Form SPD-1002-1

DUKE POWER COMPANY
PROCEDURE PREPARATION
PROCESS RECORD

(1) ID No: AP/1/A/5500/01
Change(s) 0 to
0 Incorporated

(2) STATION: McGuire Nuclear

(3) PROCEDURE TITLE: Reactor Trip

(4) PREPARED BY: Bill Reeside DATE: 6-28-82

(5) REVIEWED BY: Len Fulcher DATE: 6/28/82

Cross-Disciplinary Review By: ______ N/R: LTS

(6) TEMPORARY APPROVAL (IF NECESSARY):

By: ______ (SRO) Date: ______

By: ______ Date: ______

(7) APPROVED BY: [signature] Date: 6-28-82

(8) MISCELLANEOUS:

Reviewed/Approved By: ______ Date: ______

Reviewed/Approved By: ______ Date: ______

Date/Initial

Verified with Control Copy ____/____

DUKE POWER COMPANY
McGUIRE NUCLEAR STATION
REACTOR TRIP

1.0 Symptoms

1.1 Any Alarm on Reactor Trip First Out Panel.

1.2 All rod bottom lights are illuminated.

1.3 Nuclear Instrumentation indicating a rapid decrease in Neutron flux.

2.0 Immediate Action

2.1 Automatic

2.1.1 All rods drop into core.

2.1.2 Turbine-Generator trip.

2. .3 Feedwater Isolation when Tavg decreases to 564°F.

2.1.4 Steam Dumps Arm-Actuate and/or Main Steam PORV lift.

2.1.5 CA Pumps start and feed all S/G's if 2/4 Lo-Lo level exist in 1/4 S/G's.

2.2 Manual

NOTE

If Reactor Power does not decrease rapidly and control rods are not inserted, or turbine fails to trip, this is an "Anticipated Transient Without Scram" event.

2.2.1 If the reactor or turbine fails to trip when required, proceed to AP/0/A/5500/34 (*Actions Required for an Anticipated Transient Without Scram Event*) and perform applicable steps concurrent with subsequent steps in this procedure.

2.2.2 Verify Feedwater Isolation when Tavg decreases to ≤564°F.

2.2.3 Verify Steam Dumps arm-actuate and/or Main Steam PORV's operate ≥1125 psig.

2.2.4 Secure all boron dilution operations.

2.2.5 Verify CA Pump start and feed all Steam Generators if 2/4 Lo-Lo level exists in 1/4 S/G's.

3.0 Subsequent Action

3.1 Verify all required Immediate Actions have occurred.

NOTE

AP/1/A/5500/02 (Turbine Generator Trip) should be run concurrent with this procedure if applicable.

3.2 If any pressurizer PORV's open on high pressurizer pressure, ensure reseating at 2315 psig decreasing.

NOTE

If PORV fails to close and pressure is less than 2315 psig, close the associated PORV isolation valves.

3.3 Ensure the CA System flow to Steam Generators. If not, manually start the motor driven CA Pumps.

NOTE

If the CA Pumps receive an auto-start signal depress the Train "A" and Train "B" CA Modulating Valves Resets in order to regulate flow to the steam generators.

3.4 Select "Reset" on the Moisture Separator Reheater Panel and prepare the MSR's for a hot start or shutdown per OP/1/B/6250/11 (Moisture Separator Reheater).

3.5 Verify no-load PZR level (25%) and Pressure (2235 psig) are restored and maintained. Verify charging and letdown flow normal.

3.6 Verify Tave is maintained $\geq$557°F and <562°F.

3.7 Verify Steam Generator levels are in the narrow range and S/G pressures are ~1090 psig.

3.8 Announce occurrence over plant paging system.

3.9 Note the cause of the trip on the first out panel before resetting the alarm.

3.10 If all rods are not fully inserted, borate 150 ppm for each rod not inserted per OP/1/A/6150/09 (Boron Concentration Control).

3.11 Transfer NR-45 to one source range channel and one intermediate range channel for indication. Ensure a negative period and decaying count rate.

CAUTION

Ensure Primary and Secondary Systems have stabilized before transferring Steam Dump Controller to Pressure Mode.

3.12 Place or verify the Steam Dump M/A Station in AUTO. Transfer Steam Dump Controller to PRESSURE MODE. Verify steam dumps operate to control steam pressure at approximately 1092 psig.

