Indian Point Energy Center

2 correct answers on Administrative JPM for emergency event classification.

The JPM in contention was designed to elicit the applicant to determine that 3 barriers have or have the potential for failure. Based on the design, the JPM answer key required the event to be classified as a General Emergency.

The conditions provided in the JPM to the applicants clearly indicate that the RCS barrier has been lost (LOCA in progress) and the Fuel Barrier has a potential loss (Degraded Core Cooling FR-C.2 has been entered and exited). The Containment barrier conditions are provided as Containment Pressure 2.8 psig stable.

A high percentage of the applicants applied the following thought process when evaluating the Containment Barrier:

LOCA occurs and E-0 is entered. Time spent in E-0 is approximately 10-15 minutes. Based on the conditions provided, FR-C.2 is entered upon exit of E-0. Time spent in FR-C.2 is approximately 10-15 minutes. FR-P.1 is entered when FR-C.2 is exited. Time spent in FR-P.1 is approximately 5-10 minutes. E-1 is entered when FR-P.1 is exited.

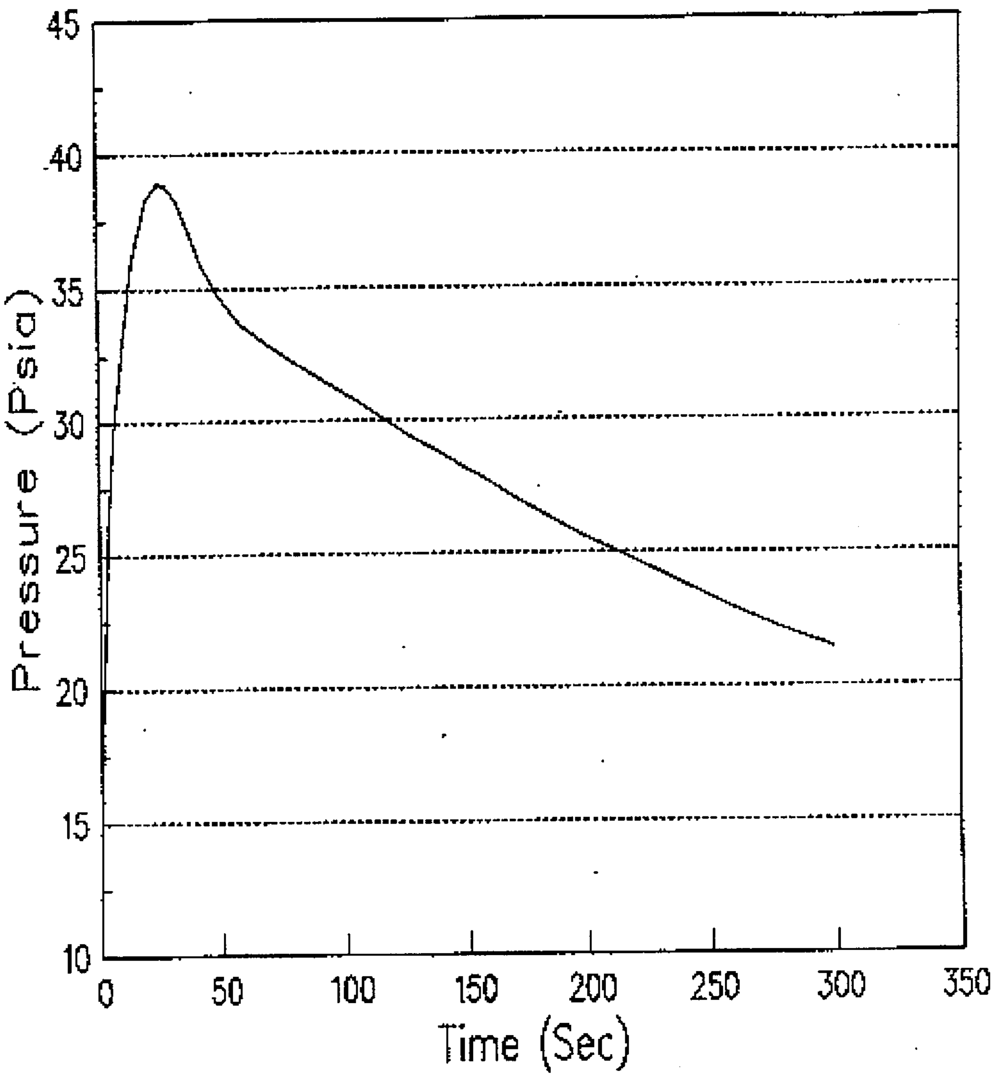
Based on 25-40 minutes into the accident sequence, with equipment functioning as specified in the JPM conditions, containment pressure will be in the single digits. A review of the FSAR and simulated accident on the IP2 simulation fully support this containment pressure response.

The applicants that applied this mental model came to the correct event classification of Site Area Emergency – Loss or potential loss of 2 barriers with the Containment Barrier remaining intact.

