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DUKE POWER

April 26, 1989

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Duke Power Company
McGuire Nuclear Station
Docket No. 50-369
Unit 1 Restart

This letter documents discussions I and several of my staff members held with D. B. Matthews and other NRC staff members during an April 25, 1989 telephone conference call. The subject of our call was the actions required prior to the restart of McGuire Unit 1 from the March 7, 1989 steam generator B tube rupture event. These actions were also discussed with the NRC on April 13, 1989 at a meeting held in White Flint and confirmed in an NRC letter dated April 19, 1989.

Based upon agreements reached during our April 25 conference call the following response to each of the actions contained the April 19 letter is provided.

(1) Further meetings and reporting regarding cause of the tube failure and integration of these results into effective corrective actions.

Response: We will meet with the NRC staff, again in White Flint, on May 5, 1989 to present and discuss the Unit 1 steam generator B tube report. The restart of Unit 1 is contingent upon receiving your concurrence at this meeting. We would like to do everything possible to facilitate the NRC's ability to support our restart effort. Prior to our meeting, we will talk with E. L. Murphy of the NRC staff daily to keep him informed of the steam generator tube report status. We will attempt to provide Emmett a draft of the report by May 3.

(2) Further meetings regarding tube plugs.

Response: The report mentioned in Item 1 will contain a section addressing tube plugs in the McGuire steam generators. We will also discuss the plugs during our presentations at the May 5 meeting.

(3) Completion of procedural changes for entry conditions into Emergency Operating Procedures as proposed during the meeting.

Response: We have addressed the procedure related actions from our April 13 meeting as well as the procedure related findings of the April 10, 1989 AIT report. These are hereby attached to this letter for NRC review.

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(4) Completion of certain items from the NRC's Augmented Inspection Team Report of April 10, 1989 to be determined from your discussions with the Team Leader.

Response: We will coordinate the resolution of these items with the NRC Resident Inspector at McGuire.

We will rely upon the daily telephone calls mentioned in Item 1 to identify any considerations for changing the above schedule. Please let me know if there is any additional information we can provide prior to our May 5 meeting that will assist in expediting the NRC's review of the McGuire Unit 1 restart.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

JSW/353/td

cc: Mr. S. D. Ebneter
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Mr. P. K. VanDoorn
NRC Resident Inspector
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McGuire Operations Response To Procedure Concerns and Findings from the AIT Audit

The following are McGuire Operations responses to the procedural concerns and findings addressed in the AIT Report. The concerns and findings will be listed first with the AIT Report page number and then McGuire's response. All responses will address the procedures as Unit 1 procedures but all statements are true for both Unit 1 and Unit 2 procedures.

1. **Concern from page 13 and 14**

   Upon receipt of an alarm on the B steamline radiation monitor, 1-EMF-25, the reactor operator at the controls verified the validity of the alarm before checking other parameters. He then checked the pressurizer level, reactor coolant makeup flow, steam generator levels and main feedwater regulator valve positions. In addition to all these including a condition that the shift referred to as the "classical S/G tube leak symptoms", the "condenser air ejector exhaust high gas radiation" annunciator alarmed. The unit supervisor concluded that the unit was experiencing a S/G tube leak and directed that AP/10 be implemented. Two of the four symptoms listed in that procedure were received: steam line high radiation and air ejector high gaseous radiation. However, the symptoms identified in AP/10 do not include increasing S/G level, decreasing feedwater flow or feedwater valve position, although the shift crew stressed during interviews that these indications were the deciding parameters. The definitive SGTR symptoms which the operator relied on to determine which AP to use, feedwater flow and steam generator level, are included in step 6 of AP/10, well into the procedural instruction. Although this event fell within the intended boundaries of the Westinghouse Owners' Group guidelines for procedures, it was handled by independent operator diagnosis and resultant direct usage of a nonemergency operating procedure.

Response:

AP/1/A/5500/10 (NC System Leakage Within the Capacity of Both NV Pumps) has been rewritten and approved and now includes decreasing feedwater flow and feedwater regulation valve position as symptoms for Case I "Steam Generator Tube Leakage". See Attachment 1, AP/1/A/5500/10, page 5 of 41. In addition, it is questionable if this event fell within the intended boundaries of the Westinghouse Owners Groups (WOG) ERG's. The entry conditions for the Safety Injection portion of ERG E-0 are the plant specific setpoint or requirement for safety injection have been met, the SI annunciator light lit or SI pumps running. See Attachment 2, pages 2 and 3 of 6. During this transient, none of MNS SI setpoints were reached, the plant specific requirement of pressurizer level less than 5% was not reached and only two out of six SI pumps were running, the normal charging pump and the standby charging pump. In addition, the WOG background document for E-0 "Reactor Trip or Safety Injection" has a paragraph explaining that the operators are expected to take manual action for anomalous conditions during power operations. These actions would include taking manual control of the automatic control systems, turning on additional charging pumps, reducing power level, etc. If these types of actions do not alleviate the trend toward a reactor trip or safety injection, the operator is permitted to trip the reactor and, if necessary, actuate safety injection. See Attachment 2, page 4 of 6.

2. **Concern from page 14**

One of the immediate operator actions after identification of the incident and entry into AP/10 was to reduce electrical load by reducing main generator power. AP/10 does not give direction to accomplish this task. The operators stated that they knew from training that this action should be performed. They did not use any procedure for this action and, therefore, had to ask the unit supervisor to determine the rate at which he wanted the load reduced. The needed rate of load reduction was analyzed and determined by the unit supervisor. This analysis placed additional burden on this individual during response to the event.

Response:

AP/1/A/5500/10 did not require a generator load decrease nor did it give a rate for the load decrease. The operators have been trained to perform this task. In the new AP/10, a step to initiate a load decrease to remove the unit off line has been added. This step is prefaced by a note explaining the load reduction rate should be determined based on the leak size and on the ability to remove the unit in a controlled manner. See Attachment 1, pages 6 and 21 of 41.

3. **Concern from page 14 and 15**

The operators considered initiating SI. They concluded however, it would not be advantageous if SI were initiated. Additional CRO manpower would be required to monitor the successful initiation of SI. In addition, the operators were uneasy regarding the dependability of the RN supply to the unaffected unit due to logic wiring problems experienced in the past. They also considered SI, when not mandatory, to be an unnecessary challenge to safety related equipment (i.e., containment isolation and diesel generator start). This preference not to manually initiate SI is reflected both in their AP and in their training.

Response:

The operators were trying to state Duke's philosophy of not challenging safety systems if they are not needed. In stating this philosophy, the operators were trying to explain that when safety injection is initiated, it is an event unto itself in which hundreds of components are required to start, stop, realign, etc. All of these components must be verified to be functioning properly prior to proceding with the procedure to mitigate the initial event. The operators gave the AIT Inspectors an example of an event on Unit 1, which was an intermitent ground in the A train solid state protection system cabinet. This intermitent ground gave a partial A train safety injection where only a portion of the A train components realigned and no B train components realigned. This event caused significant operational problems for both the operating unit and the affected unit.

The operators are not uneasy about initiating safety injection if it is required by plant status or procedures. McGuire's procedures and training do not and have not discouraged the operators from initiating safety injection when required. The difference at McGuire was the threshold or setpoint at which manual safety injection should be initiated. The actuation of automatic safety injection is dictated by the accident analysis and the manual initiation setpoint was more conservative than the automatic setpoint.

4. **Concern from page 15**

In the "immediate actions" section of AP/10, "response not obtained" for the step that requires the operator to manually initiate SI, there is no guidance to the operator on where to enter the procedure for SI.

Response:

The step in AP/10 which requires the operator to manually initiate safety injection does not identify which step to enter EP/1/A/5000/01 "Safety Injection" because the procedure is entered at the beginning. This concept is a given concept in that if an operator either trips the reactor, initiates safety injection or receives an automatic signal, the operator proceeds to the beginning of either the reactor trip procedure or safety injection procedure.

5. **Concern from page 15**

Step 3 of AP-10 directed determination of whether S/G blowdown isolation was required based solely on whether 1-EMF-34 (blowdown sample high rad alarm) was lit. Since it was not lit the operator did not verify S/G auto isolation nor manually isolate blowdown from any of the generators.

Response:

AP/10 now has a step to isolate blowdown on the ruptured steam generator and does not rely solely on the EMF to isolate blowdown. See Attachment 1 page 7 of 41.

6. **Concern from page 15**

AP/10 step 3 uses 1-EMF-34 as the sole determinant of whether S/G blowdown isolation is required. Then, in step 7b, after identifying the affected generator, the procedure does not isolate blowdown on the affected generator as part of the generator's isolation. This is similar to McGuire's EP/04 where, after identifying the ruptured generator in step 1, the subsequent steps isolate main steam to the ruptured generator but does not isolate blowdown. This is a significant safety-related deviation from the Westinghouse Owners' Group guideline E-3, SGTR, which requires, after identifying the ruptured generator, that its blowdown be isolated.

Response:

AP/10 now has a step to isolate blowdown on the ruptured steam generator and does not rely solely on the EMF to isolate blowdown. See Attachment 1 page 7 of 41. EP/1/A/5000/04 "Steam Generator Tube Rupture" does not have a specific step to isolate blowdown on the ruptured steam generator because it is done automatically by safety injection and checked by EP/1/A/5000/01, "Safety Injection". The check of the ESF Monitor Light Panel in EP/01 verifies that Containment Phase A Isolation Train A/B have been aligned properly and if the Phase A components have not been aligned the operator manually aligns those misaligned components. The steam generator blowdown valves do get isolated on a Phase A Containment Isolation signal. Therefore this action is done automatically and verified by the operator. See Attachment 3 page 3 of 3. Hence, this is not a safety significant deviation from the WOG guidelines.

7. **Concern from page 15**

Early in the implementation of AP-10 the shift manager (STA) entered the control room and began monitoring the critical safety functions of SPDS. This was an appropriate action but not specified by the procedure. Also, although the licensee has indicated that they have a fully operational SPDS, there were two parameters that were inaccurately displayed by SPDS during the event because of faulty computer logic. NC integrity was being displayed as a "red path" (extreme challenge to this safety function; immediate operator action is required) and, core cooling was being displayed as a "yellow path"; indicating that his critical safety function was in an off-normal state and might require operator attention. The AIT was informed that there were several software problems with the SPDS.

Response:

McGuire's SPDS is operable but the McGuire Operator Aid Computer is not safety related. Operations has an emergency procedure EP/1/A/5000/10, "Critical Safety Function Status Trees" which is the controlling procedure for monitoring Critical Safety Functions which uses SPDS as a convenient aid if the OAC is available. See Attachment 4. In the case where a software problem with the computerized SPDS gives an invalid alarm, the operators would check to see if the alarm is valid utilizing the status trees. If the alarm was invalid as it was in the AIT examples, the operators would ignore it and notify reactor group of the problem as was done in this case.

8. **Concern from page 15 and 16**

At step 7 of AP/10 the operator was directed to "shut down and cooldown the unit using OP/1/A/6100/02, Controlling Procedure for Unit Shutdown", in conjunction with the remaining steps of AP/10. This OP is about 50 pages long, yet no direction is given to the operator in AP/10 regarding which page or section of the OP to enter. Thus, the operator entered the procedure where he felt it was appropriate.

Response:

The last step in AP/10 now has the operator utilize one of the three emergency subprocedures for cooling down the ruptured steam generator EP/1/A/5000/4.1 "SGTR Cooldown Using Steam Dump, EP/1/A/5000/4.2 "SGTR Cooldown Using Backfill" or EP/1/A/5000/4.3 "SGTR Cooldown Using Blowdown". In conjunction, the operator is referred to OP/1/A/6100/02 "Controlling Procedure for Unit Shutdown" Enclosure 4.2 to perform applicable steps. EP/4.1, EP/4.2 and EP/4.3 are entered at the first step so no entry step reference is needed. See Attachment 1, page 14 of 41. There is no consistent way of addressing which step in OP/02 to enter due to the nature of every transient being different and having the plant end up in a slightly different status as far as what components are running, etc. To put this situation in perspective, at this time in the procedure, primary pressure and the ruptured steam generator pressure are equalized and the primary system is cooled down below the saturation temperature and pressure for steam line PORV's or safety reliefs to lift. The immediate transient has been handled and cooldown options are being decided. A licensed operator is capable of deciding where OP/02 should be entered to match where the plant is currently. In addition, the EP subprocedure is the controlling procedure with the shutdown OP being used for reference.

9. **Concern from page 16**

After the unit was off line, AP/10 directed the operator to isolate the affected steam generator. AP/10, at step 7b, directs the operator to "Close (main steam) isolation and bypass valves". By training and convention the operator knew this meant to open the by-pass, close the MSIV, then slowly close the by-pass to prevent a pressure transient.

Response:

The operator opened the MSIV bypass valves before closing the MSIV to avoid a pressure transient to prevent a pressure spike which could have lifted the steamline PORV or Safety Reliefs. Even though this opened another steamline valve the MSIV's and MSIV bypass valves are fail close valves. This action adds approximately 20 seconds to isolating the steamline which is well worth the effort to avoid lifting a steamline PORV or Safety which would cause a direct release to the public. Also, the ruptured steam generator steamline was isolated within 11 minutes of the leak which is well within our safety analysis assumed time of 30 minutes.

10. **Concern from page 16**

Near the end of AP/10, the operator was directed to "dump steam to condenser by slowly opening steam isolate bypass valve on ruptured generator". Due to the brevity and lack of specificity of this instruction the operator opted to reference EP/4.1 where there was more detailed guidance. One of the difficulties of this procedural transition (or parallel usage) is that the two types of documents may not have a consistent set of definitions. For example, AP/10 step 7d refers to "...faulted S/G pressure..." when referring to the generator with the tube leak and at point 7f of the same page refers to the "ruptured" generator as the one with the tube leak. The EPs carefully use these terms to indicate a generator with a secondary leak as "faulted" whereas "ruptured" is used to refer to a generator with a primary to secondary leak through one or more tubes. Also, the concurrent use of procedures increases the physical and mental burden of the US who performed as the "Procedure Reader".

Response:

The last step in AP/10 now has the operator utilize one of the three steam generator tube rupture cooldown subprocedures. See Attachment 1, page 14 of 41. These procedures offer more detailed guidance on cooling the ruptured steam generator. As far as the concern over the lack of consistent definitions, the one example addressed in the report was an error in our procedure. McGuire Operations strives for consistency not only among our emergency procedures but all of our procedures.

11. **Concern from page 17**

Step, 7.f.1 of AP/10, listed an alternative to dumping steam from the ruptured generator to the condenser, that alternative would be blowing down through the BB recycle system. Due to the operators lack of confidence in the BB recycle system's Hx integrity they chose to dump steam to the condenser.

Response:

This concern was initiated by the Technical Support Center not the operators. The TSC had confidence in the BB recycle Hx integrity for short term use but not long term use. To put this in perspective, long term use would be identified as use for a period of greater than a week. The decision to not use the BB recycle system was based on the fact that the TSC determined blowing down the ruptured steam generator through normal blowdown was a better option. In addition, the ruptured steam generator was not steamed to the condenser after it was isolated. This statement in the AIT Report is in error.

12. **Concern from page 16**

Step 7.f.1, unlike the step in EP/4.1, makes no reference to performing an offsite dose calculation prior to dumping steam from a ruptured generator to the condenser. EP/4.1 contains a caution indicating that such a calculation should be done. The shift supervisor indicated at the time that he did not intend to have the dose calculation performed prior to steaming because EP/4.1 stated "should" and therefore was not a requirement.

Response:

The last step in AP/10 now has the operator utilize one of the three steam generator tube rupture cooldown subprocedures. If the operator chooses EP/4.1, that subprocedure has always contained the caution that an offsite dose evaluation (not a calculation) should be performed prior to using the procedure. This caution is consistent with the WOG guidelines which state that the evaluation should be done. See Attachment 2, pages 5 and 6 of 6.

13. Concern from page 16 and 17

Cooldown per OP/02 was delayed initially since primary boron sample results were not available until 2 hrs and 44 minutes after the trip (2:30 a.m.). The Boron sample concentration was not high enough to allow cooldown below 200°F so cooldown was not resumed. Boron concentration was high enough to initiate cooldown to an intermediate temperature but the operators were unaware of this option until 3 hrs 34 minutes after the trip. Cooldown was started 5 minutes after this option was realized.

Response:

The shutdown OP and the reactivity balance OP have been changed to clarify the operator's ability to cooldown to intermediate temperatures as long as shutdown margin is maintained for those intermediate temperatures. See Attachment 5.

14. Concern from page 17

After the reactor trip, primary system pressure was maintained above 1000 psig while S/G B pressure decreased to approximately 800 psig. This continued for 4 3/4 hours. This was because step 2.33 of OP/02 and the cooldown curves require primary system temperature to be below 425° prior to decreasing pressure below 1000 psig (LOCA FSAR requirement). The operators did not become aware of a note immediately before step 2.33 allowing pressure to be reduced to 750 psi with shift supervisor approval under extenuating circumstances.

Response:

With the clarification that the cooldown can be initiated to intermediate temperatures, the plant should not get in a position that the cooldown curve requirement at 425^{O}F and 1000 psig would be a problem. In addition, this incident will be covered in operator requalification to further make the operators aware of the note allowing the shift supervisor to reduce primary pressure to 750 psig under extenuating circumstances.

15. **Concern from page 17**

Prior to commencing cooldown using S/G backfill (10:15 a.m.), the SRI asked the reactor engineer if shutdown margin projections had been made due to the impending dilution. The engineer indicated that operations personnel had indicated they did not need one but he thought it was a good ideas. He then provided the information.

Response:

Operations personnel in the TSC were well aware of the need for increased boron concentration needs for the impending dilution due to S/G backfill. This was addressed in EP/4.2. Operations personnel apparently failed to communicate this fact to the Reactor Engineer. The Operations personnel were taking the shutdown boron concentration given by the Reactor Engineer and adding the required boron concentration addition.

16. **Concern from page 17**

The procedure finally selected by the TSC to depressurize the NC System and S/G B was EP/4.2, "SGTR Cooldown Using Backfill". Step 9 if EP/4.2 (checking for void in upper head) contains a sub step (b) that requires the operator to continue monitoring for upper head void while going on to the next step in the procedure. This does not assure that attention is given to monitoring for voids while going on to another major action (i.e., NC system depressurization).

Response:

The step in EP/4.2 will be clarified to continue monitoring for upper head voids during the remainder of the procedure. This was understood before but not clearly stated in the procedure.

17. **Concern from page 17 and 18**

Based on a review of the sequence of events, operator and plant personnel interviews, and a control room walkthrough with members of the operating crew, the AIT concluded that considering the training and procedure impediments the operating crew performed adequately in mitigating this particular event. The crew followed steps prescribed in the station procedures, however, the procedures, were found to have significant weaknesses which could result in unnecessary releases of radioactivity to the environment should future SGTR events occur. The mitigative strategy which McGuire used for coping with this event deviated substantially from the Westinghouse Owners' Group Emergency Response Guidelines.

Response:

The event was handled extremely well by the operating crew. Decisions were made in the TSC over which cooldown option to use for the ruptured steam generator. Procedures needed and have been enhanced but most of the enhancement is in procedure structure. The comment that the mitigative strategy deviated substantially from the WOG guidelines has been addressed by Concern 1.

18. **Finding from page 28**

The operating crew performed adequately in mitigating this event despite procedural weaknesses which caused the operator to select portions of additional procedures that contained more detailed guidance.

Response:

As stated in Concern 17, the operating crew performed outstandingly, not adequately. In addition, Operations personnel realized that enhancements need to be made to some of Operations' procedures and have enhanced or are enhancing procedures from the lessons which were learned from this event.

19. Finding from page 28

Operators failed to promptly identify the magnitude of the reactor coolant leak, to cooldown and to equalize pressure.

Response:

Operations did not identify the magnitude of the leak immediately. Identifying the exact magnitude of a leak while pressurizer level is decreasing due to the leak and primary system cooldown due to a load decrease, charging is being increased, letdown is being decreased and the Volume Control Tank level is changing is difficult. The operators concern was, "Is the leak greater than 50 gpm (Alert Classification) and if greater than 50 gpm can pressurizer level be maintained?" Quantifying whether the leak is a 150 gpm leak or 450 gpm leak is not utmost in the operator's mind nor is it of utmost importance. The operator must first control the plant and then worry about quantifying the exact leakage rate.

The operator did cooldown and depressurize the primary system to the ruptured steam generator pressure promptly. The resultant cooldown to cold shutdown was done in a controlled deliberate manner at the direction of the TSC to minimize possible errors which could result in unnecessarily jeopardizing the plant or the public. In a steam generator tube rupture event, the critical action is to equalize ruptured steam generator pressure with primary pressure. The resultant cooldown can and should be done in a calm, slow, deliberate manner.

20. **Finding from page 28**

Procedures and training discouraged operators from safety injecting. Although SI was not needed in this event, procedures and training should be reviewed to assure operators will SI when appropriate in the future.

Response:

Procedure and training do not and have not discouraged the operator from initiating safety injection when safety injection is required. Training has been done on our procedures and has supported our procedures. AP/10 is and was one of the most widely and often used procedures during simulator requal training. The difference at McGuire was at which point the operator was required to initiate manual safety injection. McGuire chose to have the operator maximize charging by reducing letdown flow, starting a second charging pump and opening the high head injection line isolation valves. If pressurizer level still decreased to less than 5% level the operator would manually activate safety injection. The revised AP/10 has been written to allow the operator to start the second charging pump, reduce or isolate letdown flow but will not allow manually opening the high head injection line isolation valves. See Attachment 1 pages 5 and 6 of 41.

21. **Finding from page 28**

Operators lacked confidence that certain systems would function following an SI, also considered unnecessary challenge to safety related equipment.

Response:

Refer to Concern 3.

22. **Finding from page 28**

The McGuire strategy for coping with this event deviates from the WOG guidelines in several significant aspects. Overall it addresses to an accident which is within the scope of the WOG EOPs. Guidelines for emergency operating procedures as an abnormal event rather than an emergency.

Response:

Refer to Concern 1.

23. **Finding from page 28**

Some important operator actions required to mitigate the event are not specified in AP/10. Among these are reduction in load, monitoring of critical safety functions, isolation of the affected generator, depressurization by dumping steam to the condenser and offsite dose calculation prior to dumping steam to the condenser from the affected generator.

