UNION ELECTRIC

1901 Gratiot Street, St. Louis

November 27, 1985

Mr. Hugh L. Thompson, Jr.
Director, Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

ULNRC-1215

Dear Mr. Thompson:

CALLAWAY PLANT, DOCKET NUMBER 50-483
RESPONSE TO GENERIC LETTER 85-12
IMPLEMENTATION OF THE TMI ACTION ITEM II.K.3.5
"AUTOMATIC TRIP OF REACTOR COOLANT PUMPS"

Union Electric Company received the subject NRC Generic Letter 85-12 in July, 1985. The Generic Letter finds acceptable the Westinghouse Owner's Group generic response to the concerns of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps", and requests additional information to complete the Callaway Plant specific review. The Generic Letter requested that licensees furnish the plant specific information listed in Sections A, B, and C of the Generic Letter attachment entitled, IV IMPLEMENTATION.

This submittal provides the requested information. The responses to Section A and B address those items common to the SNUPPS plants. The response to Section C is Callaway site specific.

If additional information is required, please let us know.

Very truly yours,

Donald F. Schnell

Donald F. Schnell

DJW/ljr

Enclosure

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STATE OF MISSOURI)
)   S S
CITY OF ST. LOUIS)

Donald F. Schnell, of lawful age, being first duly sworn upon oath says that he is Vice President-Nuclear and an officer of Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Donald F. Schnell
Donald F. Schnell
Vice President
Nuclear

SUBSCRIBED and sworn to before me this 27th day of November, 1985.

Barbara J. Pfaff
BARBARA J. PFAFF
NOTARY PUBLIC, STATE OF MISSOURI
MY COMMISSION EXPIRES APRIL 22, 1989
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RESPONSE CONCERNING IMPLEMENTATION OF THE REACTOR COOLANT PUMP (RCP) TRIP CRITERIA

A. Determination of RCP Trip Criteria

1. NRC Request

Identify the instrumentation to be used to determine the RCP trip setpoint, including the degree of redundancy of each parameter signal needed for the criterion chosen.

SNUPPS Response to A.1

In reference 4, SNUPPS notified the NRC that reactor coolant system (RCS) pressure has been chosen as the trip parameter for the SNUPPS plants. There are three wide-range pressure indicators that are available to the operator, two that receive their signals from the nuclear incore instrumentation guide tubes at the seal table (BBPI 403 and 405), and one that receives its signal from the top of the reactor vessel (BBPI 406). These redundant, Class 1E transmitters are located outside of the containment. Each of the transmitters is associated with a different Separation Group of SNUPPS plant instrumentation (Ref. FSAR Section 7.1). The design features of this pressure instrumentation (consistent with the function, location, and environmental conditions) have been reviewed and availability is adequately assured for accident mitigation.

2. NRC Request

Identify the instrumentation uncertainties for both normal and adverse containment parameters. Describe the basis for the selection of the adverse containment parameters. Address, as appropriate, local conditions such as fluid jets or pipe whip which might influence the instrumentation reliability.

SNUPPS Response to A.2

Instrumentation uncertainty for wide-range pressure indicators PI-403, PI-405, and PI-406 was determined using the statistical methodology, as described in reference 3, previously approved by the NRC. An operator reading error of one-half the smallest instrument scale dimension was statistically included. The resulting

error for normal environmental conditions is 3.5%, which gives an instrument uncertainty of 105 psi.

The RCS wide-range pressure transmitters in the SNUPPS plants are supplied by Westinghouse and are located outside containment in areas where they will not be adversely affected by accident conditions inside containment other than radiation, or accident conditions outside containment other than a potential local auxiliary steam line break which could adversely affect one of the three transmitters. However, implementation of the RCP trip criteria is not required for an auxiliary steam line break event. The total integrated radiation dose for a six-month post-LOCA period in the vicinity of these transmitters does not exceed the Westinghouse threshold criteria for a harsh radiation environment. Therefore, the instrument uncertainty for normal environmental conditions applies for all cases when the RCP trip criteria may be invoked.

3. NRC Request

In addressing the selection of the criterion, consideration to uncertainties associated with the WOG supplied analyses values must be provided. These uncertainties include both uncertainties in the computer program results and uncertainties resulting from plant-specific features not representative of the generic data group.

