Enclosure
to U-602643

Special Assessment:
Confirmatory Action Letter Dated September 11, 1996

Prepared by: _______________ 9/21/96
Date

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Date

Executive Summary

On September 5, 1996, Clinton Power Station personnel, in pursuit of an increasing trend of reactor coolant leakage, formulated plans and performed procedures to mitigate the consequences of excessive reactor coolant leakage by shifting the operation of the Reactor Recirculation System from two-loop to single-loop operation. During the course of these actions, the reactor coolant leakage rate increased to rates which are not allowed by the Plant's Technical Specifications (T.S.-3.4.5.b ≤5 gpm) resulting in the declaration of a Notification of Unusual Event. Subsequently, the Reactor Recirculation pump seal failed causing leakage rates to again increase beyond limits allowed by the Technical Specifications (T.S.-3.4.5.d ≤2 gpm increase within 24 hours).

On September 9, 1996, the Nuclear Assessment Department began an investigation and evaluation of the events surrounding the activities of the above described event and the follow-on of the reactor shutdown. Our assessment identified the following conditions:

- Management personnel were not conservative in the operation of the plant. Numerous opportunities existed for management to evaluate uncertainties and the need for continued reactor operation, but these were not recognized or acted upon.

- Management did not properly establish, enforce or set the proper example for procedure compliance.

- Oversight of the overall picture of plant conditions and actions surrounding the event was ineffective.

- There was inadequate planning and evaluation of the potential consequences prior to performing an infrequently performed operation.

- Management tolerated long standing equipment problems that contributed to uncertainty of some plant conditions.

- IP personnel were not timely in identifying a potential procedure non-compliance and management did not sufficiently pursue other indications that would lead to management recognition of the significance of non-conservative reactor operation. It should be noted that IP was prompted by the NRC to pursue the issues of procedural compliance.

Many of these concerns were previously identified in letter RFP96-026 on June 10, 1996, Assessment of On-line Maintenance. Details of the above conclusions are contained in the detailed chronology (Attachment A) and the evaluation that follows.

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This section has been revised to incorporate additional information and evaluation from further investigation performed by the Nuclear Assessment Department. Revisions are not marked by revision bars.

I. Evaluation

This analysis of the evolution and events surrounding the September 5, 1996 Reactor Recirculation pump seal failure at Illinois Power's Clinton Power Station evaluates the following: Decisions made and actions taken and their basis; contingencies evaluated related to the decisions and their consequences, management involvement and oversight.

1730 to 1800
Action: Pre-Evolution Briefing
Basis: The pre-evolution briefing was conducted to ensure that personnel involved in the evolution understood their roles and responsibilities, to review the applicable procedures, and to discuss potential problems that might be encountered. The detailed Chronology, Attachment A, provides attendees and topics covered.
Management Involvement: The Assistant Director-Operations (ADO) (also acting as Director-Operations at this time) was present at the briefing and had previously provided the Shift Supervisor, (SS) with his written expectations (Attachment 1 to the Chronology).
Evaluation: The brief was considered by the personnel in attendance to be one of the best they had attended. Although contingencies were discussed, it is not clear that specific actions to be taken for each contingency were covered. It appears the focus was on completing the isolation and determining whether or not the Drywell Floor Drain (DWRF) leakage rate stabilized below the administrative limit of 4.5 gpm. This limit had been established by Operations management with the Plant Manager's concurrence and promulgated via the expectations paper provided to the SS. Although the potential for leakage to increase was discussed, it appears that it was assumed the crew would proceed with isolation of the loop and then evaluate the remaining steady state leakage. It does not appear that the discussion addressed that an orderly shutdown or SCRAM would be required if an upper limit of leakage was reached. Although there are indications that CPS 3302.01 REACTOR RECIRCULATION (RR), the procedure for operating the Reactor Recirculation System (including isolating one loop), was reviewed prior to and during the pre-evolution briefing, the reviews did not identify what effect each step of the procedure could have on the rate of RR pump "B" seal leakage. If this had occurred, the crew might have known in advance that they would most likely have to enter Technical Specification 3.4.5 Condition A and a Notification of Unusual Event (NOUE) if they proceeded with the evolution. This might have caused additional questions to be asked and analysis to be performed, and the specific procedure steps to be followed under these conditions to be considered. The personnel involved in the event believed that the single loop isolation could be performed but the mindset seemed to be that the DWRF leakage would decrease, or at worst, remain the same. There was no consideration for an incremental increase in flow and therefore there was no contingency plan for actions to be taken if flow increased. In general, the focus of the crew and management appears to have been on the performance of the loop isolation. For this reason, the evolution and worst case possibilities from

a reactor safety/primary coolant boundary perspective were not adequately analyzed, including limits that would require the initiation of pre-determined conservative actions. This is considered a lack of safety focus and inadequate planning.

Additionally, the expectations provided to the SS by the ADO included provisions to perform a double isolation of the Reactor Water Cleanup system (RT) if leak-by from RT to RR was suspected. The double isolation of RT is not addressed in CPS 3303.01, "REACTOR WATER CLEANUP". Although PMSO-043, "VERBAL INSTRUCTIONS TO CONDUCT OPERATIONS", allows management to provide simple instructions that do not contradict existing procedures, isolation of the RR loop was a preplanned evolution and the procedure should have been revised to incorporate the additional guidance prior to beginning the evolution. Furthermore, PMSO-043 requires that verbal instructions provided by the ADO be logged. No log entry was made. This is a violation of PMSO-043 and has been documented in CR 1-96-09-079. It is also another example of inadequate pre-planning.

The evolution was begun knowing that equipment problems existed. One of the isolation valves, the "B" RR Loop Discharge Valve (1B33-F067B), had previously provided dual indication when closed, leaving its status indeterminate in terms of whether or not it was fully closed. Furthermore, the V-notch systems utilized to determine both DWRF leakage and Drywell equipment drain (DWRE) leakage were inoperable. This is an example of Management willingness to accept operations with degraded equipment.

Finally, during the pre-evolution analysis and the pre-evolution briefing, the personnel evaluating and planning the overall evolution did not sufficiently consider conditions that could prevent its success. It should have been known that some amount of leak-by existed in the isolation valves for the "B" RR loop. Management is at fault for requesting an evolution that had a limited potential for success. All organizations involved in the planning, preparations, and performance were at fault for not challenging the plan more aggressively.

1805-2009

Action: Reduced Reactor Power to prepare for implementing RR B Loop isolation.
Basis: Procedure CPS 3302.01 Section 8.2.1, "RR Loop Shutdown During Plant Operations", requires that reactor power be at a level specified by Management and Nuclear Engineering and evaluated against the Power to Flow Map prior to performing a pre-planned loop isolation.
Management Involvement: The ADO was present in the main control room area.
Evaluation: Reduction in power was accomplished in accordance with procedures in an orderly manner.

2009

Action: Shutdown of "B" RR pump in accordance with CPS 3302.01. This included shutdown of RR Pump B (1B33-C001B), closure of the pump B discharge valve (1B33-F067B), and fully opening the Flow Control Valve (1B33-F060B) per steps 8.2.1.3, 8.2.1.4, and 8.2.1.5.

