TECHNICAL STAFF ANALYSIS REPORT

ON

TECHNICAL ASSESSMENT OF OPERATING,
ABNORMAL, AND EMERGENCY PROCEDURES

TO

PRESIDENT'S COMMISSION ON THE

ACCIDENT AT THREE MILE ISLAND

ADVANCE COPY

NOT FOR PUBLIC RELEASE

BEFORE AMs, WEDNESDAY, OCTOBER 31, 1979

7910300387

1181 179

TECHNICAL ASSESSMENT OF OPERATING, ABNORMAL, AND EMERGENCY PROCEDURES

BY

RONALD M. EYTCHISON

MEMBER

TECHNICAL ASSESSMENT TASK FORCE

OCTOBER 1979

WASHINGTON, D.C.

1181 180

This document is solely the work of the Commission staff and does not necessarily represent the views of the President's Commission or any member of the Commission.

This pre-publication copy is a final document and will be subject only to minor editorial changes in its published form.

TABLE OF CONTENTS

Abnormal Procedure 2203-2.6, Post-Accident Hydrogen Control, Revision 1, June 23, 1978

Emergency Procedure 2202-2.2, Loss of Steam Generator Feed, Revision 3, Oct. 13, 1978

Emergency Procedure 2202-1.5, Pressurizer System Failure, Revision 3, Sept. 29, 1978

Emergency Procedure 2202-1.1, Reactor Trip, Revision 6, Oct. 25, 1978

Emergency Procedure 2202-1.3, Loss of Reactor Coolant/Reactor Coolant System Pressure, Revision 11, Oct. 6, 1978

FINDINGS

ACRONYMS

APPENDIX A -- Three Mile Island Nuclear Station, Unit #2 Emergency Procedure 2202-1.5, Pressurizer System Failure

APPENDIX B -- Three Mile Island Nuclear Station, Unit #2 Emergency Procedure 2202-1.3, Loss of Reactor Coolant/Reactor Coolant System Pressure

APPENDIX C - Duke Power Company, Oconee Nuclear Station, Loss of Coolant

SUMMARY

As a part of the effort to identify and evaluate the possible causes for the Three Mile Island accident, an analysis of operating, abnormal, and emergency procedures was conducted by the staff. Of the 70 procedures included in these categories, 15 procedures were judged to be significant because they either were in use at the onset of the accident or became applicable as events took place. These procedures were evaluated for technical accuracy and adequacy with respect to the transient and its aftermath.

Evaluation of seven of the 15 procedures indicated that although they may be deficient in minor respects, they are adequate for their intended purpose. The procedures which are judged to be in this category are all operating procedures. They include: power operations, decay heat removal system, decay heat removal via OTSG (once through steam generator), core flooding system, reactor building spray, emergency feedwater, and safety and safety features actuation system. The procedure for decay heat removal using the steam generators was considered to be clearly written and provide a relatively simple, straightforward method for removing decay heat either with or without reactor coolant pumps in operation.

One operating procedure, one abnormal procedure, and two emergency procedures were believed to be usable, although they contain significant deficiencies that could cause confusion or lack of correct action:

- First, the reactor coolant pump operation procedure contains provisions that require tripping the pumps when vibration exceeds certain values. There is no discussion of unusual circumstances that might warrant pump operation with excessive vibration. Additionally, the vibration criteria conflict with those in another procedure. This procedure also does not provide clear instructions concerning whether the pumps should be tripped under low pressure, loss-of-coolant accident conditions.

- Abnormal procedure 2203-2.2 on turbine trip is deficient because it recognizes that the pilot-operated relief valve

(PORV) will open on a turbine trip, but it does not include any precaution to ensure that the valve shuts. Also, the operator is required to let down coolant as necessary following a turbine trip to prevent the pressurizer level from exceeding 240 inches. This might contribute to operator action to avoid high pressurizer levels following a turbine trip, as occurred on March 28, 1979.

- The emergency procedure for loss of steam generator feed requires immediate tripping of the reactor following loss of both feedwater pumps, regardless of reactor power level. Additionally, the procedure does not recognize that the PORV would open, although such an occurrence would be very likely.

- Review of the reactor trip emergency procedure determined that its most significant shortcoming is lack of direction to determine the cause for the reactor trip.

The third categorization in this analysis identified one operating procedure, one abnormal procedure, and two emergency procedures that were evaluated as inadequate:

- The pressurizer operation procedure emphasizes that the operators are not permitted by the technical specifications--a part of the operating license--to exceed a pressurizer level of 385 inches in mode 3--the condition that the reactor plant was in following the reactor trip. There are no exceptions indicated in the procedure for emergency conditions. Thus, operators might be influenced by this procedure in their actions if a phenomenon not predicted by the procedure, such as rising pressurizer level following a reactor trip, were to take place. Although operators' actions should be governed by all of the symptoms available, this procedure is judged to be inadequate.

- The procedure for post-accident hydrogen control--abnormal procedure 2203.2.6--fails to recognize that hydrogen can be generated rapidly, as occurred at TMI. It also does not recognize the difficulty of placing hydrogen recombiners into operation following an accident.

- Emergency procedure 2202.1.5 on pressurizer system failures is very confusing in its organization. Also, symptoms are significantly incomplete, misleading, or erroneous. Two sections of the procedure concerning a stuck-open PORV or stuck-open code safety valve should be in the loss-of-coolant accident (LOCA) procedure.

- From the standpoint of the TMI accident, perhaps the most important procedure was that for loss of reactor coolant/reactor coolant system pressure--emergency procedure 2202-1.3. The procedure does not provide the operators with objectives. It is difficult to use because the operator can be confused as to which section is applicable. A section on small-break LOCA response is misplaced, illogical, and cannot be followed.

The operator is required to bypass safeguards actuation and throttle high pressure injection, regardless of the severity of the accident. The procedure does not ensure that containment is isolated promptly.

Other general deficiencies were noted in the review of these procedures. Many minor substantive errors, typographical errors, and imprecise or sloppy terminology are not consistent with the quality required in nuclear power plant procedures. There also was noted a general emphasis on procedures to avoid equipment, component, or system damage and a lack of emphasis on keeping the core cooled.

INTRODUCTION

In accordance with the requirements of 10 CFR 50, Appendix B, commercial nuclear power plants are required to be operated as described in the operating and emergency procedures. Therefore, in considering the possible causes of the accident at Three Mile Island, investigation of procedures--as well as personnel, design, and equipment factors--was determined to be relevant. The technical accuracy and the adequacy of pertinent procedures deserved assessment to determine whether or not the procedures offered sufficient guidance for the control room operator.

TMI-2 has a total of 30 operating procedures, 25 emergency procedures, and 15 plant abnormal procedures. For the purposes of this analysis,it was decided that only those significant procedures that were in use at the onset of the accident or that were relevant as the accident progressed would be evaluated. Such procedures were determined to include nine operating procedures, two plant abnormal procedures, and four emergency procedures. Analysis of the procedures therefore was limited to study and evaluation of these 15 procedures.

ANALYSIS

The accident that occurred at Three Mile Island included a number of equipment, system, and overall plant events involving operating, abnormal, and emergency procedures. These procedures were intended to provide guidance to the operators concerning normal and abnormal plant operation, and if an emergency situation arose, the procedures were to direct mitigating actions to minimize the probability of plant damage and to ensure public safety.

The procedures that are evaluated in this paper according to their technical propriety and adequacy are listed below. They include significant procedures that were in use just before the onset of the accident, procedures that were not referred to by the operators but that were pertinent.

- power operations -- OP#2102-2.1
- pressurizer operation -- OP#2103-1.3
- reactor coolant pump operation -- OP#2103-1.4
- decay heat removal system -- OP#2104-1.3

1181 186

- decay heat removal via once through steam generator (OTSG) -- OP#2102-3.3
- core flooding system -- OP#2104-1.1
- reactor building spray -- OP#2104-1.4
- emergency feedwater -- OP#2104-6.3
- safety features actuation system -- OP#2105-1.3
- turbine trip -- AB#2203-2.2
- post-accident hydrogen control -- AB#2203-2.6
- loss of steam generator feed -- EP#2202-2.2
- pressurizer system failure -- EP#2202-1.5
- reactor trip -- EP#2202-1.1
- loss of reactor coolant/reactor coolant system pressure -- EP#2202-1.3

OPERATING PROCEDURE 2102-2.1, POWER OPERATIONS, REVISION 11, MARCH 20, 1979

The limits and precautions section of operating procedure (OP) 2102-2.1 states that in case a safety limit is exceeded (2.1), an automatic safety system does not function as required (2.2), or a limiting condition for operation (LCO) is not met, the shift supervisor shall notify the station/unit superintendent. It is not clear whether the procedure intends for the station manager, the unit superintendent, or both to be notified of such unsafe conditions.

The terms "operation" and "steady-state operation" are used but are not defined.

Limit 2.7, dated April 18, 1978, and section 4.1, dated June 17, 1977, state that the core thermal power shall not exceed 2,772 megawatts. In fact, core thermal power was restricted to 2,568 MW(t) until late 1978, pending demonstration of full compliance with 10 CFR 50.46 and 10 CFR 50, Appendix K, concerning a small-break LOCA.

A prerequisite for reactor plant power operations, as listed in section 3.12 is that three independent steam generator auxiliary feedwater pumps and associated flow paths shall be operable. The term "operable" as defined in section 1.0 of the TMI-2 technical specifications, means that the system, subsystem, train, component, or device shall be capable of performing its specified functions.

Section 4.12 requires that shift logs be maintained in accordance with administrative procedure 1012.

Evaluation

The analysis of OP 2102-2.1 on power operations concludes that the procedure was adequate for the intended purpose.

OPERATING PROCEDURE 2103-1.3, PRESSURIZER OPERATIONS, REVISION 3, JULY 19, 1978

Paragraph 2.1.8 of the procedure's limits and precautions states:

The pressurizer/RC [reactor coolant] systems must not be filled with coolant to solid conditions (400 inches) at any time except as required for system hydrostatic tests.

Paragraph 2.2.7 of the limits and precautions requires:

While in modes 1, 2 and 3, the Pressurizer shall be operable with:

a) Steam bubble, and

b) A water volume between 240 and 1,330 cubic feet (45 and 385 inches).

These requirements are consonant with the Babcock & Wilcox (B&W) limits and precautions and with the operating license technical specifications

Section 4.2.4 of the procedure provides instructions for equalizing pressurizer and reactor coolant (RCS) system boron concentrations. This portion of the procedure would be used in case of pilot-operated relief valve (PORV) or code safety valve seat leakage, such as was the case on the morning of March 28, 1979. Essentially, boron concentration equalization involves turning on heater bank 4 and opening the spray valve to permit spray flow into the pressurizer steam space.

Evaluation

Review of OP 2103-1.3, as well as pertinent portions of the technical specifications, emphasizes that the operators are required to avoid permitting the pressurizer level to exceed 385 inches in mode 3--the condition that the plant was in following reactor trip. Clearly, placing the core in jeopardy to avoid permitting the pressurizer from going solid is not the intent of either this procedure or the technical specification. However, there are no exceptions indicated in the procedure for emergency conditions. Thus operators might be influenced in their actions by this procedure if a phenomenon not predicted by the procedure--such as rising pressurizer level following a reactor trip--were to take place. Although operators' actions should be governed by all of the symptoms available to them, this procedure is judged to be inadequate.

OPERATING PROCEDURE 2103-1.4, REACTOR COOLANT PUMP OPERATION, REVISION 6, AUGUST 16, 1979

Of all the procedures reviewed, OP#2103-1.4 has the most extensive list of limits and precautions--about eight pages in length. Many of these are related to pump vibrations, including:

- The pump manufacturer shall be notified when RC pump steady state vibration measured at the pump coupling reaches 15 mils peak to peak (section 2.1.1.15).

- Reactor coolant pumps must be tripped if motor stand vibration exceeds 3 mils (section 2.2.4.7), if shaft vibration of greater than 20 mils continues for 4 hours (section 2.2.4.7), or if shaft vibration exceeds 30 mils (section 2.2.4.8).

