is established, personnel are no longer required to limit access.

2. DECONTAMINATE personnel who have been contaminated.

3. DECONTAMINATE the Area.

E. DISCUSSION

Radioactive Surface Contamination could be present due to small or large contamination releases. Spilling a sample bottle of contaminated liquid, welding or grinding on contaminated systems and leakage from certain systems are some probable causes.

Action is directed toward stopping the leak, warning personnel, isolating the cause and minimizing the spread. The person finding the contamination should determine the source of the contamination and eliminate the source of the contamination, if possible, e.g. uprighting a spilled sample bottle.

FUEL ELEMENT FAILURE

A. SYMPTOMS

1. Control Room Panel 1N62-P600(2N62-P600) Alarms:

   a. OFF GAS SYS OUTLET AND DRAIN ISOLATED.

   b. OFF GAS POST TREATMENT RAD HI.

   c. CARBON BED VAULT RAD HI.

   d. STATION VENT STACK RAD HI/HI-HI.

   e. BUILDING VENT EXHAUST RAD HI.

   f. OFF GAS SYSTEM WILL ISOLATE.

   g. TECH SPECS LIMIT WILL BE EXCEEDED.

2. Other Alarms:

   a. DRYWELL SUP CHAMBER RADIOACT HI.

   b. MAIN STEAM LINE RAD HI.

   c. Any Area or Ventilation Exhaust Radiation Monitor Alarms.

3. Other Symptoms:

   a. A significant increase in Off-Gas or Stack Gas Activity not attributable to plant evolutions.

   b. An increase in long-lived activity followed by a decrease in the ratio of short-lived activity to long-lived activity (Off-Gas grab sample).

   c. An increase in the Iodine isotopic content of the Reactor Water.

   d. An increase in the gross gamma activity of the Reactor Water.

   e. Increase in Dose Rate from the following plant areas:

1) Turbine Floor.

2) Feedwater Heater Area.

3) Steam Header and Moisture Separator Area.

4) Condensate Demineralizer Area.

5) Reactor Feed Pump Area.

6) Condensate Pump Area.

7) General Plant Background.

8) Primary Containment.

9) Steam Tunnel.

10) Reactor Building.

B. AUTOMATIC ACTIONS

1. Possible Main Steam High Radiation (3x normal) Actions.

a. Reactor SCRAM.

b. ISOLATE Vessel and Containment Isolation Groups 1 and 3.

2. Possible Off Gas Post Treatment High Radiation Actions.

a. Off Gas System Isolations.

1) Off Gas Discharge Valve

2) Off Gas Condenser Drain Valves

3) Hold Up Line

C. IMMEDIATE OPERATOR ACTIONS

1. If Main Steam High Radiation is reached, verify actions, do NOT continue this procedure, and refer to LGP 3-2, Reactor Scram.

2. If Main Steam High Radiation is NOT reached, verify Off Gas Charcoal Absorber Train is in service.

3. As necessary,

   a. NOTIFY and EVACUATE personnel.

   b. RESTRICT access.

   c. NOTIFY Shift Engineer or GSEP Station Director to classify and initiate GSEP if required.

   d. NOTIFY Rad/Chem to survey and sample.

D. SUBSEQUENT OPERATOR ACTIONS

1. OBSERVE the Continuous Air Monitors for trends.

2. CHECK local areas for increased Dose Rates during operation with defective fuel, such as the Turbine Area, the Reactor Building, the Reactor Water Cleanup Demineralizer Area, the Condensate Demineralizer Area, the Feedwater Heater Compartments, the Drywell and Turbine Building Atmosphere Sampling Stations, and the general background around the Plant Ventilation Stack.

3. INCREASE the frequency of analysis for Reactor water Chemistry during operation with defective fuel.

4. When shutting down and opening the Reactor Vessel with a known "leaker" present, special care should be taken to vent the Reactor Vessel Head correctly to minimize the release of fission gases during head removal. Any handling of suspect fuel elements should be done with caution. Reactor Water should be kept as cold as possible during the entire operation unless element sipping is to be done. The following items are methods to minimize radiation exposure and should be included in the normal shutdown for refueling sequence:

   a. SHUTDOWN the Reactor in the normal manner.

   b. PLACE the Primary Containment Vent and Purge System in service.

   c. MAINTAIN Reactor Water Temperature at or below 105°F for at least 24 hours before removing the Reactor Vessel Head.