3.13 Verify Volume Control Tank level is being maintained.

3.14 Reset Hi Flux at Shutdown Alarm when neutron flux decreases below setpoint.

3.15 Notify Chemistry to obtain a NC System boron sample. Perform a reactivity balance calculation and maintain a shutdown margin equal to or greater than 1.6% $\Delta k/k$-per OP/0/A/6100/06 (Reactivity Balance Calculation).

3.16 Notify Plant Manager or Superintendent of Operations per Station Directive 3.1.6(Notifying Management of Operating Conditions).

3.17 Notify NRC Operation Center by ENS phone within one hour as described in Station Directive 3.1.4 (Conduct of Operations).

3.18 Place boilers in operation per OP/1/B/6250/07B (Electric Boilers) as necessary.

3.18.1 Close 1AS-9 (C Htr. Bleed to AS). As 1AS-120 (Aux. Elec. Blr. A & B to AS Isol.) is opened slowly, throttle close 1AS-12 (SM to AS).

3.19 Shutdown the MG sets per OP/1/A/6150/08 (Rod Control).

3.20 Reset the "Negative Rate Trip" bistables on the power range drawers.

3.21 Close the reactor trip breakers.

3.22 Take manual control and close the following valves:

1CF-32 (A S/G CF Cntrl. Vlv.)
1CF-23 (B S/G CF Cntrl. Vlv.)
1CF-20 (C S/G CF Cntrl. Vlv.)
1CF-17 (D S/G CF Cntrl. Vlv.)
1CF-104 (A S/G CF Cntrl. Vlv. Bypass)
1CF-105 (B S/G CF Cntrl. Vlv. Bypass)
1CF-106 (C S/G CF Cntrl. Vlv. Bypass)
1CF-107 (D S/G CF Cntrl. Vlv. Bypass)

3.23 Reset Train A & B CF Isolation.

3.24 Open the following valves:

1CF-126-B (A S/G CF to CA Nozzle Isol.)
1CF-127-B (B S/G CF to CA Nozzle Isol.)
1CF-128-B (C S/G CF to CA Nozzle Isol.)
1CF-129-B (D S/G CF to CA Nozzle Isol.)

3.25 Start a Feedwater Pump per OP/1/A/6250/01 (Condensate and Feedwater System) and secure Auxiliary Feedwater per OP/1/A/6250/02 (Auxiliary Feedwater Syste when desired and maintained steam generator levels at no load value.

3.26 If thermal power output was >15% at time of reactor trip, notify primary chemistry to perform isotopic analysis for iodine in accordance with Tech Spec 4.4.9. Note that this sample must be taken and analyzed no sooner than 2 hours after the trip and no later than 6 hours after the trip.

3.27 Notify HP to perform required radioactive gaseous waste sampling in accordance with Tech. Spec. 4.11.2.1.2.

3.28 Notify Projects and Licensing Engineer to contact site NRC inspector and inform of trip. If P&L engineer cannot be reached then notify site NRC inspector of trip.

3.29 Determine the cause of the reactor trip and correct the problem. If restart is desired, proceed to OP/1/A/6100/05 (Reactor Trip Recovery).

NOTE

If Reactor Trip occurred during startup and <15% power, restart may commence per OP/1/A/6100/01 (Controlling Procedure for Unit Startup).

3.30 If shutdown is necessary, proceed to OP/1/A/6100/02 (Controlling Procedure for Unit Shutdown).

DUKE POWER COMPANY
NUCLEAR SAFETY EVALUATION CHECK LIST

(1) STATION: MG-2-0119 UNIT: 1 ______ 2 ✓ ______ 3 ______
OTHER: ______

(2) CHECK LIST APPLICABLE TO: (NSM) MG-2-0119 (Same MG-1-1161)
☐ C/E Div. ☐ M&N Div. ☑ Elect. Div. ☑ All Divs.

(3) SAFETY EVALUATION – PART A

The item to which this evaluation is applicable represents:

(RE)* Yes ✓ No ______ A change to the station or procedures as described in the FSAR?
(SRAL) Yes ______ No ✓ A test or experiment not described in the FSAR?

If the answer to the above is "Yes", attach a detailed description of the item being evaluated and an identification of the affected section(s) of the FSAR. SEE NSM FOR DETAILED DESCRIPTION.
CHAPTER 7 OF FSAR IS AFFECTED.

