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VPNPD-85-541
NRC-85-127

December 4, 1985

Mr. H. R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Attention: Mr. Edward Butcher, Acting Chief
Operating Reactors, Branch No. 3

Gentlemen:

DOCKET NOS. 50-266 AND 50-301
RESPONSE TO GENERIC LETTER NO. 85-12
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Generic Letter No. 85-12, dated June 28, 1985, pertains to the implementation of TMI Action Plan Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps". The Action Plan item requires an automatic trip of the reactor coolant pumps (RCP's) to minimize the loss of system inventory during a small-break loss-of-coolant accident (SBLOCA). However, continued operation of the RCP's is a major aid to operators during steam generator tube rupture (SGTR) and non-LOCA events. The NRC recognized that automatic trip of the RCP's may not be the most desirable solution and suggested that other solutions be considered.

In report SECY-82-475, "Staff Resolution of the Reactor Coolant Pump Issue", dated November 30, 1982, the NRC provided guidelines and criteria for resolution of TMI Action Plan Item II.K.3.5. The NRC concluded that industry could develop a RCP trip setpoint that would alert the operators to trip the RCP's for a SBLOCA, but would not require RCP trip for those events where forced circulation and pressurizer pressure control are a major aid to the operators. Resolution of the issue is intended to ensure that the mode of RCP operation selected by the licensee has a sound technical basis, meets the NRC's rules and regulations, and keeps the RCP's running, if possible, for SGTR and non-LOCA events.

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The Westinghouse Owners Group (WOG) submitted two reports to the NRC describing a generic method for selecting a setpoint for manual RCP trip. The information provided by the WOG for justification of manual RCP trip has been accepted by the NRC on a generic basis. Generic Letter No. 85-12 states that a suitable RCP trip criterion can be selected by each licensee to minimize RCP trip during SGTR and non-LOCA events, while providing for RCP trip during a SBLOCA. The NRC requests that Westinghouse licensees use the WOG methodology to select and implement an appropriate RCP trip criterion.

In accordance with the WOG methodology, Wisconsin Electric has selected subcooling of the reactor coolant system as the parameter to be used to determine when a manual trip of the RCP's should be done. The calculated RCP trip setpoints are included in the upgraded emergency operating procedures which were implemented at the Point Beach Nuclear Plant on July 1, 1985.

Generic Letter No. 85-12 also cites the need for plant-specific information which has not been provided under the WOG generic program. An outline of the plant-specific information requested is contained in Section IV, Implementation, of the Safety Evaluation Report (SER) attached to Generic Letter No. 85-12. The objective of this submittal is to respond to each of the items identified in the SER outline. The attachment to this letter is a restatement of each item in the SER outline and our response to that item.

Please contact us if you have any questions regarding the plant-specific information we have provided.

Very truly yours,

C W Fay

C. W. Fay
Vice President
Nuclear Power

Attachment

Copy to NRC Resident Inspector

12/04/85

Attachment

RESPONSE TO GENERIC LETTER NO. 85-12
REQUEST FOR PLANT-SPECIFIC
REACTOR COOLANT PUMP TRIP INFORMATION
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

A. Determination of RCP Trip Criteria

A.1 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.

RESPONSE

Two completely redundant, qualified RCS subcooling monitors are used to determine the RCP trip setpoint. The subcooling monitor requires input from the RCS pressure and temperature instruments to calculate the degree of subcooling. Pressure input is obtained from a RCS wide-range pressure transmitter. Temperature input is obtained from either the core exit thermocouples or the resistance temperature detector (RTD) located in the hot leg associated with the pressure transmitter. The Emergency Operating Procedures (EOP's) specify the use of the subcooling setpoint based on the temperature measured by the core exit thermocouples. A switch is available to select the RTD input if necessary.

A set of four core exit thermocouple signals are averaged to provide input to one subcooling monitor. Four different thermocouple signals are averaged to provide input to the second subcooling monitor. There is one thermocouple from each of the four quadrants of the reactor core in each set of four thermocouples.

