MISCELLANEOUS CORRESPONDENCE

FOR THE PRAIRIE ISLAND INITIAL EXAMINATION

THE WEEKS OF SEPTEMBER 10 AND 17, 2001

cc: D. Hills

NSP

Northern States Power Company

Prairie Island Training Center
1660 Wakonade Drive West
Welch, MN 55089-9638

September 20, 1999

US Nuclear Regulatory Commission
Document Control
Washington, D.C. 60532

Dear Sir,

In response to the NRC letter of August 20, 1999, "Preparation and Scheduling of Operator Licensing Examinations", please see the attached Operator Licensing Examination Data for our plant.

If you should have any questions, please contact Jim Lash at (651) 388-1165 Ext. 4053.

Sincerely,

Dennis Westphal
Operations Training Superintendent

cc: David E. Hills, Chief
Operations Branch
US NRC Region III

SEP 27 1999

AL 99-03

ORM 536 (…99) U.S NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO 3150-0131 EXPIRES 07/31/2002

Estimated burden per response to comply with this voluntary information collection request: 1 hour. This information collection is used to plan budgets and resources for operator examinations. Send comments regarding burden estimate to the Records Management Branch (T-6 E6) U S Nuclear Regulatory Commission, Washington, DC 20555-0001 or by internet email to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0131), Office of Management and Budget Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

OPERATOR LICENSING EXAMINATION DATA

FACILITY Prairie Island Nuclear Generating Plant

NRC REGION III

A. PROPOSED EXAMINATION PREPARATION SCHEDULE

PROPOSED NUMBER	CY 00	CY 01	CY 02	CY 03
ESTIMATED NUMBER OF LICENSEE-PREPARED EXAMINATIONS	1	1	1	1

B. INITIAL OPERATOR LICENSING EXAMINATIONS

PROPOSED NUMBER	CY 00	CY 01	CY 02	CY 03
NUMBER OF REACTOR OPERATORS		6	6	6
NUMBER OF SENIOR REACTOR OPERATORS-INSTANT		1	1	1
NUMBER OF SENIOR REACTOR OPERATORS-UPGRADE	4	2	2	2
NUMBER OF SENIOR REACTOR OPERATORS-LIMITED				
PROPOSED DATES				
PRIMARY DATE	May 15	Sep 17	Mid September	Mid September
ALTERNATE DATE		OCT	OCT	OCT

C. PROPOSED GENERIC FUNDAMENTALS EXAMINATION (GFE) SCHEDULE

PROPOSED NUMBER	CY 00		CY 01	
	FIRST (Apr)	SECOND (Jul / Oct)	FIRST	SECOND
ESTIMATED NUMBER OF CANDIDATES		— / 9		7

 PRINTED ON RECYCLED PAPER THIS FORM WAS DESIGNED USING INFORMS

March 13, 2000

Mr. M. Wadley
President, Nuclear Generation
Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

Dear Mr. Wadley:

In response to D. Westphal's letter dated September 20, 1999, and a telephone conversation with John Kempke on February 24, 2000, we have tentatively scheduled an initial licensing examination for your operator license applicants at the Prairie Island Nuclear Generating Plant during the week of September 17, 2001. Validation of the examination will occur at the station during the week of August 27, 2001. In the unlikely event that we are unable to support the examination during the scheduled week, we will inform you immediately upon discovery of such conditions and make arrangements to administer the examination at a mutually acceptable date.

As stated in the enclosure to Mr. Westphal's letter dated September 20, 1999, your staff will develop the examination.

Please inform us at your earliest opportunity if you discover you are unable to support the examination on the scheduled dates.

A supplementary letter will be sent to the training department approximately 120 days prior to the examination outlining examination security expectations, listing the materials required by the NRC to conduct the examination, reconfirming the examination dates, and reconfirming the number of candidates you have in the training program. If you have any questions concerning this information, please contact Mary Ann Bies of my staff at 630-829-9711.

Sincerely,

David E. Hills

David E. Hills, Chief
Operations Branch

Docket Nos. 50-282; 50-306
License Nos. DPR-42; DPR-60

cc: Site General Manager, Prairie Island
Plant Manager, Prairie Island
S. Minn, Commissioner, Minnesota
Department of Public Service
State Liaison Officer, State of Wisconsin
Tribal Council, Prairie Island Dakota Community
J. N. Jensen, Training Department

DOCUMENT NAME: G:DRS\PRA03130.WPD
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	RIII	C	RIII	N	RIII	N		
NAME	MABies:jp mab		AMStone AMS		DHills DEH			
DATE	03/9/00		03/9/00		03/13/00			

OFFICIAL RECORD COPY

<u>Distribution</u>:
J. Caldwell, RIII
SRI Prairie Island
DRP
DRS
RIII PRR
PUBLIC IE-42
Docket File
GREENS
D. C. Trimble, NRR:DIPM:IOLB

May 10, 2001

Mr. J. Sorensen
Site Vice-President
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

Dear Sorensen:

In a telephone conversation on May 4, 2001, between Mr. Jay Hopkins, Senior Examiner and Mr. Joseph Loesch, Principal Technical Instructor, arrangements were made for the administration of licensing examinations at the Prairie Island Nuclear Generating Plant the week of September 10, 2001. In addition, the NRC will make an examination validation visit to your facility the week of August 27, 2001.

As agreed during the telephone conversation, your staff will prepare the examinations based on the guidelines in Revision 8, Supplement 1, of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," dated April 2001. The NRC regional office will discuss with your staff any changes that might be necessary before the examinations are administered.

To meet the above schedule, it will be necessary for your staff to furnish the examination outlines no later than June 27, 2001. The written examinations, operating tests, and the supporting reference materials identified in Attachment 2 of ES-201 will be due by July 27, 2001. Pursuant to 10 CFR 55.40(b)(3), an authorized representative of the facility licensee shall approve the outlines, examinations, and tests before they are submitted to the NRC for review and approval. All materials shall be complete and ready to use. Any delay in receiving the required examination and reference materials, or the submittal of inadequate or incomplete materials, may cause the examinations to be rescheduled.

In order to conduct the requested written examinations and operating tests, it will be necessary for your staff to provide adequate space and accommodations in accordance with ES-402, and to make the simulation facility available on the dates noted above. In accordance with ES-302, your staff should retain the original simulator performance data (e.g., system pressures, temperatures, and levels) generated during the dynamic operating tests until the examination results are final.

Appendix E of NUREG-1021 contains a number of NRC policies and guidelines that will be in effect while the written examinations and operating tests are being administered.

To permit timely NRC review and evaluation, your staff should submit preliminary reactor operator and senior reactor operator license applications (Office of Management and Budget (OMB) approval number 3150-0090), medical certifications (OMB approval number 3150-0024), and waiver requests (if any) (OMB approval number 3150-0090) at least 30 days before the first examination date. If the applications are not received at least 30 days before the examination date, a postponement may be necessary. Signed applications certifying that all training has been completed should be submitted at least 14 days before the first examination date.

This letter contains information collections that are subject to the *Paperwork Reduction Act of 1995* (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget, approval number 3150-0018, which expires on April 30, 2003.

The public reporting burden for this collection is estimated to average 500 hours per response, including the time for reviewing instructions, gathering and maintaining the data needed, writing the examinations, and completing and reviewing the collection of information. Send comments on any aspect of this collection of information, including suggestions for reducing the burden, to the Information and Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001, or by Internet electronic mail at BJS1@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0018), Office of Management and Budget, Washington, D.C. 20503.

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

Thank you for your cooperation in this matter. Mr. Joseph Loesch has been advised of the policies and guidelines referenced in this letter. If you have any questions regarding the NRC's examination procedures and guidelines, please contact Mr. Jay Hopkins at 630-829-9739, or me at 630-829-9733.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

David E. Hills

David E. Hills, Chief
Operations Branch
Division of Reactor Safety

Docket Nos. 50-282; 50-306
License Nos. DPR-42; DPR-60

See Attached Distribution

DOCUMENT NAME: G:DRS\ PRA05j001drs ol.wpd
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	RIII	E	RIII	N				
NAME	JHopkins:sd	JH	DHills	DH				
DATE	05/7/01		05/16/01					

OFFICIAL RECORD COPY

cc: Plant Manager, Prairie Island
M. Wadley, Chief Nuclear Officer
Site Licensing Manager
Nuclear Asset Manager
J. Malcolm, Commissioner, Minnesota
Department of Health
State Liaison Officer, State of Wisconsin
Tribal Council, Prairie Island Indian Community
J. Silberg, Esquire
Shawn, Pittman, Potts, and Trowbridge
A. Neblett, Assistant Attorney General
Office of the Attorney General
S. Bloom, Administrator
Goodhue County Courthouse
Commissioner, Minnesota Department
of Commerce
J. Jensen, Training Department

<u>ADAMS Distribution</u>:
CMC1
DFT
DCT
TJK3
GEG
HBC
SPR
DRPIII
DRSIII
PLB1
JRK1
M. Bies (Concurrence Copy)

NMC
Committed to Nuclear Excellence

Nuclear Management Company, LLC
Prairie Island Nuclear Generating Plant
1717 Wakonade Dr. East • Welch MN 55089

June 22, 2001

10CFR55.5

Regional Administrator
U S Nuclear Regulatory Commission
Region III
801 Warrenville Road
Lisle, IL 60532-4351

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Examination Material for Prairie Island Initial Operator License Examination
Week of September 10, 2001

Enclosed are the integrated examination outlines for the initial operator license examinations to be administered at our facility the week of September 10, 2001. This information is being provided in accordance with the guidelines ES-201 of NUREG 1021, "Operating License Examination Standard for Power Reactors", Revision 8, Supplement 1.

