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Florida Power
CORPORATION

May 2, 1980

File: 3-0-1-a

Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Nuclear Safety Task Force Priority Items and Confirmatory Order for Crystal River Unit 3, dated April 14, 1980

Dear Sir:

In our meeting with you on April 24, 1980, Florida Power Corporation committed to submit the Nuclear Safety Task Force Priority Items and responses on or about May 2, 1980. This letter fulfills that commitment.

In addition, the Confirmatory Order for Crystal River - Unit 3, dated April 14, 1980, is addressed and is answered in final part. The first part of the Order was completed upon the submittal of the response to I&E Bulletin 79-27 on April 25, 1980.

This submittal also addresses the FPC Corrective Action List prior to Plant Startup; B&W Recommendations; NUREG-0667, Transient Response of Babcock & Wilcox-Designated Reactors; NRC Confirmatory Order; and INPO/NSAC Recommendations.

Enclosure A reviews the purpose of the Nuclear Safety Task Force, lists the recommendations from five other organizations that were addressed, and gives a matrix to cross-reference between the Nuclear Safety Task Force Priority Items and the other five lists.

Enclosure B of this submittal gives the fifty-one (51) Priority Items of the Nuclear Safety Task Force and their responses.

Enclosure C discussed the items of the other five lists from Enclosure A that are not discussed in Enclosure B.

If you have any questions about this submittal, please contact this office.

Sincerely,

FLORIDA POWER CORPORATION

Patsy Y. Baynard

Patsy Y. Baynard
Manager
Nuclear Support Services

RMBekcF01(D1)

Attachment

cc: Director
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Suite 3100
101 Marietta Street
Atlanta, GA 30303

STATE OF FLORIDA

COUNTY OF PINELLAS

Dr. P. Y. Baynard states that she is the Manager, Nuclear Support Services, of Florida Power Corporation; that she is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of her knowledge, information, and belief.

Patsy Y. Baynard
Dr. P. Y. Baynard

Subscribed and sworn to before me, a Notary Public in and for the State and County above named, this 2nd day of May, 1980.

Catherine S. Cameron
Notary Public

Notary Public, State of Florida at Large,
My Commission Expires: August 8, 1983

CameronNotary 3(D12)

ENCLOSURE A

Following the February 26, 1980 transient at Crystal River Unit 3 (CR-3), the Nuclear Safety Review Task Force was established by and to report to Mr. J. A. Hancock, Assistant Vice President of Nuclear Operations, Florida Power Corporation. The purpose of the Task Force was to reexamine the adequacy of various systems, procedures and training at CR-3; identify and develop recommended changes; and recommend appropriate priorities of implementation. The priority of implementation process assures that needed corrective actions will be accomplished prior to restart of CR-3 after this refueling outage.

The Nuclear Safety Review Task Force has prepared a list of 51 priority items that must be answered prior to startup. In addition, there are five other lists of questions and concerns from various organizations that have been developed after the formation of the Task Force. These lists are summarized below and restated in Tables A-1, A-2, A-3, A-4 and A-5. A cross reference matrix was developed to show which response to each of the 51 priority items also answers an item in the 5 additional lists. The cross reference matrix is given in Table A-6 and the responses are discussed in Enclosure B. The items in Tables 2, A-3, and A-5, which are not addressed in Enclosure B are discussed in Enclosure C.

1. FPC Corrective Action List Prior to Plant Startup

   This list was the result of Request 2, specific to Crystal River Unit 3 from Harold R. Denton on March 6, 1980. Request 2 stated "Provide a list of proposed actions at CR-3 as a result of this event". These items are listed in Table A-1 and cross-referenced to Enclosure B by Matrix Table A-6.

2. B&W Recommendations

   Babcock and Wilcox reviewed the past history of NNI/ICS power supply failures and developed a list of recommendations to lessen the vulnerability of the Reactor Plant to these events. These items are listed in Table A-2 and discussed in Enclosure B and C.

3. NUREG-0667, Transient Response of Babcock & Wilcox-Designed Reactors

   The NRC formed a B&W Reactor Transient Response Task Force to study transients and consequences of malfunctions and failures of the Integrated Control System (ICS) and Non-Nuclear Instrumentation (NNI). The report is NUREG-0667, Transient Response of Babcock & Wilcox-Designed Reactors. The recommendations of this report are listed in Table A-3 and discussed in Enclosures B and C.

4. NRC Confirmatory Order

The NRC issued a Confirmatory Order for Crystal River Unit 3 on April 14, 1980. The order commitments are listed in Table A-4 and discussed in Enclosure B.

5. INPO/NSAC Recommendations

The INPO and NSAC staff completed a report on the CR-3 incident of February 26, 1980. The report was sent to Harold R. Denton by E. P. Wilkinson, and E. L. Zebroski on March 11, 1980. The corrective actions requested are listed in Table A-5 and discussed in Enclosures B and C.

TABLE A-1

FPC CORRECTIVE ACTION LIST
PRIOR TO PLANT STARTUP
(Refer to Matrix Table A-6 for corresponding response in Enclosure B)

ITEM 1

Thorough testing of the NNI(X) system to determine cause of initial failure.

ITEM 2

Review PORV circuitry to assure that credible power failures do not cause the PORV to open when it is not required to open. Review power supply independence between PORV and PORV isolation block valve to assure that failure which would affect PORV does not eliminate the possibility of PORV isolation block action.

ITEM 3

Modify Pressurizer Spray Valve so that NNI power failure will close valve.

ITEM 4

Provide positive indication of all three (3) Relief Valves.

ITEM 5

Establish procedural controls of selectable sources for indication and control.

ITEM 6

Train all operators and I&C technicians in response to NNI and ICS failures.

ITEM 7

Move 120 VAC ICS(X) power to Vital Bus.

ITEM 8

Repair Events Recorder System.

ITEM 9

Initiate a more extensive surveillance program on the Events Recorder System.

ITEM 10

Provide operator with redundant indications of main plant parameters.

TABLE A-1 (Continued)

ITEM 11

Annunciate loss of power to X and Y buses and ICS.

ITEM 12

Develop and institute a test program for changes in designs and modifications.

ITEM 13

Install indicating lights on all vital bus feeds.

ITEM 14

Modify vital bus panels for quick fuse replacement.

ITEM 15

Modify EF Pump auto start circuit and reactor trip circuit so that any power failure will not prevent activation on low SG level (control grade).

ITEM 16

PORV valve position indicating lights for solenoid will be added.

ITEM 17

Visually inspect the lower portion of the steam generator support skirts and anchor bolts. Remove any corrosive residue observed.

ITEM 18

Inspect the pressurizer heater bundles for seal leakage. Electrically check the pressurizer heater elements for continuity.

ITEM 19

A fatigue analysis will be performed on the pressurizer heaters to demonstrate that the 40-year design life was not adversely impacted by this transient.

ITEM 20

The relief valve loadings on the pressurizer relief nozzles should be determined and their effect will be assessed.

ITEM 21

The CRDMs will be checked for proper insulation resistance prior to their return to service.

TABLE A-1 (Continued)

ITEM 22

a. Perform a visual inspection of the pressurizer relief system (i.e., PORV, both code safety valves, and the discharge piping). Inspection of the discharge piping system, including hangers, should be performed to ensure that no gross distortions have occurred.

b. Confirm by calculation that the structural loads imposed on the valves and pressurizer as a result of the extended period of discharge to the quench tanks are acceptable.

c. Disassembly, inspection and refurbishment (as necessary) of the PORV and the code safety valves.

ITEM 23

Provide diverse containment isolation.

TABLE A-2

B&W RECOMMENDATIONS

1. Following the loss of an NNI or ICS power supply, the Emergency Feedwater System shall be automatically initiated as needed and controlled to maintain proper heat removal by steam generators.

2. A failure of an NNI power supply shall not cause the PORV to open nor shall it prevent the PORV isolation valve from functioning.

3. Following the loss of an NNI or ICS power supply, the capability of stopping main feedwater to prevent excessive main feedwater addition shall be available.

4. Following the loss of an NNI or ICS power supply, the capability of maintaining or restoring steam pressure in at least one steam generator shall be available.

5. The failure of NNI/ICS power supplies shall not prevent safety or protection systems from operating or prevent manual override of safety or protection systems.

6. The operator shall be trained and provided with sufficient information to identify a failed power supply and its related instruments. The remaining plant instrumentation and controls shall be sufficient to place and maintain the plant in a safe hot shutdown condition.

7. The capability to isolate letdown on loss of NNI or ICS power supplies shall be maintained.

8. A failure of an NNI power supply shall not cause the spray valve to open nor shall it prevent the spray block valve from functioning.

9. A loss of an NNI or ICS power supply shall not cause the pressurizer heater to fail on, or remain on, when the pressurizer level is low.

10. Seal injection and return (as required to prevent RCP seal damage) to the Reactor Coolant Pump shall be maintained upon an NNI/ICS power failure.

11. Review NNI/ICS systems and associated AC distribution systems fault protection requirements to minimize fault propagation.

12. Field changes to NNI/ICS systems shall be performed in accordance with a formal design control process.

13. NNI/ICS field changes should include reference(s) to identified installation and maintenance precautions.

14. Each utility should perform a management review to assure that proper alteration and maintenance practices are in place.