Note that abnormal procedure (AP) 2203-1.4, reactor coolant pump and motor emergencies, and OP 2101-1.1 on nuclear plant limits and precautions, both discuss shaft vibration and have limits that differ from those given in OP#2103-1.4.

The procedure requires [In Section 4.3.2] that net positive suction head (NPSH) for the pumps be maintained in accordance with a curve provided in the procedure during low pressure condition in the RCS. Thus, at an RCS temperature of 582°F (nominal), the minimum allowed pressure for reactor coolant pump operation would be greater than 1,400 psig.

Evaluation

OP 2103-1.4 appears to be adequate for ensuring the proper operation of reactor coolant pumps. The conflicting criteria for what constitutes unacceptable pump or shaft vibration needs to be resolved. Guidance concerning the pressure temperature relationship at which pumps must be tripped is quite clear.

The NRC Investigative Report on the Three Mile Island accident (NUREG 0600) states categorically that reactor coolant pumps should have been tripped immediately when pressure dropped to 1,200 psig. This is in conflict with the statements contained in Inspection and Enforcement Bulletin 79-05A, which indicate that operators should not have tripped all reactor coolant pumps, even when flow had degraded significantly. It is suspected that the change in philosophy was the result of a B&W analysis after the accident, which determined that if high pressure injection initiates because of a low pressure condition in the reactor coolant system, all reactor coolant pumps should be tripped immediately.

OPERATING PROCEDURE 2104-1.3, DECAY HEAT REMOVAL SYSTEM, REVISION 11, JUNE 23, 1978

The decay heat removal system (DHRS) is designed to remove decay heat and sensible heat from the reactor coolant system during the latter stages of plant cooldown. In the event of a LOCA, the system injects borated water from the borated water storage tank (BWST) into the reactor vessel. For long-term emergency cooling, the system can take suction from the reactor building sump.

Operating procedure 2104-1.3 describes how the DHRS is used to fulfill these functions. Because the system is designed to operate at less than 340 psig, regardless of the RCS temperature, it must remain isolated from the RCS when system pressure and temperature are above prescribed values. The prerequisites of OP 2104-1.3 for placing the DHRS in operation state that the reactor coolant system should be cooled down to about 250°F and depressurized to less than 320 psig.

The DHRS normally is lined up for engineered safety (ES) actuation and will start when pressure drops below 1,650 psig or reactor building pressure increases to 4 psig. The system will operate in the recirculation mode, taking suction from either the BWST or the reactor building sump, until the reactor coolant system pressure drops to approximately 250 psig, at which time the system provides low pressure injection to the RCS.

This operating procedure also provides for lon erm core circulation modes to prevent boron concentration eft cts after a loss-of-coolant accident. One of four long-term circulation modes should be placed into operation within 24 hours of the LOCA.

Evaluation

Operating procedure 2104-1.3 adequately describes how to operate the decay heat removal system for its intended purposes. The procedure was not used on M rch 28, 1979, because the required prerequisite conditions could not be achieved.

OPERATING PROCEDURE 2102-3.3, DECAY HEAT REMOVAL VIA OTSG, REVISION 6, April 17, 1978

This procedure provides references, limits and precaut ons, prerequisites, and procedural steps for removing reactor decay heat using either reactor coolant pumps in operation or natural circulation cooling.

Prerequisites for placing the procedure in operation are listed in section 3.0 and are:

- The reactor is at "hot shutdown" (mode 3).
- The OTSG level is being maintained at 97-99 percent in the operating range by means of the main or emergency feedwater pumps.
- Decay heat is being removed via the turbine bypass valves, with the turbine header pressure set point at 855 psig and maintaining the reactor coolant system temperature at 532°F.

To maintain reactor coolant system temperature or to cool the plant down with reactor coolant pumps in operation, it is only necessary to change the setting on the turbine header pressure set point to the desired value.

The procedure for removing decay heat by natural circulation cooling assumes the following initial conditions:

- reactor coolant pumps tripped, reactor tripped, or turbine tripped;

1181 190

- steam pressure is being maintained at the turbine header set point (855 psig plus 125 psig following reactor trip), dumping steam to the main condenser through the turbine bypass valves or to the atmosphere if there is a low vacuum condition in the condenser;

- emergency feedwater pumps are maintaining OTSG level at 50 percent in the operating range; and

- pressure temperature limits are being maintained in accordance with figure 1.5.2. of the procedure.

Decay heat removal using natural circulation methods is accomplished simply by using the turbine header pressure set point to adjust OTSG pressure and thereby maintain RCS temperature within the pressure/temperature limitations of figure 1.5.2. This ensures adequate subcooling.

Evaluation

Operating procedure 2102-3.3 adequately describes procedures that will remove decay heat and either maintain a reactor coolant system temperature or cool the plant down to a desired temperature. The procedure is quite simple and straightforward. In fact, decay heat removal is accomplished with either forced or natural circulation by manipulation of only one control--the turbine header pressure set point.

OPERATING PROCEDURE 2104-1.1, CORE FLOODING SYSTEM

This procedure provides specific references, limits and precautions, prerequisites, and procedural steps covering system startup and shutdown.

Evaluation

The core flooding system is essentially a passive reservoir of water. This procedure adequately describes how the system is made ready for use and secured.

OPERATING PROCEDURE 2104-1.4, REACTOR BUILDING SPRAY

Operating procedure 2104-1.4 includes references, limits and precautions, prerequisites, and procedural steps covering operation of the reactor building spray system.

Evaluation

Operating procedure 2104-1.4 is adequate for the intended purpose of preparing the reactor building spray system for use and securing it.

OPERATING PROCEDURE 2104-6.3, EMERGENCY FEEDWATER, REVISION 4, JUNE 8, 1978

This procedure includes specific references, limits and precautions, prerequisites, and procedural steps for making the emergency feedwater system ready for operation. The procedure does not discuss operation of the system.

Two of the limits and precautions are of interest:

- 2.2.2 -- the emergency feedwater pumps will be put into standby during a unit heatup, after the first main feedwater pump has been placed in service, and EF-V11A and B have been placed in AUTO per 2102-1.1.

- 2.2.3 -- the maximum allowable number of cycles of the auxiliary feedwater nozzles in the OTSG is 80 for 80°F feedwater and 40 cycles for 40°F feedwater.

EP 2202-1.3 includes a system valve lineup that specifies that emergency feedwater header isolation valves, EF-V12A and EF-V12B, should be open (see Appendix B).

Evaluation

The procedure appears to be adequate for making the emergency feedwater system ready for service, but it does not include any provisions for system operation.

The limit concerning the maximum number of times that cold water can be injected into the OTSGs would preclude conducting emergency feedwater pumps surveillance unless a valve were shut in the flow path to preclude flow into the steam generator.

OP#2104-6.3 is not rigorous in its terminology. Use of the terms "emergency feedwater pump," "emergency feed pump," "main feedwater pump," "main feed pump," "main feedwater (pump)," "emergency feedwater," "auxiliary feedwater," "main steam," "MS," "auxiliary steam," and "aux steam," in the same document could lead to confusion.

OPERATING PROCEDURE 2105-1.3, SAFETY FEATURES ACTUATION SYSTEM, REVISION 2, OCTOBER 25, 1978

This procedure includes references, limits and precautions, prerequisites, and procedural steps for system startup, normal operation, and shutdown.

Evaluation

Operating procedure 2105-1.3 is adequate to carry out the intended purposes.

ABNORMAL PROCEDURE 2202-2.2, TURBINE TRIP, REVISION 7, OCTOBER 25, 1978

Abnormal procedure 2202-2.2 describes the immediate automatic actions, immediate operator actions, and followup operator actions to be taken in case of a turbine trip.

1181 192

Immediate automatic action step 2.ØA.3 states, "Pressurizer Power Operated Relief Valve open . . ." indicating that PORV opening normally would be expected on a turbine trip. Step 2.0A.5 indicates that if both main feedwater pumps have tripped, the steam-driven emergency feedwater pump and two motor-driven emergency feedwater pumps will start.

Immediate operator actions following a turbine trip include:

- verify that the turbine stop valves are closed and generator and field breakers are open;

- verify the start of the seal oil backup pump, the turbine gear oil pump, and the bearing lift pumps, and the closure of the extraction steam valves; and

- monitor pressurizer level and reactor coolant system pressure and temperature.

Followup operator action in section 3.0 includes:

- utilize pressurizer beaters and spray to control reactor coolant pressure at 2,155 psig and the steam header setpoint at 885 psig to control average coolant temperature at 582°F; adjust make-up and let-down flows to control pressurizer level at 240 inches; adjust feed flow to control OTSG levels at 30 inches; and

- if the turbine trip is due to a loss of both feed pumps, verify that emergency feed pumps have started and are delivering water to the OTSGs; control EF-V11A and B to maintain OTSG levels at 30 inches;

Evaluation

Followup action in AP 2203-2.2 does not include verifying the reclosure of the PORV, which apparently normally opens on a turbine trip. Otherwise, the procedure is adequate. Of interest is that the procedure recognizes that one of the likely causes of a turbine trip is loss of both running main feedwater pumps. In such an event, the procedure provides for making sure that emergency feedflow exists to both steam generators. The turbine trip procedure is written so as to minimize the amount of time that the plant is off the line. For instance, it prescribes that steam header pressure be maintained at 885 psig. The procedure also directs the operator to control pressurizer level at 240 inches, using let-down, if necessary. Whether this contributed to operator "mindset"--do not let pressurizer level go high following a turbine trip--cannot be determined.

ABNORMAL PROCEDURE 2203-2.6, POST-ACCIDENT HYDROGEN CONTROL, REVISION 1, JUNE 23, 1978

This procedure stipulates the taking of daily air samples for hydrogen in the reactor building following an accident and after the containment pressure reduces to normal and the activity level is reduced. Followup action calls for installing the hydrogen recombiner as the hydrogen concentration increases. If the hydrogen concentration cannot be maintained lower than 3.5 percent, purging of the reactor building must be commenced.

Evaluation

Abnormal procedure 2203-2.6 is seriously deficient in several respects:

- The term "accident" is not defined.
- No action is taken to even measure the hydrogen concentration until reactor building pressure has returned to normal and activity level is reduced.
- The procedure does not address the prompt generation of hydrogen in containment as it occurred at TMI. This appears to be a major oversight in view of the fact that a hydrogen burn actually took place on the first day of the accident.
- The procedure does not recognize the great difficulty in placing the hydrogen recombiner in operation because of radiation emanating from the recombiner and associated piping.

EMERGENCY PROCEDURE 2202-2.2, LOSS OF STEAM GENERATOR FEED, REVISION 3, OCTOBER 13, 1978

Emergency procedure (EP) 2202-2.2 lists symptoms, immediate actions, and followup actions for a loss of feedwater (FW) flow to both steam generators and a loss of flow to one steam generator.

Actions for a loss of both feed pumps include the following:

- 2.0A.1 -- Automatic Actions:
  - -- if loss of FW is due to loss of both feed pumps:
    - Reactor/turbine trip due to high RC pressure.
    - Emergency feed pumps EF-P-1, EF-P-2A, and EF-P-2B start and maintain OTSG level at 30 inches (S/U range indication).
    - If loss of FW is due to valves closing, ICS trips to track due to FW X-Limits.
- 2.0B.1 Manual Actions:
  - -- If loss of FW is due to loss of both feed pumps:
    - Trip the reactor.
    - Verify turbine trip and stop valves closed.

1181 194

- Verify EF-P-1, EF-P-2A, and EF-P-2B start as evidenced by pump discharge pressures.

- Verify emergency feedwater valves (EF-IIA(B)) are on automatic and controlling level at 30 inches.

Evaluation

The procedure requires that the reactor should be tripped manually in case of any loss of both feedwater pumps, regardless of whether an automatic trip took place or not. This raises the question of why an automatic reactor trip circuit was not installed in the reactor protection system for a loss of both feed pumps.

It is not clear why verification of emergency feed pump operation and feed flow is listed under manual immediate actions rather than followup actions, or why there is a difference in this respect between the automatic and manual cases.

EP 2202-2.2 does not make any mention of the almost certain operation of the PORV or checking to ensure that it functioned properly.

Otherwise, this procedure provides adequate guidance to the operator following loss of feedwater.