NUREG 1021 physical security requirements state that the enclosed examination materials shall be withheld from public disclosure until after the examination is complete.

In this letter we have made no new Nuclear Regulatory Commission commitments.

Please contact Joe Loesch (651-388-1165, Ext. 4084) or Doug Smith (651-388-1165, Ext. 2579) if you have any questions related to this letter.

Joel P. Sorensen

Joel P. Sorensen
Site Vice President
Prairie Island Nuclear Generating Plant

c: Jay Hopkins w/attachments

Attachments (to Jay Hopkins only):

1. ES-201-2 Exam Outline Quality Checklist
2. Notes for satisfaction of ES 201-2 criteria
3. ES 301-1 SRO / RO Administrative Topics Outlines (two pages)
4. ES 301-2 SRO-U / SRO-I / RO Walkthrough Outlines (three pages)
5. ES 301-4 Simulator Scenario Quality Checklist
6. ES-D-1 Scenario Outlines (four pages)
7. ES 301-5 Transient and Event Checklist
8. Explanation of Random Generation Technique and resolution of K/A suppression issue.
9. ES-401-3 PWR SRO Original Examination Outline
10. ES-401-4 PWR RO Original Examination Outline
11. ES-401-5 SRO / RO Original Generic Knowledge and Abilities Outlines (two pages)
12. ES-401-10 Record of Rejected K/As (two pages)
13. Outlines used for replacement K/As (labeled "replacement" and "second replacement")
14. Skyscraper totals after replacement K/As (two pages)

NMC
Committed to Nuclear Excellence

Nuclear Management Company, LLC
Prairie Island Nuclear Generating Plant
1717 Wakonade Dr. East • Welch MN 55089

July 20, 2001

10CFR55.5

Regional Administrator
U S Nuclear Regulatory Commission
Region III
801 Warrenville Road
Lisle, IL 60532-4351

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Examination Materials for Prairie Island Initial Operator License Examination Week of September 10, 2001

Enclosed are the validated examination materials for the initial operator license examinations to be administered at our facility the week of September 10, 2001. This information is being provided in accordance with the guidelines ES-201 of NUREG 1021, "Operating License Examination Standard for Power Reactors," Revision 8, Supplement 1.

NUREG 1021 physical security requirements state that the enclosed examination materials shall be withheld from public disclosure until after the examination is complete.

In this letter we have made no new Nuclear Regulatory Commission commitments.

Please contact Joe Loesch (651-388-1165, Ext. 4084) or Doug Smith (651-388-1165, Ext. 2579) if you have any questions related to this letter.

Joel P. Sorensen

Joel P. Sorensen
Site Vice President
Prairie Island Nuclear Generating Plant

c: Jay Hopkins w/enclosures

Enclosures (to Jay Hopkins only):

Enclosure 1 – Prairie Island 2001 Simulator Scenarios
- Form ES-301-3 Operating Test Quality Checklist
- Form ES-301-4 Simulator Scenario Quality Checklist
- Form ES-301-5 Transient and Event Checklist
- Form ES-301-6 Competencies Checklist
- Form ES-D-1 Scenario Outlines (4 pages, 1 with each scenario)
- 2001 NRC Exam Scenarios (total of 4 scenario guides)
- Supporting Information (with each scenario)

Enclosure 2 - Prairie Island 2001 JPMs
- Form ES-301-1 Administrative Topics Outline (RO)
- RO Administrative JPMs (total of 5 JPM guides)
- Form ES-301-1 Administrative Topics Outline (SRO)
- SRO Administrative JPMs (total of 5 JPM guides)
- Form ES-301-2 Control Room Systems and Facility Walk-Through Test Outlines (3 pages)
- Control Room Systems and Facility Walk-Through JPMs (total of 10 JPM guides)

Enclosure 3 - Prairie Island 2001 NRC Exam RO/SRO Written Examination with Supporting Information
- Form ES-401-7 Written Examination Quality Checklist (2 pages)
- Form ES-401-10 Record of Rejected K/As (2 pages)
- NRC License and Audit Exam Development Best Practice Systematic Cognitive Level Ranking (2 pages)
- Combined RO/SRO Written Examination with Supporting Information (127 questions)

Enclosure 4 - Prairie Island 2001 NRC Exam RO Written
- Forms ES-401-4; ES-401-5 PWR RO Examination Outline
- Form ES-401-8 Site-Specific Written Examination Cover Sheet
- RO Written Examination Without Supporting Information (100 questions)

Enclosure 5 - Prairie Island 2001 NRC Exam SRO Written
- Forms ES-401-3; ES-401-5 PWR SRO Examination Outline
- Form ES-401-8 Site-Specific Written Examination Cover Sheet
- SRO Written Examination Without Supporting Information (100 questions)

Enclosure 6 - Prairie Island Miscellaneous Plant Drawings

Enclosure 7 - Prairie Island Unit 1 Emergency Procedures (volume 1)

Enclosure 8 - Prairie Island Unit 1 Emergency Procedures (volume 2)

Enclosure 9 - Prairie Island Technical Specifications

NMC
Committed to Nuclear Excellence

Nuclear Management Company, LLC
Prairie Island Nuclear Generating Plant
1717 Wakonade Dr. East • Welch MN 55089

September 25, 2001

10 CFR 55

David E. Hills
Chief-Operations Branch
U.S. Nuclear Regulatory Commission
Region III
801 Warrenville Road
Lisle, IL 60532-4351

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Dear Mr. Hills:

Prairie Island has completed its review of the license written examination conducted on September 21, 2001 and has the enclosed comments. We are recommending changes to three questions.

This letter meets the requirements of NUREG-1021, ES-402 parts E-4 through E-6. In this letter we have made no NRC commitments.

If you have any questions, please contact John Kempkes at (651) 388-1165 x5031.

Sincerely,

for

Mano Nazar
Site Vice President
Prairie Island Nuclear Generating Plant

cc: Regional Administrator, Region III
Dave Pelton, Region III
(Lead Examiner)

Encl: Facility Comments
Reference Documentation (1-14)

OCT 0 2 2001

Facility Comments

Record #18 (RO#91, SRO #24)

Question:
The following conditions exist on Unit 1:

-The unit is in HOT SHUTDOWN during a normal cooldown
-RCS temperature is 520 degF
-Pressurizer pressure is 1700 psig
-At this point, all Unit 1 4.16 KV buses lose power (Loss of All AC Power)

How would emergency procedure 1ECA-0.0 "Loss of Safeguards AC Power" be used in this situation?

a. ENTER 1ECA-0.0 immediately upon verification of loss of power to buses 15 and 16.
b. ENTER 1ECA-0.0 ONLY if RCS temperature rises above 540 degF.
c. ENTER 1ECA-0.0 ONLY if a safety injection signal occurs also.
d. ENTER 1ECA-0.0 ONLY if power is NOT restored to EITHER bus 15 or bus 16 when RCS temperature reaches 350 degF.

The correct answer listed in the exam is (c).

Facility Comment: The question puts the applicant in a situation where EOP's do not initially apply. The reactor is already tripped and Safety Injection signals (except containment pressure and MANUAL) are blocked, so initially the entry conditions to E-0 (Ref. 1) are not met. ECA-0.0 is not a direct entry procedure until after the completion of E-0 step 3. ECA-0.0 is the one plant procedure that is designed to respond to the situation described.

With no operator action, several things occur that will lead to plant degradation and a reduction in safety:

- All makeup flow to the RCS is stopped.
- Reactor coolant pump seals lose cooling and will eventually cause a small LOCA to containment. This will lead to a reduction in subcooling and pressurizer level.
- Station batteries begin to support instrument and vital loads.

The listed correct answer (c) is not a conservative response due to the long period of time until these conditions are met. Waiting for an SI would degrade plant conditions significantly. SWI-O-0,Conduct of Operations, FP-OP-COO-01(Ref. 2), states the expectation that operators will place the plant in a safe condition prior to reaching the automatic setpoint.

Generic WOG guidance concerning EOP applicability states (pg 20) (Ref. 3) that EOPs are written for transients initiated at a "hot" or "at power" condition, and would result in either Reactor Protection or Engineered Safeguards System actuation. As a result, although the EOP is applicable in modes 1-4 (pg 22)(Ref. 4), some instructions within the EOP may not be applicable. In this case, the reactor is already shutdown and the turbine is already tripped.