Response:

Refer to Concerns 2, 5, 6, 8 and 12. Also SPDS is monitored on transfers out of EP/1 "Safety Injection". Since we have changed AP/10 so the high head injection line isolation valves can not be manually opened, for a leak greater than the capacity of the normal charging line safety injection will be manually initiated and the operator is directed to EP/01.

24. **Finding from page 29**

The transitions from AP/10 to other procedures are lacking in detail or not identified at all. Among these are:

- AP/01, "Rx trip" procedure
- OP/02, "Controlling Procedure for Unit Shutdown"
- EP/4.1, "SGTR Cooldown Using Steam Dump"
- EP/4.2, "SGTR Cooldown Using Backfill"
- EP/4.3, "SGTR Cooldown Using Blowdown"

Response:

Refer to Concerns 4 and 8.

25. Finding from page 29

AP/10 and EP/4.3 contain steps directing the operator to use the Blowdown Recycle System which could potentially result in establishing an unmonitored release pathway.

Response:

AP/10 now directs the operator to utilize one of the three steam generator tube rupture subprocedures to cooldown the rupture steam generator. OP/1/A/6250/08 "Steam Generator Blowdown" will be changed to have health physics personnel take grab samples if the BB Recycle Hx is in use. EP/4.3 places the BB Recycle Hx in service per OP/08. The BB Recycle Hx's on both Unit 1 and Unit 2 were taken out of service and red tagged to the Operations Superintendent on 4/21/89. These Hx's will be placed back in service after the procedures are changed. Operations and Radiation Protection procedures will be changed by May 5, 1989.

26. Finding from page 29

AP/10 does not required an assessment of offsite dose prior to dumping steam from the affected generator to the condenser.

Response:

See Concern 12.

27. Concern from page 29

System crossties caused increased radiological problems.

Response:

An additional Enclosure, Enclosure 6 "Minimizing Secondary Side Contamination" has been added to AP/10 to minimize (as much as possible) secondary side contamination due to system crossties. See Attachment 1, pages 39, 40 and 41 of 41.

28. Finding from page 29

Operators were not knowledgeable of two important provisions for cooling down and depressurizing in unusual situations. Specifically, these were (1) boron concentration required for intermediate temperature cooldown and, (2) procedural option to depressurize to 750 psi before cooling down.

Response:

See Concerns 13 and 14.

29. Finding from page 29

EP/04 does not isolate blowdown on the ruptured generator. This is a significant safety-related deviation.

Response:

See Concern 6.

30. **Finding from page 29**

The overall mitigative strategy used to deal with this >500 gpm tube rupture deviates substantially from the WOG Emergency Response Guidelines.

Response:

See Concern 1.

In addition to the responses on the AIT Report, the commitment on procedural enhancements from the April 13, 1989 Duke Power presentation to the NRC staff have been included. Attachment 6 has those commitments. The procedure changes which were made previously are in Attachment 5. The procedure enhancement to AP/10 which were scheduled to be completed by May 1, 1989 are done and are shown on Attachment 1. The procedure enhancements which were scheduled for a June 30, 1989 completion are working now.

List of Attachments

Attachment 1 - Revised AP/1/A/5000/10 "NC System Leakage Within The Capacity of Both NV Pumps"

Attachment 2 - Selected Pages From The Westinghouse Owners' Group Emergency Response"

Attachment 3 - Selected Pages from EP/1/A/5000/01 "Safety Injection

Attachment 4 - EP/1/A/500/10 "Critical Safety Function Status Trees"

Attachment 5 - Selected Pages From OP/1/A/6100/02 "Controlling Procedure for Unit Shutdown" and OP/0/A/6100/06 "Reactivity Balance Calculation"

Attachment 6 - Procedural Commitments Made in the April 13, 1989 Washington, D.C. Duke Power Presentation to the NRC Staff

Attachment 7 - Superceded Copy of AP/1/A/5000/10 and Selected Pages From the Superceded OP/1/A/6100/02

Attachment 1

Revised AP/1/A/5000/10 "NC System Leakage Within The Capacity of Both NV Pumps.

Form 34731 (R5-88)

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No. AP/1/A/5500/10
Change(s) 0 to 0 Incorporated

PREPARATION

(2) Station McGuire Nuclear Station

(3) Procedure Title NC System Leakage Within The Capacity of Both NV Pumps

(4) Prepared By Len Firebaugh Date 3/1/89

(5) Reviewed By Michael Weiner Date 4/26/89

Cross-Disciplinary Review By N/R MN

(6) Temporary Approval *(if necessary)*

By (SRO) Date

By Date

(7) Approved By Bruce Travis Date 4/26/89

(8) Miscellaneous

Reviewed/Approved By ETQS mw 4/26/89 Date

Reviewed/Approved By Date

(9) Comments (For procedure reissue indicate whether additional changes, other than previously approved changes, are included. Attach additional pages, if necessary.)
Additional Changes Included. ☒ Yes
☐ No

(10) Compared with Control Copy Date

(11) Requires change to FSAR not identified in 10CFR50.59 evaluation? ☐ Yes
If "yes", attach detailed explanation. ☒ No

Completion

(12) Date(s) Performed

(13) Procedure Completion Verification

☐ Yes ☐ N/A Check lists and/or blanks properly initialed, signed, dated or filled in N/A or N/R, as appropriate?

☐ Yes ☐ N/A Listed enclosures attached?

☐ Yes ☐ N/A Data sheets attached, completed, dated and signed?

☐ Yes ☐ N/A Charts, graphs, etc. attached and properly dated, identified and marked?

☐ Yes ☐ N/A Procedure requirements met?

Verified By Date

(14) Procedure Completion Approved Date

(15) Remarks *(attach additional pages, if necessary)*

TABLE OF CONTENTS

A. Purpose

This procedure covers the required operator actions for NC leakage greater than Tech Specs but where the Charging Pumps are capable of maintaining Pzr water level and the Pzr heaters are capable of maintaining system pressure under the following conditions:

Case I   Steam Generator Tube Leakage

Case II   NC System Leakage

Form 34913 (8-82)

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case I Steam Generator Tube Leakage	PAGE NO. 2 OF 23

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

B. Symptoms

- "1EMF 33, Cond Air Eject Exh Hi Rad" alarm
- "1EMF 34, S/G Sample Hi Rad" alarm
- "1EMF 24, 25, 26, 27 S/G A, B, C, D Steamline Hi Rad" alarm
- Increase in frequency of auto makeup to VCT
- Feedw ter flow and CF Reg valve position indication decreasing in any S/G.

C. Immediate Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 1. Check Pzr level - AT OR INCREASING TO PROGRAMMED LEVEL.	**IF** level decreasing, **THEN** perform the following to maintain level: a. Ensure 1NV-238 (Charging Line Flow Control) opening in "Auto". b. Start additional NV Pump. c. Reduce letdown to 45 GPM orifice or isolate letdown if necessary. **IF** level decreases below 5%, **THEN** manually trip Reactor and initiate SI. **GO TO** EP/1/A/5000/01, SAFETY INJECTION.
_____ 2. Check Pzr pressure - AT OR INCREASING TO 2235 PSIG.	**IF** less than 2210 PSIG, **THEN** ensure backup heaters on. **IF** pressure approaches 1945 PSIG, **THEN** trip Reactor.

D. Subsequent Actions

CAUTION If Pzr level cannot be maintained, (less than 5% and decreasing) then SI should be manually initiated.

_____ 1. Announce occurrence on paging system.

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ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 2. Check Pzr level - STABLE OR INCREASING.	**IF** level decreasing with maximum charging flow, **THEN** manually trip Reactor and initiate SI. **GO TO** EP/1/A/5000/01, SAFETY INJECTION.

NOTE Load reduction rate should be determined based on leak rate flow and on ability to take the unit off line in a controlled manner.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 3. Begin unit load reduction to remove unit off line.	
_____ 4. **WHEN** "VCT Level" less than 16%, **THEN** open 1NV-221A or 222B (NV Pumps Suct From FWST).	
_____ 5. Begin emergency boration for SDM considerations during cooldown in step 11.	
_____ 6. **REFER TO** RP/0/A/5700/00, CLASSIFICATION OF EMERGENCY.	
_____ 7. Identify ruptured S/G: ♦ Check S/G levels - ANY INCREASING IN AN UNCONTROLLED MANNER OR ♦ Check 1EMF-24, 25, 26, 27, S/G A(B, C, D) Steamline Hi Rad monitors - ANY ABOVE NORMAL OR ♦ Check CF Flow - LOWER IN ANY S/G COMPARED TO ALL OR ♦ **PER** OP/1/A/6250/08, STEAM GENERATOR BLOWDOWN, ENCLOSURE 4.3.	Do not proceed until ruptured S/G is identified.

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ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 8. After unit is off line, isolate flow from ruptured S/G:	
a. Close MSIV and MSIV Bypass valves on ruptured S/G.	a. Close MSIV and MSIV Bypass valves on nonruptured S/G. Close the following to isolate the SM header: ♦ Condenser and Atmospheric Dump valves ♦ 1SM-14 (SM To CSAE) ♦ 1SM-15 (SM To 2ND STG RHTRS) ♦ 1AS-12 (SM To AS) ♦ 1TL-3 (SM To Stm Seal Isol). Dispatch operator to locally close: ♦ 1SP-1 (SM To CF Pump 1A Isol) ♦ 1SP-2 (SM To CF Pump 1B Isol).
b. Close steamline drain on ruptured S/G: ♦ 1SM-83, 84, 95, 101(A(B,C,D) SM Line Drain).	
c. Isolate blowdown on ruptured S/G.	
d. Dispatch operator to locally close valves on ruptured S/G: ♦ 1SA-1 (SM 1C To TD CA Pump Manual Isol) and 1SA-77 (SM 1C To TD CA Pump Loop Seal Isol) ♦ 1SA-2 (SM 1B To TD CA Pump Manual Isol) and 1SA-78 (SM 1B To TD CA Pump Loop Seal Isol).	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
e. Maintain ruptured S/G pressure - LESS THAN 1125 PSIG:	
1) Verify SM PORVs in "Auto" and closed.	1) <u>IF</u> SM PORV fails to close at 1092 PSIG, <u>THEN</u> close its SM PORV Isol valve.
2) Open MSIV Bypass valve to maintain pressure.	2) <u>IF</u> condenser not available, <u>THEN</u> verify SM PORV cycles to prevent opening safety valves.
_____ 9. Check ruptured S/G level:	
a. "S/G NR Lvl" - GREATER THAN 5%	a. Maintain feed flow to ruptured S/G until level greater than 5%.
b. Stop feed flow to ruptured S/G.	
_____ 10. Check intact S/G levels:	
a. "S/G NR Lvl" - GREATER THAN 5%.	a. Maintain total feed flow greater than 450 GPM until level greater than 5% in at least one intact S/G.
b. Control feed flow to maintain levels - AT NO LOAD.	

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ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION ♦ **Isolation of the ruptured S/G steam lines must be complete before continuing to step 11.**

♦ **Administrative cooldown rate of 50°F/hr may be exceeded during cooldown in step 11.**

NOTE ♦ Blocking the steamline low pressure SI signal will enable the steamline pressure rate, main steam isolation signal.

♦ Cooldown and depressurization in steps 11 and 12 should be performed concurrently to minimize break flow while maintaining subcooling.

_____ 11. Initiate NC System cooldown:

a. Maintain cooldown to ensure NC System stays 20°F subcooled until depressurization in step 12 is completed.

b. Dump steam to condenser from intact S/Gs.

Response not obtained: b. Dump steam from intact S/Gs SM PORVs.

c. **WHEN** "Pzr Press" less than 1955 PSIG, **THEN** verify "P-11 Pressurizer S/I Block Permissive" status light is lit.

1) Depress the following "Block" pushbuttons and verify "Blocked" lights - LIT:

♦ "Pzr SI Trn A (B) Block"

♦ "Stm Line SI Trn A (B) Block".

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ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 12. Depressurize NC System to minimize break flow:	
a. Normal Pzr spray - AVAILABLE	a. <u>IF</u> letdown in service, <u>THEN</u> use NV Aux spray. <u>IF</u> letdown not in service, <u>THEN</u> use one Pzr PORV. <u>IF</u> no Pzr PORVs available, <u>THEN</u> use NV Aux spray.
b. Depressurize until any of the following are satisfied before continuing with this procedure: ♦ NC pressure - LESS THAN RUPTURED S/G PRESSURE OR ♦ Ruptured S/G level - CONSTANT OR DECREASING OR ♦ Pzr level - GREATER THAN 95%.	
c. Close spray valves or PORV.	c. Stop NC Pump in loop with open spray valve or close Pzr PORV Isol valve for any PORV that will not close.

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ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 13. Check if charging flow can be reduced:	
a. Pzr level - GREATER THAN 25% AND INCREASING	a. Maintain charging flow until level greater than 25%.
b. Reduce charging flow to maintain Pzr level constant.	
_____ 14. Check VCT Makeup Control System:	
a. IF started, THEN stop emergency borating when SDM is adequate PER Data Book Table 6.5.	
b. Ensure makeup set for greater than NC System boron concentration.	
c. Ensure "NC Sys M/U Controller" in "Auto" and place "NC System Makeup" switch to "Start".	

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ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 15. Establish Letdown:	Establish excess letdown:
a. Open letdown line isolation valves: • 1NV-1A and 2A (NC L/D Isol To Regn Hx) • 1NV-7B (Letdown Cont Isol Outside).	1. Ensure open: • 1KC-305B (Excess Letdn Hx Sup Otsd Isol) • 1KC-315B (Excess L/D Hx Ret Hdr C/I Otsd) • 1NV-94A and 95B (NC Pmps Seal Ret C/I Inside/Otsd).
b. Place 1NV-124 (Letdown Press Control) in "Man" and close.	2. Place 1NV-27B (Excess L/D Hx Otlt 3-Way Cntrl) to "VCT" position.
c. Crack open 1NV-459A (A L/D Orifice Outlet Flo Cntrl) and adjust 1NV-124 to maintain "L/D Press" at 350 PSIG.	3. Open 1NV-24B and 25B (C NC Loop To Exs L/D Hx Isol).
d. **WHEN** pressure can be maintained without continual adjustment to 1NV-124, **THEN** slightly open 1NV-459A a little more while monitoring pressure.	4. Slowly open 1NV-26 (Excess L/D Hx Outlet Cntrl) to maintain "Excess L/D Hx Temp" less than 200°F.
e. Continue above procedure until pressure stabilizes quickly after 1NV-459A adjustments and desired flow is achieved.	
f. After letdown line is pressurized align letdown valves for desired flowrate: • 75 GPM, 1NV-458A (B L/D Orif Otlt Cont Isol) - OPEN/CLOSED • 45 GPM, 1NV-457A (C L/D Orif Otlt Cont Isol) - OPEN/CLOSED • Variable, 1NV-459A (A L/D Orif Otlt Cont Isol) - CLOSED.	
g. Verify "L/D Press" at 350 PSIG and place 1NV-124 in "Auto".	

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ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 16. Maintain stable plant conditions:	
a. NC pressure - STABLE	a. Operate Pzr heaters and sprays.
b. Pzr level - 25%	b. Control charging and letdown.
c. Intact S/G levels - AT NO LOAD	c. Control feed flow.
d. NC temperatures - STABLE.	d. Operate Condenser Dumps.
_____ 17. Minimize secondary side contamination:	
a. If available transfer AS header supply to Unit 2: 1) Close 1AS-9 (C Htr Bleed to AS) and 1AS-12 (SM To AS). 2) Open 1HM-95 (AS To A & B FWPT).	a. Dispatch operator to locally place Aux Electric Boiler in operation <u>PER</u> OP/1/A/6250/07B, AUX ELECTRIC BOILERS, Enclosure 4.2, then do the following: 1) Close 1AS-9 (C Htr Bleed to AS) and 1AS-12 (SM To AS). 2) Open 1HM-95 (AS To A & B FWPT).
b. <u>IF</u> B S/G ruptured, dispatch operator to locally close 1SM-85 (Stm Line 1B DH Drn Orifice Inlet).	
c. <u>REFER TO</u> Enclosure 6.	

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AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case I Steam Generator Tube Leakage	PAGE NO. 11 OF 23

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION
- **If any NC Pump is running, then the preferred cooldown method is EP/1/A/5000/4.2, SGTR COOLDOWN USING BACKFILL.**
- **If water may exist in main steamlines, then EP/1/A/5000/4.1 SGTR COOLDOWN USING STEAM DUMP should not be used.**
- **It is strongly recommended that the condenser be available if EP/1/A/5000/4.1, SGTR COOLDOWN USING STEAM DUMP is used. Otherwise an evaluation of using the ruptured S/G SM PORV must be made.**

____ 18. Continue plant cooldown to cold shutdown:

a. Select cooldown method for ruptured S/G:

- EP/1/A/5000/4.1, SGTR COOLDOWN USING STEAM DUMP

OR

- EP/1/A/5000/4.2, SGTR COOLDOWN USING BACKFILL

OR

- EP/1/A/5000/4.3, SGTR COOLDOWN USING BLOWDOWN.

b. REFER TO OP/1/A/6100/02, CONTROLLING PROCEDURE FOR UNIT SHUTDOWN, Enclosure 4.2 and perform applicable steps.

END

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AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II NC System Leakage	PAGE NO. 12 OF 23

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

B. Symptoms

- Increase in frequency of auto makeup to VCT
- Increased leakrate results from PT/1/A/4150/01B, REACTOR COOLANT LEAKAGE CALCULATIONS
- Increased radiation from any of the following:
  - "1EMF38 Containment Part Hi Rad" alarm
  - "1EMF39 Containment Gas Hi Rad" alarm
  - "1EMF40 Containment Iodine Hi Rad" alarm
  - "EMF41 Aux Bldg Vent Hi Rad" alarm
  - "1EMF46 A(B) Train A(B) KC Hi Rad" alarm.
- Increased levels in any of the following:
  - "Cont Flr/Eqp Sump A(B) Level"
  - ND and NS Sump
  - NCDT
  - RHT
  - KC Surge Tank
  - PRT.
- Increased temperatures on any of the following:
  - "Pzr PORV Disch Hi Temp" alarm
  - "Pzr Safety Discharge Hi Temp" alarm
  - "Rx Vessel Flange Leak Off Hi Temp" alarm
  - "Letdown Relief Hi Temp" alarm.
  - PRT
  - Containment.
- "NC Pump A(B,C,D) Thermal Barrier Outlet Hi Flow" computer alarm
- Letdown or charging line flow or pressure abnormal.

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ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Immediate Actions	
_____ 1. Check Pzr level - AT OR INCREASING TO PROGRAMMED LEVEL.	**IF** level decreasing, **THEN** perform the following to maintain level: a. Ensure 1NV-238 (Charging Line Flow Control) opening in "Auto". b. Start additional NV Pump. c. Reduce letdown to 45 GPM orifice or isolate letdown if necessary. **IF** level decreases below 5%, **THEN** manually trip Reactor and initiate SI. **GO TO** EP/1/A/5000/01, SAFETY INJECTION.
_____ 2. Check Pzr pressure - AT OR INCREASING TO 2235 PSIG.	**IF** less than 2210 PSIG, **THEN** ensure backup heaters on. **IF** pressure approaches 1945 PSIG, **THEN** trip Reactor.

D. Subsequent Actions

CAUTION **If Pzr level cannot be maintained, (less than 5% and decreasing) then Safety Injection should be manually initiated.**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 1. Announce occurrence on paging system.	
_____ 2. Check Pzr level - STABLE OR INCREASING.	**IF** level decreasing with maximum charging flow, **THEN** manually trip Reactor and initiate SI. **GO TO** EP/1/A/5000/01, SAFETY INJECTION.
_____ 3. **REFER TO** RP/0/A/5700/00, CLASSIFICATION OF EMERGENCY.	
_____ 4. **WHEN** "VCT Level" less than 16%, **THEN** open 1NV-221A or 222B (NV Pumps Suct From FWST).	

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ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 5. Check if Containment ventilation isolation required:	
a. 1EMF 38, 39 or 40 - IN ALARM	a. **GO TO** step 6.
b. Stop VP Fans.	
c. Stop any VQ release in progress.	
_____ 6. Check "EMF41 Aux Bldg Vent Hi Rad" - IN ALARM	**GO TO** step 7.
a. Verify "1ABF-D-3 VA Filter Exh Bypass Dmpr Trn A(B)" closed lights - LIT.	
b. Verify "2ABF-D-3 VA Filter Exh Bypass Dmpr Train A(B)" closed lights - LIT.	
_____ 7. Check "1EMF-46 KC Hx Outlet" - IN ALARM	**GO TO** step 8.
a. Dispatch operator to locally verify 1KC-122 (KC Surge Tank Vent) - CLOSED.	

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ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 8. Check "NC Pmp A(B, C, D) Therm Bar KC Outlet Flow" computer alarm - IN ALARM	GO TO step 9.
a. Verify the following valve closes on affected pump: • A, 1KC-394A (A NC Pump Therm Bar Otlt) OR • B, 1KC-364B (B NC Pump Therm Bar Otlt) OR • C, 1KC-345A (C NC Pump Therm Bar Otlt) OR • D, 1KC-413B (D NC Pump Therm Bar Otlt).	
b. Verify "1A(B, C, D) NC Pump L/B Temp" remains less than 225°F.	b. IF greater than 225°F, THEN trip NC Pump.
_____ 9. IF required to stop leak, THEN isolate letdown:	GO TO step 10.
a. **Close 1NV-1A and 2A (NC L/D Isol to Regen Hx) to isolate normal letdown.**	
b. **Verify charging flow -** DECREASING TO MINIMUM.	b. Take manual control and reduce charging flow to 32 GPM.
c. Establish excess letdown PER OP/1/A/6200/01, CHEMICAL AND VOLUME CONTROL, Enclosure 4.8.	
d. Power operation may continue as long as NC System activity and chemistry requirements are met.	

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ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 10. IF required to stop leak, THEN isolate normal charging.	GO TO step 11.
a. Isolate letdown:	
1) Close 1NV-1A and 2A (NC L/D Isol To Regen Hx).	
b. Isolate charging:	
1) Close 1NV-244A and 245B (Charging Line Cont Isol Otsd)	
2) Manually throttle 1NV-238 (Charging Line Flow Control) to maintain 8 GPM seal injection flow per NC Pump.	
c. Establish excess letdown PER OP/1/A/6200/01, CHEMICAL AND VOLUME CONTROL, Enclosure 4.2.	
d. Power operation may continue as long as NC System activity and chemistry requirements are met.	