EMERGENCY PROCEDURE 2202-1.5, PRESSURIZER SYSTEM FAILURE

Emergency Procedure 2202-1.5 includes seven sections, each of which lists symptoms, immediate action, and followup action for the following casualties:

A. leaking pilot-operated (electromatic) relief valve (RC-R2)

B. inoperative pilot-operated (electromatic) relief valve (RC-R2)

C. leaking code relief valve (RC-R1A or RC-R1B)

D. inoperative code relief valve

E. inoperative pressurizer heaters

F. malfunction in pressurizer level indication or control

G. pressurizer spray valve failure

At some time during the accident on March 28, 1979, actual or suspected conditions applicable to sections A, B, C, E, and F existed.

Evaluation

Analysis of EP 2202-1.5 points out a number of deficiencies:

° The procedure is hard to use because there is no introductory section that indicates the contents or scope. Although the procedure is entitled "pressurizer system failure," it includes a variety of problems that might not be associated with a failure of the pressurizer system. There also should be an index.

- Section A. (Leaking PORV)

  -- Symptoms do not indicate that one cannot tell whether high relief valve discharge line temperatures are due to the leaking of a PORV or a code safety valve.

  -- The symptoms suggest incorrectly that there is no indication of reactor coolant drain tank temperature in the control room.

  -- The immediate action section requires the PORV isolation valve (RC-V2) to be shut any time the discharge line temperature exceeds 130°F.

  -- There is no mention in followup action of the need to recirculate water through the pressurizer to equalize boron concentrations.

- Section B (Inoperative PORV)

  -- The symptoms for a stuck open PORV do not include a discussion of position indication. The procedure assumes that this will be noted by the operator.

  -- The symptoms for a stuck open PORV do not mention: increasing RC drain tank level, rapidly decreasing make-up tank level, possible increasing pressurizer level, or decreasing RCS pressure.

  -- Possible increasing reactor building temperature, pressure and sump level.

  --- Automatic action indicates that all pressurizer heaters will be on below 2,105 psig. No mention is made about the possible loss of pressurizer heaters, which reportedly has been a chronic problem and was severe during the TMI accident.

  -- The action section does not include a warning against premature interruption of high pressure injection (HPI).

  -- The procedure requires the PORV block valve to be shut if the PORV itself fails shut.

  -- The entire section on a stuck open PORV should be in the loss-of-coolant emergency procedure. This was precisely the cause of the loss-of-coolant accident at Three Mile Island.

- Section C (Leaking Code Relief Valve)

-- The term "code relief" valve rather than the proper term "code safety" valve is used throughout. This improper terminology could cause confusion.

-- The same comment made above for a leaking PORV can be made here. Symptoms indicate erroneously that you can identify whether the leaking valve is the PORV or a code relief (safety) valve without taking any further action.

-- Followup action directs that a reactor coolant leakage rate measurement be taken. However, no mention is made of performing subsequent leakage rate measurements to determine trends.

° Section D (Inoperative Code Relief Valve)

-- The first two symptoms--code relief fails to open, code relief fails to close causes--are rather than symptoms.

-- Symptoms do not include RCS pressure dropping, increasing reactor coolant drain tank level, rapidly decreasing make-up tank level, possible increasing pressurizer level, and possible increasing reactor building temperature, pressure, or sump level.

-- For a fail-to-open code relief, the procedure does not mention as automatic action the de-energization of pressurizer heaters as it does in section B.

-- The term "safety injection" is used rather than "high pressure injection" as in section B.

-- In followup action, the procedure cautions against inserting any positive reactivity. With the reactor already fully shut down, it is not clear how positive reactivity would be inserted or what difference it would make.

-- The terms "code safety" and "code relief" are used interchangeably.

° Section E (Inoperative Pressurizer Heaters)

-- Followup action requires that if pressure cannot be maintained with the remaining heaters, continue load reduction to shutdown and possibly to a cooldown condition. This directive is not clear-cut and positive. It should state exactly the criteria to determine if the plant can stay in hot shutdown or if it must be placed in cold shutdown.

° Section F (Malfunction in Pressurizer Level Indication or Control)

1181 197

-- Symptom F.1.2 states, "Rapid change is indicated/recorded level due to loss of compensation or loss of power or d/p cell failure or other malfunction." This statement, coupled with the title of the section, suggests that the procedure is applicable to the case in which there is an actual pressurizer level control malfunction as opposed to a level indication malfunction. However, the procedure action paragraphs provide no guidance for actual pressurizer level control problems.

<u>EMERGENCY PROCEDURE 2202-1.1, REACTOR TRIP,
REVISION 6, OCTOBER 25, 1978</u>

This procedure provides symptoms and immediate and followup action steps for a reactor trip. Action required includes the following:

- Manually trip the reactor.
- Verify all in-limit lights are actuated, with the exception of group 8 rods.
- Close let-down valve MU-V376.
- Start a second make-up pump.
- Open MU-V16B as necessary to maintain 100 inches in the pressurizer.
- Verify that pressurizer heaters are off at 80 inches in the pressurizer.
- Announce "reactor trip" on page system.
- Monitor make-up tank level and maintain a level higher than 55 inches.
- Verify that pressurizer heaters and spray have returned RCS pressure to normal operating pressure (2,155 psig).
- Reduce pressurizer level set point to 100 inches.
- Verify that turbine bypass control valves are maintaining header pressure at 1,010 psig.
- Verify normal electrical lineup.
- Check that all RMS channels are normal.
- If reactor startup is not intended within 4 hours, raise OTSG level to 97 to 99 percent in the operating range.

<u>Evaluation</u>

Emergency procedure 2202-1.1 on reactor trip, in conjunction with AP 2203-2.2 (turbine trip), provides adequate guidance to operators for this casualty. The obvious intent of the procedure is to place or verify the plant in a safe condition and to minimize the length of the outage.

A major shortcoming of the procedure is that it does not mention determining the cause of the reactor trip and correcting it. For example, the trip could have been caused by high RCS pressure, low RCS pressure, high reactor coolant outlet temperature, or high reactor building pressure. Each of these conditions could be the result of a hazardous set of circumstances would that have to be dealt with to stabilize the plant.

EMERGENCY PROCEDURE 2202-1.3, LOSS OF REACTOR COOLANT/REACTOR COOLANT SYSTEM PRESSURE, REVISION 11, OCTOBER 6, 1978

Background

The most serious accident that can occur in a pressurized water power reactor is a loss-of-coolant accident (LOCA) or loss-of-coolant pressure. Either could result in core damage due to the lack of heat removal. Emergency procedure 2202-1.3 is intended to ensure that in case of the most severe LOCA or loss-of-coolant pressure that systems and operators will function to prevent core melt down, to maintain the integrity of the containment, and to ensure that the public is not exposed to radiation in excess of 10 CFR 100 limits. Similarly, the procedure should mitigate any LOCA or loss-of-coolant pressure of lesser severity.

Emergency procedure 2202-1.3 has as its analytical basis the Final Safety Analysis Report (FSAR), Section 6. The FSAR assumes that the worst case break is a 5 ft^2 break in a reactor coolant system hot leg. Cold leg breaks of various sizes are analyzed but are not considered as severe because less energy is released for a given size break.

A LOCA/loss-of-pressure event could be described as falling into one of the following three categories of severity, and the procedure must provide for each one:

1. A Leak. Mass loss from a leak can be accommodated from normal pressurizer water inventory and normal system pressure control. No safety system operation is necessary. Plant shutdown might be required due to technical specification limits on leakage, but the shutdown would be deliberate and orderly using normal procedures.

2. Small-Break LOCA. Engineered safeguards systems automatically activate and deliver water to the reactor coolant system. The break size is such that reactor coolant system pressure is above the shut off head of the low pressure injection pumps. All injection water is delivered by the high pressure injection systems. Decay heat is removed by fluid exiting the break, by heat up of high pressure injection water, by heat up of the auxiliary feedwater delivered to the steam generators, and possibly by steam release from the steam

generators. Containment isolation upon pressure rise in containment may not occur or may be delayed significantly.

3. Large-Break LOCA. Engineered safeguards systems automatically activate, as does the containment isolation signal from containment building pressure. The low pressure injection system delivers significant mass flow to the reactor coolant systems. Core flood tanks discharge to the reactor coolant system. Steam generators do not remove heat.

EVALUATION

General

The procedure does not identify an objective. The ultimate goal is not discussed. Symptoms are not listed in any priority, either by importance or by likelihood of occurrence. There is no statement to the effect that not all symptoms need be observed for the event to be taking place. The format of the procedure is difficult to follow; this has been exacerbated by the illogical insertion of a small break LOCA subprocedure. Terminology is confusing. For instance the following terms are used interchangeably to express the same phenomenon: high pressure injection, HPI, safety injection, ESF (engineered safety features).

The procedure has no guide to indicate what is contained inside; there is no list in the beginning of cases (similar sets of conditions which would dictate similar courses of action).

The procedure does not list any initial conditions or applicability such as reactor critical; reactor at power; reactor in hot standby; modes 1, 2, 3, 4, and so on.

Specific Comments

The first category of the procedure is "Leak within Capability of System Operation." This is confusing because it does not specify what system is capable of operating--makeup system, high pressure injection system, reactor coolant system, or nuclear steam supply system.

- Symptoms

  -- Symptom 1.1 is described as "Initial loss of reactor coolant pressure and decrease in pressurizer level becoming stable after a short period of time." This symptom assumes that correct action has been taken to correct the loss of pressure and level.

  -- The symptoms do not include reactor coolant drain tank pressure or temperature increases increasing reactor building pressure or reactor coolant system leakage calculations.

- Manual action makes no provision for tripping the reactor on low reactor coolant system pressure even if the pressurizer level is not yet low.

- Manual initiation of high pressure injection does not mention the small-break LOCA response which is appropriate in this section.

- If high pressure injection is initiated manually the operator is required to throttle HPI as necessary to maintain pressurizer level at 220 inches (that is, not permit it to increase).

- Step 3.2.8.1 says, "THROTTLE HPI string(s) flow rate to at least 500 gpm each 250 gpm per leg)." This is confusing.

- The procedure does not mention ensuring that containment isolation is set. The last statement in section A following a three-page discussion of how to place the decay heat removal system in operation is, "Reactor Building Isolation Initiated." It is not clear whether this is merely a statement or a much belated procedural step.

- Section B of the procedure is entitled "Leak or Rupture of Significant Size Such that Engineered Safety Features Systems are Automatically Initiated." It is not clear that this includes the procedure for a small-break LOCA.

- Symptoms do not:

  -- recognize that pressurizer level may increase, as it did on March 28, 1979;

  -- mention a high make-up flow alarm;

  -- mention reactor coolant drain tank high temperature or pressure; or

  -- mention a possible decrease in reactor coolant system flow.

- The procedure provides symptoms that are intended to determine whether a reactor coolant system leak, a steam leak, or an OTSG tube rupture are giving the observed symptoms. It is not clear why a steam leak would result in high pressure injection. Symptoms of an OTSG tube failure do not include increasing steam generator water level or pressure.

- The procedure states in one place (B.1.1.3) that safety injection commences at 1,640 psig and in another place (B.2.1.3) at 1,600 psig.

- Under manual action is a section on small-break LOCA response:

  -- This section is misplaced.

  -- The operator is required to verify that a small-break LOCA exists. How this is to be determined is not indicated.

  -- The section has its own illogical symptoms (of a small-break LOCA with single failure), which are:

1181 201

- Safety features actuation system (SFAS) initiation and only one make-up pump (MUP) started; or

- SFAS initiation and loss of 2-1E or 2-2E.

-- A caution note begins: "If the LOCA was ES with loss of MUP. . . ." This is confusing.

-- The small-break LOCA procedure was the result of misinterpreted instructions issued in May 1978 from B&W concerning a worst case small-break LOCA. The procedure as written is illogical and confusing and cannot be followed.

- Step 3.2 requires that a site emergency be declared for any leak or rupture that results in ESF actuation.

- Following action does not make any provision for shifting to low pressure injection or shifting injection pump suction to the reactor building sump when the borated water storage tank goes dry.

- The procedure requires that engineered safeguards be bypassed regardless of the seriousness of the leak or rupture in order to prevent exceeding HPI pump runout. This means, in effect, that avoiding runout limits is more important than continuing high pressure injection at the maximum rate.

