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50-270 Oconee Nuclear Station, Unit 2, Duke Power Co. 05000270
50-287 Oconee Nuclear Station, Unit 3, Duke Power Co. 05000287

AUTH.NAME	AUTHOR AFFILIATION
PARKER,W.O.	Duke Power Co.
RECIP.NAME	RECIPIENT AFFILIATION
DENTON,H.R.	Office of Nuclear Reactor Regulation, Director
EISENHUT,D.G.	Division of Licensing

SUBJECT: Forwards "Reactor Vessel Pressurized Thermal Shock Evaluation." Summary of responses to NRC 810821 & 1218 requests encl. S66 RPT

DISTRIBUTION CODE: A049S COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 44+253
TITLE: Thermal Shock to Reactor Vessel

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	NRR/DHFS DIR		1	1	NRR/DHFS/PTRB		1	1
	NRR/DL DIR		1	1	NRR/DL/ADSA		1	1
	NRR/DL/ORAB	11	1	0	NRR/DSI DIR		1	1
	NRR/DSI/RAB		1	1	NRR/DSI/RSB		1	1
	NRR/DST DIR		1	1	NRR/DST/GIB		1	1
	REG FILE	05	1	1	RES BASDEKAS		1	1
	RES VAGINS,M		1	1	RES/DET		1	1
	RES/DRA		1	1				
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DUKE POWER COMPANY
POWER BUILDING
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WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

January 15, 1982

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. D. G. Eisenhut, Director
Division of Licensing

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Sir:

By letter dated August 21, 1981, the Staff provided its understanding of the pressurized thermal shock concern. Certain specific information was requested 60 days after the above letter and was provided by a Duke letter dated October 20, 1981. Additionally, within 150 days of the above NRC letter, the Staff requested a plan that would define actions and schedules for resolution of this issue and analyses supporting continued operation. The Staff letter dated December 18, 1981 requested certain additional information be provided with the "150-day" response.

As the Staff is aware, Duke has made several submittals regarding this issue over the last year. Based on the generic analysis that was conducted in 1979-80, Duke determined that a plant specific evaluation would be necessary to appropriately characterize the thermal shock events for Oconee. This effort was initiated early in 1981 and is provided as an attachment to this letter. The attached report entitled "Oconee Nuclear Station, Reactor Vessel Pressurized Thermal Shock Evaluation", DPC-RS-1001, provides the basis upon which Duke Power Company has determined that continued safe operation of Oconee Unit 1 is justifiable through the remaining service life of the plant or through about the year 2007. Similar results are expected for the remaining two Oconee units.

This report provides an extensive evaluation of the pressurized reactor vessel thermal shock concern for several postulated accidents as specifically applied to the Oconee 1 reactor vessel. Assumptions of material properties used in the evaluation are based on data obtained through the B&W Owners Integrated Reactor Vessel Materials Surveillance Program. The assumed operator actions are consistent with existing procedures and are assumed to be performed in a time frame that is consistent with the actions required.

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The results of this evaluation indicate that the Oconee 1 reactor vessel integrity would not be compromised due to anticipated transients, moderate frequency events, and a number of low probability accidents during the anticipated lifetime of the reactor. Also, a detailed evaluation of various

DUKE POWER COMPANY
OCONEE NUCLEAR STATION
January 15, 1982 Submittal
Attachment 1

The following summarizes the Duke Power Company responses to these Staff requests concerning pressurized thermal shock:

NRC letter dated August 21, 1981
Enclosure

1. Geometry - In addition to discussions contained in the attached report, geometrical descriptions are provided in Chapters 3 and 4 of the Oconee FSAR; BAW-1511P, "Irradiation-Induced Reduction in Charpy Upper-Shelf Energy of Reactor Vessel Welds", submitted by Duke letter dated March 23, 1981, and BAW-1436, "Analysis of Capsule OCI-E, Oconee Nuclear Station, Unit 1", submitted by Duke letter dated February 21, 1978 and referenced in NRC Safety Evaluation Report dated March 19, 1979. (License Amendments DPR-38-71, DPR-47-71, and DPR-55-68)

2. Material Description - Same as above

3. Neutron Source - Same as above

4. Vessel Fluence - Same as above

5. Surveillance Capsules - In addition to discussions contained in the attached report, capsule descriptions are contained in BAW-1436 and BAW-10006, Rev. 3, Reactor Vessel Material Surveillance Program, approved by the NRC by letter dated January 15, 1975 and submitted on the Oconee dockets by Duke letter dated August 31, 1977. The Staff evaluation of the vessel material surveillance program is contained in amendments 44, 44, and 41 to the Oconee Facility Operating Licenses.

6. Vessel Welds - Same as above

7. Systems Analysis - Contained in the attached report

NRC letter dated December 18, 1981
Enclosure (1)

Duke responses to Items 1 and 2 of the NRC August 21, 1981 letter were accepted by the Staff.

3 & 4. RT_{NDT} Limit and Basis for the Limit -

Duke Power suggests that the evaluation required to assure acceptable long-term low risk of vessel failure due to pressurized thermal shock

be included within the currently established Appendix G to 10 CFR 50. Such an approach would require defining an appropriate overcooling transient response (temperature, pressure vs. time) of low enough probability of occurrence that would be periodically evaluated as required by Appendix G. This approach would recognize the changing techology used to evaluate reactor vessel integrity as well as accomodate changes in plant operations and equipment during plant service life.

5. Operator Actions - The Staff states that based on review of the Duke "60-day" response, the degree of emphasis which is currently placed on the need for changes in procedures, training, and management involvement could not be determined. An expanded response was requested.

   Response - Duke Power has been aggressively pursuing the resolution of this concern since early 1981. Based on the results of the analyses that are described in the attached document no major changes in existing plant procedures are considered necessary (the appropriate plant procedures are attached). The station procedure which will implement the Abnormal Transient Operating Guideline will include appropriate operator instructions for mitigation of overcooling transients. While the plant operating personnel have been made aware of the thermal shock concern, no formal training has been conducted on the issue. Duke considers that the sound knowledge the operators possess of the current plant procedures is acceptable for the short term. However, as alluded to in the transmittal letter, Duke intends to augment the operator understanding of the thermal shock concern. To this end, the necessary background information on reactor vessel integrity will be developed for incorporation as appropriate in both the classes for RO and SRO licences as well as the requalification classes. Concurrently, copies of the attached report have been provided to the Oconee Superintendent of Operations for immediate use in augmenting operator awareness of this issue.

NRC letter dated December 18, 1981

Enclosure (2)

1. Consistent with the time available and the tasks accomplished prior to this letter, the information requested concerning the identification of the PTS events considered and probability of occurrence is included in the attached report.

   All potential transients/accidents which are relevant to the pressurized thermal shock issue were investigated in the course of our analysis. Specifically, small break loss-of-coolant accident and overcooling transients induced by various secondary system malfunctions were considered. No new analysis was performed for loss-of-coolant accidents of a large break nature in consideration of the original design analysis.

initiating events and failures of important systems and operator actions which lead to severe thermal shock indicates that accident sequences beyond those considered in the reactor vessel integrity analysis are very low in frequency. When the conditional probabilities that these events lead to vessel failure and that the failure results in the inability to maintain core cooling are factored in, the contribution of pressurized thermal shock to the likelihood of core damage is seen to be quite small. Accordingly, Duke Power Company considers that the evaluations contained in this report adequately address the pressurized thermal shock concern for Oconee Unit 1 and that extensive additional analyses and evaluations of potential plant modifications are unnecessary at this time.