CAUTION

Do not lower Reactor water Temperature below 90°F if the Reactor Vessel Head Bolts are under tension.

d. MONITOR the Reactor Building atmosphere continuously when opening the Reactor Vessel to atmosphere.

e. PROCEED with unbolting the Reactor Vessel Head if air sample activity remains less than the Maximum Permissible Concentration (MPC).

f. TAKE frequent air samples during unbolting of the Reactor Vessel Head.

g. REMOVE the Reactor Vessel Head using filter masks or independent air supplied masks for personnel protection as recommended by the Rad/Chem Department.

h. After sipping, REMOVE the suspect fuel and PLACE it in the special basket for transfer to the Fuel Storage Pool. On recommendation of the Rad/Chem Department, those elements identified as "gross leakers" should be placed in sealed containers and stored under water until shipped offsite.

E. DISCUSSION

A massive fuel element failure should not normally occur unless a severe condition which exceeds fuel strain limits or thermal limits occurs. Automatic action is initiated by the Reactor Protection and Reactor Containment Isolation Systems. Action is directed toward ensuring the appropriate automatic action and taking steps to prevent environmental release. Subsequent recovery includes assessing the damage, completing shutdown, and preparing a corrective maintenance plan.

A minor fuel element failure will normally be indicated by increasing coolant fission product activity and by increasing area and stack release radiation levels. Positive action is required to minimize the effects of the condition. Consideration should be given to reducing load, or to shutting down and replacing the leaky fuel. Routine operations should additionally be conducted so

as to minimize personnel exposure due to changing plant radiation levels.

The action following a COMPLETE CORE FAILURE IS NOT COVERED IN THIS PROCEDURE, since it is covered in GSEP.

BOMB THREAT RESPONSE

A. SYMPTOMS

1. Bomb threat received.

B. AUTOMATIC ACTIONS

1. None.

C. IMMEDIATE OPERATOR ACTIONS

1. If phone call:

   a. Do NOT hang up phone.

   b. Attempt to transfer to Shift Engineer.

2. Obtain and complete hard copy of checklist (Attachment A).

3. Notify Shift Engineer or GSEP Station Director to classify and initiate GSEP as necessary.

D. SUBSEQUENT OPERATOR ACTIONS

1. NOTIFY Security of any support required.

CAUTION

DO NOT touch or attempt t inspect or move a device suspected of being a bomb. Clear area and request Security to obtain demolition expert.

2. CONDUCT Bomb Search in accordance with Station Bomb Search Procedures.

   a. Each Department should be notified of the threat and possible locations of the device.

   b. The preferred action by each Department is to have personnel remain in their current location and review that area for any recent changes that may indicate the location of the device.

   c. Personnel should not be assembled in general areas.

d. The Station Director will evaluate the threat and determine if any evacuation is necessary.

E. DISCUSSION

Once a bomb threat has been received, the Shift Engineer must decide what actions are necessary to protect personnel and equipment based on his evaluation of the validity of the threat. The more information he has available to him the better his decisions and actions will be. For this reason transfer of a bomb threat phone call to him should be attempted so he can talk directly with the person making the threat.

The first decision the Shift Engineer must make is what level of validity to attach to the call. Next he must decide what verification of the call is necessary such as visual sighting of or search for the bomb device. Once the location has been determined, he must weigh the possible consequences of the device detonating and take actions to mitigate those consequences.

In addition he must decide what assistance is needed from outside sources. For example, the inspection and removal of a suspected bomb must only be done by experienced, qualified individuals.

LGA-17
Revision 2
July 11, 1980
3 (final)

ATTACHMENT A

BOMB THREAT CALL CHECKOFF LIST

1. Date_____, and Time..._____Number of rings 1,2,3
   Length_____

2. Exact wording of the threat:_____
   _____
   _____
   _____

3. Questions to ask of caller:
   a. When is bomb going to explode?_____
   b. Where is it right now?_____
   c. What does it look like?_____
   d. What kind of bomb is it?_____
   e. Did you place the bomb?_____
   f. Why?_____
   g. What is your name?_____
   h. What is your address?_____

4. Caller's: Sex____, Approx. Age____, & Race_____.

5. Caller's Voice:

CALM...._____	LAUGHING._____	LISP....._____	DISGUISED.._____
ANGRY.._____	CRYING.._____	RASPY....._____	ACCENT....._____
EXCITED_____	NORMAL.._____	DEEP....._____	FAMILIAR.._____
SLOW.._____	DISTINCT_____	RAGGED....._____	IF VOICE IS FAMILIAR,
RAPID._____	SLURRED._____	CLEARING THROAT_____	WHO DID IT SOUND LIKE?
SOFT._____	NASAL.._____	DEEP BREATHING_____	_____
LOUD._____	STUTTER_____	CRACKING VOICE_____	_____