Basis: This is the sequence of steps for RR pump shutdown prescribed by the procedure.
Management Involvement: The ADO was still available in the area.
Evaluation: Based on previous experience that closure of the discharge valve resulted in dual indication and the inability to tell whether the valve had fully closed, electricians and the VOTES engineer had instrumented the valve in advance to provide more positive indication of closure. The indication included motor current traces and stroke time. This was considered good pre-planning for the evolution. When shut, all indications provided were that the valve fully closed including a green only position indication on the Main Control Room panel. At this time, the pump seal pressures, P1 and P2, began to equalize. The reason for this was not fully understood, nor was this expected to occur. The crew began to analyze what could be causing the equalization. The system engineer determined that the lower seal (P1) had failed, causing pressure to equalize throughout the seal cavity. It is not clear that the system engineer effectively communicated this conclusion to the SS when this occurred.

At this point, DWRF leakage had not increased and the crew believed the evolution was proceeding as expected with the exception of the seal pressure equalization.

2009-2030
Action: Initiate RR loop isolation by closure of the Reactor Water Cleanup (RT) recirculation loop "B" suction valve (1G33-F106) and RR Pump B Discharge Valve (1B33-F067B) steps 8.2.4.1 and 8.2.4.2 of CPS 3302.01. Continue with step 8.2.4.3 which is an "If" - "Then" statement. This statement says that IF "loop suction isolation is required due to an emergency condition (system seal leak)", THEN " a) shut 1B33-F023A(B), pump suction valve. b) shut 1B33-F075A(B), pump A(B) seal staging shutoff valve." When the "IF" condition is satisfied, the "THEN" actions are performed. When the "IF" condition is not satisfied, the individual(s) performing the procedure proceeds to step 8.2.4.4. Step 8.2.4.4 provides the conditions required prior to isolating CRD Seal Injection Flow. The conditions specified for the idle loop are:

° Normal conditions
Cooldown to <250°F

° Emergency conditions (loop shutdown from 8.2.3)
Cooldown to <250°F; or in the event the RR seals are failed, loop depressurized to ~ drywell pressure.

Basis: This is the sequence of steps prescribed by procedure CPS 3302.01 for single loop isolation.
Management Involvement: The ADO was still present in the general area.
Evaluation: Section 8.2.4 contains an asterisk (*) which per Step. 8.1.11.4 of 1005.14, "FORMATTING OF CPS PROCEDURES AND DOCUMENTS", states that asterisk sections are to be performed in the sequence that they are written. Step 8.2.4.1 was properly completed. Step 8.2.4.2, which closes the discharge valve, had already been completed per the pump shutdown procedure section, Step 8.2.1.4.

Step 8.2.4.3 is in an "IF-THEN" format such that unless the conditions of the IF statement are met (in this case described as "an emergency condition"), the actions of the THEN statement are

not to be performed. Since the SS did not consider himself in an emergency condition due to seal leakage, he proceeded to step 8.2.4.4. Proceeding to step 8.2.4.4 was appropriate because the IF portion of Step 8.2.4.3 was not satisfied.

Step 8.2.4.4 provides conditions to be satisfied prior to proceeding to the next step in the procedure. The order of steps 8.2.4.4, 8.2.4.5, and 8.2.4.6 is somewhat confusing because although 8.2.4.4 precedes 8.2.4.5, the wording of the conditions included in 8.2.4.4 does not refer to the actions described in 8.2.4.5, but to those described in 8.2.4.6. The SS interpreted Step 8.2.4.4 to be more of a caution than a step, and then incorrectly proceeded to Step 8.2.4.5 without having satisfied either condition in Step 8.2.4.4. This was a violation of procedure CPS 3302.01 and has been documented in Condition Report 1-96-09-078.

2030

Action: Closed the pump "B" Seal Staging Shutoff Valve (1B33-F075B) per Step 8.2.4.5 of CPS 3302.01.

Basis: When the crew did not meet the required conditions for Step 8.2.4.4, they incorrectly omitted the step and proceeded to Step 8.2.4.5, which directs closure of the Seal Staging Shutoff Valve. As noted above, because neither condition of step 8.2.4.4 was satisfied, this was out-of-sequence and therefore a procedure violation of CPS 3302.01 (CR 1-96-09-078). The crew thought that closing the Seal Staging Shutoff Valve would force more cold Control Rod Drive (CRD) water into the partially isolated loop, thus accelerating the rate at which the loop cooled.

Management Involvement: The ADO was still available in the area.

Evaluation: This step is part of the routine sequence for isolation of a reactor recirculation loop. However, the planning for the evolution and/or the pre-evolution brief did not consider all of the effects that shutting 1B33-F075B would have on a degraded seal. This step of the procedure was written under the assumption that the seal was in normal working order. Prior to shutting 1B33-F075B, DWRF leakage (unidentified leakage per Technical Specifications) was approximately 4.7 gpm. Twenty-five minutes after shutting 1B33-FO75B, DWRF leakage increased to approximately 5.5 gpm. 1B33-F075B diverts approximately 1 gpm of RR "B" seal leak-off to the DWRE, which is considered identified leakage by the Technical Specifications. Shutting this valve provided approximately one (1.0) additional gpm of flow that could either go into the loop or out through the seal. It is believed that most or all of the 1.0 gpm of flow did exit through the seal, thus increasing the flow to the DWRF which was added to the already existing unidentified leakage. When DWRF leakage exceeded 5 gpm, Condition A of Technical Specification 3.4.5 was entered. This requires restoration of unidentified leakage to less than 5 gpm within four hours. Additionally, unidentified leakage greater than 5 gpm requires a Notification of Unusual Event (NOUE) be declared in accordance with the CPS Emergency Plan per procedure EC-02, "EMERGENCY CLASSIFICATIONS". Had the procedure been sufficiently analyzed prior to the evolution, personnel might have identified that closure of the seal staging valve could increase leakage to above the action level. This would have allowed further analysis regarding the merit of proceeding with the evolution and contingencies should Technical Specifications Action statements be reached.

2055
Action: The crew entered CPS 4000.01, "REACTOR COOLANT LEAKAGE" off-normal procedure, entered the Action Statements required by TS 3.4.5, and declared the NOUE.
Basis: Unidentified leakage above 5 gpm requires entry into the Action Statements of Technical Specification 3.4.5 and declaration of an NOUE per Emergency Plan Procedure, EC-02.
Management Involvement: The ADO was closely following the events at this time and began conferring more often with the Shift Supervisor.
Evaluation: The initial actions taken at this time were appropriate and in compliance with the governing procedures. The entrance into the off-normal procedure, TS Action statement, and NOUE were an opportunity for shift management to assess overall plant conditions. When 4001.01, "REACTOR COOLANT LEAKAGE," was entered, Step 4.4, which directs the control room staff to notify Radiation Protection to help in the identification of the source of leakage, was not performed. This is a violation of Procedure CPS 4001.01 and is documented on Condition Report 1-96-09-114.
Additionally, the Shift Supervisor asked the STA if he had time to perform the required notifications. The STA affirmed that he did and took the responsibility to make the notifications. In hindsight, both the Shift Supervisor and the STA indicated that they should have had the responsibility assigned to an operator, because this placed the STA in a role different than he normally fulfilled and may have precluded him from exercising a more general oversight role.

2116-2127
Action: The crew depressurized the Drywell using Mixing Compressor "A" and then secured Mixing Compressor "A" when Drywell pressure was reduced to 0.23 psig (Time 2127).
Basis: This provided additional margin between the drywell pressure and the auto scram setpoint for this parameter.
Management Involvement: The ADO continued to closely follow the events and confer with the Shift Supervisor.
Evaluation: This was good anticipation and preplanning by the shift crew.