There are, however, three areas where additional attention and/or continued vigilance are necessary for amelioration of the reactor vessel integrity concern. First, it is important to ensure that the operators have sufficient training and that the procedures are adequate to recognize incipient overcooling events and to prevent the occurrence of severe thermal shock events. Although the analyses reported herein are based on actions required in the existing procedures, it is our intent to augment the operator understanding of the thermal shock behavior and to incorporate additional guidance in the procedures. Second, the ongoing RV material and fluence surveillance program will provide periodic confirmation that the accumulated fluence and changes in material properties are consistent with design predictions. Third, the operational occurrences will be monitored on a continuing basis to recognize any unanticipated precursor events and to confirm that the type and frequency of occurrence of events are acceptable. This is being achieved by the operating experience evaluation program now in place.

Duke Power Company considers that the information and references contained in the attached report are responsive to the Staff's request for additional information contained in the August 21, 1981 letter. By letter dated December 18, 1981, the Staff provided amplification to the initial request. Responses are provided to this second letter to the extent possible within the limited time which was made available. Attachment 1 provides a summary of the Duke Power Company responses to both of the Staff requests.

I declare under penalty of perjury that the statements set forth herein are true and correct to the best of my knowledge, executed on January 15, 1982.

Very truly yours,

William O. Parker, Jr.

RLG/jfw
Attachments

cc: NRC Regional Administrator
R. Breen, NSAC
T. Marston, EPRI
R. C. Kryter, ORNL
P. Higgins, AIF
J. Taylor, B&W
P. Rahe, Westinghouse
E. Scherer, Combustion Engineering
E. P. Wilkinson, INPO
R. Bernero, NRC/RES

The approach utilized in the identification of PTS occurring from secondary system malfunction is as follows:

For the overcooling transients, three general categories of events--steam generator feeds, steam pressure control failures including small steam line breaks and large steam line breaks were identified as having the potential for leading to pressurized thermal shock. Each of these categories were further scrutanized in the context of additional deliterious system failures.

Accident sequences which were deemed to have an extremely low probability of occurrence were eliminated from further consideration. The resulting thirteen cases of overcooling transients were analyzed in detail with respect to the transient thermal-hydraulic behavior. From the calculated thermal-hydraulic response of these thirteen events, four sequences were determined to be bounding and these were were selected for detailed vessel fracture mechanics analysis.

An evaluation of the frequency of occurrence of accident sequences which were analyzed and not analyzed with respect to vessel integrity was performed using the data, models and insight gained from the Oconee PRA program. This evaluation considered the frequency of initiating events, probabilities of system failures (based on fault tree analysis), and the unreliability of operator actions. This evaluation confirmed that those PTS accident sequences not considered in the vessel integrity analysis and which could pose a more severe threat to the vessel integrity have indeed very low frequency of occurrence (10^{-6} - 10^{-4} per reactor year). Furthermore, the conditional probability of vessel failure for these severe events are believed to be low (10^{-2} - 10^{-3}) as an upper bound. Therefore, there is reasonable assurance that vessel failure would not occur during the anticipated lifetime of the Oconee 1 reactor.

Additional details of this matter is contained in Chapters 2.0 and 9.0 of the attached report.

2. The shift to an 18-month LBP fuel cycle design with Oconee 1 Cycle 6 resulted in a significant decrease in peripheral fuel assembly power, with a corresponding decrease in core leakage flux and fluence at the vessel wall. Details of this effect on fluence are included in Chapter 7.0 of the attached document.

3. Based on the fracture mechanics results obtained for the various transient analyses, as performed on all vessel welds, no significant change in resulting service life is expected. The limited time available to perform a detailed analysis of sensitivity has precluded a more quantitative response.

4a. Detailed discussions of assumptions used in the transient analyses are included in Chapters 2.0 and 3.0 (Overcooling Transient Analyses and SBLOCA Analyses, respectively) of the attached report. Appropriate plant procedures, Loss of Reactor Coolant EP/0/A/1800/4 and Steam Supply System Rupture EP/0/A/1800/08, are attached to this letter. Similar information regarding operator guidance is contained in the draft Oconee Abnormal Transient Operating Guidelines, currently under Staff review. The training to establish operator readiness to cope with PTS events is described in the supplemental response to item 5 of the August 21, 1981 letter provided earlier in this submittal.

b. The effects of warm prestressing are discussed in Chapter 8.0 of the attached report. While credit for warm prestressing extends the allowable service life for the vessel to 32 EFPY for SBLOCA, a similar service life would be expected for postulated SBLOCA transients if downcomer temperatures predicted by improved mix codes currently in development are warm enough.

In addition to the above, the Staff requested that four possible actions be considered and evaluated. Based on the results contained in the attached report, no plant modifications are considered necessary. The following specific statements are provided in response to the four possible actions:

1. Duke Power Company considers that the current fuel management program, as described in the attached report, eliminates any need for further fuel management changes to reduce the rate of neutron radiation damage.

2. Duke Power Company considers that the results contained in the attached report support the decision to maintain ECC injection water temperature at the current levels for at least the next several years. With improved mixing calculational techniques currently in development, the current water temperature may be acceptable for the remaining service life of the plant.

3. Duke Power Company has supported an EPRI sponsored research program to evaluate in-place annealing, and will be reviewing the results of such work as they become available to determine feasibility of implementation. However, based on the results contained in the attached report, it is considered that in-place annealing is not required to assure safe operation of Oconee 1 through the design lifetime of the plant.

4. Duke Power Company considers that, based on the analyses conducted to date and documented in the attached report, no control system

changes are necessary to mitigate thermal shock or to control repressurization. Operator actions that are realistic, along with existing plant information and procedures, are sufficient to mitigate such transients and to reduce the likelihood of vessel failure.

Form SPD-1002-1

DUKE POWER COMPANY
PROCEDURE PREPARATION
PROCESS RECORD

(1) ID No: EP/0/A/1800/08
Change(s) 12 to
-- Incorporated

(2) STATION: OCONEE NUCLEAR STATION

(3) PROCEDURE TITLE: STEAM SUPPLY SYSTEM RUPTURE

(4) PREPARED BY: Dennis I Gordon DATE: 12-31-81

(5) REVIEWED BY: Neal Edwards DATE: 12/31/81

Cross-Disciplinary Review By: ______ N/R: NFE

(6) TEMPORARY APPROVAL (IF NECESSARY):

By: ______ (SRO) Date: ______

By: ______ Date: ______

(7) APPROVED BY: J N Pope Date: 12-31-81

(8) MISCELLANEOUS:

Reviewed/~~Approved~~ By: Ed Payne Date: 12-31-81

Reviewed/~~Approved~~ By: [illegible] Date: 12/31/81

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

STEAM SUPPLY SYSTEM RUPTURE

1.0 Symptoms

1.1 Unexpected increase in Reactor Power.

1.2 Rapid decrease in T_{av}, Pressurizer level, Reactor Coolant Pressure, Steam Pressure, and Electrical Load.

1.3 Increased Reactor Building pressure and temperature (line break in R.B.).

1.4 Increased Penetration Room pressure and Temp (line break in P.R.).

2.0 Immediate Action

2.1 Automatic

2.1.1 Reactor trip/Turbine-Generator Trip

2.1.2 Possible ES actuation (Channels 1 through 8).

2.2 Manual

2.2.1 If ES Channels 1 & 2 have actuated because of a low pressure condition in the RC System, TRIP all RC Pumps.

2.2.2 Isolate Both Steam generators by:
Placing MAIN, STARTUP FDW, EMERG FDW, and TURBINE BYPASS valves in manual and close.