The most correct action would thus be to implement 1ECA-0.0 immediately, with the understanding that E-0 steps 1 and 2 were completed during the shutdown procedure and that some steps in ECA-0.0 may not apply. As a result, Prairie Island requests that the correct answer be changed to (a), which paraphrases the guidance of E-0 step 3.

Further support that (c) cannot be the correct answer can be found by looking at the expected alarms that would occur. For instance, the loss of all charging pumps will cause C47015-0207, "12 RCP LABYRINTH SEAL LO dP" (Ref. 5) to alarm. The alarm response guide will direct the operator to C12.1 AOP1, LOSS OF RCP SEAL INJECTION (Ref. 6). Since Component Cooling has also been lost, step 2.4.1 will direct the operators to 1C3 AOP2, LOSS OF RCP SEAL COOLING (Ref. 7). No equipment is available to correct the condition due to the loss of power, so as soon as bearing temperatures reach 200 degF (step

Facility Comments

2.4.2) the operators are directed to trip the reactor and "Initiate 1E-0, Reactor Trip and Safety Injection." This would be met within the first few minutes of the event and would occur well before a Safety Injection was needed. Other paths (i.e., pressurizer level deviation C47012-0507 (Ref. 8) to 1C4AOP1, Reactor Coolant Leak (Ref. 9), to 1E-0) also would provide entry to the E-procedures. If the above change (accepting (a) as the correct answer) is not accepted, we recommend the question be deleted due to no correct answer existing.

Facility Comments

Record #28 (RO#46, SRO #48)

Question:
Given the following conditions on Unit 1:

- A plant heatup is in progress following a refueling outage.
- RCS temperature is 230 degF
- Both RCPs are running
- 11 RHR pump is in service with RCS heatup being controlled using 11 RHR heat exchanger
- 11 CC surge tank level is +8 inches and rising
- 1R-39, CC SYSTEM LIQUID MONITOR, is indicating normally

Which of the following actions should be taken to address this condition?

a. Verify MV-32088, 11 CC SURGE TANK VENT, is CLOSED.
b. Close CV-31245 AND CV-31246, RCP THERM BARRIER CLNT OUTL.
c. Verify both CC pumps are operating AND **initiate** CC flow through 12 RHR heat exchanger.
d. Open 1HC-624, 11 RHR HX RC OUTLET FCV, to limit heatup rate.

The correct answer listed in the exam is (c).

Facility Comment:
The conditions of this question put the applicant in a situation which is not allowed by plant procedures. The normal shutdown cooling lineup has the manual crosstie valves to both RHR HX's open and CC flow to both HX's in service. Control of heatup rate is done by operation of both RHR HX FCV's. The valves are initially aligned to OPEN during preparation for Phase II Cooldown in 1C14 (Ref. 10) and not returned to AUTO until the RHR system is realigned for ECCS operation during startup per 1C15 (Ref. 11).

In addition, at 225 degF during the heatup, C1.2 step 5.4.8 (Ref. 12) has the operator VERIFY that both CC pumps are operating with CC flow through both RHR HX's.

As a result, if the applicant is aware of the procedure limitation (1C15 3.6) and the previously completed steps, (c) would be eliminated as a correct answer because it clearly implies that CC flow was NOT established. It is not fair to the applicant to give a plant condition and expect him to assume that required actions and limitations are not being complied with.

Answers (a) and (b) are not correct because there is no indication of leakage from the RCS to CC, and (d) is not correct because increasing flow through the heat exchanger will cause a greater heat load to the CC system. Prairie Island recommends the question be deleted due to no correct answer existing.

Facility Comments

Record #72 (RO#41, SRO N/A)

Question:
Using the attached diagram of the Dual Channel Drawer for 1R7, which lights are lit when the Operation Selector Switch is placed in the CHECK SOURCE position?

a. Orange, Blue AND both red lights
b. Orange AND Blue lights ONLY
c. Orange AND BOTH Red lights ONLY
d. Orange, Blue, AND High Alarm Red lights ONLY

Listed correct answer is (d), indicating Power (orange), Channel Test (blue) and High Alarm (red) lights are lit.

The key states that the check source being moved into alignment with the detector causes the high alarm to actuate. A review of plant documents reveals this is not the case.

The check source reading is recorded during the performance of SP1243, Radiation Monitoring Quarterly Source Test (Ref. 13). As shown on Page 25, the reading with the check source (corrected reading) is 6.0E-3 R/hr. The high alarm setpoint is 5.0E-2 R/hr (Ref. 14). The check source reading is nearly a decade below the high alarm setpoint.

The simulator accurately reflects the plant setpoints, with the high alarm NOT coming in.

As a result, Prairie Island requests the correct answer be changed to (b), Orange AND Blue lights ONLY.

Ref. 1
Page 1

REACTOR TRIP OR SAFETY INJECTION

A. PURPOSE

This procedure provides actions to verify proper response of the automatic protection systems following manual or automatic actuation of a REACTOR TRIP OR SAFETY INJECTION, to assess plant conditions, and to identify the appropriate recovery procedure.

B. ENTRY CONDITIONS

1. REACTOR TRIP OR SAFETY INJECTION

2. Transition entry from:

   1ES-0.1, Step 1
   1ES-0.1, Step 10
   1ES-0.3A, Step 1
   1ES-0.3B, Step 1
   1ES-0.4, Step 1
   1FR-I.2, Step 5

C. ATTACHMENTS:

TABLE E0-1: Auto Actions Guide
ATTACHMENT G: Unit 1 Containment Isolation Valve Locations
ATTACHMENT J: Isolate Unit 1 Moisture Separator Reheaters

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE *Circled step numbers show IMMEDIATE ACTION steps.*	
1	Verify Reactor Trip: • Reactor trip and bypass breakers - OPEN • Neutron flux - DECREASING • Rod position indicators - ZERO • Rod bottom lights - LIT	Manually trip reactor. <u>IF</u> reactor power <u>NOT</u> LESS THAN 5%, <u>THEN</u> go to 1FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, Step 1.
2	Verify Turbine Trip: a. Both turbine stop valves - CLOSED	a. Manually trip turbine. <u>IF</u> <u>NOT</u>, <u>THEN</u> verify all turbine control valves closed. <u>IF</u> <u>NOT</u>, <u>THEN</u> manually close control valves. <u>IF</u> <u>NOT</u>, <u>THEN</u> close MSIVs and bypass valves. <u>IF</u> <u>NOT</u>, <u>THEN</u> locally trip turbine (local trip lever on turbine pedestal).

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	Verify Safeguard Buses Energized:	
	a. Safeguard buses - AT LEAST ONE ENERGIZED	a. Go to 1ECA-0.0, LOSS OF ALL SAFEGUARDS AC POWER, Step 1
	b. Safeguard buses - BOTH ENERGIZED	b. Initiate action to restore power to deenergized safeguard bus per 1C20.5 AOP1, REENERGIZING 4.16KV BUS 15 <u>OR</u> 1C20.5 AOP2, REENERGIZING 4.16KV BUS 16.
4	Check If SI Is Actuated: • "SI ACTUATED" status light - LIT -OR- • Any SI first-out annunciators - LIT	Check if SI is required. <u>IF</u> SI is required, <u>THEN</u> manually actuate. <u>IF</u> SI is <u>NOT</u> required, <u>THEN</u> go to 1ES-0.1, REACTOR TRIP RECOVERY, Step 1.

Ref. 2

3.3 Shift Managers

The Shift Manager is the senior management representative on shift and is responsible for the safe operation of the plant at all times. The Shift Manager represents the Plant Manager and Operations Manager when they are not on site. He is responsible for:

3.3.1 Safe operation of the plant.

3.3.2 Assisting in the development, implementation and maintenance of high standards and management expectations.

3.3.3 Providing effective leadership and stewardship of their assigned crew.

3.4 Operations Group Personnel

Operations Group personnel are responsible for performing operational activities in the manner described by the expectations and standards established by this procedure.

4.0 DEFINITIONS

None

5.0 REQUIREMENTS

5.1 Conduct of Operations General Requirements

5.1.1 Plant operations shall be conducted in a manner that establishes nuclear and personal safety as the highest priority while additionally employing a conservative decision making process.

5.1.2 If a protective system setpoint is being approached in an uncontrolled manner, the Operator at the Controls is expected to take action as necessary to correct the problem or place the plant in a safe condition prior to the process reaching the setpoint.

5.1.3 Operations personnel are expected to accept instrument readings and indications as accurate unless proven otherwise. When unusual or unexpected indications occur, operators will check any redundant or backup indications that may be available to validate the instrument response. Operators will respond to instrument readings and indications in a timely, conservative and deliberate manner as required by operating procedures.

Ref. 3

7. MODES OF APPLICABILITY OF THE ERGs

The ERG network was developed to accommodate emergency transients occurring at a "hot" or "at power" condition. The transients envisioned would result in either Reactor Protection System or Engineered Safeguards Systems actuation, with some corresponding, or subsequent, operator action needed. The guidance for operator action is based upon having the safety-related equipment required by Technical Specifications for MODE 1 or MODE 2 operation available for use. For transients initiating during other MODEs of operation, the same complement of equipment cannot be assumed available, so that some instructions within the ERGs may not be applicable. Just as earlier steps in the ERGs generally involve response to the transient, later steps generally relate to getting the plant to a cold shutdown condition. If the plant is already substantially cooled down and depressurized, some instructions may not be applicable.