TABLE A-3

NUREG-0667, TRANSIENT RESPONSE OF BABCOCK & WILCOX-DESIGNED REACTORS

Auxiliary Feedwater (AFW) System

1. AFW system upgrade to safety grade.

2. AFW system automatic initiation and control.

3. Addition of motor-driven AFW pump for Davis-Besse.

4. Modifications to steam line break detection and mitigation system.

Instrumentation and Control

5. Improvements in plant control system (ICS/NNI).

6. Selected data set of principal plant parameters for operator.

7. Increased usage of incore thermocouples.

8. High radiation signal initiation of containment isolation.

Design and Operational Matters

9. System response to maintain pressurizer level on scale and pressure above HPI setpoint.

10. Sensitivity studies of operational modifications.

11. Modifications to eliminate immediate manual actions for emergency procedures.

12. Qualified I&C Technician on duty.

13. Operator training on Crystal River 3 event.

14. Guidelines for loss of NNI/ICS.

15. Mandatory one-week simulator training for operators as part of requalification program.

16. Evaluation of RCP restart criteria.

17. Alternative solution to PORV unreliability/safety system challenge rate concerns.

18. IREP Crystal River Study.

TABLE A-3 (Continued)

General Areas for Improvement

19. Performance criteria for anticipated transients.

20. Continued evaluation of need to trip RCPs during small break loss-of-coolant accidents.

21. Reevaluate location of AFW injection into OTSG.

22. Staff study of personnel-related LERs with respect to high number for B&W plants.

TABLE A-4

NRC CONFIRMATORY ORDER COMMITMENTS

II-1 Action which will allow the operator to cope with various combinations of loss of instrumentation and control functions. This includes changes in (A) equipment and control systems to give clear indications of functions which are lost or unreliable, (B) procedures and training to assure positive and safe manual response by the operator in the event that competent instruments are unavailable.

II-2 Determination of the effects of various combinations of loss of instrumentation and control functions by design review analysis and verification by test.

II-3 Correction of electrical deficiencies which may allow the power-operated relief valve and pressurizer spray valve to open on non-nuclear instrumentation power failures, such as the event which occurred at Crystal River Unit 3 on February 26, 1980.

In addition, this NRC Confirmatory Order required a written response to IE Bulletin 79-27 to be submitted by April 26, 1980 in accord with Florida Power Corporation's commitment. FPC submitted this response on April 25, 1980.

TABLE A-5

INPO and NSAC RECOMMENDATIONS

I.	TRAINING
I. A	Procedural requirements for declaration of appropriate emergencies should be emphasized in plant training sessions.
I. B	Review power supply failures and their effects on control systems. Include events such as ICS related malfunctions at Crystal River in plant training sessions and in simulator training.
I. C	Instrument technician work practices and their potential impact on plant safety should be reviewed in plant training sessions. Attention should be given to events similar to the 3/20/78 and 1/5/79 transients at the Rancho Seco plant where overcooling resulted from maintenance technician actions.
II.	PROCEDURES
II. A	Promulgate written procedures for switching instruments between power supplies, in the event of power supply failures and promulgate a procedure designating the preferred bus for each instrument.
II. B	Procedures for Steam Generator rupture matrix or its equivalent should be reviewed in conjunction with post-TMI requirements on steam-driven emergency feedwater pumps to determine if aggravating effects exist during loss of heat sink.
II. C	Procedures for orderly plant shutdown following loss of power supply should be prepared or reviewed/revised as necessary. Reactor system cooldown limits, and the basis for those limits should be reviewed.
II. D	The Industry should further analyze and resolve with the NRC the current reactor coolant pump trip procedures to be followed during a small break LOCA. Mandated procedures can be counterproductive to safety if they are not sufficiently discriminating to specific circumstances, and to specific plant designs.
II. E	The Industry should review the current High Pressure Injection pump requirements and resolve any procedural issues with the NRC. Procedures which avoid or minimize challenges to safety valves, primary system, (and eventually to the containment building itself) are needed.

TABLE A-5 (Continued)

Mandated procedures can be counterproductive to safety if they are not sufficiently discriminating to specific circumstances and to plant designs.

II. F Procedures for declaration of emergencies should be reviewed to determine if responsibility for monitoring plant conditions which lead to declaration of a specific emergency category should be assigned to a specific individual. It is suggested that this individual would also be responsible for immediately informing the senior person in charge at the time when these conditions for emergencies and emergency notification have been met.

III. PLANT SYSTEMS AND HARDWARE

The following list of problems should be investigated and corrective action taken as required.

III. A Loss of Power Supply

III. A. 1 Need for backup or bus transfer capabilities if a fault trips instrumentation and control power supplies.

III. A. 2 Coupling of indication, control and computer input signals, e.g., loss of power to ICS, NNI, or RCS results in loss of control board indication of many signals.

III. A. 3 PORV opening and its failure modes due to voltage variation resulting in loss of proper setpoint reference.

III. A. 4 Susceptibility of control systems to incorrect information caused by electrical faults, e.g., choking off feedwater to steam generators, withdrawing rods, and opening the turbine throttle.

III. A. 5 Instrument loops are selected by a switch in the control room. Designs should be reviewed, and wherever practical, field-tested to determine the effects of a loss of power to one of the instrument loops, and to establish the absence of cross-contamination of multiple power supplies in the instrument and control functions.

III. A. 6 The coincidence of having a mid-scale operating point and mid-scale instrument failure on loss of power gives uncertain information, e.g., loss of EFW auto start because steam generator level indication appeared to be higher than actual.

III. A. 7 Assignment of instruments to specific busses should insure as much redundancy as possible.

TABLE A-5 (Continued)

III. B	Data Handling and Display
III. B. 1	The adequacy of data handling and display systems should be reviewed. Examples of specific problems encountered during this event were as follows:
III. B. 1. a	Many instances of alarm conditions returning to a normal state without any prior indication of having reached an alarm state.
III. B. 1. b	Computer printout loss due to overload.
III. B. 1. c	The system monitoring the in-core temperatures automatically prints any temperature which indicates in excess of 700°F. The basis for selecting 700°F should be reviewed to determine if this number should be revised, since data was lost during the transient.
III. B. 2	Plant transient monitoring and recording. Plant transient records independent of process computer, to provide a tape record of main plant parameters, are desirable for all plants. They are desirable on an earliest practicable schedule.
	Steam Generator System
III. C	The Steam Generator rupture matrix or equivalent should be reviewed and changed as necessary to prevent actuation of isolation and loss of heat sink for events which do not actually involve ruptures in the steam generator system.

TABLE A-6

CR-3 CROSS REFERENCE MATRIX

Safety Task Force Priority Items	FPC Corrective Actions for Startup	B&W Recommendations	NUREG-0667 Recommendations	NRC Confirmatory Order	INPO NSAC Recommendations
1	1				
2	5	6	5	II-1	II.A, III.A.2
3	9				
4	12			II-2	III.A.5
5	17				
6	18				
7	21				
8	22				
9					II.E
10					
11					
12					
13					
14					
15				II-1	
16	6	3	5&13	II-1	I.B, II.A, II.C
17			13	II-1	
18	8				III.B.1.a
19					III.B.1.b
20			9		
21			12		
22					
23					
24					
25					
26					
27		7		II-1	III.A.6
28				II-1	
29					
30				II-1	
31		12			
32		13			
33				II-2	
34	10	1&6	5&6	II-1	III.A.2, III.A.7
35		4	10	II-2	III.A.6

TABLE A-6 (Continued)

CR-3 CROSS REFERENCE MATRIX

Safety Task Force Priority Items	FPC Corrective Actions for Startup	B&W Recommendations	NUREG-0667 Recommendations	NRC Confirm-atory Order	INPO NSAC Recommendations
36		11			III.A.1
37					
38	4				
39	14				
40	15	1	1&2		III.A.6
41	16			II-3	
42	19				
43	20				
44	23				
45			1&2		
46	2	2		II-3	III.A.3
47	3	8		II-3	
48	7		5		
49	11	6	5	II-1	
50	13				
51			4		II.B, III.C

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ENCLOSURE B

SAFETY TASK FORCE PRIORITY ITEMS

The first priority areas of inquiry for the Nuclear Safety Review Task Force were the subjects of power and power supply failures and the close-coupling of the NSSS with the secondary side of the plant. The objective was to strengthen identified weak points that would otherwise significantly increase the chances of a core damage incident. The concern for the "man/machine interface" emphasized the viewpoint of the operator as a major factor in review on an equal status consideration with the hardware side. It was not intended for the Task Force to examine all safety considerations identified in the FSAR on a nonprioritized basis; however, the Task Force was not precluded from any area of inquiry that analysis/strategy lead them.

ITEM 1

Thorough testing of the NNI(X) system to determine cause of initial failure.

RESPONSE

The following five items were checked on the NNI(X) system to determine cause of failure. Procedure PT-454 was developed to examine the system.

1. Verified auctioneering diodes and power supply capability to carry bus voltage.

2. Determined AC voltage level to produce 22 VDC on bus, which trips power supply monitor - 86.8 VAC.

3. Tested to determine if S_1 and S_2 could be closed with load on bus and power supply monitor in service. S_1 and S_2 time delay 0.5 seconds.