2.2.3 Verify automatic actions have occurred; if not, perform steps manually.

CAUTION: Do not override Automatic Actions of Engineered Safety Features unless con-

tinued operation will result in unsafe plant conditions or will threaten reactor vessel integrity. (Refer to Enclosure 1).

3.0 Subsequent Action

3.1 Immediately on completion of necessary immediate manual action steps, alternate instrument channels, if available, shall be checked to confirm the key parameter readings that are asterisk (*).

3.2 Determine affected steam generator by monitoring S/G level and S/G pressure. The affected steam generator will have a more rapid decrease in both level and pressure.

3.3 Using the STARTUP FDW or EMER FDW valve on the unaffected Steam Generator, slowly fill to bring level to:

25 inches S/U level (RCP's on)

50% Op Range (RCP's off)

and place valve in Auto.

CAUTION: High RB temperatures will cause level instruments to indicate high (Encl. 2).

3.4 Maintain reactor coolant temperature constant with turbine bypass valves on unaffected steam generator, to prevent over filling of pressurizer.

3.5 Monitor nuclear instruments (*) to insure neutron level is decreasing.

3.6 If the RC System is 50 degrees subcooled, start one (1) RC pump in each loop per OP/1103/06, Reactor Coolant pump operation.

or

3.7 Verify natural circulation is achieved by insuring the following:

3.7.1 RCS is $50^{O}F$ subcooled on Subcooled Margin Monitors (*) and remains so. T_h indication (*) and Enclosure #1 can also be used to determine subcooled margin. Backup readings for Subcooled Margin Monitors are located on Display Group 22.

3.7.2 Feedwater level is at ∿ 50% on the operating range. (*)

3.7.3 The Δt between T_{hot} (*) and T_{cold} (*) is ∿ $30^{O}F$ to $40^{O}F$.

3.7.4 Main steam pressure (*) is at ∿ 1000 psi.

3.7.5 Incore thermocouples (*) are not increasing.

3.7.6 Feedwater valves indicates OPEN and flow is being observed entering unaffected steam generator.

3.8 If the HPI system has been actuated because of a low pressure condition, it must remain in operation until either:

3.8.1 Both LPI pumps are in operation and flowing at a rate in excess of 1000 gpm each on header flow gauge (*) and the situation has been stable for 20 minutes,

or

3.8.2 All hot and cold leg temperatures (*) are at least 50 degrees below the saturation temperature for the existing RCS pressure on wide range pressure (*). If 50 degrees subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated (refer to Enclosure 1).

3.9 If the transient is of a magnitude that the LPI Pumps have been activated, Operator action may be required to alter the operation of the LPI Pumps due to a maximum run time of thirty (30) minutes without indicated flow.

3.10 If action is taken to alter the LPI Pump operation, caution must be exercised to either:

3.10.1 Reset the ES Channels to allow the LPI system to be re-initiated

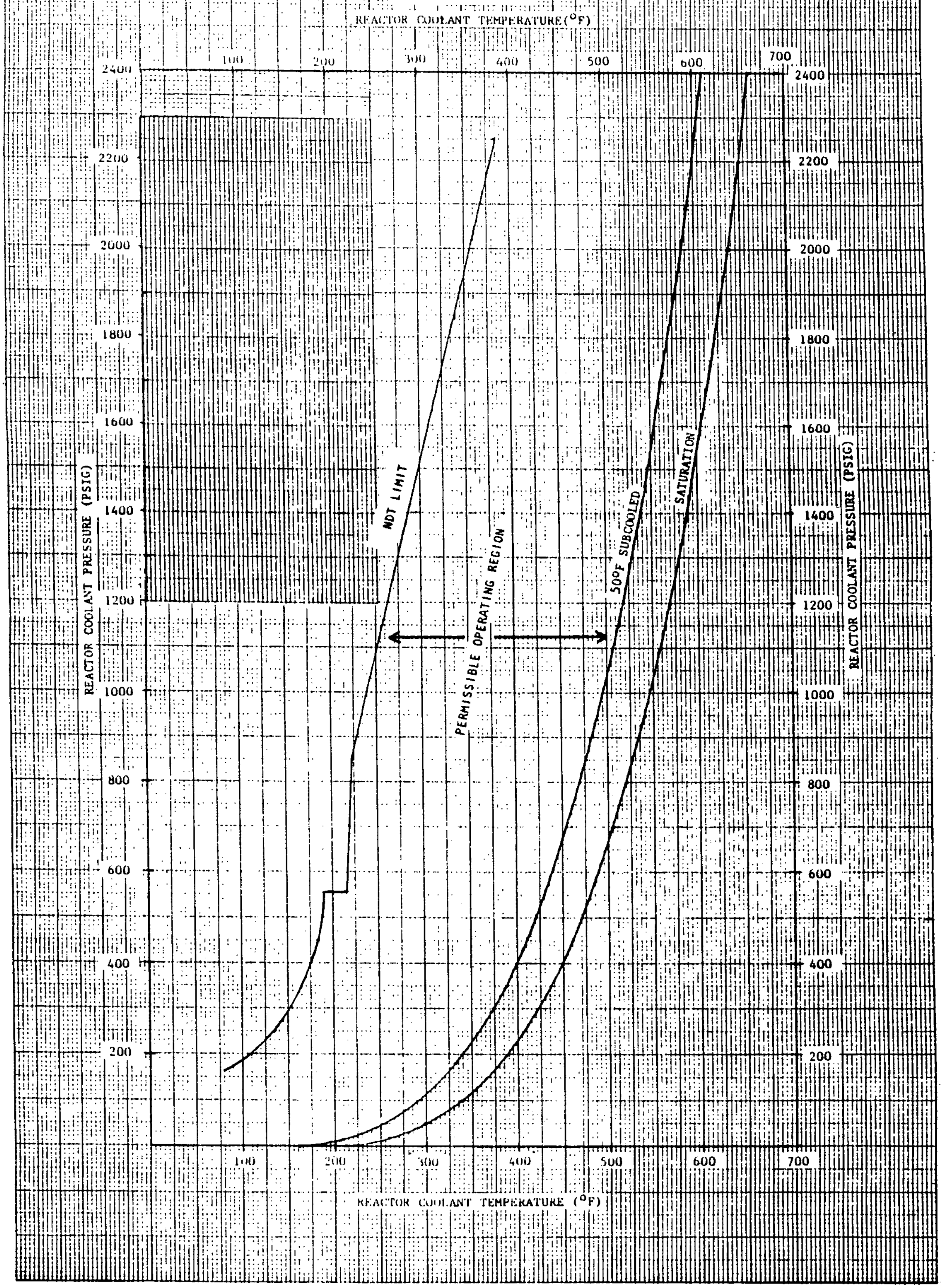
or

3.10.2 Manually re-initiate the LPI system if the ES Channels cannot be reset, but the plant transient progresses to a point that requires that LPI system to be in operation.