If a licensee determines that the WOG alternative criteria are marginal for preventing unneeded RCP trip, it is recommended that a more discriminating plant-specific procedure be developed. For example, use of the NRC-required inadequate-core-cooling instrumentation may be useful to indicate the need for RCP trip. Licensees should take credit for all equipment (instrumentation) available to the operators for which the licensee has sufficient confidence that it will be operable during the expected conditions.

SNUPPS Response to A.3

The LOFTRAN Computer code was used to perform the alternate RCP trip criteria analyses. Both Steam Generator Tube Rupture (SGTR) and non-LOCA events were simulated in these analyses. Results from the SGTR analyses were used to obtain all of the trip parameters. LOFTRAN is a Westinghouse licensed code used for FSAR

SGTR and non-LOCA analyses. The code has been validated against the January, 1982 SGTR event at the Ginna plant. Results of this validation show that LOFTRAN can accurately predict RCS pressure, RCS temperatures and secondary pressures, especially in the first ten minutes of the transient. This is the critical time period when minimum pressure and subcooling is experienced.

The major causes of uncertainties and conservatism in the computer program results, assuming no changes in the initial plant conditions (i.e., full power, pressurizer level, all SI and AFW pumps run) are due to either calculated models or inputs to LOFTRAN. The following are considered to have the most impact on the determination of the RCP trip criteria:

1. Break flow
2. SI flow
3. Decay heat
4. Auxiliary feedwater flow

The following sections provide an evaluation of the uncertainties associated with each of these items.

To conservatively simulate a double ended tube rupture in safety analyses, the break flow model used in LOFTRAN includes a substantial amount of conservatism (i.e, predicts higher break flow than actually expected). Westinghouse has performed analyses and developed a more realistic break flow model that has been validated against the Ginna SGTR tube rupture data. The break flow model used in the W̲OG analyses has been shown to be approximately 30% conservative when the effect of the higher predicted break flow is compared to the more realistic model. The consequence of the higher predicted break flow is a lower than expected predicted minimum pressure.

The SI flow inputs used were derived from best estimate calculations, assuming all SI trains operating. An evaluation of the calculational methodology shows that these inputs have a maximum uncertainty of ± 10%.

The decay heat model used in the W̲OG analyses was based on the 1971 ANS 5.1 standard. When compared with the more recent 1979 ANS 5.1 decay heat inputs, the values used in the W̲OG analyses are higher by about 5%. To determine the effect of the uncertainty due to the decay heat model, a sensitivity study was conducted for the

SGTR. The results of this study show that a 20% decrease in decay heat resulted in only a 1% decrease in RCS pressure for the first 10 minutes of the transient. Since RCS temperature is controlled by the steam dump, it is not affected by the decay heat model uncertainty.

The AFW flow rate input used in the WOG analyses are best estimate values, assuming that all auxiliary feed pumps are running, minimum pump start delay, and no throttling. To evaluate the uncertainties with AFW flow rate, a sensitivity study was performed. Results from the two loop plant study show that a 64% increase in AFW flow resulted in only an 8% decrease in minimum RCS pressure. Results from the 3 loop plant study show that a 27% increase in AFW flow resulted in only a 3% decrease in minimum RCS pressure.

The effects of all these uncertainties in the models and input parameters were evaluated, and it was concluded that the contributions from the break flow conservatism and the SI uncertainty dominate. The calculated overall uncertainty in the WOG analysis for the SNUPPS plants, as a result of these considerations, is a -150 psig to +150 psig for the minimum RCS pressure RCP trip setpoint. Due to the minimal effects from the decay heat model and AFW input uncertainties on the RCS pressure uncertainty, the calculational uncertainty result includes only the effects of the uncertainties due to the break flow model and SI flow inputs.

There are no uncertainties resulting from plant-specific features not representative of the generic data group. RCP trip on RCS pressure has been selected as the appropriate trip parameter. RCS pressure provides ample margin (in excess of 300 psig) to the trip setpoint for the non-LOCA accidents that were evaluated in the WOG analyses.