Background Sounds:

STREET....._____	HOUSE...._____	CLEAR....._____.
CROCKERY..._____	MOTOR...._____	STATIC....._____
VOICES....._____	OFFICE..._____	LOCAL....._____
P.A. SYSTEM._____	FACTORY.._____	LONG DISTANCE._____
MUSIC....._____	ANIMAL..._____	BOOTH....._____

OTHER.....:_____

Threat Language:

WELL SPOKEN.._____	INCOHERENT....._____
FOUL....._____	TAPED....._____
IRRATIONAL._____	MESSAGE READ BY THREAT MAKER._____

5. Remarks:_____
   _____
   _____

6. Notify immediately the Shift Engineer (Ext. 202 or 203) or during regular daytime hours the Superintendent (Ext. 212).

7. Fill out completely: Name_____Position_____
   Date_____Phone Number_____

TRANSIENT WITH FAILURE TO SCRAM

A. SYMPTOMS

1. The following true signal(s) due to a Reactor transient are indicated by alarm or indication and the required full scram does not insert control rods as indicated on the full core display, rod position printout on the computer, or four rod display:

   a. Reactor Low Water Level 3 (12.5").

   b. High Drywell Pressure 1.69 psig.

   c. Scram Discharge Volume High Water Level (Scram Alarm).

   d. Main Steam Line High Radiation (later*).

   e. Turbine Stop Valve Closure.

   f. Control Valve Closure (Turbine Generator Load Reject).

   g. High Power Thermal Trip (Flow biased APRM).

   h. High Neutron Flux (120% in RUN, 15% in Startup).

   i. MSIV Closure (less than 90% open in Run).

   j. High Reactor Pressure above 1043 psig.

2. Reactor pressure and/or neutron flux indication increases abruptly, and may go off-scale on recorders and meters. This is the key indication to recognizing an ATWS event.

3. Safety-Relief Valves may Lift.

4. Other Indications:

   a. Increasing Drywell Pressure and Temperature.

   b. Increasing Suppression Pool Temperature.

   c. Possible increase in containment radiation levels.

d. Possible High Stack and Off-Gas release rates.

e. Recirculation Pumps trip with jet pump and core flow decreasing.

B. AUTOMATIC ACTIONS

1. 1076# Reactor Vessel Pressure and above actuates various safety relief valves, with possible low level setback (LLS).

2. 1120# Reactor Vessel Pressure initiates Reactor Recirculation Pump TRIP (ATWS).

3. Level 3, Reactor Vessel Water Level 12.5" Actions:

a. Reactor SCRAM signal.

b. TRIP Recirc Pumps to 15 Hz.

c. ADS Permissive.

d. ISOLATE vessel and containment groups 4, 5, 7.

e. SETBACK programmed level control to 18".

4. Level 2, Reactor Vessel Water Level -50" Actions:

a. AUTO START RCIC.

b. AUTO START HPCS and HPCS Diesel Generator.

c. TRIPS Recirc Pumps OFF (ATWS).

d. ISOLATE Vessel and Containment Groups 1, 2, 3, 5.

5. Level 1, Reactor Vessel Water Level -129" Actions:

a. AUTOSTART RHR LPCI Mode.

b. AUTOSTART LPCS.

c. AUTOSTART Diesel Generator 0 and 1A(2A).

d. ADS Permissive.

6. 1.69# Containment Pressure Actions:

   a. ISOLATE Vessel and Containment Groups 2, 4, 6, 7. Also Group 9 if 57# Reactor Pressure.

   b. Reactor SCRAM signal.

C. IMMEDIATE OPERATOR ACTIONS

1. Manually SCRAM Reactor.

   a. ARM and DEPRESS Manual pushbuttons.

   b. PLACE mode switch in Shutdown.

   c. If SCRAM, all rods in, and power decaying, go to LGP 3-2, Reactor Scram, and do NOT continue this procedure.

2. Within ONE minute of the start of the event, if no Scram:

   a. START BOTH SBLC Pumps.

   b. VERIFY RWCU ISOLATION.

3. VERIFY actions if reached.

   a. 1076# Safety/Relief Actuations.

   b. 1120# Recirc Pump Trip.

   c. Level 3.

   d. Level 2.

   e. Level 1.

   f. 1.69# Containment.

