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THOMPSON,H.L. Office of Nuclear Reactor Regulation, Director (post 851125

SUBJECT: Forwards info re reactor coolant pump trip criteria, potential problems & operator training & procedures, in response to Generic Ltr 85-12, "Implementation of TMI Action Item II.K.3.5."

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November 27, 1985
AEP:NRC:0785C

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
NUREG-0737 - II.K.3.5
REACTOR COOLANT PUMP TRIP (GENERIC LETTER 85-12)

Mr. Hugh L. Thompson, Jr., Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Thompson:

This letter and its attachments respond to your June 28, 1985 letter (Generic Letter 85-12) addressed to all applicants and licensees with Westinghouse Designed Nuclear Steam Supply Systems. With that letter we received your safety evaluation of the Westinghouse Owners Group (WOG) submittals on Generic Letters 83-10 C and D, which included your request for additional plant-specific information.

Our previous submittals on the topic of Reactor Coolant Pump (RCP) trip (AEP:NRC:0785, 0785A, 0785B), dated June 2, 1983, May 30, 1984, and June 25, 1984, along with the WOG submittals (OG-110, 117, 137) were made pursuant to Generic Letters 83-10 C and D, "Resolution of TMI Action Item II.K.3.5." This submittal complements that information by responding to your plant-specific requests made in Generic Letter 85-12, "Implementation of TMI Action Item II.K.3.5."

Of the three alternative RCP trip criteria evaluated by the WOG, Indiana & Michigan Electric Company has selected for the Donald C. Cook Nuclear Plant the option of manually tripping the RCPs based upon the criterion of Reactor Coolant System (RCS) pressure. The RCPs are to be tripped during post-accident conditions at an RCS pressure of less than 1250 psig.

Our response, which follows, parallels the implementation section format used in your safety evaluation included with Generic Letter 85-12. The following supplements are attached for your review:

Attachment 1: Determination of RCP Trip Criteria
Attachment 2: Potential Reactor Coolant Pump Problems
Attachment 3: Operator Training and Procedures (RCP Trip)

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This document has been prepared following Corporate procedures which incorporate a reasonable set of controls to insure its accuracy and completeness prior to signature by the undersigned.

Very truly yours,

M. P. Alexich
Vice President
11/27/85

cm

cc: John E. Dolan
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Bruchmann
G. Charnoff
NRC Resident Inspector - Bridgman

Attachment 1

To

AEP:NRC:0785C

RESPONSES TO NRC REQUESTS IN GENERIC LETTER 85-12
FOR ADDITIONAL INFORMATION ON RCP TRIP CRITERIA

Section A - Determination of RCP Trip Criteria

Request A.1: IDENTIFY THE INSTRUMENTATION TO BE USED TO DETERMINE THE RCP TRIP SET POINT, INCLUDING THE DEGREE OF REDUNDANCY OF EACH PARAMETER SIGNAL NEEDED FOR THE CRITERION CHOSEN.

Request 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.

Responses to A.1 & A.2

The instrumentation used to determine the Reactor Coolant Pump (RCP) trip setpoint for Units 1 and 2 of the Cook Nuclear Plant consists of two redundant wide-range Reactor Coolant System pressure detectors (NPS-121 & NPS-122). The uncertainty for these instruments is 3.25% of span for normal containment conditions and 12.70% of span for adverse containment conditions. The parameters used to determine adverse containment conditions are containment pressure $\geq$ 1.1 psig or containment radiation $\geq 10^5$ Rad/hr.

The containment pressure setpoint of 1.1 psig initiates safety injection and Phase A containment isolation. It is a conservative value for determining adverse containment conditions. At containment temperatures up to 228°F, the inaccuracies of the environmentally qualified instrumentation are relatively insignificant ($\gtrsim$ 3.0%). Assuming saturated conditions, 228°F corresponds to a containment pressure of $\approx$ 5 psig. To allow for containment pressure instrument inaccuracies and to provide a conservative value, the existing setpoint of 1.1 psig is used to determine adverse containment conditions.

For total integrated radiation doses up to 10^6 Rad, the inaccuracy of the environmentally qualified instrumentation is insignificant. The adverse containment setpoint of 10^5 Rad/hr is conservative. When this value is reached, adverse containment values will be used unless the integrated radiation dose is verified less than 10^6 Rad.