B. Potential Reactor Coolant Pump Problems

1. NRC Request

Assure that containment isolation, including inadvertent isolation, will not cause problems if it occurs for non-LOCA transients and accidents.

a. Demonstrate that, if water services needed for RCP operations are terminated, they can be restored fast

enough once a non-LOCA situation is confirmed to prevent seal damage or failure.

b. Confirm that containment isolation with continued pump operation will not lead to seal or pump damage or failure.

SNUPPS Response to B.1.a

The automatic (Hi-3) containment isolation signal isolates component cooling water to and from each RCP motor and thermal barrier. Automatic (Hi-1) or manual containment isolation isolates RCP seal water return to the Chemical and volume Control System (CVCS) but does not isolate seal water injection to the RCP. Although the RCP seal system can operate for some time with seal water injection only, the RCP motor bearings are more limiting and are qualified for 10 minutes operation without component cooling water with no resultant damage (reference 10). Ten minutes provides adequate time for the operator to determine either that a non-LOCA accident has occured and to restore component cooling water flow to the RCP or to determine that a LOCA has occured and to trip the RCPs.

SNUPPS Response to B.1.b

Containment isolation (Hi-3) does not isolate seal water injection, but does isolate component cooling water to the reactor coolant pump and seal water return to the CVCS. The limiting components associated with the RCP under these conditions are the motor bearings which are qualified for 10 minutes operation without resultant damage. As described in the response to B.1.a., 10 minutes provides adequate time to restore component cooling water flow or to trip the RCPs as appropriate.

2. NRC Request

Identify the components available to trip the RCPs, including relays, power supplies, and breakers. Assure that RCP trip, when determined to be necessary, will occur. If necessary, as a result of the location of any critical component, include the effects of adverse containment conditions on RCP trip reliability. Describe the basis for the adverse containment parameters selected.

SNUPPS Response to B.2

The components available to manually trip the RCPs are listed on Attachment A.

To trip one RCP, three active devices must function: the hand-switch on the MCB; the trip coil mechanism; and the breaker. The Control and Turbine Buildings are not subject to harsh environmental conditions resulting from a LOCA in the containment.

There are no components required for the function of tripping the RCPs located inside containment. The only equipment inside containment associated with interrupting power to the RCP motors are the RCP motors, electrical power cable to the motors, the in-containment portion of the electrical penetration assemblies carrying power to the pump motors, the differential current transformers (on the power cables) and associated electrical cable and electrical penetration assemblies for the differential relay current circuits. It is unlikely that adverse environmental conditions could affect the above equipment prior to initiation of a manual trip; however, degradation of either power cables or differential relay current circuits would most likely result in an RCP trip signal generated by the differential relay. Once the RCP breakers are open, the RCPs receive no power and further degradation of the equipment inside containment will not result in the breakers reclosing.

Adverse containment parameters assumed in this evaluation are those post-LOCA conditions of temperature, pressure, humidity, radiation, chemical spray, and potential submergence provided in the SNUPPS NUREG-0588 Submittal (reference 5).

C. Operator Training and Procedures (RCP Trip)

1. NRC Request

Describe the operator training program for RCP trip. Include the general philosophy regarding the need to trip pumps versus the desire to keep pumps running.

CALLAWAY Response to C.1

The Callaway Training Department provides in-depth training on Plant Emergency procedures and their bases to students in initial License Training and Licensed

Operator Requalification Training. Part of their training includes a detailed study of Reactor Coolant Pump trip criteria. Subjects discussed in this area are as follows:

- Large break versus small break LOCA concerns

  This section includes information on the effects of large break and small break LOCA's, time frames for several site LOCA's, and the possibility of core uncovery.

- Integrated mass loss versus time of RCP trip

  This section includes information on the effects of RCS mass loss with RCP's running during LOCA's.

- Continuous RCP operation

  This section includes information on the effects of RCS voiding.

- Bases for RCP trip criteria

  The bases for RCP trip criteria are discussed in detail. Information on this topic is derived from the Westinghouse Owner's Group analyses of RCP trip criteria.

- Applicability of RCP trip steps

  The applicability of RCP trip steps during controlled cooldowns and other plant evolutions is discussed.