4. MAINTAIN Reactor Pressure and Level.

5. Within TEN minutes of the event:

   a. START RHR Service Water A & B Loops (LOP-RH-05).

   b. Place both RHR A & B in Suppression Pool Cooling Mode (LOP-RH-15).

D. SUBSEQUENT OPERATOR ACTIONS

1. VERIFY the following indications:

   a. Reactor power is decreasing.

   b. Reactor pressure is below 1076 psig.

   c. Reactor level is above Level 2 (-50"). CONTROL Reactor level with RCIC, HPCS, or Feedwater flow, if possible.

   d. Core flow decreases to natural circulation level, consistant with operating map.

   e. If containment pressure reaches or exceeds 30 psig, USE wetwell spray to CONTROL containment pressure.

   f. Both Standby Liquid Control Pumps are running and Standby Liquid Control Solution Tank level is decreasing.

CAUTION

1. De-energizing RPS busses will result in a loss of neutron monitoring instrumentation.

2. The following attempts to scram the Reactor are to be performed concurrently if manpower is sufficient.

2. CHECK Control Rod Position Indication. If the control rods have not fully inserted, PROCEED to scram the Reactor in the following order:

   a. DE-ENERGIZE RPS Subchannel Logic by opening breakers from control room panels 1H13-P609 and 1H13-P611 (2H13-P609 and 2H13-P611).

   b. TRIP RPS Scram Logic Breakers C8-2A and C8-2B at the RPS Distribution Bus in the Auxiliary Electric Room.

   c. OPEN the following RPS Power Supply Breakers for 2 minutes locally at the RPS Power Supply Buses and MG Set control panels and then RECLOSE:

1) RPS MG Set "A" output breaker.

2) RPS MG Set "B" output breaker.

3) CB-1, RPS Alternate Power Supply Breaker from MCC 1B25-1.

d. INDIVIDUALLY SCRAM Control Rods at Local Hydraulic Control Units (HCU's) by placing Branch Junction Module Switches to the Scram-Test position.

e. ISOLATE air from the scram air system by closing 1C11-F095 (2C22-F095); Scram Air Supply Valve.

f. MANUALLY INSERT Control Rods from Control Room Panel 1H13-P603 (2H13-P603) using the Reactor Manual Control System.

3. CHECK Stack Gas Release Rate and Off-Gas Release Rate. INITIATE GSEP if necessary.

4. AFTER the Reactor is shutdown to the level where the only source of power is decay heat, PROCEED to stabilize Plant Condition in HOT SHUTDOWN as follows:

CAUTION

Do not shutdown SBLC injection once it has been started until the SBLC Solution Tank is verified to be empty.

a. SAMPLE Reactor Coolant frequently to VERIFY Boron concentration is above the level determined to maintain the plant shutdown (above 750 ppm with all rods out).

b. PERFORM either step D.4.b.1 or D.4.b.2 as follows:

1) MAINTAIN Reactor at 1000 to 1050 psig by operating a Relief Valve and removing heat from the containment using Suppression Pool Cooling.

2) MAINTAIN Reactor at 1000 to 1050 psig as follows:

a) VERIFY that Boron concentration in the Reactor will be sufficient to maintain the Reactor shutdown after accounting for a normal startup of the Steam Condensing Mode of RHR.

b) STARTUP the Steam Condensing Mode of RHR in accordance with LOP-RH-07, Steam Condensing Startup and Operation.

5. When the Reactor is to be shutdown to COLD SHUTDOWN, PROCEED using the following considerations:

a. Sufficient negative reactivity has been inserted to the Reactor to account for the positive reactivity effects temperature and dilution.

NOTE

Carryover should not significantly affect Reactor Boron concentration.

b. If the plant is not contaminated and the Reactor is not isolated, a normal shutdown and cooldown in accordance with LGP 2-1 can be performed.

c. CAUTION must be taken that the unborated water in the RHR Shutdown Cooling lines does not temporarily dilute the boron in the core to allow criticality as follows:

1) ESTABLISH an excess boron concentration to accomodate for the effect of RHR dilution (30% excess is required above the 750-1000 PPM concentration).

2) STARTUP the Reactor Recirculation Pumps in SLOW speed in order to homogenize vessel Boron concentration. If the Primary Containment is isolated, the isolation signal must be reset in order to supply RBCCW to the Recirc. Pumps.