2130
Action: The crew returned to step 8.2.4.3.a of CPS 3302.01 and closed the RR pump suction valve (1B33-F023B). Because the seal staging shutoff valve (1B33-F075B) had been previously closed in step 8.2.4.5 at time 2030, the second part of this step, 8.2.4.3.b, had already been completed.
Basis: The crew returned to Step 8.2.4.3.a because the SS, with concurrence of the ADO, considered the NOUE to meet the criteria (Emergency Condition) for performing this step.
Management Involvement: Prior to returning to Step 8.2.4.3.a, the Shift Supervisor discussed the condition with the ADO. The ADO concurred that the Unusual Event did constitute an emergency condition, which they thought justified returning to the step.
Evaluation: Returning to step 8.2.4.3.a at this point in time is not considered appropriate. The asterisk at the beginning of Section 8.2.4 requires that the steps of the procedure be performed in the sequence specified. Although the plant was in an Unusual Event and Technical Specification Action Statement, this did not justify returning to Step 8.2.4.3.

The situation presented to the crew at this time is not adequately addressed by section 8.2.4 of CPS 3302.01, at least in part due to the fact that the SS had incorrectly completed step 8.2.4.5 prior to meeting either condition specified in step 8.2.4.4. The SS had to reduce the DWRF leakage to below 5 gpm within four hours or begin a shutdown of the reactor. He could not complete the loop isolation because he did not meet the conditions of step 8.2.4.4 and was thus unable to proceed to step 8.2.4.5. He knew that the cooldown time for the partially isolated loop would exceed the four hours allowed by Technical Specification 3.4.5 Condition A. The SS was also implementing CPS 4001.01, "REACTOR COOLANT LEAKAGE". He believed it was important to isolate the reactor coolant leakage which had caused him to enter the Technical Specification Action Statement and declare the NOUE. Because he did not have the loop completely isolated and considered himself to be in an emergency situation, he believed he was justified in returning to step 8.2.4.3a, which directs the closure of the "B" RR Pump Suction Valve, (1B33-F023B).

At this time, the SS, ADO or STA in their role of monitoring the overall plant conditions from a broader perspective, should have recognized that the appropriate action was to shut down the reactor.

Instead, the SS with concurrence of the ADO made the decision, based on considering the plant to be in an emergency condition, to return to Step 8.2.4.3 and complete the portion (8.2.4.3.a) which had not already been completed. This is a violation of procedure CPS 3302.01 (CR-1-06-09-078) and was also not the appropriate conservative action.

Although performance of Step 8.2.4.3.a, which closed the "B" RR Pump Suction Valve (1B33-F023B) was a procedure violation, it completed the isolation of the "B" RR Loop from the reactor in compliance with CPS 4001.01, "REACTOR COOLANT LEAKAGE".

2130-2159
Action: Main Control Room personnel monitored plant conditions, specifically, DWRF and RR "B" seal pressures for change. During this period of time, (2144) unidentified reactor coolant leakage exceeded a 2 gpm increase in 24 hours. Unidentified reactor coolant leakage indicated approximately 6 gpm. Twenty-four hours prior to this, the reading was 3.8 gpm. Technical Specification 3.4.5 Condition A was re-entered for the 2 gpm leakage increase. RR "B" pump seal pressures during this period were steady at 980 psig on P1 and P2.
Basis: Personnel were monitoring DWRF leakage and seal pressures to determine whether closure of the "B" loop suction valve had an effect.
Management Involvement: Within this general time frame, the Illinois Power Company Vice President-Nuclear called the Shift Supervisor to obtain a status after having been notified via his pager that an emergency had been declared. He was informed that the loop had been isolated and that we were in single loop operation. In the conversation, it was understood that if the leakage did not come down, a plant shutdown would be required. The ADO was still present and conferring with the SS periodically for updates and to provide suggestions if requested.

Evaluation: During this time, Technical Specification 3.4.5 Condition A was re-entered for the 2 gpm leakage increase. This should have been another indicator that it was time to reassess the situation and at least evaluate whether a shutdown of the reactor should be considered. Because the leakage rate was basically steady state, and there was no perceived immediate risk, management attention continued to focus on DWRF leakage reduction. The plant had experienced DWRF leakage increases twice, entrance into Technical Specification Action Statements twice, declaration of a NOUE, and entrance into an off-normal procedure (CPS 4001.01, "REACTOR COOLANT LEAKAGE"), yet none of these occurrences precipitated management or shift supervision stepping back and evaluating the overall plant status and potential concerns related to reactor safety. Management was involved in the evolution from a perspective of assisting with ideas for leak isolation instead of taking a broader view of overall plant safety.

2159

Action: Closed the Control Rod Drive (CRD) supply isolation to RR pump "B" (1C11-F026B) per CPS 3302.01, step 8.2.4.6. This completed the RR loop isolation per section 8.2.4 of CPS 3302.01.

Basis: The SS and ADO, with input and discussion from the crew, had evaluated that the isolated loop was not going to depressurize through the seal with CRD flow valved in. Because DWRF leakage was determined to be approximately the same as CRD make-up flow, it was thought that the leakage could not be reduced without isolating the CRD make-up flow. Additionally, the shift was focused on reducing DWRF leakage to comply with CPS 4001.01. One concern with maintaining CRD injection was that overpressurization and further damage to the seal could occur. The Shift Supervisor considered that after the loop was isolated from the reactor, pressure from CRD injection would either cause pressure to increase on the seal or that seal pressure would start to slowly decrease. If the seal started to pressurize, he thought it would mean leakage also was coming from outside the loop isolation boundaries. If it started to decrease, it would mean the seal leak rate was now greater than the CRD injection rate. This was assuming that the loop isolation valves were not leaking by. He expected increased pressure to either lift the injection relief valve, or require reopening the suction valve so pressure could be released to the reactor. The potential for catastrophic seal failure when CRD seal flow was secured and the resulting potential for drywell contamination from the reactor water in the loop depressurizing through the seal was discussed with the RR System Engineer and the ADO. They decided that catastrophic seal failure was not likely and that further degradation of an already damaged seal was acceptable. The evaluation that catastrophic seal failure was not likely was based on the System Engineer's evaluation that isolating the CRD injection would increase the stress on the seal and possibly incrementally increase damage, but that the seal would probably not have a severe failure. They also decided that isolating the RR pump "B" seal leakage was more important than the potential to contaminate the drywell. They were also confident because of the full-closed position indication in the MCR and the data from the instrumented "B" loop suction and discharge valves, that the "B" recirculation loop was securely isolated. An additional consideration was that should a catastrophic failure of the seal occur, that the isolated finite volume of coolant contained in the loop would quickly depressurize, and that any immediate

increase in initial DWRF drain flow rate would soon decrease. This assumed that the isolation valves did not leak-by.

Management Involvement: The Shift Supervisor, with concurrence of the ADO, made the decision to close the CRD supply isolation to RR pump "B" (1C11-F026B).

Evaluation: Shift Management was focused on the details of the evolution rather than maintaining a focus on the broader plant status. The SS and ADO had assumed the role of managing the evolution and trying to solve the problems being encountered, rather than maintaining their proper oversight function. Closure of the CRD supply was another error for two reasons. First, management, the SS, and the crew continued attempts to reduce DWRF leakage to meet the requirements of CPS 4001.01 by isolating CRD. They were caught up in the evolution and appeared to be thinking of the current leakage as "unisolated reactor coolant leakage". They failed to recognize that the intent of CPS 4001.01 was to isolate reactor coolant leakage which was completed when the suction valve was closed. Isolating CRD injection did nothing to reduce reactor coolant leakage because closure of the "B" RR Loop Suction Valve had completed the isolation of the loop from the reactor (except for valve leak-by). If anything, it replaced the CRD water leaking out of the seals with reactor coolant leakage. Because the CRD injection also provides some cooling to the seal, this jeopardized an already degraded seal even further.