The bases for the Callaway Plant Emergency procedures RCP trip criteria is discussed in detail in both initial training and retraining. The significance of running RCP's during Loss of Coolant Accidents is discussed, with the effects of RCP operation during small break Loss of Coolant Accidents emphasized. Voiding in the Reactor Coolant System during pump on and pump off conditions is reviewed in the Callaway Mitigation Core Damage course.

In addition to this training, the bases for specific steps in all major Emergency Procedures are discussed. These steps include RCP trip criteria for the various procedures. Classroom training on these topics is reinforced through the use of simulator exercises.

Overall, approximately 30 hours of classroom material is presented on these topics.

The philosophy regarding RCP trip criteria presented to students during this training is consistent with that presented in the Westinghouse Owner's Group Background Documents for the Emergency Procedures. Students are trained to evaluate plant conditions during emergencies and follow the guidance provided in the Plant Emergency Procedures.

While the desirability of maintaining RCS forced flow for specific accidents (i.e., S/G Tube Ruptures) is discussed, the training reinforces the concept that actions performed during emergency conditions shall be in accordance with plant procedures.

2. NRC Request

Identify those procedures which include RCP trip related operations:

(a) RCP trip using WOG alternate criteria
(b) RCP restart
(c) Decay heat removal by natural circulation
(d) Primary system void removal
(e) Use of steam generators with and without RCPs operating
(f) RCP trip for other reasons

Callaway Response to C.2

The responses to this section reference numerous Operation Department procedures. Therefore, Attachment B provides a listing of applicable procedures and their titles for easy reference.

(2.a) The procedures identified below generically direct operators to trip all Reactor Coolant Pumps (RCP) when both of the following conditions exist: a) Charging pumps or Safety Injection pumps - AT LEAST ONE RUNNING and b) Reactor Coolant System (RCS) pressure is less than 1400 psig. This is an incorporation of the WOG alternate criteria.

E-0	ES-0.4
ES-0.0	E-1
ES-0.1	E-3
ES-0.2	ECA-2.1

(2.b) Callaway procedures incorporate the following generic instructions and cautions for restarting the reactor coolant pumps:

Cautions

- After any attempt to start where the motor has failed to achieve full speed before it is stopped, a restart should not be attempted until the motor has been allowed to cool by standing idle for a period of not less than thirty minutes.

- Two successive RCP starts are permitted, provided the motor is allowed to coast to a stop between starts.

- A third RCP start may be made when the winding and the core have cooled by running for a period of 20 minutes or by standing idle for a period of 45 minutes.

- When three starts or attempted starts have been made within a two-hour period, then a fourth start should not be made until the motor has been allowed to cool, by standing idle for at least one hour.

- Start only one RCP at a time.

Instructions

- Ensure the following:

  a. RCS pressure > 325 PSIG
  b. 13.8 kV buses energized (PA01 and PA02)
  c. #1 seal dP > 200 PSID
  d. VCT pressure $\geq$ 15 PSIG
  e. CVCS in operation with seal injection 6-13 gpm per pump
  f. #1 seal leak off $\geq$ 0.2 GPM
  g. CCW in operation, supplying cooling water to the following:

     (1) Thermal barrier heat exchangers
     (2) Upper and lower bearing oil coolers
     (3) Motor air coolers

  h. All annunciator alarms clear for RCP operation

- Start the RCP Oil Lift Pump

  NOTE A pressure interlock prevents starting the RCP unless a minimum oil pressure of 700 PSIG is available to the motor thrust bearing oil lift system. This interlock is satisfied when the white light on the respective oil lift pump control switch is lit.

- After the Oil Lift Pump has been running for two minutes, start the RCP.

- After the RCP has been running for one minute, stop the Oil Lift Pump.

Subsequent steps include directions to go to other steps of the procedure if either an RCP is running or if an RCP cannot be started. The following list of procedures are identified as containing the above instructions and cautions:

ES-0.1	ECA-3.1
ES-0.2	ECA-3.2
ES-0.4	ECA-3.3
ES-1.1	FR-C.1
ES-1.2	FR-I.3
E-3	FR-P.1
ECA-2.1	

In addition, one additional procedure, OTN-BB-00003, also contains the above instructions and cautions while providing detail on equipment lineups, required valve positions, and stopping the RCPs at designated loads.