3) START RHR Shutdown Cooling flow to the Reactor Vessel gradually by throttling OPEN the Shutdown Cooling Injection Valve (The RHR Pump Minimum Flow Valve must be

overridden in the closed position to prevent the loss of borated water).

d. Do not exceed a 100°F/hr cooldown rate (TS).

e. When flooding the Reactor Vessel up to the steam dome, use a source of water borated to the same concentration as the water in the Reactor to prevent Reactor boron concentration dilution. The SBLC Solution Tank can be used. However, if using the SBLC Pumps, Reactor flooding will require one to two days. An alternate pump can be used.

f. Concentration levels of Boron in the Reactor Vessel will be 750-1000 ppm. The minimum solubility of Boron in water at 32°F is greater than 5000 ppm.

g. If a fuel element failure is suspected, refer to LGA-16, Fuel Element Failure.

E. DISCUSSION

An ATWS (Anticipated Transient without Scram) is extremely unlikely but will require prompt operator action to mitigate the consequences. Operator concerns are as follows: 1) VERIFY that Recirculation pumps trip, 2) Shutdown of Reactor, 3) Limit Reactor peak pressure 4) Maintain the core covered, 5) Limit the temperature of the Suppression Chamber, and 6) Long-term cooldown.

The operator must attempt to scram the Reactor with the most readily available means. Upon recognizing that the Reactor does not scram, the operator should INITIATE STANDBY LIQUID CONTROL (SBLC) WITHIN TWO MINUTES OF THE EVENT to minimize Reactor power production, which would heat-up the containment. HPCS or RCIC operation is necessary to maintain the core covered if feed flow is stopped, and should be initiated if Level 2 (-50 inches) is approached. Suppression Pool Cooling using two RHR Heat Exchangers must be initiated to ensure that Suppression Chamber temperature limits are not exceeded.

Subsequently, the operator must insert enough negative reactivity into the Reactor so that an uncontrolled restart will not occur. Thus, a cooldown must not be initiated until Control Rods are inserted or Boron concentration is determined satisfactory. The consequences

of this accident to the containment and environment must be evaluated. GSEP should be initiated if necessary.

Assuming 100% Reactor power and a two minute time delay to the start of Boron injection, Suppression Pool temperature will peak at 177°F after 28 minutes with the MSIV's closed or 105°F after two minutes with Bypass Valve capability. Containment pressure will peak at 3.5 psig with MSIV closure or .3 psig with Bypass Valve capability. Hot shutdown should be achieved within 18 minutes of Boron injection. However, once boron injection is started, it must be run to completion: DO NOT SHUTDOWN SBLC UNTIL POSITIVE VERIFICATION THAT THE SBLC SOLUTION TANK IS EMPTY.

It must be noted that FAILURE OF A MANUAL SCRAM WITHOUT AN ABNORMAL TRANSIENT REQUIRES THAT REACTOR RECIRCULATION PUMPS REMAIN OPERATING TO EXPEDITE BORON MIXING. The release limits of 10 CFR 100 apply to the ATWS event.