Critical Safety Function monitoring using the Status Trees also assumes a MODE 1 or MODE 2 initial condition, followed by some Reactor Protection System actuation, to result in a subcritical reactor. Use of the trees can be extended beyond this original intent, but with an understanding of the intent of each tree. For example, the Heat Sink tree assumes the steam generators are required for heat removal by steaming. If all reactor decay heat is being removed by the RHR System, the steam generators are not required in their normal capacity. So SG availability is really not required to be satisfied. Yet the tree would indicate an SG in either wet layup or in dry layup to be abnormal.

To clarify the usability of the ERGs for transients originating during other than the assumed initial operating MODES, a detailed review of the entire network has been performed. The results are presented in the following table. In some cases, slight modifications would be needed to extend the applicable range of a guideline beyond its original intent.

Ref. 4

TABLE 1 (Cont)

MODES OF APPLICABILITY

Guideline Designator	Applicable Modes*	Comments
E-3 ES-3.1 ES-3.2 ES-3.3	1, 2, 3 1, 2, 3 1, 2, 3 1, 2, 3	Assumes RHR System not in service. Modifications may be necessary if otherwise.
ECA-0.0 ECA-0.1 ECA-0.2	1, 2, 3, 4 1, 2, 3, 4 1, 2, 3, 4	Assumes RCS partly hot and pressurized. Problems with RCP seals are minimal if RCS is cold and depressurized.
ECA-1.1 ECA-1.2	1, 2, 3 1, 2, 3	Assumes RHR System not in service; entry is limited by stated conditions. Modifications necessary if otherwise.
ECA-2.1	1, 2, 3, 4	Assumes RHR System not in service, hot conditions. Not of concern if SGs will not steam.
ECA-3.1 ECA-3.2 ECA-3.3	1, 2, 3 1, 2, 3 1, 2, 3	Assumes RHR System not in service. Modifications may be necessary if otherwise.

Ref. 5

TITLE:	ALARM RESPONSE PROCEDURE	C47015
		Rev. 32

ANNUNCIATOR LOCATION: 47015-0207

12 RCP LABYRINTH SEAL LO ΔP	Alarm Labyrinth Seal Differential Pressure Low	
12 REACTOR COOLANT PUMP LABYRINTH SEAL LOW DIFFERENTIAL PRESSURE SER Input Point: 016 Address: 8W16	Approximate Setpoints Tripped: < 15" H_2O	Reset: Not Specified

AUTOMATIC ACTIONS

NONE

INITIAL ACTIONS

1. **Check** differential pressure low.
2. IF due to inadequate seal injection flow, THEN **increase** seal injection flow to clear alarm (6 gpm normal at low pressure and 8 gpm at high pressure).
3. IF due to loss of seal injection flow , THEN **refer** to C12.1 AOP1, LOSS OF RCP SEAL INJECTION.
4. IF caused by excessive #1 seal leakage, THEN **refer** to 1C3 AOP3, FAILURE OF A REACTOR COOLANT PUMP SEAL.

SUBSEQUENT ACTIONS

Effect necessary repairs AND **return** system to normal.

INSTRUMENTS & REFERENCES

1. Actuating device (**1PT-124**).
2. Flow Diagram XH-1-38.
3. Logic Diagram NF-40781 Sheet 2.
4. Schematic Diagram NE-40011 Sheet 153.
5. C3, REACTOR COOLANT PUMP.
6. 1C3 AOP3, FAILURE OF A REACTOR COOLANT PUMP SEAL.

C	LOSS OF RCP SEAL INJECTION	NUMBER: C12.1 AOP1
		REV: 10
		Page 3 of 9

1.0 PURPOSE

This malfunction results from failure of all charging pumps, failure of a piece of equipment or a valve, or pipe break in the Seal Injection System. Alternate cooling is available to the reactor coolant pumps (RCPs) from the thermal barrier heat exchanger. Reactor coolant will flow across the labyrinth seal and thermal barrier heat exchanger to the RCP #1 seal.

2.0 PROCEDURES

2.1 Symptoms

2.1.1 Annunciator **47015** [47515]-**0103**, 11[21] CHARGING PUMP OVERLOAD TRIP

2.1.2 Annunciator **47015** [47515]-**0104**, 12[22] CHARGING PUMP OVERLOAD TRIP

2.1.3 Annunciator **47015** [47515]-**0105**, 13[23] CHARGING PUMP OVERLOAD TRIP

2.1.4 Annunciator **47015** [47515]-**0203**, CHARGING PUMP IN AUTO HI/LO SPEED

2.1.5 Annunciator **47015** [47515]-**0206**, 11[21] RCP LABYRINTH SEAL LO ΔP

2.1.6 Annunciator **47015** [47515]-**0207**, 12[22] RCP LABYRINTH SEAL LO ΔP

2.1.7 Annunciator **47015** [47515]-**0208**, 11[21] RCP NO. 1 SEAL INLT OR OUTL HI TEMP

2.1.8 Annunciator **47015** [47515]-**0209**, 12[22] RCP NO. 1 SEAL INLT OR OUTL HI TEMP

2.1.9 Annunciator **47015** [47515]-**0306**, 11[21] RCP SEAL LEAKOFF HI FLOW

2.1.10 Annunciator **47015** [47515]-**0307**, 12[22] RCP SEAL LEAKOFF HI FLOW

2.1.11 Annunciator **47015** [47515]-**0406**, 11[21] RCP SEAL LEAKOFF LO FLOW

2.1.12 Annunciator **47015** [47515]-**0407**, 12[22] RCP SEAL LEAKOFF LO FLOW

2.1.13 Seal water injection low flow indication

2.2 Automatic Actions

NONE

2.3 Immediate Manual Actions

NONE

Ref. 6 Page 2

2.4 Subsequent Manual Actions

2.4.1 IF both seal injection AND CC have been lost or CC flow is < 195 gpm, THEN **refer** to 1C3 AOP2 [2C3 AOP2], Loss of RCP Seal Cooling. ______

2.4.2 IF seal injection is lost due to loss of a charging pump, THEN **perform** the following:

A. **Start** any charging pump and/or **adjust CV-31198 [CV-31211]**, CHARGING LINE FLOW CONTROL VALVE, to restore seal injection. ______

B. IF letdown isolated and restoration is desired, THEN **restore** letdown per C12.1, Letdown, Charging, and Seal Water Injection. ______

2.4.3 IF seal injection is lost due to **CV-31198 [CV-31211]**, CHG LINE FLOW CONT, failing OPEN, THEN **perform** Step 2.4.3.A OR Step 2.4.3.B to restore seal injection:

A. If desired **isolate** charging using the regen heat exchanger charging line outlet control valve as follows:

1. **Remove** normal letdown from service by CLOSING the following valves:

   CV-31226 [CV-31230], LETDOWN LINE ISOL, using **CS-46165 [CS-49536]** ______

   CV-31255 [CV-31279], LETDOWN LINE ISOL, using **CS-46133 [CS-49667]**. ______

2. If necessary **reduce** charging pump speed to prevent lifting the charging pump relief valve when the next step is performed. ______

3. **CLOSE CV-31328 [CV-31420]**, REGEN HX CHG LINE OUTL using **CS-46296 [CS-49578]**. ______

B. If desired **throttle** the manual inlet or outlet valve for the failed charging line control valve:

VC-7-9 [2VC-7-9], CHARGING LINE CV-31198 [CV-31211] OUTLET ______

OR

VC-7-11 [2VC-7-11] CHARGING LINE CV-31198 [CV-31211] INLET ______

C Section	TITLE LOSS OF RCP SEAL COOLING	NUMBER: 1C3 AOP2 REV: 4 Page 3 of 6

1.0 PURPOSE

This procedure provides guidance for action necessary to mitigate the consequences of a loss of RCP seal cooling.

2.0 PROCEDURES

2.1 Symptoms

Both CC flow AND seal injection have been lost to the RCPs.

2.2 Automatic Actions

NONE

NOTE:	Loss of only component cooling is covered in 1C14 AOP1, Loss of Component Cooling. Loss of only seal injection is covered in C12.1 AOP1, Loss of RCP Seal Injection.