The second reason that this was an error is that the conditions required by the procedure prior to isolating the CRD injection flow were not satisfied. The two conditions dealt with allowing the loop to cool below 250°F as a protection to the seal, and preventing contamination and airborne activity in the drywell by not allowing the pressurized reactor water in the isolated loop to depressurize through the seal. However, the Shift Supervisor and the Assistant Director of Operations believed that awaiting these conditions would preclude isolation of the leak per CPS 4001.01. Both conditions were evaluated, and the judgments involved had some technical merit, but did not justify the procedure violation. A more conservative approach would have been to implement a controlled reactor shutdown while monitoring leakage and allowing temperature to decrease. It was evident that the temperature of the isolated loop could not be expected to decrease to below 250°F, nor could it be expected that the loop would depressurize in the near term. For these reasons, it was evident that the conditions required to isolate CRD flow would not be satisfied prior to entrance into the plant shutdown required by Technical Specification 3.4.5. Because the crew and management oversight were focused on reducing the leak and not on overall plant safety, isolating the CRD flow appeared reasonable and justifiable.

When the crew closed 1C11-F026B CRD supply Isolation to RR Pump "B" per step 8.2.4.6 of 3302.01 without having met the conditions of Step 8.2.4.4, a Condition Report should have been written to document the deviation from the procedure per Step 6.1 of 1001.05, "AUTHORITY AND RESPONSIBILITIES OF THE REACTOR OPERATOR FOR SAFE OPERATION AND SHUTDOWN". This was a violation of procedure CPS 1001.05 and CPS 3302.01 and has been documented in Condition Report 1-96-09-078.

The input provided by the System Engineer to the SS which indicated that closure of the CRD injection valve was not likely to cause severe failure, does not appear to have had a rigorous

technical basis. It is recommended that this decision receive additional evaluation by the Nuclear Station Engineering Department.

As noted above, the crew and management present expected that even if the seal should fail, that the loop coolant would quickly depressurize and any increase in DWRF flow would soon decrease. It is not clear that the crew or management evaluated the situation or established a contingency in the event the floor drain flow rate did not decrease. The crew appeared to be confident that they could anticipate and predict what would occur and did not take the time to evaluate what the specific action should be, should the unexpected occur.

Additionally, interviews with the Reactor Operators (RO's) indicated that one of the RO's made a suggestion to first throttle CRD supply back by 2 gpm. However, this suggestion, which was identified directly to the Shift Supervisor, does not appear to have been evaluated for implementation prior to securing the CRD injection. This would have been a more conservative approach, and the lower procedure limit of 3 gpm could have been tried. Procedure CPS 3302.01, step c8.1.1.2 requires that CRD seal injection flow be between 3 and 5 gpm. The crew could have written a TPD to allow throttling the valve and this would have maintained compliance with the procedure and also allowed an evaluation of reduced CRD flow to the seal.

If not before, at this time in the event previous operator training on industry lessons learned should have helped the operators realize they were proceeding in a non-conservative manner. Specifically, lessons learned from INPO SOER 94-1, "Nonconservative Decisions and Equipment Performance Problems Result in a Reactor Scram, Two Safety Injections, and Water-Solid Conditions," were clearly relevant to the situation facing the crew. A comparison to the conclusions from the SOER and the situation facing the crew is as follows (conclusions are paraphrased):

- SOER Conclusion: Operators were challenged by degrading plant conditions and equipment malfunctions that were not directly covered by procedural guidance. Additionally, the operating crew did not perceive these conditions as deviating significantly from the expected, and they did not take prompt action to place the plant in a stable condition as a conservative response.

- CPS Similarities
  - RR "B" pump seal pressures (P1 and P2) equalized
  - Incremental increase in RR "B" pump seal leakage leading to
    - Technical Specification 3.4.5 Condition A
    - NOUE
    - Entry into CPS 4001.01, "Reactor Coolant Leakage"
  - Second entry into Technical Specification 3.4.5 (Condition "d" for leakage increase > 2 gpm in 24 hours)
  - Seal depressurization (from ≅ 980 psi to ≅ 280 psi)
  - Need to start mixing compressor to depressurize the drywell
  - Evacuation of containment
  - Leak detection display going to "white data"

- SOER Conclusion: The operating crew demonstrated weaknesses in the following areas:

  - Supervisory oversight
  - Teamwork
  - Communications
  - Resource usage

- CPS Similarities:
  - Supervisory oversight
    - Focused on leakage, did not maintain broad oversight.
    - No specific limits for leakage provided.
    - Concurred with procedure violation.

  - Teamwork
    - Some discussions were held outside the "at the controls" area by the SS and ADO which prevented other crew members from hearing the discussion and providing input.

  - Communications
    - It's not clear the SS understood what he was being told when the system engineer (SE) informed him that the RR "B" pump seals (P1 and P2) had equalized in pressure because the lower seal, P1, had failed. It is also not clear that the SE explained this clearly.
    - The suggestion to throttle CRD injection vice isolate it was not remembered or evaluated by the SS.
    - It is not clear that questions/concerns related to the double isolation of the RT system were adequately resolved.

  - Resource Usage
    - The STA was assigned to make the state and federal notifications when the NOUE was declared, which gave him a line responsibility instead of his continuing in a purely oversight/advisory role.

SOER Conclusion: Equipment malfunctions and degraded equipment complicated analysis and understanding of plant status.

CPS Similarities:
- Inoperable V-notch
- Loss of leak detection indication provided by the LD-027 modification
- RR Pump "B" Lower Seal (P1) failure
- RR Loop "B" Discharge/Suction valve leak-by

SOER Conclusion: Procedure deficiencies and weaknesses contributed to Human Performance Deficiency.

CPS Similarities:

- Step 8.2.4.4 of CPS 3302.01 addresses conditions that appear to apply to step 8.2.4.6.
- Procedure did not clearly/correctly define at what point an "Emergency" existed which is needed to understand step 8.2.4.3.
- Step 8.2.4.4 does not provide clear guidance on what constitutes a "Failed RR Seal."
- The procedure does not provide guidance on actions to be taken if incremental seal failure or increasing leakage occurred during the evolution.

SOER Conclusion: Knowledge and skill deficiencies contributed to the event.

CPS Similarities:

- Operators did not understand the operation of LD-027 indication.
- This assessor could not determine that operators had ever practiced routine loop isolation in the simulator (with or without incremental leakage increase).
- It is not clear operators have been given guidance on the conditions required before they should consider permitting themselves to deviate from a procedure on an "Emergency" basis.
- The simulator did not model:
  - V-notch indication inoperability
  - Leak detection indication (from LD-027 modification) going to white data when leakage exceeded 8 gpm
- Simulator exercises generally deal with severe accidents, not events that slowly, continually degrade incrementally.

2217

Action: Annunciator 5003.01,-1K was received. This indicated a seal cooler outlet temperature alarm on the "B" reactor recirculation pump. The temperature identified was 146°F.
Basis: With isolation of the CRD complete, this was expected.
Management Involvement: It was approximately 2200 when the Plant Manager arrived at the Main Control Room. At this time, the crew and management present were still focused on reducing leakage and were presently monitoring the affects of having closed the CRD injection valve.
Evaluation: This was another opportunity to evaluate overall plant conditions and conservative actions that might be taken. Indications are that the crew and management continued to focus on reducing the leakage.