- The procedure includes no cautionary note or other guidance concerning ensured adequate core cooling.

- The procedure is silent with regard to when HPI can be secured if it initiated automatically.

In summary, the LOCA/loss-of-pressure emergency procedure may not be adequate to ensure that the integrity of the core will be maintained in case of a LOCA.

Appendix B is a copy of emergency procedure 2202-1.3 for reference.

Appendix C is a copy of a portion of the loss-of-reactor coolant procedure from another utility. Included for comparison is a list of cases considered by the procedure and a copy of Case A7: "Small Break--no Feedwater - No RC Pumps - Reactor Trip," a worst case small-break situation.

FINDINGS

Analysis of the technical aspects of the operating and emergency procedures that were used or that were applicable on March 28, 1979, at TMI-2 suggests the following findings:

- Although they may be deficient in minor respects, the procedures listed below are adequate for their intended purpose:

-- Operating procedure 2102-2.1, power operations

-- Operating procedure 2104-1.3, decay heat removal system

-- Operating procedure 2102-3.3, decay heat removal via OTSG

-- Operating procedure 2104-1.1, core flooding system

-- Operating procedure 2104-1.4, reactor building spray

-- Operating procedure 2104-6.3, emergency feedwater

-- Operating procedure 2105-1.3, safety features actuation system

The following procedures contain significant deficiencies which could cause confusion or lack of action but would not preclude their use by thinking operators:

-- Operating procedure 2103-1.4, reactor coolant pump operation.

- Precludes pump operation with excessive vibration.

- Whether pump should be tripped under low pressure, LOCA condition was not clear.

-- Abnormal procedure 2203-2.2, turbine trip.

- Does not require operator to verify that the PORV is shut although it is expected to open.

- The operator is directed to use let-down, as necessary, to preclude pressurizer level from exceeding 240 inches following a turbine trip.

-- Emergency procedure 2202-2.2, loss of steam generator feed:

- Requires immediate manual reactor trip on loss of both feedwater pumps.

- Does not require verification of proper PORV operation.

-- Emergency procedure 2202-1.1, reactor trip:

- The procedure makes no provision for determining the cause of the reactor trip and correcting it.

The following procedures were so deficient as to be inadequate:

-- Operating procedure 2103-1.3, pressurizer operation:

- States the pressurizer may not be taken solid for any reason except hydrostatic tests.

1181 203

-- Abnormal procedure 2203-2.6, post accident hydrogen control:

- The procedure does not recognize the rapid generation of hydrogen as occurred at TMI.

- The procedure does not recognize any difficulties which might be encountered in placing the hydrogen recombiner in operation.

-- Emergency procedure 2202-1.5, pressurizer system failures:

- The procedures basic structure is very confusing; some sections should be in the loss-of-coolant procedure; symptoms are significantly incomplete, misleading, or erroneous.

- No guidance is given for actual pressurizer level control problems.

(3) Terminology is sloppy.

-- Emergency procedure 2202-1.3, loss-of-reactor-coolant/reactor coolant system pressure:

- Procedure lacks objectives.

- Symptoms are incomplete, misleading, or erroneous.

- The procedure is difficult to use. Cases are not defined.

- The operator is required to throttle HPI to prevent pump runout, regardless of the severity of the accident.

- The procedure does not promptly ensure that containment is isolated.

- A section on Small Break LOCA response is illogical and cannot be followed.

- No cautionary guidance is included regarding core covering and cooling.

° Operators were prohibited by the technical specifications from permitting the pressurizer to go solid.

° Some procedures emphasize avoiding equipment damage over keeping the core covered with water or maintaining core cooling.

° The procedure for decay heat removal via the OTSG is simple, straight-forward and, if followed, can be used to cool the core either with or without running reactor coolant pumps.

- Procedures recognize that the PORV will open following a turbine trip.

- Procedures in general are written to minimize "outage" time and maximize "plant availability."

- The turbine trip procedure requires that if the cause of the trip was loss of feedwater flow the operator should verify emergency feedwater flow to the steam generators.

- In addition to the major, substantive deficiencies cited in the analysis, Metropolitan Edison procedures contain many minor errors in substance, typographical errors, imprecision or sloppiness in terminology, format deficiencies, and the like, which reflect a lack of quality essential in nuclear power plant procedures.

°In reviewing Three Mile Island procedures, there is no evidence that operating experience (lessons learned) at the Island or at other Babcock & Wilcox plants was incorporated into operating procedures.

ACRONYMS

ABP	abnormal procedure
BWST	borated water storage tank
CFR	Code of Federal Regulations
DHRS	decay heat removal system
EF	emergency feedwater
EP	emergency procedure
ES/ESF	engineered safety features
FA	fuel assembly
FSAR	Final Safety Analysis Report
FW	feedwater
HPI	high pressure injection
LOCA	loss-of-coolant accident
MS	main steam
MU	make-up
MUP	make-up pump
NRC	Nuclear Regulatory Commission
OP	Operating procedure
OTSG	once-through steam generator
PORV	pilot-operated relief valve
RB	reactor building
RC	reactor coolant
RCS	reactor coolant system
SFAS	safety features actuation system

1181 206

VI. APPENDICES

A. Operating Procedure 2202-1.5.

B. Emergency Procedure 2202-1.3.

C. Portions of Loss of Coolant Accident Emergency Procedure from Oconee Nuclear Station.

CONTROLLED COPY
TRAINING DEPT.

2202-1.5
Revision 3
09/29/78

APPENDIX A -1

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.5
PRESSURIZER SYSTEM FAILURE

Table of Effective Pages

Page	Date	Revision	Page	Date	Revision	Page	Date	Revision
1.0	06/22/77	1	26.0			51.0		
2.0	06/22/77	1	27.0			52.0		
3.0	06/22/77	1	28.0			53.0		
4.0	11/17/77	2	29.0			54.0		
5.0	09/29/78	3	30.0			55.0		
6.0	03/11/77	0	31.0			56.0		
7.0	03/11/77	0	32.0			57.0		
8.0	03/11/77	0	33.0			58.0		
9.0	03/11/77	0	34.0			59.0		
10.0	03/11/77	0	35.0			60.0		
11.0	03/11/77	0	36.0			61.0		
12.0			37.0			62.0		
13.0			38.0			63.0		
14.0			39.0			64.0		
15.0			40.0			65.0		
16.0			41.0			66.0		
17.0			42.0			67.0		
18.0			43.0			68.0		
19.0			44.0			69.0		
20.0			45.0			70.0		
21.0			46.0			71.0		
22.0			47.0			72.0		
23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

[illegible] Date 9/28/7[illegible]
Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

[illegible] Date [illegible]

Manager Generation Quality Assurance Approval NA Date —

1181 208

TMI 55-A [illegible] 8/77

THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PR[illegible] 2202-1.5

PRESSURIZER SY[illegible]LURE

SECTION A Leaking Pilot Operated (electromatic) Relief Valve (RC-R2)

A.1 SYMPTOMS

1. Relief valve discharge line temperature exceeding the normal 130°F. Alarms on computer at 200°F.
2. RC drain tank pressure above normal on the control room radwaste disposal control panel and temperature above normal on the local radwaste disposal control panel.
3. RC System makeup flow above normal for the variable letdown flow and RC pump seal in-leakage conditions.
4. Boric Acid concentration continually increasing in the pressurizer.

A.2 IMMEDIATE ACTIONS

A. Automatic Actions

1. None.

B. Manual Actions

1. Close the Electromatic Relief Isolation Valve, RC-V2.

A.3 FOLLOW-UP ACTION

1. Repair during next shutdown.
2. Limit rate of change on ICS to less than 1% per minute while RCV is closed except for runbacks.

SECTION B Inoperative Pilot operated (electromatic) Relief Valve (RC-R2)

B.1 SYMPTOMS

1. RC System pressure is above 2255 psig and RC-R2 fails to open.
2. RC System pressure is below 2205 psig and RC-R2 fails to close.
3. RC-R2 discharge line temperature is above the 200°F alarm. Computer Point (402)
4. The RC drain tank pressure and temperature are above normal on the control room radwaste disposal control panel 8A.

B.2 IMMEDIATE ACTION

A. Automatic Action

1. For a failed closed RC-R2:
   a. Pressurizer heaters off at 2160 psig. Spray valve RC-V1 is open above 2205 psig.
   b. Reactor trip occurs at 2355 psig.
   c. Pressurizer code relief valves open at 2450 psig.
2. For a failed open RC-R2:
   a. All pressurizer heater banks on full below 2105 psig.
   b. Reactor trips at 1900 psig or variable pressure temperature.
   c. High Pressure Injection is actuated at 1600 psig.

B. Manual Action

1. For a failed close RC-R2:
   a. Shift spray valve RC-V1 to "MANUAL" and open further for additional spray flow.
   b. Insure all pressurizer heaters off above 2160 psig.

c. If reactor power is being changed (except for a runback) stop the power change until pressure is returned to normal.

d. Isolate RC-R2 by closing RC-V2.

2. For a failed open RC-R2:

a. Close Electromatic Relief Isolation Valve (RC-V2).

b. Insure all pressurizer heaters on below 2105 psig.

B.3 FOLLOW-UP ACTION

1. Return system pressure and temperature to normal.

2. Reduce ICS Rate of Change to less than 1% per minute (except for Runbacks.)

SECTION C Leaking Code Relief Valve (RC-R1A or RC-R1B)

C.1 SYMPTOMS

1. Code relief valve discharge line temperature(s) exceeding the computer normal 130°F. Computer alarms at 200°F. Computer Point (403) (404)
2. RC drain tank pressure and temperature above normal on the control room radwaste disposal control panel 8A.
3. RC System makeup flow is above normal for the variable letdown flow and RC pump seal in-leakage conditions.
4. Boric Acid Concentration continually increasing in the Pressurizer.

C.2 IMMEDIATE ACTION

A. Automatic Action

1. None.

B. Manual Action

1. Determine RC leakage according to 2301-3D3.

C.3 FOLLOW-UP ACTION

1. If RC system identified leakage is in excess of 10 gpm, reduce the leakage rate to within limits within 4 hours or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.
2. It will be necessary to recirculate the pressurizer through the spray valve to equalize Boron concentration.
3. Place Code Relief Discharge Line temperatures on Analog Trend Recorder.

SECTION D Inoperative Code Relief Valve (RC-R1A or RC-R1B)

D.1 SYMPTOMS

1. Code relief valve(s) fail to open when RC pressure is above 2450 psig.
2. Code relief valve(s) fail to close when RC pressure is below 2325 psig.
3. Code relief valve(s) discharge line temperature is above 200°F alarm.
4. The RC Drain Tank pressure and temperature are above normal on the control room radwaste disposal control panel.
5. RC system makeup flow is above normal for the variable letdown flow and the RC pump seal in-leakage conditions.

D.2 IMMEDIATE ACTION

A. Automatic Action

1. For a fail to open code relief valve:
    a. Reactor trip occurred at 2355 psig.
    b. Spray valve RC-V1 opened above 2205 psig.
2. For a fail to close code relief valve:
    a. Reactor trip occurs at 1900 psig or on variable P/T.
    b. Increased makeup flow.
    c. All pressurizer heaters energized.
    d. Safety Injection is actuated at 1600 psig.

B. Manual Action

1. For a fail to open code relief valve:
    a. Place pressurizer spray valve in "MANUAL" and open further for additional spray flow.

b. Verify pressurizer heaters are "OFF" at plant control panel.

2. For a fail to close code relief valve:

a. Turn all heaters "ON" at plant control panel.

b. Isolate letdown flow at plant control panel by "CLOSING" MU-V376.

c. Open DH-V5A. Start MU-P1A if necessary. Attempt to control pressurizer level using MU-V16B.

d. Manually initiate safety injection if required to maintain pressurizer level.

D.3 FOLLOW-UP ACTION

1. For a fail to open code relief valve:

a. Proceed with cooldown.

