BOSTON Edison COMPAN

NUCLEAR OPERATIONS DEPARTMENT
PILGRIM NUCLEAR POWER STATION
Procedure 1.3.37

POST TRIP REVIEWS

List of Effective Pages

List of Attachments

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I. PURPOSE

To provide instructions to personnel to perform Post TRIP Reviews (PTR) following unplanned reactor trips. Adherence to this procedure will ensure that consistent data will be collected, and that a uniform analysis and decision process will be applied after a reactor trip and prior to granting permission to restart.

II. DISCUSSION

The analysis of the Salem Nuclear Station event of 1983 revealed that an unrecognized failure to scram (ATWS) event took place. Evaluations of the event by both the NRC and INPO have resulted in recommendations aimed at standardizing post trip reviews. BECo commitments to both INPO and NRC are directed to proceduralizing and standardizing post trip reviews. This procedure will formalize the existing post trip review method at PNPS.

The purpose of a post trip review is to determine the plant's readiness to return to power after an unscheduled reactor trip. Station personnel must reasonably determine the cause of the trip, verify proper functioning of safety related and other equipment during the trip, and ensure that the trip did not have a detrimental effect on the plant.

Post trip reviews can also serve to provide lessons learned to the plant staff and other utilities.

III. REFERENCES

A. INPO Good Practice OP-211 "Post Trip Reviews" Draft.

B. US NRC Generic Letter 83-28 "Generic Implications of Salem ATWS Events"

C. BECo letter "Response to Generic Letter 83-28"

D. Nuclear Operations Procedure NOP 3301 "Conduct of Operations"

E. PNPS Operations Manual procedures:

1. 1.3.3 "Authority to Shutdown and Startup Station"
2. 1.3.9 "Reports"
3. 1.3.12 "Notification and Recall of Personnel"
4. 2.2.17 "Communications"

IV. APPLICABILITY

The requirements of this procedure will apply to all unplanned trips from the "RUN" mode. The Chief Operations Engineer or the Nuclear Operations Manager may require that elements of this procedure be followed for other unit problems from other operating conditions, such as unexplained power reductions.

V. PREREQUISITES

A. The post trip review (PTR) will be initiated after plant conditions are stabilized. The PTR shall not distract the Watch Engineer, the Operating Supervisor, the STA or operating personnel from their primary responsibility of monitoring plant parameters and maintaining the plant in a safe condition.

B. Sufficient information shall be collected from personnel involved in the unit trip prior to permitting relief by the oncoming shift.

VI. RESPONSIBILITIES

A. Nuclear Watch Engineer is responsible for ensuring that the post trip review is initiated. He is also responsible, along with the NOS and STA, for the investigation phase of the PTR and review of the results.

B. Nuclear Operating Supervisor is responsible as part of the investigation phase to record the actions taken and preliminary information relating to the initiating event. He is also responsible for directing the Shift Administrative Assistant in obtaining statements from operating personnel and others involved in the trip.

C. Shift Technical Advisor is responsible with the NWE and NOS for trip investigation. This includes interpretation of the Process Computer Data Recall Log and Sequence of Event Log. Additionally the STA will record the data required on Attachment B.

D. Shift Administrative Assistant is responsible for collecting the required recorder charts and obtaining statements of involved personnel. Additionally the SAA is responsible to ensure that the PTR package is retained for records and distributed to the Nuclear Operations Manager and the OSS&P Group Leader.

E. Operations Personnel and others (i.e., I&C Technicians) involved in the unplanned trip are responsible for providing the SAA with objective statements that describe their observations of and/or participation in the trip.

F. The Nuclear Operations Manager or his designate is responsible to authorize restart of the reactor.

G. The ORC shall review all post trip reviews. In the case of trips classified as Type 1 or 2, the ORC shall review the trip at its next scheduled meeting. If the trip is classified as Type 3, the NOM will convene ORC to provide independent assessment of the event.

VII. POST TRIP REVIEW PROCEDURE

NOTE: Event notification to appropriate agencies or persons shall be made consistent with procedure 2.2.17 "Communications".