4. Interrupted AC power to S_1 and S_2 at V8DP to determine power will restore without tripping S_1 and S_2.

5. Visually inspected and tested all Type 820 modules in NNI(X) with 24 VDC supplies.

The first four items were completed by March 2, 1980, and all results were satisfactory.

The following is the results of Item 5:

1. A complete module-by-module check was made of all NNI(X) modules.

2. On Wednesday, March 5, 1980, a buffer module was found in the NNI(X) cabinets with the pin connectors touching between the +24 VDC and ground. The land on the printed circuit board was burned away.

ITEM 1 (Cont'd)

RESPONSE

3. The actual module that failed was installed on February 15, 1980, and was associated with the P_{sat}-T_{sat} Monitors. An investigation revealed that the failure in "X" power did not impair the function of the redundant channel in the "Y" cabinets, nor did any non-NNI T_{sat} equipment contribute to the incident.

4. All buffer modules in the "Y" cabinets were also inspected for any problems.

ITEM 2

Establish procedural controls of selectable sources for indication and control.

RESPONSE

Our investigation to-date of the February 26, 1980, transient at CR-3 indicates a need for the following procedural changes which will be implemented prior to restart:

1. Emergency or Abnormal Procedures
(Including recognition and response)

   a. Loss of vital bus power to a non-nuclear instrumentation bus.

   b. Loss of vital bus power to integrated control system.

2. Surveillance Procedures

   a. Events Recorder System.

   b. Instrument systems power supply and function switch positions.

3. Functional Test Procedures

   a. Subcooling monitor.

   b. Redundant instrument availability.

ITEM 3

Initiate a more extensive surveillance program on the Events Recorder System.

ITEM 3 (Cont'd)

RESPONSE

A surveillance procedure, SP-505, has been developed to perform a periodic functional check of the Events Recorder/Annunciator System and will be implemented before startup.

ITEM 4

Develop and institute a test program for changes in designs and modifications.

RESPONSE

The development and institution of a test program for changes in designs and modifications requested in Item 12 of the FPC Corrective Action List Prior to Plant Start is governed by existing Compliance Procedure CP-114, "Procedure for Preparation and Control of Permanent Modifications, Temporary Modifications, Deviations, and MAR Functional Test Procedures." CP-114 will be the controlling document in developing and performing the functional test for redundant instrument modifications. MAR 80-3-64 is the specific MAR package detailing the modifications to be made on the redundant instruments for major plant parameters.

Specific test procedures will be developed, issued, and approved as part of the construction work package authorizing and detailing the installation once the system modifications/redesigns have been finalized.

ITEM 5

Visually inspect the lower portion of the steam generator support skirts and anchor bolts. Remove any corrosive residue observed.

RESPONSE

Visual inspection and cleaning of the lower portion of the steam generator support skirts and anchor bolts was completed on April 12, 1980.

ITEM 6

Inspect the pressurizer heater bundles for seal leakage. Electrically check the pressurizer heater elements for continuity.

RESPONSE

Visual inspection of the pressurizer heater bundles for seal leakage was completed April 12, 1980. The heater element continuity checks were completed April 29, 1980. No problems were found.

ITEM 7

The CRDMs will be checked for proper insulation resistance prior to their return to service.

RESPONSE

The insulation resistance of the CRDMs has been checked and found to be within specifications.

ITEM 8

a. Perform a visual inspection of the pressurizer relief system (i.e., PORV, both code safety valves, and the discharge piping). Inspection of the discharge piping system, including hangers, should be performed to ensure that no gross distortions have occurred.

b. Confirm by calculation that the structural loads imposed on the valves and pressurizer as a result of the extended period of discharge to the quench tanks are acceptable.

c. Disassembly, inspection, and refurbishment (as necessary) of the PORV and the code safety valves.

RESPONSE

a. The pressurizer relief system including hangers for the discharge piping system was visually inspected and no gross distortions were detected. Work was completed on April 29, 1980.

b. Calculations have been performed by GAI which demonstrated that structural loads on the valves and pressurizer were within acceptable limits.

c. The disassembly, inspection and necessary refurbishment of the PORV and code safety valves is in progress. The PORV is presently being rebuilt and the replacement of the code safety valves is complete and are being checked as of April 30, 1980.

ITEM 9

Review procedures covering when HPI flow can be cut back and secured during a small break or overcooling transient.

RESPONSE

Modifications to the small break and non-LOCA overcooling transient procedures will be completed to only require 20°F subcooling. (A T_{sat} meter exists with digital display for ease of verification.) The guidelines will be submitted to the NRC for approval prior to implementation.

ITEM 10

Review procedures covering OTSG tube rupture in accordance with revised B&W Tube Rupture Guidelines and Small Break Guidelines.

RESPONSE

Interim draft guidance on handling a steam generator tube rupture was prepared and provided to Crystal River. A key item in the recommendation section is early recognition and identification of a steam generator tube rupture incident.

The final guidelines on handling a steam generator tube rupture will be submitted to the NRC for approval prior to implementation.

ITEM 11

Review procedures concerning proper OTSG level at HPI and manual RCP Trip in accordance with revised B&W Small Break Guidelines.

RESPONSE

For overcooling conditions, the Small Break Guidelines will allow an RCP to be restarted in each loop if the required conditions are met. The operation of the RCP removes the OTSG level requirement for natural circulation. The CR-3 Small Break Guidelines have similar requirements as ANO-1.

The Emergency Procedures and Abnormal Procedure will be revised to provide the operator with a means of making a decision on what is causing ESFAS actuation: either a LOCA or a overcooling transient.

The revised Small Break Guide will be submitted to the NRC for approval prior to implementation.

ITEM 12

Review procedures concerning when to initiate HPI cooling upon total loss of secondary heat removal capability.

RESPONSE

Installation of EFW flow indicators was accomplished during the Summer 1979 outage. Should a LOFW include the EFW, as seen on the EFW flow indicators, and EFW is not capable of being started, then HPI cooling should be initiated. However, if OTSG level is above the level limit and steam pressure is greater than 600 psi then efforts should be made to manually start EFW. However, HPI cooling should only be used if EFW and MFW are lost and not regainable.

ITEM 12 (Cont'd)

RESPONSE

The operator will be given guidelines as to what OTSG pressure and level can be before HPI cooling is initiated. Prior to this time, he will be instructed to attempt to start EFW.

ITEM 13

Provide procedures and training for recovery from EFW actuation to avoid OTSG overfill.

RESPONSE

A section entitled "Recovery from Emergency Feedwater Actuation" is being incorporated into EP-108 "Loss of Steam Generator Feed" and will be completed before startup.

Emergency Procedure EP-113 "Plant Shutdown from Outside Control Room" will be revised to include adequate operator verification of proper feedwater water system status prior to leaving the control room.

Emergency Procedure EP-105 "Steam Supply Rupture" will be revised to state that Main Steam Isolation Valves will not automatically open. Also, new valve closing instructions will be developed for the operator.

Emergency Procedure EP-101 "Unit Blackout" required changes are covered in Item 30.

Abnomal Procedure AP-112 "Loss of Electrical Supplies" will be revised to recognize loss of the startup transformers as unit blackout.

Emergency Procedure EP-108 "Loss of Steam Generator Feed" will be revised to verify feedwater valve status earlier in the response.

ITEM 14

Establish minimum conditions for voluntarily entering degraded modes of operation.

RESPONSE

Administrative Instruction 500 "Conduct of Operations" has been revised to provide the following guidelines:

1. Plant stable and under control within existing equipment, procedures and personnel capability.

2. Surveillance of redundant equipment will be determined to be operable before removing the degraded equipment from service.

ITEM 14 (Cont'd)

RESPONSE

3. Additional compensating measures shall be considered (i.e., dedicated operator, etc.).

4. Any equipment out of service that causes entry into an action statement of Technical Specifications shall be worked around the clock with all resources necessary to repair the equipment and place it back in service in the shortest possible time.

5. If an emergency diesel is to be taken out of service, no equipment in the opposite Emergency Safeguards train can be out of service.

6. If "A" emergency diesel is taken out of service, the steam-driven EF pump shall be determined to be operable.

ITEM 15

Revise procedures to require ICS Rod withdrawal inhibit reset upon RPS reset.

RESPONSE

Procedures will be revised to require ICS Rod withdrawal inhibit reset upon RPS reset prior to restart.

ITEM 16

Train all operators and I&C technicians in response to NNI and ICS failures.

RESPONSE

The Training Department at CR-3 will work with engineering to develop a course to be presented to all licensed personnel and technicians. The course will include as a minimum:

- A review of NNI/ICS, including all proposed modificaions.

- A review of what indications are available to the operator during system upsets and NNI/ICS power losses.

- A review of control system interactions caused by NNI/ICS power losses.

- A review of Emergency Procedures and Abnormal Procedures necessary to shut down the plant with emphasis on how to regain manual control during NNI/ICS power losses.

ITEM 16 (Cont'd)

RESPONSE

- A review of all NNI/ICS event trees developed by the Nuclear Safety Task Force.

ITEM 17

FPC shall train operators on CR-3 sequence of events, concentrating on ICS response to failed NNI and how lessons learned from TMI-2 affected transient and on plant changes that are being made as a result of the CR-3 event.

RESPONSE

The Training Department at CR-3 will work with engineering to develop a course to be presented to all licensed operators. This course will include as a minimum:

- Training on all modifications that result from the 2/26/80 incident.
- Training on how modifications will affect plant response and control.
- Training on all procedure changes that occur as a result of these modifications.
- The operators will also be trained on the sequence of events, ICS response to failed NNI, and how lessons learned from TMI-2 affected the transient.