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ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 11. Attempt to identify and isolate leak:	
a. Check "Pzr PORV Disch Hi Temp" - IN ALARM	a. <u>GO TO</u> step b.
1) Verify Pzr PORV's - CLOSED	1) Close Pzr PORV's.
2) Determine which valve is leaking by monitoring "PORV Relief Valve Temp" while cycling a PORV isolation and its sample valve one at a time: ♦ 1NC-33A (Pzr PORV Isol) and 270 (Pzr Relief Hrd Sample Isol) ♦ 1NC-35B (Pzr PORV Isol) and 269 (Pzr Relief Hdr Sample Isol) ♦ 1NC-31B (Pzr PORV Isol) and 271 (Pzr Relief Hrd Sample Isol).	
b. Check "Cold Leg Accumulator Level" - INCREASING	b. <u>GO TO</u> step c.
1) Close Accumulator isolation valve OR Drain accumulator <u>PER</u> OP/1/A/6200/09, ACCUMULATOR OPERATION, Enclosure 4.2.	
c. Check "Pzr Relief Tank Level (Temp)" - INCREASING ABOVE NORMAL	c. <u>GO TO</u> step d.
1) Check inputs to PRT <u>PER</u> Enclosure 1.	

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ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
d. Check NCDT level or temperature - INCREASING ABOVE NORMAL	d. **GO TO** step e.
1) Check inputs to NCDT **PER** Enclosure 2.	
e. Check "Cont Flr/Eqp Sump A(B) Level" - INCREASING ABOVE NORMAL	e. **GO TO** step f.
1) Check inputs to sumps **PER** Enclosure 3.	
f. Check inputs to Aux Building Sumps from NV System **PER** Enclosure 4.	f. **GO TO** step g.
g. Check ND System - IN SERVICE	g. **GO TO** step 12.
1) Check inputs to Aux Building Sumps from ND System **PER** Enclosure 5.	
_____ 12. **IF** leak can not be isolated **AND** unit shutdown is required, **THEN** notify NRC via red phone **PER** RP/0/A/5700/10, NRC IMMEDIATE NOTIFICATION REQUIREMENTS.	**IF** leak is isolated, **THEN** consult Unit 1 Operations Manager for further actions and end this procedure.

NOTE Load reduction rate should be determined based on leak rate flow and on ability to take the unit off line in a controlled manner.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 13. Begin unit load reduction to remove unit off line.	
_____ 14. Check NC Pumps - ALL RUNNING.	Start all available NC Pumps **PER** OP/1/A/6150/02A, NC PUMP OPERATION.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE
- Blocking the steamline low pressure SI signal will enable the steamline pressure rate, main steam isolation signal.
- If an NC Pump is running, administrative cooldown rate of 50°F/HR may be exceeded in the following steps.

_____ 15. Begin NC System cooldown to 200°F:

a. **REFER TO** OP/1/A/6100/02, CONTROLLING PROCEDURE FOR UNIT SHUTDOWN, Enclosure 4.2 and perform applicable steps.

b. Maintain cooldown rate - LESS THAN 100°F/HR (50°F/HR with no NC Pumps running)

c. Dump steam to condenser.

RNO: c. Dump steam using SM PORVs.

d. Control CA flow to maintain "S/G NR Lvl" - AT NO LOAD.

e. Maintain NC System boron concentration - GREATER THAN SDM REQUIREMENTS OF DATA BOOK TABLE 6.5.

f. **WHEN** "Pzr Press" less than 1955 PSIG, **THEN** verify "P-11 Pressurizer S/I Block Permissive" status light is lit.

RNO: f. Continue with step 16.

1) Depress the following "Block" pushbuttons and verify "Blocked" lights - LIT:

- "Pzr SI Trn A (B) Block"
- "Stm Line SI Trn A (B) Block".

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AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II NC System Leakage	PAGE NO. 20 OF 23

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 16. Check if cold leg accumulators should be isolated:	
a. NC System subcooling - GREATER THAN 0°F AND NC pressure - LESS THAN 1000 PSIG	a. Continue with this procedure while monitoring pressure and subcooling. **WHEN** both conditions are met, **THEN** do step b.
b. Place the following switches to "Enable" and close the respective valve: ♦ "Pwr Discon For 1NI-54A" ♦ "Pwr Discon For 1NI-76A" ♦ "Pwr Discon For 1NI-65B" ♦ "Pwr Discon For 1NI-88B".	b. Vent any unisolated accumulator: 1) Open isolation valve on affected accumulator: ♦ 1NI-50 (A CL Accum N2 Supply Isol) ♦ 1NI-61 (B CL Accum N2 Supply Isol) ♦ 1NI-72 (C CL Accum N2 Supply Isol) ♦ 1NI-84 (D CL Accum N2 Supply Isol) 2) Open 1NI-83 (CL Acc N2 Hdr Atmos Vent Isol).
____ 17. Check if one NV Pump should be stopped:	
a. NC System subcooling - GREATER THAN 50°F AND Pzr level - GREATER THAN 5%	a. Continue dumping steam. **WHEN** subcooling greater than 5(**AND** Pzr level greater than 5% **THEN** **GO TO** step b.
b. Both NV Pumps - RUNNING	b. **GO TO** step 19.
c. Stop one NV Pump.	

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ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 18. Check NC System conditions:	
a. Subcooling - GREATER THAN 0°F AND Pzr level - GREATER THAN 5%	a. Restart NV Pump and continue dumping steam. **WHEN** NC System subcooling greater than 75°F **AND** Pzr level greater than 5% **THEN RETURN TO** step 17.
____ 19. Adjust charging flow to maintain NC System subcooling and Pzr level during cooldown.	
____ 20. Establish letdown:	Establish excess letdown:
a. Open letdown line isolation valves: ♦ 1NV-1A and 2A (NC L/D Isol To Regen Hx) ♦ 1NV-7B (Letdown Cont Isol Outside).	1) Ensure open: ♦ 1KC-305B (Excess Letdn Hx Sup Otsd Isol) ♦ 1KC-315B (Excess L/D Hx Ret Hdr C/I Otsd) ♦ 1NV-94A (NC Pumps Seal Ret C/I Inside) ♦ 1NV-95B (NC Pumps Seal Ret C/I Otsd).
b. Place 1NV-124 (Letdown Press Control) in "Man" and close.	
c. Crack open 1NV-459A (A L/D Orif Outlet Flow Cntrl) and adjust 1NV-124 to maintain "L/D Press" at 350 PSIG.	2) Place 1NV-27B (Excess L/D Hx Otlt 3-Way Cntrl) to "VCT" position.
d. **WHEN** pressure can be maintained without continual adjustment to 1NV-124, **THEN** slightly open 1NV-459A a little more while monitoring pressure.	3) Open 1NV-24B and 25B (C NC Loop To Exs L/D Hx Isol).
e. Continue above procedure until pressure stabilizes quickly after 1NV-459A adjustments and desired flow is achieved.	4) Slowly open 1NV-26 (Excess L/D Hx Outlet Cntrl). Adjust valve to maintain "Excess L/D Hx Temp" less than 200°F.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
f. After letdown line is pressurized align letdown valves for desired flowrate: ♦ 75 GPM - 1NV-458A (B L/D Orif Otlt Cont Isol) - OPEN/CLOSED ♦ 45 GPM - 1NV-457A (C L/D Orif Otlt Cont Isol) - OPEN/CLOSED ♦ Variable - 1NV-459A (A L/D Orif Otlt Cont Isol) - CLOSED.	
g. Verify "L/D Press" at 350 PSIG and place 1NV-124 in "Auto".	
_____ 21. Control Pzr pressure:	
a. Energize Pzr heaters and operate normal spray to maintain pressure within Data Book Curve 1.6.	a. <u>IF</u> letdown in service, <u>THEN</u> use 1NV-21A (NV Spray To Pzr Isol). <u>IF</u> letdown not in service, <u>THEN</u> use one Pzr PORV.
_____ 22. Maintain NC System subcooling - GREATER THAN 50°F.	Establish 50°F subcooling: a. Limit NC System cooldown rate to less than 100°F/HR (50°F/HR if no NC Pumps running). b. Dump steam to condenser OR Dump steam with SM PORVs. <u>IF</u> cooldown is not adequate to restore subcooling, <u>THEN</u> increase NC System pressure.

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ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 23. Check if ND System can be placed in service:	
a. NC hot leg temperatures - LESS THAN 350°F AND NC pressure - LESS THAN 385 PSIG	a. Continue dumping steam. Do not proceed until conditions met.
b. Place ND System in service <u>PER</u> OP/1/A/6200/04, RESIDUAL HEAT REMOVAL, Enclosure 4.1.	b. Continue dumping steam while trying to place ND System in service.
____ 24. Prior to going below 300°F place low temperature over pressure protection system in service:	
a. NC pressure - LESS THAN 325 PSIG	
b. Place "PORV Overpress Protection Select 1NC-34A (23B)" switch to "Low Press".	
c. Verify 1NI-430A and 431B (N_2 To 1NC-34A (32B) From A(B) CL Accum) - OPEN.	
____ 25. Use ND System and S/Gs to continue cooldown to less than 200°F.	
____ 26. <u>WHEN</u> NC System cooldown complete, <u>THEN</u> stop NC Pumps <u>AND</u> depressurize to stop break flow.	
____ 27. Evaluate long term plant status:	
a. Further actions should be at the discretion of the TSC.	

<u>END</u>

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACTIY OF BOTH NV PUMPS Case II - Enclosure 1 Possible NC System Leakage Paths To PRT	PAGE NO. 1 OF 2

Valve Number	Nomenclature	Position	Initial
	POSSIBLE NC SYSTEM LEAKAGE PATHS TO PRT		
	OUTSIDE CONTAINMENT		
	OUTSIDE CONTAINMENT		
1ND-56	ND HX 1A OUTLET TO NI SYSTEM COLD LEG INJECTION SAFETY RELIEF		
1ND-61	ND HX OUTLET TO NI SYSTEM HOT LEG INJECTION SAFETY RELIEF		
1ND-64	ND HX 1B OUTLET TO NI SYSTEM COLD LEG INJECTION SAFETY RELIEF		
1NS-2	NS PUMP 1B SUCTION SAFETY RELIEF		
1NS-19	NS PUMP 1A SUCTION SAFETY RELIEF		
1NI-102	SAFETY INJECTION PUMPS SUCTION HDR SAFETY RELIEF		
1NI-119	SAFETY INJECTION PUMP 1A DISCHARGE SAFETY RELIEF		
1NI-151	SAFETY INJECTION PUMP 1B DISCHARGE SAFETY RELIEF		
1NI-161	SAFETY INJECTION PUMPS COLD LEG INJECTION HDR SAFETY RELIEF		
1NV-229	CENTRIFUGAL CHARGING PUMPS SUCTION HDR SAFETY RELIEF		
	INSIDE CONTAINMENT		
1NC-1	PZR RELIEF VALVE		
1NC-2	PZR RELIEF VALVE		
1NC-3	PZR RELIEF VALVE		
1NC-32B	PZR PORV		
1NC-34A	PZR PORV		
1NC-36B	PZR PORV		
1NC-43	PRESSURIZER #1 VENT		
1NC-119	PRESSURIZER #1 SEAL LOOP DRAIN HEADER		

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACTIY OF BOTH NV PUMPS Case II - Enclosure 1 Possible NC System Leakage Paths To PRT	PAGE NO. 2 OF 2

VALVE CHECKLIST

VALVE NUMBER	NOMENCLATURE	INITIAL
1NC-272A,C	TRN 1A HEAD VENT TO PRT ISOL	
1NC-274B	TRN 1B HEAD VENT TO PRT ISOL	
1ND-3	NC LOOP 3 DISCHARGE TO ND SYSTEM SAFETY RELIEF	
1NV-6	LETDOWN LINE SAFETY RELIEF	
1NV-93	NC PUMPS SEAL RETURN HDR SAFETY RELIEF	

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II - Enclosure 2 Possible NC System Leakage Paths To NCDT	PAGE NO. 1 OF 1

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
Possible NC System Leakage Paths To NCDT				
1NV-27B	Excess L/D Hx Otlt 3-Way Cntrl	RB Pipechase 105°	VCT	
1NI-224	Accumulator 1A Drain Isol	RB 725' 40°	Closed	
1NI-226	Accumulator 1B Drain Isol	RB 725' 140°	Closed	
1NI-228	Accumulator 1C Drain Isol	RB 725' 220°	Closed	
1NI-230	Accumulator 1D Drain Isol	RB 725' 317°	Closed	
1NB-352	Reactor Makeup Water Storage Tank #1 Outlet Relief To NCDT			
NC Pump	1A #3 Seal			
NC Pump	1A Standpipe			
NC Pump	1B #3 Seal			
NC Pump	1B Standpipe			
NC Pump	1C #3 Seal			
NC Pump	1C Standpipe			
NC Pump	1D #3 Seal			
NC Pump	1D Standpipe			
	Valve Steam Leakoff from RB valves			
	Rx Vessel Head O Ring Seal			

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II - Enclosure 3 Possible NC System Leakage Paths to Containment Sumps	PAGE NO. 1 OF 1

VALVE CHECKLIST

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
1NM-67	PZR Sample Hdr Cont Pent Relief	755', 120°		
1NM-68	NC Hot Leg Sample Header Cont Pent Relief			
1NM-69	NI Accumulators Sample Hdr Cont Pen Relief	730', 115°		
1NV-102	Excess Letdown Hx #1 Tube Drain	Pipechase 115° 6' up	Closed	
1NV-108	Regenerative Hx #1 Overflow	Pipechase 105° 5' up	Closed	
1NV-110	Regenerative Hx #1 Drain	Pipechase 105° 5' up	Closed	

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II - Enclosure 4 Possible NV System Leakage Paths in Auxiliary Building	PAGE NO. 1 OF 6

VALVE CHECKLIST

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL

NOTE This Enclosure is to be used as a guide. Consideration should be given to any recent change in NV System alignment.

Check Seal Leakoff on following valves:

ND & NS ROOMS SUMP

1NV-95B	NC Pumps Seal Ret C/I OTSD	744' Midget Hole		
1NV-127A	L/D Hx Outlet 3-Way Temp Cntrl	NC FILTER ROOM		
1NV-137A	NC Filters OTLT 3 Way Cntrl	Outside VCT Rm. So. Wall		
1NV-141A	VCT Outlet Isolation	OTSD S Wall of VCT under grating		
1NV-142B	VCT Outlet Isol	OTSD SE Wall of VCT under grating		
1NV-803	PD Pump Outlet Isol	722' S. of PD Pump		
1NV-219	PD Pump Disch Isol	PD Pump Rm		
1NV-240	Regen Hx Tube Inlet Cntrl Isol	722 HH-59 & JJ-60		
1NV-241	Seal Inj Flow Control	Above BW Pumps		
1NV-242	Regen Hx Tube Side Inlt Cntrl Isol	Above BW Pumps		
1NV-243	Regen Hx Tube Side Inlt Cntrl Bypass	Above BW Pumps		

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPAC TY OF BOTH NV PUMPS Case II - Enclosure 4 Possible NV System Leakage Paths in Auxiliary Building	PAGE NO. 2 OF 6

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
1NV-244A	Charging Line Cont Isol OTSD	Above BW Pumps		
1NV-245B	Charging Line Cont Isol OTSD	West of BW Pumps		
1NV-431	Seal Water Inj Filters Bypass	A Seal Inj Rm		
1NV-230	Cent Charging Pump B Suct	726' SE of 1B CCP 12' Off floor		
1NV-224	Cent Charging Pump A Suct.	726' HH-57 & JJ-58 W of 1B CCP 12' Off Floor		
1NV-804	Cent Charging Pump B Outlet Isol	Right of 1B CCP		
1NV-232	Cent Charging Pump B Disch	724' NW of 1B CCP		
1NV-226	Cent Charging Pump A Disch	726' NE of 1B CCP		
1NV-802	Cent Charging Pump A Outlet Isol	NE 5' Above 1A CCP		
1NV-235	Cent Charging Pump B To Seal Inj Filter	NW of 1B CCP		
1NV-236	Cent Charging Pump A To Seal Inj Filter	NE of 1A CCP		
1NV-237	Cent Charging Pumps Disch To Control Isol	N of PD Pump		
1NV-238	Charging Line Flow Control	N of PD Pump		
1NV-239	Cent Charging Pumps Disch			

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II - Enclosure 4 Possible NV System Leakage Paths in Auxiliary Building	PAGE NO. 3 OF 6

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
	Control Isol			
1NV-347	NR System Flow Control	Seal Inj Filter Room		
1NV-121	ND Letdown Control			
1NV-221A	NV Pumps Suct From FWST	20' N of BW Pump		
1NV-222B	NV Pumps Suct From FWST	20' N of BW Pump		
	RECYCLE HOLDUP TANK			
1NV-7B	Letdown Cont Isol Outside			
1NV-8	L/D Reheat Hx Tubeside	SE of L/D Hx		
	Back pressure Cntrl Isol			
1NV-9	L/D Reheat Hx Tubeside	W of L/D Hx		
	Back pressure Cntrl Isol			
1NV-10	L/D Reheat Hx Tubeside	W of L/D Hx		
	Back Pressure Cntrl Isol			
1NV-11	L/D Reheat Hx Tubeside	SW of L/D Hx		
	Back Pressure Cntrl Isol			
1NV-476	LP Letdown Control Inlet	S of L/D Hx		
	Isol			
1NV-124	Letdown Press Control	L/D Hx Rm		
1NV-477	LP Letdown Control	L/D Hx Rm		
	Outlet Isol			

Check vent and drain boundary valves closed:

	WASTE DRAIN TANK			

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II - Enclosure 4 Possible NV System Leakage Paths in Auxiliary Building	PAGE NO. 4 OF 6

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
1NV-184	Letdown Hx Tube Drain To WDT	L/D Hx Rm	CLOSED	
1NV-205	Seal Water Filter Drain To WDT	Seal Ret Filter Rm	CLOSED	
1NV-272	PD Pump Drain To WDT	Otsd PD Pump Rm	CLOSED	
1NV-310	Seal Water Inj Filters Drain To WDT	B Seal Inj Rm	CLOSED	
1NV-299	Charging Pump B Drain To WDT	E of 1B CCP	CLOSED	
1NV-330	NC Filter Drain To WDT	E of B NC Filters	CLOSED	
1NV-356	Mixed And Cation Bed Demin Outlet Line Drain To WDT	A Mixed Bed Rm	CLOSED	
	WASTE EVAPORATOR FEED TANK			
1NV-181	Letdown Hx Tube Overflow	L/D Hx Rm	CLOSED	
1NV-185	Letdown Hx Tube Drain To WEFT	LD Hx Rm	CLOSED	
1NV-145	VCT Outlet Drain	Below VCT	CLOSED	
1NV-210	Seal Water Hx Tube Overflow		CLOSED	
1NV-204	Seal Water Filter Drain To WEFT		CLOSED	
1NV-215	Seal Water Hx Tube Drain TO WEFT	Seal Water Hx Rm	CLOSED	
1NV-309	Seal Water Inj Filters	B Seal Inj Filter Rm	CLOSED	

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II - Enclosure 4 Possible NV System Leakage Paths in Auxiliary Building	PAGE NO. 5 OF 6

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
	Drain To WEFT			
1NV-329	NC Filters Drain To WEFT	B NC Filter Rm	CLOSED	
1NV-335	Mixed Bed Demin A Backflush			
	Drain			
1NV-333	Mixed Bed Demin A Backflush		CLOSED	
	Outlet Isol			
1NV-340	Mixed Bed Demin B	B Mixed Bed Rm	CLOSED	
	Backflush Outlet Isol			
1NV-373	Mixed Bed Demin B	B Mixed Bed Rm	CLOSED	
	Outlet Line Drain			
1NV-365	Cation Bed Demin	733 Pipechase between NR &	CLOSED	
	Sluicing Resin Isol	NV DIM		
1NV-354	Mixed Bed Demin A Outlet	A Mixed Bed Rm	CLOSED	
	Line Drain			
1NV-366	Cation Bed Demin Outlet	Cation Bed Rm	CLOSED	
	Line Drain			
1NV-357	Mixed & Cation Bed	A Mixed Bed Rm	CLOSED	
	Demins Outlet Line Drain			
	To WEFT			
	WASTE EVAPORATOR FEED TANK SUMP A			
1NV-296	Charging Pump B Overflow		CLOSED	
1NV-300	Charging Pump B Drain To		CLOSED	
	WEFT Sump A			
1NV-285	Charging Pump A Overflow		CLOSED	
1NV-289	Charging Pump A Drain To		CLOSED	
	WEFT Sump A			

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II - Enclosure 4 Possible NV System Leakage Paths in Auxiliary Building	PAGE NO. 6 OF 6

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
	SPENT RESIN TANK			
1NV-349	Mixed Bed Demin A Sluicing Resin Isol		CLOSED	
1NV-350	Mixed Bed Demin A Sluicing Resin Isol		CLOSED	

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS CASE II - Enclosure 5 Possible ND System Leakage Paths in Auxiliary Building	PAGE NO. 1 OF 2

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
	ND & NS ROOMS SUMP Check seal leakoff on following valves			
1ND-4B	B ND Pmp Suct From FWST Or NC	Aux 695' FF-59 & GG-60		
1ND-19A	A ND Pmp Suct From FWST Or NC	Aux 695' GG-59 & HH-60		
1ND-9	ND Pump B Disch	N of Pump 12' up		
1ND-24	ND Pump A Disch	W of Pump 12' up		
1ND-26	ND Hx A Inlet	E of Hx 6' up		
1ND-14	B ND Hx Outlet	Aux 733' LL-61		
1ND-29	A ND Hx Outlet	Aux 733' LL-60 & MM-61		
1ND-30A	Train 1A ND To Hot Leg Isol	Aux 733' LL-60 & MM-61		
1ND-58A	Train 1A ND To NV & NI Pumps	Aux 733' LL-60 & JJ-61		
1ND-15B	Train 1B ND To Hot Leg Isol	Aux 733' KK-60 & LL-61		
1ND-35	ND To FWST Isol	15' E of KK-59, 12' up		
1ND-34	A & B ND Hx Bypass	Aux 733 KK-60 & LL-61		
1ND-33	A ND Hx Bypass	Aux 733 LL-60 & MM-61		
1ND-18	B ND Hx Bypass	Aux 733 KK-60 & LL-61		
1ND-11	ND Hx B Inlet	W of Hx 4' up		
1NI-173A	Train 1A ND To A & B CL	Aux 733' FF-59 & GG-60		
1NI-178B	Train 1B ND To C&D CL	Aux 733' HH-60 & JJ-61		
1NI-184B	RB Sump To Train 1B ND & NS	Aux 716' EE-58 & FF-59		

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS CASE II - Enclosure 5 Possible ND System Leakage Paths in Auxiliary Building	PAGE NO. 2 OF 2

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
1NI-185A	RB Sump To Train 1A ND & NS	Aux 716' FF-59 & GG-60		
	WASTE EVAPORATOR FEED TANK Check Drain Boundary Valves Closed			
1ND-52	ND HX A Drain Hdr	S of Hx	Closed	
1ND-46	ND Hx B Drain Hdr	S of Hx	Closed	
ND & NS ROOMS SUMP				
1ND-51	ND Pump A Drain Hdr	SW of Pump	Closed	
1ND-45	ND Pump B Drain Hdr	S of Pump	Closed	
1ND-69	ND & NS System Drain	RB 860' Rx Dome	Closed	

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Enclosure 6 Minimizing Secondary Side Contamination	PAGE NO. 1 OF 3

1. Notify the following to initiate their S/G tube leak response procedures:

   HP Shift (4282)

   RDW Shift (4305)

   CT Lab (4362).