A. General

Post trip review is a 5 step process as follows:

1. Data Collection
2. Trip Investigation
3. Event Classification/Safety Assessment
4. Restart Authorization
5. Information Feedback

B. Data Collection

1. The object of the data collection is to assemble enough information to reconstruct the trip, assess the response of systems, and identify the root cause of the event.

2. The SAA shall collect the following hard copy data:

   a. Process Computer Data Recall Log (Attachment F identifies points on this Log)

   b. Process Computer Sequence of Events Log and Alarm Typer Output

   c. Recorder charts from:

      i. One APRM recorder
      ii. Feedwater flow
      iii. Wide (or) narrow range reactor pressure
      iv. Reactor level
      v. Other recorders as identified by the STA or NWE

3. When recorder charts are collected, photo copies of the original strip may be made. Chart speed, reference time, date, scale values and pen channels will be recorded on the chart or copy.

4. The SAA in conjunction with or under the direction of the NOS shall record statements from each individual involved in the event. These statements shall be obtained only after the plant is in a stable condition. Statements should include facts concerning the event relative to pretrip conditions or activities, initial indications of a problem, initial actions taken, equipment automatic and manual operation, observed malfunctions or procedural deficiencies.

5. It may be appropriate for the NWE to interview involved personnel or to collect information from assembled individuals as a group.

6. The STA shall complete part 1 of the PTR DATA Summary (Attachment B). If additional information is needed it shall be obtained at this time.

7. Collected data and personnel statements shall be assembled into a package and delivered to the NWE to begin the trip investigation.

C. Trip Investigation (Attachment C)

1. The NWE, NOS, and STA shall reconstruct the event by preparing a chronology of the event. (Attachment C, part 1)

2. The NWE and NOS shall review the data package for proper system performance and note that appropriate automatic functions and equipment operation took place. The reviewers should look beyond the obvious indications to diagnose the cause of the trip and determine acceptability to restart the unit.

3. The STA shall analyze the data to determine if critical parameters remained within the bounds of the FSAR or the Cycle Reload Analysis. Peak reactor pressure, lowest water level, drywell pressure, steamline radiation are examples of what shall be recorded for this analysis. Departures from the bounds of the FSAR or reload analysis shall be noted on part 3 of Attachment C and brought to the immediate attention of the NWE. A potential unreviewed safety question may exist.

4. The NWE will complete the Scram Report (Attachment A) and summarize the event. The scram report shall identify the probable cause of the event, and identification of systems with inadequate performance (if applicable). This Scram Report shall become the cover sheet for the PTR package.

5. Any equipment or processes identified with inadequate performance or abnormal response shall be reported on separate Failure and Malfunction Reports according to procedure 1.3.24.

D. Event Classification/Safety Assessment

1. The preliminary Safety Assessment (Attachment D) form shall be completed by the STA and Watch Engineer.

2. The Event shall be classified as Type 1, 2, or 3 by the following criteria:

   a. Type 1 - The cause of the trip is positively known and has been or is in the process of being, corrected; all safety related and other important equipment functioned properly during the trip.

   b. Type 2 - The cause of the trip is positively known and has been or is in the process of being, corrected except some safety related or other important equipment did not function properly. The malfunction has been or is in the process of correction or a Tech. Spec. Constraint does not prohibit startup.

   c. Type 3 - The cause of the trip is not positively known and/or some safety related or important equipment functioned abnormally during the trip, or the malfunction cannot be readily corrected, or startup is precluded due to Tech. Spec. Constraints, or the transient did not remain within the bounds of the FSAR or Reload Analysis.

3. The NOM and the COE will be notified of the classification of the event and their concurrence shall be obtained. If the event is classified as Type 3, the COE or his designate will take charge of the investigation until the cause of the event and corrective action has been determined.

E. Restart Authorization

1. The NOM will be informed of the results of the PTR and classification. The NOM may then authorize restart if the event is classified as Type 1 or 2.

2. If the NOM is not satisfied with the results of the PTR he will take actions necessary to satisfy his concerns.

3. In the case of a Type 3 event the NOM will convene the ORC for further evaluation and independent review of the event prior to authorizing restart.