ITEM 18

Repair Events Recorder System.

RESPONSE

Analysis of the Events Recorder System following the February 26, 1980 trip of CR-3, revealed that if more than 16 events occurred simultaneously, everything beyond 16 was lost. Subsequent troubleshooting isolated the cause of this to a defective Sequential Memory module connector problem. The problem has been corrected and work initiated to install equipment necessary for periodically exercising the full capabilities of the Sequential Memory circuitry. Surveillance Procedure SP-505 referenced in Item 3 will be used to check the operability of the Events Recorder.

ITEM 19

Review the present design of computer alarm printout to help eliminate overload and printout time delay.

RESPONSE

Three deficiencies were considered in the responsiveness of the plant computer system during plant transients:

1. Lack of timeliness with which alarms are printed.
2. Large number of useless alarms.
3. Probability of completely filling the alarm buffer during longer plant transients.

The importance of the computer as an alarming device for situations requiring prompt action was weighed against the need for recording as much data as possible for later analysis.

It was decided that the alarm printout is more important as a data logger than as an operator tool during transients. Thus, Part 2 above is insignificant and improvements need only be made in the area of Parts 1 and 3. Therefore the system program has been modified so that all alarms will be automatically diverted to the line printer upon plant trip.

ITEM 20

Adjust secondary steam relief valve blowdown settings.

RESPONSE

B&W analysis and changes in the main steam line relief valve settings at the Davis-Besse Plant have shown that a less than 5% blowdown following a reactor trip can significantly increase the probability of the pressurizer level remaining on scale. Therefore, the main steam line relief valve settings at Crystal River will be modified to obtain less than 5% blowdown following a reactor trip in an effort to maintain the pressurizer level on scale.

ITEM 21

Provide around-the-clock I&C technician coverage.

RESPONSE

Around-the-clock I&C coverage will be established as soon as additional people can be hired and trained. We are looking for I&C technicians for the special maintenance crew. Until the special maintenance crew can be formed, we will use the premium payment clause of the contract and the regular crew. It is planned to have all shifts covered when we start up, if possible.

ITEM 22

Perform corrective action regarding potential shorting in safety system from improperly installed fiber clamp.

RESPONSE

This item will be completed before startup following the current refueling outage.

ITEM 23

Provide temporary backup air system for main feedwater startup control valves.

RESPONSE

A portable diesel-driven air compressor will be utilized as a temporary backup air system for main feedwater startup control valves until an evaluation can be made for a long-term upgrade. This will provide greater instrument air system reliability for controlling of MFW and EFW valves during a loss of offsite power.

ITEM 24

Review and revise procedures for immediate operator action to trip DH pumps upon spurious closing of the DH dropline valves.

RESPONSE

The Decay Heat pumps will be tripped anytime their suction supply valves are not in the open position. This will minimize pump damage due to cavitation. Operating Procedure OP-404 and Emergency Procedure EP-112 are being revised to provide these procedures.

ITEM 25

FPC management shall perform an evaluation of the number of exempt personnel that should hold operators licenses.

RESPONSE

The concern is a walkout of Bargaining Unit licensed operators during an incident, and management having to operate the plant. Since this is a concern whenever union contract negotiations occur, an evaluation has been completed. The results are:

1. To operate the plant without Bargaining Unit personnel required 3 shifts of 8 operators of which one must have an SRO

ITEM 25 (Cont'd)

RESPONSE

and 2 must have an RO license (i.e., a total of 9 licensed exempt personnel).

2. Exempt licensed personnel available:

   7 Shift Supervisors
   6 Asst. Shift Supervisors
   1 Operations Superintendent
   1 Operations Engineer
   1 Manager of Training
   3 Training Instructors
   1 Maintenance Superintendent
   1 Maintenance Staff Engineer
   1 Maintenance Planner
   1 Technical Services Superintendents
   23 Total (which is inclusive of the compliment required in 1 above)

3. There are approximately 15 exempt maintenance and technical support personnel that are not licensed but are capable of performing nonlicensed operator duties.

ITEM 26

Establish administrative controls to minimize access to containment in Mode 1.

RESPONSE

The procedure for administrative control of containment entry will be revised prior to startup.

ITEM 27

Include in operator training and plant procedures methods of isolating letdown and makeup in the event of loss of ICS or NNI power supplies.

RESPONSE

The operator can isolate letdown from the control room by closure of makeup system valve MUV-49 or valves MUV-40 and MUV-41. These valves are the system containment isolation valves and are powered by Engineered Safeguard power which is independent of NNI and ICS power supplies.

Operators will be advised by training and procedure that these valves are available to isolate letdown in the event of a loss of NNI or ICS power supplies.

ITEM 28

Provide capability to facilitate operator action in event of loss of power to the ICS which results in a spurious interlock precluding restart of reactor coolant pumps.

RESPONSE

Loss of power to the ICS causes the ICS "Reactor Power Greater Than 22%" relay to operate, which is a start permissive interlock for the reactor coolant pumps.

The RCP start logic will be modified to provide bypass of permissive interlocks in an emergency condition. This bypass will be under administrative control with a key lock switch for each pump and will be located on the main control board.

ITEM 29

Visually inspect AFW nozzle collars and repair as required.

RESPONSE

The AFW nozzle collars inspection and repair was completed on April 29, 1980.

ITEM 30

Review restart of critical items on loss of offsite power without ESFAS actuation and revise applicable procedures.

RESPONSE

We are presently performing a review to determine which critical items are affected by a loss of offsite power without ESFAS actuation. This review will be completed and all necessary modifications and procedure changes will be implemented prior to startup.

ITEM 31

Field changes to NNI/ICS Systems should be performed in accordance with design control requirements.

RESPONSE

FPC will develop and/or revise procedures prior to restart to implement this item as part of the design control system. FPC presently has a design control system for modifications to safety-related systems. The NNI/ICS are not safety-related systems per our definition. However, due to their importance, we are including these systems in our design control program. This revision to our program will insure adequate design review and control over modifications to this system consistent with safety-related changes that are made at CR-3.

ITEM 32

NNI/ICS changes should include specific reference(s) to installation and maintenance precautions identified by the equipment supplier.

RESPONSE

FPC will develop procedures prior to restart to implement this item as part of the design control system.

ITEM 33

Check 820 signal monitor output for seal-in problems.

RESPONSE

A Bailey 820 signal monitor output seal-in caused the PORV to open in the February 26 incident. We are reviewing other NNI/ICS circuits to determine if similar problems exist and implementing corrective action as required prior to startup.

ITEM 34

Provide operator with redundant indications of main plant parameters.

RESPONSE

The main plant parameter NNI system instrumentation consists of signals originated at 120 VAC-powered transmitters, signal conditioning by 24 VDC-powered electronic modules, and display by 120 VAC-powered indicators or recorders. NNI(X) and NNI(Y) supply all three types of power. The signal conditioning modules were mixed in instances between NNI(X) and NNI(Y). These vital instrument loops will be revised to achieve separation from the transmitter through conditioning to new indicators. These new indicators are self-powered and require no 120 VAC for operation and are an addition to existing main control board (MCB) indicators.

The new indicators with NNI "X" and "Y" power availability lights will be located on the ICS portion of the MCB above the existing annunciator windows. Attachment 1 details location of these new indicators and NNI instrument loop design criteria to assure the availability of vital indication in the event of a power failure.

ITEM 35

Provide override closure of atmospheric dump valves upon loss of ICS power.

ITEM 35 (Cont'd)

RESPONSE

Upon loss of ICS power, the atmospheric dump valves fail mid-position. A circuit modification will be made prior to startup to provide automatic closure of these valves upon loss of ICS power to preclude uncontrolled secondary side blowdown. This new circuitry will, upon loss of ICS power, automatically vent control air to these pneumatic-operated valves causing valve closure.

ITEM 36

Provide automatic bus transfer switches for NNI (X) and ICS AC power, normally on a vital bus with auto transfer to regulated instrument bus. Dual 24 VDC supplies will be powered from a vital bus and regulated instrument bus.

RESPONSE

Loss of 120 VAC power causes both indication and control losses in the NNI/ICS systems. We will install automatic transfer switches for the NNI (X) and ICS AC power with the vital bus as the preferred supply and transfer to the regulated instrument bus upon loss of vital bus failure. See Attachment 2 for further clarification.

ITEM 37

Provide subcooling monitors with reliable backup. One monitor shall be operable on loss of either inverter A, B, C, or D or offsite power.

RESPONSE

Dual subcooling monitors at CR-3 provide redundant indication of saturation margin. Loss of power due to either an inverter or offsite failure would not cause loss of both subcooling meters. Our subcooling monitor power and signal sources have been verified to provide this reliability.

Details concerning the design of the subcooling monitors at CR-3 were provided in our January 11, 1980 response regarding NUREG-0578.

ITEM 38

Provide positive indication of all three pressurizer relief valves.

RESPONSE

In direct response to NUREG-0578, Item 2.1.3.a, FPC has purchased from Babcock & Wilcox a Valve Monitoring System. This system incorporates acoustical monitoring techniques to provide the reactor operator with indication of flow through the valve. The equipment is very similar to the existing Loose Parts Monitoring System supplied by Babcock & Wilcox.