3.11 Insure shutdown margin of the Reactor Coolant System, per OP/1103/15, Reactivity Balance Calculation.

3.12 Initiate plant cooldown per OP/1102/10, Controlling Procedure for Unit Shutdown.

ENCLOSURE 1


REACTOR COOLANT TEMPERATURE (°F)
100
200
300
400
500
600
700
2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
REACTOR COOLANT PRESSURE (PSIG)
NDT LIMIT
PERMISSIBLE OPERATING REGION
50°F SUBCOOLED
SATURATION
REACTOR COOLANT TEMPERATURE (°F)

ENCLOSURE 2

EP/0/A/1800/8

LEVEL CORRECTIONS FOR POST ACCIDENT TEMPERATURE EFFECTS

Corrections to indicated level, % of Full Scale

Reactor Bldg. Temp. °F	Pressurizer Level	Steam Generator Startup	Steam Generator Operate	Steam Generator Full	Core Flood Tank
100	1.6	0.3	0.5	0.0	0.0
150	1.9	2.3	1.8	1.3	1.3
200	3.7	5.0	3.6	3.0	3.0
250	5.9	8.2	5.7	5.2	5.1
300	8.3	12.0	8.1	7.6	7.5
350	11.2	16.2	10.8	10.4	10.3
400	14.4	20.0	13.9	13.6	13.4

Pressurizer or OTSG Downcomer Temp. °F	Pressurizer Level	OTSG Level
68	1.00	1.00
150	1.02	1.02
200	1.04	1.04
250	1.06	1.06
300	1.09	1.09
350	1.12	1.12
400	1.17	1.17
450	1.23	1.21
500	1.31	1.31
550	1.42	1.36
600	1.61	
650	1.99	

Actual Level = Indicated Level - (Correction x Span)
Correction = R.B. Temp. Correction x Pressurizer or OTSG Temp. Correction

NOTE: Use only R.B. Temp. Correction for Core Flood

Span = Instrument Full Scale Reading

Example:

Steam Generator Startup Level Span	250"
Reactor Building Temp.	250°
Downcomer Temp.	300°
Indicated Level	220"
Correction for R.B. Temp.	8.2%
Correction for Downcomer Temp.	1.09%

Correction Factor = 8.2 (1.09) = 8.938%

Actual Level = 220" - (8.938% x 250) - 197.66"

Form SPD-1002-1

DUKE POWER COMPANY
PROCEDURE PREPARATION
PROCESS RECORD

(1) ID No: EP/0/A/1800/4
Change(s) — to
— Incorporated

(2) STATION: Oconee Nucl. Station

(3) PROCEDURE TITLE: Loss of Reactor Coolant

(4) PREPARED BY: Bobby Pollard DATE: 8-31-81

(5) REVIEWED BY: JA Ridgeway DATE: 9-2-81

Cross-Disciplinary Review By: ______ N/R: GAR

(6) TEMPORARY APPROVAL (IF NECESSARY):

By: ______ (SRO) Date: ______

By: ______ Date: ______

(7) APPROVED BY: JN Pope Date: 12-10-81

(8) MISCELLANEOUS:

Reviewed/~~Approved~~ By: W.M. Harris Date: 9-3-81

Reviewed/~~Approved~~ By: Terry Campbell Date: 9-17-81

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

LOSS OF REACTOR COOLANT

Considers the following cases:

Case A Excessive RC System Leakage

Case B Small Break

Case C RC System Rupture

Anytime during the use of this procedure if either of the following indications of inadequate core cooling exist, go to Inadequate Core Cooling OP/0/A/1106/35.

1. Hot Leg RTD's read superheated for the existing RCS pressure.
2. Incore thermocouple temperature reads superheated for the existing RCS pressure (Display Group 29 and Enclosure 2).
3. Subcooled Margin Monitors indicate 0 with red background.

Case A: Excessive RC System Leakage

1.0 Symptoms

1.1 Decreasing pressurizer level.

1.2 LDST level low or decreasing more than normal.

1.3 RCS leakage calculation indicates leak.

1.4 Reactor Building sump level increasing.

1.5 RIA alarms inside containment.

1.6 Increasing Quench Tank pressure and temperature.

2.0 Immediate Action

2.1 Automatic

None

2.2 Manual

NOTE: Any asterisk (*) parameters in the below sections shall be verified by checking alternate instrument channels as soon as all other immediate actions are performed.

2.2.1 Close (1)(2)(3)HP-5 (Letdown Isolation) if required.

2.2.2 Throttle open (1)(2)(3)HP-26 ("A" Loop Inj.) as required to maintain pressurizer level (*).

2.2.2.1 Open (1)(2)(3)HP-410 ("A" Bypass Inj.) as required to maintain pressurizer level (*) if (1)(2)(3)HP-26 fails closed.

2.2.3 Start standby HPI pump, if necessary, to maintain RC pumps seal flow.

2.2.4 Initiate makeup to LDST. If LDST level (*) cannot be maintained open (1)(2)(3)HP-24 ("A" Suction from BWST) to provide suction to the HPI pump(s) from the BWST.

2.2.5 Insure the RC System is $\geq$ 35°F subcooled by the Subcooled Margin Monitors (*) and remains so during power operations.

2.2.6 Close (1)(2)(3)RC-4 (Power Operated Relief Block).

3.0 Subsequent Action

3.1 Verify control room indications support the alarms received, alternate instrument channels shall be checked to confirm key (*) parameters (i.e., system temperatures, pressures, and pressurizer level). Backup readings for Subcooled Margin Monitors are located on Display Group 22. Subcooled margin may also be calculated using Loop T_h indication and Enclosure 1.

3.2 Evaluate the leakage and, if greater than Technical Specification 3.1.6 limits, initiate shutdown per OP/1102/10 (Unit Shutdown) to bring the unit to cold shutdown. Rate of shutdown should be as rapid as necessary up to and including a manual reactor trip.

3.3 If RCS pressure increases to pressurizer relief valves setpoints, verify PORV and pressurizer code reliefs reseated by observing pressurizer relief valves flow monitors.

Case B: Small Break

1.0 Symptoms

1.1 Decreasing pressurizer level with excessive reactor coolant system makeup and low letdown storage tank level.

1.2 Decreasing RCS pressure.

1.3 Quench tank high level, high temperature and high pressure (rupture disk may be blown).

1.4 Increasing Reactor Building sump level and temperature, or increasing HAWT and LAWT levels if leak is in Auxiliary Building.