4. In the case of Type 1 or 2 events the ORC will review the PTR at the next scheduled meeting.

F. Information Feedback

1. After the PTR is completed, the SAA will ensure that the assembled package is forwarded as follows:

a. Original to the NOM

b. Copy to ORC Secretary for ORC review

c. Copy to OSS&P Group Leader - for operating experience review per NOP 8401

2. After ORC review is complete the OSS&P Group Leader will review the PTR package, summarize it and route the information within the Nuclear Organization. Additionally any information of general interest to the industry will be entered into the operating experience category of NUCLEAR NOTEPAD.

VIII. ACCEPTANCE CRITERIA

A. The PTR will be performed as described in the procedure.

B. Event Classification will key appropriate actions prior to restart of the unit.

IX. ATTACHMENTS

A. Scram Report

B. PTR Data Summary

C. Investigation and Evaluation

D. Preliminary Safety Assessment/Startup Authorization

E. Sample PTR Package Organization

F. Data Recall Log Point ID Summary

BOSTON EDISON COMPANY

PILGRIM STATION

RTYPE · G7

SCRAM REPORT

Date ______ Time ______ Number ______

—CAUSE—	—MODE SWITCH— POSITION	—REACTOR STATUS—	—FLOW CONTROL STATUS—
☐ Operator Error ☐ Testing Error ☐ Equipment Failure ☐ Other ______	☐ Run ☐ Start-Up ☐ Refuel	☐ Critical ______ MWT ☐ Subcritical	☐ Loop Manual ☐ Master Manual ☐ Master Automatic

—STATION STATUS—	—SCRAM TIMES— (for monitored CRD'S)	—STATION CONDITIONS—
☐ Starting-Up ☐ Shutting-Down ☐ Changing Power ☐ Steady State ☐ Surveillance Testing ☐ Other ______	☐ none available Average ______ Sec. Fastest Rod ______ Time ______ Sec. Slowest Rod ______ Time ______ Sec.	Core Flow (263-10) ______ Lb/ Reactor Pressure (640-27) ______ Ps Steam Flow (640-27) ______ Lb/ Vessel Level (640-26) ______ Inch Feedwater Flow (640-26) ______ Lb/ Off Gas Activity Level ______ Mr/ Stack Gas Activity Level ______ C Building Vent Activity Level ______ C Generator Output ______ MWe (gross) ☐

Prompt Notification Via Telephone Required ______ T/Limit ______ T/Notified ______ Person Contacted ______

On Call Individual ______ Required ______ 1 hr. ______

______ NRC ______ Required ______ 1 hr. ______

Brief Description of Scram ______

Corrective Action Taken ______

Watch Engineer ______

1.3.37A-1

BECO. FORM X5066

ORIGINAL · STATION MANAGER TO D.C.C.

PNPS PTR DATA SUMMARY

Date of Occurrence ___________ Time of Occurrence ___________

By STA: _______________________

_______________/_______________
Date Time

Part 1 INITIAL CONDITIONS

The status of safety systems and a selected set of important plant parameters, pump running combinations control switch positions, chemistry results, and radiation readings that exist prior to the unscheduled reactor trip must be recorded. The data to be selected should be based on the following considerations:

- o the data is not directly available on control room strip charts or computer printouts
- o the data is necessary to ascertain the cause of the trip or abnormal response and proper functioning of safety-related equipment
- o the data is necessary to effectively reconstruct plant status prior to the trip

<u>Examples</u>

(a)	Reactor Power (mw_{th})*	__________%
(b)	Unit Generator Load*	__________
(c)	Mode Switch Position*	__________
(d)	Reactor Vessel Pressure*	__________
(e)	Reactor Feed Pumps Operating (Circle)	A B C
(f)	Vessel Level* ___ on Instrument _______	__________ in
(g)	Main Circulating Water Pumps Running (Circle)	A B

* Also contained on Scram Report

(h) Status of Control Stations (Circle)

1. Recirculation pump control mode (Circle)	Master Manual	Local Manual
2. Vessel level control (Circle)	Auto	Manual
3. Turbine Control		
Pressure Setpoint	__________	
Load Limiter Setpoint	__________	

(i) Torus Temperature (C7) __________ °F

(j) Off normal status of any trains/ portions of a safety system prior to event :