2222-0000 (Friday, September 6, 1996)

Action: The reactor recirculation pump "B" seal pressure decreased rapidly from approximately 950 psig to approximately 280 psig within a few seconds. A drywell air cooler drain flow alarm was received. Drywell pressure started to increase and the "B" mixing compressor was started. The containment was evacuated per CPS 4001.01. At basically the same time (2223), white data was indicated on the computer point for DWRF flow rate. Drywell pressure peaked at 0.45 psig

then decreased to a pressure of 0.12 psig as steam was being condensed in the drywell. The "B" mixing compressor was secured when drywell pressure reached 0.40 psig on its downward trend. The calculated value for DWRF leakage peaked at 38.1 gpm (now considered 23 gpm but at the time the more conservative value was used), then decreased to 14 gpm. At 2255 the shift entered actions for inoperable leak detection instrumentation per Technical Specifications Section 3.4.7.

During this time period, the seal pressure decreased rapidly, leading to increased leakage. Reconstruction of the occurrences during this time frame from interviews indicates the following. After closure of the CRD isolation valve, leakage was decreasing slightly. When the seal depressurized, leakage increased rapidly, but the crew could not immediately determine the rate because the input to the operator console Display Control System (DCS) caused DCS data to become "white data" and the strip chart recorder pegged high at 7.98 gpm.

Recognizing that white data was not dependable, the SS directed the STA to begin manually calculating the leakage rate. While accumulating data for pump start and stop time, he monitored time between pump runs and determined that the leakage rate was decreasing. Based on the STA's notes and memory, he informed the SS at approximately 2250 that the leakage rate was decreasing. It was at this time that he began inputting the data to the program that calculates the actual flow rates. At about 2300, he got the results which provided him with the following trend for DWRF leakage (in gpm): 6.5, 5.6, 24.0, 38.1, 20.8, 15.7. The STA immediately informed the SS that the leakage was 15.7 gpm with a downward trend. He told the SS that leakage had peaked at about 38 gpm and emphasized that this was a very conservative figure, with the actual leakage probably being much less. The SS thought this indicated the leakage was decreasing rapidly and that the effects of the seal depressurization were proceeding as expected.

The SS and STA discussed the leakage to evaluate the potential that an Emergency Classification Level of ALERT had been reached. They used the highest DWRF leakage rate of 38.1 gpm and the normal/historical DWRE Leakage Rate of 8-10 gpm and determined that the 50 gpm total leakage action level, which requires declaration of an ALERT, had not been exceeded.

Basis: Starting a mixing compressor and evacuation of the Containment are standard procedure when drywell pressure begins to increase. With all DWRF indication inoperable, there was no other option than to perform manual calculations of DWRF leakage.
Management Involvement: The Plant Manager and ADO were still available in the general area and were monitoring the overall actions of the crew. The crew and management present saw that the leakage was getting better and that loop depressurization was proceeding as expected. This indicated to them that the evolution was under control. Also, the fact that the calculated DWRF leakage indicated a decreasing rate influenced the crew and management present to believe that leakage might be brought within Technical Specification limits.
Evaluation: Initiation of the mixing compressor and evacuation of the containment were appropriate steps to take. When the computer point for drywell floor drain flow rate became "white data", the Operators knew that the data was not dependable, but did not understand the reason it went to "white data". The Shift Supervisor immediately directed the Shift Technical Advisor (STA) to begin manual calculations for the DWRF flowrate.

Although several indicators were received that should have driven management to step back and evaluate overall reactor safety, they were not recognized or acted upon. These indicators include two DWRF leakage rate increases, two entrances into Technical Specification Action Statements, entrance into an off-normal procedure (CPS 4001.01), declaration of a NOUE, inability to complete the procedure while performing the steps in the specified order, rapid depressurization of the "B" RR pump seal, receipt of a drywell air cooler drain flow alarm, an increase in Drywell pressure necessitating the start of a mixing compressor, and having the computer point for DWRF become white data which required entrance to Technical Specification 3.4.7 for inoperable leak detection instrumentation. The focus continued to be on successful loop isolation and reduction of the leakage, rather than overall conservative operation.

Additionally, none of the management or shift crew present understood that the DWRF leakage instrumentation in use was not designed to provide indication above 8.0 gpm. This caused considerable confusion and several discussions occurred to try and resolve this unanticipated event. This occurred because an alternate system of drywell floor drain sump monitoring, Plant Modification LD-27, was installed and released for operations in September of 1995. The intended function of the LD-27 system is to provide a backup Drywell floor drain sump monitoring system to measure low flow rates. The range of the recorder which provides the indication is zero to eight gallons per minute. Engineering Change documents associated with this change and the subsequent modification were reviewed by the Training Department. However, these reviews failed to identify that the simulator recorder for this system should indicate white data if its value exceeds 8.0 gpm. The inadequate review by the Training Department is documented in CR 1-96-09-134. Therefore, training material did not cover this feature of the system. For this reason, Operator requalification training and affected CPS procedures did not cover this specific aspect (range 0-8 gpm) of the system, nor was the simulator updated to incorporate this data limitation. Hence, most operating personnel were unaware that the LD-27 DCS indication goes to white data when RF leakage exceeds 8.0 gpm.

The interviews indicated that both the SS and STA understood that the ALERT classification is based on total leakage, both identified (DWRE) and unidentified (DWRF). However, they depended on historical data for the value of DWRE leakage and did not perform any calculations to determine the actual leakage rate. Because the assumed value of DWRE plus the calculated value of DWRF was equal to about 48 gpm, the plant was close to the total leakage value which requires an ALERT. The SS and STA incorrectly assumed the DWRE leakage value and did not verify it by calculation. They did not continue to track total leakage, because they had calculations indicating that DWRF leakage was decreasing and that the peak value for DWRF of 38 gpm was very conservative and probably actually much lower. This was also incorrect because they were depending on historical DWRE leakage and there is no evidence that DWRE leakage was being monitored. The failure to track total leakage, monitor DWRE leakage and verify DWRE leakage by calculation is documented in CR-1-96-09-137.

During the evolution, the shift crew and management present thought they were in control of the event at all times. This assumption led the personnel in command of the evolution to analyze the events, as they occurred, step by step, and provide technical justification to themselves on why the

actions being taken were prudent and reasonable, rather than stepping back to consider unexpected occurrences.

It is not clear that at any time from the beginning of the evolution to isolate the "B" reactor recirculation loop until this time in the chronology, that any member of the shift crew or management, present or contacted during the event, focused on anything other than isolation of the loop and reducing the leakage. This is the **most significant issue and consideration from the overall evaluation**. At no time during the period just described did any crew member or management individual present or contacted question whether we were doing the right thing in regard to our principles of conservative decision making with reactor safety as our number one priority.

0000 (9/6/96)

Action: The offgoing shift crew held a critique to analyze the events.

Basis: The critique was held to evaluate the actions and events leading to exceeding the Technical Specification limits which required declaration of a NOUE.

Management Involvement: The Plant Manager and ADO were present at the Critique, although the Plant Manager left before the critique was completed. The offgoing SS was the critique chairman.

Evaluation: The critique was ineffective and did not perform an adequate self-assessment of the event. The critique did not identify any of the procedure violations which occurred during the event. The failure to track total leakage and confirm the value of DWRE by calculation was not identified. The SS and ADO focus on the details of the evolution and their failure to maintain a broader perspective was not identified. The attendance was limited to only those involved and no outside departments, including Licensing and Nuclear Assessment, were involved. This reduced the potential for a critical self evaluation. The critique should have been delayed if necessary to assemble the appropriate personnel.

0039

Action: The crew performed a Double Isolation of the Reactor Water Cleanup (RWCU) System from the 'B' Reactor Recirculation Loop.