(2.c) The following procedures include RCP trip related operations for decay heat removal by natural circulation:

ES-0.1	ECA-2.1
ES-0.2	ECA-3.1
ES-0.4	ECA-3.2
ES-1.1	ECA-3.3
ES-1.2	FR-P.1
E-3	

Each of these procedures has a checklist with which to verify Natural Circulation. Checklist Items include:

- RCS subcooling - MORE SUBCOOLED THAN INSTRUMENT ERROR

  Two methods for determination of subcooling are provided. One method is the RCS SUBCOOLING METER ERROR CORRECTION calculation. The other is use of RCS SUBCOOLING CURVES.

- Steam pressure - STABLE.

- RCS hot leg temperature - STABLE OR SLOWLY DECREASING.

- RCS cold leg temperature - NEAR SATURATION TEMPERATURE FOR STEAM PRESSURE.

NOTE (1) Approximately 20 minutes will be required to establish stable natural circulation conditions.

(2) Hot leg temperatures are expected to initially respond by increasing to 575 deg. F and then stabilizing.

- Core exit thermocouples (TC) - STABLE OR SLOWLY DECREASING.

- IF natural circulation is verified, THEN return to the appropriate step of the procedure.

- IF natural circulation is NOT verified, THEN increase dumping steam and verify natural circulation from trended values.

Specifically, Procedure ES-0.2 provides actions to perform a natural circulation RCS cooldown and depressurization to cold shutdown, with no accident in progress under requirements that will preclude any upper head void formation. This procedure is entered from ES-0.1 and ECA-0.1 when it has been determined that a natural circulation cooldown is required.

Specifically, Procedure ES-0.4 provides actions to continue plant cooldown and depressurization to cold shutdown, with no accident in progress, under conditions that allow for the potential formation of a void in the upper head region without a vessel level system available to monitor void growth. This procedure is entered from ES-0.2.

Specifically, Procedure ECA-0.1 provides the following list of steps to ensure natural circulation, and if the proper response is not obtained, further procedural steps increase the dumping of steam from intact steam generators.

- RCS subcooling - PRESSURE AND TEMPERATURE WITHIN PERMISSIBLE RANGE. Use RCS subcooling curves to ensure adequate subcooling. For normal containment, use core exit TC's. For adverse containment, use RCS wide range RTD's.

- Steam Generator (S/G) pressures - STABLE OR DECREASING.

- RCS hot leg temperatures - STABLE OR DECREASING.

- Core exit TC's - STABLE OR DECREASING.

- RCS cold leg temperatures - AT SATURATION TEMPERATURE FOR S/G PRESSURE.

The procedures OTN-BB-00003, OTG-ZZ-00001 and OTG-ZZ-00006 address natural circulation in the PRECAUTIONS AND LIMITATIONS SECTION. They recommend if one RCP cannot be started, that natural circulation using the steam dump to condenser or atmosphere be utilized and verified via the following indications:

- RCS subcooling based on core exit TC's - GREATER THAN 11 DEG. F.

- S/G pressures - STABLE OR DECREASING.

- RCS hot leg temperatures - STABLE OR DECREASING.

- RCS cold leg temperatures - AT SATURATION TEMPERATURE for S/G PRESSURES.

- Core outlet temperatures - AT LEAST 10 DEG. F. BELOW SATURATION TEMPERATURE.

(2.d) Primary system void removal, as related to RCP trip operations, is accomplished in the procedures listed below. These procedures are entered from procedure CSF-1.

ES-0.4
FR-I.3

Description of ES-0.4 was given in the response to 2.c.

FR-I.3 provides actions to respond to voids in the reactor vessel and is entered from CSF-1 when the pressurizer level is at or above normal and the reactor vessel is less than full.

CSF-1 is a procedure that provides two functions: 1) provides general surveillance under all sets of unusual or abnormal conditions that can lead to or result from initiation of safety injection, 2) and it directs operator guidance in those rare events that go beyond the design basis of the Engineered Safeguards Systems and the Emergency Operating Procedures and Emergency Contingency Actions.

(2.e) No specific procedures address use of the S/G's with the RCP's operating. When the RCP's are not operating, natural circulation is required and is described in ES-0.2 and ES-0.4. In both cases, S/G parameters are used to verify natural circulation.