Table 1 lists the tag numbers and a description of the instrumentation used to determine the RCP trip setpoint.

A.2 Identify the instrumentation uncertainties for both normal and adverse containment conditions. 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.

RESPONSE

Instrumentation uncertainties for normal and adverse containment conditions are identified in Table 2.

Criteria for transition from normal to adverse containment conditions are selected so that the environmental uncertainty for normal containment conditions is small, but allows conditions resulting from less severe accidents, including SBLOCA, to be addressed within the normal containment conditions classification. A temperature of more than 180°F or a radiation exposure rate of more than 10^5 R/HR are the bounding values used for qualification of equipment as described in Wyle Laboratories Test Report No. 45592-4 entitled, "Foxboro N-E10 Series Pressure Transmitters". The bounding values of pressure, temperature, and radiation exposure exceed the worst conditions possible for a credible accident at Point Beach. Using the most adverse containment conditions to calculate instrument uncertainty in any adverse conditions ensures that the calculated uncertainty will be larger than the actual uncertainty.

Wisconsin Electric has not identified a local condition that might influence the reliability of the subcooling monitor. The pressure transmitters are located outside of the primary shield wall and are, therefore, protected from pipe whip, fluid jets, and missiles. Core exit thermocouples are located in the upper plenum of the reactor vessel. The thermocouples are designed to function in the fluid conditions expected in the upper plenum which are more harsh than the most severe containment conditions. There are no pipes or missiles in the upper plenum that could damage the core exit thermocouples. The thermocouple connectors and extension cables outside of the reactor vessel are not subject to pipe whip, fluid jets, or missiles. The RTD's could be lost due to pipe whip or a fluid jet, but the RTD's are not normally used with the subcooling monitor and no more than one channel would be affected at one time.

A.3 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.

RESPONSE

The LOFTRAN computer code was used to perform the alternate RCP trip criteria analyses. 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. The

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 determined.

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 safety injection (SI), and auxiliary feedwater (AFW) pumps running), are due to either 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. AFW 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 substantial conservatism (i.e., predicts higher break flow than actually expected). Westinghouse has developed a more realistic break flow model that has been validated against the Ginna SGTR data. The more realistic break flow model shows that the flow model used in the WOG analyses is approximately 30% too high. The consequence of the higher predicted break flow is a predicted minimum pressure which is lower than expected.

The SI flow is derived from best estimate calculations, assuming all SI trains operating. An evaluation of the methodology shows that the SI flow calculation has a maximum uncertainty of ±10%.

The decay heat model used in the WOG analyses is 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 WOG analyses are about 5% higher. To determine the effect of the uncertainty due to the decay heat model, a sensitivity study was conducted for 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 ten 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 inputs 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 three-loop plant study show that a 27% increase in AFW flow resulted in only a 3% decrease in minimum RCS pressure, a 2% decrease in minimum RCS subcooling, and a 2% decrease in pressure differential.

The effects of all these uncertainties have been evaluated and the conclusion is that the contributions from the break flow conservatism and the SI uncertainty dominate. Uncertainty in the decay heat model and the AFW flow have a minimal effect on the RCP setpoint and are not included in the overall uncertainty. The calculated overall uncertainty in the WOG analyses as a result of these considerations for Point Beach is +3 to +10°F for the minimum RCS subcooling RCP trip setpoint.

Some features of the Point Beach Nuclear Plant are different than the features used in the generic analysis. The largest difference is in the RCS pressure. Point Beach is capable of operating at either 2250 or 2000 PSIA. The generic analysis used operating conditions close to the higher pressure operating conditions of Point Beach. Sensitivity studies show that a lower operating pressure does not have a significant effect on the transient response of the plant.

The WOG also considered differences in SI flow, AFW flow, core power, and fuel type. When a difference existed, the WOG selected the value that would lead to the most limiting result or performed additional analysis to evaluate the effect of the difference. The combination of worst features of all plants in the generic group adds conservatism to the results. The conservatism is reflected in the one sided, positive uncertainty of the WOG analysis for the RCP trip setpoint at Point Beach.