ITEM 38 (Cont'd)

RESPONSE

This design provides for two transducers mounted on each safety valve and the PORV. Each of these transducers will be wired from the containment to the PORV/T_{sat} monitoring cabinet, located in the 4160 V SWGR Room. Within this cabinet will be three channels (one for each valve) of signal conditioning with local indication, alarm, and selectable audio monitor. Only one transducer will be normally monitored on each valve. The other is manually selectable for comparison of performance or in the event of transducer failure. Each channel will also provide remote analog indication and annunciator events recorder high alarm functions. This analog indicator for each channel will be mounted on the ICS section of the main control board. A common annunciator window will also be located on this section. The events recorder will provide CRT and hard copy indication of valves that actuate.

The valve monitoring/T_{sat} cabinet will be powered from a vital source with all cable routing meeting seismic requirements. Florida Power Corporation is participating with Babcock & Wilcox in a generic program to environmentally qualify the valve monitoring equipment. This program is presently scheduled to be completed by October, 1980.

ITEM 39

Modify vital bus panels for quick fuse replacement.

RESPONSE

The replacement of any vital bus fuse requires the removal of a panel front including the access door to the breaker. This will be revised such that this entire panel front is hinged for quick access for replacement.

ITEM 40

Modify EF pump auto start circuit and reactor trip circuit so that any power failure will not prevent actuation on low SG level (control grade).

RESPONSE

The control grade emergency feedwater pump auto start and anticipatory reactor trip on low-low steam generator level will be modified to provide Auto Start and reactor trip during any single power failure.

ITEM 41

PORV valve position indicating lights for solenoid will be added.

ITEM 41 (Cont'd)

RESPONSE

PORV position indicating lights indicate selector switch position, high pressure setpoint select, and low pressure setpoint select. These indicating lights will be modified to provide indication that opening has been commanded. This, combined with the new acoustical monitors, and the existing pressure setpoint indication will provide the operator with the capability to cross check indications for verification of PORV position.

ITEM 42

A fatigue analysis will be performed on the pressurizer heaters to demonstrate that the 40-year design life was not adversely impacted by this treatment.

RESPONSE

During the February 26, 1980 transient experienced at Crystal River 3, it was postulated that some of the pressurizer heaters may have been exposed to a saturated steam environment and concurrently energized (Ref. Transient Assessment Report). B&W recommended that the heaters be electrically checked for continuity, the bundles inspected for possible leakage and a 40 year fatigue analysis performed. Althougth it was not the intent of B&W, FPC committed to the NRC to perform the fatigue analysis prior to restart. B&W now believes this fatigue analysis is not necessary because it has been confirmed that the heater bundles were not uncovered. Therefore, Florida Power Corporation will not pursue this analysis any further.

ITEM 43

The relief valve loadings on the pressurizer relief nozzles should be determined and their effect will be assessed.

RESPONSE

This has been accomplished by the GAI calculations (see Item 8) which demonstrated that piping loads were within design considerations and as such require no further analysis of pressurizer relief nozzles.

ITEM 44

Provide diverse containment isolation.

ITEM 44 (Cont'd)

RESPONSE

Florida Power Corporation, in its April 12, 1979 response to Item 6 of IE Bulletin 79-05A, identified essential and nonessential systems with regard to containment isolation and core cooling. Essential systems were defined as those systems which are required for core cooling capability and, therefore, should not be automatically isolated on HPI actuation. For the valves listed in our April 12 response, which receive no ES signal and are normally closed and remain closed following the accident conditions, no further action is required.

The nonessential valves, listed in our response, which receive a containment isolation signal (4 psig RB pressure) will be provided with a diverse containment isolation parameter by the addition of an auto-close isolation signal, based on automatic HPI actuation. This diverse containment isolation signal will satisfy safety-grade requirements and resetting of HPI will not result in the automatic loss of containment isolation.

ITEM 45

Electrically interlock motor-driven EFW pump to start on loss of MFW.

RESPONSE

The motor-driven EFW pump is presently interlocked to start on loss of main feedwater provided that offsite power is available. The emergency diesel generator block loading sequence will be modified to include the auto start of the motor-driven emergency feedwater pump when offsite power is not available.

Details of this design change are included in our May 2, 1980 submittal to Mr. Robert Reid of the NRC staff.

ITEM 46

Review PORV circuitry to assure that credible power failures do not cause the PORV to open when it is not required to open. Review power supply independence between PORV and PORV isolation block valve to assure that failure which would affect PORV does not eliminate the possibility of PORV isolation block action.

RESPONSE

The PORV circuitry will be modified to include a loss of NNI power interlock to interrupt 125 VDC control power to the PORV solenoid. Loss

ITEM 46 (Cont'd)

RESPONSE

of either NNI(X) 120 VAC or 24 VDC will cause the auxiliary relay to de-energize and block the automatic opening circuit of the PORV.

The power source to the PORV and block valve are independent. The power supply to the PORV solenoid is 125 VDC, while the block valve is a motor-operated valve, fed from a 480 VAC Engineered Safeguards motor control center.

The control switch for the PORV will be relocated to the Main Control Board. This switch will have auto (NNI control), open and close positions. Positions override NNI commands and do not rely on NNI power.

ITEM 47

Modify Pressurizer Spray valve so that NNI power failure will close the valve.

RESPONSE

The Pressurizer Spray Valve will be modified to include an interlock from the auxiliary relay described above for the PORV. Loss of NNI(X) AC or DC power will cause the spray valve to close. The operator will be able to manually operate the valve from the main control board to either open or close the valve. Failure of the NNI(X) power will not prevent manual control.

ITEM 48

Move 120 VAC ICS(X) power to Vital Bus.

RESPONSE

The ICS(X) 120 VAC power supply is presently fed from a regulated power source. A change to incorporate an Automatic Transfer Switch (ATS) between a vital bus power source and a regulated source is being provided.

This scheme provides reliable power to the ICS (see Item 36 for further clarification of this modification).

ITEM 49

Annunciate loss of power to NNI(X) and NNI(Y) buses and ICS.

ITEM 49 (Cont'd)

RESPONSE

There were no provisions to independently annunciate the loss of ICS, NNI(X) or NNI(Y) 120 VAC and 24 VDC power sources. Separate annunciator windows will be provided to inform the operator of a loss of the NNI(X), NNI(Y), or ICS power sources. In addition, power available indicating lights are provided on the main control board adjacent to the added indicators such that the absence of a light indicates the adjacent indicators are without power.

ITEM 50

Install indicating lights on all vital bus feeds.

RESPONSE

An indicating light for each branch circuit supplied from safety-related vital distribution panels will be provided. The indicating lamps and isolating fuses will be installed in a box which will be mounted near the distribution panels and easily visible to control room personnel. There is a separate box for each distribution panel.