2. Minimize draining into Turbine Building Sump by dispatching operator to locally perform the following:

   A) Secure CSAE drains to TB sump as follows:

   Open 1CS-26 (CSAE After Condenser Drn Isol)

   Close: 1ZJ-24 (CSAE 1A After Condenser Lo Pt Drn)

   1ZJ-25 (CSAE 1B After Condenser Lo Pt Drn)

   1ZJ-26 (CSAE 1C After Condenser Lo Pt Drn).

   B) Align WZ Sump Pumps to pump to Unit 2 only:

   1) In WZ Sump A place Pump A in "Off" and Pump B in "Auto".

   2) In WZ Sump B place Pump A in "Auto" and Pump B in "Off".

   3) <u>IF</u> either Unit 2 pump is out of service, <u>THEN</u> manually pump only enough water to Unit 1 to keep Hi level alarm cleared.

   C) Secure any other components draining into TB Sump.

   D) Open 1CS-62 (NB + WL Cond To Unit 2 CST) and close 1CS-61 (NB + WL Cond To Unit 1 CST)

   E) Align 1CB-197 (Aux Electric Boiler Blowdown 3 Way Divert) to Unit 2 Turbine Building Sump when it has been determined that blowdown is not contaminated.

   F) Align Aux Electric Boiler Feed Pumps miniflow to Unit 2 CST:

   a. Open 1CB-108 (Aux Electric Blr A and B Feed Pump Miniflow to Unit CST Isol)

   b. Close 1CB-101 (Aux Electric Blr A and B Feed Pump Miniflow to Unit 1 CST Isol).

G) Align Aux Electric Boiler Feed Pump Suction to Unit 2 UST or YM:

a. Open 1CB-93 (Unit 2 UST To Any Electric Blr Isol)

OR

1CB-135 (Demin Water to Aux Electric Blr Isol)

b. Close 1CB-91 (Unit 1 UST To Aux Electric Blr Isol).

3. IF Unit 2 condensate available to supply CA storage tank, **THEN** dispatch operator to locally perform the following:

A) Throttle open 1CA-158 (Unit 2 CM To CA Storage Tank Isol) and close 1CA-157 (Unit 1 CM TO CA Storage Tank Isol).

B) Open 1CA-154 (CA Storage Tank Overflow To Unit 2 CST Isol) and close 1CA-153 (CA Storage Tank Overflow To Unit 1 CST Isol).

4. Close 1CA-6 (CA Sup From CA Storage Tank).

CAUTION **Constant communication should be maintained with Radwaste Chemistry if pumpover from Turbine Bldg sump to Floor Drain Tank is required.**

5. TB Sump will overflow into hotwell pit. If equipment damage in hotwell pit is imminent (Amertap pumps) and before pump out to RC is allowed, then locally realign to pump to Floor Drain Tank as follows:

A) Close 1WP-6 (TB Sump Pumps Dis. To WC Isol)

B) Verify Radwaste Chemistry has made alignment to FDT and pump over only enough volume to prevent equipment damage in hotwell pit.

C) When Radwaste Chemistry can no longer receive water to Waste System, stop the pump over from Turbine Bldg. Sump. If pump out to WC is not possible per HP, prepare for flooding of eqiupment in hotwell sump. (Amertap etc.).

6. When HP sample results allow pumping to RC, manually pump out sump PER OP/1/B/6400/01A, CONDENSER CIRCULATING WATER AND LOW LEVEL INTAKE, Enclosure 4.11.

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Enclosure 6 Minimizing Secondary Side Contamination	PAGE NO. 3 OF 3

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

7. IF Unit 1 CST is in danger of overflowing or if Unit 2 is in danger of losing condensate, THEN coordinate with HP to pump to Unit 2 CST PER OP/1/A/6250/01, CONDENSATE AND FEEDWATER, Enclosure 4.10.

8. Realign the following for normal operation as required after condition is cleared:

   A) CSAE drains to TB sump

   B) WZ Pumps to Auto

   C) TB Sump Pumps

   D) CA Storage Tank supply and overflow to desired unit

   E) Open 1CA-6 (CA Sup From CA Storage Tank).

   F) NB and WL evaporator condensate to desired unit

   G) Aux Electric Boiler to desired unit.

Attachment 2

Selected Pages from the Westinghouse Owners' Group Emergency Response Guidelines.

Number	Title	Rev.Issue/Date
E-0	REACTOR TRIP OR SAFETY INJECTION	HP-Rev.1A 1 July 1987

A. PURPOSE

This guideline provides actions to verify proper response of the automatic protection systems following manual or automatic actuation of a reactor trip or safety injection, to assess plant conditions, and to identify the appropriate recovery guideline.

B. SYMPTOMS OR ENTRY CONDITIONS

1) The following are symptoms that require a reactor trip, if one has not occurred:
[Enter plant specific setpoints and requirements]

2) The following are symptoms of a reactor trip:

a. Any reactor trip annunciator lit.

b. Rapid decrease in neutron level indicated by nuclear instrumentation.

c. All shutdown and control rods are fully inserted. Rod bottom lights are lit.

3) The following are symptoms that require a reactor trip and safety injection, if one has not occurred:
[Enter plant specific setpoints and requirements].

4) The following are symptoms of a reactor trip and safety injection:

a. Any SI annunciator lit.

b. SI pumps running.

a. [Enter plant specific list].

1. INTRODUCTION

Guideline E-0, REACTOR TRIP OR SAFETY INJECTION, provides actions to verify proper response of the automatic protection systems following manual or automatic actuation of a reactor trip or safety injection, to assess plant conditions, and to identify the appropriate Optimal Recovery Guideline.

Guideline E-0 is to be entered when any of the following occur:

1) A reactor trip is required as determined by plant specific setpoints or requirements being exceeded.
2) A reactor trip has occurred as determined by the plant annunciators, neutron flux instrumentation, and control rod position indicators.
3) A safety injection is required as determined by plant specific setpoints or requirements being exceeded.
4) A safety injection has occurred as determined by the plant annunciators, SI pump status, or other plant specific means.

Once E-0 is entered, it is not exited until there is a direct transition to an Optimal Recovery Guideline (ORG) as directed by the symptoms being monitored in E-0 or to a Function Restoration Guideline (FRG) as directed by the Critical Safety Function Status Trees or symptoms being monitored in E-0.

2. DESCRIPTION

Guideline E-0, REACTOR TRIP OR SAFETY INJECTION, provides the operator with the necessary guidance to verify that all automatic actions have occurred as designed and presents the diagnostic sequence to be followed in the identification of the appropriate Optimal Recovery Guideline. These include:

1. ECA-0.0, LOSS OF ALL AC POWER
2. ES-0.1, REACTOR TRIP RESPONSE
3. E-1, LOSS OF REACTOR OR SECONDARY COOLANT
4. E-2, FAULTED STEAM GENERATOR ISOLATION
5. E-3, STEAM GENERATOR TUBE RUPTURE
6. ES-1.1, SI TERMINATION
7. ECA-1.2, LOCA OUTSIDE CONTAINMENT

It is expected that the operator will attempt to take manual actions to correct for anomalous conditions during power operation. Such actions would include taking manual control of the automatic control systems, turning on additional charging pumps, reducing power level, etc. If these types of actions do not alleviate the trend toward a reactor trip or safety injection, the operator is permitted to trip the reactor and, if necessary, actuate safety injection.

The reactor protection equipment is designed to safely shut down the reactor in the event that the anomalous condition cannot be corrected. The safety injection system is designed to provide emergency core cooling water and boration to maintain a safe reactor shutdown condition. The plant safeguards systems operate with offsite electrical power or from onsite emergency diesel-electric power, should offsite power not be available. The operator will enter E-0 on ~~a reactor trip or~~ safety injection, whether the signal was automatic or a result of manual actuation.

Through symptom-based diagnosis, the operator is directed to the proper Optimal Recovery Guideline to facilitate optimal recovery. Transient descriptions are provided in the appropriate background documents.

Number	Title	Rev.Issue/Date
ES-3.3	POST-SGTR COOLDOWN USING STEAM DUMP	HP-Rev.1A 1 July 1987

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

- *Steam should not be released from any ruptured SG if water may exist in its steamline.*
- *An offsite dose evaluation should be completed prior to using this guideline.*

NOTE *Foldout page should be open.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Turn On PRZR Heaters As Necessary To Saturate PRZR Water At Ruptured (SG)s Pressure**	
2	**Check If SI Accumulators Should Be Isolated:**	
	a. Check the following: o RCS subcooling based on core exit TCs - GREATER THAN (1)°F [(2)°F FOR ADVERSE CONTAINMENT] o PRZR level - GREATER THAN (3)% [(4)% FOR ADVERSE CONTAINMENT]	a. Go to ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT SUBCOOLED RECOVERY DESIRED, Step 1.
	b. Check power to isolation valves - AVAILABLE	b. Restore power to isolation valves.
	c. Close all SI accumulator isolation valves	c. Vent any unisolated accumulators.
3	**Verify Adequate Shutdown Margin:**	
	a. Sample ruptured SG(s)	
	b. Sample RCS	
	c. Shutdown margin - ADEQUATE	c. Borate as necessary.

STEP DESCRIPTION TABLE FOR ES-3.3 STEP 1 - CAUTION 2

CAUTION: An offsite dose evaluation should be completed prior to using the guideline.

PURPOSE: To alert the operator that this guideline will result in releases of radiological effluents. The consequences of this release should be evaluated before using this guideline

BASIS:

Subsequent steps require the release of contaminated steam from the ruptured steam generator. The potential radiological consequences of this action should be evaluated to minimize offsite exposures and demonstrate conformance to 10CFR20 limitations, if possible. This evaluation should consider pre-event primary coolant activity, meterological conditions, and steam release path.

ACTIONS:

Alert appropriate plant personnel

INSTRUMENTATION:

N/A

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

N/A

Attachment 3

Selected Pages from EP/1/A/5000/01 "Safety Injection".

(1) ID No. EP/1/A/5000/01
Change(s) 0 to
2 Incorporated

DUKE POWER COMPANY
PROCEDURE PROCESS RECORD

INFORMATION ONLY

PREPARATION

(2) STATION McGuire Nuclear

(3) PROCEDURE TITLE Safety Injection

(4) PREPARED BY Len Firebaugh DATE 2/24/88

(5) REVIEWED BY Bill Pitesa DATE 3-30-88

Cross-Disciplinary Review By ________ N/R BP

(6) TEMPORARY APPROVAL (If Necessary)

By ________ (SRO) DATE ________

By ________ DATE ________

(7) APPROVED BY Bruce Travis DATE 4/13/88

(8) MISCELLANEOUS

Reviewed/Approved By ETQS Bill Pitesa DATE 3-30-88

Reviewed/Approved By ________ DATE ________

(9) COMMENTS (For procedure reissue indicate whether additional changes, other than previously approved changes, are included. Attach additional pages, if necessary.) ADDITIONAL CHANGES INCLUDED: ☑ Yes ☐ No

(10) COMPARED WITH CONTROL COPY ________ DATE ________

COMPLETION

(11) DATE(S) PERFORMED ________

(12) PROCEDURE COMPLETION VERIFICATION

☐ Yes ☐ N/A Check lists and/or blanks properly initialed, signed, dated or filled in N/A or N/R, as appropriate?

☐ Yes ☐ N/A Listed enclosures attached?

☐ Yes ☐ N/A Data sheets attached, completed, dated and signed?

☐ Yes ☐ N/A Charts, graphs, etc. attached and properly dated, identified and marked?

☐ Yes ☐ N/A Acceptance criteria met?

VERIFIED BY ________ DATE ________

(13) PROCEDURE COMPLETION APPROVED ________ DATE ________

(14) REMARKS (Attach additional pages, if necessary.)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 5. Verify Load Sequencers actuated: • Status light "E/S Load Seq Actuated Train A" - LIT • Status light "E/S Load Seq Actuated Train B" - LIT.	Manually initiate SI.

D. Subsequent Actions

____ 1. Initiate RP/0/A/5700/01, NOTIFICATION OF UNUSUAL EVENT.

CAUTION Monitor lights may not be aligned properly for other than initial entry into this procedure.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 2. Check ESF Monitor Light Panel:	
a. Groups 1, 2, 5, 7 - DARK	a. Manually align equipment as required. IF "Safety Inject Train A/B" lit, THEN check OAC Tech Spec program 13 to determine misaligned valves. IF OAC is out of service, THEN complete Enclosure 2.
b. Groups 3 AND 6 - LIT	b. Manually open valves in group 3 AND/OR close valves in group 6 as required.
c. Ss and St components in group 4 - LIT.	c. Manually align equipment as required. IF "Cont Isol Phase A Train A/B" NOT lit, THEN manually initiate Phase A isolation. IF still NOT lit, THEN check OAC Tech Spec program 13 to determine misaligned valves.

Attachment 4

EP/1/A/5000/10 "Critical Safety Function Status Trees"

INFORMATION ONLY

Form 34731 (10-81)
(Formerly SPD-1002-1)

DUKE POWER COMPANY
PROCEDURE PREPARATION
PROCESS RECORD

(1) ID No: EP/1/A/5000/10
Change(s) 0 to
0 Incorporated

(2) STATION: McGuire

(3) PROCEDURE TITLE: Critical Safety Function Status Trees

(4) PREPARED BY: Len Firebaugh DATE: November 26, 1984

(5) REVIEWED BY: [signature] DATE: 11-30-84

Cross-Disciplinary Review By: ______ N/R: [initials]

(6) TEMPORARY APPROVAL (IF NECESSARY):

By: ______ (SRO) Date: ______

By: ______ Date: ______

(7) APPROVED BY: George W. Cage Date: 11/30/84

(8) MISCELLANEOUS:

Reviewed/Approved By: ______ Date: ______

Reviewed/Approved By: ______ Date: ______

Form 34912 (8-82)

EP/1/A/5000/10 REV 0	CRITICAL SAFETY FUNCTION STATUS TREES	PAGE NO. 1 OF 5

A. Purpose

To provide guidance on how to monitor the plant safety status by use of logic diagrams that cover the six basic safety functions.

B. Entry Conditions

- o EP/1/A/5000/01, SAFETY INJECTION, step 21, when SI cannot be terminated and cause has not been determined
- o On any transition out of EP/1/A/5000/01, SAFETY INJECTION.

Form 34913 (8-82)

EP/1/A/5000/10 REV 0	CRITICAL SAFETY FUNCTION STATUS TREES	PAGE NO. 2 OF 5

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. Immediate Actions

None

D. Subsequent Actions

1. Critical Safety Functions

a. The six Critical Safety Functions (CSF) and associated procedures in order of priority are:

1) Subcriticality - EP/1/A/5000/11, SUBCRITICALITY

2) Core Cooling - EP/1/A/5000/12, CORE COOLING

3) Heat Sink - EP/1/A/5000/13, SECONDARY HEAT SINK

4) Integrity - EP/1/A/5000/14, NC SYSTEM INTEGRITY

5) Containment - EP/1/A/5000/15, CONTAINMENT

6) Inventory - EP/1/A/5000/16, NC SYSTEM INVENTORY.

b. Each CSF has a corresponding status tree to enable the function to be monitored and to warn the operator if a safety parameter is being challenged. (Enclosures 1-6)

Form 34813 (8-82)

EP/1/A/5000/10 REV 0	CRITICAL SAFETY FUNCTION STATUS TREES	PAGE NO. 3 OF 5

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2. The CSF status trees should be monitored as follows:

   a. Normally the CSFs are continuously monitored and displayed in the Control Room by the OAC. Any change in state of the CSF will be alarmed on the computer and the alarm video displays will change color. Tech Spec Programs 21 through 26 should be used to determine which EP to implement.

   b. IF the OAC is unavailable, THEN status trees should be monitored manually as follows:

      1) Monitor status trees (Enclosure 1-6) when a SI signal is present and log status on Enclosure 7.

      2) Tree scanning should be continuous if any condition is coded higher than yellow or there is a significant change in plant status.

      3) IF no condition is coded higher than yellow, THEN tree scanning intervals should not exceed 10 minutes.

NOTE Operator discretion is required in use of status trees. It is possible certain accidents might produce non-green status conditions which cannot be corrected.

3. The rules of priority for implementing EPs referenced by the status trees are as follows:

   a. The importance of any non-green

Form 34913 (8-82)

EP/1/A/5000/10 REV 0	CRITICAL SAFETY FUNCTION STATUS TREES	PAGE NO. 4 OF 5

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

condition relative to any other condition of the same color is indicated by the order of the trees as given in step 1a.

b. IF a red path is encountered, THEN initiate indicated procedure to defend or recover the challenged CSF:

1) A red CSF requires immediate attention and departure from any conflicting Emergency Procedure in effect.

2) IF during execution of a lower priority red path procedure, a red path of higher priority arises, THEN address the higher priority red path first.

c. IF an orange path is encountered, THEN note associated procedure AND check remaining trees for a red path.
IF NO red path exists THEN initiate appropriate orange path procedure:

1) An orange CSF requires prompt attention and departure from any conflicting Emergency Procedure in effect.

2) When highest priority orange path procedure is complete, scan trees for a red path before going to next orange path procedure.

3) IF during the execution of an orange path procedure, a red path arises, THEN suspend orange AND implement red path procedure.

d. IF a yellow path is encountered,

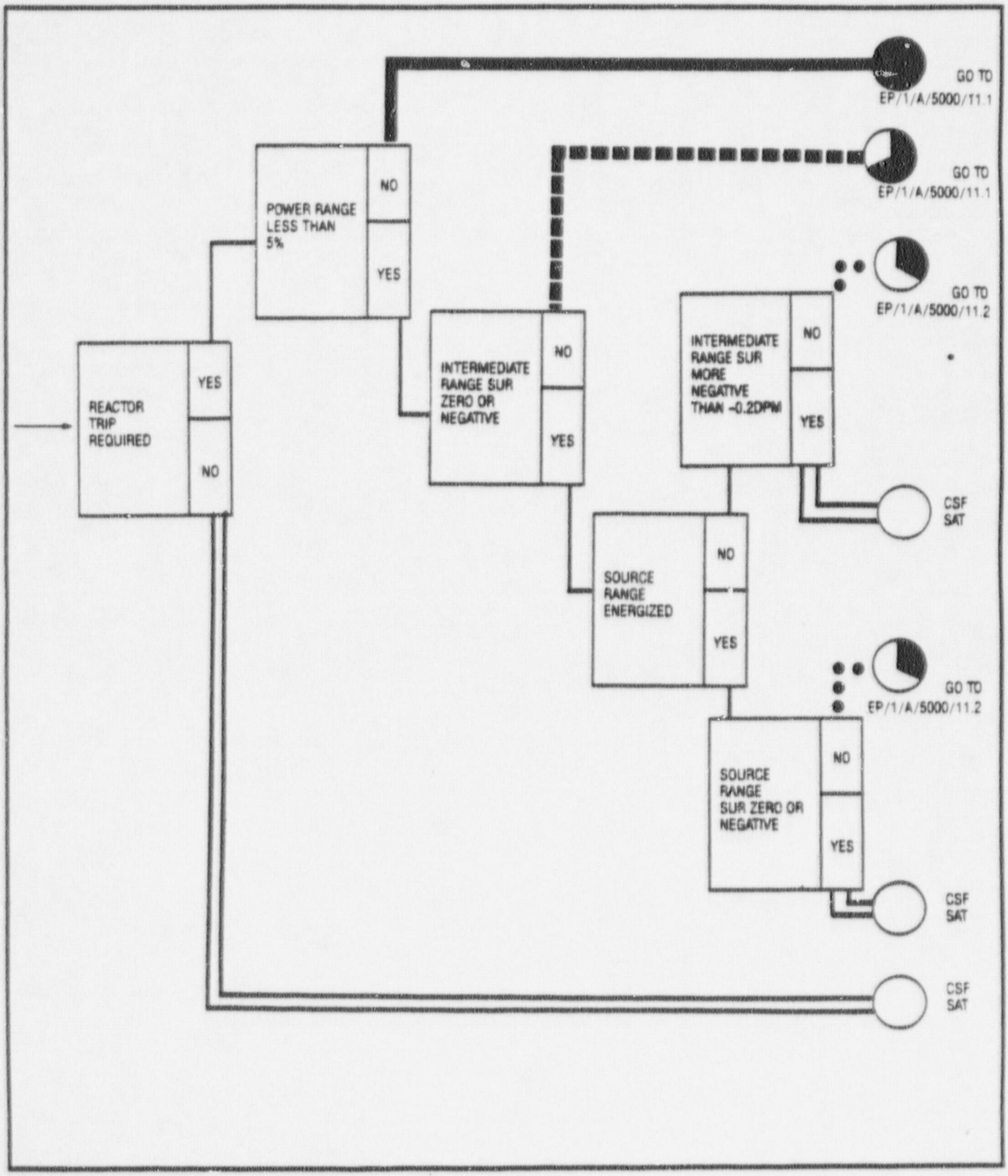
Form 34913 (8-82)