B. Potential Reactor Coolant Pump Problems

B.1 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.

RESPONSE

A containment isolation signal at Point Beach does not terminate component cooling water or seal injection for RCP operations. Therefore, containment isolation does not cause a problem if it occurs for non-LOCA transients and accidents. The RCP's are able to continue operation after containment isolation without increased danger of damage or failure of the pump or seal.

B.2 Identify the components required to trip the RCP's, 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.

RESPONSE

RCP trip is accomplished by energizing the trip coil of the 4160 volt breaker which feeds the pump. Several actions will cause this coil to energize and, therefore, trip the RCP.

1. Manual operation of a control board switch by an operator.

2. Operation of the breaker lockout relay. This relay is in turn energized by individual overcurrent relays on each phase of the supply to the reactor coolant pumps.

3. Operation of one of two underfrequency relays which monitor bus frequency on the bus supplying the RCP coupled with operation of one of two similar relays for the companion bus. The companion bus is that which feeds the other reactor coolant pump and has the same power source.

4. Operation of an auxiliary relay which in turn is energized when a bus section lockout relay picks up or when both undervoltage relays on the bus section supplying the RCP drop out.

Power for the trip coil of the breakers which supply the RCP's and any of the associated relays mentioned above is from an uninterrupted DC source with battery backup. This power can be switched to a similar alternate source by operation of a switch at the switchgear itself.

None of the components required to trip the RCP's are located in containment or in an area expected to experience a harsh environment. In addition, RCP trip is not required for a steam line break outside of containment as documented in Chapter 14 of the Point Beach FSAR. Therefore, reliability of the RCP trip components when required for SBLOCA is not affected by adverse conditions.

If manual RCP trip cannot be accomplished by the operator using the RCP control switch, then the non-safeguards 4160 Volt bus supplying the RCP's could be deenergized by manually opening the feeder breakers from the control room. As a last resort, the RCP breakers or the bus feeder breakers could be locally opened at the switchgear in the turbine hall. Therefore, manual RCP trip can always be accomplished when required.

C. Operator Training and Procedures (RCP Trip)

C.1 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.

RESPONSE

Point Beach Nuclear Plant operators regularly participate in training programs related to RCP trip. Over the past four years four separate training programs related to RCP trip have been conducted for Point Beach operators. The training programs listed below ensure that the operators are aware of the RCP trip criteria and the rationale for tripping the RCP's versus the desire to keep the RCP's running.

1. "Mitigating Core Damage", September 1981. A course taught by Westinghouse for nuclear plant operators included a discussion of RCP operation.

2. "Emergency Operating Procedures", course TRCR-31, 1984. All licensed operators completed this course on EOP's, including RCP criteria.

3. "Licensed Operator Requalification", July 28 through September 7, 1985. Part of scheduled operator requalification training concentrated on the background document for the new EOP's, including the trip criteria and the reason for tripping or not tripping the RCP.

4. "Plant Operations Review", course TRCR-30, September 8 through October 19, 1985. Reinforces training on EOP's by using the procedures on a simulator. An assessment period and discussion covered the reasons for RCP trip criteria and the consequences of tripping or not tripping the RCP.

C.2 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 RCP's operating.
f. RCP trip for other reasons.

RESPONSE

Table 3 identifies the emergency operating procedures (EOP's) which include RCP trip-related operations. The table is divided into six parts corresponding to the six items listed in the NRC request. The EOP's used at Point Beach are based on the WOG emergency response guidelines.