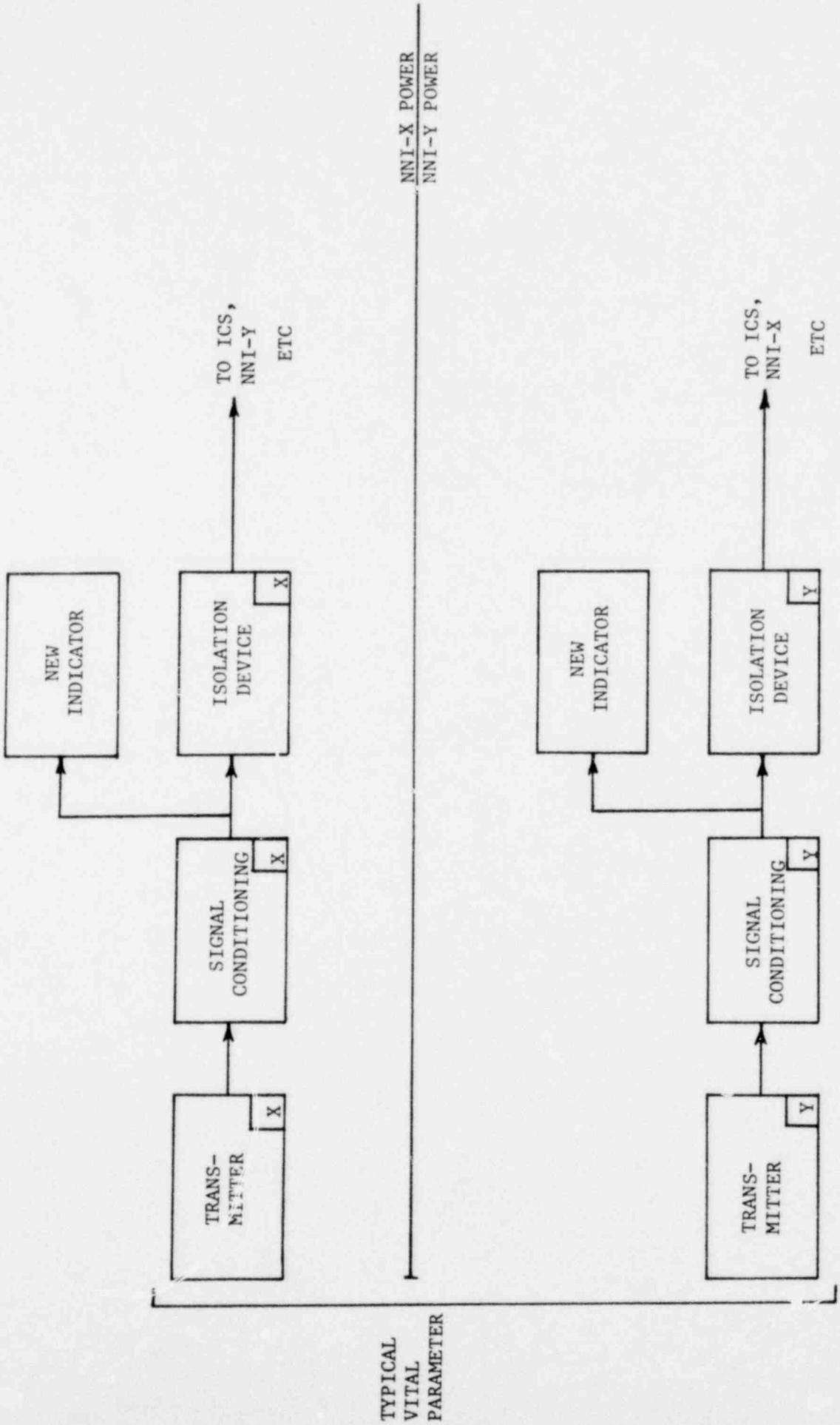
ITEM 51

Evaluate the OTSG Rupture Matrix with the intent to remove the signal from FWV-161 and FWV-162 to assure a passive EF flow path to both OTSGs on initiation of EFW.

RESPONSE

The CR-3 steam line rupture matrix presently isolates FWV-161 and FWV-162 upon actuation. Concern is that this isolation of emergency feedwater may not be in the best interest of overall plant safety. This concern is based upon the higher industry operating experiences of actuation due to generator dry-out or RC system cooldown rather than a steam line rupture.

The protection afforded by isolation of EFW should be evaluated against the increased reliability of EFW by not isolating EFW. This evaluation is being performed by GAI and B&W and will be submitted for NRC review upon completion. If the evaluation supports not isolating EFW via the rupture matrix, a minor control change will remove these valves from the steam line rupture matrix actuation logic.


NNI-X POWER
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CONDITIONING
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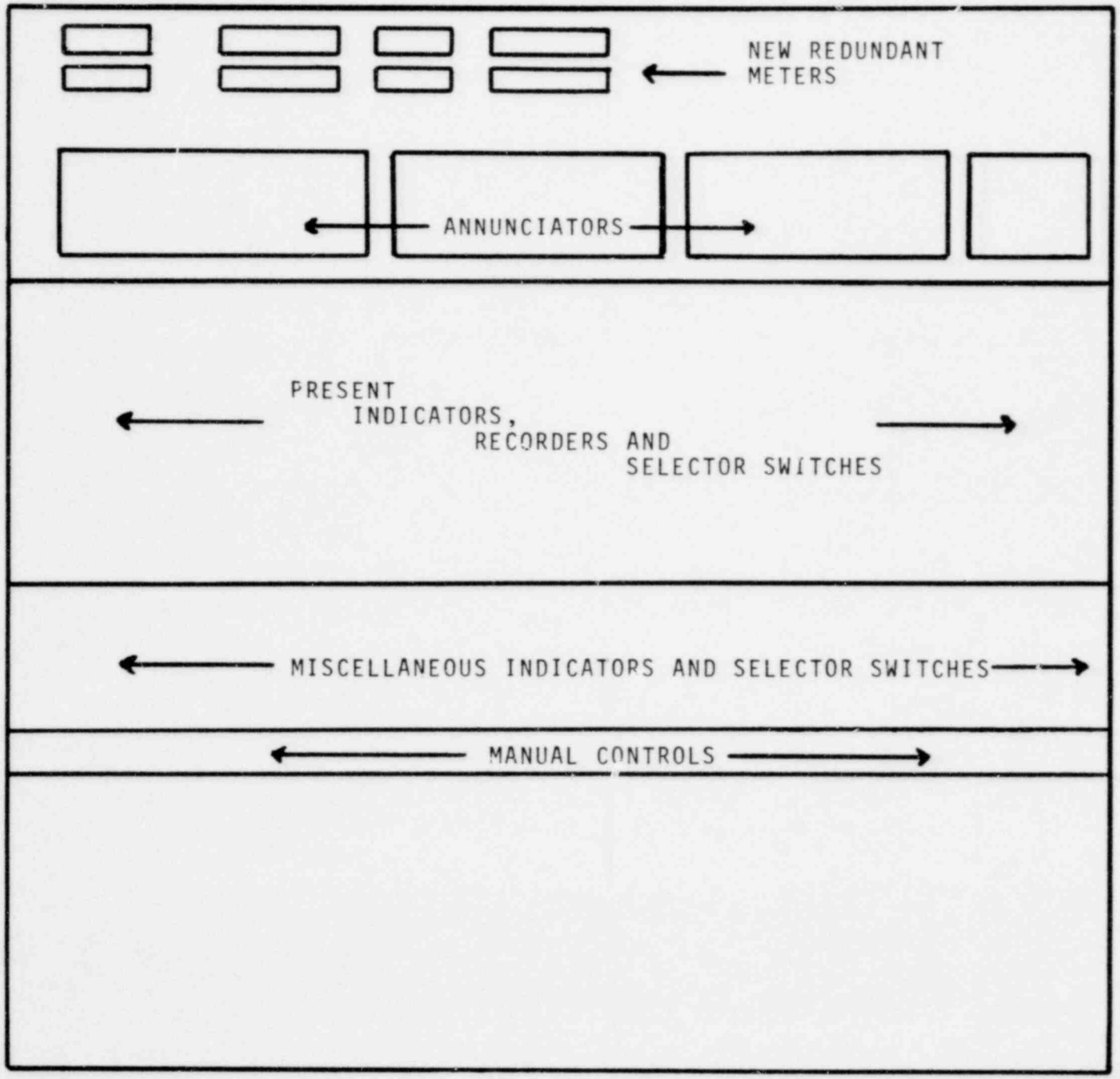
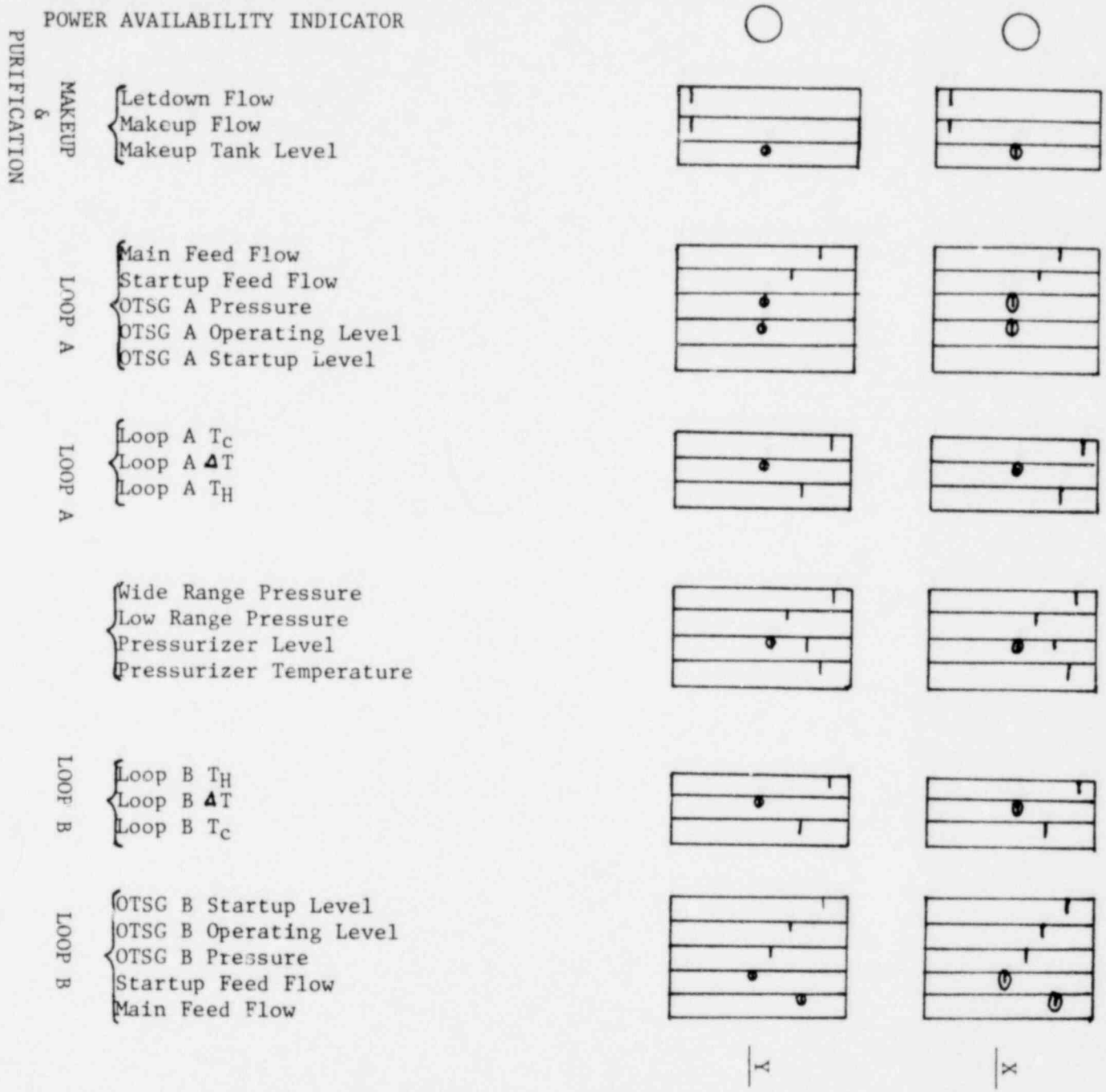
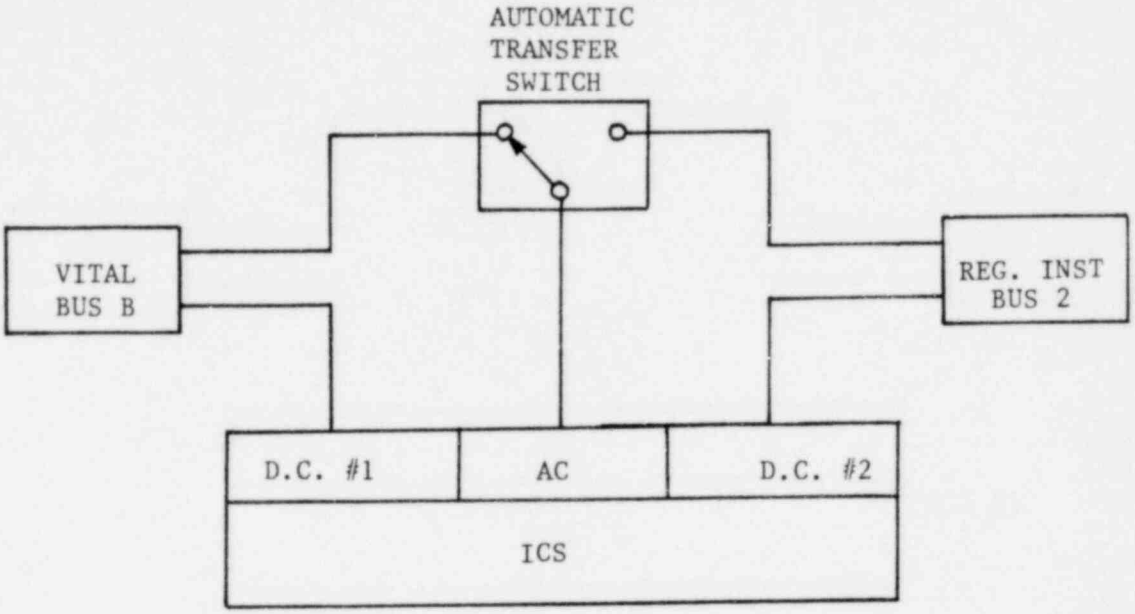