EP/1/A/5000/10 REV 0	CRITICAL SAFETY FUNCTION STATUS TREES	PAGE NO. 5 OF 5

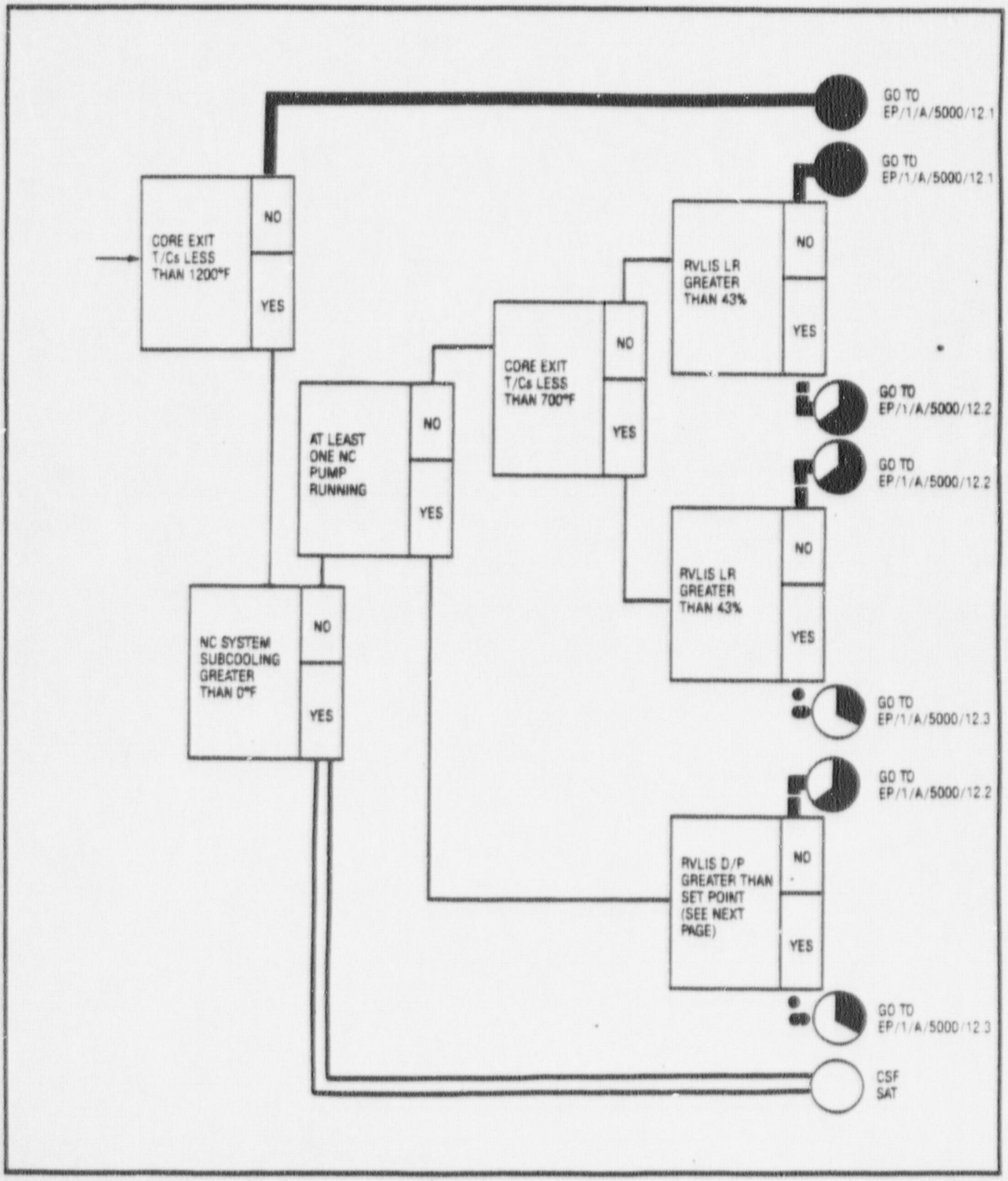
ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
<u>THEN</u> note nature of deficiency of <u>CSF</u> <u>AND</u> check remaining trees for a higher priority. <u>WHEN</u> practical initiate actions needed to fully restore indicated CSF.	

<u>END</u>

EP/1/A/5000/10	CRITICAL SAFETY FUNCTION STATUS TREES ENCLOSURE 1 — SUBCRITICALITY	PAGE NO. 1 OF 1



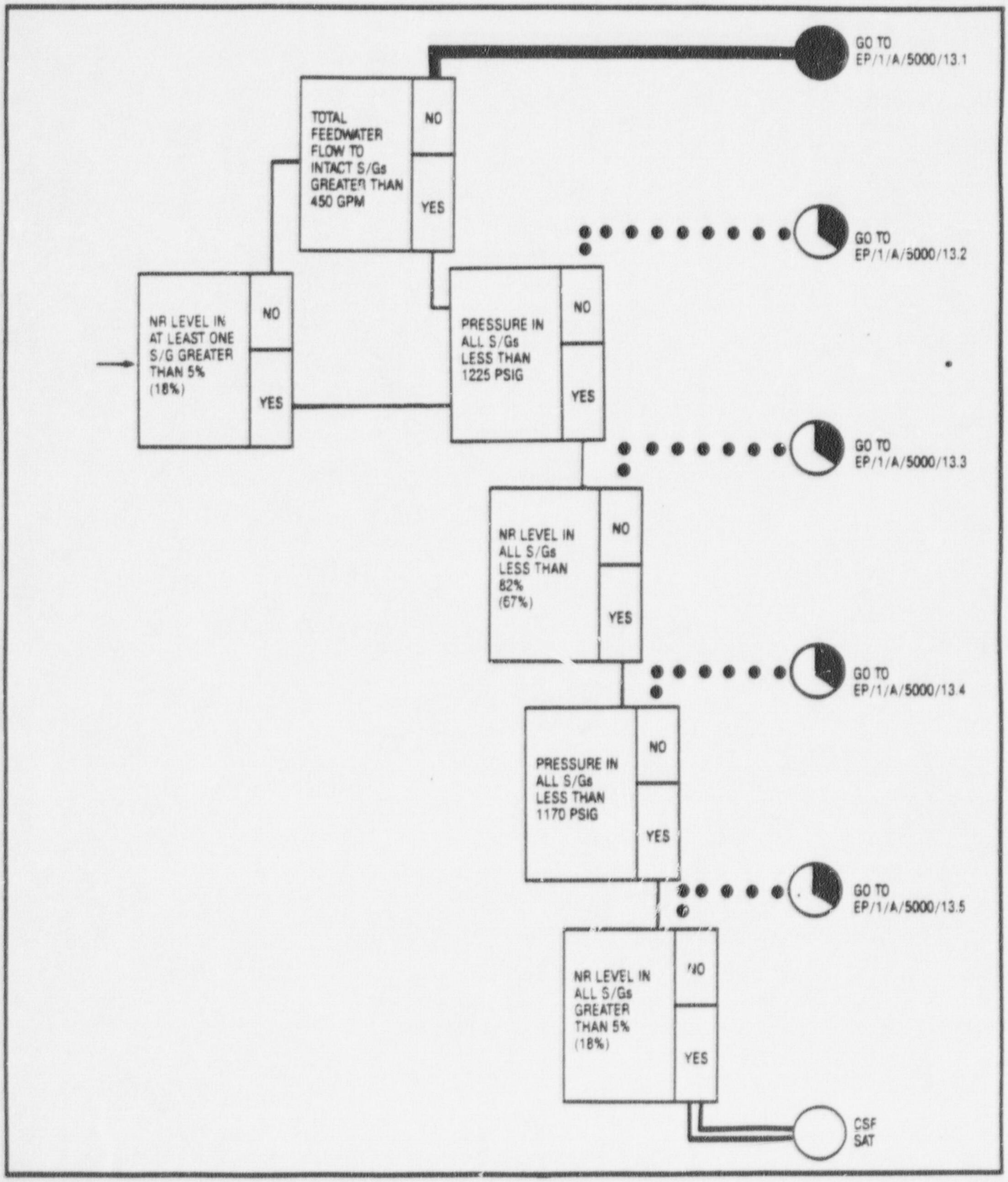
EP/1/A/5000/10	CRITICAL SAFETY FUNCTION STATUS TREES ENCLOSURE 2 — CORE COOLING	PAGE NO. 1 OF 2



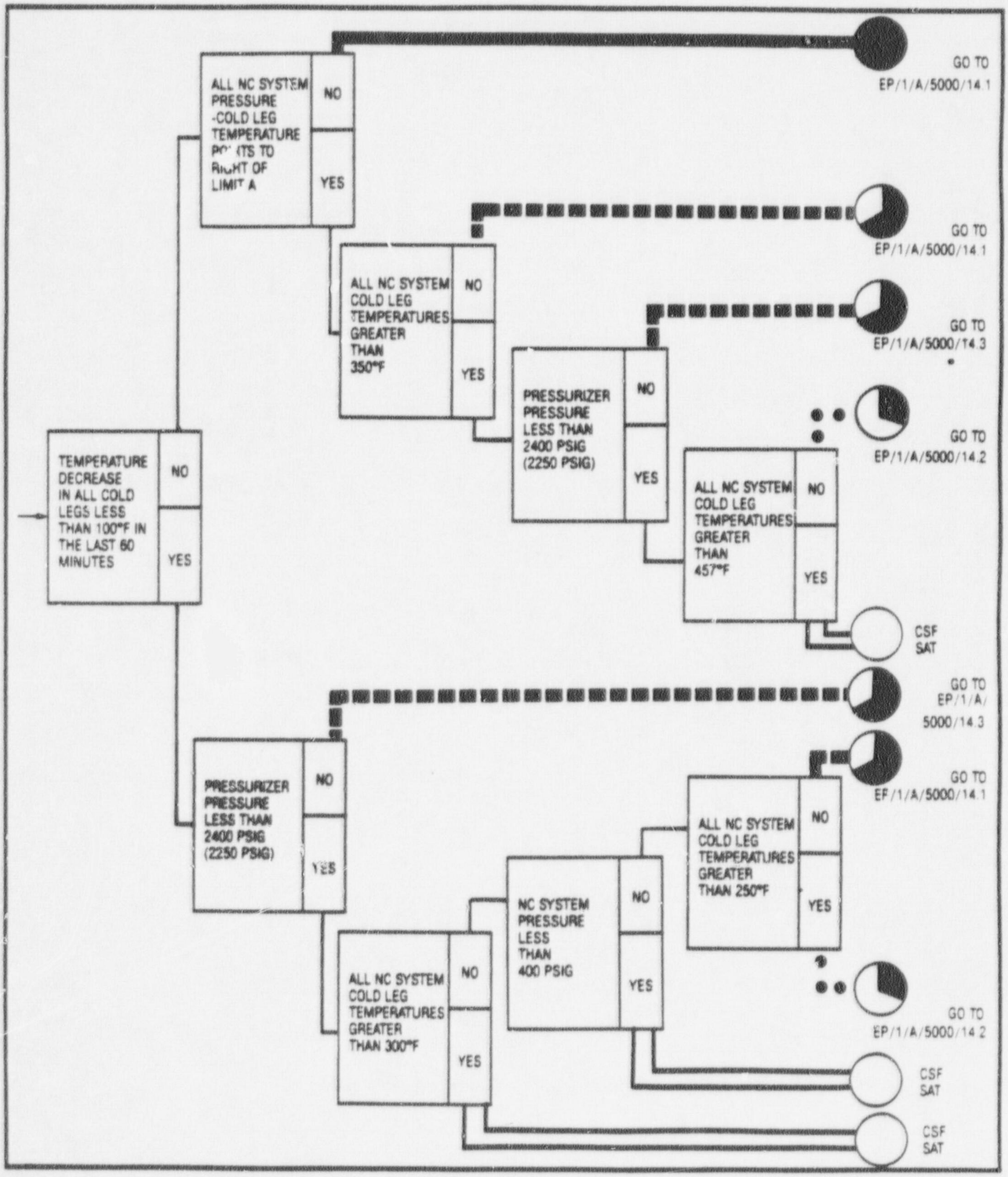
RVLIS D/P
SETPOINTS FOR
DEGRADED CORE COOLING

Number of NC Pumps Running	Channel A with NC Pump A		Channel B with NC Pump C	
	Running	Not Running	Running	Not Running
4	80%	—	80%	—
3	60%	35%	60%	35%
2	45%	23%	45%	23%
1	35%	15%	35%	15%

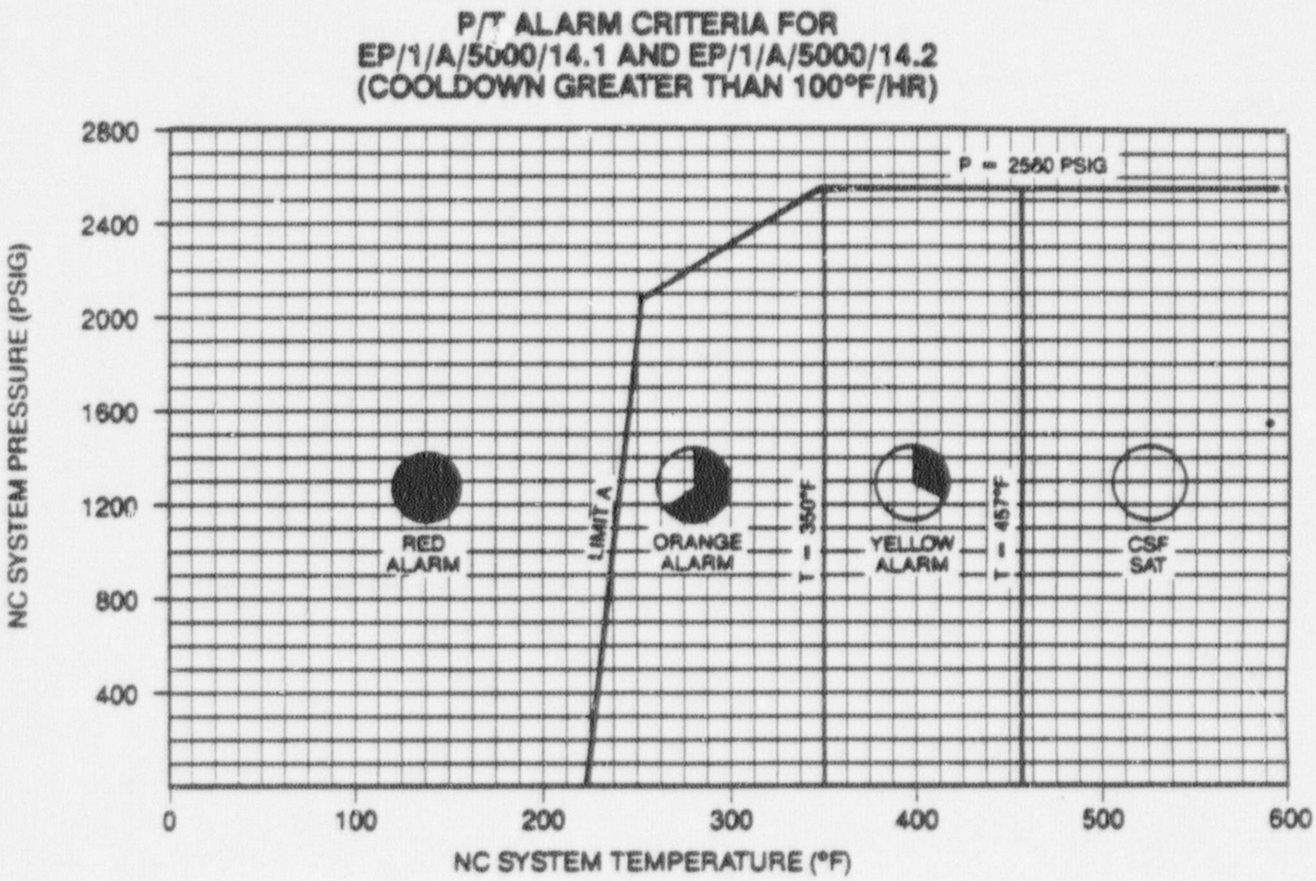
EP/1/A/5000/10	CRITICAL SAFETY FUNCTION STATUS TREES ENCLOSURE 3 — HEAT SINK	PAGE NO. 1 OF 1



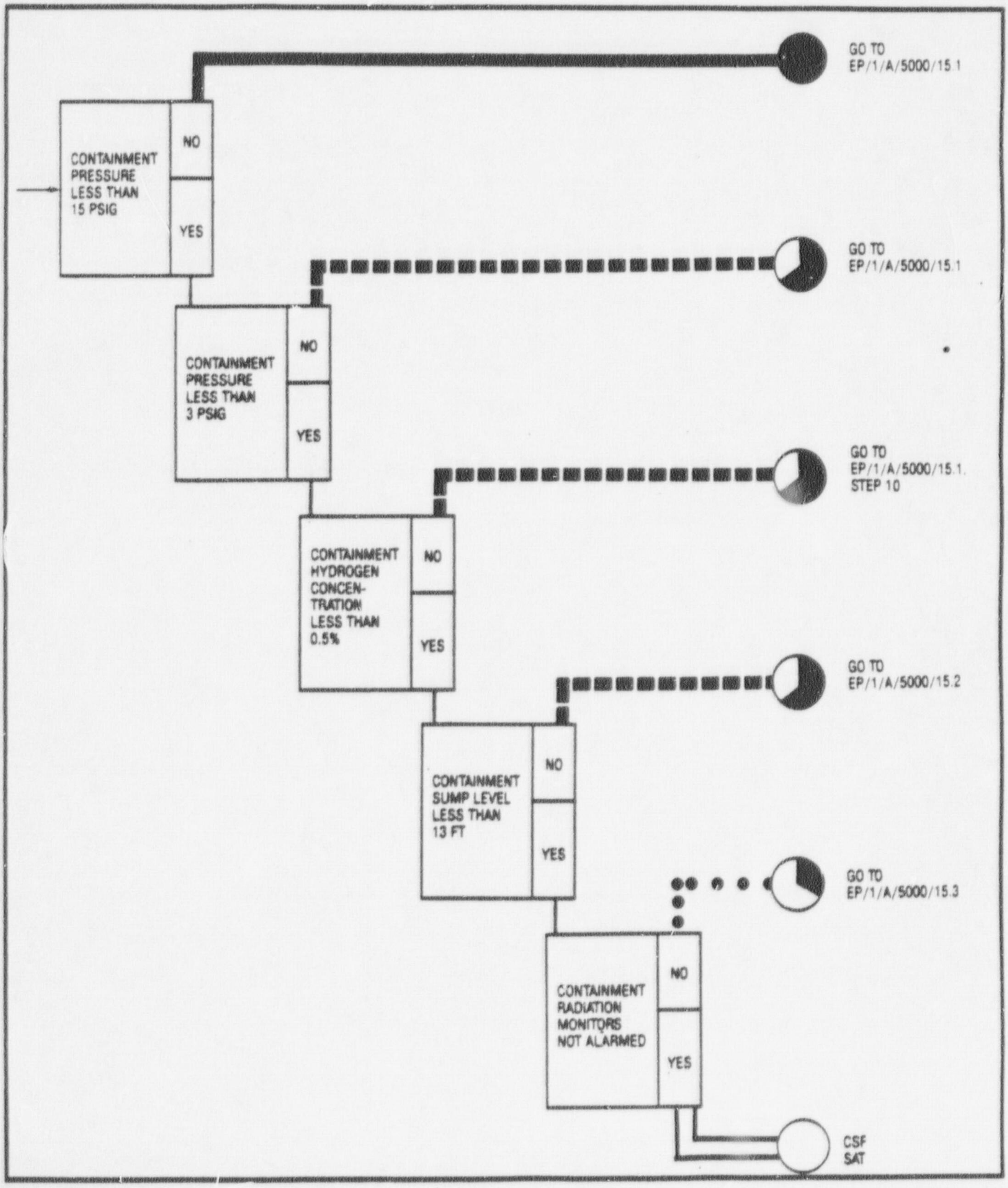
EP/1/A/5000/10	CRITICAL SAFETY FUNCTION STATUS TREES ENCLOSURE 4 — INTEGRITY	PAGE NO. 1 OF 2



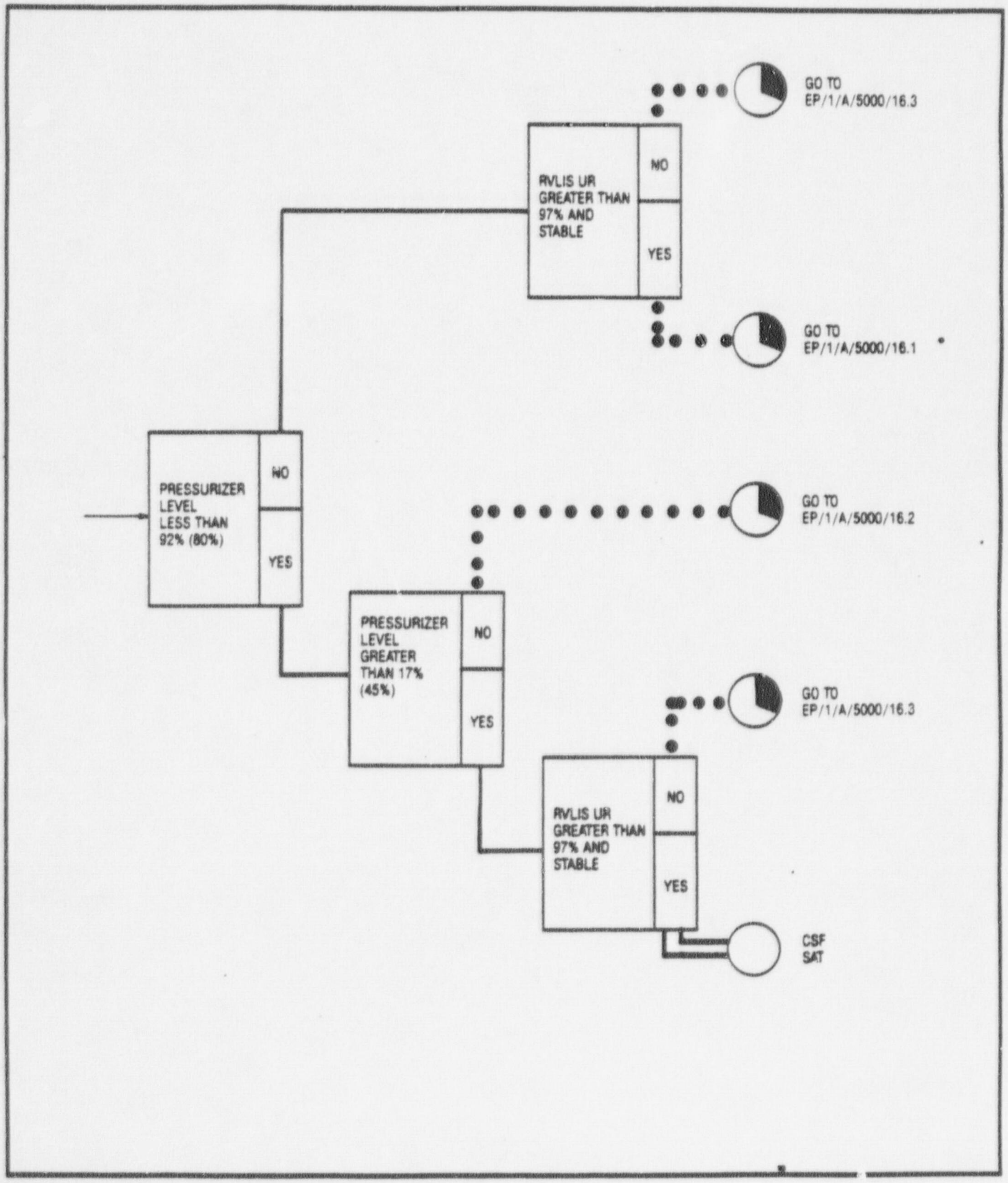
EP/1/A/5000/10	CRITICAL SAFETY FUNCTION STATUS TREES ENCLOSURE 4 — INTEGRITY	PAGE NO. 2 OF 2