The wide-range pressure transmitters are located on opposite sides of the circular containment, at an elevation of 616'6", between the primary shield and the secondary shield barrier. By visualizing the containment as a circle comprised of four circumferential quadrants in the plan view, NPS-121 and NPS-122 are located in adjacent quadrants separated by an arc of approximately 150 degrees. The high-energy systems in the vicinity of these transmitters are the 30" main steam (MS) line at elevation 635'-0" and the 14" feedwater (FW) line at elevation 624'-3". These lines are protected from rupture and pipe whip by a system

Responses to A.1 & A.2 (cont.)

of five rupture restraints on the MS lines and four rupture restraints on the FW line, together with penetration anchors at the secondary shield wall. The rupture restraints for the MS and FW piping are designed to resist a pipe break force due to rupture within the main steamline or the feedwater line.

Review of Regulatory Guide 1.46 criteria for postulating break points on high-energy lines inside containment indicates that breaks are to be postulated at the terminal ends of the MS and FW lines (at the shield wall) and at a minimum of two intermediate points based upon a total stress criteria. On this basis, there exists a possibility of fluid jets following a pipe rupture that could affect the continued operation of the wide-range pressure transmitters NPS-121 and NPS-122.

However, should the unusual event of instrumentation damage occur from a pipe break fluid jet in one quadrant, the redundant instrumentation in the adjacent quadrant would serve as a backup.

Request 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 A.3: The LOFTRAN computer code was used to perform the alternate Reactor Coolant Pump (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 but three 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. The results of this validation show that LOFTRAN can accurately predict Reactor Coolant System (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 are 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 and auxiliary feedwater pumps functional), are due to either models or inputs to LOFTRAN. The following items are considered to have the most impact on the determination of the RCP trip criteria:

1. Break flow
2. Safety Injection (SI) Flow
3. Decay heat
4. Auxiliary Feedwater (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 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 WOG 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%.

Response A.3: (cont.)

The decay heat model used in the WOG 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 WOG 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 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 all auxiliary feed pumps 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, a 3% decrease in minimum RCS subcooling, and an 8% decrease in minimum pressure differential. 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 with 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 analyses as a result of these considerations for the D. C. Cook units is +30 to +200 psig for the minimum RCS pressure RCP trip setpoint. Due to the minimal effects from the Decay Heat Model and AFW input, these results include only the effects of the uncertainties due to the break flow model and SI flow inputs.

Attachment 2

To

AEP:NRC:0785C

RESPONSES TO NRC REQUESTS IN GENERIC LETTER 85-12
FOR ADDITIONAL INFORMATION ON RCP TRIP CRITERIA

Section B - Potential Reactor Coolant Pump Problems

Request B.1: ASSURE THAT CONTAINMENT ISOLATION, INCLUDING INADVERTENT ISOLATION, WILL NOT CAUSE PROBLEMS IF IT OCCURS FOR NON-LOCA TRANSIENTS AND ACCIDENTS. 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. CONFIRM THAT CONTAINMENT ISOLATION WITH CONTINUED PUMP OPERATION WILL NOT LEAD TO SEAL OR PUMP DAMAGE OR FAILURE.

Response B.1: The above concerns have been addressed in letters AEP:NRC:0185B, dated July 25, 1979, and AEP:NRC:0785, dated June 2, 1983. As stated in those letters, a Phase A containment isolation signal is initiated by a safety injection signal from the Reactor Protection System and/or containment pressure or manually. A Phase B containment isolation signal is initiated by containment pressure or manually.

The only Reactor Coolant Pump (RCP) valves that are affected upon a Phase A signal are the No. 1 seal leak-off header containment isolation valves. These valves close and the flow is redirected through a safety valve set at 150 psig to the pressurizer relief tank. Operation of the pump is acceptable under this condition for a prolonged period of time. No limit is required to restore the leak-off lines.

