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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

June 22, 1979

Mr James G Keppler
Director - Region III
Office of Inspection and Enforcement
United States Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Mr Keppler:

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Sockets No. 50-282 and No. 50-306

In response to Items 1 through 13 of IE Bulletin 79-06A - Revision 1, and a subsequent NRC request for additional information, the following is being submitted as a revision to our initial response:

Item 1

All licensed operators and plant management and supervisors with operational responsibilities at the Prairie Island Nuclear Plant, with the exception of three individuals, have attended a presentation of the Three Mile Island - Unit 2 (TMI-2) accident of March 28, 1979. This presentation was audio taped and was presented to the three individuals who did not attend.

The presentation covered the chronological sequence of events for the first 16 hours after the accident at TMI-2, review of IE Bulletin 79-06A with emphasis on Items 1.a and 1.b, and a question and answer session. Attendance was taken at these sessions, and records are being maintained in the Training Section Files.

Item 2

The emergency procedures for coping with transients and accidents have been reviewed and revised to recognize the possibility of forming voids in the Reactor Coolant System. The emergency procedures contain specific instructions that mitigate the consequences of void formation and enhance natural circulation. A training memo has been developed to inform the operators of the basis for the instructions in the procedures and warns of void formation. The training memo also describes what instruments to monitor to detect the occurrence of voids. Because we believe this material would "clutter" the action statements contained in the actual procedures, we believe the material belongs in a "basis training" document. The items below address the specific requests contained in the June 7 draft request for additional information working paper.

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NORTHERN STATES POWER COMPANY

Mr James G Keppler
Page 2
June 22, 1979

- A. A training memo has been sent to all NRC licensed plant personnel. The memo warns the operator of the possibility of forming voids in the Reactor Coolant System and contains the basis for the instructions in the revised procedures that are used for mitigating the consequences of void formation and for enhancing natural circulation. The memo also describes the instrumentation an operator can use to detect void formation which includes:

1. Low RC flow with RCP(s) operating
2. Diverging T_{hot} and T_{cold} temperature
3. Increasing temperature as indicated on the core thermocouples
4. Source Range Monitor Indicator increase

Instrument indications available to verify that natural circulation is occurring. These indications are:

1. RCS $\Delta T \leq$ full power ΔT
2. Constant or decreasing RCS temperature
3. Steam generator pressure following Reactor Coolant temperature

- B. Our procedures have been revised to provide actions possible for terminating conditions leading to void formation. Specifically, our procedures:

1. Verify Safeguards Auto Actions have occurred (all safeguards equipment is in operation).
2. Verify Auxiliary Feedwater Flow to both Steam Generators.
3. Contain a note which instructs the operator not to defeat auto actions unless a failure of auto actions has occurred or continued operation of safeguards equipment could result in unsafe plant conditions.
4. Instruct the operator to leave one reactor coolant pump running, if available.

POOR ORIGINAL

576041

NORTHERN STATES POWER COMPANY

Mr James G Keppler

Page 3

June 22, 1979

5. Do not allow termination of S.I. flow unless the following conditions are met:
 - a. RCS pressure > 2000 psig.
 - b. RCS pressure is increasing.
 - c. S/G narrow range level indication is seen on at least one S/G.
 - d. Pressurizer level \geq 50%.
 6. Instruct the operator to manually restart S.I. pumps if the RCS pressure decreases in excess of 200 psi; or if the RCS hot and cold leg temperatures cannot be maintained 50° lower than the saturation temperature for the existing RCS pressure following termination of S.I. flow.
- C. Our procedures have been revised to ensure that the necessary actions which can be taken to improve core cooling and ensure natural circulation are taken. Specifically, our procedures:
1. Verify Auxiliary Feedwater Flow to both Steam Generators.
 2. Instruct the operator to leave one reactor coolant pump running, if available.
 3. Do not allow termination of S.I. flow unless the following conditions are met:
 - a. RCS pressure > 2000 psig.
 - b. RCS pressure is increasing.
 - c. S/G narrow range level indication is seen on at least one S/G.
 - d. Pressurizer level \geq 50%.
 4. Instruct the operator to manually restart S.I. pumps if the RCS pressure decreases in excess of 200 psi; or if the RCS hot and cold leg temperatures cannot be maintained 50° lower than the saturation temperature for the existing RCS pressure following termination of S.I. flow.

Final procedure revisions will comply with agreements reached between the Westinghouse PWR Owners Group and the NRC.