From Oper. 28 — Details

1. RPS ____________
2. ECCS ____________
3. SBGTS ____________
4. Emergency Buses/Diesels ____________
5. DC Buses ____________

1. Testing/Surveillances in Progress from Oper. 28 or other

Test Number	Status/Step

Part 2 PLANT RESPONSE

Data selected to be documented for determining plant response should include the following:

- o list of strip charts to be retained
- o printouts from devices such as the process control computer, alarm printer, and event recorders
- o safety systems activations and performance information
- o manual, radiological, and control system actions

Examples

(a) Obtain a copy of the applicable parameter plots given below for every Event:

Panel

905 1. 1 Channel APRM (recorders 750-10A, B, C or D)

905 2. Recorder 640-26 - Reactor Vessel Level (black pen), Feedwater Flow (red pen)

905 3. Reactor Vessel Pressure

a. Narrow range - 640-28 (red pen)

b. Wide range - 640-27 (black pen)
Steam flow is also on this recorder.

(b) Optionally the following may be requested depending on the Event:

Panel		Parameter
905	1.	Core Flow FR 263-110
C-1	2.	Control valve position ZR 9027
903	3.	Recirc. pump suction TR 151 A and B
904	4.	Drywell pressure TRU 9045
905	5.	Torus Level LR 5038

6. Relief/Safety Valve Temperature TR 260-20

7. Reactor Water conductivity CAS-129-25

8. Main Steam Line Radiation PR 1705-11

9. ECCS System Performance (if system actuated)

<u>Parameter</u>

HPCI Flow
RCIC Flow
LPCI Flow
CS Flow

10. Condenser Vacuum PR 3392

11. Vessel Metal Temperature

(c) Obtain a printout from:

1. Process Computer Sequence of Events Log

2. Process Computer Data Recall Log

3. Alarm Printer Output (with c.1 above)

4. Control Rod Position (program OD-7 output)

(d) Safety System Actuation and Performance

1. Reactor Protection System

RPS Trips giving scram _______________ Actuation Time __:__

2. Containment Isolations

	Cause	Time
Group I	_____	____
Group II	_____	____
Group III	_____	____
Group IV	_____	____
Group V	_____	____
Group VI	_____	____

3. ECCS		Actuation Signal	Time
	HPCI	______	______
	RCIC	______	______
	CS	______	______
	LPCI	______	______
	ADS	______	______

(e) Control Systems (Circle)

1. Turbine Runback	Yes	No
2. Turbine Trip Time ____ Signal ____	Yes	No
3. Recirculation Pump Runback	Yes	No
4. Recirculation Pump Trip	Yes	No

Cause ______________________________
____
____

(f) Manual Actions

Were any control stations taken from auto to manual? (Specify station time at time/sequence) Yes No
____
____
____
____

(g) Radiological Response (include abnormal area radiation monitoring, process radiation monitoring and environmental radiation monitoring indications ______________
____

(h) Chemistry Response 1. Reactor Coolant Chemistry
____
____

Other STA Comments:

____
____
____
____
____
____

INVESTIGATION AND EVALUATION

Date ______ ______ Time ______________

Part 1 Chronology of Event

Attach written copy

Part 2 PROBABLE CAUSE OF TRIP ______________________________

Comments: ______________________________

Attach statements of involved personnel.

By Watch Engineer ______________________________ NOS ______ STA ______

Part 3 UNEXPECTED ASPECT OF TRANSIENT BEHAVIOR (if event compared with previous similar transient note, the transient with which compared)

Reload Analysis or FSAR Transient Page Number <u>Compared With</u> ______________

-or- Previous Trip on ______________ / ______________

Date Time

____

By: STA ______________________________ F&M issued if required by: ______________

Part 4 IDENTIFICATION OF SYSTEMS WITH INADEQUATE PERFORMANCE

System/Component	Description of Problem
______________	______________________________
______________	______________________________
______________	______________________________

______________________________ / ______________

WE Signature Date Time

______________________________ / ______________

NOS Signature Date Time

Note: A separate F&M is required for each malfunction above.