2.3 Immediate Manual Actions

NONE

2.4 Subsequent Manual Actions

2.4.1 IF bearing water temperature is less then 200°F, THEN:

A. **Attempt** to START any charging pump to restore seal injection. ______

B. **Attempt** to START any component cooling pump to restore seal cooling. ______

C. **OPEN** OR **verify OPEN** the RCP thermal barrier CC outlet valves:

- **CV-31245**, 11 RCP PUMP THERMAL BARRIER CLNT OUTLET. ______
- **CV-31246**, 12 RCP PUMP THERMAL BARRIER CLNT OUTLET. ______

D. IF necessary, THEN **notify** the Aux. Bldg, operators that CC and seal injection to the RCPs have been lost AND **direct** them to investigate. ________

E. IF either seal injection OR thermal barrier CC flow is restored, THEN **continue** with Step 2.5 Recovery Actions. ________

2.4.2 IF bearing water temperature reaches 200°F, THEN:

A. **Trip** the reactor, THEN:

- **Initiate** 1E-0, Reactor Trip or Safety Injection ________

AND

- **Complete** Steps 2.4.2.B, C, D, and E. ________

B. **Stop** the affected RCP(s). ________

C. **Transfer** the affected RCP spray valve(s) to "MANUAL" AND CLOSE:

- **CV-31224**, A PRZR SPRAY ________
- **CV-31225**, B PRZR SPRAY ________

D. IF both RCPs are affected, THEN **CLOSE** one seal return containment isolation valve:

- **MV-32199**, SEAL WTR RT CONT ISOL ________

OR

- **MV-32166**, SEAL WTR RT CONT ISOL ________

E. IF only one RCP is affected, THEN **CLOSE** the number 1 seal leakoff isolation valve for the affected RCP:

- **CV-31335**, 11 RCP SEAL LEAKOFF ISOL ________

OR

- **CV-31336**, 12 RCP SEAL LEAKOFF ISOL ________

Ref. 8

TITLE:	ALARM RESPONSE PROCEDURE	C47012
		Rev. 37
		Page 1 of 2

ANNUNCIATOR LOCATION: 47012-0507

PRZR LVL DEVIATION	Alarm Level Deviation from Program Level	
PRESSURIZER LEVEL DEVIATION SER Input Point: 206 Address: 6W43	Approximate Setpoints	
	Tripped	Reset
	± 10%	Not Specified

AUTOMATIC ACTIONS

Pressurizer high level backup heaters energize on high level deviation.

INITIAL ACTIONS

1. **Check** all pressurizer level channels to verify level deviation.
2. IF actual pressurizer level deviation exists, THEN **perform** the following:
   A. **Verify** charging pump in automatic and charging flow proper for the level.
   B. IF necessary, THEN **place** charging pump speed control in "MANUAL".
   C. **Control** charging pump speed as necessary to restore pressurizer level to programmed level.
3. IF due to an instrument failure, THEN **refer** to 1C51, INSTRUMENT FAILURE GUIDE - UNIT 1.

SUBSEQUENT ACTIONS

1. IF necessary, THEN **run** an RCS leak rate test.
2. IF necessary, THEN **refer** to 1C4 AOP1, REACTOR COOLANT LEAK.
3. IF condition was caused by malfunction of automatic pressurizer level control system, THEN **notify** System Engineer.
4. **Effect** repairs AND **return** system to normal.

Ref. 9 Page 1

REACTOR COOLANT LEAK

1.0 PURPOSE

This procedure describes the symptoms associated with small reactor coolant leakage, the methodology for determining the path of such leakage, and the necessary corrective action.

NOTE: **T.S.3.1.C (IT.S.3.4.14) RCS Leakage, should be consulted anytime detected or suspected leakage from the reactor coolant system is being investigated or evaluated.**

2.0 PROCEDURES

2.1 Symptoms

2.1.1 Control Room

A. Decreasing pressurizer level or level deviation alarm.

B. Charging pump speed increase due to increased abnormal make-up.

C. Decreasing VCT level.

D. Daily leak rate surveillance.

E. Increased radiation levels on 1R-2, 7, 11 ,12, 15, 19, 22, 30, 37 or 39.

F. Annunciator **47016-0502** OR **47016-0503**, RHR PUMP HI PRESS.

G. Annunciator **47015-0203**, CHARGING PUMP IN AUTO HI/LO SPEED.

2.1.2 Local

A. Boric acid accumulation.

B. Leaking pipes - water or steam.

2.2 Automatic Actions

2.2.1 Charging pump speed increase in response to decreasing pressurizer level.

2.2.2 VCT automatic make-up.

2.2.3 Letdown isolation on low pressurizer level.

Ref. 9 Page 2

2.3 Immediate Manual Actions

NONE

2.4 Subsequent Manual Actions

2.4.1 <u>IF</u>, at any time, RCS inventory can not be maintained by available charging flow, <u>THEN</u> manually **trip** the reactor <u>AND</u> **go** to 1E-0, Reactor Trip or Safety Injection. ______

S 2.4.2 **Start** additional charging pumps as needed to control pressurizer level. ______

S 2.4.3 <u>IF</u> VCT level cannot be maintained by the make-up system, <u>THEN</u> **align** charging pump suction to the RWST. ______

2.4.4 Use ERCS "LEAK" Program and/or Control Board indications to determine approximate leak rate. ______

2.4.5 **Determine** the location of the leak using Figure 1 and the associated tables. ______

2.4.6 **Isolate** the leak, if possible. ______

2.4.7 <u>IF</u> leakage is identified as steam generator tube leakage, <u>THEN</u> **refer** to 1C4 AOP2, Steam Generator Tube Leak. ______

2.4.8 <u>IF</u> unable to identify the source of the leak using tables 1, 2, 3, or 4 <u>THEN</u> attempt to locate the leak by sequentially isolating service systems from the RCS using steps A thru E below.

A. **Remove** Letdown from service as follows:

1. **CLOSE or verify** CLOSED the following:

a. **CV-31325**, 40 GPM ISOLATION, using **CS-46170.** ______

b. **CV-31326**, 40 GPM ISOLATION, using **CS-46171.** ______

c. **CV-31327**, 80 GPM ISOLATION, using **CS-46174.** ______

Ref. 10

NOTE:	Both SI pumps may be in "PULLOUT" and the control switch covers installed when RCS temperature is less than 310° F (Reference 7.1.5).

5.4.7 WHEN RCS temperature is greater than 218° F, THEN perform the following:

A. **Remove** the switch cover on 11 SI Pump. ________

B. **Clear** the Secure Card on 11 SI Pump (S.C. installed per 1C1.3). ________

C. **Place CS-46178**, 11 SI PUMP, in "AUTO." ________

5.4.8 WHEN RCS temperature is greater than 225°F, THEN **verify** the following:

A. Both CC pumps are operating. ________

B. CC flow through both RHR heat exchangers. ________

5.4.9 Using PINGP 1012 as the basis list, **log** CLOSURE of the total OPENINGS in ABSVZ for valves 3 inches or less in the ABSVZ openings log. (Initiated per 1C1.3.) ________

CAUTION:	DO NOT ALLOW VCT PRESSURE TO DROP BELOW 20 PSIG DURING THE PERFORMANCE OF STEP 5.4.10.

5.4.10 **Form** a Hydrogen blanket in the VCT per C12.4, VCT Gas Control, and **repeat** as necessary to establish an RCS Hydrogen concentration per Duty Chemist. ________

5.4.11 WHEN SG pressure exceeds 40 psig, THEN **complete** C1.1.27-1C, Part III Main and Auxiliary Steam Unit 1 Checklist. ________

3.4 After alignment for Phase II cooldown, initiate RHR flow slowly to avoid thermal shock to RHR System components and piping.

3.5 Loss of RHR suction from the loops will reduce core cooling capability and may cause damage to the RHR pumps because of cavitation. To minimize this possibility, half the breakers for the RCS to RHR motor valves are placed in the "OFF" position when the RCS is floating on the PRT or drained. See 1C1.2 and 1C1.3.

3.6 The flow control valves on the outlets of the RHR heat exchangers are not zero leakage valves. Therefore, if the RHR System is above approximately 225°F, boiling of the CC in the out-of-service RHR heat exchanger can occur. This may result in water hammer and/or rising CC surge tank level. Both CC pumps should be operated when the RHR System is above 225°F and both RHR heat exchangers should have CC flow through them.

3.7 If a RHR pump is started with the RCS solid, RCS pressure may fluctuate due to temperature differences between the RCS and RHR. RCS pressure should be closely monitored as the RHR system is returned to service.

3.8 If a RHR pump is operating in a high flow condition, (greater than 2500 gpm) venting of the suction piping should not be performed. Under these conditions, venting of the suction will draw air into the system rather than relieving any potential gas.

3.9 DO NOT OPEN both CC motor inlet valves to the RHR heat exchangers unless both CC pumps are running.

3.10 Both RHR HX outlet crossover valves (**RH-2-5** and **RH-2-6**), must be OPEN in order for the RHR pump discharges to be crosstied. IF either of these valves is CLOSED, THEN 11 RHR Pump will NOT be able to discharge to the Loop B cold leg return.

3.11 When supplying Component Cooling to a RHR heat exchanger, the same train CC pump must be used, or the CC pump suction cross connect valves must be OPEN, due to pump suction piping limitations.

3.12 The CC HX TCVs have travel stops installed to limit the open travel. To support Phase II cooldown, the travel stops may be adjusted in the OPEN direction for up to 7 days. This has been evaluated in the safety evaluation for Mod 95CL04.

C	COMPONENT COOLING SYSTEM - UNIT 1	NUMBER: 1C14
		REV: 17
		Page 11 of 22

NOTE:	Component Cooling flow through the RHR heat exchangers should indicate between 2600 and 2800 gpm.