POOR ORIGINAL

576042

NORTHERN STATES POWER COMPANY

Mr James G Keppler

Page 4

June 22, 1979

Item 3

Instructions in the form of an approved Special Order were immediately issued upon receipt of IE Bulletin 79-06A, which directs the control room operator to manually initiate S.I. on low pressure (1815 psig). Also, the last revision of our Emergency Procedure E1.0, Safety Injection Initiation, contains the statement to "initiate SI manually if Auto SI did not occur at activation setpoint".

A design change was initiated for modification of the existing pressurizer pressure coincident with pressurizer level safety injection actuation logic to a two-of-three low pressurizer pressure actuation logic. Implementation of this design change was on hold pending NRC review of the safety evaluation and approval of the technical specification change request.

In the interim, one pressurizer level input was placed in trip on Unit 2. This interim measure was to be performed on Unit 1 if the actuation logic change was not approved prior to completion of the refueling outage in progress.

On April 26, the Region III office contacted the plant advising that all pressurizer levels should be placed in trip; this action was completed with proper reviews before the end of that day.

Modification of the safeguards circuitry was accomplished by May 2 on Unit 1 and May 8 on Unit 2 to allow Safety Injection on two-out-of-three low pressurizer pressure.

Item 4

The Containment Isolation (CI) signal is initiated by low steam line pressure in either loop, coincident low pressurizer level, and pressure or high containment pressure. These are the same initiating signals which actuate safety injection. In addition, manual 1 of 2 safety injection actuation from the control board will initiate CI. CI can also be initiated manually from the control board using a 1 of 2 manual CI control.

Manual CI initiates Containment Ventilation Isolation (CVI). CVI can also be initiated by manual containment spray actuation, manual or automatic safety injection initiating signals, and high containment air particulate monitor activity or high containment gas monitor activity.

Those actuating safety injection signals, either automatic or manual, must be reset on a "train" basis; i.e., the manual reset consists of two momentary controls on the control board — one for each train. The containment radiation monitors which actuate CVI must be individually and manually reset

POOR ORIGINAL

576043

NORTHERN STATES POWER COMPANY

Mr James G Keppler
Page 5
June 22, 1979

from the control room. In addition to the safety injection reset and the radiation monitor reset just described, the CI function and the CVI function must be individually and manually reset from the control board on a "train" basis.

Therefore, components actuated by the CI signal and components actuated by CVI are all individually sealed in (latched) so that loss of the actuating signal will not cause these components to return to the position held prior to the advent of the actuating signal. CI and CVI can only be reset on a "train" basis by manual controls on the main control board.

Following a safety injection signal, either manual or automatic, subsequent safety injection signals are blocked by a 90 sec time delay. After the safety injection signal has been manually reset and the time delay interlock is satisfied, then the reactor trip breakers must be reset to enable subsequent automatic safety injection signals.

Emergency procedures contain a manual immediate action which requires the operator to verify that the CI status panel is lit. Subsequent actions require the safety injection signal to be reset only after first verifying that all startup of safeguards equipment has been completed and that the Safety Injection Pumps' suction has transferred from the Boric Acid Storage tanks to the Refueling Water Storage Tank.

The containment isolation initiation design and procedures have been reviewed and no changes were necessary to permit containment isolation of all lines whose isolation does not degrade needed safety features or cooling capability upon automatic initiation of safety injection.

Similarly, per the above concern the automatic containment isolation does not isolate those lines whose isolation would degrade needed safety features or cooling capability; however, the design does isolate all unneeded lines following Safety Injection Initiation. There are no manual valving containment isolation requirements.

Further detail is given in Attachment 1.

Item 5

The auxiliary feedwater system at Prairie Island is automatically initiated.

Item 6

Procedures have been prepared and implemented to alert the operator to the symptoms of a stuck open power operated relief valve (PORV) and instruct the operator to isolate a stuck open valve with its associated remotely controlled motor operated isolation valve when the RCS pressure drops to the PORV automatic closure setpoint.

5176044

POOR ORIGINAL

NORTHERN STATES POWER COMPANY

Mr James G Keppler
Page 6
June 22, 1979

Item 7

Operating procedures which are also used as training instructions have been reviewed and revised to provide guidance for continued operation of engineered safety features after automatic initiation and provide instructions for Reactor Coolant Pump Operation. The items below address the specific requests contained in the June 7 draft request for additional information working paper.