EP/1/A/5000/10	CRITICAL SAFETY FUNCTION STATUS TREES ENCLOSURE 5 — CONTAINMENT	PAGE NO. 1 OF 1



EP/1/A/5000/10	CRITICAL SAFETY FUNCTION STATUS TREES ENCLOSURE 6 — INVENTORY	PAGE NO. 1 OF 1



Form 34812 (8-82)

EP/1/A/5000/10 REV 0	CRITICAL SAFETY FUNCTION STATUS TREES Enclosure 7 Status Tree Log	PAGE NO. 1 OF 1

R - Red

O - Orange

Y - Yellow

G - Green

TIME	SUBCRITICALITY	CORE COOLING	HEAT SINK	NC INTEGRITY	CONTAINMENT	NC INVENTORY	INITIALS

Performed by: ______________________ Date: ______

Attachment 5

Selected Pages From OP/1/A/6100/02 "Controlling Procedure For Unit Shutdown" and OP/0/A/6100/06 "Reactivity Balance Calculation"

Form 34731 (R5-88)

MASTER FILE

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No. OP/1/A/6100/02
Change(s) 0 to 72 Incorporated

PREPARATION

(2) Station McGuire Nuclear Station

(3) Procedure Title Controlling Procedure for Unit Shutdown

(4) Prepared By Len Firebaugh Date 3/30/89

(5) Reviewed By [signature] Date 4/3/89

Cross-Disciplinary Review By ______ N/R [initials]

(6) Temporary Approval *(if necessary)*

By ______ (SRO) Date ______

By ______ Date ______

(7) Approved By [signature] Date 4/6/89

(8) Miscellaneous

Reviewed/~~Approved~~ By Michael Werner Date 4/4/89

Reviewed/~~Approved~~ By ETQS [initials] Date 4/3/89

(9) Comments (For procedure reissue indicate whether additional changes, other than previously approved changes, are included. Attach additional pages, if necessary.)

Additional Changes Included. ☒ ~~Yes~~
☐ No

(10) Compared with Control Copy ______ Date ______

(11) Requires change to FSAR not identified in 10CFR50.59 evaluation? ☐ Yes
If "yes", attach detailed explanation. ☒ ~~No~~

Completion

(12) Date(s) Performed ______

(13) Procedure Completion Verification

☐ Yes ☐ N/A Check lists and/or blanks properly initialed, signed, dated or filled in N/A or N/R, as appropriate?

☐ Yes ☐ N/A Listed enclosures attached?

☐ Yes ☐ N/A Data sheets attached, completed, dated and signed?

☐ Yes ☐ N/A Charts, graphs, etc. attached and properly dated, identified and marked?

☐ Yes ☐ N/A Procedure requirements met?

Verified By ______ Date ______

(14) Procedure Completion Approved ______ Date ______

(15) Remarks *(attach additional pages, if necessary)*

2.15.1.1 Ensure the "Operation Selector" for all 6 detectors is in the "Off" position.

2.15.1.2 Open and tag the 120 VAC main power breaker on the panel.

2.16 Begin boration of the NC System per OP/1/A/6150/09 (Boron Concentration Control) to ensure that the SDM requirements of Data Book Table 6.5 can be maintained during cooldown.

2.17 Have IAE do the following:

_____ 2.17.1 When the neutron level decays to the normal shutdown counts, verify "High Flux At Shutdown" alarm bistable is set at one-half decade above normal shutdown source counts, and reinstate "High Flux At Shutdown" alarm.

_____ 2.18 After the "High Flux At Shutdown" alarm has been reinstated, insert the Shutdown Banks per OP/1/A/6150/08 (Rod Control).

_____ 2.19 Remove both MG sets from service per OP/1/A/6150/08 (Rod Control).

_____ 2.20 As soon as access to lower containment is possible, close 1NC-24 (Reactor Vessel Head Gasket Leakoff Drain Manual Block) to prevent NCDT H_2 from escaping to containment during cooldown.

_____ 2.21 Place Acoustic Emission Leak Monitor in "Manual" to prevent spurious alarms during shutdown.

_____ 2.22 If required, perform PT/1/A/4250/01A (Main Steam Isolation Valve Movement Test)

<u>CAUTION</u> Ensure VCT Makeup blended flow boron concentration is adjusted to a value greater than required SDM boron concentration whenever VCT makeup controls are set for normal makeup.

Form 35283 (R5-88)

Duke Power Company
Procedure Major Change
PROCESS RECORD

(1) ID No. OP/0/A/6100/06
Change No. 35
Permanent/Restricted To

(2) Station McGuire

(3) Procedure Title Reactivity Balance Calculation

(4) Section(s) of Procedure Affected: Section 4.2, Enclosure 5.5, Enclosure 5.7

(5) Description of Change: *(Attach additional pages, if necessary).*

A) After the fourth sentence in the note at the beginning of Section 4.2, add
"It is permissible to calculate Shutdown Margin for an intermediate temperature per Section 4.2.1. However, before cooling down below this temperature, Shutdown Margin shall be re-calculated for a new intermediate temperature per Section 4.2.1 or for cooldown below 200°F per Section 4.2.3. In either case, Shutdown Margin must be calculated per Section 4.2.3 prior to cooldown below 200°F."

B) On Enclosure 5.5, change "Note" to "Notes" and number existing note as 1).

C) Add to Notes at top of Enclosure 5.5:

2) This enclosure is to be used to calculate Shutdown Margin for temperatures between 200°F and 557°F.

3) Prior to cooling down below 200°F, Shutdown Margin must be calculated per Enclosure 5.7.

D) Add to top of Enclosure 5.7:

NOTE: This enclosure is to be used to calculate Shutdown Margin prior to cooling down below 200°F.

(6) Reason for Change

To clarify the use of Enclosures 5.5 and 5.7 for calculating Shutdown Margin

(7) Prepared By M. [signature] Date 4/4/89

(8) 10CFR50.59 Evaluation
Attach completed 10CFR50.59 evaluation form.

(9) Requires change to FSAR not identified in 10CFR50.59 evaluation? ☐ Yes
If "yes", attach detailed explanation. ☒ No

(10) Reviewed By ______ Date ______
Cross-Disciplinary Review By [signature] 4/4/89 N/R OPS

(11) Temporary Approval *(if necessary)*
By ______ (SRO) Date ______
By ______ Date ______

(12) Approved By [signature] Date 4/5/89

(13) Miscellaneous
Reviewed/Approved By ______ Date ______
Reviewed/Approved By ______ Date ______

DUKE POWER COMPANY
PROCEDURE MAJOR CHANGE
PROCESS RECORD CONTINUATION FORM

ID No: OP/0/A/6100/06
Change No: 35
Page 2 of 2

5) CONTINUED

E) On Enclosure 5.5, renumber step 12. as 12.a. and add
"12.b. If NC Boron concentration (5. above) is less than Adjusted Shutdown Boron Concentration (10. above) and Temperature (3. above) is between 540°F and 557°F, then Shutdown Margin may be recalculated per Enclosure 5.6."

F) On Enclosure 5.5, add after step 11,
"NOTE: Perform either 12.a or 12.b"

Form 34731 (R5-88)

Duke Power Company
PROCEDURE PROCESS RECORD

CONTROL COPY

(1) ID No. OP/0/A/6100/06
Change(s) 0 to 33 Incorporated

PREPARATION

(2) Station McGuire

(3) Procedure Title Reactivity Balance Calculation

(4) Prepared By M. Todd Kibbe Date 10/20/88

(5) Reviewed By L J Klinker Date 10/24/88

Cross-Disciplinary Review By [signature] N/R 12/9/88

(6) Temporary Approval *(if necessary)*

By ______ (SRO) Date ______

By ______ Date ______

(7) Approved By Bruce Hamilton Date 12/9/88

(8) Miscellaneous

Reviewed/Approved By M. S. Kitlan Jr Date 12/9/88

Reviewed/Approved By ______ Date ______

(9) Comments (For procedure reissue indicate whether additional changes, other than previously approved changes, are included. Attach additional pages, if necessary.)

Additional Changes Included. ☒ Yes
☐ No

(10) Compared with Control Copy ______ Date ______

(11) Requires change to FSAR not identified in 10CFR50.59 evaluation? ☐ Yes
If "yes", attach detailed explanation. ☒ No

Completion

(12) Date(s) Performed ______

(13) Procedure Completion Verification

☐ Yes ☐ N/A Check lists and/or blanks properly initialed, signed, dated or filled in N/A or N/R, as appropriate?

☐ Yes ☐ N/A Listed enclosures attached?

☐ Yes ☐ N/A Data sheets attached, completed, dated and signed?

☐ Yes ☐ N/A Charts, graphs, etc. attached and properly dated, identified and marked?

☐ Yes ☐ N/A Procedure requirements met?

Verified By ______ Date ______

(14) Procedure Completion Approved ______ Date ______

(15) Remarks *(attach additional pages, if necessary)*

4.1.2.7 Determine 557°F rod worth of control rods at their present position from Data Book Curve 6.3.3.

4.1.2.8 Obtain maximum reactivity effect of flux redistribution at zero power at any time in core life from note at bottom of Data Book Table 6.3.2.

4.1.2.9 Sum values obtained in Steps 4.1.2.5, 4.1.2.6, 4.1.2.7, and 4.1.2.8.

4.1.2.10 Determine required shutdown margin by adding value of 4.1.2.4 to 1300 pcm . Value in Step 4.1.2.9 shall be more positive than this value per MNS Technical Specification 3.1.1.1. If value in Step 4.1.2.9 is not more positive than this value, borate per appropriate Station procedure.

4.1.2.11 Forward a copy of all completed Enclosure(s) 5.4 to Reactor Unit by next working day.

4.2 Unit Shutdown

CAUTION

Perform all shutdown margin calculations and adjust boron concentration prior to cooling down below 550°F.

NOTE: For temperatures between 200°F and 557°F, shutdown margin calculations should be performed per Section 4.2.1. If temperature will remain between 500°F and 557°F and calculations of Section 4.2.1 show inadequate shutdown margin exists, then shutdown margin may be calculated with credit for xenon included. In this case, perform Section 4.2.2. If cooldown below 200°F is expected, shutdown margin should be calculated per Section 4.2.3. If shutdown banks are to be withdrawn prior to adjusting NC boron concentration per ECB, criticality may be possible even though adequate shutdown margin exists. Ensure criticality will not occur by completing Section 4.2.4.

It is permissible to calculate Shutdown Margin for an intermediate temperature per Section 4.2.1. However, before cooling down below this temperature, Shutdown Margin shall be re-calculated for a new intermediate temperature per Section 4.2.1 or for cooldown below 200°F per Section 4.2.3. In either case, Shutdown Margin must be calculated per Section 4.2.3 prior to cooldown below 200°F.

CH# 35
MTK
4/6/89

4.2.1 Unit shutdown, Tave > 200°F, No Xenon Credit included

Complete Enclosure 5.5 as follows:

4.2.1.1 Record unit and cycle.

4.2.1.2 Record cycle burnup from OAC points P1457 and P1458.

ENCLOSURE 5.5
SHUTDOWN MARGIN - UNIT SHUTDOWN,
TAVE >200°F, NO XENON CREDIT INCLUDED

NOTES: 1) Perform prior to cooling down below 550°F.
2) This enclosure is to be used to calculate Shutdown Margin for temperatures between 200°F and 557
3) Prior to cooling down below 200°F, Shutdown Margin must be calculated per Enclosure 5.7. CH#35 MTK 4/6/89

1. Unit _______ Cycle _______

2. Cycle Burnup _______ EFPD _______ MWD/MTU

3. Lowest Temperature Expected _______ °F

4. 1.3% Shutdown Margin Boron (from tabular data Data Book Curve 6.5 for burnup 2. above and temperature 3. above). _______ ppm

5. NC Boron Concentration _______ ppm

NOTE: If one or more rods are known to be inoperable, perform Steps 6. through 9. Otherwise mark Steps 6., 7. and 8. as N/A, mark Step 9. as 0 ppm and proceed to Step 10.

6. Number of known inoperable rods _______ inoperable rods

7. Stuck Rod Worth (from Data Book Table 6.3.2 line B interpolated to present burnup 2. above) -_______ pcm

8. Differential Boron Worth (from Data Book Curve 6.2 for present burnup 2.above) -_______ pcm/ppm

9. Stuck Rod Penalty +_______ ppm
$\left(\frac{\text{6. above x 7. above}}{\text{8. above}}\right)$

10. Adjusted Shutdown Boron Concentration (4. above + 9. above) _______ ppm

11. If NC Boron Concentration (5. above) is greater than or equal to Adjusted Shutdown Boron Concentration (10. above), adequate shutdown margin exists at temperature 3. above.

CH#35 MTK 4/6/89 NOTE: Perform either 12.a or 12.b

12.a. If NC Boron concentration (5. above) is less than Adjusted Shutdown Boron Concentration (10. above) and it is desired to decrease temperature to that of 3. above, then NC Boron Concentration must be adjusted equal to or greater than 10. above.

12.b. If NC Boron concentration (5. above) is less than adjusted Shutdown Boron Concentration (10. above) and Temperature (3. above) is between 500°F and 557°F, then Shutdown Margin may be recalculated per Enclosure 5.6.

CH#35 MJK 4/6/89

Calculated By _______________ Date/Time _______/_______

Checked By _______________ Date/Time _______/_______

13. Forward a copy of all completed Enclosure(s) 5.5 to Reactor Unit by next working day.

ENCLOSURE 5.7
SHUTDOWN MARGIN - UNIT SHUTDOWN,
TAVE < 200°F, NO XENON
CREDIT INCLUDED

NOTE: This enclosure is to be used to calculate Shutdown Margin prior to cooling down below 200°F.

CH#35 MTC 4/6/89

1. Unit ______ Cycle ______

2. Cycle Burnup ______ EFPD ______ MWD/MTU

3. NC Boron Concentration ______ ppm

4. Shutdown Boron Concentration (from tabular data of Data Book Curve 6.5, 1.0% shutdown at 68°F or 1.3% shutdown at 200°F, whichever is greater) ______ ppm

NOTE: If one or more rods are known to be inoperable, perform Steps 5. through 8. Otherwise, mark Steps 5., 6. and 7. as N/A, mark Step 8. as 0 ppm and proceed to Step 9.

5. Number of known inoperable rods ______ inoperable rods

6. Stuck Rod Worth (from Data Book Table 6.3.2 line B interpolated to present burnup 2. above) -______ pcm

7. Differential Boron Worth (from Data Book Curve 6.2 for present burnup 2. above) -______ pcm/ppm

8. Stuck Rod Penalty +______ ppm
(5. above x 6. above)
7. above

9. Adjusted Shutdown Boron Concentration (4. above + 8. above) ______ ppm

10. If NC Boron Concentration (3. above) is greater than or equal to Adjusted Shutdown Boron Concentration (9. above), adequate shutdown margin exists for cooldown.

11. If NC Boron concentration (3. above) is less than Adjusted Shutdown Boron Concentration (9. above) and it is desired to cooldown below 200°F, then NC Boron Concentration must be adjusted equal to or greater than 9. above.

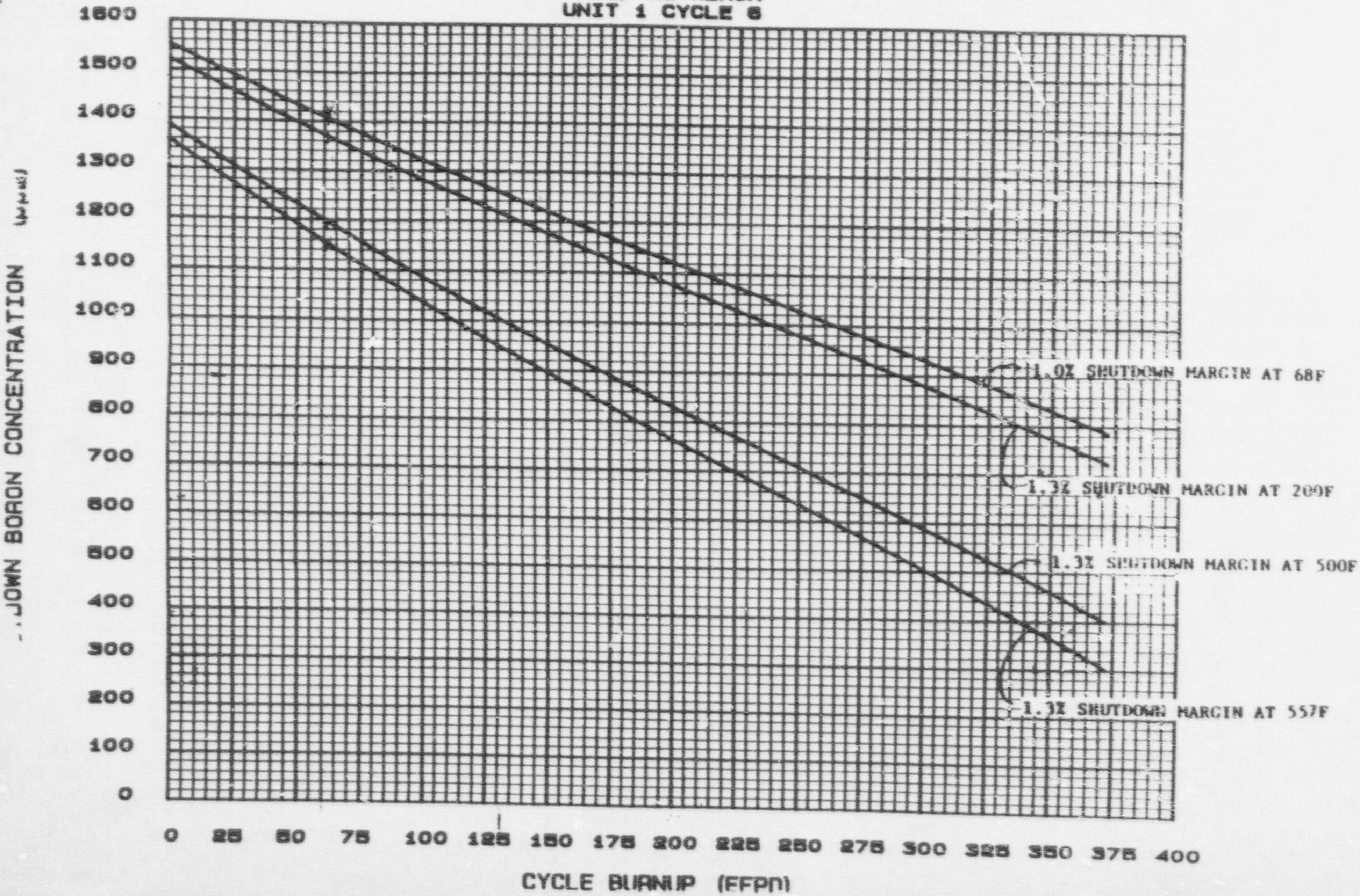
Calculated By ______________ Date/Time ________/________

Checked By ______________ Date/Time ________/________

12. Forward a copy of all completed Enclosure(s) 5.7 to Reactor Unit by next working day.

OP/1/A/6100/22
ENCLOSURE 4.3
CURVE 6.5
SHUTDOWN BORON CONCENTRATION
ARI, NO XENON
UNIT 1 CYCLE 6

NOTE: 100 PPM CONSERVATIVE
NOTE: INCLUDES 1 STUCK ROD



CN#2
LK
11/23/88

OP/1/A/6100/22
Enclosure 4.3
CURVE 6.5
SHUTDOWN BORON CONCENTRATION
ARI, NO XENON *
MCGUIRE 1 CYCLE 6

1.3% Shutdown Margin

Ch#94
LK
9/12/88

CORE AVERGE TEMP. (° F)