NOTE:	Component Cooling motor inlet valves to the RHR heat exchangers open automatically (if their control switches are left in the AUTO position) when the associated RHR pump is started.

CC INLET TO "OPEN"

5.4.4 **OPEN MV-32093**, 11 RHR HX CC INLT, using **CS-46023** to start Component Cooling flow through the heat exchanger. ______

5.4.5 **OPEN MV-32094**, 12 RHR HX CC INLT, using **CS-46027** to start Component Cooling flow through the heat exchanger. ______

THIS IS DONE DURING UNIT COOLDOWN TO PREPARE TO PLACE RHR IN SERVICE. EXCEPT AS DIRECTED BY WORK ORDER, VALVES REMAIN IN THIS CONDITION (OPEN) UNTIL RHR IS REALIGNED FOR SFGDS INJECTION DURING PLANT HEATUP.

Ref. 11 Page 3

5.7 Shutdown and Alignment for Safeguards Operation

This procedure describes the steps necessary to shutdown the RHR System from decay heat removal, cool it down, and align it for automatic low head safety injection.

This procedure is entered from 1C1.2, Unit 1 Startup.

5.7.1 **Verify** a pressurizer bubble has been formed. ________

5.7.2 **Verify** at least one RCP is running. ________

5.7.3 **CLOSE** the in-service RHR HX outlet flow control valve by adjusting its associated manual controller as soon as the primary system heatup has commenced.

CV-31235, 11 RHR HX RC OUTLET FLOW (1HC-624)

OR

CV-31236, 12 RHR HX RC OUTLET FLOW (1HC-625) ________

5.7.4 **CLOSE MV-32066**, RHR TO RC LOOP B COLD LEG by placing **CS-46225** to the "CLOSE" position. ________

5.7.5 **Verify** all letdown orifice isolation valves are OPEN.

CV-31325, LTDN ORIFICE ISOL 40 GPM ________ |

CV-31326, LTDN ORIFICE ISOL 40 GPM ________ |

CV-31327, LTDN ORIFICE ISOL 80 GPM ________

5.7.6 **Clear** secure cards on for Shift Supervisor, and **OPEN** the RHR to reactor vessel injection motor valves:

CS-46223, MV-32064, RHR TO RX VSL S.C.________ ________

CS-46224, MV-32065, RHR TO RX VSL S.C.________ ________

5.7.7 **OPEN** the RHR pump suction valves from the RWST:

MV-32084, RWST TO 11 RHR PUMP, using **CS-46202** ________

MV-32085, RWST TO 12 RHR PUMP, using **CS-46203** ________

5.7.8 **CLOSE** the Loop A RHR suction isolation valves:

MV-32164, LOOP A HOT LEG TO RHR, using **CS-46226** ______

MV-32165, LOOP A HOT LEG TO RHR, using **CS-46228** ______

5.7.9 **CLOSE** the Loop B RHR suction isolation valves:

MV-32230, LOOP B HOT LEG TO RHR ______

MV-32231, LOOP B HOT LEG TO RHR ______

5.7.10 **CLOSE MV-32234** RHR TO LTDN LINE, by placing **CS-46175** in the "CLOSE" position. ______

5.7.11 **Verify** an adequate number of Cooling Water Pumps are running to maintain CLG WTR header pressure. ______

5.7.12 **Start** the standby CC pump by momentarily placing the control switch in "START":

CS-46036, 11 CC WTR PUMP

OR

CS-46037, 12 CC WTR PUMP ______

NOTE:	**CC flow through an operating RHR heat exchanger should indicate between 2600 and 2800 gpm.**

NOTE:	**The CC inlet valve to the RHR heat exchanger OPENS automatically (if the control switch is in "AUTO") when the associated RHR pump is started.**

5.7.13 **Start** the standby RHR pump by momentarily placing its control switch in the "START" position:

CS-46184, 11 RHR PUMP

OR

CS-46185, 12 RHR PUMP ______

5.7.14 Slowly **OPEN** both RHR HX outlet flow control valves by adjusting their associated manual controllers.

CV-31235, 11 RHR HX RC OUTLET FLOW (1HC-624) ______

CV-31236, 12 RHR HX RC OUTLET FLOW (1HC-625) ______

5.7.15 **Transfer 1HC-626A** to "MANUAL" and slowly **CLOSE CV-31237**, 11/12 RHR HX BYPASS FLOW. ________

5.7.16 **Adjust 1HC-626A** auto setpoint to zero. ________

5.7.17 **CLOSE** the following valves:

RH-2-3, 12 RHR HX CROSSOVER INLET ________

RH-2-4, 11 RHR HX CROSSOVER INLET ________

RH-2-5, 11/12 RHR HX CROSSOVER OUTLET ________

RH-2-6, 11/12 RHR HX CROSSOVER OUTLET ________

5.7.18 <u>WHEN</u> RHR loop temperature is less then 200°F, as indicated on **1TR-630**, RHR TO RCS/CVCS LTDN (Green Pen), <u>THEN</u> **stop** both RHR pumps by momentarily placing the control switch in "STOP":

CS-46184, 11 RHR PUMP ________

CS-46185, 12 RHR PUMP ________

5.7.19 **Perform** SP 1369, Exercising 11 & 12 RHR Pump Suction Line Check Valves (for <u>both</u> refueling outages and CSD outages). ________

5.7.20 **Turn** the MCC breakers for the RHR loop isolation valves to the "OFF" position and **replace** the Safeguard Hold Cards on the breakers:

HC 1-180, **1LA1-B1**, 1 RCS LP A HOT LEG RHR SPLY (INSIDE) MV-32164 ________

HC 1-181, **1LA1-D1**, 1 RCS LP B COLD LEG RHR INJ MV-32066 ________

HC 1-182, **1LA2-C1**, 1 RCS LP B HOT LEG RHR SPLY (INSIDE) MV-32230 ________

Ref. 12

5.7.21 IF not previously done, THEN **place MV-32234**, RHR TO LTDN LINE, breaker **1LA2-B3** to "OFF." ______

5.7.22 **Verify** the CC HX TCV travel stops are in place:

CV-31381, 11 CC HX CLG WTR OUTLET CV ______

CV-31411, 12 CC HX CLG WTR OUTLET CV ______

NOTE:	When the CC pump is stopped, hold the control switch in the "STOP" position until CC System pressure stabilizes above 75 psig.

5.7.23 IF it is desired to **stop** one CC pump, THEN **perform** the following:

UNTIL NEW IN "OPEN"

A. **Place** the associated RHR HX CC inlet motor valve in "AUTO" and **verify** the motor valve "CLOSES."

CS-46023, 11 RHR HX CC INLET MV-32093

OR

CS-46027, 12 RHR HX CC INLET MV-32094 ______

B. **Stop** the desired CC pump.

CS-46036, 11 CC WTR PUMP

OR

CS-46037, 12 CC WTR PUMP ______

C. **CLOSE** the associated CC HX Cooling Water inlet motor valve.

CS-46044, 11 CC HX CLG WTR INLET MV-32145

OR

CS-46047, 12 CC HX CLG WTR INLET MV-32146 ______

2001 3897 0820

Ref. 13 Page 1

F010807011

..IE ISLAND NUCLEAR GENERATING PLANT
.NORTHERN STATES POWER COMPANY

SURVEILLANCE PROCEDURES

SP	RADIATION MONITORING QUARTERLY SOURCE TEST	SP 1243 REV: 4
		WO: 105757
		Page 1 of 27 4 dz

SYSTEMS:	RD

RESULTS/COMMENTS:

Minimum filming requirements: Coversheet and Table 1.

Work Order Initiated: YES ______ NO X WO No. ________

Test Performance:

Performed By: AH / BH / [signature] Date: 7/31/01
(Signature or Initials)

Additional Requirements:

None

Review Of Acceptability:
Acceptance Criteria Met? YES/~~NO~~ RAD Protection Supervisor: T. Egan

SP Completion: Test is Complete
SP Surveillance Schedule Satisfied: YES/NO Surv. Admin: [signature] 9-1-01

Other Actions for Consideration:

System Engineer Review: JRStrickland Date: 8/3/01

5AWI 1.5.0 compliance. 4 page(s) total to be retained. System Engineer: JRH Date: 8/3/01

O.C. REVIEW DATE: 1-12-00	APPROVED BY: AJohnson	DATE: 1-11-00

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
NORTHERN STATES POWER COMPANY

SURVIELLANCE PROCEDUI

SP	TITLE: RADIATION MONITORING QUARTERLY SOURCE TEST	NUMBER: SP 1243
		REV: 4
		Page 25 of 27

TABLE 1 - RADIATION MONITORING QUARTERLY SOURCE TEST DATA

Train "B" Rack 1RM1, 1RM2, 21 -(Unit 1 Side)