NEW REDUNDANT
METERS
ANNUNCIATORS
PRESENT
INDICATORS,
RECORDERS AND
SELECTOR SWITCHES
MISCELLANEOUS INDICATORS AND SELECTOR SWITCHES
MANUAL CONTROLS

POWER AVAILABILITY INDICATOR

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BUS 2
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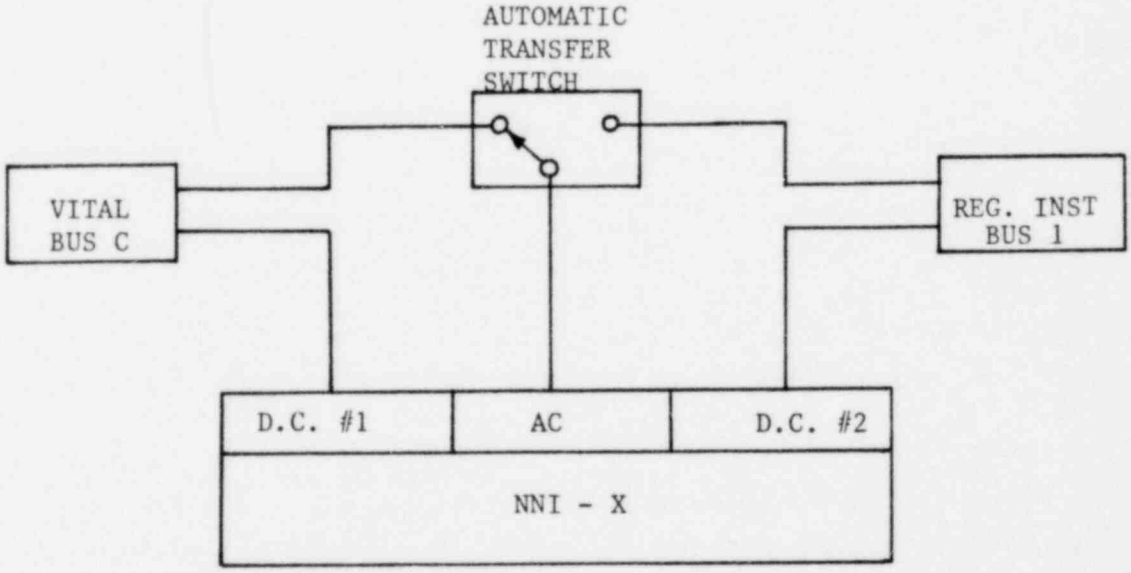
D.C. #2
ICS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TRANSFER
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BUS 1
D.C. #1
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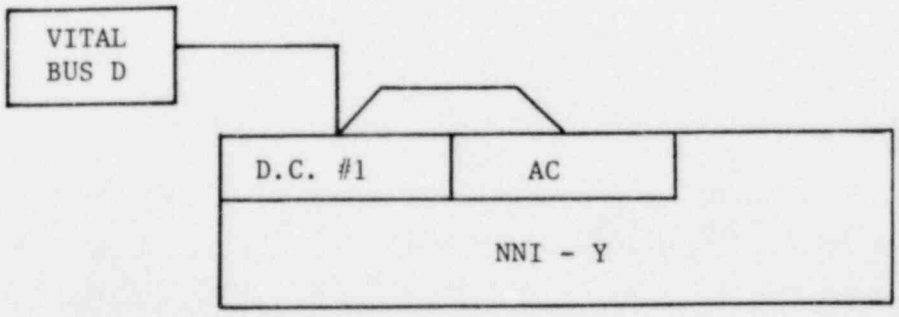
D.C. #2
NNI - X


VITAL
BUS D
D.C. #1
AC
NNI - Y

ENCLOSURE C

The response to items in the B&W Recommendations, NUREG-0667 Recommendations, and INPO/NSAC Recommendations that were not addressed by the Nuclear Safety Task Force Priority Items (Enclosure B) are discussed in this enclosure.

B&W RECOMMENDATIONS NOT COVERED ON TASK FORCE LIST

ITEM 5

The failure of NNI/ICS power supplies shall not prevent safety or protection systems from operating or prevent manual override of safety or protection systems.

RESPONSE

The NNI/ICS supplies no signals to the RPS or ES systems, nor can ICS or NNI failures affect ES through signals supplied to the NNI. Therefore, this concern does not apply to CR-3. Additionally, there is no RPS override. The ESF channel override has no connection with the NNI/ICS systems.

ITEM 9

A loss of an NNI or ICS power supply shall not cause the pressurizer heater to fail on, or remain on, when the pressurizer level is low.

RESPONSE

Loss of NNI power could cause heater groups 7 through 13 to come on. This is not considered to be a safety concern and as such not a startup restraint. The problem of heater burnout on low pressurizer level has been evaluated, and it was determined that heater well failure does not occur before the heating element fails and terminates the temperature increase. Therefore RC system integrity will not be unacceptably degraded during this incident.

ITEM 10

Seal injection and return (as required to prevent RCP seal damage) to the Reactor Coolant Pump shall be maintained upon an NNI/ICS power failure.

RESPONSE

This item requires additional B&W study before implementation and is not considered to be an immediate safety concern. Therefore, this is a long term consideration and not a startup restraint.

ITEM 14

Each utility should perform a management review to assure that proper alteration and maintenance practices are in place.

RESPONSE

The Nuclear Safety Review Task Force has recommended the Nuclear General Review Committee of FPC direct a review/audit of existing procedures concerning change and maintenance practices.

Klein(NRC5280)DN90

B&W RECOMMENDATIONS NOT COVERED ON TASK FORCE LIST

ITEM 14 (Cont'd)

RESPONSE

Since the Quality Programs Department (QPD) personnel are familiar with the use of the existing procedures, they will perform this audit for the NGRC. The Quality Programs Department will submit a final report to the NGRC for committee action. From this QPD Report, the NGRC will ascertain key areas that could be improved or corrected.

NUREG-0667 RECOMMENDATIONS NOT COVERED ON THE TASK FORCE LIST

ITEM 3

Addition of a motor-driven AFW pump for Davis-Besse.

RESPONSE

This recommendation does not apply to CR-3.

ITEM 7

Increased usage of incore thermocouples:

RESPONSE

Each T_{sat} meter, as presently installed, has two hot leg and two cold leg temperature inputs and a pressure input from each loop.