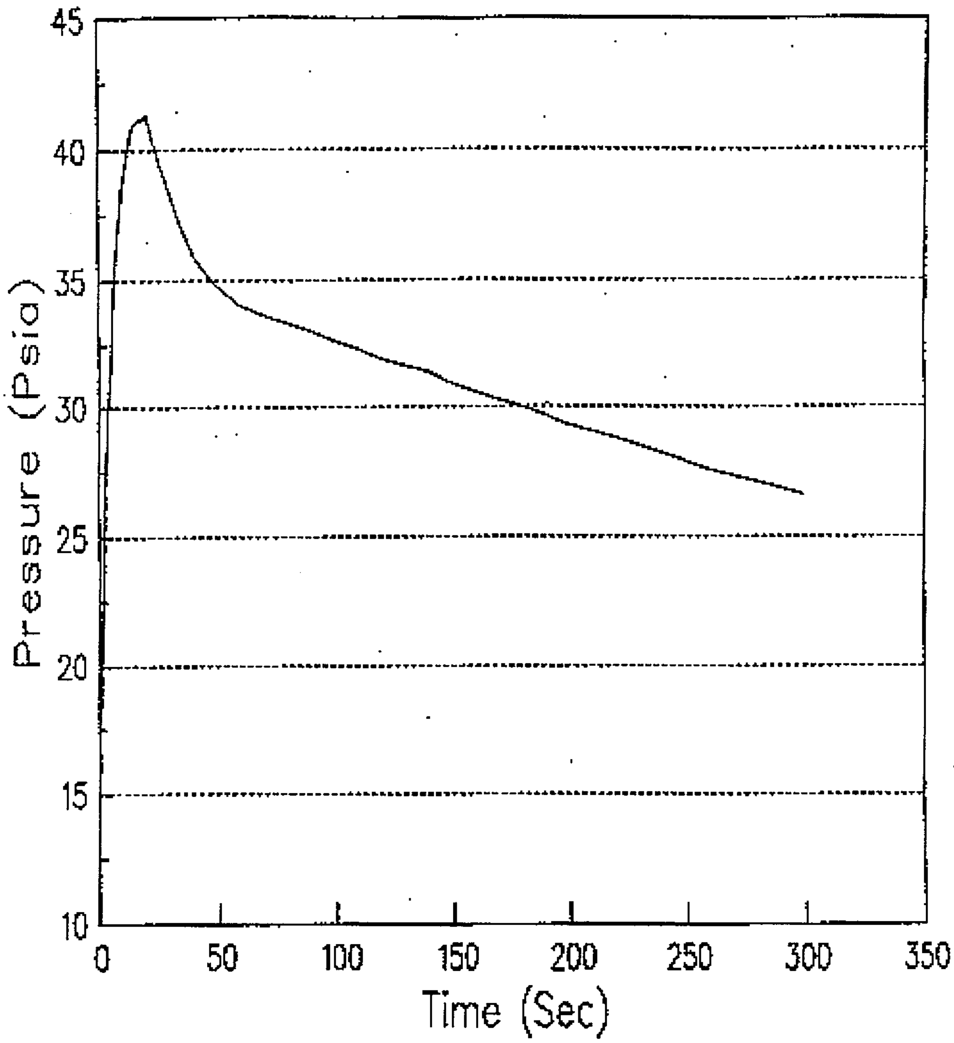
Containment Pressure Response Curves from the FSAR and Simulation are attached.

F. WILSON

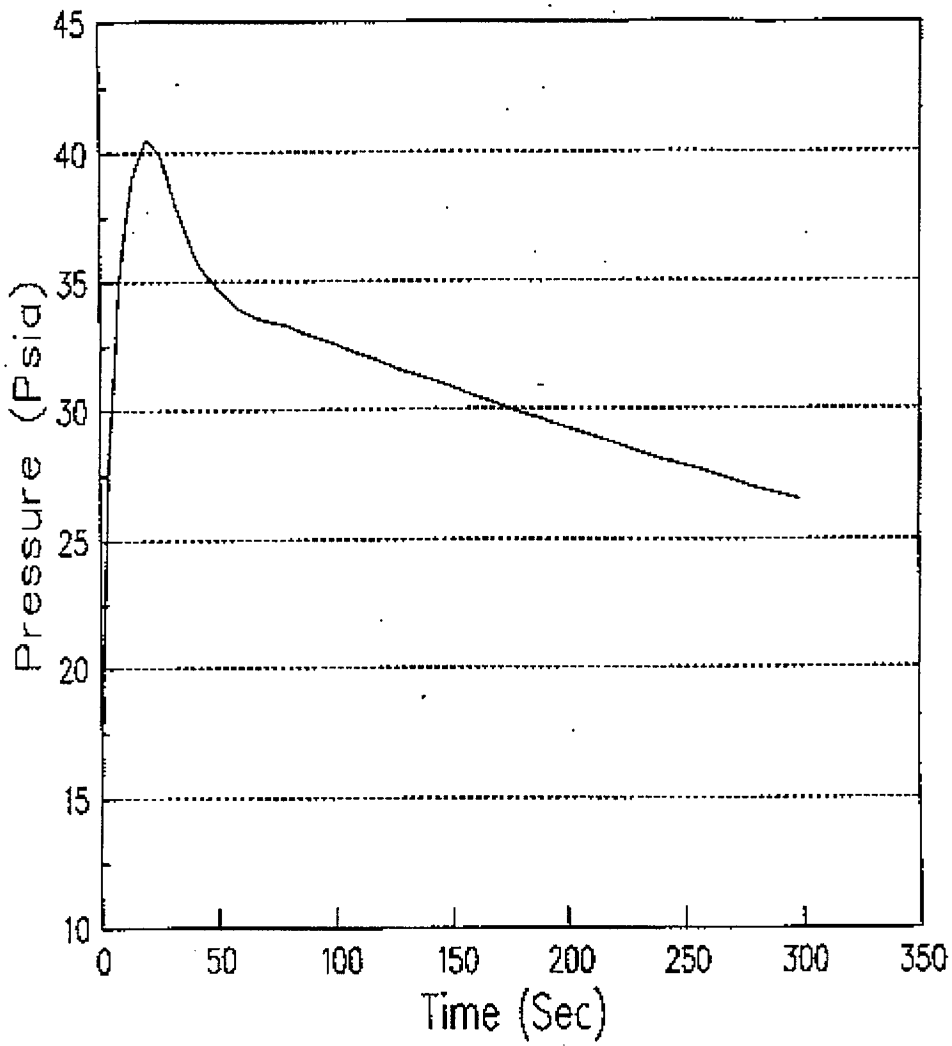
SUPERINTENDENT OF
OPERATIONS TRAINING
MARCH 28 2003