1.5 Increasing Reactor Building temperature and pressure.

1.6 Increasing RIA's. (Reactor Building, Auxiliary Building, Stack)

1.7 Additional symptoms during heatup or cooldown.

1.7.1 RCS temperature increasing with minimum letdown and pressurizer level decreasing.

1.7.2 Cooldown rate of $\leq$ 100°F/hr and cannot maintain level in letdown storage tank.

NOTE: Anytime during the use of this procedure, if either of the following indications of inadequate core cooling exist, go to Inadequate Core Cooling OP/O/A/1106/35.

1. Hot Leg RTD's read superheated for the existing RCS pressure.
2. Incore thermocouple temperature reads superheated for the existing RCS pressure (Display Group 29 and Enclosure 2).
3. Subcooled Margin Monitors indicate 0 with red background.

2.0 <u>Immediate Actions</u>

2.1 Automatic

2.1.1 Reactor Trip, Turbine Trip

2.1.2 Possible E.S. Actuation

2.2 Manual

NOTE: Any asterisk (*) parameters in the below sections shall be verified by checking alternate instrument channels as soon as all other immediate actions are performed.

2.2.1 If E.S. has been initiated automatically because of low RC system pressure (*), <u>immediately</u> secure all RC pumps.

NOTE: Securing RC pump operation takes precedence over all other immediate manual actions.

2.2.2 Verify automatic actions have occurred; if not, perform manually. If ES has been bypassed due to heatup or cooldown, initiate manually, as required, to maintain pressurizer level and Reactor Coolant System pressure.

2.2.3 Check immediately for flow indication (*) on both HPI emergency injection lines.

2.2.3.1 If no flow is indicated in "B" loop, open (1)(2)(3)HP-409 ("B" Bypass Injection) within 10 minutes of ES actuation.

2.2.3.2 If no flow is indicated in "A" loop, open (1)(2)(3)HP-410 ("A" Bypass Injection) within 10 minutes of ES actuation.

2.2.4 Verify appropriate OTSG Level is being maintained by feedwater control. (Low level limit with RC pumps and 50% on the operating range without RC pumps.)

3.0 Subsequent Action

3.1 Verify control room indications support the alarm received. Alternate instrument channels shall be checked to confirm key (*) parameters (i.e., system temperatures, pressures and pressurizer level). Backup readings for Subcooled Margin Monitors are located on Display Group 22. Subcooled margin may also be calculated using Loop T_h indication and Enclosure 1.

CAUTION: High Reactor Building Temperature will cause level instruments to indicate high. Apply the correction factors supplied in Enclosure 3 to the appropriate instruments. Insure pressurizer heaters are manually off until pressurizer level is at least 100 inches to prevent heater burnout.

3.2 Monitor system pressure and temperature. If indication of inadequate core cooling exists go to OP/O/A/1106/35, Inadequate Core Cooling. If saturated conditions occur, initiate HPI. (See Display Group 29 and Enclosure 2.)

3.3 The operator should determine as quickly as possible if the cause for the depressurization is due to either a LOCA or non-LOCA (over cooling) event. Overcooling event symptoms that will differ from that of a small break or leak will be rapidly decreasing reactor coolant temperatures along with a decrease in pressure. If RC System Subcooling cannot be verified or maintained and if Engineered Safeguards (ES) actuate because of low RC system pressure, then the RC pumps will be tripped. For non-LOCA events:

3.3.1 Correct cause of the over-cooling event, then immediately restart a RC pump in each loop if the RCS is $\geq$ 50°F subcooled.

3.3.2 Control or reduce main steam pressure with the turbine bypass valves (or the atmospheric dumps) to stabilize RCS heatup and prevent the pressurizer from going solid.

NOTE: Non-LOCA (over cooling) conditions could be the result of overfeeding the S/G's or a steam line break. Go to EP/0/A/1800/8 Steam Supply System Rupture, if necessary.

3.3.3 As long as the RCS is maintained 50°F subcooled, throttle makeup and letdown flow to maintain pressurizer level at ∿ 220 inches.

3.3.4 Using turbine bypass and feedwater system, control the main steam pressure as needed to limit plant heatup until RC pressure control can be reestablished with the pressurizer.

3.3.5 When pressure control is re-established, use normal startup or shutdown (OP/1102/01 or OP/1102/10) procedures to establish desired plant conditions.

3.4 If startup transformer being shared with another unit is supplying auxiliary loads on both units, manually shed non-essential* loads on both units to avoid over-rating of the transformer or an undervoltage condition on the 4160 volt buses (maintain > 3600 volts).

*NOTE: Loads to be shed:

1RCP in each loop (unaffected unit)

all but one CCW pump

all but one HW pump

all but one CBP

all running RBCU fans (unaffected unit)

all aux RB Fans

suspend any RB Purges

suspend any Performance testing

3.5 Carefully monitor 4160 volt feeder buses to insure a minimum of 3600 volts is maintained. However, voltages between 3200 and 3600 volts are allowable for short periods not to exceed 4 hours.

3.6 Maintain a temperature versus time plot and a temperature versus pressure plot on a saturation diagram (Enclosure 1). Use both hot leg RTD's and the highest incore thermocouple reading to track the unit's condition through cooldown.

CAUTION: Do not override Automatic Actions of Engineered Safety Features unless continued operation will result in unsafe plant conditions or will threaten reactor vessel integrity. (Refer to Enclosure 1).

1. If 50°F subcooling criteria is met, throttle HPI flow using (1)(2)(3)HP-26 and (1)(2)(3)HP-27 (if (1)(2)(3) HP-26 or (1)(2)(3)HP-27 fail, use (1)(2)(3)HP-410 and (1)(2)(3)HP-409) to keep RC Pressure/Temperature in Region II with RCP's off or Region I or II with RCP(s) on.

   CAUTION: In the event that (1)(2)(3)HP-409 or (1)(2)(3) HP-410 do not close, secure HPI pumps, close the affected valve(s), then reinitiate HPI flow.

2. If RC pumps are operating, use Tc RTD's. If RC pumps are not operating, use: Average of the 5 highest reading incore thermocouples.

   CAUTION: If the HPI System has actuated because of low pressure conditions, it must remain in operation until one of the following criteria is satisfied:

   1. The LPI System is in operation and flowing at a rate in excess of 1000 gpm in each line and the situation has been stable for 20 minutes;

      or

   2. All hot and cold leg temperatures are at least 50°F below the saturation temperature for the existing RCS pressure (see Enclosure 1), and the action is necessary to prevent the indicated pressurizer level from going off scale high.

   3. If the 50°F subcooling cannot be maintained, the HPI shall be reactivated.

   4. Throttle (1)(2)(3)HP-26 & (1)(2)(3)HP-27 to maintain HPI flow not greater than 500 gpm per HPIP to prevent cavitation.

3.7 Anytime during the use of this procedure, if either of the following indications of inadequate core cooling exist, go to Inadequate Core Cooling OP/O/A/1106/35.

   1. Hot Leg RTD's read superheated for the existing RCS pressure.

2. Incore thermocouple temperature reads superheated for the existing RCS pressure (Display Group 29 and Enclosure 2).

3. Subcooled Margin Monitors indicate 0 with red background.

3.8 If any parameter causing ES actuation returns to a state that would allow the ES channel to be reset, the appropriate ES channels (analog and digital) must be manually reset.