(4) SAFETY EVALUATION – PART B (SRAL)

Yes ______ No ✓ Will this item require a change to the station Technical Specifications?

If the answer to the above is "Yes", identify the specification(s) affected and/or attach the applicable page(s) with the change(s) indicated.

(5) SAFETY EVALUATION – PART C (SRAL)

As a result of the item to which this evaluation is applicable:

Yes ______ No ✓ Will the probability of an accident previously evaluated in the FSAR be increased?
Yes ______ No ✓ Will the consequences of an accident previously evaluated in the FSAR be increased?
Yes ______ No ✓ May the possibility of an accident which is different than any already evaluated in the FSAR be created?
Yes ______ No ✓ Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
Yes ______ No ✓ Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
Yes ______ No ✓ May the possibility of malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
Yes ______ No ✓ Will the margin of safety as defined in the bases to any technical Specification be reduced?

If the answer to any of the preceding is "Yes", an unreviewed safety question is involved. Justify the conclusion that an unreviewed safety question is or is not involved. Attach additional pages as necessary.

(6) PREPARED BY (SRAL): R. D. Carroll DATE: 3/28/83

(7) REVIEWED BY (SRAL): G. D. Rowland DATE: 3/28/83

(8) REVIEWED BY (RE): J. E. Snipes DATE: 3-25-83
Ernest E Estep SRC 3-29-83

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*RE = Responsible Engineer
SRAL = Safety Review, Analysis & Licensing Division

ATTACHMENT

In the case of the modification to add the capacity for automatic actuation of the reactor trip breaker by means of the shunt trip mechanism, the following logic was applied to each of the items in section (5) of the Nuclear Safety Evaluation Check List (Attached).

1. Will the probability of an accident previously evaluated in the FSAR be increased?

   No. The shunt trip mechanism was already in place in the manual trip circuit and malfunction is no more probable now than previously. Addition of the automatic trip by means of the shunt trip will not interfere with, degrade, or replace automatic reactor trip by means of the undervoltage trip mechanism. For the above reasons, no previously evaluated accident is more probable.

2. Will the consequences of an accident previously evaluated in the FSAR be increased?

   No. There will be no degradation of the reactor trip mechanism or its associated systems as a result of this modification. Implementation of this modification will enhance the reliability of the automatic reactor trip function.

3. May the possibility of an accident which is different than any already evaluated in the FSAR be created?

   No. This modification enables automatic actuation of the shunt trip. The shunt trip was already in place and in use as a manual trip for the reactor trip breakers. Enabling automatic initiation of both the undervoltage and shunt trips will increase the reliability of the reactor trip function and will not degrade the reactor trip system.

4. Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

   No. As stated previously, the shunt trip was already in place and is no more likely to fail now than before. Interference of the shunt trip mechanism with the undervoltage trip mechanism is not considered credible. The circuitry enabling receipt of the automatic trip signal by the shunt trip mechanism was designed to the high standards of the ESF system.

5. Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR by increased?

   No. As stated previously, there will be no degradation of the reactor trip mechanism or its associated systems as a result of this modification. Implementation of this modification will enhance the reliability of the automatic reactor trip function.

6. May the possibility of malfunction of equipment important to safety different than any already evaluated in the FSAR be created?

   No. As stated previously, the only new malfunction mechanism would be

interference of the shunt trip mechanism with the undervoltage trip mechanism. This is not considered credible.

7. Will the margin of safety as defined in the based to any Technical Specification be reduced?

   No. The reactor will trip at least as reliably and quickly as without this modification.

The addition of the capability for automatic actuation of the reactor trip breaker by means of the shunt trip mechanism will enhance the reliability of the reactor trip function. All trip mechanism problems to date have occurred in the undervoltage trip mechanism. Although enabling automatic initiation of the shunt trip is not an alternative to the undervoltage trip or a fix for the undervoltage trip problems, implementation of this modification provides additional assurance that the reactor will trip when an automatic trip signal is generated. The shunt trip mechanism is highly reliable and implementation of this modification has increased Duke Power's confidence in the ability of the reactor trip system to trip the reactor on demand.

NOTE: This package was developed to provide documentation of the logic applied in determining that an unreviewed safety question does not exist due to the modification enabling automatic trip of the reactor trip breaker by means of the shunt trip mechanism.