PROC. NO.			REV	DATE	DISKETTE
LOA	AA-01	DELETED	01	4/79	01
LOA	AA-02	OPERATION DURING TORNADO WARNING	00	11/78	01
LOA	AA-03	FAILURE OF THE COOLING LAKE DIKE	01 ~~00~~	11/30/79SL ~~11/78~~	01
LOA	AA-04	ACTION TO BE TAKEN IN THE EVENT OF AN OIL SPILL TO THE ILLINOIS RIVER OR ON THE ILLINOIS RIVER	01 ~~00~~	11/15/79SL ~~1/79~~	01
LOA	AA-05	PLANT OPERATION WITH THE CONTROL ROOM INACCESSIBLE	01 ~~00~~	2/1/80SL ~~11/78~~	01
LOA	AA-06	ACTION TO BE TAKEN IN THE EVENT OF AN OIL SPILL TO THE COOLING LAKE	01 ~~00~~	1/3/80SL ~~11/78~~	01
LOA	AP-01	LOSS OF SYSTEM AUXILIARY TRANSFORMER, SAT 142 (242), DURING POWER OPERATION	01 ~~00~~	1/8/0SL ~~4/79~~	02
LOA	AP-02	FAILURE OF BUS 141Y (241Y) OR BUS 142Y (242Y) TO TRANSFER TO UNIT AUXILIARY TRANSFORMER, UAT 141 (241) UPON LOSS OF POWER FROM SYSTEM AUXILIARY TRANSFORMER SAT 142 (242)	01 ~~00~~	1/8/0SL ~~4/79~~	02
LOA	AP-03	LOSS OF A 4 KV ESS BUS	01 ~~00~~	1/8/0SL ~~4/79~~	02
LOA	AP-04	LOSS OF A NON ESS BUS	01 ~~00~~	1/8/0SL ~~4/79~~	02
	AP-06	Actions On Transformer Trouble Alarm	00	2/80SL.	
LOA	AR-01	ACTION ON AN AREA RADIATION MONITOR ALARM	00	10/77	01
LOA	CD-01	LOSS OF CONDENSATE PUMP	01 ~~00~~	1/8/0SL ~~9/77~~	01
LOA	CG-01	LOSS OF CYCLED CONDENSATE GLAND WATER	01 ~~00~~	1/8/0SL ~~10/77~~	01
LOA	CP-01	OPERATION OF THE CONDENSATE POLISHING SYSTEM FOLLOWING AN EXCESSIVE CONDENSER TUBE LEAK	00	3/79	01
LOA	CP-02	CONTINUED OPERATION WITH EXCESSIVE CONDENSATE POLISHED DIFFERENTIAL PRESSURE	01 ~~00~~	1/8/0SL ~~3/79~~	01
LOA	CW-01	LOSS OF ALL CIRCULATING WATER PUMPS	01 ~~00~~	1/1/79SL ~~11/77~~	01
LOA	CW-02	ABNORMAL CONDENSER WATERBOX DIFFERENTIAL PRESSURE	01 ~~00~~	1/1/79SL ~~10/77~~	01
LOA	CW-03	CIRCULATING WATER HIGH INLET TEMPERATURE	01 ~~00~~	1/1/79SL ~~10/77~~	01
LOA	CY-01	HIGH CONDUCTIVITY IN CYCLED CONDENSATE STORAGE	00	11/77	01

6/12/79

LOA INDEX

PROC. NO.	TITLE	REV.	REV DATE	PAGE DISKETT
LOA DC-01	250 VDC SYSTEM FAILURE	01 ~~00~~	11/79SL ~~11/77~~	01
LOA DC-02	125 VDC SYSTEM FAILURE	00	11/77	01
LOA DC-03	48/24 VDC SYSTEM FAILURE	01 ~~00~~	11/80SL ~~10/77~~	01
LOA EH-01	~~MALFUNCTION OF THE PRESSURE CONTROL SYSTEM~~ Press. Reg. Failure upgrade	01 ~~00~~	11/80SL ~~4/79~~	02
LOA EH-02	TURBINE CONTROL VALVE FAILURE	01 ~~00~~	11/80SL ~~3/79~~	02
EH-03 rev'd upg.				
LOA FC-01	LOSS OF FUEL POOL COOLING	01 ~~00~~	11/80S ~~4/79~~	01
LOA FC-02	LOSS OF NORMAL LEVEL CONTROL IN THE FUEL POOL	01 ~~00~~	11/80S ~~5/79~~	03
LOA FW-01	LOSS OF FEEDWATER HEATERS	00	5/79	03
LOA GA-01	LOSS OF HYDROGEN COOLERS	01 ~~00~~	11/80SL ~~10/77~~	01
LOA GA-02	LOSS OF GEN H2 TEMPERATURE CONTROL	01 ~~00~~	11/80S ~~3/79~~	03
LOA GC-01	LOSS OF GENERATOR STATOR COOLING	SL ~~00~~ 01	6/79SL ~~3/79~~	03
GS-01 Steam Seal Evaporator Malfunct. R.O. 1/81				
LOA HD-01	LOSS OF PUMPED FORWARD FLOW HEATER DRAIN	01 ~~00~~	11/80SL ~~4/79~~	01
LOA HD-02	OPERATION WITH REDUCED PUMPED FORWARD HEATER DRAIN FLOW	01 ~~00~~	11/80SL ~~4/79~~	01
OA HD-03&05 (over)				
OA HS-01	LOSS OF MAIN HYDROGEN SEAL OIL PUMP	00	11/77	01
OA HY-01	LOSS OF GENERATOR HYDROGEN PURITY	00	5/79	00
OA IA-01	LOSS OF INSTRUMENT AIR	01 ~~00~~	11/80SL ~~3/79~~	02
OA IN-01	LOSS OF NORMAL DRYWELL PNEUMATIC AIR SUPPLY	01 ~~00~~	2/80SL ~~4/79~~	02
OA IN-02	LOSS OF 100# DYRWELL PNEUMATIC AIR SUPPLY	01 ~~00~~	2/80SL ~~5/79~~	03
OA MC-01	CLEAN CONDENSATE STORAGE TANK HIGH CONDUCTIVITY	00	11/77	01
OA NB-01	PRIMARY SYSTEM LEAKS	01 ~~00~~	3/80SL ~~4/79~~	03
OA NB-02	FAILURE OF A RELIEF VALVE TO SEAT PROPERLY Or Inadvertent Actuation of a Safety Relief Vlv	01 ~~00~~	5/80SL ~~5/79~~	01
OA NB-03	INADVERTANT ACTUATION OF A SAFETY RELIEF VALVE	~~00~~ 01	~~4/79~~ 13-80SL	03
OA NB-04	REACTOR COOLANT HIGH CONDUCTIVITY	~~00~~ 01	~~4/79~~ 1/80SL	03