- A. The Emergency Procedures have been revised to comply with the bulletin. Specifically, caution statements have been added which state, "DO NOT DEFEAT AUTO ACTIONS unless you can determine a failure of auto circuits has occurred or continued operation of safeguards equipment will result in unsafe plant conditions". Furthermore, specific SI termination criteria were included in each recovery procedure which assures certain plant conditions must exist before S.I. termination and which assure the plant is in a condition which would not threaten reactor vessel integrity.
- B. 1. Prairie Island Emergency Procedures call for the continued operation of the RHR pumps as long as the system is delivering flow to the reactor coolant system (an exception would be where the pumps must be temporarily stopped and re-aligned during the recirculation switchover processes).

However, in the event that the RCS pressure stabilizes at a value above the shutoff head of the RHR pumps, instructions are provided to stop the RHR pumps. They can then be manually restarted, if required during injection phase, and for recirculation phase operation.

2. The Emergency Procedures provide several criteria for the termination of SI. Specifically to terminate SI, the following conditions must be met:
 - a. RCS pressure > 2000 psig
 - b. RCS pressure increasing
 - c. Narrow range level in at least 1 steam generator
 - d. Pressurizer level \geq 50%

POOR ORIGINAL

576045

NORTHERN STATES POWER COMPANY

Mr James G Keppler
Page 7
June 22, 1979

In addition, if a 200 psig RCS pressure drop or if 50°F subcooling cannot be maintained after SI termination, re-initiation of SI flow is directed.

Following a steamline break, alternative criteria for SI termination are furnished to address the need for SI termination to prevent overpressurization of the RCS. However, if after SI termination a 50°F subcooling cannot be maintained or a 200 psig RCS depressurization occurs, instructions are given to re-initiate SI flow.

- C. Our revised Loss of Reactor Coolant Emergency Procedure contains instructions to stop one Reactor Coolant Pump if both are running. Our review has shown that R.C. Pumps can be operated after Containment Isolation.
- D. The following specific parameters are identified for operator use in evaluating plant conditions and are included in appropriate operating procedures:

- Wide Range RCS Temperature
- Wide Range RCS Pressure
- Steam Generator Water Level
- Containment Pressure
- RWST Level
- Pressurizer Level
- Boric Acid Tank Level
- Containment Sump Level

Final Procedure revisions will comply with agreements reached between the Westinghouse PWR Owners Group and the NRC.

Item 8a

Actual safety related valve positions were verified by running the safeguards hold checklist. The position of safety related valves were found to be in their proper position, ready to respond to a safety signal.

Item 8b

All safety related Operating Procedures, Surveillance Procedures, Preventive

POOR ORIGINAL

576046

NORTHERN STATES POWER COMPANY

Mr James G Keppler
Page 8
June 22, 1979

Maintenance Procedures, and Special Operating Procedures were reviewed. No procedures left valves in unsafe positions. The review verified all valves were left in their proper safeguard positions.

Several procedures were refined to reduce even further the potential for unsafe conditions. A summary of the changes is listed below:

1. PM 3119-1 "Component Cooling Pump Annual Inspection"

The procedure was upgraded to reduce the possibility of valve and maintenance errors.

2. SP 1163 "Component Cooling Valve Test"

The status of valves MV-32120 and -32122 was verified.

3. SP 1143, 2143 "FW Isolation and FW Trip Verification"

The procedure was changed to include verification of several valve positions.

4. SP 2144, 2146 "SI 24X Relay Contact"

Position of control switches is now verified.

5. SP 1147, 2147 "SI 21X Relay Contact"

Procedure changed to specify plant condition at which test can be run.

6. SP 1190 "Emergency DG Oil Storage Tank"

The procedure was refined to verify fuel system is relined after sample is drawn.

7. SP 1186 "Diesel Generator Operability Test"

The procedure was upgraded to give the operator specific instructions on what to do in case of loss of offsite power.

8. SP 1093 "Diesel Generator Manual Test"

Same as 7.

9. PM 3001-2 "Diesel Generator Annual Inspection"

Procedure now verifies that DG auxiliary systems are returned to service after PM on the DG is complete.

576047

POOR ORIGINAL

NORTHERN STATES POWER COMPANY

Mr James G Keppler
Page 9
June 22, 1979

10. C 18 "Engineered Safeguards Systems"

The procedure was changed to eliminate the possibility of leaving the SI pump discharge valves closed.

11. SP 1088, 2088 "SI Pump Test"

Same as 10.