3.9 If the transient is of a magnitude that the LPI Pumps have been activiated, but the pressure of the Reactor Coolant System is too high to allow flow with the LPI Pumps, Operator Action may be required to alter the operation of the LPI Pumps due to a maximum run time of thirty (30) minutes without indicated flow.

3.10 If action is taken to alter the LPI Pump operation, caution <u>must</u> be exercised to either:

3.10.1 Reset the ES Channels to allow the LPI System to re-initiate.

<u>or</u>

3.10.2 Manually re-initiate the LPI System if the ES Channels cannot be reset, but the plant transient progresses to a point that requires the LPI System to be in operation.

3.11 Identification and Early Control

3.11.1 Attempt to locate and isolate leak if possible. Close (1)(2)(3)HP-5 (Letdown Isolation), (1)(2)(3)RC-4 (Power Operated Relief Block), and if RCS pressure increased to pressurizer relief valves setpoints, verify PORV and pressurizer code reliefs reseated by observing pressurizer relief valves flow monitors.

NOTE: Pressurizer Level may be increasing due to RCS reaching saturated conditions or a break on top of the pressurizer.

3.11.2 If RC pressure decreases continuously, verify that core flood tanks and low pressure injection have actuated as needed.

NOTE: If the BWST should decrease to the Lo-Lo Level alarm point before the RCS pressure is low enough to allow the LPI system to pump water into the RCS, open valves (1)(2)(3)LP-19 and (1)(2)(3)LP-20, then close valves (1)(2)(3)LP-21 and (1)(2)(3)LP-22. NOTE: For Unit 3 only: To open 3LP-19 and 3LP-20, press and hold the 3LP-19 and 3LP-20 interlock bypass switch while opening 3LP-19 and 3LP-20. Open valves (1)(2)(3)LP-15 and (1)(2)(3)LP-16 to provide HPI pumps suction from the reactor building emergency sump through the LPI pumps and LPI coolers.

3.11.3 If high activity is detected in a S/G, isolate the leaking generator. Do not isolate both S/G's.

3.11.4 Component cooling water, low pressure service water and HPI seal injection should be maintained to the RC pumps to insure continued service or the ability to restart the RC pumps.

3.11.5 If an intersystem LOCA should exist where the check valves in the Low Pressure Injection System leak causing an over-pressurization of the LPI System, close LP-17 ("A" LP Header

to Reactor Building Isolation) or LP-18 ("B" LP Header to Reactor Building Isolation) on the affected Header and isolate the LPI System break. The described intersystem LOCA could be identified by:

LPI Pump Diff. Press Lo Alarm.

Area Radiation Monitor Alarms Outside Reactor Building.

NEO Report of leakage probably in LPI Cooler Rooms.

Rapid increase in LAWT.

LPI Pump Amps High.

3.11.6 Determine availability of Reactor Coolant pumps and main and auxiliary feedwater systems. If RCP's are not operating, go to 3.12. If RCP's are operating, go to 3.13.

3.12 Actions if RCP's are not operating:

3.12.1 If feedwater is not available, restore feedwater as soon as possible.

3.12.2 Open (1)(2)(3)RC-4 and (1)(2)(3)RC-66 and maintain maximum HPI flow ($\sim$ 500 gpm per pump).

NOTE: If the PORV cannot be actuated, the safeties will relieve pressure.

3.12.3 When feedwater is restored, raise OTSG levels to 95% on the operating range by throttling (1)(2)(3)FDW-315 and (1)(2)(3)FDW-316. Close (1)(2)(3)RC-4 and (1)(2)(3)RC-66.

NOTE: Verify (1)(2)(3)RC-66 reseated by checking PORV flow monitor.

3.12.4 If RC pressure is decreasing, following secondary side pressure decreases and with primary system temperatures

indicating saturated conditions, raise OTSG level to 95% on the operating range, continue cooldown via reflux boiling action until an RC pressure of 150 psig is reached and proceed to LPI cooling by going to OP/0/A/1106/35, Inadequate Core Cooling.

NOTE 1: Verify Reflux boiling in OTSG's by observing:

1. T_c at saturation temperature for OTSG pressure.
2. Primary delta T, either $(T_H - T_c)$ or (Incore thermocouple - T_c) ∿ 30 to 40°F.
3. OTSG level being maintained at 95% on the operating range.

NOTE 2: Loss of OTSG pressure, increasing incore thermocouple or T_H and RCS pressure increasing are indications that primary to secondary coupling is lost. If this occurs, proceed to step 3.12.6.b (RCP bump) to regain reflux boiling action. If T_H or incore thermocouples read superheated, or Subcooled Margin Monitors indicate 0 with red background, proceed immediately to OP/0/A/1106/35, Inadequate Core Cooling.

3.12.5 If RC pressure stops decreasing in response to secondary side pressure decrease and the reactor coolant system becomes subcooled, check to see that the following conditions are both satisfied:

1. All hot and cold leg temperatures are below the saturation temperature for the existing RCS pressure; and

2. The hot and cold leg temperatures are decreasing in response to steam generator secondary temperature decrease.

If these conditions are satisfied and remain satisfied, raise OTSG level to 95% on the operating range, continue cooldown via natural circulation to achieve an RCS Temperature (T_c) of 240°F and proceed to OP/1102/10 Unit Shutdown for normal LPI decay heat removal. Bypass ES Low Pressure injection and block core flood actuation at RC pressure of 700 psig.

NOTE 1: Verify natural circulation in the RCS by observing:

1. RCS is 50°F subcooled by Subcooled Margin Monitors and remains so or see Enclosure #1.
2. Feedwater level is at ∿ 95% on the operating range.
3. The ΔT between T_{hot} and T_{cold} is ∿ 30°F to 40°F.
4. Main steam pressure is at ∿ 1000 psi.
5. Incore thermocouples are not increasing.
6. Feedwater valves indicate OPEN and flow is being observed entering steam generators.

CAUTION: OTSG levels greater than 95% on the operating range must be avoided to preclude feedwater carryover into the steamlines.

NOTE 2: Loss of OTSG pressure, increasing incore thermocouple or T_H and RCS pressure increasing are indications that primary to secondary coupling is lost. If this occurs, proceed to Step 3.12.6.b (RCP bump) to regain natural circulation. If T_H or Incore thermocouples read superheated, or Subcooled Margin Monitors indicate 0 with red background, proceed to OP/0/A/1106/35, Inadequate Core Cooling.

3.12.6 Restore RCP flow (one pump per loop), when possible, per following instructions. If RCP's cannot be operated and RCS pressure is increasing, proceed to step 3.12.11.

CAUTION: Verify adequate LPSW, CC and Seal Injection to the Reactor Coolant pump before starting.

a. Starting a pump is permissible at RC pressure > 1600 psig; even if saturation conditions exist.

NOTE: The break is small enough to be below the critical break range where the core would become uncovered if the RCP's tripped off.

or

b. If RC pressure exceeds S/G steam pressure by 600 psi or more, "bump" one RC pump for ∿ 10 seconds (a loss of OTSG pressure and RCS pressure increasing, indicates a loss of primary to secondary coupling). Allow RCS pressure to stabilize and continue cooldown. If RC

pressure again exceeds S/G steam pressure by 600 psi, wait at least 15 minutes and bump another pump. Bump alternate pumps so that no pump is bumped more than once in an hour. This may be repeated with an interval of 15 minutes up to 5 times. If after the 4th RCP Bump natural circulation or reflux boiling action is still not re-established, wait 15 minutes and start a RCP and allow the pump to continue in operation; or

c. If: 1. RCS pressure has stabilized for >1 hour.
2. Steam pressure is < 100 psig.
3. RCS pressure is > 250 psig.