In addition to the above temperature inputs, FPC is also providing, independently, the hottest of 5 core exit T/Cs to each T_{sat} meter. These T/Cs are selected from each core quadrant and the central region. This addition will give the operator the capability to selectively observe saturation margin of the hottest T/C, T_h or T_c against pressure in either loop. The alarm on reduced saturation margin will be from the hottest temperature input and selected loop pressure. Installation of the modification is scheduled to be completed prior to restart.

CR-3 also has automatic printout of all incore thermocouples on activation of high temperature setpoint. FPC is working on a modification to automatically initiate this printout on reactor trip.

ITEM 8

High radiation signal initiation of containment isolation.

RESPONSE

Containment purge isolation at CR-3 is presently actuated upon high radiation, 4 psig RB pressure or HPI actuation. The RB pressure and HPI actuation systems are redundant and safety-grade. FPC is presently performing an evaluation of the purge system at CR-3 in response to NRC questions contained in Mr. Reid's letter of February 29, 1980. Upon completion of our evaluation, FPC will identify any modifications necessary to permit continuous purging at CR-3.

NUREG-0667 RECOMMENDATIONS NOT COVERED ON THE TASK FORCE LIST
(Continued)

ITEM 11

Modifications to eliminate immediate manual actions for emergency procedures.

RESPONSE

The design philosophy of CR-3 relies on well-trained, intelligent operators as the most effective means of responding to unforeseen situations. Where the time requirement for operator action is too short, automatic activation modifications have been made. A long term study is required in order to establish additional modification requirements.

ITEM 14

Guidelines for loss of NNI/ICS.

RESPONSE

This item requires B&W evaluation. However, FPC is developing plant procedures for the operators to follow in the event of NNI/ICS losses. Training in these procedures will be completed prior to startup after this refueling outage.

ITEM 15

Mandatory one-week simulator training for operators as part of requalification program.

RESPONSE

This recommendation was made a part of the requalification program at Crystal River Unit 3 in 1979.

ITEM 16

Evaluation of RCP restart criteria.

RESPONSE

This item was referred to the NRC staff for evaluation.

NUREG-0667 RECOMMENDATIONS NOT COVERED ON THE TASK FORCE LIST
(Continued)

ITEM 17

Alternative solution to PORV unreliability/safety system challenge rate concerns.

RESPONSE

This item was referred to the NRC staff for evaluation.

ITEM 18

IREP Crystal River study.

RESPONSE

This item was referred to the NRC staff for evaluation.

ITEM 19

Performance criteria for anticipated transients.

RESPONSE

This item was referred to the NRC staff for evaluation.

ITEM 20

Continued evaluation of need to trip RCPs during small break loss-of-coolant accidents.

RESPONSE

This item was referred to the NRC staff for evaluation.

ITEM 21

Reevaluate location of AFW injection into OTSG.

RESPONSE

This item requires B&W evaluation and is a long term study.

NUREG-0667 RECOMMENDATIONS NOT COVERED ON THE TASK FORCE LIST
(Continued)

ITEM 22

Staff study of personnel-related LERs with respect to high number for B&W plants.

RESPONSE

This item was referred to the NRC staff for evaluation.

An assessment of licensed operator errors was completed in support of the Crystal River - Unit 3 Nuclear Safety Task Force by P. E. Dietz, Institute for Nuclear Power Operations, in his May 1, 1980, letter to Dr. P. Y. Baynard.

Recommendation 22 of the DRAFT NUREG-0667, April 2, 1980, Transient Response of Babcock and Wilcox - Designed Reactors, urges the performance of an analysis of the number of licensee event reports attributed to licensed personnel error to determine the significance and cause of the higher number associated with the operation of B&W facilities. As noted in the report, LERs have only been categorized by licensed personnel error since January 1978.

The following table is from the NRC report and covers LERs between January 1978 and January 1980.

TABLE 1

LER OUTPUT ON LICENSED OPERATOR

EVENTS FOR 1978 and 1979

VENDOR	TOTAL PLANTS	LERs	AVERAGE/PLANT
B & W	9	58	6.44
CE	8	45	5.63
GE	25	142	5.68
W	25	131	5.24

RESPONSE (Cont'd)

To address the expressed concern, INPO performed an independent analysis of the LER data. A search of the LER Data Base provided by NRC yielded the following information on licensed operator errors between January 1978 and March 1980.

TABLE 2

LER OUTPUT ON LICENSED OPERATOR

EVENTS BETWEEN JANUARY 1978

AND MARCH 1980

VENDOR	TOTAL PLANTS	LERs	AVERAGE/PLANT
B & W	9	70	7.8
CE	8	53	6.6
GE	25	151	6.0
W	25	146	5.8

A cursory review of the data might indicate that licensed operators generate more LERs at B & W plants than at the other vendor-designed facilities. As noted in NUREG-0667 the reported LERs decreased significantly with age of the plant, those having already undergone the first several years break-in period generally submitting the fewest LERs. Accordingly, INPO attempted to analyze the data in such a way as to remove the bias caused by plant maturity. One approach was to eliminate from consideration plants more mature than any B&W plant. The starting date for each vendor's first plant is shown in Table 3.

TABLE 3

YEAR FIRST PLANT TAKEN CRITICAL

VENDOR	YEAR
B & W	1973
CE	1971
GE	1959
W	1960

If only the plants that have been taken critical for the first time since 1973 are considered the B & W plants no longer generate the most licensed operator errors.

TABLE 4

LICENSED OPERATOR ERRORS AT

PLANTS CRITICAL FOR THE

FIRST TIME SINCE JANUARY 1973

VENDOR	TOTAL PLANTS	LERs	AVERAGE/PLANT
B & W	9	70	7.8
CE	6	30	5.0
GE	12	100	8.3
W	16	119	7.4

Analyzing only LERs since 1978, is like taking a time exposure picture of operating history. By looking at a single time period, one can see all plant data affected by the same industry wide influences such as the TMI-2 accident. The number of LERs varies with plant age. No one vendor-designed plant has a tendency to generate more licensed operator errors (LERs) than the others. In fact the data fluctuations for both GE and Westinghouse plants are larger than for B & W plants, and each bounds the B & W fluctuation.

INPO concludes that the LER data does not support a concern that the error rate for licensed operators at B & W plants is greater than for other nuclear plants, once the bias due to plant maturity has been removed.

The INPO letter will be forwarded to the NRC upon receipt. It will include graphic representaions for clarity of conclusions.

INPO/NSAC CONCERNS NOT COVERED ON THE TASK FORCE LIST

ITEM I.A

Procedural requirement for declaration of appropriate emergencies should be emphasized in plant training sessions.

RESPONSE

This is covered during :ocemse Training and Requalification Training at CR-3.

ITEM I.C

Instrument technician work practices and their potential impact on plant safety should be reviewed in plant training sessions. Attention should be given to events similar to the 3/20/78 and 1/5/79 transients at the Rancho Seco plant where overcooling resulted from maintenance technician actions.

RESPONSE

This is covered during Craft Systems Training at CR-3 by implementing a revised program that includes the above transients.

ITEM II.D

The industry should further analyze and resolve with the NRC the current reactor coolant pump trip procedures to be followed during a small break LOCA. Mandated procedures can be counterproductive to safety if they are not sufficiently discriminating to specific circumstances, and to specific plant designs.

RESPONSE

This item, which is also covered by NUREG-0667 Recommendation No. 20 requires further analysis to resolve the concern.

ITEM II.F

Procedures for declaration of emergencies should be reviewed to determine if responsibility for monitoring plant conditions which lead to declaration of a specific emergency category should be assigned to a specific individual. It is suggested that this individual would also be responsible for immediately informing the senior person in charge at the time when these conditions for emergencies and emergency notification have been met.

RESPONSE

The Emergency Coordinator is responsible for the declaration of a specific emergency category based on plant conditions. The Shift Supervisor acts as the temporary Emergency Coordinator until the Nuclear

ITEM II.F (Cont'd)

RESPONSE

Plant Manager or his designee assumes the role as the Emergency Coordinator. The temporary Emergency Coordinator is responsible for immediately notifying the Nuclear Plant Manager or his designee when an emergency has been declared.

ITEM III.A.4

Susceptibility of control systems to incorrect information caused by electrical faults, e.g., choking off feedwater to steam generators, withdrawing rods, and opening the turbine throttle.

RESPONSE

This is a long term item which requires additional analysis.

ITEM III.B.1.C

The system monitoring the incore temperatures automatically prints any temperature which indicates in excess of 700°F. The basis for selecting 700°F should be reviewed to determine if this number should be revised, since data was lost during the transient.

RESPONSE

The automatic thermocouple printout is for information only and is not safety related. A review of the printer start logic and setpoints is underway. FPC's action to resolve this concern is described in our response to NUREG-0667 Recommendation No. 7 of this submittal.

ITEM III.B.2

Plant transient monitoring and recording. Plant transient records independent of process computer, to provide a tape record of main plant parameters, are desirable for all plants. They are desirable on an earliest practicable schedule.

RESPONSE

FPC intends to install the B&W RECALL System equipment as part of the Technical Support Center upgrade per NUREG-0578. The RECALL System records 12 hours of 160 analog and 65 digital inputs and writes over the previous record continuously. Upon an incident signal, such as Reactor Trip, RECALL will record the next 11 hours of data but will retain the hour prior to the incident signal. Additional details of this system will be provided per our discussions with the NRC staff concerning our implementation of NUREG-0578.