Basis: The crew believed that a contributor to the RR pump "B" seal leak could be leak-by of the 1G33-F106 Recirculation Loop "B" suction valve. In an attempt to further isolate RWCU from the "B" RR loop, 1G33-F100, Recirculation Loop "A" suction and 1G33-F102 RWCU Recirculation Suction throttle valves were closed.

Management Involvement: The ADO had provided a list of management expectations for the single loop isolation evolution, which included a contingency for double isolating RWCU. He presented this to and discussed it with the Shift Supervisor prior to the pre-evolution briefing.

Evaluation: The list of management expectations was drafted by the ADO and had the concurrence of the Plant Manager. These expectations were formulated to ensure the Shift Supervisor and management had the same focus concerning the loop isolation evolution. One expectation was to consider double isolating RWCU from the RR loops to determine if 1G33-F106 could be leaking-by. This guideline was not technically reviewed to determine if current procedures supported the evolution to be performed. This presented a proposed contingency action to the shift crew that was not specifically covered by station procedures.

During the pre-evolution brief prior to beginning the loop isolation, questions raised regarding the adequacy of the procedure by crew members were not adequately addressed and resolved by shift supervision. Following the closing of 1G33-F100 and 1G33-F102, the resultant RR "B" Pump Seal leakage did not change. However, the system continued to be operated in this abnormal configuration for nearly 19 hours. This presents mixed signals to station staff relative to management's expectations concerning procedure compliance.

A review of station procedures and Standing Orders was performed. PMSO-043, "VERBAL INSTRUCTION TO CONDUCT OPERATIONS", provides direction for manipulations such as leak isolation that are not prohibited by, but are also not specifically outlined in, existing procedures. Neither a procedure change or implementation of PMSO-043 were used for this pre-planned evolution which is a violation of PMSO-043. CR 1-96-09-079 documents the extended operation of RWCU in the abnormal configuration. The failure to log the direction provided by the ADO as required by PMSO-43 is also addressed in CR 1-96-09-079.

0055-0623

Action: The four hour Technical Specification 3.4.5 Required Action A.1 completion time limit to reduce the leakage to below 5 gpm or commence a shutdown to Mode 3 expired at 0055. An orderly shutdown was commenced per 3005.01, "UNIT POWER CHANGES". This included reducing reactor power from 55% to 23% with reactor coolant flow and control rod insertion, shutdown of the B Reactor Feed Pump, transfer of the A Reactor Recirculation Pump to slow speed, shifting from 3-element feedwater control to single element, removing the Moisture Separators from service, and dealing with a reported fire in the Service Building which turned out to be light smoke only.

Basis: The unidentified leakage in the drywell of greater than 5 gallons per minute necessitated a unit shutdown per Technical Specification 3.4.5. The shutdown was performed in accordance with procedure CPS 3005.01, "UNIT POWER CHANGES".

Management Involvement: The Shift Supervisor, Assistant Director Plant Operations and Plant Manager discussed the shutdown evolution with emphasis on following procedures and performing the shutdown carefully. The Supervisor Operations Support was present during this time to provide management oversight of the shutdown. The Plant Manager left at 0117.

Evaluation: The shutdown proceeded as planned. The crew maintained focus, followed procedures, exercised proper 3 part communications, and fully implemented the STAR self checking technique. However, at 0228, reactor power was 55% and was decreased to 38% at 0310. CPS 3005.01, Step 6.1.b addressed ODCM SR 3.4.1.1 and 3.4.1.2 requirements to perform gaseous sampling when a thermal power change exceeds 15% of rated thermal power within a one hour period. Chemistry was notified of the impending power decrease at 0228, but was never informed of the >15% power change which occurred by 0310. This violation of CPS procedure 3005.01 was documented on Condition Report 1-96-09-045.

The Supervisor Operations Support filled out an OSO-086, "OPERATIONS SELF ASSESSMENT TASK CARD", identifying the fact that the numerous feedwater heater level alarms due to the low power level and the 3 part communications between the Operator and the Line Assistant Shift Supervisor during the acknowledgment and resetting of those alarms seemed

more an exercise in verbatim compliance than a good operating practice. It seemed as though compliance to the expectations of OSO-090, "EXPECTATIONS FOR CREW MEMBERS", was a distraction to the operation of the plant.

0623

Action: EOP 4402.01, "PRIMARY CONTAINMENT CONTROL", was entered due to receiving a High Suppression Pool Level alarm.
Basis: The crew appropriately entered EOP 4402.01 when the High Suppression Pool Level entry condition was received. Technical Specification 3.6.2.2, Suppression Pool Water Level Condition A requires the suppression pool level to be restored to within limits in 2 hours.
Management Involvement: The Supervisor Operations Support was the Operations Management representative present to monitor plant shutdown activities.
Evaluation: During the mid shift (2300-0700), the Shift Supervisor and crew were primarily focused on shutting down the reactor and entering Mode 3. The 12 hour Technical Specification time limit of Technical Specification 3.4.5, Condition c to be in Mode 3 by 1255 that day, was in effect. The Motor Driven Reactor Feed Pump was started at 0550 in preparation for shifting reactor vessel feed from Turbine Driven Reactor Feed Pump B to the Motor Driven Reactor Feed Pump which requires a concentrated effort by the control room staff. At 0612, a report of a fire in the Service Building was received by the control room staff. This diverted the crew's attention and required manning the fire brigade. It was later determined that no fire existed, but light smoke was present. The High Suppression Pool Level alarm occurred at 0623.

Since the April 9, 1996 reactor scram and Mode 3 operation with main steam safety relief valves, there has been increased safety relief valve leakage, necessitating more frequent use of suppression pool cooling and water transfer to radwaste in order to maintain the required suppression pool temperature and level. There is no evidence that the crew was expecting the high suppression pool level to occur.

While the crew's focus during abnormal conditions must be to deal with those situations, oversight of overall plant conditions during normal evolutions could help the crew anticipate other needed manipulations. This is an indication that, although busy with the immediate tasks at hand in the control room, Shift Supervision and the Shift Technical Advisor were distracted from their primary responsibility of oversight and review of all plant parameters, resulting in the entry into an Emergency Operating Procedure.

Additionally, when EOP 4402.01 was entered, Technical Specification 3.6.2.2 Action Statements should have been entered and documented in the Control Room Log. Condition Report 1-96-09-115 documents failure to log entrance into the LCO.

0644-0945

Action: Continued the plant shutdown per CPS 3006.01, "UNIT SHUTDOWN". This included starting and subsequently securing RHR Pump B from Suppression Pool Cooling, starting the A

Electrode Boiler, performing a shift turnover from mid shift to day shift, placing the Motor Driven Feed Pump in single element automatic and shutting down the A Reactor Feed Pump.
Basis: The plant shutdown was continued in accordance with procedure 3006.01, "UNIT SHUTDOWN".
Management Involvement: The Supervisor Operations Support was present on mid shift until shift turnover, and the Operations Task Coordinator was available in the control room area to provide management oversight on day shift.
Evaluation: There were no noted problems observed while performing these varied steps in the shutdown sequence. The crew maintained focus, followed procedures, exercised proper 3 part communications, and fully implemented the STAR self -checking technique.