Table 1

INSTRUMENTATION USED TO DETERMINE THE RCP TRIP SETPOINT

Description of Instrument	TAG NUMBERS Unit 1 Yellow Power Source	Unit 1 White Power Source	Unit 2 Yellow Power Source	Unit 2 White Power Source
RCS Subcooling Monitor	1TM971	1TM970	2TM971	2TM970
Pressure Transmitter	1PT-420B	1PT-420A	2PT-420B	2PT-420A
Analog Module (I/V) N-2AI-I2V	1PQ-420B	1PQ-420A	2PQ-420B	2PQ-420A
Function Generator (F(x)) N-2AP+SGC	1TM-971-6	1TM-970-6	2TM-971-6	2TM-970-6
Hot Leg RTD	1TE-451D	1TE-450D	2TE-451D	2TE-450D
Analog Module (R/V) N-2AI-P2V	1TM-451D	1TM-450D	2TM-451D	2TM-450D
Core Exit Thermocouples				
Quadrant I	1TE-23	1TE-25	2TE-23	2TE-25
Quadrant II	1TE-21	1TE-26	2TE-21	2TE-26
Quadrant III	1TE-31	1TE-36	2TE-31	2TE-36
Quadrant IV	1TE-10	1TE-34	2TE-10	2TE-13
Analog Modules (mV/V) N-2AI-T2V	1TM-23	1TM-25	2TM-23	2TM-25
	1TM-21	1TM-26	2TM-21	2TM-26
	1TM-31	1TM-36	2TM-31	2TM-36
	1TM-10	1TM-34	2TM-10	2TM-13
Summation Analog Module N-2AP+SUM	1LM-495-9	1LM-494-9	2LM-495-9	2LM-494-9
ISOL N-2AI-C2L	1TY-971-1	1TY-970-1	2TY-971-1	2TY-970-1
Selector N-2AX-DSR	1TY-971	1TY-970	2TY-971	2TY-970

Table 1 (continued)

Description of Instrument	TAG NUMBERS			
	Unit 1 Yellow Power Source	Unit 1 White Power Source	Unit 2 Yellow Power Source	Unit 2 White Power Source
Summation Analog Module N-2AP+SUM	1TM-971-3	1TM-970-3	2TM-971-3	2TM-970-3
Analog Modules (H/H) N-2AP+ALM-AR	1TC-971-A 1TC-971-B	1TC-970-A 1TC-970-B	2TC-971-A 2TC-971-B	2TC-970-A 2TC-970-B
Analog Modules (T) N-2AO-L2C-R	1TY-971-A 1TY-971-B	1TY-970-A 1TY-970-B	2TY-971-A 2TY-971-B	2TY-970-A 2TY-970-B
Alarms (15°F) (30°F)	1D-2-3 1D-1-3	1D-2-3 1D-1-3	2D-2-7 2D-1-7	2D-2-7 2D-1-7
Analog Module (V/I) N-2AO-VAI	1TM-971-4	1TM-970-4	2TM-971-4	2TM-970-4
Indicator	1TI-971	1TI-970	2TI-971	2TI-970
Analog Module (V/V) N-2AO-VAI	1TM-971-5	1TM-970-5	2TM-971-5	2TM-970-5
Analog Module (T) N-2AO-L2C-R	1TY-971-2/3	1TY-970-2/3	2TY-971-2/3	2TY-970-2/3

Table 2

INSTRUMENT UNCERTAINTIES FOR RCS SUBCOOLING MONITOR

Instrument Description	Tag Number	Uncertainties Normal	Uncertainties Adverse
Pressure Transmitter	1 PT-420A 1 PT-420B 2 PT-420A 2 PT-420B	±0.96%	±8.16%
Analog Module N-2AI-I2V	1 PQ-420A 1 PQ-420B 2 PQ-420A 2 PQ-420B	±0.75%	±0.75%
Function Generator N-2AP+SGC	1 TM-970-6 1 TM-971-6 2 TM-970-6 2 TM-971-6	±4.785%[1]	+3.829% -5.910%
Core Exit Thermocouples	See Table 1		
Analog Module N-2AI-T2V	See Table 1	±1.71%[2]	±1.71%
Analog Module N-2AP-SUM	1 LM-494-9 1 LM-495-9 2 LM-494-9 2 LM-495-9		
Analog Module N-2AP-SUM	1 TM-970-3 1 TM-971-3 2 TM-970-3 2 TM-971-3	±2.763%[3]	+2.893 -2.605
Analog Module N-2AO-VAI	1 TM-970-4 1 TM-971-4 2 TM-970-4 2 TM-971-4	±0.866%	±0.866%
Indicator	1 TI-970 1 TI-971 2 TI-970 2 TI-971	±2.15%	±2.15%