BURNUP (EFPD)	200	225	250	275	300	325	350	375	400	425	450	475	500	525	550	557
0	1523	1512	1501	1490	1479	1468	1457	1446	1435	1424	1413	1401	1389	1376	1363	1359
20	1471	1459	1447	1436	1424	1412	1400	1388	1376	1363	1350	1337	1323	1308	1293	1289
40	1421	1409	1396	1384	1372	1359	1346	1333	1319	1305	1291	1276	1260	1244	1227	1222
60	1373	1360	1347	1334	1321	1308	1294	1280	1265	1250	1234	1217	1200	1182	1163	1157
80	1327	1314	1300	1287	1273	1258	1244	1228	1213	1196	1179	1161	1142	1122	1102	1096
100	1282	1268	1255	1240	1226	1211	1195	1179	1162	1144	1126	1106	1086	1065	1043	1036
120	1239	1225	1210	1196	1180	1164	1148	1131	1113	1094	1075	1054	1032	1010	986	979
140	1197	1182	1167	1152	1136	1119	1102	1084	1065	1046	1025	1003	980	956	930	923
160	1156	1141	1125	1109	1093	1075	1057	1039	1019	998	976	953	929	904	877	869
180	1116	1100	1084	1068	1050	1032	1013	994	973	952	929	905	879	853	824	816
200	1076	1060	1043	1026	1008	990	970	950	928	906	882	857	830	802	772	763
220	1037	1020	1003	985	967	947	927	906	883	860	836	810	782	752	721	711
240	997	980	963	944	925	905	884	862	839	815	790	763	734	703	669	660
260	958	941	922	903	984	863	841	818	794	770	743	715	685	653	618	608
280	918	900	882	862	842	820	798	774	750	724	697	668	637	603	566	556
300	878	859	840	820	799	777	754	730	704	678	650	620	587	552	514	503
320	837	818	798	778	756	734	710	685	658	631	602	571	537	501	461	449
340	794	775	755	734	712	689	664	639	611	583	553	521	486	448	407	394
360	751	732	711	690	667	643	618	591	563	534	503	469	433	394	351	338
372	724	705	684	662	639	615	589	562	533	504	472	438	400	360	316	303

Ch#92
LK
9/23/88

* NO XE, EQ SM

NOTE: MOST REACTIVE ROD STUCK OUT

100 PPM CONSERVATIVE

OP/1/A/6100/22
Enclosure 4.3
CURVE 6.5
SHUTDOWN BORON CONCENTRATION
ARI, NO XENON *
MCGUIRE 1 CYCLE 6

10% Shutdown Margin

Ch #94
LK
9/12/88

BURNUP (EFPD)	CORE AVERGE TEMP. (° F) 68	70	80	90	100	110	120	130	140	150	160	170	180	190	200
0	1553	1552	1549	1545	1541	1537	1533	1529	1525	1520	1516	1512	1507	1503	1498
20	1503	1502	1498	1494	1490	1486	1482	1478	1474	1469	1465	1460	1456	1451	1446
40	1455	1454	1450	1446	1442	1438	1434	1429	1425	1420	1416	1411	1406	1401	1396
60	1409	1408	1404	1400	1395	1391	1387	1382	1378	1373	1369	1364	1359	1354	1349
80	1365	1364	1359	1355	1351	1346	1342	1337	1333	1328	1323	1318	1313	1308	1303
100	1322	1321	1316	1312	1307	1303	1298	1294	1289	1284	1279	1274	1269	1264	1258
120	1280	1279	1275	1270	1266	1261	1256	1252	1247	1242	1237	1231	1226	1221	1215
140	1239	1238	1234	1229	1225	1220	1215	1211	1206	1200	1195	1190	1184	1179	1173
160	1199	1198	1194	1190	1185	1180	1175	1170	1165	1160	1155	1149	1143	1138	1132
180	1160	1159	1155	1151	1146	1141	1136	1131	1126	1120	1115	1109	1103	1098	1092
200	1122	1121	1117	1112	1108	1103	1098	1092	1087	1081	1075	1070	1064	1058	1052
220	1083	1083	1079	1074	1069	1064	1059	1054	1048	1042	1036	1030	1024	1019	1013
240	1045	1045	1041	1036	1031	1026	1021	1015	1009	1004	997	991	985	979	973
260	1007	1007	1003	998	993	988	983	977	971	965	959	952	946	940	934
280	969	968	964	960	955	950	944	938	932	926	920	913	907	901	894
300	930	930	925	921	916	911	905	899	893	886	880	874	867	861	854
320	891	890	886	881	876	871	865	859	853	846	840	833	827	820	814
340	851	850	846	841	836	830	824	818	812	806	799	792	785	779	772
360	810	809	804	799	794	788	783	776	770	764	757	750	743	736	729
372	785	784	779	774	768	763	757	751	744	739	731	724	717	710	703

Ch #92
LK
11/23/87

* NO XE, EQ SM

NOTE: MOST REACTIVE ROD STUCK OUT

NOTE: 100 PPM CONSERVATIVE

Attachment 6

Procedural Committments Made in the April 13, 1989
Washington, D.C. Duke Power Presentation to the NRC Staff

EMERGENCY PROCEDURES

- ERG DEVIATION DOCUMENT

- Projected Schedule for Completion

  Deviation document scheduled completion is June 30, 1989.

  It was Duke's opinion that a deviation document did exist and that we shared Catawba's documentation. This was based on the fact that both sites share a common design and similar safety analysis.

- Safety Injection Initiation

  McGuire has changed the threshold for manual initiation of safety injection. The previous threshold was pressurizer level less than five percent after a second charging pump was started and cold leg injection valves were opened; the new threshold is pressurizer level decreasing after a second charging is started and injecting through the normal charging flow path.

- Other Procedure Enhancements

  I. Procedure changes already completed

  A. OP/1,2/A/6100/02, "Controlling Procedure for Unit Shutdown"

  1. Revised procedure step to more clearly allow cooldown initiation prior to meeting the Shutdown Margin for Cold Shutdown as long as Shutdown Margin is maintained throughout the cooldown.

  B. OP/0/A/6100/06, "Reactivity Balance Calculation"

  1. Revised procedure step to more clearly allow cooldown initiation prior to meeting the Shutdown Margin for Cold Shutdown as long as Shutdown Margin is maintained throughout the cooldown.

  II. Procedure changes to be completed by May 1, 1989

  A. AP/1,2/A/5500/10, "NC System Leakage Within the Capacity of Both NV Pumps - Case 1 Steam Generator Tube Leakage"

  1. Revised procedure to require operator to initiate manual Safety Injection and go to EP/1,2/A/5000/01 instead of manually opening NI-9A and NI-10B (NC Cold Leg Injection from NV) when maximum charging is not maintaining pressurizer level.

2. Revised subseque. .ctions to more clearly resemble EP/1,2/A/5000/04, "Steam Generator Tube Rupture".

3. Revised the last step to direct the operator to cooldown the ruptured Steam Generator using EP/1,2/A/5000/4.1, "SGTR Cooldown Using Steam Dump", EP/1,2/A/5000/4.2, "SGTR Cooldown Using Backfill" or EP/1,2/A/5000/4.3, "SGTR Cooldown Using Blowdown".

4. Added new step to begin unit load reduction.

5. Added new step and enclosure to minimize secondary contamination.

6. Added Caution to allow operator to exceed 50°F/hr cooldown rate.

7. Added step to isolate blowdown on the ruptured steam generator.

B. AP/1,2/A/5500/10, "NC System Leakage Within the Capacity of Both NV Pumps - Case 2 Reactor Coolant System Leakage"

1. Revised procedure to require operator to initiate manual Safety Injection and go to EP/1,2/A/5000/01, "Safety Injection" instead of manually opening NI-9A and NI-10B (NC Cold Leg Injection from NV) when maximum charging is not maintaining pressurizer level.

2. Revised subsequent actions to more closely resemble EP/1,2/A/5000/2.2, "Post LOCA Cooldown and Depressurization"

III. Procedure changes to be implemented with the Emergency and Abnormal Procedure total reissue currently scheduled for June 30, 1989 (waiting on simulator for validation)

A. EP/1,2/A/5000/04, "Steam Generator Tube Rupture"

1. All Reactor Coolant Pumps are left operating for cooldown.

2. New enclosure for minimizing secondary contamination.

B. EP/1,2/A/5000/4.2, "SGTR Cooldown Using Backfill"

1. Revised procedure step to more clearly allow cooldown initiated as long as Shutdown Margin is maintained throughout the cooldown.

2. Revised procedure to stop the reactor coolant pump on the ruptured Steam Generator after placing residual heat removal in service. This helps maintain ruptured Steam Generator pressure elevated and hence Reactor Coolant System pressure to allow Reactor Coolant Pumps on the intact steam generators to be operated until the Reactor Coolant System Temperature is less 160°F.

C. EP/1,2/A/5000/4.3, "SGTR Cooldown Using Blowdown"

1. Revising entire procedure to utilize normal Blowdown instead of the Blowdown Recycle System.

D. AP/1,2/A/5500/01, "Reactor Trip"

1. Deleting procedure and incorporating Reactor Trip in EP/1,2/A/5000/01, "Safety Injection".

Attachment 7

Superceded Copy of AP/1/A/5000/10 and Selected Pages
From the Superceded OP/1/A/6100/02

(1) ID No. AP/1/A/5500/10
Change(s) 0 to
0 Incorporated

DUKE POWER COMPANY
PROCEDURE PROCESS RECORD

INFORMATION ONLY

PREPARATION

(2) STATION McGuire Nuclear

(3) PROCEDURE TITLE NC System Leakage Within Capacity of Both NV Pumps

(4) PREPARED BY Len Firebaugh DATE 1/29/88

(5) REVIEWED BY [signature] DATE 3-3-88

Cross-Disciplinary Review By ____ N/R DCS

(6) TEMPORARY APPROVAL (If Necessary)

By ____ (SRO) DATE ____

By ____ DATE ____

(7) APPROVED BY Bruce Travis DATE 3/4/88

(8) MISCELLANEOUS

Reviewed/Approved By ____ DATE ____

Reviewed/~~Approved~~ By RH ETBS DATE 3-6-88

(9) COMMENTS (For procedure reissue indicate whether additional changes, other than previously approved changes, are included. Attach additional pages, if necessary.) ADDITIONAL CHANGES INCLUDED. ☑ Yes ☐ No

(10) COMPARED WITH CONTROL COPY ____ DATE ____

COMPLETION

(11) DATE(S) PERFORMED ____

(12) PROCEDURE COMPLETION VERIFICATION

☐ Yes ☐ N/A Check lists and/or blanks properly initialed, signed, dated or filled in N/A or N/R, as appropriate?

☐ Yes ☐ N/A Listed enclosures attached?

☐ Yes ☐ N/A Data sheets attached, completed, dated and signed?

☐ Yes ☐ N/A Charts, graphs, etc. attached and properly dated, identified and marked?

☐ Yes ☐ N/A Acceptance criteria met?

VERIFIED BY ____ DATE ____

(13) PROCEDURE COMPLETION APPROVED ____ DATE ____

(14) REMARKS (Attach additional pages, if necessary.)

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN THE CAPACITY OF BOTH NV PUMPS	PAGE NO. i

TABLE OF CONTENTS

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN THE CAPACITY OF DOTH NV PUMPS	PAGE NO. 1 OF 17

A. Purpose

This procedure covers the required operator actions for NC leakage greater than Tech Specs but where the Charging Pumps are capable of maintaining Pzr water level and the Pzr heaters are capable of maintaining system pressure under the following conditions:

Case I Steam Generator Tube Leakage
Case II NC System Leakage
Case III Letdown Or Charging Line Leakage
Case IV Leakage Into KC System.

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case I Steam Generator Tube Leakage	PAGE NO. 2 OF 17

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B. Symptoms	
• "1EMF-33, Cond AE Exh Hi Gas Rad" alarm	
• "1EMF-34, SG Sample Hi Rad" alarm	
• "1EMF-24 25 26 27 Steam Line Hi Rad" alarm	
• Increase in frequency of auto makeup to VCT.	
C. Immediate Actions	
_____ 1. Check Pzr Level - AT OR INCREASING TO PROGRAMMED LEVEL.	IF level decreasing, THEN perform the following to maintain level: a. Ensure #1 PD Pump speed increasing OR 1NV-238 (Charging Line Flow Control) opening. b. Start additional NV Pumps c. Reduce letdown to 45 GPM orifice. IF level decreases below 5%, THEN manually initiate SI AND go to EP/1/A/5000/01, SAFETY INJECTION.
_____ 2. Check Pzr Press - AT OR INCREASING TO 2235 PSIG.	IF less than 2210 PSIG, THEN ensure backup heaters on. IF pressure approaches 1945 PSIG, THEN trip Reactor AND refer to AP/1/A/5500/01, REACTOR TRIP.
D. Subsequent Actions	

CAUTION If Pzr level cannot be maintained, (less than 5% and decreasing) then SI should be manually initiated.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 1. Announce occurrence on paging system.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 2. Check Pzr Level - STABLE OR INCREASING.	IF level decreasing with maximum charging flow, THEN: a. Manually trip Turbine AND Reactor. b. Open 1NI-9A AND 10B (NC Cold Leg Inj From NV). c. Swap charging pump suction to FWST: 1) Open 1NV-221A AND 222B (NV Pumps Suct From FWST) 2) Close 1NV-141A AND 142B (VCT Outlet Isol).
_____ 3. Check if S/G blowdown isolation required:	
a. "1EMF-34 S/G Sample Hi Rad" alarm - LIT	a. Go to step 4.
b. Verify S/G BB Auto Isol valves - CLOSED: • S/G A, 1BB-119 • S/G B, 1BB-120 • S/G C, 1BB-121 • S/G D, 1BB-122.	b. Manually close valves.
_____ 4. Refer to RP/0/A/5700/01, NOTIFICATION OF UNUSUAL EVENT.	
_____ 5. Notify HP to determine activity released from air ejectors.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 6. Identify affected S/G: • Decrease in CF flow • Increase in S/G level • Increase in S/G pressure • Per OP/1/A/6250/08, STEAM GENERATOR BLOWDOWN.	
____ 7. Perform the following steps in conjunction with shutdown and cooldown of unit per OP/1/A/6100/02, CONTROLLING PROCEDURE FOR UNIT SHUTDOWN:	
a. IF Unit 2 available to supply AS header, THEN:	a. Supply AS header with Aux Electric Boiler:
1) Ensure the following valves - OPEN: • 1AS-74 (Unit 1 Aux Stm Hdr Isol) • 2AS-74 (Unit 2 Aux Stm Hdr Isol) • 1AS-253 (Unit 1 And 2 Aux Stm Hdr Crosstie)	1) Place boilers in operation per OP/1/B/6250/07B, AUX ELECTRIC BOILER 2) Ensure open: • 1AS-74 (Unit 1 Aux Stm Hdr Isol) • 2AS-74 (Unit 2 Aux Stm Hdr Isol) • 1AS-253 (Unit 1 And 2 Aux Stm Hdr Crosstie)
2) Close 1AS-9 (C-Htr Bleed to AS) AND 1AS-12 (SM To AS).	
3) Open 1HM-95 (AS To "A" and "B" FWPT)	3) Open 1AS-120 (Aux Elec Blr A And B To AS Isol)
4) Locally verify proper operation of 2AS-11 (Unit 2 Main Steam To Aux Steam Hdr Control).	4) Close 1AS-9 (C Htr Bleed to AS) AND slowly throttle closed 1AS-12 (SM To AS)
	5) Open 1HM-95 (AS To "A" and "B" FWPT).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

b. After unit is off line, isolate affected S/G:

- S/G 1A
  1) Close SM Isol AND Bypass Valves, 1SM-7AB, 12
  2) Control feed flow to maintain S/G NR Lvl greater than 38%.
- S/G 1B
  1) Close SM Isol AND Bypass Valves, 1SM-5AB, 11
  2) Control feed flow to maintain S/G NR Lvl greater than 38%.
  3) Locally close 1SA-2 (SM To #1 TD CA Pump)
- S/G 1C
  1) Close SM Isol AND Bypass Valves, 1SM-3AB, 10
  2) Control feed flow to maintain S/G NR Lvl greater than 38%.
  3) Locally close 1SA-1 (SM To #1 TD CA Pump)
- S/G 1D
  1) Close SM Isol AND Bypass Valves, 1SM-1AB, 9
  2) Control feed flow to maintain S/G NR Lvl greater than 38%.

c. Cooldown NC System to less than 507°F.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
d. Reduce NC System pressure to faulted S/G pressure while blocking SI per shutdown procedure.	
e. Continue plant cooldown per shutdown procedure.	
f. Depressurize NC System and ruptured S/G simultaneously:	
1) Dump steam to condenser by slowly opening SM Isol Bypass valve on ruptured S/G. OR Initiate blowdown to recycle system per OP/1/A/6250/08 S/G BLOWDOWN.	
2) Reduce NC pressure to maintain equal to ruptured S/G pressure.	

END

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

B. Symptoms

- Increase in frequency of auto makeup to VCT
- Increased leakrate results from PT/1/A/4150/01B, REACTOR COOLANT LEAKAGE CALCULATIONS
- Cont Flr/Eqp Sump Level increase
- "1EMF-38 Containment HI Part Rad" alarm
- "1EMF-39 Containment HI Gas Rad" alarm
- "1EMF-40 Containment HI Iod Rad" alarm
- "Pzr PORV Disch Hi Temp" alarm
- "Pzr Safety Discharge Hi Temp" alarm
- PRT temperature increase
- PRT level increase
- Containment temperature increase
- Containment humidity increase
- "Rx Vessel Flange Leak Off Hi Temp" alarm

C. Immediate Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 1. Check Pzr Level - AT OR INCREASING TO PROGRAMMED LEVEL.	IF level decreasing, THEN perform the following to maintain level: a. Ensure #1 PD Pump speed increasing OR 1NV-238 (Charging Line Flow Control) opening. b. Start additional NV Pumps c. Reduce letdown to 45 GPM orifice. IF level decreases below 5%, THEN manually initiate SI AND go to EP/1/A/5000/01, SAFETY INJECTION.

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II NC System Leakage	PAGE NO. 8 OF 17

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 2. Check Pzr Press - AT OR INCREASING TO 2235 PSIG.	IF less than 2210 PSIG, THEN ensure backup heaters on. IF pressure approaches 1945 PSIG, THEN trip Reactor AND refer to AP/1/A/5500/01, REACTOR TRIP.

D. Subsequent Actions

CAUTION If Pzr level cannot be maintained, (less than 5% and decreasing) then Safety Injection should be manually initiated.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 1. Announce occurrence on paging system.	
____ 2. Check Pzr Level - STABLE OR INCREASING.	IF level decreasing with maximum charging flow, THEN: a. Manually trip Turbine AND Reactor. b. Open 1NI-9A AND 10B (NC Cold Leg Inj From NV). c. Swap charging pump suction to FWST: 1) Open 1NV-221A AND 222B (NV Pumps Suct From FWST) 2) Close 1NV-141A AND 142B (VCT Outlet Isol).
____ 3. Check if Containment ventilation isolation required:	
a. EMF 38, 39 OR 40 - IN ALARM	a. Go to step 4.
b. Stop VP Fans.	
c. Stop any VQ release in progress.	
____ 4. Refer to RP/0/A/5700/01, NOTIFICATION OF UNUSUAL EVENT.	

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II NC System Leakage	PAGE NO. 9 OF 17

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 5. Attempt to identify and isolate leak:	
a. Check "Pzr PORV Disch Hi Temp" - IN ALARM	a. Go to Step b.
1) Verify Pzr PORV's - CLOSED	1) Close Pzr PORV's.
2) Monitor PORV Relief Valve Temp and cycle Pzr PORV Isol <u>AND</u> Relief Hdr Sample valves to determine leak path: • 1NC-33A <u>AND</u> 270 • 1NC-35B <u>AND</u> 269 • 1NC-31B <u>AND</u> 271.	
b. Check Cold Leg Accumulator Level - INCREASING	b. Go to step c.
1) Close CL Accum Disch Isol Valve <u>OR</u> Drain accumulator per OP/1/A/6200/09, ACCUMULATOR OPERATION.	
c. Check Pzr Relief Tank Level <u>OR</u> Temp - INCREASING ABOVE NORMAL	c. Go to step d.
1) Check inputs to PRT per Enclosure 1.	

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II NC System Leakage	PAGE NO. 10 OF 17

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
d. Check NCDT level OR temperature - INCREASING ABOVE NORMAL	d. Go to step e.
1) Check inputs to NCDT per Enclosure 2.	
e. Check Cont Flr/Eqp Sump Level - INCREASING ABOVE NORMAL	e. Go to step f.
1) Check inputs to sumps per Enclosure 3.	
f. Check inputs to Aux Building Sumps from NV System per Enclosure 4.	f. Go to step g.
g. Check ND System - IN SERVICE	g. Go to step 6.
1) Check inputs to Aux Building Sumps from ND System per Enclosure 5.	
_____ 6. IF unit shutdown is required by Tech Specs, THEN notify NRC via red phone per RP/0/A/5700/10, NRC IMMEDIATE NOTIFICATION REQUIREMENTS.	

END

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case III Letdown Or Charging Line Leakage	PAGE NO. 11 OF 17

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B. Symptoms	
• VCT level decrease or abnormal increase in frequency of auto makeup	
• Increase levels in: • ND/NS Sump • NCDT • RHT	
• "EMF-41 Aux Bldg Hi Gas Rad" alarm	
• "Letdown Relief Hi Temp" alarm.	
• Letdown or charging flows abnormal.	
C. Immediate Actions	
____ 1. Check Pzr Level - AT OR INCREASING TO PROGRAMMED LEVEL.	IF level decreasing, THEN perform the following to maintain level: a. Ensure #1 PD Pump speed increasing OR 1NV-238 (Charging Line Flow Control) opening. b. Start additional NV Pumps c. Reduce letdown to 45 GPM orifice. IF level decreases below 5%, THEN manually initiate SI AND go to EP/1/A/5000/01, SAFETY INJECTION.
____ 2. Check Pzr Press - AT OR INCREASING TO 2235 PSIG.	IF less than 2210 PSIG, THEN ensure backup heaters on. IF pressure approaches 1945 PSIG, THEN trip Reactor AND refer to AP/1/A/5500/01, REACTOR TRIP.
D. Subsequent Actions	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
<u>CAUTION</u> If Pzr level cannot be maintained, (less than 5% and decreasing) then Safety Injection should be manually initiated.	
____ 1. Announce occurrence on paging system.	
____ 2. Check Pzr Level - STABLE OR INCREASING.	<u>IF</u> level decreasing with maximum charging flow, <u>THEN</u>: a. Manually trip Turbine <u>AND</u> Reactor. b. Open 1NI-9A <u>AND</u> 10B (NC Cold Leg Inj From NV). c. Swap charging pump suction to FWST: 1) Open 1NV-221A <u>AND</u> 222B (NV Pumps Suct From FWST) 2) Close 1NV-141A <u>AND</u> 142B (VCT Outlet Isol).
____ 3. Check "EMF-41 Aux Bldg Hi Gas Rad" - IN ALARM a. Verify 1ABF-D-3 VA Filter Exh Bypass Dmpr Trn A/B closed lights - LIT. b. Verify 2ABF-D-3 VA Filter Exh Bypass Dmpr Trn A/B closed lights - LIT.	Go to step 4.
____ 4. Refer to RP/0/A/5700/01, NOTIFICATION OF UNUSUAL EVENT.	
____ 5. Attempt to identify and isolate leak.	