(2.f) The following discussion addresses RCP trip procedures for other reasons. The following procedures generically direct operators to trip any RCP if component cooling water to that pump is lost to the RCP motor for greater than 2 minutes or if the upper or lower bearing temperatures reach 195 deg. F:

E-0	ES-0.2
ES-0.0	ES-0.4
ES-0.1	ECA-2.1

The next set of procedures generically directs operators to trip the affected RCP's when the number one seal differential pressure is less than 215 PSID OR if the number one seal leakoff flow is less than 1.25 GPM:

ES-1.2	ECA-1.1
ES-3.1	ECA-3.1
ES-3.2	ECA-3.2
ES-3.3	ECA-3.3

The next three procedures are entered from procedure CSF-1 and are described separately since the RCP trip criteria differs for each.

- FR-C.1 - In response to a degraded core, if at least two RCS hot leg temperatures are less than 350 deg. F, the operators are directed to trip all RCP's since they are no longer needed for core cooling. Also, this procedure has operators trip all RCP's upon anticipation of the loss of number one seal requirements with which further operation would damage the RCP's. Loss of the seal is anticipated with depressurization of the S/G's to atmospheric pressure.

- FR-C.2 - One step in the procedure directs operators to trip all RCP's upon anticipated loss of number one seal requirements and a later step directs them to trip the pumps after SI flow has been established and verified to provide core cooling.

- FR-H.1 - This procedure has operators trip all RCP's in order to extend the time to restore feed flow to the S/G's since RCP operation results in heat addition to the RCS water.

Procedure, OTN-BB-00001, contains RCP trip related criteria in the section titled "Dynamic Venting of the RCS." Within this section each pump is required to run for two minutes after reaching full speed or until seal DP reaches 200 PSID and is then stopped.

The procedure directs operators to trip the affected pump if RCP seal differential pressure decreases rapidly or approaches 200 PSID.

Procedure OTN-BB-00003 directs operators to trip the affected RCP immediately if vibration exceeds 5 MILS.

Finally Procedure, OTO-BB-00002, describes symptoms, probable causes, and the operator actions required for RCP off-normal conditions. It directs operators to immediately trip the affected RCP if any of the following conditions exist:

- If vibration exceeds 5 MILS on the frame, 20 MILS on the shaft.

- If on a loss of seal injection, any of the following occur:

  a. Number one seal outlet temp. > 235 deg. F.
  b. Seal injection temp. > 150 deg. F.
  c. Thermal barrier cooling water inlet temp. > 105 deg. F.
  d. Pump lower bearing > 225 deg. F.
  e. Motor bearing cooling water temp. > 120 deg. F.
  f. Motor bearing temp. > 195 deg. F.
  g. If reactor power < 48% and a loss of CCW to one RCP motor exists for $\geq$ 2 minutes or if upper or lower bearing temperature reaches 195 deg. F.

- If reactor power is > 48% and a loss of CCN to one RCP motor exists for > 2 minutes, or if uppor or lower bearing temperature reaches 195 deg. F., the operator is directed to trip the reactor and turbine, then trip the affected RCP(s).

- It further directs operators to reduce power to < 48% and then trip the affected RCP(s) if the seal ΔP is < 200 PSID or if frame vibration is above 3 MILS and increasing at a rate of 2 MILS/HR or if

shaft vibration is above 15 MILS and increasing at a rate of 1 MIL/HR.

Subsequent operator actions give additional guidance for the following off-normal conditions:

a. Number one seal - Leak-off flow high

b. Number one seal - Leak-off flow low

c. Number two seal - Leak-off flow high

d. Number three seal - Standpipe High/Low Level

e. Loss of Seal Injection

f. Loss of Component Cooling Water

References

1. NRC Generic Letter 85-12, dated 6/28/85

2. SLNRC 83-0021, dated 4/22/83, Response to NRC Generic Letter No. 83-10c.

3. SLNRC 84-0050, dated 3/23/84, Response to NRC Questions on Setpoint Methodology for SNUPPS.

4. SLNRC 84-66, dated 4/13/84, Final Response to NRC Generic Letter No. 83-10c.

5. SNUPPS Report of Independent Review of Environmental Qualification Programs to NUREG-0588, SLNRC 83-15, dated 3/10/83 as revised by SLNRC 83-30, dated 5/27/83 and SLNRC 84-13, dated 2/1/84.