Footnotes

(1) The net uncertainty in function generator N-2AP+SGC is given by:

$$E'_{FUNC} = [(E_{FUNC})^2 + (E_{DRIFT})^2]^{\frac{1}{2}}$$

Where:

$$E_{FUNC} = \{C_2 \sum_{i=1}^{j} g_i E_{CV})^2 + 2\sum_{i=1}^{j} (C_2 g_i E_{CV})^2 + \sum_{i=1}^{j} [Eg_i (S_I + a - b_i)/(S_{LB}(i) + a - b_i)]^2$$

$$+ E_{SGC}{}^2 + E_{CV}{}^2\}^{\frac{1}{2}}$$

E_{DRIFT} = 0.5% = uncertainty due to drift

E_{CV} = 0.1% = uncertainty of calibration voltage

E_{SGC} = 0.5% = inherent module accuracy

$Eg_i = f(S_{LB})$ = normalized gain uncertainty

S_{LB} = input voltage during calibration

S_I = input voltage

The uncertainty used in determining the reactor coolant pump trip criteria is evaluated at the cold leg accumulator pressure (700 psig) to bound small break LOCA conditions.

(2) The net uncertainty in indicated average core exit temperture as input to the sucooling monitor is combined as:

$$1.71\% \geq \{[(0.866)^2 + 1.04(0.79)^2][4(0.25)^2] + (1.6)^2\}^{\frac{1}{2}}$$

Where:

0.79% = uncertainty of the core exit thermocouples

0.866% = uncertainty of analog module N-2AI-T2V

0.25% = uncertainty in gain for each process stream

1.60% = uncertainty in analog module N-2AP-SUM including drift

(3) The net uncertainty in analog module N-2AP-VAI is given by:

$$E'_{MOD} = [(E_{MOD})^2 + (E_{DRIFT})^2]^{\frac{1}{2}}$$

Where:

$$E_{MOD} = [(E_{SUM})^2 + (E_{CV})^2 + (AEg_a/A_{CB})^2 + (BEg_b/B_{CB})^2]^{\frac{1}{2}}$$

E_{DRIFT} = 0.5% = uncertainty due to drift

E_{SUM} = 0.5% = inherent module accuracy

E_{CV} = 0.1% = uncertainty of calibration voltage

A = input voltage from process stream A

B = input voltage from process stream B

CB = refers to calibration process

g_A = gain for process stream A

g_B = gain for process stream B

E = output bias

The uncertainty used in determining the reactor coolant pump trip criteria is evaluated at the cold leg accumulator pressure (700 psig) to bound small break LOCA conditions.

Table 3

PROCEDURES WHICH INCLUDE RCP TRIP-RELATED OPERATIONS

a. RCP TRIP USING WOG ALTERNATIVE CRITERIA

Procedure Number	Procedure Title
EOP-0	Reactor Trip or Safety Injection
EOP-1	Loss of Reactor or Secondary Coolant
EOP-3	Steam Generator Tube Rupture
ECA-2.1	Uncontrolled Depressurization of Both Steam Generators

b. RCP RESTART

Procedure Number	Procedure Title
EOP-0.1	Reactor Trip Response
EOP-0.2	Natural Circulation Cooldown
EOP-0.3	Natural Circulation Cooldown with Steam Void in Vessel
EOP-1.1	SI Termination
EOP-1.2	Small Break LOCA Cooldown and Depressurization
EOP-3	Steam Generator Tube Rupture
ECA-2.1	Uncontrolled Depressurization of Both 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
CSP-C.1	Response to Inadequate Core Cooling
CSP-P.1	Response to Imminent Pressurized Thermal Shock Condition
CSP-I.3	Response to Voids in Reactor Vessel