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case III Letdown Or Charging Line Leakage	PAGE NO. 13 OF 17

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 6. Check if letdown should be isolated:	
a. Leak on letdown line that can <u>NOT</u> be isolated by any other means	a. Go to step 7.
b. Isolate normal letdown:	
1) Close:	
• 1NV-1A (NC L/D Isol to Regen Hx)	
• 1NV-2A (NC L/D Isol to Regen Hx)	
• 1NV-241 (Seal Inj Flow Control).	
c. Adjust PD Pump speed Control <u>OR</u> Manually throttle 1NV-238 (Charging Line Flow Control) to maintain 8 GPM seal injection flow per NC Pump.	
d. Establish excess letdown per OP/1/A/6200/01, CHEMICAL AND VOLUME CONTROL.	
e. Power operation may continue as long as NC System activity and chemistry requirements are met.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 7. Check if charging headers should be isolated:	
a. Leak on charging line that can NOT be isolated by any other means	a. IF leak can be isolated, THEN operation may continue.
b. Isolate letdown:	
1) Close NC L/D Isol to Regen Hx valves: • 1NV-1A • 1NV-2A	
c. Isolate charging:	
1) Close Charging Line Cont Isol OTSD valves: • 1NV-244A • 1NV-245B.	
2) Adjust PD Pump speed control OR Manually throttle 1NV-238 (Charging Line Flow Control) to maintain 8 GPM seal injection flow per NC Pump	
d. Establish excess letdown per OP/1/A/6200/01, CHEMICAL AND VOLUME CONTROL.	
e. Power operation may continue as long as NC System activity and chemistry requirements are met.	

END

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case IV Leakage Into KC System	PAGE NO. 15 OF 17

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B. Symptoms	
• Increase in KC Surge Tank Level	
• "EMF-46 Comp Cool Hi Rad" alarm	
• "NC Pump Thermal Barrier Outlet Hi Flow" alarm	
• Increased frequency of Auto Makeup To VCT.	
C. Immediate Actions	
____ 1. Check Pzr Level - AT OR INCREASING TO PROGRAMMED LEVEL.	IF level decreasing, THEN perform the following to maintain level: a. Ensure #1 PD Pump speed increasing OR 1NV-238 (Charging Line Flow Control) opening. b. Start additional NV Pumps c. Reduce letdown to 45 GPM orifice. IF level decreases below 5%, THEN manually initiate SI AND go to EP/1/A/5000/01, SAFETY INJECTION.
____ 2. Check Pzr Press - AT OR INCREASING TO 2235 PSIG.	IF less than 2210 PSIG, THEN ensure backup heaters on. IF pressure approaches 1945 PSIG, THEN trip Reactor AND refer to AP/1/A/5500/01, REACTOR TRIP.
D. Subsequent Actions	
____ 1. Announce occurrence on paging system.	

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case IV Leakage Into KC System	PAGE NO. 16 OF 17

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 2. Check Pzr Level - STABLE OR INCREASING.	IF level decreasing with maximum charging flow, THEN: a. Manually trip Turbine AND Reactor b. Open 1NI-9A AND 10B (NC Cold Leg Inj From NV). c. Swap charging pump suction to FWST: 1) Open 1NV-221A AND 222B (NV Pumps Suct From FWST) 2) Close 1NV-141A AND 142B (VCT Outlet Isol).
_____ 3. Check "EMF-46 KC Hx Outlet" - IN ALARM a. Locally verify 1KC-122 (KC Surge Tank Vent) - CLOSED.	Go to step 4.
_____ 4. Check if any high "NC Pmp Therm Bar KC Outlet Flow" computer alarm is in: a. Verify NC Pump Therm Bar Otlt valve closes on affected pump: • A, 1KC-394A • B, 1KC-364B • C, 1KC-345A • D, 1KC-413B.	Go to step 5.
b. Verify NC Pump L/B Temp remains less than 225°F.	b. IF greater than 225°F, THEN trip NC Pump.
_____ 5. Refer to RP/0/A/5700/01, NOTIFICATION OF UNUSUAL EVENT.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

_____ 6. IF leak in Letdown Hx, THEN:

a. Close:

- 1NV-457A, 458A, 459A, (L/D Orif Otlt Cont Isol)
- 1NV-241 (Seal Inj Flow Control).

b. Adjust PD Pump speed control
OR
Manually throttle 1NV-238 (charging Line Flow Control) to maintain 8 GPM seal injection flow per NC Pump

c. Establish excess letdown per OP/1/A/6200/01, CHEMICAL AND VOLUME CONTROL

d. Power operation may continue as long as NC System activity and chemistry requirements are met.

END

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II - Enclosure 1 Possible NC System Leakage Paths To PRT	PAGE NO. 1 OF 2

VALVE NUMBER	NOMENCLATURE	INITIAL
	POSSIBLE NC SYSTEM LEAKAGE PATHS TO PRT	
	OUTSIDE CONTAINMENT	
	OUTSIDE CONTAINMENT	
1ND-56	ND HX 1A OUTLET TO NI SYSTEM COLD LEG INJECTION SAFETY RELIEF	
1ND-61	ND HX OUTLET TO NI SYSTEM HOT LEG INJECTION SAFETY RELIEF	
1ND-64	ND HX 1B OUTLET TO NI SYSTEM COLD LEG INJECTION SAFETY RELIEF	
1NS-2	NS PUMP 1B SUCTION SAFETY RELIEF	
1NS-19	NS PUMP 1A SUCTION SAFETY RELIEF	
1NI-102	SAFETY INJECTION PUMPS SUCTION HDR SAFETY RELIEF	
1NI-119	SAFETY INJECTION PUMP 1A DISCHARGE SAFETY RELIEF	
1NI-151	SAFETY INJECTION PUMP 1B DISCHARGE SAFETY RELIEF	
1NI-161	SAFETY INJECTION PUMPS COLD LEG INJECTION HDR SAFETY RELIEF	
1NV-229	CENTRIFUGAL CHARGING PUMPS SUCTION HDR SAFETY RELIEF	
	INSIDE CONTAINMENT	
1NC-1	PZR RELIEF VALVE	
1NC-2	PZR RELIEF VALVE	
1NC-3	PZR RELIEF VALVE	
1NC-32B	PZR PORV	
1NC-34A	PZR PORV	
1NC-36B	PZR PORV	
1NC-43	PRESSURIZER #1 VENT	
1NC-119	PRESSURIZER #1 SEAL LOOP DRAIN HEADER	

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACTIY OF BOTH NV PUMPS Case II - Enclosure 1 Possible NC System Leakage Paths To PRT	PAGE NO. 2 OF 2

VALVE NUMBER	NOMENCLATURE	INITIAL
1NC-272A,C	TRN 1A HEAD VENT TO PRT ISOL	
1NC-274B	TRN 1B HEAD VENT TO PRT ISOL	
1ND-3	NC LOOP 3 DISCHARGE TO ND SYSTEM SAFETY RELIEF	
1NV-6	LETDOWN LINE SAFETY RELIEF	
1NV-93	NC PUMPS SEAL RETURN HDR SAFETY RELIEF	

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II - Enclosure 2 Possible NC System Leakage Paths To NCDT	PAGE NO. 1 OF 1

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
Possible NC System Leakage Paths To NCDT				
1NV-27B	Excess L/D Hx Otlt	RB Pipechase 105°	VCT	
	3-Way Cntrl			
1NI-224	Accumulator 1A Drain	RB 725' 40°	Closed	
	Isol			
1NI-226	Accumulator 1B Drain	RB 725' 140°	Closed	
	Isol			
1NI-228	Accumulator 1C Drain	RB 725' 220°	Closed	
	Isol			
1NI-230	Accumulator 1D Drain	RB 725' 317°	Closed	
	Isol			
1NB-352	Reactor Makeup Water			
	Storage Tank #1 Outlet			
	Relief To NCDT			
NC Pump	1A #3 Seal			
NC Pump	1A Standpipe			
NC Pump	1B #3 Seal			
NC Pump	1B Standpipe			
NC Pump	1C #3 Seal			
NC Pump	1C Standpipe			
NC Pump	1D #3 Seal			
NC Pump	1D Standpipe			
	Valve Steam Leakoff from			
	RB valves			
	Rx Vessel Head O Ring Seal			

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II - Enclosure 3 Possible NC System Leakage Paths to Containment Sumps	PAGE NO. 1 OF 1

VALVE CHECKLIST

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
1NM-67	PZR Sample Hdr Cont Pent Relief	755', 120°		
1NM-68	NC Hot Leg Sample Header Cont Pent Relief			
1NM-69	NI Accumulators Sample Hdr Cont Pent Relief	730', 115°		
1NV-102	Excess Letdown Hx #1 Tube Drain	Pipechase 115° 6' up	Closed	
1NV-108	Regenerative Hx #1 Overflow	Pipechase 105° 5' up	Closed	
1NV-110	Regenerative Hx #1 Drain	Pipechase 105° 5' up	Closed	
1NI-336	UHI Check Valve Test Line Safety Relief	RB 750' 236°		

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II - Enclosure 4 Possible NV System Leakage Paths in Auxiliary Building	PAGE NO. 1 OF 6

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
NOTE	This Enclosure is to be used as a guide. Consideration should be given to any recent change in NV System alignment.			
Check Seal Leakoff on following valves:				
	ND & NS ROOMS SUMP			
1NV-95B	NC Pumps Seal Ret C/I OTSD	744' Midget Hole		
1NV-127A	L/D Hx Outlet 3-Way Temp Cntrl	NC FILTER ROOM		
1NV-137A	NC Filters OTLT 3 Way Cntrl	Outside VCT Rm. So. Wall		
1NV-141A	VCT Outlet Isolation	OTSD S Wall of VCT under grating		
1NV-142B	VCT Outlet Isol	OTSD SE Wall of VCT under grating		
1NV-803	PD Pump Outlet Isol	722' S. of PD Pump		
1NV-219	PD Pump Disch Isol			
1NV-240	Regen Hx Tube Inlet Cntrl Isol	722 HH-59 & JJ-60		
1NV-241	Seal Inj Flow Control	Above BW Pumps		
1NV-242	Regen Hx Tube Side Inlt Cntrl Isol	Above BW Pumps		
1NV-243	Regen Hx Tube Side Inlt Cntrl Bypass	Above BW Pumps		

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II - Enclosure 4 Possible NV System Leakage Paths in Auxiliary Building	PAGE NO. 2 OF 6

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
1NV-244A	Charging Line Cont Isol OTSD	Above BW Pumps		
1NV-245B	Charging Line Cont Isol OTSD	West of BW Pumps		
1NV-431	Seal Water Inj Filters Bypass	A Seal Inj Rm		
1NV-230	Cent Charging Pump B Suct	Off floor	726' SE of 1B CCP 12'	
1NV-224	Cent Charging Pump A Suct.	726' HH-57 & JJ-58 W of 1B CCP 12' Off Floor		
1NV-804	Cent Charging Pump B Outlet Isol	Right of 1B CCP		
1NV-232	Cent Charging Pump B Disch	724' NW of 1B CCP		
1NV-226	Cent Charging Pump A Disch	726' NE of 1B CCP		
1NV-802	Cent Charging Pump A Outlet Isol	NE 5' Above 1A CCP		
1NV-235	Cent Charging Pump B To Seal Inj Filter	NW of 1B CCP		
1NV-236	Cent Charging Pump A To Seal Inj Filter	NE of 1A CCP		
1NV-237	Cent Charging Pumps Disch To Control Isol	N of PD Pump		
1NV-238	Charging Line Flow Control	N of PD Pump		
1NV-239	Cent Charging Pumps Disch			

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II - Enclosure 4 Possible NV System Leakage Paths in Auxiliary Building	PAGE NO. 3 OF 6

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
	Control Isol			
1NV-347	NR System Flow Control	Seal Inj Filter Room		
1NV-121	ND Letdown Control			
1NV-221A	NV Pumps Suct From FWST	20' N of BW Pump		
1NV-222B	NV Pumps Suct From FWST	20' N of BW Pump		
	RECYCLE HOLDUP TANK			
1NV-7B	Letdown Cont Isol Outside			
1NV-8	L/D Reheat Hx Tubeside	SE of L/D Hx		
	Back pressure Cntrl Isol			
1NV-9	L/D Reheat Hx Tubeside	W of L/D Hx		
	Back pressure Cntrl Isol			
1NV-10	L/D Reheat Hx Tubeside	W of L/D Hx		
	Back Pressure Cntrl Isol			
1NV-11	L/D Reheat Hx Tubeside	SW of L/D Hx		
	Back Pressure Cntrl Isol			
1NV-476	LP Letdown Control Inlet	S of L/D Hx		
	Isol			
1NV-124	Letdown Press Control	L/D Hx Rm		
1NV-477	LP Letdown Control	L/D Hx Rm		
	Outlet Isol			

Check vent and drain boundary valves closed:

WASTE DRAIN TANK

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II - Enclosure 4 Possible NV System Leakage Paths in Auxiliary Building	PAGE NO. 4 OF 6

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
1NV-184	Letdown Hx Tube Drain To WDT	L/D Hx Rm	CLOSED	
1NV-205	Seal Water Filter Drain To WDT	Seal Ret Filter Rm	CLOSED	
1NV-272	PD Pump Drain To WDT	Otsd PD Pump Rm	CLOSED	
1NV-310	Seal Water Inj Filters Drain To WDT	B Seal Inj Rm	CLOSED	
1NV-299	Charging Pump B Drain To WDT	E of 1B CCP	CLOSED	
1NV-330	NC Filter Drain To WDT	E of B NC Filters	CLOSED	
1NV-356	Mixed And Cation Bed Demin Outlet Line Drain To WDT	A Mixed Bed Rm	CLOSED	
	WASTE EVAPORATOR FEED TANK			
1NV-181	Letdown Hx Tube Overflow	L/D Hx Rm	CLOSED	
1NV-185	Letdown Hx Tube Drain To WEFT	LD Hx Rm	CLOSED	
1NV-145	VCT Outlet Drain	Below VCT	CLOSED	
1NV-210	Seal Water Hx Tube Overflow		CLOSED	
1NV-204	Seal Water Filter Drain To WEFT		CLOSED	
1NV-215	Seal Water Hx Tube Drain TO WEFT	Seal Water Hx Rm	CLOSED	
1NV-309	Seal Water Inj Filters	B Seal Inj Filter Rm	CLOSED	

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS Case II - Enclosure 4 Possible NV System Leakage Paths in Auxiliary Building	PAGE NO. 5 OF 6

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
	Drain To WEFT			
1NV-329	NC Filters Drain To WEFT	B NC Filter Rm	CLOSED	
1NV-335	Mixed Bed Demin A Backflush Drain			
1NV-333	Mixed Bed Demin A Backflush Outlet Isol		CLOSED	
1NV-340	Mixed Bed Demin B Backflush Outlet Isol	B Mixed Bed Rm	CLOSED	
1NV-373	Mixed Bed Demin B Outlet Line Drain	B Mixed Bed Rm	CLOSED	
1NV-365	Cation Bed Demin Sluicing Resin Isol	733 Pipechase between NR & NV DIM	CLOSED	
1NV-354	Mixed Bed Demin A Outlet Line Drain	A Mixed Bed Rm	CLOSED	
1NV-366	Cation Bed Demin Outlet Line Drain	Cation Bed Rm	CLOSED	
1NV-357	Mixed & Cation Bed Demins Outlet Line Drain To WEFT	A Mixed Bed Rm	CLOSED	
	WASTE EVAPORATOR FEED TANK SUMP A			
1NV-296	Charging Pump B Overflow		CLOSED	
1NV-300	Charging Pump B Drain To WEFT Sump A		CLOSED	
1NV-285	Charging Pump A Overflow		CLOSED	
1NV-289	Charging Pump A Drain To WEFT Sump A		CLOSED	

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
	SPENT RESIN TANK			
1NV-349	Mixed Bed Demin A Sluicing Resin Isol		CLOSED	
1NV-350	Mixed Bed Demin A Sluicing Resin Isol		CLOSED	

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS CASE II - Enclosure 5 Possible ND System Leakage Paths in Auxiliary Building	PAGE NO. 1 OF 2

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
	ND & NS ROOMS SUMP Check seal leakoff on following valves			
1ND-4B	B ND Pmp Suct From FWST Or NC	Aux 695' FF-59 & GG-60		
1ND-19A	A ND Pmp Suct From FWST Or NC	Aux 695' GG-59 & HH-60		
1ND-9	ND Pump B Disch	N of Pump 12' up		
1ND-24	ND Pump A Disch	W of Pump 12' up		
1ND-26	ND Hx A Inlet	E of Hx 6' up		
1ND-14	B ND Hx Outlet	Aux 733' LL-61		
1ND-29	A ND Hx Outlet	Aux 733' LL-60 & MM-61		
1ND-30A	Train 1A ND To Hot Leg Isol	Aux 733' LL-60 & MM-61		
1ND-58A	Train 1A ND To NV & NI Pumps	Aux 733' LL-60 & JJ-61		
1ND-15B	Train 1B ND To Hot Leg Isol	Aux 733' KK-60 & LL-61		
1ND-35	ND To FWST Isol	15' E of KK-59, 12' up		
1ND-34	A & B ND Hx Bypass	Aux 733 KK-60 & LL-61		
1ND-33	A ND Hx Bypass	Aux 733 LL-60 & MM-61		
1ND-18	B ND Hx Bypass	Aux 733 KK-60 & LL-61		
1ND-11	ND Hx B Inlet	W of Hx 4' up		
1NI-173A	Train 1A ND To A & B CL	Aux 733' FF-59 & GG-60		
1NI-178B	Train 1B ND To C&D CL	Aux 733' HH-60 & JJ-61		
1NI-184B	RB Sump To Train 1B ND & NS	Aux 716' EE-58 & FF-59		

AP/1/A/5500/10	NC SYSTEM LEAKAGE WITHIN CAPACITY OF BOTH NV PUMPS CASE II - Enclosure 5 Possible ND System Leakage Paths in Auxiliary Building	PAGE NO. 2 OF 2

VALVE NUMBER	NOMENCLATURE	VALVE LOCATION	POSITION	INITIAL
1NI-185A	RB Sump To Train 1A ND & NS	Aux 716' FF-59 & GG-60		
	WASTE EVAPORATOR FEED TANK Check Drain Boundary Valves Closed			
1ND-52	ND HX A Drain Hdr	S of Hx	Closed	
1ND-46	ND Hx B Drain Hdr	S of Hx	Closed	
ND & NS ROOMS SUMP				
1ND-51	ND Pump A Drain Hdr	SW of Pump	Closed	
1ND-45	ND Pump B Drain Hdr	S of Pump	Closed	
1ND-69	ND & NS System Drain	RB 860' Rx Dome	Closed	

Form 34731 (R5-88)

MASTER FILE

Duke Power Company
PROCEDURE PROCESS RECORD

(1) ID No. OP/1/A/6100/02
Change(s) 0 to 67 Incorporated

SUPERCEDED

PREPARATION

(2) Station McGuire Nuclear Station

(3) Procedure Title Controlling Procedure for Unit Shutdown

(4) Prepared By Verida Bellamy Date 1/20/89

(5) Reviewed By [signature] Date 1/20/89

Cross-Disciplinary Review By N/R

(6) Temporary Approval *(if necessary)*

By (SRO) Date

By Date

(7) Approved By Bruce Travis Date 2/1/89

(8) Miscellaneous

Reviewed/Approved By [signature] Date 1-20-89

Reviewed/~~Approved~~ By ETQS Date 1/20/89

(9) Comments (For procedure reissue indicate whether additional changes, other than previously approved changes, are included. Attach additional pages, if necessary.)

Additional Changes Included. ☒ Yes ☐ No

(10) Compared with Control Copy Date

(11) Requires change to FSAR not identified in 10CFR50.59 evaluation? ☐ Yes ☒ No
If "yes", attach detailed explanation.

Completion

(12) Date(s) Performed

(13) Procedure Completion Verification

☐ Yes ☐ N/A Check lists and/or blanks properly initialed, signed, dated or filled in N/A or N/R, as appropriate?

☐ Yes ☐ N/A Listed enclosures attached?

☐ Yes ☐ N/A Data sheets attached, completed, dated and signed?

☐ Yes ☐ N/A Charts, graphs, etc. attached and properly dated, identified and marked?

☐ Yes ☐ N/A Procedure requirements met?

Verified By Date

(14) Procedure Completion Approved Date

(15) Remarks *(attach additional pages, if necessary)*

2.15.1.1 Ensure the "Operation Selector" for all 6 detectors is in the "Off" position.

2.15.1.2 Open and tag the 120 VAC main power breaker on the panel.

_____ 2.16 Determine required boron concentration to establish greater than or equal to 1.3% Delta k/k shutdown margin at desired temperature per Table 6.5 of OP/1/A/6100/22 (Unit 1 Data Book). If cooldown below 200°F is anticipated, ensure greater than or equal to 1% Delta k/k shutdown margin at 68°F prior to reduction below 200°F.

CAUTION The following step must be completed and NC System boron concentration verified prior to initiating NC System cooldown.

_____ 2.17 Borate the NC System per OP/1/A/6150/09 (Boron Concentration Control) or OP/1/A/6200/02 (BTRS) to establish the appropriate shutdown margin.

2.18 Have IAE do the following:

_____ 2.18.1 When the neutron level decays to the normal shutdown counts, verify "High Flux At Shutdown" alarm bistable is set at one-half decade above normal shutdown source counts, and reinstate "High Flux At Shutdown" alarm.

_____ 2.19 After the "High Flux At Shutdown" alarm has been reinstated, insert the Shutdown Banks per OP/1/A/6150/08 (Rod Control).

_____ 2.20 Remove both MG sets from service per OP/1/A/6150/08 (Rod Control).

_____ 2.21 As soon as access to lower containment is possible, close 1NC-24 (Reactor Vessel Head Gasket Leakoff Drain Manual Block) to prevent NCDT H_2 from escaping to containment during cooldown.