2. For a fail to close code relief valve:

a. Hold pressurizer, if possible, at or greater than 220 inches with Safety Injection.

b. Proceed with cooldown.

c. With no pressurizer code safety valve operable, immediately suspend all operations involving positive reactivity changes and place an operable DHR Loop into operation in the shutdown cooling mode.

d. With a pressurizer code safety valve inoperable, either:

1. Restore the inoperable valve to operable status within 15 minutes or

2. Be in Hot Shutdown within 12 hours.

1181 214

SECTION E Inoperative Pressurizer Heaters

E.1 SYMPTOMS

1. Heater banks fail to energize or de-energize if RC pressure is at heater bank setpoint.

	Bank 1	Bank 2	Bank 3	Bank 4	Bank 5	Units
ON	2150	2145	2135	2120	2105	PSIG
OFF	2160	2155	2155	2140	2125	PSIG

NOTE: Banks 1, 2, and 3 are full on at "ON" setpoint.

2. Pressurizer level Lo-Lo alarm at 80 inches.
3. Pressurizer heater power supply ground alarm.
4. Abnormal console indicating lights for the heating groups.
5. High (2255 psig) or low (2055 psig) pressure alarms.

E.2 IMMEDIATE ACTION

A. Automatic Action

1. For energized heaters and rising pressure:
   a. Pressurizer spray valve (RC-V1) open (red and green console jog button lights).
2. For loss of heaters and decreasing pressure: None.

B. Manual Action

1. If control malfunction is suspected:
   a. Place heater controller in "MANUAL".
2. For energized heaters and rising pressure:
   a. Attempt to de-energize all heaters except Banks 1 or 2. (Groups 12 or 13 respectively).
3. For loss of heaters and decreasing pressures:
   a. Attempt to energize backup heaters from plant control panel.
   b. If a. is unsuccessful, begin reducing unit load.

E.3 FOLLOW-UP ACTION

1. For energized heaters and rising pressure:

   a. Open heater breakers in question at the pressurizer heater control centers except for Banks 1 or 2 (Groups 12 or 13 respectively).

   b. Control RC pressure at the normal 2155 psig set point with the pressurizer spray valve (RC-V1) in "MANUAL".

2. For loss of heaters and decreasing pressure:

   a. Determine cause.

   b. If pressure cannot be maintained with the remaining heaters, continue load reduction to shutdown and possibly cooldown condition.

   c. Close RC-V3 and reopen periodically to maintain spray line temperature greater than 540°F.

SECTION F Malfunction In Pressurizer Level Indication or Control

F.1 SYMPTOMS

1. Disagreement between the console recorder level readouts of more than 12 inches.
2. Rapid change in indicated/recorded level due to loss of compensation or loss of power or d/p cell failure or other malfunction.

F.2 IMMEDIATE ACTION

A. Automatic Action

1. If indication fails low:
   Pressurizer heaters trip @ 30 inches, makeup valve MU-V17 opens, and RC pressure increases.
2. If indication fails high:
   Makeup valve MU-V17 closes.

B. Manual Action

1. When any two of three console recorder level transmitter readouts disagree by more than 12 inches, take manual control of level and then select the third transmitter for indication.
2. Re-energize heaters if tripped due to malfunction.

F.3 FOLLOW-UP ACTION

1. If the switching level transmitters has not rectified the condition, switch to the alternate temperature detector.
2. If pressurizer level recorder indication is lost, select another transmitter or use the computer for level indication.

SECTION G Pressurizer Spray Valve Failure (RC-V1)

G.1 SYMPTOMS

1. Pressurizer spray valve (RC-V1) fails to open when the RC system pressure is greater than 2205 psig.
2. Pressurizer spray valve (RC-V1) is open when the RC System is less than 2155 psig.

G.2 IMMEDIATE ACTION

A. Automatic Action

1. RC system pressure greater than 2255 psig activates RC-R2 electromatic relief and the high pressure alarm.
2. RC-V1 failing open (in auto) causes RC system pressure to stablize at approximately 2100 psig with all heater "on".
3. Failure when manually opened beyond the automatic limit position causes continued pressure drop and alarm at 2055.

B. Manual Action

1. Control RC-V1 opening or closing in "MANUAL" with jog buttons.
2. If the spray valve has failed open, control pressure by closing the pressurizer spray isolation valve (RC-V3).

NOTE: If the pressurizer spray isolation valve (RC-V3) is closed, it must be periodically cycled to keep the spray line warm. Cycle RC-V3 is open as necessary to stay above RC pressurizer spray line temperature alarm of 540^{o}F. (Computer point 0405).

CAUTION: Do not exeed a ΔT of 410°F between pressurizer temperature and reactor coolant hot leg temperature.

3. Reduce rate of ICS load change to less than 1% per minute.

G.3 FOLLOW-UP ACTION

1. Continue plant operation with reduced rate of load change.
2. Check thermal overload on RC-V1 and reset if necessary.

G/r m)
117

(26)

CONTROLLED COPY
CENTRAL FILE

THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-1.3

LOSS OF REACTOR COOLANT/REACTOR COOLANT SYSTEM PRESSURE

Table of Effective Pages

Page	Date	Revision	Page	Date	Revision	Page	Date	Revision
1.0	06/22/77	1						
2.0	05/12/78	8						
3.0	12/30/77	3						
4.0	12/30/77	3						
5.0	06/22/77	1						
6.0	03/01/78	5						
7.0	09/29/78	10						
8.0	05/12/78	8						
9.0	06/20/78	9						
10.0	05/12/78	8						
10.1	05/12/78	8						
10.2	05/12/78	8						
11.0	06/22/77	1						
12.0	06/22/77	1						
13.0	04/18/78	7						
14.0	10/06/78	11						
15.0	04/18/78	7						
1? 0	12/30/77	3						
[illegible]	06/22/77	1						
[illegible]	04/18/78	7						
19.0	12/30/77	3						
20.0	12/30/77	3						
21.0	12/30/77	3						
22.0	12/30/77	3						
23.0	06/22/77	1						

8113020 08

Unit 1	Unit 2
Unit 1 Staff Recommends Approval Approval NA Date —— Cognizant Dept. Head	Unit 2 Staff Recommends Approval Approval NA Date —— Cognizant Dept. Head
Unit 1 PORC Recommends Approval NA Date —— Chairman of PORC	Unit 2 PORC Recommends Approval RP Warren Date 10/6/78 V - Chairman of PORC
Unit 1 Superintendent Approval NA Date ——	Unit 2 Superintendent Approval [illegible signature] Date 10/6/78

Manager Generation Quality Assurance Approval NA Date ——

THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-1.3

LOSS OF REACTOR COOLANT/REACTOR COOLANT SYSTEM

PRESSURE

A. Leak or Rupture Within Capability of System Operation.

1.0 SYMPTOMS

1.1 Initial loss of reactor coolant pressure & decrease in pressurizer level becoming stable after short period of time.

1.2 Possible reactor building high radiation and/or temp. alarm.

1.3 Possible reactor building sump high level alarm.

1.4 Make-up tank level decreasing > 1" in 3 min.

1.5 Possible makeup line high flow alarm. ?

1.6 RB Fan Drip Pan Level Hi Alarms.

NOTE: The operator may distinguish between a loss of coolant inside containment, an OTSG tube rupture and a steam line break by the following symptoms which are unique to the aforementioned accidents.

1. Loss of coolant inside Rx Bldg. - particulate, iodine & gas monitor alarm on HP-R-227 "Reactor Building Air Sample".

2. OTSG tube rupture - gas monitor alarm on VA-R-748.

3. Steam line break

(1) Low condensate storage tank level alarm - and or low hot well level alarm.

(2) FW Latch System Actuation.

2.0 IMMEDIATE ACTION

2.1 Automatic Action:

2.1.1 MU-V17 will open to compensate for reduced pressurizer level.

2.1.2 Additional pressurizer heaters will come on in response to reduced reactor coolant pressure.

2.2 Manual Action

2.2.1 Verify MU-V17 open and pressurizer heaters on.

2.2.2 "CLOSE" MU-V376 letdown isolation valve, & "START" the backup MU pump, if required.

2.2.3 Reduce load at 10% minute & proceed with normal shutdown.

2.2.4 "LINE-UP" waste transfer pump from a R.C. Bleed Holdup Tank & pump to the makeup tank to maintain required level.

2.2.5 If for any reason the operator cannot maintain Make-up Tank and Pressurizer levels above their respective low level alarm setpoints, "TRIP" the reactor, "INITIATE" Safety Injection manually (push buttons on panel 3), & then "Close" MU-V12.

3.0 FOLLOW UP ACTION

3.1 Safety Injection Not Initiated.

3.1.1 Initiate unit shutdown & cooldown per 2102-3.1 and 2102-3.2 respectively.

3.2 Safety Injection Manually Initiated (HPI and LPI).

3.2.1 Verify that the Makeup Pumps & Decay Heat Removal Pumps start satisfactorily.

3.2.1.1 Close MU-V12 and MU-V18.

3.2.2 Bypass the SAFETY INJECTION by DEPRESSING the Group Reset Pushbuttons & "THROTTLE" MU-V16A/B/C/D as necessary to maintain 220" pressurizer level and not exceed 250 GPM/HPI flow leg.

3.2.3 If MU pump flow drops below 95 GPM, trip excess MU pumps.

<u>NOTE</u>: HPI String A flow is the sum of MU 23 FE1&2. HPI String B flow is the sum of MU23 FE 3&4.

3.2.4 Verify that Safety Injection equipment is in its ESF position as shown in Table A-1.

3.2.5 <u>CAUTION</u>: Continued operation depends upon the capability to maintain pressurizer level and RCS pressure above the 1640 PSIG Safety Injection Actuation setpoint.

1. If pressurizer level can be maintained above the low level alarm point and the RCS pressure above the Safety Injection Actuation point, then proceed to step 3.2.6.
2. If pressurizer level cannot be maintained above the low level alarm point and the RCS pressure above the Safety Injection Actuation point, then the plant has suffered a major rupture and operation should continue according to Part B - Leak or Rupture of Significant Size Such that Engineered Safety Features Systems are Automatically Initiated.

3.2.6 With the pressurizer level and RCS pressure being maintained within allowable limits, initiate plant shutdown and cooldown per 2102-3.1 and 2102-3.2, respectively.

<u>NOTE</u>: The HPI System is being used for makeup control and valves MU-V16A/B/C/D will have to be throttled to maintain pressurizer level. As RCS pressure decreases, it may be possible to return to the normal makeup flowpath and secure HPI. If MU pump flow drops below 95 GPM as a result of throttling, "Open" MU-V36 & 37 to provide MU pump recirculation path to MU Tank. Monitor MU Tank level and open MU-V12 as required.

1181 223

3.2.7 At the time the DH System is to be brought on line for normal cooling only one DH string should be used for decay heat removal (i.e. - recirculation from the RC System). The other DH string should be maintained on standby for use in recirculating water from the RB sump to the RC system.

<u>NOTE</u>: Trip Reactor Coolant Pumps before R.C. Pressure decreases below pump NPSH (See figure 1 of 2102-3.1/3.2.

3.2.8 When the Borated Water Storage Tank level decreases to 12' as indicated on panel 8, Shift the MU/HPI pump(s) suction from the BWST to the RB sump if RCS Pressure is greater than 200 psig as follows: (assume DH string A(B) is being used for decay heat removal and DH string B(A) is being maintained on standby):

3.2.8.1 If not already done, "THROTTLE" HPI string(s) flow rate to at least 500 gpm each < 250 gpm per leg) using control valves MU-V16A/B/C/D (or MU-V17 if flow has been returned to the normal makeup flow path). Flow rate indication and valve control in control room on Panel 8 and 3, respectively.

3.2.8.2 "OPEN" valve DH-V7B(A) in crossover line from LPI String B(A) to HPI string B(A) (suction of HPI pumps)..

"REPOSITION" HPI flow control valves MU-V16A/B/C/D (or MU-V17 if flow has been returned to the normal makeup flowpath). HPI flow would increase because of increased HPI pump suction pressure.

3.2.8.3 When the BWST level decreases to 7', verify automatic transfer to the RB sump is initiated. Verify OPEN suction valve for string B(A), DH-V6B(A) from the RB sump.