EH-03 Rw.0 3/80 Turb. Cont. Valve(s) Failed Open
On Press. Reg. Failed Down Scale.

HD-03 Rw.0 1/80 Loss of H.P. Feed Wtr. Htr.
HD-04 Rw.0 1/80 Loss of A L.P. Htr. Strings
HD-05 Rw.0 1/80 Htr. Dr. Tnk. Level Hi/Low

PROC. NO.	TITLE	REV.	REV DATE	DISKETTE
LOA NB-05	RECOVERY FROM AN ECCS INITIATION UNDER POST ACCIDENT CONDITIONS	~~00~~ 01	~~4/79~~ 11/80 SL	03
LOA NR-01	SRM AND IRM INSERT OR WITHDRAW FAILURE	~~00~~ 01	~~4/79~~ 11/80 SL	03
LOA NR-02	LPRM FAILURE/LPRM HIGH FLUX OR LPRM DOWNSCALE	~~00~~ 01	~~4/79~~ 11/79 SL	03
LOA NR-03	LOSS OF NEUTRON FLUX INDICATION	~~00~~ 01	~~4/79~~ 10/79 SL	03
LOA OG-01	LOAD REDUCTION FOR OFF GAS EMERGENCY	~~00~~ 01	~~4/79~~ 11/80 SL	03
LOA OG-02	OFF GAS HYDROGEN EXPLOSION	00	4/79	03
LOA OG-03	ACTIONS TO BE TAKEN IN THE EVENT OF A FIRE IN THE OFF GAS CHARCOAL ADSORBER TRAINS	~~00~~ 01	~~4/79~~ 3/80 SL	02
LOA OG-04	LOSS OF ONE OR BOTH HYDROGEN ANALYZERS	00	3/79	03
LOA OG-05	FAILURE OF THE STEAM JET AIR EJECTOR STEAM PRESSURE CONTROL	00	~~0/00~~ 6/79 SL	03
PA-01	Misc. Auto Cont Sys. (LD PA06J, Pwr. Failure	00	6/80 SL.	
LOA PC-01	LOSS OF PRIMARY AND/OR SECONDARY CONTAINMENT INTEGRITY	00	3/79	03
LOA PR-01	RELEASE RATE SPIKES AFTER POWER CHANGE	00	11/77	01
LOA PR-02	RELEASE RATE EXPONENTIAL WITH POWER	00	10/77	01
LOA PR-03	HIGH RELEASE RATE	00	11/77	01
LOA RD-01	STUCK CONTROL ROD	~~00~~ 01	~~4/79~~ 11/80 SL	03
LOA RD-02	UNCOUPLED CONTROL ROD	00	5/79	03
LOA RD-03	MISPOSITIONED CONTROL ROD	~~00~~ 01	~~4/79~~ 11/80 SL	03
LOA RD-04	CONTROL ROD DRIVE SYSTEM FLOW CONTROL FAILURE	00	5/79	03
LOA RD-05	CONTROL ROD DRIVE STABILIZER	00	5/79	03
LOA RH-01	LOSS OF SHUTDOWN COOLING	00	3/79	02
LOA RH-02	LOSS OF SUPPRESSION POOL COOLING	~~00~~ 01	~~3/79~~ 11/80 SL	02
LOA RH-03	LOSS OF RHR SERVICE WATER	00	3/79	02
LOA RI-01	RCIC FAILS TO START ON AUTOMATIC INITIATION SIGNAL	00	10/77	01
LOA RI-02	RCIC FAILS TO PUMP WATER AFTER AUTOMATIC START	00	11/77	01