A Phase B signal, which would be expected to follow a Phase A signal, affects RCP operation by closing all service water valves to the pumps. Component Cooling Water (CCW) to and from the pump thermal barrier and motor oil coolers and Non-Essential Service Water (NESW) to and from the motor air coolers is terminated upon Phase B actuation. In conjunction with the seal leak-off header isolation, the reactor coolant pumps must be tripped as soon as operating parameters exceed their established limits. In particular, pump bearing temperature cannot exceed 225°F. Operating procedures exist based on pump manufacturer's instructions to trip the reactor coolant pumps upon exceeding pump/motor operating parameters which is likely to occur with a Phase A actuation followed by a Phase B actuation or upon an inadvertent manual Phase B actuation.

Monitoring of pump and seal parameters is required to determine the need to trip the pumps if operating conditions deteriorate where pump/seal damage can occur. This criterion is to be followed for all normal, abnormal, and emergency conditions.

Response B.1: (cont.)

Restoring water services (Component Cooling Water) to the RCPs can be accomplished approximately five minutes after Phase B. It is a matter of resetting the containment isolation Phase B signal and opening the motor-operated supply and return valves. Each action is performed from the control room. The limiting factor appears to be the time required to reduce containment pressure below the Phase B setpoint. Seal injection is supplied throughout any of these events including Phase B isolation, which should preclude seal damage.

The following is an excerpt from the Generic Issue section of the Emergency Response Guidelines Executive Volume (RCP Trip/Restart Topic):

> "If seal injection flow and CCW flow have been interrupted for an appreciable period of time (for example, greater than 1 hour) while the RCS remains water-filled at high pressure and the RCPs have been tripped, two concerns will have to be addressed if RCP restart is to be attempted. First, if the RCS has remained water-filled, the seals will most likely have been exposed to temperatures well above their normal operating temperatures. Reestablishing seal injection flow under this situation must be done slowly if high thermal stresses are to be avoided. Thermal shock can result in seal failure and the potential bending of RCP shafts. This concern would not be as critical or may be absent if CCW flow to the thermal barrier had been maintained at design conditions.
>
> "The second concern is that any grit, crud and corrosion products present in the reactor coolant may accumulate in the No. 1 seal region if seal injection flow is lost or interrupted for appreciable time intervals. If grit deposits form in the seals, the potential exists for seal damage and scoring during restart. This, in turn, could lead to No. 1 seal failure or abnormally high seal leak-off flow.
>
> "Thus, if an RCP is tripped in response to an ERG instruction, it is advantageous to maintain seal injection flow and CCW flow to the thermal barrier. If both services have been lost for appreciable time intervals, it is necessary to reestablish seal cooling carefully to avoid potential mechanical damage."

The "concerns" related above are addressed in appropriate steps of Westinghouse Owners Group Emergency Response Guideline-based, D. C. Cook-specific EOPs. These steps outline the steps necessary to "re-establish seal cooling carefully."

Request B.2: IDENTIFY THE COMPONENTS REQUIRED 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.

Response B.2: Each RCP can be electrically tripped, remote from the circuit breaker, via its control switch located in the control room. This means of tripping the breaker requires the availability of 250VDC breaker control circuit power. The breakers can also be tripped mechanically at the breaker. This means of tripping does not require the 250VDC power.

Request B.2 states: "Assure that RCP trip, when determined to be necessary, will occur." Although the RCPs are not considered to be Class 1E, the circuit breaker and control switches are of the same pedigree as Class 1E devices and were purchased 1E. The 250VDC distribution used for these circuits is classified as a 1E source.

The breakers are automatically tripped for electrical faults and 4kV bus underfrequency conditions.

The major components which are required to electrically trip the RCPs are listed in Tables 2.1 and 2.2 which follow. None of these components are located inside containment.

TABLE 2.1
UNIT 1 COMPONENTS REQUIRED TO TRIP REACTOR COOLANT PUMPS

RCP #1

- 4kV ACB 1B9
- Control switch 101-B9
- Cabinet MCAB circuit 2, 100A fuses
- 35A fuses at ACB 1B9
- Cable 6783-1, 101-B9 to ACB 1B9
- Cable 3422R-1, MCAB to ACB 1B6

RCP #2

- 4kV ACB 1C2
- Control switch 101-C2
- Cabinet MCCD circuit 1, 100A fuses
- 35A fuses at ACB 1C2
- Cable 4959-1, 101-C2 to ACB 1C2
- Cable 3496G-1, MCCD to ACB 1C8