3.2.8.4 When the suction valve from the RB sump DH-V6 B(A), is fully open, then "CLOSE" the ECCS suction valve, DH-V5B(A), from the BSWT (valve controls and position indication in control room). The ECCS B(A) string is now in "piggy-Back" operation providing makeup to the RCS from the RB sump as required.

3.2.9 After R.C.S. pressure decreases to ≈ 200 psig, throttle HPI discharge flow by throttling MU-V16A/B/C/D. Observe that LPI pumps now deliver water to RCS via DH-V4A/B.

3.2.10 When MU-V16A/B/C/D (HPI flow valves) are closed, stop the Hi pressure injection pumps & close DH-V7A & 7B from the LPI pump discharge. Injection flow path is now as follows:

Spill coolant to RB sump, RB sump to LPI pumps,
LPI pumps to RCS via DH-V4A/B.

3.2.11 Throttle DH-V128A & B as required to maintain 220" pressurizer level and max. LPI pump flow of 3000-3300 gpm. Within about 24 hours, establish a long-term cooling circulation mode as described in 2104-1.3 and listed below.

Mode 1 Forced circulation using decay Heat drop line.

Mode 2 Gravity draining reactor coolant hot leg to the Reactor Building sump via the D.H. drop line.

Mode 3 Hot leg injection using Pressurizer Auxiliary Spray Line.

Mode 4 Reverse flow through the Decay Heat Drop line into "B" Reactor Coolant Loop Hot Leg.

3.2.12 Evaluate radiation levels & initiate action for Site Emergency as outlined in the TMI radiation emergency plan.

3.2.13 Reactor Building Isolation Initiated

1. Refer to Section B, 3.0 & complete all steps.

B. LEAK OR RUPTURE OF SIGNIFICANT SIZE SUCH THAT ENGINEERED SAFETY FEATURES SYSTEMS ARE AUTOMATICALLY INITIATED.

1.0 SYMPTOMS

1.1 Rapid continuing decrease of reactor coolant pressure.

(1) Lo alarm 2055 psig.

(2) Lo-Lo-alarm 1700 psig.

(3) Safety Injection actuation at 1640 psig.

1.2 Rapid continuing decrease of pressurizer level.

(1) Lo alarm 200".

(2) Lo-Lo alarm 80" (Interlock heater shutoff).

1.3 Hi radiation alarm in Reactor Building.

1.4 Reactor Building Ambient Temperature Alarm.

1.5 Hi Reactor Building Sump level.

1.6 Hi Reactor Building pressure (R.C.S or main steam line rupture).

1.7 Rapidly decreasing make-up tank level.

1.8 Both core flood tanks levels & pressures are decreasing.

NOTE: The operator may distinguish between a loss of coolant inside containment, an OTSG tube rupture and a steam line break by the following symptoms which are unique to the aforementioned accidents.

1. Loss of coolant inside Rx Bldg. - particulate, iodine gas monitor alarm on HP-R-227 "Reactor Building Air Sample."

2. OTSG tube rupture - Gas monitor alarm on VA-R-748.

3. Steam break inside Rx Bldg:

(1) Low condensate storage tank level alarm - and or low hot well level alarm.

(2) FW Latch System Actuation.

2.0 IMMEDIATE ACTION

2.1 Automatic Action.

2.1.1 Reactor trip 1900 psig.

2.1.2 Turbine Trip.

2.1.3 Safety Injection initiated @ 1600 psig R.C.S pressure, or 4 psig Reactor Building pressure.

2.1.4 Both Core Flood Tank levels & pressures may decrease depending upon rupture size and R.C.S. pressure. ($\leq$ 600 psig).

2.1.5 Reactor Building Isolation & Cooling initiated. (R.B. Press. $\geq$ 4 psig).

2.1.6 Reactor Building Spray if the Reactor Building pressure is greater than 30 psig.

2.2 Manual Action.

2.2.1 "CLOSE" MU-V12 and MU-V18.

2.2.2 Small Break LOCA Response

2.2.2.1 Within 2 minutes of the LOCA the CRO dedicated to recognition of a small break LOCA must complete the following:

a. Verify that small break LOCA with single failure symptoms exist.

Symptoms: 1. SFAS initiation and only one MUP started, or

2. SFAS initiation and loss of 2-1E or 2-2E

1181 227

b. DISPATCH designated LOCA Response Primary A.O. to OPEN MUP Discharge Cross-connect.

c. PROCEED to MU-V16A & B or MU-V16C & D.

d. Within 5 minutes of the LOCA the MUP discharge crc connect valve must be opened off its closed seat and one of the MU-V16 valves on the side of the single failure must be opened 2 turns.

2.2.2.2 CRO at MU-V16A & B or MU-V16C & D must ESTABLISH communications with the Control Room.

2.2.2.3 Once in communication with the control room the CRO at MU-V16A & B or C & D continue to open the valves to establish 125 gpm per leg; while the control room CRO THROTTLES MU-V16C & D or A & B to prevent pump runout.

<u>NOTE</u>: If the LOCA was E.S. with loss of MUP all MU-V16 valves would initially be open. The operator at the controls must in this case throttle all 4 MU-V16 valves to 125 gpm/leg. Also in this case the CRO dedicated to Small Break LOCA would not need to go to the MU-V16 valves. He should go to the MUP discharge cross connect valves and assist the A.O. to speed up the opening of the MUP discharge cross connect.

SMALL BREAK LOCA ACTION TIMES

EVENT	TIME From Occurance
Recognition	$\leq$ 2 minutes
CRO to MU-16's	$\leq$ 4.5 minutes

SMALL BREAK LOCA ACTION TIMES (Cont'd)

EVENT	TIME From Occurance
AO to MUP Discharge X-Connects	$\leq$ 3.5 minutes
Communications Established With CRO at MU-V16's	$\leq$ 5.0 minutes
One of the Single Failure Side MU-V16's Open 2 turns	$\leq$ 5.0 minutes
Discharge X-Connect OFF Closed Seat	$\leq$ 5.0 minutes
Discharge X-Connect Open	$\leq$ 10.0 minutes
MU-V16's Throttled to 125 gpm/leg	$\leq$ 10.0 minutes

2.2.3 Verify Hi pressure injection is operating properly as evidenced by injection flow in all four legs. (MU-V16A/B/C/D). Flow indicated on MU23 FE1,2,3,4.

2.2.4 "TRIP" reactor coolant pumps before reaching 1200 psig.

2.2.5 Verify Reactor Building Cooling and Isolation is operating properly.

3.0 FOLLOW UP ACTION

3.1 Verify that all E.S.F. equipment is in its ESF position, by observing that all equipment status lights indicate as shown in Table B-1.

3.1.1 Check locked valve status book and verify closed or close the following manual containment isolation valves MU-V330, MU-V364, CF-V114A, CFV114B, CF-V145, CF-V146, DH-V187, and DW-V28.

NOTE: Should any component not operate properly, attempt to actuate it at its remote switch in the Control Room. If it still does not operate, & the component has a local control station attempt to operate the component locally.

3.2 Notify Shift Foreman, who notifies all Station personnel over the cross-tied PA system that a site emergency has occured.

3.3 0 to 20 or 30 minutes past LOCA until sump recircualtion is initiated: Control Room operator continuously monitors the following:

1181 229

3.3.1 Liquid levels in the:

1. Borated Water Storage Tank, DH-T1, (DH-3-LI 1/2).
2. Sodium Hydroxide Tank, DH-T2, (DH-7-LI).

3.3.2 Safety Features flow rate in each of the following:

1. Two Low Pressure (Decay Heat) Injection lines, DH-1-FI 1 and 2.
2. Four High Pressure (Makeup) Injection lines, MU-23-FI 1, 2, 3, and 4.
3. Two Reactor Building Spray injection lines, BS-1-FI and 2.
4. Four of five reactor building emergency cooling river water lines AH-FI-5620, 5621, 5522, 5623, or 5624 respectively.

3.3.3 Reactor Building environmental indications:

1. Temperature, recorder on Panel 25.
2. Pressure, recorder on Panel 3.

3.4 "DEFEAT" any <u>two</u> channels of Reactor Building Isolation and Cooling, then bypass all three Safety Injection Channels.

<u>CAUTION</u>: If normal power is lost while operating in the injection Mode from the BWST, RB Isolation and Cooling must be manually initiated, when either the BUS 2-1E or 2-2E Undervoltage alarm is received to ensure proper diesel generator load sequencing.

3.5 "THROTTLE" as required to prevent pump runout:

1. Hi press. inj. flow (MU-16A/B/C/D) 0-250 GPM/LEG.

<u>CAUTION</u>: If MU pump flow drops below 95 GPM trip the excess MU pumps.

1181 230

2. Lo Press. Inj. Flow (DH-V128A/B) 3000-3300 GPM/PMP.

3. Building spray flow (BS-V1A/B) 1400-1700 GPM/PMP.

<u>NOTE</u>: Hi flow alarms should actuate as a warning to throttle flows.

<u>CAUTION</u>: The actions to be taken for switching suction from the BWST to the R.B. sump depend upon the number of operating ECCS injection strings and the delivered flowrates in these injection strings. Based upon the existing situation, in the ECCS, proceed as outlined below to perform switch over of suction to the RB sump:

	Situation	Go to Step
1.	Both LPI strings are operating and indicated flow in each is above 750 gpm.	3.6
2.	Both LPI strings are operating but indicated flow in each is below 750 gpm.	3.7
3.	One LPI string is inoperative.	3.8

<u>NOTE</u>: The main objective when switching suction from the BWST to the RB sump is to maintain ECCS flow through two flow paths.

3.6 Both LPI Strings are Operating and Indicated Flow in Each is Above 750 GPM.

3.6.1 When the BWST level reaches approximately 12', initiate the following steps:

3.6.2 If not already done "THROTTLE" LPI strings flow rates back to 3000 GPM each using control valves DH-V128 A &B (flow rate indication and valve control in control room).

3.6.2.1 If not already done, THROTTLE BS pump's flows back to 1600 gpm per pump. This must be done prior to taking suction from the RB sump.

3.6.3 "SHUT OFF" HPI pumps (pump control in control room).

3.6.4 Verify the ECCS suction valves DH-V6A & B from RB sump automatically open at BWST level of 7'.

3.6.5 When suction valves from RB sump are full open, "CLOSE the ECCS suction valves (DH-V5A & B) from the BWST.

3.6.6 "REPOSITION" LPI flow control valves (DH-V128 A&B) to obtain 3000 GPM each string if necessary. (Flow rate could change due to change in suction sources).

3.6.7 Proceed to step 3.9.

3.7 Both LPI Strings Are Operating But Indicated Flow In Each Is Below 750 GPM.

3.7.1 When the BWST level reaches approximately 12', initiate the following steps.

3.7.2 If not already done, "THROTTLE" HPI strings' flow rates back to 500 GPM per pump each using control valves MU-V16A, B, C, and D (flow rate indication and valve control in control room).

3.7.2.1 If not already done, THROTTLE BS pump's flows back to 1600 GPM per pump. This must be done prior to taking suction from the RB sump.

3.7.3 "OPEN" valves DH-V7A and B in crossover line from LPI line to suction of HPI pumps (valve control and position indication in control room). "Reposition" HPI flow control valves (MU-V16A,B,C, & D) to obtain 500 GPM each string. (HPI flow would increase because of increased HPI pump suction pressure).

3.7.4 VERIFY the ECCS suction valves (DH-V6A & B) from RB sump automatically open at BWST level of 7'.

3.7.5 When suction valves from RB sump are fully open, "Close" the ECCS suction valves (DH-V5A & B) from the BWST. The ECCS is now in "piggy-back" operation.

3.7.6 Proceed to step 3.9.

<u>NOTE</u>: Once the flow in each LPI string exceeds 750 gpm, the HPI pumps can be "SHUT OFF" and valves DH-V7A & B can be "CLOSED".

3.8 One LPI String is Inoperative

3.8.1 The BWST 7' automatic transfer to the RB sump is reached in approximately 55-80 minutes from initiation of ECCS injection, depending upon reason for string failure (i.e. - local LPI failure or diesel failure). Prior to actuation of the lo-lo level alarm, initiate the following steps.

3.8.2 Using the controls in the control room, attempt to "START UP" the non-operating LPI String. If successful, proceed to step 3.6. If not successful, proceed to step 3.8.3 below.