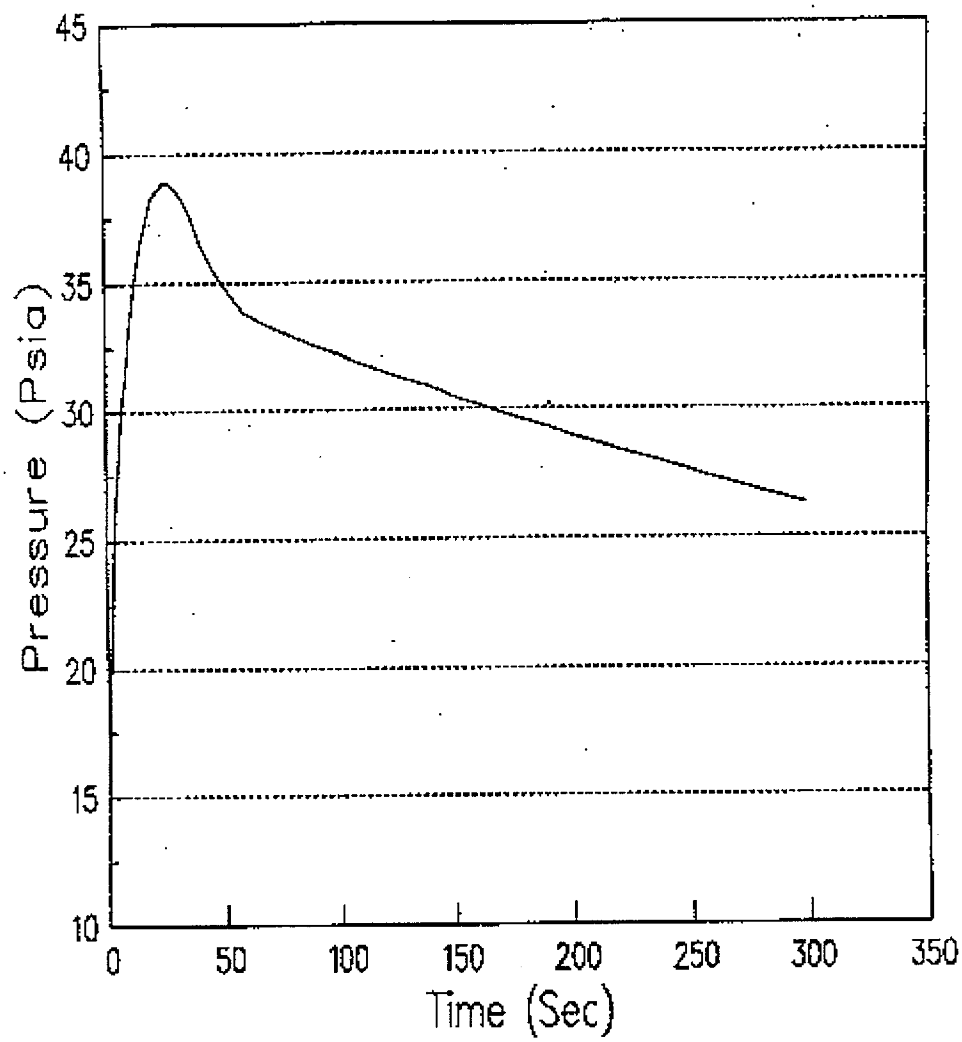
INDIAN POINT 3	FSAR UPDATE
C_D = 0.4 LARGE BREAK LOCA CONTAINMENT PRESSURE MAX SI	
REV. 1 JUNE, 1999	FIGURE No. 14.3-45a