Radiation Monitor	Desired	Back Ground	Low Limit	Corrected Reading	High Limit	Units	Date	Time	Comments/ Notes/WO#
R-1	2.3E-03	2.0E-04	1.3E-03	3.0E-03	4.1E-03	R/hr	7/31/2001	10:33:00 AM	
1R-2	2.0E-02	7.0E-03	1.1E-02	1.5E-02	3.5E-02	R/hr	7/31/2001	10:08:00 AM	
R-3	3.5E-03	5.0E-04	2.0E-03	5.0E-03	6.2E-03	R/hr	7/31/2001	8:50:00 AM	
R-4	1.0E-01	8.0E-04	5.6E-02	9.0E-02	1.8E-01	R/hr	7/31/2001	9:26:00 AM	
R-5	9.0E-02	3.0E-04	5.1E-02	7.0E-02	1.6E-01	R/hr	7/31/2001	8:36:00 AM	
R-6	1.0E-02	7.0E-04	5.6E-03	1.0E-02	1.8E-02	R/hr	7/31/2001	8:52:00 AM	
1R-7	7.0E-03	2.0E-03	3.9E-03	6.0E-03	1.2E-02	R/hr	7/31/2001	10:10:00 AM	
R-8	8.0E-03	5.0E-04	4.5E-03	7.0E-03	1.4E-02	R/hr	7/31/2001	9:46:00 AM	
1R-9	1.0E+00	5.0E-03	5.6E-01	9.0E-01	1.8E+00	R/hr	7/31/2001	8:47:00 AM	
R-16	1.2E+03	4.0E+01	6.8E+02	1.0E+03	2.1E+03	CPM	7/31/2001	10:35:00 AM	
R-41	2.0E+03	2.0E+01	1.1E+03	3.0E+03	3.6E+03	CPM	7/31/2001	10:13:00 AM	
R-24	9.0E+03	2.0E+02	5.1E+03	8.0E+03	1.6E+04	CPM	7/31/2001	10:29:00 AM	
R-27	6.0E+03	4.0E+01	3.4E+03	5.0E+03	1.1E+04	CPM	7/31/2001	9:39:00 AM	
R-29	1.0E+02	4.0E-01	5.6E+01	9.0E+01	1.8E+02	mR/hr	7/31/2001	9:49:00 AM	
R-36	8.0E+01	2.0E+00	4.5E+01	9.0E+01	1.4E+02	mR/hr	7/31/2001	9:42:00 AM	
2R-39	7.0E+03	2.0E+01	3.9E+03	6.0E+03	1.2E+04	CPM	7/31/2001	9:36:00 AM	
1R-50	3.0E+02	3.0E-01	1.2E+02	2.0E+02	4.8E+02	mR/hr	7/31/2001	10:16:00 AM	

2

NOTE:	Notify System Engineer if actual readings are outside the tolerance.

Ref. 13 Page 2

Ref. 14

Calibration Card / Setpoint File - [Setpoint Master]

File Edit Module Report Window Help

Eqp ID...	1RM-07	IR-07 INCORE SEAL TABLE AREA RAD METER							
Unit	0	System	RD	FSystem	RD	Critical	N	Dept	INST
Setpoint #	1	Function	HI ALM					CURRENT	
Rev #	1	09/23/1999							

	Value	Eng Units	Dec	Sci Not
Setpoint	5.00E-02	R/HR	2	☑
Analog Eqv	5.00E-02	R/HR	2	☑
Nominal SP				☐
SP Basis				

WO ID... 9614736

Direction: Increasing

OPS Doc...:

SAR / T.S.: 0

OC Review:

Meeting #:

Reason for Change OR Notes: THIS SET POINT WAS LAST CHANGED BY SPCR 255 SEE RECORDS MANAGEMENT FOR PRIOR HISTORY

Requested By	OBRM05	09/23/1999
Approved By	SHLD01	09/23/1999
Changed By	SHLD01	09/23/1999

Print | Approve | Change | Refresh | Delete | Request | Save | Close

Ready

NMC
Committed to Nuclear Excellence

Nuclear Management Company, LLC
Prairie Island Nuclear Generating Plant
1717 Wakonade Dr. East • Welch MN 55089

October 2, 2001

10 CFR 55

Mr. D. E. Hills
Chief-Operations Branch
U.S. Nuclear Regulatory Commission
Region III
801 Warrenville Road
Lisle, IL 60532-4351

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Dear Mr. Hills:

On September 21, 2001 the NRC conducted a written operator licensing examination at the Prairie Island Nuclear Generating Plant. On September 25, 2001 Prairie Island submitted a letter to the NRC recommending changes to three exam questions. Discussions between your staff and our training staff identified that additional clarification and supporting information was needed regarding record question #18 (RO#91, SRO#24), before the NRC could adequately consider and rule on our recommended changes.

Please refer to the enclosures for a summary of our station management's expectations regarding what an acceptable course of action would be in this situation. Also attached is a revised facility comment regarding record question #18 (RO#91, SRO#24). The facility comment was revised following a review of Prairie Island plant specific guidance and Westinghouse Owners Group generic guidance.

In this letter we have made 4 NRC commitments, indicated in bold italics in Attachment 1.

OCT 0 5 2001

If you have any questions, please contact Bob Gillespie at (651) 388-1165 x4036.

Sincerely,

Mano K. Nazar
Site Vice President
Prairie Island Nuclear Generating Plant

cc: Regional Administrator, Region III
Dave Pelton, Region III, Chief Examiner

Attachments:
1. Prairie Island Management Expectations
2. Facility Comments for Record Question #18
3. Reference Documentation (1-5)

Attachment 1

Prairie Island Management Expectations regarding implementation of ECA-0.0

Prairie Island has conducted a thorough review of the conditions presented in the record question #18, and is confirming answer (a) is the correct response expected from our operators.

Given the hypothetical condition where our plant is in HOT SHUTDOWN, and in the process of a normal cooldown, when a loss of all AC electrical power occurs, we would expect the implementation of ECA-0.0 to restore power. This response is based on the Prairie Island Emergency Operating Procedure Users Guide statement that:

> *If at any time a complete loss of power on the ac emergency busses takes place, the operator will enter procedure ECA-0.0, LOSS OF ALL AC POWER. This includes any time during the performance of any other EOP.*

We have reviewed this response with several of our staff to determine if this would be the actual response received, given our existing procedural direction and training. While we are satisfied that our operators would have implemented ECA-0.0 as the proper procedure, we have determined that inconsistent methods to reach/implement ECA-0.0 could have been used. For this reason, Prairie Island is clarifying our operating procedures and operator training to ensure a consistent and prompt transition to ECA-0.0.

The following specific actions are in progress:

1. ***Perform just-in-time training for licensed operators via PINGP 1224, CREW MEETING REVIEW OF NOTEWORTHY EVENT/NEAR MISS/CHANGE regarding the procedure use clarifications. (Initiated 9/28/01; all crews review as they return to watchstanding duties. Completion date 10/12/01)***

2. ***Revise ECA-0.0 Purpose and Entry Conditions page. The second entry condition will be rephrased to a statement such as "Anytime power is lost to both safeguards buses after the reactor and turbine are verified tripped." (Completion date 11/2/01)***

3. ***Revise SWI-O-10, "Operations Manual Usage" administrative procedure to emphasize that ECA-0.0 is a direct entry emergency operating procedure under shutdown conditions when loss of all AC power occurs. (Completion date 11/16/01)***

4. ***Revise lesson plans (P8197L-010 and P8197L-011) to reflect ECA-0.0 and SWI-O-10 clarifications outlined above. (Completion date 11/30/01)***

Attachment 2

Facility Comments for Record Question#18 (RO#91, SRO #24)

Question:
The following conditions exist on Unit 1:

-The unit is in HOT SHUTDOWN during a normal cooldown
-RCS temperature is 520 degF
-Pressurizer pressure is 1700 psig
-At this point, all Unit 1 4.16 KV buses lose power (Loss of All AC Power)

How would emergency procedure 1ECA-0.0 "Loss of Safeguards AC Power" be used in this situation?

a. ENTER 1ECA-0.0 immediately upon verification of loss of power to buses 15 and 16.
b. ENTER 1ECA-0.0 ONLY if RCS temperature rises above 540 degF.
c. ENTER 1ECA-0.0 ONLY if a safety injection signal occurs also.
d. ENTER 1ECA-0.0 ONLY if power is NOT restored to EITHER bus 15 or bus 16 when RCS temperature reaches 350 degF.

The correct answer listed in the exam is (c).

Facility Comment:

The correct answer should be changed to (a).

There are two entry paths for the operator to enter ECA-0.0 in the given plant condition. The Prairie Island Emergency Operating Procedure Users Guide (Ref. 1) states one of the entry conditions is "If at any time a complete loss of power on the ac emergency busses takes place, the operator will enter procedure ECA-0.0...." Thus, if the operator diagnoses loss of power to the safeguards busses, an entry would be made to ECA-0.0. The second path is by performing the immediate manual action steps of C14AOP1 (Ref. 2) for RCP high bearing temperature: the operator would trip the reactor, which is already shutdown, and initiate E-0, trip the RCPs and perform the remaining steps of C14AOP1. At Step 3 of E-0 (Ref. 3) the operator verifies loss of power to the safeguards busses and enters ECA-0.0. In each case the entry occurs upon verification of loss of power to the safeguards busses.