RCP #4

- 4kV ACB 1A4
- Control switch 101-A4
- Cabinet MCAB circuit 1, 100A fuses
- 35A fuses at ACB 1A4
- Cable 4963-1, 101-A4 to ACB 1A4
- Cable 3421R-1, MCAB to ACB 1A6

RCP #3

- 4kV ACB 1D9
- Control switch 101-D9
- Cabinet MCCD circuit 2, 100A fuses
- 35A fuses at ACB 1D9
- Cable 5649-1, 101-D9 to ACB 1D9
- Cable 3497G-1, MCCD to ACB 1D6

Components Common to Both RCP #1 and RCP #4

- Battery 1AB
- 600A fuses at battery 1AB fuse box
- Cabinet MCAB
- Cabinet MDAB
- Cabinet TDAB
- Battery 1AB shunt 1 & 2
- Cable 8367R-1, MCAB to MDAB
- Cable 8365R-1, MDAB to TDAB
- Cable 8352R-1, TDAB to shunt box
- Cables 8350R-1 & 8351R-1, shunt box to battery 1AB, via 600A fuse box

Components Common to Both RCP #2 and RCP #3

- Battery 1CD
- 600A fuses at battery 1CD fuse box
- Cabinet MCCD
- Cabinet MDCD
- Cabinet TDCD
- Battery 1CD shunt 1 & 2
- Cable 8367G-1, MCCD to MDCD
- Cable 8365G-1, MDCD to TDCD
- Cable 8352G-1, TDCD to shunt box
- Cable 8350G-1, & 8351G-1, shunt box to battery 1CD via 600A fuse box

TABLE 2.2
UNIT 2 COMPONENTS REQUIRED TO TRIP REACTOR COOLANT PUMPS

RCP #1

- 4kV ACB 2B9
- Control switch 101-B9
- Cabinet MCAB circuit 2, 100A fuses
- 35A fuses at ACB 2B9
- Cable 6783-2, 101-B9 to ACB 2B9
- Cable 3422R-2, MCAB to ACB 2B6

RCP #2

- 4kV ACB 2C2
- Control switch 101-C2
- Cabinet MCCD circuit 1, 100A fuses
- 35A fuses at ACB 2C2
- Cable 4959-2, 101-C2 to ACB 2C2
- Cable 3496G-2, MCCD to ACB 2C8

RCP #4

- 4kV ACB 2A4
- Control switch 101-A4
- Cabinet MCAB circuit 1, 100A fuses
- 35A fuses at ACB 2A4
- Cable 4963-2, 101-A4 to ACB 2A4
- Cable 3421R-2, MCAB to ACB 2A6

RCP #3

- 4kV ACB 2D9
- Control switch 101-D9
- Cabinet MCCD circuit 2, 100A fuses
- 35A fuses at ACB 2D9
- Cable 5649-2, 101-D9 to ACB 2D9
- Cable 3497G-2, MCCD to ACB 2D6

Components Common to Both RCP #1 and RCP #4

- Battery 2AB
- 600A fuses at battery 2AB fuse box
- Cabinet MCAB
- Cabinet MDAB
- Cabinet TDAB
- Battery 2AB shunt 1 & 2
- Cable 8357R-2, MCAB to MDAB
- Cable 8365R-2, MDAB to TDAB
- Cable 8352R-2, TDAB to shunt box
- Cables 8350R-2 & 8351R-2, shunt box to battery 2AB, via 600A fuse box

Components Common to Both RCP #2 and RCP #3

- Battery 2CD
- 600A fuses at battery 2CD fuse box
- Cabinet MCCD
- Cabinet MDCD
- Cabinet TDCD
- Battery 2CD shunt 1 & 2
- Cable 8367G-2, MCCD to MDCD
- Cable 8365G-2, MDCD to TDCD
- Cable 8352G-2, TDCD to shunt box
- Cable 8350G-2, & 8351G-2, shunt box to battery 2CD via 600A fuse box

Attachment 3

To

AEP:NRC:0785C

RESPONSES TO NRC REQUESTS IN GENERIC LETTER 85-12
FOR ADDITIONAL INFORMATION ON RCP TRIP CRITERIA

Section C - Operator Training and Procedures (RCP Trip)

Request 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 C.1: The following is taken from the lesson plan used to train plant operations personnel on the emergency operating procedures. This part of the lesson plan covers the basis for tripping the RCPs during an uncontrolled primary system depressurization, and states our philosophy on this matter.