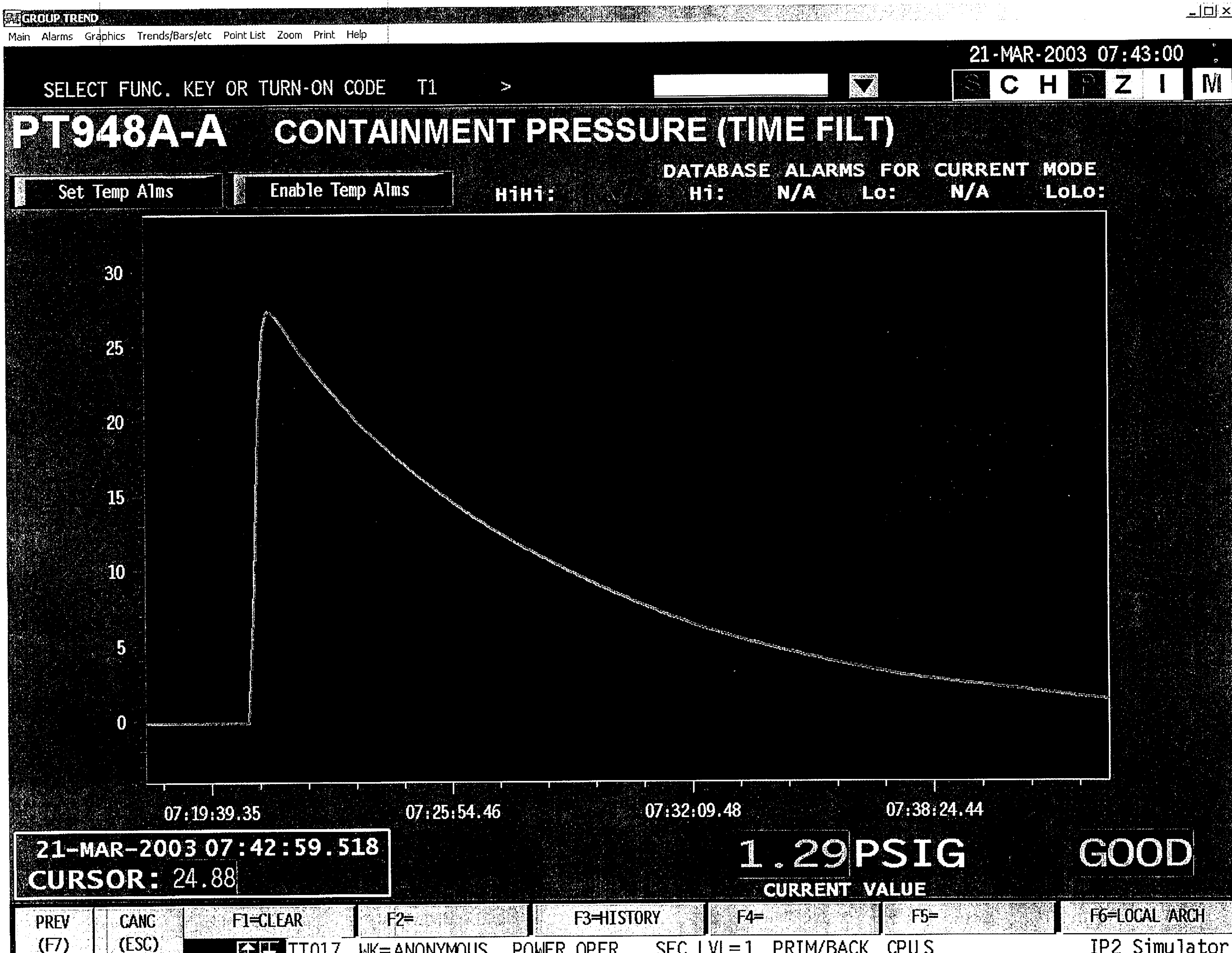
INDIAN POINT 3	FSAR UPDATE
CD = 0.8 LARGE BREAK LOCA CONTAINMENT PRESSURE	
REV. 1 JUNE, 1999	FIGURE No. 14.3-43



INDIAN POINT 3	FSAR UPDATE
CD = 0.6 LARGE BREAK LOCA CONTAINMENT PRESSURE	
REV. 3 JUNE, 1999	FIGURE No. 14.3-44



INDIAN POINT 3	FSAR UPDATE
CD = 0.4 LARGE BREAK LOCA CONTAINMENT PRESSURE	
REV. 3 JUNE, 1999	FIGURE No. 14.3-45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Main Alarms Graphics Trends/Bars/etc Point List Zoom Print Help
21-MAR-2003 07:43:00
SELECT FUNC. KEY OR TURN-ON CODE T1 >
C H Z I M
PT948A-A CONTAINMENT PRESSURE (TIME FILT)
Set Temp Alms
Enable Temp Alms
DATABASE ALARMS FOR CURRENT MODE
HiHi: Hi: N/A Lo: N/A LoLo:
30
25
20
15
10
5
0
07:19:39.35
07:25:54.46
07:32:09.48
07:38:24.44
21-MAR-2003 07:42:59.518
CURSOR: 24.88
1.29 PSIG
CURRENT VALUE
GOOD
PREV (F7)
CANC (ESC)
F1=CLEAR
F2=
F3=HISTORY
F4=
F5=
F6=LOCAL ARCH
TT017 WK=ANONYMOUS POWER OPER SEC LVL=1 PRIM/BACK CPU S
IP2 Simulator

Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
IPEC Training
P.O. Box 308
Buchanan, NY 10511
Tel 914 736 8000

March 28, 2003
IP-TNG-03-018

Operational Safety Branch
Division of Reactor Safety
United States Nuclear Regulatory Commission
Region 1
475 Allendale Road
King of Prussia
Pennsylvania, 19406-1415
Att: Mr. R J Conte

SENIOR REACTOR AND REACTOR OPERATOR INITIAL EXAMINATION – INDIAN POINT STATION, UNITS 2 AND 3

Dear Mr. Conte,

In accordance with NUREG 1021, please find attached the following examination documents as specified in Section ES-501, for the Indian Point Energy Center Reactor Operator and Senior Reactor Operator written examinations completed on March 21, 2003:

- The graded written examinations (each applicant's original answer and examination cover sheets) plus a clean copy of each applicant's answer sheet
- The master examination(s) and answer key(s). No changes were made while administering or grading the examination(s)
- Questions asked by and answers given to the applicants during the written examination
- No substantive comments were made by the applicants after the written examination
- The written examination seating chart
- A completed Form ES-403-1, "Written Examination Grading Quality Checklist"

See attached test Exam Analysis Bill ~~Frank~~ Altic / Frank Wilson

NOTE: per telecon post exam analysis reflects no correct answer for Question # 97 Question deleted [illegible] 4/9/03

- The written examination performance analysis. No question contentions are ~~being~~ submitted at this time. Please note that the utility reserves the right to submit contentions at a later date.