The entry condition provided at the beginning of the Westinghouse Owners Group (WOG) generic ECA-0.0 (Ref. 4) is "This guideline is entered from E-0, REACTOR TRIP OR SAFETY INJECTION, Step 3, on the indication that all ac emergency busses are deenergized." The entry based on loss of power to the busses without proceeding through E-0 is performed by the conditions stated in the WOG Users Guide and Prairie Island Users Guide. This same path is available and expected for Prairie Island. Prairie Island entry conditions for ECA-0.0 (Ref. 5) also states "Anytime power is lost to both safeguards buses after completion of 1E-0, Step 3." This entry ensures the first two steps of E-0 are performed when entry to the EOP network is with a reactor trip.

Answer (b) is incorrect because ECA-0.0 is applicable as an entry procedure above 200 degF.

Answer (c) is incorrect because a safety injection signal does not have to be generated to enter ECA-0.0. In fact, for the conditions stated, the safety injection signal is blocked. Since the PI users guide directs entry to ECA-0.0 and a path is available through AOPs and E-0, the use of the word "ONLY" makes (c) incorrect.

Answer (d) is incorrect for the same reason as answer (b).

Thus based on the above, answer (a) is the only correct choice.

PI EOP USERS GUIDE REF. 1

If the SUR was negative but more positive than a -.2 DPM. the user would follow this branch to a yellow coded terminus and find the appropriate procedure to be FR-S.2. Since the status priority is YELLOW, the user will continue to monitor the remaining trees and deal with any RED or ORANGE condition which might be encountered. If no other condition coded higher than YELLOW was present, then the operator would decide if the FRP will be performed or delayed. Often, a YELLOW condition is indicative of an off-normal condition which will be corrected with actions already in progress. At other times a YELLOW condition may be symptomatic of an abnormality in a single RCS loop on a single SG, as an example, and may be considered acceptable for the particular transient in progress. Whatever the case, the operator makes the decision whether to implement the FRP or wait.

If Log Power Range was less than 10^{-5}% and the source range and Log Power Range SUR are zero or negative, the user will follow the branches to a GREEN coded terminus annotated "CSF SAT". The user then evaluates the next status tree in the sequence.

If Log Power Range.was less than 10^{-5}% and the source range on Log Power Range SURs are positive, the user will follow branches to a YELLOW coded terminus. The appropriate procedure in this case is FR-S.2. The operator again would use his judgment regarding implementation of FR-S.2 while continuing recovery operations and observing Rules of Priority for the other Status Trees.

6.0 CONTROL ROOM USAGE OF THE EOP NETWORK

While the previous sections discussed the separate usage of individual procedures and evaluation of Status Trees, this section presents the intended overall usage of the entire EOP network.

Entry into the Emergency Operating Procedure set is limited to two specific conditions:

- o If at any time a reactor trip or safety injection occurs or is required, the operator will enter guideline E-0, REACTOR TRIP OR SAFETY INJECTION.

- * o If at any time a complete loss of power on the ac emergency busses takes place, the operator will enter procedure ECA-0.0, LOSS OF ALL AC POWER. This includes any time during the performance of <u>any</u> other EOP.

The entry into E-0 is expected to be the one more frequently used, so it is described first:

The operator enters at Step 1 and proceeds through E-0, following the rules of procedure usage as described above, with two possible outcomes:

- o He remains in E-0 and is directed by an action step to begin monitoring the Status Trees, or

- o He transfers to some other procedure, at which point he begins to monitor the Status Trees.

2.2 Automatic Actions

2.2.1 Auto start of standby CC pump on low discharge header pressure.

2.2.2 Auto OPEN **CV-31432**, CC SURGE TNK M-U CV, on low surge tank level.

2.3 Immediate Manual Actions

IF any RCP bearing reaches 200°F, THEN **perform** the following:

* 2.3.1 **Trip** the reactor THEN **initiate** 1E-0, Reactor Trip or Safety Injection AND **complete** the remaining steps of this procedure. ______

2.3.2 **Trip** the affected RCP(s). ______

2.4 Subsequent Manual Actions

2.4.1 IF the level in the CC Surge Tank cannot be maintained "on scale" OR CC flow cannot be maintained, THEN **perform** the following:

A. **Trip** the reactor, THEN **initiate** 1E-0, Reactor Trip or Safety Injection AND **complete** the remaining steps of this procedure. ______

B. **Trip** both RCPs. ______

C. **Isolate** letdown. ______

D. **Reduce** charging flow to minimum to supply the RCP seals. ______

2.4.2 IF necessary, THEN **verify** auto start of the standby CC pump. ______

2.4.3 **Verify** the CC surge tank level control valve is controlling level in the surge tank. IF not, THEN **restore** level to the CC surge tank by:

A. **Verify** OPEN or **OPEN MV-32375**, REACTOR MAKEUP TO 11 CC SURGE TANK, using **CS-46025**. ______

B. **Verify** locally that **CV-31432**, CC SURGE TANK M-U CV is OPEN (765' el. by CC Surge Tank). ______

C. **CLOSE CC-27-8**, 11 CC SURGE TNK XTIE ISOL (765' el. by CC Surge Tank). ______

D. **Stop** any CC transfers in progress. ______

 REF. 3 p. 1

REACTOR TRIP OR SAFETY INJECTION

A. PURPOSE

This procedure provides actions to verify proper response of the automatic protection systems following manual or automatic actuation of a REACTOR TRIP OR SAFETY INJECTION, to assess plant conditions, and to identify the appropriate recovery procedure.

B. ENTRY CONDITIONS

1. REACTOR TRIP OR SAFETY INJECTION
2. Transition entry from:

   1ES-0.1, Step 1
   1ES-0.1, Step 10
   1ES-0.3A, Step 1
   1ES-0.3B, Step 1
   1ES-0.4, Step 1
   1FR-I.2, Step 5

C. ATTACHMENTS:

TABLE E0-1: Auto Actions Guide
ATTACHMENT G: Unit 1 Containment Isolation Valve Locations
ATTACHMENT J: Isolate Unit 1 Moisture Separator Reheaters

REF. 3 p.2

Number:	Title:	Revision Number:
1E-0	REACTOR TRIP OR SAFETY INJECTION	REV. 19

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE *Circled step numbers show IMMEDIATE ACTION steps.*	
1	Verify Reactor Trip: • Reactor trip and bypass breakers - OPEN • Neutron flux - DECREASING • Rod position indicators - ZERO • Rod bottom lights - LIT	Manually trip reactor. IF reactor power NOT LESS THAN 5%, THEN go to 1FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, Step 1.
2	Verify Turbine Trip: a. Both turbine stop valves - CLOSED	a. Manually trip turbine. IF NOT, THEN verify all turbine control valves closed. IF NOT, THEN manually close control valves. IF NOT, THEN close MSIVs and bypass valves. IF NOT, THEN locally trip turbine (local trip lever on turbine pedestal).

REF. 3 p.3

Number:	Title:	Revision Number:
1E-0	REACTOR TRIP OR SAFETY INJECTION	REV. 19

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	Verify Safeguard Buses Energized:	
	a. Safeguard buses - AT LEAST ONE ENERGIZED	a. Go to 1ECA-0.0, LOSS OF ALL SAFEGUARDS AC POWER, Step 1
	b. Safeguard buses - BOTH ENERGIZED	b. Initiate action to restore power to deenergized safeguard bus per 1C20.5 AOP1, REENERGIZING 4.16KV BUS 15 <u>OR</u> 1C20.5 AOP2, REENERGIZING 4.16KV BUS 16.
4	Check If SI Is Actuated: • "SI ACTUATED" status light - LIT -OR- • Any SI first-out annunciators - LIT	Check if SI is required. <u>IF</u> SI is required, <u>THEN</u> manually actuate. <u>IF</u> SI is <u>NOT</u> required, <u>THEN</u> go to 1ES-0.1, REACTOR TRIP RECOVERY, Step 1.

REF. 4

Number W0G ECA-0.0	Title LOSS OF ALL AC POWER	Rev./Date LP-Rev. 1C 9/30/97

A. PURPOSE

This guideline provides actions to respond to a loss of all ac power.

B. SYMPTOMS AND ENTRY CONDITIONS

1) The symptom of a loss of all ac power is the indication that all main and emergency ac busses are deenergized.

2) This guideline is entered from E-0, REACTOR TRIP OR SAFETY INJECTION, Step 3, on the indication that all ac emergency busses are deenergized.

REF. 5

LOSS OF ALL SAFEGUARDS AC POWER

A. PURPOSE

This procedure provides actions to respond to a loss of all safeguards AC power.

B. ENTRY CONDITIONS

1. Transition entry from:

   1E-0, Step 3
   Anytime power is lost to both safeguards buses after completion of 1E-0, Step 3

C. ATTACHMENTS:

ATTACHMENT E: SG Wide Range Level - Adverse Conditions
ATTACHMENT G: Unit 1 Containment Isolation Valve Locations