PROC. NO.	TITLE	REV.	REV DATE	DISKETT
LOA RL-01	FAILURE OF REACTOR WATER LEVEL CONTROL SYSTEM IN AUTO OR SINGLE	01 ~~00~~	1/80SL ~~3/79~~	03
LOA RL-02	FAILURE OF THE TORFP M/A XFR STATION	01 ~~00~~	1/80SL ~~3/79~~	03
LOA RR-01	AUTOMATIC TRANSFER OF REACTOR RECIRCULATION HYDRAULIC POWER UNIT FROM LEAD TO BACKUP SYSTEM	00	4/79	03
RR-03 (over.) RR-01 (over.) RT-01	Loss of Reactor Wtr. Cleanup (RWCU) SYS	00 01	1/80SL 1/80SL	
LOA SA-01	LOSS OF SERVICE AIR	~~00~~	~~3/79~~	03
LOA SC-01	INADVERTANT INJECTION OF BARON INTO THE REACTOR COOLANT SYSTEM DURING COLD SHUTDOWN	00	11/79SL ~~0/00~~	01
LOA SC-02	INITIATION OF STANDBY LIQUID CONTROL	00	7/79SL ~~0/00~~	02
LOA SH-01	LOSS OF STATION HEAT RECOVERY	01 ~~00~~	11/79SL ~~3/79~~	03
LOA TG-01	TURBINE HOOD SPRAY REGULATOR VALVE FAILURE	00	3/79	01
LOA TG-02	ABNORMAL VIBRATION OR NOISE FROM TURBINE GENERATOR	00	3/79	01
LOA TG-03	LOSS OF SHAFT GROUNDING	00	3/79	01
LOA TO-01	CONTINUED OPERATION WITH ABNORMAL BEARING OR BEARING OIL TEMPERATURE	01 ~~00~~	6/79SL ~~10/77~~	01
LOA TO-02	TURBINE LUBE OIL COOLER LEAK	00	4/78	02
LOA VC-01	OPERATION OF CONTROL ROOM HVAC DURING HI RADIATION, SMOKE OR CHLORINE DETECTION	00	4/79	03
LOA VE-01	OPERATION OF AUXILIARY ELECTRIC EQUIPMENT HVAC DURING HIGH RADIATION, SMOKE OR CHLORINE DETECTION	00	4/79	03
LOA VP-02	PRIMARY CONTAINMENT COOLING PRESSURE RELIEF AFTER HIGH DRYWELL PRESSURE, NON-LOCA	00	1/80SL ~~0/00~~	02
LOA WL-01	OPERATION OF RIVER SCREEN HOUSE WITH ONE TRAVELING SCREEN INOPERATIVE	01 ~~00~~	10/79SL ~~3/79~~	01
LOA WR-01	LOSS OF REACTOR BUILDING CLOSED COOLING WATER (RBCCW)	01 ~~00~~	10/79SL ~~4/79~~	01
LOA WS-01	LOSS OF SERVICE WATER	01 ~~00~~	1/80SL ~~6/78~~	01
LOA WT-01	LOSS OF TURBINE BUILDING CLOSED COOLING WATER (TBCCW)	01 ~~00~~	11/79SL ~~3/79~~	02

RR-03	Reactor Recirc. Flow Cont. Sys. Failure to Min. Demand	R.O	3/4/D S
RR-04	Recirc. Flow Cont. Vlv. Failure - Max Demand	R.O	3/4/D SL.

PROC. NO.	TITLE	REV.	REV DATE	DI KETTI
LOA WX-01	NO CAP IN DRUM	01 ~~00~~	10/79 SL ~~4/79~~	03
LOA WX-02	NO FILL SELECTION	01 ~~00~~	10/79 SL ~~4/79~~	03
LOA ZZ-01	OPERATION DURING EARTHQUAKE CONDITIONS	01 ~~00~~	10/79 SL ~~4/79~~	01 SL
ZZ-03	Failure of the Cooling Lake Dike	00	11/79	
ZZ-07	Piping Dikes	00	10/79 SL	
ZZ-04	Action to be taken In Event of An Oil Spill to the Ill River OR On the Ill. River	00	1/80 SL	
ZZ-05	Plant Oper. With The CR Inaccessible	00	8/80 SL	
ZZ-06	Action To Be Taken In the Event Of An Oil Spill To The Cooling Lake	00	1/80 SL	
ZZ-08	Irradiated Fuel Damage While Refueling (Fuel Handling Accident)	00	1/80 SL	