- The original of Form ES-201-3, "Examination Security Agreement,"

If you have any questions, please contact Mr. Bill Altic at (914) 788-2629 for Unit 2, Mr. Steve Joubert at (914) 788-2973 for Unit 3, or me at (914) 788-2904.

Sincerely,

Frank Wilson
Superintendent – Operations Training
Indian Point Energy Center

Question 9: Common RO/SRO exams - 4 of 6 incorrect responses.

Question asks for the method to transfer feedwater control to Main Feedwater Regulating Valves.

> *PLACE selected Main Feedwater Regulator Valve Controller in AUTO. IF a Main Feedwater Regulator becomes erratic, RETURN it to MAN and RESTORE SG level to the programmed value.*

Question is valid as written.

Question 28: Common RO/SRO exams – 4 of 6 incorrect responses.

The question asks about the air compressor arrangement after a leak on the Instrument Air header.

> *The U1 station air compressors are used as the primary source of air supply to Unit 2 IA and SA systems. Unit 2 Station Air serves as a backup supply of air to the Unit 2 Instrument Air system.*

The question is valid as written.

Question 32: Common RO/SRO exam – 5 of 6 incorrect responses.

Question asks about the Interlock Override Key switch on the Fuel handling Manipulator Crane

> *Answer D is the correct answer. All other answers are incorrect.*

The question is valid as written.

Question 41: Common RO/SRO exam – 4 of 6 incorrect responses.

Question asks xenon response to a power reduction with a misaligned rod.

> *Answer C is correct.*

Question is valid as written.

Indian Point Unit 2 NRC ILO Test Item Analysis

Question 9: Common RO/SRO exams - 4 of 6 incorrect responses.

Question asks for the method to transfer feedwater control to Main Feedwater Regulating Valves.

> *PLACE selected Main Feedwater Regulator Valve Controller in AUTO. IF a Main Feedwater Regulator becomes erratic, RETURN it to MAN and RESTORE SG level to the programmed value.*

Question is valid as written.

Question 28: Common RO/SRO exams – 4 of 6 incorrect responses.

The question asks about the air compressor arrangement after a leak on the Instrument Air header.

> *The U1 station air compressors are used as the primary source of air supply to Unit 2 IA and SA systems. Unit 2 Station Air serves as a backup supply of air to the Unit 2 Instrument Air system.*

The question is valid as written.

Question 32: Common RO/SRO exam – 5 of 6 incorrect responses.

Question asks about the Interlock Override Key switch on the Fuel handling Manipulator Crane

> *Answer D is the correct answer. All other answers are incorrect.*

The question is valid as written.

Question 41: Common RO/SRO exam – 4 of 6 incorrect responses.

Question asks xenon response to a power reduction with a misaligned rod.

> *Answer C is correct.*

Question is valid as written.

Indian Point Unit 2 NRC ILO Test Item Analysis

Question 58: Common RO/SRO exam – 3 of 6 incorrect responses.

The question asks about the response to a loss of redundant power supplies to a Rod Control Power Cabinet.

Answer A is correct

Question valid as written

Question 59: Common RO/SRO exam – 3 of 6 incorrect responses.

Question asks the downstream temperature of a leaking PZR safety.

Answer D is correct

Question valid as written.

Question 60: Common RO/SRO exam – 4 of 6 incorrect responses.

Question asks ECCS flows to the loops after a break in the SI Cold Leg Discharge connection.

SI pump discharge lines contain variable orifice valves (FE-6454, 6, 7, and 8) The valves are throttled to predetermined position and welded. This assures proper flow distribution and maintains minimum and maximum flow as used in accident analysis

Question valid as written

Question 72: RO Only- 2 of 3 incorrect responses.

Question asks the approximate level of indicated Source Range level.

Answer C is the correct answer

Question is valid as written.

Indian Point Unit 2 NRC ILO Test Item Analysis

Question 77: RO Only- 2 of 3 incorrect responses.

Question asks about the response of a thermocouple display following a ventilation failure.

Answer C is correct

Question is valid as written.

Question 79: RO Only- 2 of 3 incorrect responses.

Question asks about voltage adjustments when starting a condensate pump.

Answer B is correct

Question is valid as written.

Question 82: SRO Only- 3 of 3 incorrect responses.

Question asks Temporary Procedure Changes to EOPs.

OAD 27 has been cancelled.
Answer C is correct

Question is valid as written. Needs to be updated for new procedures.

Question 86: RO Only- 2 of 3 incorrect responses.

Question asks for radiation monitor response after starting two exhaust fans.

Answer A is correct

Question is valid as written.

Question 97: SRO Only- 3 of 3 incorrect responses.

Question asks the Tech Spec requirements for a loss of both intermediate range instruments during a startup.

There is no correct answer. Intermediate range instruments are required until greater than 10% power by the IP2 technical specifications. IP2 TS requires Hot Shutdown condition but there is not a stated time limit. Recommend that Question 97 be deleted.