12. SP 1055 "Control Room Special Ventilation Test"

The procedure was changed to verify system realignment.

13. SP 1091 "Containment Fan Coil Test"

The procedure was changed to correct inaccuracies.

14. SP 1106 "Diesel Cooling Water Pump Test"

The procedure was revised to eliminate the potential for leaving the cooling water pump discharge valves closed.

15. C 35 "Cooling Water System"

Procedure now maintains the pump discharge valves open on pump start.

16. C 12 "Chemical Volume Control System"

Checklist was amended to include a 121 BA tank level instrument.

17. Cl.1.20.7-5 "Start Up Checklist"

Checklist amended to include a DG auxiliary system valve.

Our technical specifications require periodic surveillance of locked valves. We have expanded the list of locked valves which we feel are critical to proper operation of safeguards systems and we require a monthly check of all these locked valves. This is accomplished by procedure check-off. There is one area where we have taken exception to the monthly check (this does not violate our technical specifications) and that is those valves located in containment. They are double checked after a refueling shutdown or in the event maintenance requires manipulation of these valves. During normal operations all entries into containment are restricted and any required entries are accomplished under strict administrative controls.

POOR ORIGINAL

576048

NORTHERN STATES POWER COMPANY

Mr James G Keppler
Page 10
June 22, 1979

Item 9

Inadvertent release of radioactive liquids and gases due to undesired pumping or venting is precluded due to the isolation design discussed in Item 4.

Specifically, those systems designed to transfer potentially radioactive gases and liquids out of the primary containment, existing interlocks, CI or CVI functions, and those manual controls which demonstrate the continued operability of the isolation functions are tabulated as Attachment 2.

There are no manual controls used to assure the continued operability of the features (Containment Isolation and Containment Ventilation Isolation) which prevent inadvertent pumping, venting or other release of radioactive fluids. Manual actions are used if and only when Containment Isolation is no longer desirable.

Specifically, all Containment Isolation and Containment Ventilation Isolation interlocks remain operable until manually reset, and in the particular case of high radiation interlocks, additional manual resetting is necessary for each initiating radiation monitor at the appropriate monitor's control panel.

Item 10a

Removal of equipment from service is controlled by a procedure or with the Work Request Authorization. In either case administrative controls assure that the Technical Specification requirements which require verification of the operability of the redundant safety-related system be identified and verified.

We are in the process of reviewing each maintenance and test procedure to verify by test or inspection the operability of the redundant safety related system prior to the removal of any safety-related system from service. This will be accomplished by verifying that the redundant system's surveillance testing is up-to-date and that the system has not been made inoperable by testing or maintenance. We have accelerated our schedule for review and modification of the procedures and expect to be completed by June 29, 1979. A summary of the results of the review and actions taken will be submitted within two weeks after completion of this effort.

Item 10b

Our maintenance and test procedures presently require functional testing of equipment before returning to service and contain the requirements to return the system to normal with respect to all aspects affected by the maintenance or test controls.

POOR ORIGINAL

576049

NORTHERN STATES POWER COMPANY

Mr James G Keppler
Page 11
June 22, 1979

We are in the process of reviewing and modifying each maintenance and test procedure to verify system operability by a second independent inspection of those valves and switches that would affect the system's ability to perform its safety function. We have accelerated our schedule for review and modification of the procedures and expect to be completed by June 19, 1979. A summary of the results of the review and actions taken will be submitted within two weeks after completion of this effort.

Item 10c

Our maintenance and test procedures provide for explicit notification of the Shift Supervisor since his authorization of the removal of equipment from service and his review upon the return of equipment to service is required.

Relay of system status between Shift Supervisors, Lead Plant Equipment and Reactor Operators, and Plant Equipment and Reactor Operators is accomplished utilizing a formalized method. This method requires that a status list be completed and turned over to the relieving shift control room individuals. In addition separate status boards for each unit are maintained in the control room.

Item 11

Procedures have been revised to require NRC notification within one hour of the time the reactor is not in a controlled or expected condition and maintenance of an open channel of communication.

Item 12

Operating procedures are being reviewed and will be revised, if required, to provide instructions to control and/or remove generated hydrogen gas. Hydrogen can be removed from the RCS by a variety of methods including (1) RCS spray of the pressurizer steam space, (2) venting of hydrogen to the PRT via the pressurizer power operated relief valves, and/or (3) degassing of the RCS via letdown. The impact of S.I. initiated CI on the ability to degas the RCS via letdown will require further investigation. Operating procedures have been revised.