Attachment C
Page 1 of 1

PRELIMINARY SAFETY ASSESSMENT
STARTUP AUTHORIZATION

Part 1 TRANSIENT DATA FOR PERTINENT PLANT PARAMETERS

	Maximum	Minimum
(a) RCS pressure	Loop A___ B___	Loop A___ B___
(b) Reactor vessel water level		________
(c) Reactor coolant flow	Loop A___ B___	Loop A___ B___
(d) Reactor core thermal power	________	

Part 2 PRELIMINARY SAFETY ASSESSMENT (Circle)

(a) RCS pressure remained above 880 Yes No
(b) Reactor isolation occurred Yes No
(c) RCS pressure increased to safety/ relief valve operating pressure Yes No
(d) RCS temperature decrease less than $100^{o}F/hr$ Yes No
(e) HPCI/RCIC initiated Yes No
(f) ADS timer initiated Yes No
(g) Primary containment press ________ temp ________
(h) Torus water level ________ temp ________

Part 3 EVENT CONDITION

Classify trip as 1, 2 or 3 according to guidelines in procedure section.

The event on ______________ (Date) at ___:___ (Time) is a condition ______________ (I, II, III)

Signature indicates agreement with condition.

________________________ / ________
Watch Engineer Date Time

________________________ / ________
STA Date Time

NOM ________________________ COE ________________________
Notification

NOM notif.ed of event classification ____________ / ________
Date Time

NOM concurrence? YES ____ NO ____

If Type 3 COE notified to take over investigation

Time____________ By ____________________________________

Part 4 STARTUP AUTHORIZATION

Class 1, 2 EVENTS

Plant manager notified and permission granted to startup the reactor.

____________________________ ________/______

Watch Engineer Date Time

Comments: ____

____

Class 3 EVENT

ORC review of event on ______________, meeting number ______________________

Minutes of the meeting(s) are attached.

____________________________ ________________

ORC Chairman Date

Permission is granted to startup the reactor

____________________________ ________/______

NOM Date Time

Comments: ____

____

____

____

____

SAMPLE PTR PACKAGE
ORGANIZATION

1. Scram Report (Attachment A)
2. PTR Data Summary (Attachment B)
3. Recorder Chart Copies
4. Investigation Evaluation (Attachment C)
5. Chronology of Event (separate sheet)
6. Statement of Involved Personnel (separate sheet)
7. Preliminary Safety Assessment/Startup Authorization (Attachment D)

DATA RECALL LOG POINT ID SUMMARY

LINE 1

B000	APRM "A" (% PWR)
B001	APRM "C" (% PWR)
B013	Rx. Pressure (PSIG)
B014	Core Plate D/P (PSI)
B015	Rx. Core Flow (M#/H)
B017	CRD Flow (M#/H)
B018	Rx. FW Inlet Flow "A" (M#/H)
B019	Rx. FW Inlet Flow "B" (M#/H)
B024	Rx. Water Level
B025	Rx. Outlet Steam Flow (M#/H)
B028	FW Inlet Temp. "A1" (°F)
B030	FW Inlet Temp. "B1" (°F)
B038	Recirc. Flow Loop "A2" (M#/H)
B039	Recirc. Flow Loop "B1" (M#/H)
C002	Rx. Saturation Temp. (°F)
C023	Seawater Flow (GPM)
C027	Hotwell Outlet Temp. (°F)
C050	Condenser Δ T (°F)
M071	Torus Level

LINE 2

G012	Stator Cooler Inlet Temp (°F)
G013	Stator Cooler Outlet temp (°F)
G019	Alternator Air to Cooler (°F)
G020	Alternator Air from Cooler (°F)
F002	Condensate Demineralizer D/P
F007	RFP Suction Pressure (PSIG)
F011	Condensate Pump Header Pressure
F012	West Condenser Pressure (In Hg)
F013	East Condenser Pressure (In Hg)
F028	RBCCW Loop "A" Flow
F029	RBCCW Loop "B" Flow
F030	RBCCW To RHR Hx. Loop "A" Flow
F031	RBCCW To RHR Hx. Loop "B" Flow
F077	RBCCW Hx. Outlet Temp. "A"
F078	RBCCW Hx. "B" Outlet Temp
M034	Torus Pressure
M035	Drywell Pressure
M042	SSW Flow Loop "A"
M043	SSW Flow Loop "B"