No correct answer. Reference IP2 Technical Specifications Table 3.5-2

Post-it® Fax Note 7671	Date	# of pages ▶
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Co./Dept.	Co.	
Phone #	Phone #	
Fax # 610-337-6528	Fax #	

3.5 INSTRUMENTATION SYS

Operational Safety Instrumentatior

Applicability

Applies to plant instrumentation systems.

Objectives

To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specifications

3.5.1 When the plant is not in the cold shutdown condition, the Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 3.5-1.

3.5.2 For instrumentation channels, plant operation at rated power shall be permitted to continue in accordance with Tables 3.5-2 through 3.5-4. No more than one channel of a particular protection channel set shall be tested at the same time. By definition, an instrumentation channel failure shall not be regarded as a channel being tested.

3.5.3 In the event the number of channels of a particular function in service falls below the limits given in the column entitled Minimum Operable Channels, or Minimum Degree of Redundancy cannot be achieved, operation shall be limited according to the requirements shown in Column 5 or 6 of Tables 3.5-2 through 3.5-4. For on-line testing or corrective maintenance of instruments with installed bypass capability, the required minimum degree of redundancy may be reduced by one to permit testing or corrective maintenance of a channel in bypass.

3.5.4 In the event of sub-system instrumentation channel failure, Tables 3.5-2 through 3.5-4 need not be observed during the short period of time the operable sub-system channels are tested where the failed channel must be blocked to prevent unnecessary reactor trip.

3.5.5 The cover plate on the rear of the safeguards panel in the control room shall not be removed without authorization from the Watch Supervisor.

3.5.6 When the reactor coolant system is above 350°F, the instrumentation requirements as stated in Table 3.5-5 shall be met.

Basis

Instrumentation has been provided to sense accident conditions and to initiate operation of the Reactor Protection System or the Engineered Safety Features[1,4].

Safety Injection System Actuation

Protection against a loss-of-coolant or steam break accident is brought about by automatic actuation of the Safety Injection System, which provides emergency cooling and reduction of reactivity.

The loss-of-coolant accident is characterized by depressurization of the Reactor Coolant System and rapid loss of reactor coolant to the containment. The Engineered Safety Features have been designed to sense the effects of the loss-of-coolant accident by detecting low pressure and generator signals actuating the SIS active phase.

The SIS active phase is also actuated by a high containment pressure signal (Hi-Level) brought about by loss of high enthalpy coolant to the containment. This actuation signal acts as a backup to the low pressurizer pressure signal actuation of the SIS and also adds diversity to protection against loss of coolant.

Signals are also provided to actuate the SIS upon sensing the effects of a steamline-break accident. Therefore, SIS actuation following a steamline break is designed to occur upon sensing high differential steam pressure between any two steam generators or upon sensing high steamline flow in coincidence with low reactor coolant average temperature or low steamline pressure.

The increase in the extraction of RCS heat following a steamline break results in reactor coolant temperature and pressure reduction. For this reason, protection against a steamline-break accident is also provided by low pressurizer pressure signals actuating safety injection.

Protection is also provided for a steamline break in the containment by actuation of SIS upon sensing high containment pressure.

SIS actuation injects highly borated fluid into the Reactor Coolant System in order to counter the reactivity insertion brought about by cooldown of the reactor coolant which occurs during a steamline-break accident.

Containment Spray

The Engineered Safety Features actuation system also initiates containment spray upon sensing a high containment pressure signal (Hi-Hi Level). The containment spray acts to reduce containment pressure in the event of a loss-of-coolant or steamline-break accident inside the containment. The spray cools the containment directly and limits the release of fission products by absorbing iodine should it be released to the containment.

Containment spray is designed to be actuated at a higher containment pressure (approximately 50% of design containment pressure) than the SIS (2.0 psig). Since spurious actuation of containment spray is to be avoided, it is automatically initiated only on coincidence of Hi-Hi Level containment pressure sensed by both sets of two-out-of-three containment pressure signals.

Steamline Isolation

Steamline isolation signals are initiated by the Engineered Safety Features closing all steamline stop valves. In the event of a steamline break, this action prevents continuous, uncontrolled steam release from more than one steam generator by isolating the steamlines on high containment pressure (Hi-Hi Level) or high steamline flow. Protection is afforded for breaks inside or outside the containment even when it is assumed that there is a single failure in the steamline isolation system.

Feedwater Line Isolation

The feedwater lines are isolated upon actuation of the Safety Injection System in order to prevent excessive cooldown of the reactor coolant system. This mitigates the effect of an accident such as steam break which in itself causes excessive coolant temperature cooldown.

Feedwater line isolation also reduces the consequences of a steamline break inside the containment by stopping the entry of feedwater.

Setting Limits

The Engineered Safety Features Actuation System instrumentation trip setpoints Specified in Table 3.5-1 are the nominal values at which the bistables may be set for each functional unit. A setpoint for an Engineered Safety Features Actuation System or interlock function is applicable to the process rack modules and is considered to be adjusted consistent with the nominal value when the "as left" value is within the band allowed for calibration accuracy. Sensor/Transmitters are considered to be adjusted consistent with the nominal value when the "as left" value(s) at the calibration point(s) is (are) within the band allowed for calibration accuracy. This band is defined by the calibration accuracy applied in both the conservative and non-conservative directions about the trip setpoint for process rack modules and calibration point(s) for sensor/transmitters as defined by plant calibration procedures and the plant setpoint study.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, administrative limits for the setpoints have been determined. Operation with "as found" setpoints less conservative than the Trip Setpoint but within the administrative limit is acceptable since allowances have been made in the plant setpoint study to account for the applicable instrument uncertainties and the plant administrative process, including the administrative limit, verifying that the instrument performance complies with the plant setpoint study. Operation with "as found" setpoints less conservative than the administrative limit requires that further instrument operability evaluations be performed. This would include verification that the channel is capable of demonstrating operating performance within the design characteristics of the instruments through channel calibration, drift evaluations, instrument response characteristics, and other manufacturer recommended tests. Process rack modules or a sensor/transmitter found outside the "as left" band for calibration accuracy must be returned to within the band after the performance of each surveillance test.