The present method for post LOCA hydrogen control involves controlled purging from the Containment building to the Shield Building annulus and discharge via the Shield Building Ventilation System and dilution by the addition of air to containment. Operating procedures were reviewed and were determined to be adequate.

POOR ORIGINAL

576050

NORTHERN STATES POWER COMPANY

Mr James G Keppler
Page 12
June 22, 1979

Item 13

A change to Technical Specifications was proposed to change safety injection logic to two-out-of-three low pressurizer pressure. The change was issued May 1, 1979. Modification of the logic was made May 2 on Unit 1 and on May 8 on Unit 2.

A setpoint change was implemented raising the interlock setpoint for enabling actuation of the pressurizer power-operated relief valves from 2185 to 2335 psig so that failure of a single pressure channel will not cause valve opening. That is, two channels (instead of one) must provide high pressure signals for valve actuation.

Yours very truly,

DE Gilbert for L J Wachter

L J Wachter
Vice President
Power Production and
System Operation

cc: Mr G Charnoff
NRC Office of Inspection and Enforcement
Washington, D.C.

POOR ORIGINAL

576050

Attachment 1

<u>Containment Isolation by Penetration</u>	<u>Receives Containment Isolation</u>
Pressurizer Relief Tank to Gas Analyzer	Yes
Pressurizer Relief Tank Nitrogen Supply	Yes
Primary System Vent Header	Yes
Reactor Coolant Drain Tank Pump Discharge	Yes
Main Steam	*
Feedwater	Yes
Steam Generator Blowdown	Yes
Residual Heat-Out	No
Residual Heat-In	No
Letdown	Yes
Charging	No
Reactor Coolant Pump Seal Water Supply	No
Reactor Coolant Pump Seal Water Return	Yes
Pressurizer Steam Sample	Yes
Pressurizer Liquid Sample	Yes
Hot Leg Sample	Yes
Instrument Air	Yes
Reactor Coolant Drain Tank to Gas Analyzer	Yes
Containment Air Sample (In and Out)	Yes
Containment Purge (Exhaust and Supply)	Yes
Sump A Discharge	Yes
Steam Generator Blowdown Sample	Yes
Safety Injection	No
Containment Spray	No
Sump B RHR Suction	No
Nitrogen to Accumulators	Yes
Component Cooling to and from Reactor Coolant Pumps	No
Fan Coil Cooling Water Supply and Return	No
Component Cooling to and from Excess Letdown Heat Exchanger	Yes
Vacuum Breakers	Yes
Hydrogen Control Vent and Air Supply to Shield Building	No
Containment In-Service Purge Supply and Exhaust	Yes
Reactor Water Make-Up to Pressurizer Relief Tank	Yes
Auxiliary Feedwater	No
Low Head Safety Injection	No

Component cooling water supply and return for the reactor coolant pumps is not interrupted by containment isolation. Similarly, reactor coolant pump seal water supply is not interrupted. Reactor coolant pump seal water return is isolated by containment isolation; however, this does not directly disable continued running of the pumps but rather would result in undesirable relief of the return flow to the pressurizer relief tank. To permit RCP operation with the normal seal return flow path, the containment isolation valves can be reopened by the control room operator using the appropriate control switches.

*Steam line isolation occurs as a result of:

- (1) SI plus Hi-Hi Steam Flow, (2) SI plus Lo-Lo Tave plus Hi steam flow, and
- (3) Hi-Hi Containment Pressure.