3.8.3 If step 2 was unsuccessful, initiate opening the DH cross-connect isolation valves (DH-V193 A & B) as follows:

3.8.3.1 ENSURE SN-V188 is closed then OPEN DH-V112 A & B to fill the inoperable string.

NOTE: If offsite power is lost DHV193 A & B must be manually opened.

3.8.3.2 Obtain the keys for the DH cross-connect isolation valves' (DH-V193A and DH-V193B) breakers from the shift supervisor.

3.8.3.3 Proceed to 480V MCC 2-32B and MCC 2-42B.

3.8.3.4 Remove the locks from the isolation valve breakers for the DH cross-connect line.

NOTE: Local control stations for DH-V193A and are located in the Aux Bldg at Elev. 280'6" near the DH vaults.

3.8.3.5 "OPEN" the DH cross-connect isolation valve (e.g. DH-V-193A(B)) next to the operating LPI String.

3.8.3.6 "OPEN" the second isolation valve (DH-V193A(B)).

3.8.3.7 While opening the second decay heat cross-connect isolation valve, "THROTTLE" either DH-V128A or DH-V128B in the control room as required to achieve essentially equal flow rates in both D.H. injection lines. (Approximately 1500 gpm per LPI string).

3.8.3.8 If flow is established at greater than 750 gpm through each LPI string, then proceed to step 3.8.5.

3.8.3.9 If flow cannot be established through each LPI string in excess of 750 gpm using the cross-connect line before ECCS suction must be switched to the RB sump, then proceed to step 3.9.4.

3.8.4 If opening the DH cross-connect line fails to provide flow in each LPI string in excess of 750 gpm, then place one HPI string in a modified "Piggy-Back" mode with the operating LPI string as follows (assume LPI string "A"(B) is the operating string):

3.8.4.1 If not already done, "THROTTLE" HPI string "A"(B) flow rate back to 500 GPM using control valve MU-V16A and V16B.

3.8.4.2 "THROTTLE" LPI string "A"(B) flow rate to 3000 GPM using control valve DH-V128A(B).

3.8.4.3 "OPEN" valve DH-V7A(B) in crossover line from LPI line to suction of the operating HPI pump.

3.8.4.4 "REPOSITION" HPI flow control valves MU-V16A&B to otbain 250 GPM per leg HPI flow and reposi ion LPI flow control valve DH-V128A to obtain 2500 GPM LPI flow. The LPI pump is pumping design flow of 3000 GPM (2500 GPM LPI plus 500 GPM HPI).

3.8.4.5 "SHUT OFF" HPI pump in HPI string "B"(A).

3.8.4.6 Proceed to step 3.8.6.

3.8.5 "SHUT OFF" HPI pumps (pump contro in control room).

3.8.6 If not already done, THROTTLE BS pump's flows back to 1600 GPM per pump. This must be done prior to taking suction from the RB sump.

3.8.6.1 When the BWST level decreases to 7' VERIFY the ECCS suction valves (DH-V6A & B) from RB sump automatically open.

3.8.7 When suction valves f om RB sump are full open (position indication in control room), "CLOSE" the ECCS suction valves (DH-V5A & B) from the BWST.

3.8.8 "REPOSITION" LPI flow control valve(s) (DH-V128A and/or B) as required to obtain proper string flowrates. (Flow rates could change due to change in suction sources).

3.9 When the Sodium Hydroxide Tank level reaches approximately 3 ft. "CLOSE" DH-V8A & B.

3.10 Actuate Environmental Barrier System by opening EB-V11.

3.11 Within 24 hours of ECCS initiation, establish one of the long-term cooling circulation modes described in 2104-1.3 - Decay Heat Removal System, and listed below:

Mode 1 Forced circulation using decay heat drop line.

Mode 2 Gravity draining reactor coolant hot leg to the Reactor Building sump via the D.H. drop line.

Mode 3 Hot leg injection using pressurizer auxiliary spray line.

Mode 4 Reverse flow through the decay heat drop line into "B" Reactor Coolant Loop Hot leg.

4.0 LONG TERM ACTION

4.1 Verify all previous actions and carry out additional actions as outlined below.

4.2 Evaluate symptoms and determine if possible the cause of the loss of coolant.

4.3 Secure turbine, feed water, and steam systems when time permits.

4.4 Monitor for H_2 buildup and assure actuation of H_2 recombiner per 2104-6.4, Hydrogen Recombiner Operations.

4.5 Monitor R.B. Sump for pH and add Sodium Hydroxide as required thru the decay heat removal system.

4.6 As conditions permit, evaluate unit conditions, and return all non-essential equipment to its normal line up.

NOTE: Refer to the following instructions and procedures for additional information as required.

1. Radiation emergency plan site emergency in the emergency plan.
2. 2104-5.4 - Control Building HVAC.

1181 236

TABLE A-1

ESF EQUIPMENT - ESF POSITION IN THE CONTROL ROOM PANEL 13

The White light for each component should be lit to indicate that the component is in its ESF position, unless otherwise noted.

ACTUATION A

SAFETY INJECTION GP. 1

Equipment	ESF Position
DC-P-1A	ON
G2-12 (Note 1)	CLOSED
MU-P-1A (Note 2)	ON
NR-P-1A (Note 3)	ON
NS-P-1A (Note 4)	ON
T1E-2E2	OPEN
T3E-4E2	OPEN
T11E-21E2	OPEN
DC-V96A	CLOSED

SAFETY INJECTION GP. 2

Equipment	ESF Position
NR-V9A	CLOSED
DH-V4A	OPEN
DH-V5A	OPEN
DH-V8A	OPEN
DH-V100A	CLOSED
DH-V102A	OPEN
MU-P-1B (Note 2)	ON
NR-V40A	OPEN
NS-P-1C	ON
T12-22E-2	OPEN
MU-V28	CLOSED

ACTUATION B

SAFETY INJECTION GP. 1

Equipment	ESF Position
DC-P-1B	ON
G22-12 (Note 1)	CLOSED
MU-P-1B (Note 2)	ON
NR-P-1C (Note 3)	ON
NS-P-1B (Note 4)	ON
T2E-1E2	OPEN
T4E-3E2	OPEN
T21E-11E2	OPEN
DC-V96B	CLOSED

SAFETY INJECTION GP. 2

Equipment	ESF Position
NR-V9B	CLOSED
DH-V4B	OPEN
DH-V5B	OPEN
DH-V8B	OPEN
DH-V100B	CLOSED
DH-V102B	OPEN
MU-P-1C	ON
NR-V40B	OPEN
NS-P-1C (Note 4)	ON
T22E-12E-2	OPEN
HY-V55	CLOSED

1181 237

TABLE A-1

ACTUATION A

SAFETY INJECTION GP. 3 Equipment	ESF Position
MU-V36	CLOSED
MU-V16A	OPEN
MU-V16B	OPEN
DH-P-1A	ON
NR-P-1B (Note 3)	ON
T31E-41E-2	OPEN
NS-V84B	CLOSED
NR-V42A	OPEN

ACTUATION B

SAFETY INJECTION GP. 3 Equipment	ESF Position
MU-V37	CLOSED
NS-V[illegible]2	CLOSED
NS-V[illegible]7	CLOSED
DH-P-1B	ON
MU-V16C	OPEN
MU-V16D	OPEN
NR-P-1D (Note 3)	ON
T41E-31E-2	OPEN
NS-V84A	CLOSED
NR-V42B	OPEN

PANEL 8

Equipment	ESF Position	Status Light Indicator
NS-V83A	OPEN	R
NS-V83B	OPEN	R
NS-V215	CLOSED	G
NS-V216	CLOSED	G

NOTE 1: Diesel Generator Breaker will only be closed if Normal Power is lost; otherwise status indication will be Open (white).

NOTE 2: MU-P-1B will be running if normal power is available for the Actuation, for the pump that it is selected to backup.
If normal power is lost, MU-P-1B will be running, if the pump that it is selected to backup fails to start or is inoperable.

TABLE A-1

NOTE 3: The NR pump in each header selected for ES or standby will start if a pump is not operating in that header; otherwise the operating pump will remain in service.

NOTE 4: Normally NS-P-1A and 1B will start; however, NS-P-1C will start if either NS-P-1A or B (depending upon which pump it is selected to backup) fails to start or is inoperable.

TABLE B-1

ESF EQUIPMENT - ESF POSITION IN THE CONTROL ROOM
PANEL 13

The White light for each component should be lit to indicate that the component is in its ESF position, unless otherwise noted.

ACTUATION A

SAFETY INJECTION GP. 1

Equipment	ESF Position
DC-P-1A	ON
G2-12 (Note 1)	CLOSED
MU-P-1A (Note 2)	ON
NR-P-1A (Note 3)	ON
NS-P-1A (Note 4)	ON
T1E-2E2	OPEN
T3E-4E2	OPEN
T11E-21E2	OPEN
DC-V96A	CLOSED

SAFETY INJECTION GP. 2

Equipment	ESF Position
NR-V9A	CLOSED
DH-V4A	OPEN
DH-V5A	OPEN
DH-V8A	OPEN
DH-V100A	CLOSED
DH-V102A	OPEN
MU-P-1B (Note 2)	ON
NR-V40A	OPEN
NS-P-1C	ON
T12-22E-2	OPEN
MU-V28	CLOSED

ACTUATION B

SAFETY INJECTION GP. 1

Equipment	ESF Position
DC-P-1B	ON
G22-12 (Note 1)	CLOSED
MU-P-1B (Note 2)	ON
NR-P-1C (Note 3)	ON
NS-P-1B (Note 4)	ON
T2E-1E2	OPEN
T4E-3E2	OPEN
T21E-11E2	OPEN
DC-V96B	CLOSED

SAFETY INJECTION GP. 2

Equipment	ESF Position
NR-V9B	CLOSED
DH-V4B	OPEN
DH-V5B	OPEN
DH-V8B	OPEN
DH-V100B	CLOSED
DH-V102B	OPEN
MU-P-1C	ON
NR-V40B	OPEN
NS-P-1C (Note 4)	ON
T22E-12E-2	OPEN
HY-V55	CLOSED

1181 240

NOTE 1: Diesel Generator Breaker will only be closed if Normal Power is lost; otherwise status indication will be Open (white).

NOTE 2: MU-P-1B will be running if normal power is available for the Actuation, for the pump that it is selected to backup.

If normal power is lost, MU-P-1B will be running, if the pump that it is selected to backup fails to start or is inoperable.

NOTE 3: The NR pump in each header selected for ES or standby will start if a pump is not operating in that header; otherwise the operating pump will remain in service.

NOTE 4: Normally NS-P-1A and 1B will start; however, NS-P-1C will start if either NS-P-1A or B (depending upon which pump it is selected to backup) fails to start or is inoperable.

PANEL 8

Equipment	ES Position	Indication
DH-V7A(1)	Close	G
DH-V7B(1)	Close	G
NS-V83A	Open	R
NS-V83B	Open	R
NS-V215	Close	G
NS-V216	Close	G
CF-V1A	Open	R
CF-V1B	Open	R

(1) This valve may have to be opened for "piggy-back" operation. Once, opened, the Position/Indication becomes Open/R.

TABLE B-1

ACTUATION A

R.B. ISOLATION & COOLING GP. 3

Equipment	Position
IC-V2	CLOSED
IC-V5	CLOSED
MU-V2A	CLOSED
MU-V2B	CLOSED
MU-V377	CLOSED
NS-V72	CLOSED
NS-V81	CLOSED
NR-P-2A	ON
NR-V144A	OPEN
RR-P-1A	ON
RR-V25A	CLOSED
RR-V25B	CLOSED
IC-P-1A	OFF
AH-C-8A	ON
AH-E-11B	ON
AH-E-11C	ON
AH-P-1A	ON
AH-V1A	CLOSED
AH-V1B	CLOSED
AH-V4A	CLOSED
AH-V4B	CLOSED
AH-V5	CLOSED
AH-V60	CLOSED
AH-V102	CLOSED
AH-V72	CLOSED
* BS-P-1A	ON

* If RB Pressure >30 psig.