0945
Action: The B Reactor Operator transferred the 4160v and 6900v non-vital busses from the Unit Auxiliary Transformers (UAT) to the Reserve Auxiliary Transformer (RAT) per 3501.01, "HIGH VOLTAGE AUXILIARY POWER SYSTEM", step 8.2.1. The B Reactor Operator performing the evolution mistakenly turned the control switch slightly past the Neutral switch position when returning the switch to Neutral from the Closed position. This was done while transferring the 6900v 1B non-vital bus from the UAT to the RAT.
Basis: The Unit Shutdown procedure 3006.01, step 8.4.4 addresses the transfer of the 4160v and 6900v non-vital busses from the UAT to the RAT per the 3501.01 procedure. This is part of the normal shutdown sequence.
Management Involvement: A representative from the Training Department was present and monitoring the control room activities.
Evaluation: This event was strictly operator error caused by lack of attention to detail by the operator. This resulted in the opening of the 6900v 1B bus Reserve Feed Breaker, momentary deenergization of the 6900v 1B bus, and the auto reclosure of the Main Feed Breaker which reenergized the bus. This human performance error was documented in Condition Report 1-96-09-020.

The impact was the B Circulating Water Pump tripped, the Reactor Water Cleanup B & C Pumps tripped, the A Electrode Boiler tripped, Stator Water Cooling Pump A auto started, the Turbine Bearing Lift Pumps auto started, and the RPS Solenoid Inverter B trouble alarm was received. In addition, the Motor Driven Reactor Feed Pump fed off 6900v 1B was running and feeding the reactor vessel, but did not trip. This would have resulted in a reactor scram.

This resulted in delaying the shutdown process by diverting the crew's resources to resetting the RPS inverter trouble alarm, restarting the B Circulating Water Pump, restarting the A Electrode Boiler which would not run (MWR D74661) (B Electrode Boiler was eventually placed into service), shutting down the A Stator Water Cooling Pump, and dealing with a Reactor Water Cleanup System restart. This complication placed added pressure on the crew to get the plant in hot shutdown in the Technical Specification imposed 12 hour time limit.

The B Reactor Operator readily admitted his error and what he believed to be the cause. The Electrical System Engineer was present and verified this on the prints. This greatly reduced the recovery time. The forthrightness of the Operator is commendable.

1005-2316

Action: Continued the plant shutdown per CPS 3006.01, "UNIT SHUTDOWN". This included completing the 4160v and 6900v non-vital bus transfers, restart of the B Circulating Water Pump, startup of the B Electrode Boiler, reducing reactor power to 17% with control rods, removing the generator from the grid and shutting down the turbine, shifting steam seals to Auxiliary Steam, performing a reactor scram as part of the normal shutdown procedure, returning the Reactor Water Cleanup System to service, performing a cooldown to Mode 4, and exiting the Notification of Unusual Event.

When transferring steam seals to Auxiliary Steam, the 1GS041 Auxiliary Steam to Gland Seal Supply Valve would not open electrically. The operators turned off the feeder breaker to the valve and manually opened it.

Basis: The shutdown was continued in accordance with CPS 3006.01, "UNIT SHUTDOWN".

Management Involvement: The Operations Task Coordinator and Assistant Director Plant Operations were available in the control room area during these evolutions.

Evaluation: There were no noted problems observed while performing these varied steps in the shutdown sequence. The crew maintained focus, followed procedures, exercised proper 3 part communications, and fully implemented the STAR self-checking technique.

The normal reactor scram per the shutdown procedure was well briefed, all rods inserted as expected, CPS 4401.01, "RPV CONTROL" was entered at the appropriate entry condition (level 3) and exited when reactor water level recovered with normal feedwater following the scram. The cooldown rate was appropriately followed.

The logical corrective action to deenergize and then manually open the 1GS041 valve by the operators showed good trouble shooting techniques that kept the shutdown progressing with minimal delays due to equipment malfunctions. MWR D61643 was written to document the 1GS041 valve problem.

This assessment did not evaluate training of operators or the impact training methods may have had on the actions of the crew. However, during the assessment, training weaknesses were identified as follows:

- The simulator training rarely, if ever, addresses routine operations which are infrequently performed.
- Simulator exercises generally address severe events, not events that slowly and continually degrade incrementally.
- It is not clear that operators are given guidance on conditions required before they should consider it acceptable to deviate from procedures based on being in an "Emergency."

It is recommended that the Nuclear Training Department evaluate the above weaknesses and take appropriate action.

Finally, during the assessment it was identified that the ADO exceeded the limits on working hours because he stayed on site during the event. This was documented in CR 1-96-09-077.

This completes the analysis portion of this report. The six major issues that are identified as a result of this evaluation are discussed in the following Conclusion section of this report.

This section has been revised to incorporate additional information and evaluation from further investigation performed by the Nuclear Assessment Department. Revisions are marked by revision bars.

II. Conclusions

Based upon the evaluations of the specific actions and decisions that occurred during this event, the following conclusions can be drawn regarding the event as a whole.

Management personnel were not conservative in the operation of the plant. Numerous opportunities existed for management to evaluate uncertainties and the need for continued reactor operation, but these were not recognized or acted upon. (CR 1-96-09-071)

- When leakage initially exceeded 5 gpm, and a NOUE was entered, the crew, in accordance with Technical Specifications, focused on isolating the leak within the 4 hour action statement and did not give adequate consideration to initiating a reactor shutdown.
- Subsequent events complicated the situation at which time the crew could have re-evaluated the need for reactor shutdown:
  - Leakage increased to greater than a 2 gpm increase in 24 hours.
  - Operating crew entered the Emergency section of the RR loop isolation procedure.
  - Shutting the CRD injection valve (C11-F026B) to the RR pump seal.
  - Seal Failure.
  - Drywell pressure increased requiring mixing compressor startup.
  - Continued leakage greater than Technical Specification limit following seal failure.
  - Loss of indication for unidentified leakage.

- The Shift Supervisor, Assistant Director of Operations, Shift Technical Advisor and the Plant Manager did not perform the expected oversight role in considering the overall picture of plant conditions.

- Ineffective application of previous lessons learned from industry experience.

Management did not properly establish, enforce, or set the proper example for procedure compliance. (CR 1-96-09-072)

- Several procedure violations occurred:
  - Isolation of CRD injection valve to the RR pump seal.
  - Performance of procedural steps outside the prescribed sequence.
  - Performance of double isolation of RT and subsequently operating the system in a manner not specifically covered by the procedure.
  - Failure to notify RP to assist in determining leakage location.
  - Failure to take gaseous sample for a reactor power change >15% in 1 hour.
- Management provided an action plan for the evolution that had not received an independent technical review and contained a contingency step not specifically covered by the procedures.
- Critique held after the event did not identify the procedural violations.

Oversight of the overall picture of plant conditions and actions surrounding the event was ineffective. (CR 1-96-09-073)

- Shift management did not maintain proper oversight and was too involved with details regarding leak isolation.
- Senior management did not maintain proper oversight and was too focused on leak isolation.
- Management decisions regarding actions to operate the plant were determined not to be in accordance with procedures.
- During the decision process, shift management did not consistently involve all the licensed members of the operating crew and thus did not have an opportunity to hear dissenting opinions.
- No one called a "time-out" but instead focus remained on leakage isolation, consequently more conservative options (such as reactor shutdown) were not evaluated.
- Several procedure violations occurred.

There was inadequate planning and evaluation of potential consequences prior to performing an infrequently performed operation. (CR 1-97-09-074)

- Inadequate Action Plan (letter from Mosley to Shift Supervisor entitled, "Guideline for RR "B" Loop Isolation and RF Leakage Evaluation").
- Insufficient evaluation of potential for valve leak-by:
  - RT potential leak-by on isolation. (Procedure was thought to address double isolation, but did not).
  - Potential leak-by on RR isolation valves.
  - MWR's existed which indicated leak by was potentially present on both trains (D63773, dated 4/8/95 and D76000, dated 3/8/96).
- No review of what procedural steps to follow if leakage increased during the evolution.
- No upper leakage limit set for beginning an orderly shutdown.
- It is not clear that any previous evolutions to enter single loop operation, planned or emergency, were evaluated prior to this evolution.