576052

Attachment 2

PENETRATIONS USED TO TRANSFER RADIOACTIVE GASES AND LIQUIDS
OUT OF PRIMARY CONTAINMENT

PEN	COMPONENT	DESCRIPTION OR SYSTEM	HIGH RAD INTERLOCK	CNTMT ISOL SIGNAL	OPERABILITY COMMENTS (Manual, SI Reset, CI Reset, CVI Reset, Radiation Reset)
1	CV-31319	PRT to Gas Analyzer	No	Yes	*
	CV-31318	PRT to Gas analyzer	No	Yes	*
4	CV-31434	RCDT Vent Hdr	No	Yes	*
	CV-31435	RCDT Vent Hdr	No	Yes	*
5	CV-31436	11/12 RCDT Pmps Dsch Hdr	No	Yes	*
	CV-31437	11/12 RCDT Pmps Dsch Hdr	No	Yes	*
8A	MV-32040	11 SGB Isolation	No	Yes	*
8B	MV-32043	12 SGB Isolation	No	Yes	*
15	CV-31296	Przr Stm Sample	No	Yes	*
	CV-31297	Przr Stm Sample	No	Yes	*
16	CV-31298	Przr Liquid Sample	No	Yes	*
	CV-31299	Przr Liquid Sample	No	Yes	*
17	CV-31300	RCS Loop B Hot Leg Smpl	No	Yes	*
	CV-31301	RCS Loop B Hot Leg Smpl	No	Yes	*
21	CV-31545	RCDT to Gas Analyzer	No	Yes	*
	CV-31546	RCDT to Gas Analyzer	No	Yes	*
22	CV-31092	Cntmt Air Sample-In	No	Yes	*
	CV-31022	Cntmt Air Sample-In	No	Yes	*
23	CV-31019	Cntmt Air Sample-Out	No	Yes	*
	CV-31750	Cntmt Air Sample-Out	No	Yes	*
25A	CV-31570	1 Cntmt Prg Exht Isol-CV-A	Yes (CVI)	Yes (CVI)	need CVI reset, purge mode selected
	CV-31569	1 Cntmt Prg Exht Isol-CV-B	Yes (CVI)	Yes (CVI)	need CVI reset, purge mode selected
25B	CV-31313	Cntmt Purge Sply CV-A	Yes (CVI)	Yes (CVI)	need CVI reset, purge mode selected
	CV-31312	Cntmt Purge Sply CV-B	Yes (CVI)	Yes (CVI)	need CVI reset, purge mode selected
26	CV-31439	11/12 Cntmt Sump A Dsch	No	Yes	*
	CV-31438	11/12 Cntmt Sump A Dsch	No	Yes	*

Attachment 2

PENETRATIONS USED TO TRANSFER RADIOACTIVE GASES AND LIQUIDS
OUT OF PRIMARY CONTAINMENT

PEN	COMPONENT	DESCRIPTION OR SYSTEM	HIGH RAD INTERLOCK	CNTMT ISOL SIGNAL	OPERABILITY COMMENTS (Manual, SI Reset, CI Reset, CVI Reset, Radiation Reset)
27A	CV-31637	11 SGBD Sample	No	Yes	*
	CV-31402	11 SGBD Sample	No	Yes	*
	CV-31638	12 SGBD Sample	No	Yes	*
	CV-31403	12 SGBD Sample	No	Yes	*
41A	CV-31621	Cntmt Vsl Vac Brkr CV A	No	Yes	*
	CV-31624	Cntmt Vsl Vac Brkr CV A	No	No	Gravity Operated Check
41B	CV-31622	Cntmt Vsl Vac Brkr CV B	No	Yes	*
	CV-31625	Cntmt Vsl Vac Brkr CV B	No	No	Gravity Operated Check
42A	MV-32273	14 Dome Recirc/Annulus to GA	No	No	n/a
	CV-31929	14 Dome Recirc to GA	No	No	n/a
	CV-31927	14 Dome Recirc to Annulus	No	No	n/a
	MV-32276	Air to Cntmt Vsl	No	No	n/a
	SV-33991	Air to Cntmt Vsl Vent	No	No	n/a
42B	CV-31633	Cntmt In-Ser Prg Sply A	Yes (CVI) & Rad Mntr	Yes (CVI)	need CVI reset or Rad Mntr reset & manual reopen
	CV-31633	Cntmt In-Ser Prg Sply B	Yes (CVI) & Rad Mntr	Yes (CVI)	need CVI reset or Rad Mntr reset & manual reopen
43A	CV-31311	Cntmt In-Ser Prg Exh A	Yes (CVI) & Rad Mntr	Yes (CVI)	need CVI reset or Rad Mntr reset & manual reopen
	CV-31310	Cntmt In-Ser Prg Exh B	Yes (CVI) & Rad Mntr	Yes (CVI)	need CVI reset or Rad Mntr reset & manual reopen
50	MV-32271	11 Dome Recirc/Annulus to GA	No	No	n/a
	CV-31923	11 Dome Recirc to Annulus	No	No	n/a
	CV-31925	11 Dome Recirc to GA	No	No	n/a
	MV-32274	Air to Cntmt Vsl	No	No	n/a
	SV-33990	Air to Cntmt Vsl Vent	No	No	n/a

*Need CI reset and manual re-open