Bump a pump, for 10 seconds, wait 30 minutes and start an alternate pump.

3.12.7 When forced cooldown is established allow RCS pressure to stabilize.

3.12.8 Establish and maintain OTSG cooling by adjusting main steam pressure via turbine bypass valves (or atmospheric dump valves). Cooldown at 100°F/hr to achieve an RC pressure of $\leq$ 190 psig.

3.12.9 Isolate core flood tanks when 50°F subcooling is attained and RC pressure is less than 700 psig by closing (1)(2)(3)CF-1 and (1)(2)(3)CF-2 and bypass ES low pressure injection.

3.12.10 Go to OP/1102/10 Unit Shutdown for LPI cooling.

3.12.11 If a RC pump cannot be operated and RCS pressure reaches 2300 psig, open (1)(2)(3)RC-4 and (1)(2)(3)RC-66 to reduce

RC pressure to 100 psi above steam generator pressure. Repeat if necessary. Continue to cooldown with natural circulation and go to OP/1102/10, Unit Shutdown, for normal LPI decay heat removal.

NOTE: Verify (1)(2)(3)RC-66 reseated by checking PORV flow monitor.

NOTE: If the leak is of such a small size that the core is not being adequately cooled by the leak relieving heat energy, opening the PORV will create another flow path to help relieve the excess heat energy. If T_H or Incore thermocouples read superheated, or Subcooled Margin Monitors read 0 with red background, proceed to OP/0/A/1106/35, Inadequate Core Cooling.

3.13 Actions if RCP's are operating:

3.13.1 Go to one RCP per loop.

3.13.2 If feedwater is not available, maintain maximum HPI flow and restore feedwater as soon as possible.

3.13.3 If RCS pressure increases, open (1)(2)(3)RC-4 and (1)(2)(3)RC-66 and leave open.

NOTE: If the PORV cannot be activated, the safeties will relieve pressure.

3.13.4 When feedwater is recovered, restore OTSG levels in a controlled manner by throttling (1)(2)(3)FDW-315 and (1)(2)(3)FDW-316. (EFWP Discharge to emergency feed ring)

3.13.5 After feedwater is restored, shut (1)(2)(3)RC-4 and (1)(2)(3)RC-66.

NOTE: Verify (1)(2)(3)RC-66 reseated by checking PORV flow monitor.

3.13.6 Allow RC pressure to stabilize.

3.13.7 Establish and maintain OTSG cooling by adjusting main steam pressure via turbine bypass valves (or atmospheric dumps). Cooldown at less than 100°F/hr to achieve an RC pressure at 250 psig.

3.13.8 Isolate core flood tanks when 50°F subcooled and RC pressure is less than 700 psig, by closing (1)(2)(3)CF-1, (1)(2)(3)CF-2 and bypass ES Low Pressure Injection.

3.13.9 If RC pressure continues to decrease following secondary side pressure decreases and with primary system temperatures indicating saturated conditions, continue cooldown until an RC pressure of 150 psig is reached and proceed to OP/0/A/1106/35, Inadequate Core Cooling.

3.13.10 If RC pressure stops decreasing in response to secondary side pressure decrease and the reactor coolant system becomes subcooled, check to see that the following conditions are <u>both</u> satisfied:

1. All hot and cold leg temperatures are below the saturation temperature for the existing RCS pressure;

   and

2. The hot and cold leg temperatures are decreasing in response to steam generator secondary temperature decrease.

If these conditions are satisfied and remain satisfied,

continue cooldown to achieve an RCS temperature (T_c) of 240°F and proceed to OP/1102/10, Unit Shutdown, for normal LPI decay heat removal.

Case C: RC System Rupture

1.0 Symptoms

1.1 Very rapid RC system depressurization to some stable low pressure.

1.2 RIA Alarms

1.3 Increasing Reactor Building Sump level

1.4 Very rapidly decreasing pressurizer level indication

2.0 Immediate Action

2.1 Automatic

2.1.1 Reactor trip.

2.1.2 Turbine trip.

2.1.3 ES actuation.

2.2 Manual

NOTE: Any asterisk (*) parameters in the below sections shall be verified by checking alternate instrument channels as soon as all other immediate actions are performed.

2.2.1 IMMEDIATELY TRIP all RC pumps.

2.2.2 Verify automatic actions have occurred; if not, perform manually.

2.2.3 Check immediately for flow (*) indication on both HPI emergency injection lines.

2.2.3.1 If no flow is indicated in "B" loop, open (1)(2)(3)HP-409 ("B" Bypass Inj.) within 10 minutes of ES actuation.

2.2.3.2 If no flow is indicated in "A" loop, open (1)(2)(3)HP-410 ("A" Bypass Inj.) within 10 minutes of ES actuation.

2.2.4 After the valves are open, verify flow into both loops (*). If flow is not established within the above time period, rapidly depressurize the RC System so that LPI ES can be actuated below 500 psig by opening turbine bypass valves or opening (1)(2)(3)RC-66 (power operated relief valve) from ICS Cabinet 13.

3.0 Subsequent Action

3.1 Verify control room indications support the alarms received. Alternate instrument channels should be checked to confirm key (*) parameters (i.e, system temperatures, pressures and pressurizer level).

CAUTION: High Reactor Building Temperature will cause level instruments to indicate high. Apply the correction factors supplied in Enclosure 3 to the appropriate instruments. Insure pressurizer heaters are manually off until pressurizer level is at least 100 inches to prevent heater burnout.

NOTE: If RCS pressure increased to pressurizer relief valves setpoints, verify pressurizer relief valves reseated by observing pressurizer relief valves flow monitors.

3.2 If startup transformer being shared with another unit is supplying auxiliary loads on both units, manually shed non-essential* loads on both units to avoid over-rating of the transformer or an undervoltage condition on the 4160 volt buses (maintain > 3600 volts).

*NOTE: Loads to be shed:

1RCP in each loop (unaffected unit)

all but one CCW pump

all but one HW pump

all but one CBP

all running RBCU fans (unaffected unit)

all aux RB Fans

suspend any RB Purges

suspend any Performance testing

3.3 Carefully monitor 4160 volt feeder buses to insure a minimum of 3600 volts is maintained. However, voltages between 3200 and 3600 volts are allowable for short periods not to exceed 4 hours.

3.4 If any parameter causing ES actuation returns to a state that would allow the ES channel to reset, the appropriate ES channel (analog and digital) must be manually reset.

CAUTION: Do not override Automatic Actions of Engineered Safety Features unless continued operation will result in unsafe plant conditions or will threaten reactor vessel integrity. (Refer to Enclosure 1).

1. If 50°F subcooling criteria is met, throttle HPI flow using (1)(2)(3)HP-26 and (1)(2)(3)HP-27 (if (1)(2)(3) HP-26 or (1)(2)(3)HP-27 fail, use (1)(2)(3)HP-410 and (1)(2)(3)HP-409) to keep RC Pressure/Temperature in Region II with RCP's off or Region I or II with RCP(s) on.