ACTUATION B

R.B. ISOLATION & COOLING GP. 3

Equipment	Position
IC-V3	CLOSED
IC-V4	CLOSED
MU-V376	CLOSED
MU-V18	CLOSED
MU-V25	CLOSED
NS-V100	CLOSED
NR-P-2B	ON
RR-P-1C	ON
RR-V25D	CLOSED
RR-V25E	CLOSED
AH-C-8A	ON
AH-E-11E	ON
AH-P-1B	ON
AH-V2A	CLOSED
AH-V2B	CLOSED
AH-V3A	CLOSED
AH-V3B	CLOSED
AH-V6	CLOSED
AH-V61	CLOSED
AH-V61	CLOSED
AH-V63	CLOSED
AH-V71	CLOSED
IC-P-1B	OFF
* BS-P-1B	ON

TABLE B-1

ACTUATION A

R.B. ISOLATION & COOLING GP. 1

Equipment	Position
RR-V5C	OPEN
SV-V55	CLOSED
WDL-V1095	CLOSED
DC-V114	CLOSED

R.B. ISOLATION & COOLING GP. 2

Equipment	Position
AH-E-4A	ON
AH-E-11A	ON
RR-V25C	CLOSED
BS-V1A	OPEN
CA-V10	CLOSED
CA-V4A	CLOSED
CA-V9	CLOSED
RR-P-1B	ON
WDG-V199	CLOSED
WDL-V22	CLOSED
WDL-V1126	CLOSED
AH-D4092A & B	RECIRC
AH-D4092D & E	RECIRC
ED-4098	

ACTUATION B

R.B. ISOLATION & COOLING GP. 1

Equipment	Position
RR-V6C	OPEN
RR-V6D	OPEN
RR-V6E	OPEN
SV-V54	CLOSED
WDL-V1092	CLOSED
DC-V103	CLOSED
DC-V115	CLOSED

R.B. ISOLATION & COOLING GP. 2

Equipment	Position
AH-E-4B	ON
AH-E-11C	ON
RR-V25C	CLOSED
BS-V1B	OPEN
CA-V1	CLOSED
CA-V3	CLOSED
CA-V4B	CLOSED
CA-V8	CLOSED
CA-V6	CLOSED
RR-P-1D	ON
WDG-V2	CLOSED
WDL-V1125	CLOSED
WDL-V271	CLOSED
AH-D4092A & B	RECIRC
AH-D4092D & E	RECIRC
ED-4098	RECIRC

1181 243

TABLE B-1

ACTUATION A		ACTUATION B	
SAFETY INJECTION GP. 3		SAFETY INJECTION GP. 3	
Equipment	ESF Position	Equipment	ESF Position
MU-V36	CLOSED	MU-V37	CLOSED
MU-V16A	OPEN	NS-V32	CLOSED
MU-V16B	OPEN	NS-V67	CLOSED
DH-P-1A	ON	DH-P-1B	ON
NR-P-1B (Note 3)	ON	MU-V16C	OPEN
T31E-41E-2	OPEN	MU-V16D	OPEN
NS-V84B	CLOSED	NR-P-1C (Note 3)	ON
NR-V42A	OPEN	T41E-31E-2	OPEN
		NS-V84A	CLOSED
		NR-V42B	OPEN

ACTUATION A		ACTUATION B	
R.B. ISOLATION AND COOLING GP. 1		R.B. ISOLATION AND COOLING GP 1	
Equipment	Position	Equipment	Position
AH-V81	CLOSED	AH-E-11D	ON
AH-V101	CLOSED	AH-V80	CLOSED
AH-V102	CLOSED	AH-V103	CLOSED
AH-V105	CLOSED	AH-V104	CLOSED
AH-V107	CLOSED	AH-V106	CLOSED
CF-V144	CLOSED	AH-V108	CLOSED
DH-V3	CLOSED	CF-V115	CLOSED
NM-V52	CLOSED	DH-V2	CLOSED
NR-V51A	CLOSED	NM-V104	CLOSED
RR-V2A	OPEN	NR-V51B	CLOSED
RR V2B			
RR-V5A	OPEN	RR-V2C	OPEN
RR-V5B	OPEN	RR-V2D	OPEN

1181 244

PANEL 15

Equipment	ES Position	Indication
DH-V6A(1)	Close	G
DH-V6B(1)	Close	G
MU-V378(2)	Open	R
MS-V4A(3)	Open	R
MS-V4B(3)	Open	R
MS-7A(3)	Open	R
MS-V7B(3)	Open	R

(1) This valve must be opened for sump-switchover. Once opened, the Position/Indication becomes OPEN/R.

(2) This valve should be closed at the operator's first chance. Once closed, the Position/Indication becomes CLOSE/W.

(3) These valves should be closed when the steam system is secured. Once closed, the Position/Indication becomes CLOSE/G.

PANEL 25

Equipment	ES Position	Indication
AH-E12A	Off	G
AH-E12B	Off	G
AH-E19A	Off	G
AH-E19B	Off	G

Checked Control Copy ____________________

EP/0/A/1800/4

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

LOSS OF REACTOR COOLANT

Considers the following cases:

Case A1: Excessive RC System Leakage - No Reactor Trip.

Case A2: Small Break -- Feedwater-RC Pumps-No Reactor Trip.

Case A3: Small Break -- Feedwater-RC Pumps-Reactor Trip.

Case A4: Small Break -- No Feedwater-RC Pumps-Reactor Trip.

Case A5: Small Break -- Feedwater-No RC Pumps-Reactor Trip.

Case A6: Small Break -- Feedwater-No RC Pumps-RC Pressure Stabilizes at ~ Secondary Side Pressure.

Case A7: Small Break -- No Feedwater-No RC Pumps-Reactor Trip.

Case B: Rupture in Excess of Capability of Available High Pressure Injection pumps.

C-2

INFORMATION ONLY

Case A4: Small Break--No Feedwater-RC Pumps-Reactor Trip

1.0 Symptoms

1.1 Excessive RCS makeup

1.2 Decreasing RCS pressure

1.3 Reactor trip

1.4 Decreasing Pressurizer level initially. May increase later.

1.5 RIA alarms

1.6 LDST level low or decreasing more than normal

1.7 ES actuation 1-2

1.8 Increasing Reactor Building Temperature and Pressure and Rx. Bldg sump level

1.9 No feedwater flow and no S/G level

2.0 Immediate Action

2.1 Automatic

2.1.1 Reactor trip

2.1.2 Turbine trip

2.1.3 Possible ES actuation 1-2

2.2 Manual

NOTE: Any asterisk (*) parameters in the below sections shall be verified when step 3.1 of subsequent action is performed.

2.2.1 If ES Channels 1 & 2 have actuated because of a low pressure(*) condition in the RC system, IMMEDIATELY TRIP all RC pumps and ref Case A7 Section 2.2.

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

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 2).

2.2.3 Initiate ES 1-2 if it has not been actuated on ECCS signal.

CAUTION: If RC system pressure decreases below 1600 psi(*) IMMEDIATELY TRIP all RC pumps and refer to Case A7 Section 2.2.

2.2.4 Check immediately for flow indication on both HPI emergency injection lines. If no flow is indicated in "B" loop, dispatch operator to open (2)HP-116 <u>within 10 minutes of ES actuation</u>.

2.2.4.1 For Unit 3, if no flow is indicated in "B" loop, open 3HP-409 <u>within 10 minutes of ES actuation</u>.

2.2.5 If no flow is indicated in "A" loop, dispatch operator to open (2)HP-26 <u>within 10 minutes of ES actuation</u>.

2.2.5.1 For Unit 3, if no flow is indicated in "A" loop, open 3HP-410 <u>within 10 minutes of ES actuation</u>.

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

2.2.5.2 Both LPI pumps are in operation and flowing at a rate in excess of 1000 gpm on header flow gauge (*)

each and the situation has been stable for 20 minutes,

OR

2.2.5.3 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 the 50 degrees subcooling by T_h indication (*) cannot be maintained after HPI cut-off, the HPI shall be reactivated (refer to Enclosure 1). The degree of subcooling beyond 50 degrees F and length of time HPI is in operation shall be limited by the pressure/temperature consideration for the vessel integrity (refer to Enclosure 2).

NOTE: If the HPI System has been activated and RC pumps operating, at least one RCP per loop shall remain operating.

2.2.6 Maintain maximum HPI flow. (*)

2.2.7 If pressure is increasing open RC-4 (Power Operated Relief Block) and RC-66 (Power Operated Relief Valve) to maintain forced cooling with the HPI system.

Note: If RC-66 is not operable, Pressurizer Code Reliefs will relieve overpressure and maintain force flow.

2.2.8 Monitor RCS T_{hot} (*) (if on scale) or incore thermocouples (*) (Display group #29) for indication of core outlet temperature stabilization. (T_{sat} for 2500 psig = 665°F).

2.2.9 Regain feedwater as soon as possible.

3.0 Subsequent Actions

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

3.2 Once feedwater is available, commence feeding the OTSGs through the auxiliary feed nozzles and control level at ~ 25 inches on the startup range (*) and control OTSGs secondary side pressure on OTSG pressure gauge (*) at ~ 1000 psig using Turbine Bypass valves; if unavailable, the main steam relief valves.

3.3 Close the PORV, RC-66.

3.4 Regain RCS pressure control by energizing the pressurizer heaters and heating the pressurizer until the pressurizer temperature (*) indicates within the pressure temperature curve for saturation.

CAUTION: If pressurizer heaters are inoperable, control RCS pressure by throttling HP injection flow with (3)(2) HP-26 and (3)(2) HP-27.

3.4.1 If 3HP-26 and/or 3HP-27 fail, control RCS pressure by throttling 3HP-410 and/or 3HP-409 respectively.

3.5 Monitor RCS pressure carefully to ensure that the steam bubble is formed in the pressurizer.

3.6 Place pressurizer heaters in automatic.

NOTE: RCS must be maintained subcooled by T_h indication (*). (See Enclosure 1)

3.7 Borate the RC System for cold shutdown conditions per OP/1103/15, Reactivity Balance Calculation.

3.8 Go to one (1) RC pump per loop operation. One (1) pump should be the pump that supplies pressurizer spray.

3.9 De-energize pressurizer heaters and maintain OTSG cooling by adjusting steam pressure using the Turbine Bypass valves or the manual steam dumps. Cooldown at 100°F/hr. to achieve an RC pressure of 320 psig.

NOTE: Bypass ES low pressure injection and block core flood actuation at RC pressure of 700 psig.

NOTE: Plot RC pressure/RC temperature at 1/2 hour intervals on Enclosure 1 (Subcooled Curve).

3.10 Maintain RCS pressure at 320 psig and reduce RCS temperature to 240°F.

3.11 Stop one (1) RC pump.

3.12 Sample RC System for isotopic analysis and notify Superintendent of Operations of results prior to placing LPI in service.

3.13 Close (3)(2)LP-21 and (3)(2)LP-22.

3.14 Align and start "A" or "C" LPI pump in the decay heat removal mode (switchover for Unit 1&2) per OPs/1,2,3/A/1104/04. Establish > 1000 gpm flow in the "A" header.

3.15 Stop the remaining RC pump.

INFORMATION ONLY C-7

3.16 Reduce RCS pressure to 100 psig by throttling HPI flow with (3)(2) HP-26 and (3)(2) HP-27. Maintain 50°F subcooling by throttling (3)(2) LPSW-251 and (3)(2) LPSW-252.

3.16.1 If 3HP-26 and/or 3HP-27 fail, throttle HPI flow with 3HP-410 and/or 3HP-409 respectively.

3.17 Place LPI in normal decay heat removal mode per OP/1&2/A/1104/04 (Unit 1&2 only).

3.18 Open (3)(2) LP-22, close (3)(2) LP-21, start "B" LPI pump, open (3)(2) LP-18 and establish > 1000 gpm in "B" LPI train.

3.19 Secure HPI pump.

3.20 Shift LPI Pump "B" suction from the BWST to the Reactor Building Sump by opening (3)(2)LP-20 when sufficient NPSH is available.

NOTE: This is desirable to avoid unnecessary quantities of water in containment.

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

3.21 Reduce RCS temperature per OPs/1,2,3/A/ .04/04 using the decay heat removal coolers for long-term core [illegible].