6. WOG Emergency Response Guidelines Executive Volume, dated 3/21/84.

7. WOG Evaluation of Computer Code Uncertainties, dated 8/16/85.

8. The following Bechtel Drawing No.'s:

   E-03PA02 Rev. 12
   E-03PA05 Rev. 11
   E-03BB01 Rev. 13
   E-03PA14 Rev. 8
   E-01PK01 Rev. 13
   M-02BB02(Q) Rev. 18
   M-02BB04(Q) Rev. 6
   M-0G063-07

9. SNUPPS FSAR Table 3.11(B)-3.

10. SNUPPS FSAR Section 5.4.1.

11. BLUE 1985, dated 9/5/85, Circuits Penetrating Containment Excluded from T/S Table 3.8-1.

12. Instruction Manual for 13.8 kV Switchgear, E-009-0223-05, E-01001, Rev. 7.

Enclosure
ULNRC -1215
November 27, 1985

Attachment A

Equipment Available to Manually Trip

Reactor Coolant Pumps

Equipment	Location/Room #
A. Main Control Board RL021	Control Building/ 3601
1. Hand Indicating Switch (one per pump)	
B. Switchgear (PA01, PA02)	Turbine Building/ 2033 ft. el. NW
1. Breakers (one per pump)	
2. Trip Coil (one per pump) (energize-to-trip)	
3. Fuses (2) for breaker control power	
4. Breaker "a" contacts (two per breaker)	
5. Wiring/terminal blocks	
C. 125 VDC Distribution Panel (PK41, PK62) (control power to breakers)	Turbine Building/ 2033 ft. el. NW
1. Switches (#3, #4)	
2. Wiring	
D. 125 VDC Switchgear (PK01, PK02)	Turbine Building/ 2033 ft. el. NW
1. Fuses, wiring	
E. 125 VDC Batteries (PK11, PK12) (power to 125 VDC switchgear)	Turbine Building/ 2033 ft. el. NW

ATTACHMENT B

LIST OF APPLICABLE PROCEDURES

AND THEIR TITLES

Emergency Response Procedures

E-0: Reactor Trip or Safety Injection

- ES-0.0: Rediagnosis
- ES-0.1: Reactor Trip Response
- ES-0.2: Natural Circulation Cooldown
- ES-0.4: Natural Circulation Cooldown With Steam Void in Vessel(Without RVLIS)

E-1: Loss of Reactor or Secondary Coolant

- ES-1.1: SI Termination
- ES-1.2: Post LOCA Cooldown and Depressurization

E-3: Steam Generator Tube Rupture

- ES-3.1: Post-SGTR Cooldown Using Backfill
- ES-3.2: Post-SGTR Cooldown Using Blowdown
- ES-3.3: Post-SGTR Cooldown Using Steam Dump

ECA-0.1: Loss of All AC Power Recovery Without SI Required

ECA-1.1: Loss of Emergency Coolant Recirculation

ECA-2.1: Uncontrolled Depressurization of All Steam Generators

ECA-3.1: SGTR With Loss of Reactor Coolant-Subcooled Recovery Desired

ECA-3.2: SGTR With Loss of Reactor Coolant-Saturated Recovery Desired

ECA-3.3: SGTR Without Pressurizer Pressure Control

Function Restoration Procedures

FR-C.1: Response to Inadequate Core Cooling

FR-C.2: Response to Degraded Core Cooling

FR-H.1: Response to Loss of Secondary Heat Sink

FR-I.3: Response to Voids in Reactor Vessel

FR-P.1: Response to Imminent Pressurized Thermal Shock Condition

CSF-1: Critical Safety Function Status Trees

Other Procedures

OTG-ZZ-00001: Plant Heatup Cold Shutdown to Hot Standby
OTG-ZZ-00006: Plant Cooldown Hot Standby to Cold Shutdown
OTO-BB-00002: Reactor Coolant Pump Off-Normal
OTN-BB-00001: Reactor Coolant System
OTN-BB-00003: Reactor Coolant Pumps