CAUTION: In the event that (1)(2)(3)HP-409 or (1)(2)(3) HP-410 do not close, secure HPI pumps, close the affected valves(s), then reinitiate HPI flow.

2. If RC pumps are operating use Tc RTD's. If RC pumps are not operating, use: Average of the 5 highest reading incore thermocouples.

3.5 If the transient is of a magnitude that the LPI Pumps have been activiated, but the pressure of the Reactor Coolant System is too high to allow flow with the LPI Pumps, Operator Action may be required to alter the operation of the LPI Pumps due to a maximum run time of thirty (30) minutes without indicated flow.

3.6 If action is taken to alter the LPI Pump operation, caution <u>must</u> be exercised to either:

3.6.1 Reset the ES Channels to allow the LPI System to re-initiate.

<u>or</u>

3.6.2 Manually re-initiate the LPI System if the ES Channels cannot be reset, but the plant transient progresses to a point that requires the LPI System to be in operation.

3.7 Continue with Unit Shutdown per OP/1102/10.

CAUTION: System must be depressurized to less than 200 psig on RCS wide range pressure by the time the BWST is down to 3 ft. on level gauge.

3.8 Verify core flood discharges at ~600 psig.

3.9 If the HPI System has actuated because of low pressure conditions, it must remain in operation until one of the following criteria is satisfied:

1. The LPI System is in operation and flowing at a rate in excess

of 1000 GPM in each line and the situation has been stable for 20 minutes;

or

2. All hot and cold leg temperatures are at least 50°F below the saturation temperature for the existing RCS pressure, and the action is necessary to prevent the indicated pressurizer level from going off-scale high.

NOTE: If 50°F subcooling cannot be maintained, the HPI shall be reactivated.

NOTE: The degree of subcooling beyond 50°F and the length of time HPI is in operation shall be limited by Enclosure 1.

NOTE: Throttle (1)(2)(3)HP-26 and (1)(2)(3)HP-27 to maintain less than 500 GPM per HPIP to prevent HPIP cavitation.

3.10 When the BWST level decreases to the lo-lo level alarm point, open valves (1)(2)(3)LP-19 and (1)(2)(3)LP-20, then close valves (1)(2)(3)LP-21 and (1)(2)(3)LP-22.

NOTE: For Unit 3 only. To open 3LP-19 and 3LP-20, press and hold the 3LP-19 and 3LP-20 interlock bypass switch while opening 3LP-19 and 3LP-20.

3.11 Throttle LP injection valves (1)(2)(3)LP-12 and (1)(2)(3)LP-14 to prevent pump cavitation.

NOTE: High flow 4200 gpm per pump.

3.12 When the LPI System is placed in the recirculation mode, notify Chemistry to commence caustic addition within 30 minutes to the LPI System as required in CP/(1&2) (3)/A/2002/05 (Post Accident Caustic Injection into the LPI System).

3.13 Sample the RB Sump to determine boron concentration is not being diluted. If the RB Sump is being diluted, insure LPSW to RCP's is isolated. Isolate LPSW to any RBCU that has a "RBCU Rupture Stat-alarm".

3.14 When RB pressure on RB pressure gauge decreases to less than 3.0 psig, return two RB cooling fans to full speed.

3.15 Secure the RB spray system when RB pressure on RB pressure gauge decreases to atmospheric pressure or system has operated a minimum of 80 hours. Spray system may be run longer to aid in removing airborne iodine.

3.16 Monitor the hydrogen level in RB. Place RB Hydrogen Purge System in operation per OP/0/A/1104/29.

3.17 Ensure PRVS is in operation and adjust (1)(2)(3)PR-13 (Filter A Outlet) and (1)(2)(3)PR-17 (Filter B Outlet) as necessary to maintain ∿ 1000 cfm flow thru each filter train and ∿ 1.75 in. H_2O negative pressure in the Penetration Room. On Unit 2 only; secure Auxiliary Bldg. Fans 19, 20, and 21. If either Penetration Room fan has failed to start on ES initiation, open (1)(2)(3)PR-20 (PR Fans Suction Tie) at its locally mounted switch to allow the operating fan to purge thru both filters.

CAUTION: Do not exceed 1100 cfm flow thru each filter train.

3.18 Perform the following applicable step(s) on the appropriate unit within 24 hours:

3.18.1 Unit 1: Rack in breakers on valves 1LP-103 and 1LP-104 and open these valves. Verify flow through these valves.

3.18.1.1 If unable to verify flow through 1LP-103 and 1LP-104, rack in breaker on valve 1LP-105. Open 1LP-1,

1LP-2, and 1LP-105 and verify flow through 1LP-106 and 1LP-107.

3.18.2 Unit 2: Rack in breakers on 2LP-103 and 2LP-104 and open these valves. Verify flow through these valves.

3.18.2.1 If unable to verify flow through 2LP-103 and 2LP-104, open 2LP-4, 2LP-3, 2LP-2, 2LP-1, 2LP-108 and 2LP-109 and verify flow through this line.

3.18.3 Unit 3: Rack in breakers on 3LP-103 and 3LP-104 and open these valves.

3.18.3.1 If unable to verify flow through 3LP-103 and 3LP-104, open 3LP-3, 3LP-2, 3LP-1, 3LP-108 and 3LP-109 and verify flow through this line.

ENCLOSURE 3
EP/0/A/1800/04

LEVEL CORRECTIONS FOR POST ACCIDENT TEMPERATURE EFFECTS

Corrections to indicated level, % of Full Scale

Reactor Bldg. Temp. °F	Pressurizer Level	Steam Generator Startup	Steam Generator Operate	Steam Generator Full	Core Flood Tank
100	1.6	0.3	0.5	0.0	0.0
150	1.9	2.3	1.8	1.3	1.3
200	3.7	5.0	3.6	3.0	3.0
250	5.9	8.2	5.7	5.2	5.1
300	8.3	12.0	8.1	7.6	7.5
350	11.2	16.2	10.8	10.4	10.3
400	14.4	20.0	13.9	13.6	13.4

Pressurizer or OTSG Downcomer Temp. °F	Pressurizer Level	OTSG Level
68	1.00	1.00
150	1.02	1.02
200	1.04	1.04
250	1.06	1.06
300	1.09	1.09
350	1.12	1.12
400	1.17	1.17
450	1.23	1.21
500	1.31	1.31
550	1.42	1.36
600	1.61	
650	1.99	

Actual Level = Indicated Level - (Correction x Span)
Correction = R.B. Temp. Correction x Pressurizer or OTSG Temp. Correction

NOTE: Use only R.B. Temp. Correction for Core Flood

Span = Instrument Full Scale Reading

Example:		
	Steam Generator Startup Level Span	250"
	Reactor Building Temp.	250°
	Downcomer Temp.	300°
	Indicated Level	220"
	Correction for R.B. Temp.	8.2%
	Correction for Downcomer Temp.	1.09%

Correction Factor = 8.2 (1.09) = 8.938%

Actual Level = 220" - (8.938% x 250) - 197.66"