1. The Hi Level containment pressure limit is set at 2.0 psig containment pressure. Initiation of Safety Injection protects against loss-of-coolant[2,4] or steamline-break[3,4] accidents as discussed in the safety analysis.

2. The Hi-Hi Level containment pressure limit is set at about 50% of design containment pressure. Initiation of Containment Spray and Steamline Isolation protects against large loss of coolant[2] or steamline-break accidents[3] as discussed in the safety analysis.

3. The pressurizer low-pressure limit is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss-of-coolant accident as shown in the safety analysis[2].

4. The steamline high differential pressure limit is set well below the differential pressure expected in the event of a large steamline-break accident as shown in the safety analysis[3].

5. The high steamline flow limit is set at approximately 40% of the full steam flow at 0% to 20% load. Between 20% and 100% (full) load, the trip setpoint is ramped linearly with respect to first stage turbine pressure from 40% of the full steam flow to 110% of the full steam flow. These setpoints will initiate safety injection in the case of a large steamline-break accident. Coincident low T_{avg} setting limit for SIS and steamline isolation initiation is set below its hot shutdown value. The coincident steamline pressure setting limit is set below the full load operating pressure. The safety analyses show that these settings provide protection in the event of a large steamline break[3].

Instrument Operating Conditions

During plant operation, the complete instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. A channel bistable may also be placed in a bypassed mode; e.g., a two-out-of-three circuit becomes a two-out-of-two circuit. The nuclear instrumentation system channels are not intentionally placed in a tripped mode since the test signal is superimposed on the normal detector signal to test at power. Testing of the NIS power range channel requires (1) bypassing the Dropped Rod protection from NIS for the channel being tested, and (2) defeating the ΔT protection channel set that is being fed from the NIS channel, and (3) defeating the power mismatch section of T_{avg} control channels when the appropriate NIS channel is being tested. However, the Rod Position system and remaining NIS channels still provide the dropped-rod protection. Testing does not trip the system unless a trip condition exists in a concurrent channel.

The Functional Units having risk informed AOTs are identified with either (1) or (2) in column 6 of Tables 3.5-2 through 3.5-4. Risk informed AOTs for analog channels (72 hours) and logic channels (24 hours) are based on the analysis provided in Reference 5. Specification 3.5.3 allows the minimum degree of redundancy to be reduced by one for on-line testing (and corrective maintenance for inoperable instrumentation discovered during the surveillance testing) of instruments with installed bypass capability. For analog channels, this test bypass allowance is limited to 12 hours consistent with Reference 5. For logic channels, this test bypass allowance is limited to eight hours as provided in Note # of Tables 3.5-2 and 3.5-3 and consistent with Reference 5. At the end of this test bypass allowance, the requirements of Tables 3.5-2 through 3.5-4 and associated notes must be complied with. The test bypass allowance does not apply to the performance of preventative maintenance or performance of maintenance for inoperable instrumentation discovered by other means than the performance of a surveillance test.

Corrective (and not preventative) maintenance is permitted on a logic channel provided that the redundant channel is operable. For the RPS, 24 hours of such maintenance is permitted for the logic channel. This same 24 hour corrective maintenance period is permitted for the trip breaker if the logic channel requires maintenance at the same time.

References

(1) UFSAR Section 7.2

(2) UFSAR Section 14.3

(3) UFSAR Section 14.2.5

(4) Safety Evaluation accompanying the Indian Point Unit No. 2 "Application for Amendment to Operating License," sworn to on May 29, 1979 by Mr. William J. Cahill, Jr. of Consolidated Edison.

(5) WCAP-14333, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times"

Table 3.5-1

Engineered Safety Features Initiation Instrument Setting Limits

No.	Functional Unit	Channel	Setting Limits
1.	High Containment Pressure (Hi Level)	Safety Injection	≤ 2.0 psig
2.	High Containment Pressure (Hi-Hi Level)	a. Containment Spray b. Steam Line Isolation	≤ 24 psig
3.	Pressurizer Low Pressure	Safety Injection	≥ 1833 psig
4.	High Differential Pressure Between Steam Lines	Safety Injection	≤ 155 psi
5.	High Steam Flow in 2/4 Steam Lines Coincident with Low T_{avg} or Low Steam Line Pressure	a. Safety Injection b. Steam Line Isolation	≤ 40% of full steam flow at zero load ≤ 40% of full steam flow at 20% load ≤ 110% of full steam flow at full load ≥ 540°F T_{avg} ≥ 525 psig steam line pressure
6.	Steam Generator Water Level (Low-Low)	Auxiliary Feedwater	≥ 7% of narrow range instrument span each steam generator
7.	Station Blackout (Undervoltage)	Auxiliary Feedwater	≥ 40% nominal voltage
8a.	480V Emergency Bus Undervoltage (Loss of Voltage)	------	220V + 100V, -20V 3 sec ± 1 sec
8b.	480V Emergency Bus Undervoltage (Degraded Voltage)	------	421V ± 6V 180 sec ± 30 sec (no SI) 10 sec ± 2 sec (coincident