c. DECAY HEAT REMOVAL BY NATURAL CIRCULATION

Procedure Number	Procedure Title
EOP-0.1	Reactor Trip Response
EOP-0.2	Natural Circulation Cooldown
EOP-1.1	SI Termination
EOP-1.2	Small Break LOCA Cooldown and Depressurization
EOP-3	Steam Generator Tube Rupture
ECA-0.1	Loss of All AC Power Recovery Without SI Required
ECA-2.1	Uncontrolled Depressurization of Both 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

d. PRIMARY SYSTEM VOID REMOVAL

Procedure Number	Procedure Title
EOP-0.2	Natural Circulation Cooldown
EOP-1	Loss of Reactor or Secondary Coolant
CSP-I.3	Response to Voids in Reactor Vessel

e. USE OF STEAM GENERTORS WITH AND WITHOUT RCP'S OPERATING

Procedure Number	Procedure Title
EOP-0	Reactor Trip or Safety Injection
EOP-0.1	Reactor Trip Response
EOP-0.2	Natural Circulation Cooldown
EOP-0.3	Natural Circulation Cooldown With Steam Void in Vessel

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Procedure Number	Procedure Title
EOP-1	Loss of Reactor or Secondary Coolant
EOP-1.1	SI Termination
EOP-1.2	Small Break LOCA Cooldown and Depress
EOP-2	Faulted Steam Generator Isolation
EOP-3	Steam Generator Tube Rupture
EOP-3.1	Post-Steam Generator Tube Rupture Cooldown Using Feedwater
EOP-3.2	Post-Steam Generator Tube Rupture Cooldown Using Blowdown
EOP-3.3	Post-Steam Generator Tube Rupture Cooldown Using Steam Dump
ECA-0.0	Loss of All AC Power
ECA-0.1	Loss of All AC Power Recovery Without SI Required
ECA-0.2	Loss of All AC Power Recovery With SI Required
ECA-1.1	Loss of Containment Sump Recirculation
ECA-2.1	Uncontrolled Depressurization of Both 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

Procedure Number	Procedure Title
CSP-S.1	Response to Nuclear Power Generation/ATWS
CSP-C.1	Response to Inadequate Core Cooling
CSP-C.2	Response to Degraded Core Cooling
CSP-H.1	Response to Loss of Secondary Heat Sink
CSP-H.2	Response to Steam Generator Overpressure
CSP-H.3	Response to Steam Generator High Level
CSP-H.4	Response to Loss of Normal Steam Release Capabilities
CSP-H.5	Response to Steam Generator Low Level
CSP-P.1	Response to Imminent Pressurized Thermal Shock Condition
CSP-P.2	Response to Anticipated Pressurized Thermal Shock Condition

f. RCP TRIP FOR OTHER REASONS

Procedure Number	Procedure Title
EOP-0	Reactor Trip or Safety Injection
EOP-1.2	Small Break LOCA Cooldown and Depressurization
EOP-3	Steam Generator Tube Rupture
EOP-3.1	Post-Steam Generator Tube Rupture Cooldown Using Feedwater
EOP-3.2	Post-Steam Generator Tube Rupture Cooldown using Blowdown
EOP-3.3	Post-Steam Generator Tube Rupture Cooldown using Steam Dump
ECA-1.1	Loss of Containment Sump Recirculation
ECA-1.2	LOCA Outside Containment

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Procedure Number	Procedure Title
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
CSP-C.1	Response to Inadequate Core Cooling
CSP-C.2	Response to Degraded Core Cooling
CSP-H.1	Response to Loss of Secondary Heat Sink
CSP-I.1	Response to High Pressurizer Level
CSP-I.3	Response to Voids in Reactor Vessel