If RCS pressure below 1250 psig and SI flow present then RCPs are stopped.

During a SBLOCA, leak rate can exceed injection flow rate resulting in RCS mass depletion.

As mass depletes, RCS will first reach saturation, then void fraction will increase.

With RCPs running, the steam-water mixture will provide adequate core cooling.

If RCPs suddenly stop, the steam and water will separate and cause the upper portion of the core to be exposed to a steam environment and overheat.

SI flow will eventually reflood and cool the core, but damage will have already been done.

1250 psig in RCS is highest pressure at which RCS could reach saturation. By tripping the RCPs then, the conditions described above will be avoided.

Note that concern is related to mass depletion. If SI flow is not present, then RCPs are left on even if pressure is low to avoid creating the phase separation situation.

This is a continuous action.

Request 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 RCPs OPERATING
(f) RCP TRIP FOR OTHER REASONS

Response C.2.a: RCP trip using WOG alternate criteria:

OHP 4023.E-0, Rev. 0	Step 21
OHP 4023.E-1, Rev. 0	Step 1
OHP 4023.E-3, Rev. 0	Step 1
OHP 4023.ECA-2.1, Rev. 0	Step 3

Response C.2.b: RCP restart:

OHP 4023.ES-1.2, Rev. 0	Step 12, 19
OHP 4023.E-3, Rev. 0	Step 35
OHP 4023.ECA-2.1, Rev. 0	Step 31
OHP 4023.ECA-3.1, Rev. 0	Step 16, 23
OHP 4023.ECA-3.2, Rev. 0	Step 10, 17
OHP 4023.ECA-3.3, Rev. 0	Step 2
OHP 4023.FR-C.1, Rev. 0	Step 19
OHP 4023.ES-0.1, Rev. 0	Step 9
OHP 4023.ES-0.2, Rev. 0	Step 1
OHP 4023.ES-0.3, Rev. 0	Step 2
OHP 4023.ES-0.4, Rev. 0	Step 2
OHP 4023.ES-1.1, Rev. 0	Step 23
OHP 4023.FR-P.1, Rev. 0	Step 5, 12
OHP 4023.FR-I.3, Rev. 0	Step 9

Response C.2.c: Decay heat removal by natural circulation:

OHP 4023.ES-0.2
OHP 4023.ES-0.3
OHP 4023.ES-0.4
OHP 4023.ECA-0.1
OHP 4023.ES-0.1

Response C.2.d: Primary system void removal:

OHP 4023.FR-I.3, Rev. 0

Response C.2.e: **Use of steam generators with and without RCPs operating.**

(The following procedures have steps to follow if the RCPs are operating, and alternate steps to follow if the RCPs are not operating.)

OHP 4023.E-1, Rev. 0
OHP 4023.ES-1.1, Rev. 0
OHP 4023.ES-1.2, Rev. 0
OHP 4023.E-3, Rev. 0
OHP 4023.ECA-3.2, Rev. 0
OHP 4023.ECA-3.1, Rev. 0
OHP 4023.ECA-2.1, Rev. 0
OHP 4023.FR-C.2, Rev. 0

Response C.2.f: **RCP trip for other reasons**

OHP 4023.ES-1.2, Rev. 0	Step 12, 19, 29
OHP 4023.E-3, Rev. 0	Step 35
OHP 4023.ES-3.1, Rev. 0	Step 10
OHP 4023.ES-3.2, Rev. 0	Step 12
OHP 4023.ECA-1.1, Rev. 0	Step 19
OHP 4023.ECA-3.1, Rev. 0	Step 16, 23, 35
OHP 4023.ECA-3.2, Rev. 0	Step 10, 17, 29
OHP 4023.ECA-3.3, Rev. 0	Step 36
OHP 4023.FR-C.1, Rev. 0	Step 14, 22
OHP 4023.FR-C.2, Rev. 0	Step 7, 14, 18
OHP 4023.FR-H.1, Rev. 0	Step 3
OHP 4023.ES-3.3, Rev. 0	Step 12
1-OHP 4022.002.001, Rev. 1	Step 4.2.1, 4.2.2, 4.2.3, 4.2.4, 4.2.5