165

Table 3.5-2

Reactor Trip Instrumentation Limiting Operating Conditions

No.	Functional Unit	1 No. of Channels	2 No. of Channels to Trip	3 Min. Operable Channels	4 Min. Degree of Redun-dancy	5 Operator Action if Conditions of Column 3 Cannot be Met	6 Operator Action if Conditions of Column 4 Cannot be Met
1.	Manual	2	1	1	0	Maintain hot shutdown	Same as Column 5
2.	Nuclear Flux Power Range	4	2	3	2	Maintain hot shutdown	(1)
2.a	Nuclear Flux Power Range	4	2	2	1	Maintain hot physics tests only	Same as Column 5
3.	Nuclear Flux Intermediate Range	2	1	1*	0	Maintain hot shutdown	Same as Column 5
4.	Nuclear Flux Source Range	2	1	1**	0	Maintain hot shutdown	Same as Column 5
5.	Overtemperature delta T	4	2	3	2	Maintain hot shutdown	(1)
6.	Overpower delta T	4	2	3	2	Maintain hot shutdown	(1)
7.	Low Pressurizer Pressure	4	2	3	2	Maintain hot shutdown	(1)

Table 3.5-2

Reactor Trip Instrumentation Limiting Operating Conditions

No.	Functional Unit	1 No. of Channels	2 No. of Channels to Trip	3 Min. Operable Channels	4 Min. Degree of Redun-dancy	5 Operator Action if Conditions of Column 3 Cannot be Met	6 Operator Action if Conditions of Column 4 Cannot be Met
8.	Hi Pressurizer Pressure	3	2	2	1	Maintain hot shutdown	(1)
9.	Pressurizer Hi Water Level	3	2	2	1	Maintain hot shutdown	(1)
10.	Low Flow Loop ≥ 75% F.P. Low Flow Two Loops 10-75% F.P.	3/loop 3/loop	2/loop (any loop) 2/loop (any two loops)	2/operable Loop 2/operable loop	1/operable loop 1/operable loop	Maintain hot shutdown	(1)
11.	Lo-Lo Steam Generator Water Level	3/loop	2/loop	2/loop	1/loop	Maintain hot shutdown	(1)
12.	Undervoltage 6.9 kV Bus	1/bus	2	3	2	Maintain hot shutdown	(1)
13.	Low frequency 6.9 kV Bus	1/bus	2	3	2	Maintain hot shutdown***	(1)
14.	Quadrant power tilt monitors	2	NA	1	0	Log individual upper and lower ion chamber currents once/shift and after load change > 10%	Same as Column 5

Table 3.5-2

Reactor Trip Instrumentation Limiting Operating Conditions

No.	Functional Unit	1 No. of Channels	2 No. of Channels to Trip	3 Min. Operable Channels	4 Min. Degree of Redun-dancy	5 Operator Action if Conditions of Column 3 Cannot be Met	6 Operator Action if Conditions of Column 4 Cannot be Met
15.	DELETED						
16.	Control Rod Protection****	3	2	2	1	During RCS cooldown, manually open reactor trip breakers prior to T_{cold} decreasing below 381°F. Maintain reactor trip breakers open during RCS cool-down when T_{cold} is less than 381°F.	Same as Column 5
17.	Turbine Trip ≥ 35% F.P. A. Low Auto Stop Oil Pressure	3	2	2	1	Maintain reactor power below 35% F.P.	(2)
18.	Reactor Trip Logic	2	1	2#	1#	Be in hot shutdown within the next six hours.	Same as Column 5

Table 3.5-2

Reactor Trip Instrumentation Limiting Operating Conditions

No.	Functional Unit	1 No. of Channels	2 No. of Channels to Trip	3 Min. Operable Channels	4 Min. Degree of Redun-dancy	5 Operator Action if Conditions of Column 3 Cannot be Met	6 Operator Action if Conditions of Column 4 Cannot be Met
19.	Reactor Trip Breakers	2	1	2#	1#	With either diverse trip feature inoperable, or the breaker incapable of tripping for any other reason, restore it to operable conditions or, be in hot shutdown within the next six hours and open both reactor trip breakers. The breaker shall not be bypassed except for the time required for performing maintenance and/or testing to restore it to operability.	Same as Column 5

Table 3.5-2

Reactor Trip Instrumentation Limiting Operating Conditions

F.P. = Rated Power

* If two of four power range channels are greater than 10% F.P., channels are not required.

** If one of two intermediate range channels is greater than 10^{-10} amps, channels are not required.

*** 2/4 trips all four reactor coolant pumps.

**** Required only when control rods are positioned in core locations containing LOPAR fuel.

\# A reactor trip breaker and/or associated logic channel may be bypassed for maintenance or surveillance testing for up to eight hours provided the redundant reactor trip breaker and/or associated logic channel is operable.

A reactor trip breaker and associated logic channel may be bypassed for corrective maintenance for up to 24 hours if corrective maintenance is required on the logic channel, provided the redundant reactor trip breaker and/or associated logic channel is operable.

(1) Restore all channels as required by column 1 to an OPERABLE status within 72 hours or place the inoperable channel in trip. Otherwise, maintain hot shutdown.

(2) Restore all channels as required by column 1 to an OPERABLE status within 72 hours or place the inoperable channel in trip. Otherwise, reduce reactor power below 35%.

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