Management tolerated long-standing equipment problems that contributed to uncertainty of some plant conditions. (CR 1-96-09-075)

- The following long-term deficiencies were associated with this event:
  - LD system V-notch indication routinely inoperable.
  - Prolonged operation with a degraded reactor recirculation pump seal.
  - Leak-by on RR isolation valves.
  - Dual position indication on RR isolation valve.

- Clinton has made decisions to operate with degraded equipment that did not place sufficient emphasis on operational needs.

- Long-standing equipment deficiencies, such as drywell leakage system, reactor recirculation pump seals, and reactor recirculation pump suction and discharge valves, further complicated decisions to mitigate the event.

IP personnel were not timely in identifying a potential procedure non-compliance and management did not sufficiently pursue other indications that would lead to management recognition of the significance of non-conservative reactor operation. It should be noted that IP was prompted by the NRC to pursue the issues of procedural guidance. (CR 1-96-09-076)

- Self assessment was ineffective/inadequate:
  - Operations did not recognize the procedural non-compliances at the critique immediately following the event.
  - Procedural non-compliances were not fully recognized by the station until the Wednesday following the event.
  - Management personnel did not respond critically and aggressively to this event.

ATTACHMENT A

SUMMARIZED CHRONOLOGY

Date: September 05, 1996

Prior to Time Line of 1730-1800

Prior to the evolution, the Assistant Director of Operations, acting as the Director of Plant Operations, met with the Shift Supervisor and provided a written summary of the management expectations (see Attachment 1 to this Chronology) to the Shift Supervisor. These expectations were identified as a tool and were not intended to conflict or override CPS Procedures or Technical Specifications. The expectations contain the following:

1. As reactor power is decreased to <70%, monitor drywell floor drain (RF) leakage for changes.

2. As the flow control valves are manipulated, monitor drywell RF leakage for changes.

3. When the "B" reactor recirculation (RR) pump is shutdown, be prepared for seal leakage to rapidly increase and expedite loop isolation, if required.

4. Once the "B" RR pump is shutdown and the loop isolated, the following may be expected to happen:

   * Drywell RF leakage remains relatively unchanged. Then consider isolating the Reactor Water Cleanup System (RT) from the RR loops to determine if valve 1G33-F106 is leaking by its seat and adding to the leakage into the RR loop. If this does not cause the drywell RF leakage to decrease then CPS Management will decide what the next course of action will be.

   * Drywell RF leakage decreases. CPS Management will decide if continued operation in single loop is appropriate or not.

   * Drywell RF leakage increases. If the leakage exceeds 4.5 gpm, then continue to shutdown the plant and go to Mode 4. If the leakage is less than 4.5 gpm, then CPS Management will decide the next course of action.

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Elapsed time: 0 hr.

Date: 9/5/96

<u>Time: 1730-1805</u>

Plant Status:

- Rx Power: 100%
- Seal Pressure P1:
- Seal Pressure P2: 479 psig (time 1753)
- Seal Temp:
- Drywell RF Leakage manual calculation sump fill time: 4.27 gpm (time 1744)
- Drywell RF Leakage manual calculation pump run time:4.64 gpm (time 1744)

A special briefing was held from 1730 to 1800 with the following personnel in attendance:

- STA.
- Electricians who would take valve signatures to verify proper and complete closure of the valves.
- C&I Techs who would be responsible for completing the appropriate surveillances within the required time limits once we achieved single loop operation.
- The system engineer.
- "VOTES" engineer who would direct the electricians.
- A nuclear engineer (separate from the STA) who was there to discuss control rod manipulations.
- The Assistant Director Plant Operations (also filling in for the Director Plant Operations during his absence while on vacation).
- Shift Supervisor.
- Line Assistant Shift Supervisor.
- The RO's taking part in the evolution.
- Some outside area watches.

There were various things discussed and assignments made, including:

1) One RO was assigned and discussed the normal shutdown procedures and process.

2) One RO was assigned and provided a briefing on the procedures and process to be used should a catastrophic failure of the RR pump seal be experienced.

3) The RR system engineer stated that the most likely point in the evolution where catastrophic failure might occur would be when the flow control valve is closed to minimum position immediately prior to

shutting the RR pump down. (The system engineer cautioned the operating crew to not delay at this point.)

4) Also discussed was that during this evolution, an increase in the unindentified leakage could put the plant in an emergency plan action level. The specific leakage criteria was not discussed.

5) Procedures discussed during the briefing included the following:
3302.01 Reactor Recirculation
3005.01 Unit Power Changes
4001.01 Reactor Coolant Leakage
4008.01 Abnormal Reactor Coolant Flow
4100.01 Reactor Scram

6) During the procedure discussion, the briefers did refer to specific steps and actions to be taken during the evolution. Specific questions were asked and answers were provided with the assistance of the system engineer.

7) The first contingency discussed was that if catastrophic failure occurred,they would expect to see certain indications. These indications included:
- Greater amount of leakage
- Increase in drywell pressure
- Decrease in RR pump seal pressure
- Increase in Drywell RF leakage

8) The second contingency discussed was that right at the end of the day shift operators had noted a blip on a GETARS point for the feedwater system. The contingency plan was if they have a problem with the feedwater pump/pump control circuit they would take the pump off-line and operate with the one remaining pump. Personnel at the briefing indicated that this action made sense.

9) The sequence of shutting valves and movement of monitoring equipment was discussed with the electricians. Provisions were made to instrument the 1B33-F067B RR Pump "B" discharge valve with VOTES equipment in order to verify positive valve closure. This precaution was taken since there was an existing MWR (MWR D63044) related to a dual position indication problem on this valve. Since this equipment was available, it was decided to also use it on the 1B33-F023B RR Pump "B" suction valve.

Elapsed time: 35 min.

Time: 1805-2009

Plant Status:

- Rx Power: 100%

Procedures Entered/In Use: 3005.01 (Unit power Changes)

Reduced reactor power in preparation for shutting down the "B" Reactor Recirculation Pump. Power was reduced from 100% to 69%.

Elapsed time: 2 hrs. 39 min.

Time: 2009

Plant Status:

- Rx Power: 69%

Procedures Entered/In Use: 3302.01 (Reactor Recirculation)

- Seal Pressure P1: 985 psig
- Seal Pressure P2: 479 psig (time 1753)
- Seal Temperature Cavity 2: 109 °F
- Drywell RF Leakage manual calculation sump fill time: 4.27 gpm (time 1744)
- Drywell RF Leakage manual calculation pump run time: 4.64 gpm (time 1744)

Shutdown the "B" RR pump in accordance with the normal RR operating procedure, 3302.01, Step 8.2.1. The crew tripped the pump, closed the discharge valve 1B33-F067B. The procedure had an instruction to reopen 1B33-F067B in 5 minutes if it was not planned to isolate the loop. Final power level was 58%. At this time there were electricians at the valve breaker who had the valve motor instrumented to verify proper closure. Indication for proper closure included: a) verify signature traces on the valve current to verify it seated properly; b) verify the time for stroking the valve to ensure full travel (approximately 120 seconds); c) verify closure indication on control room panel. The valve was instrumented because there was some uncertainty that the main control room indicating lights would indicate full closure due to the existance of MWR D63044 as noted earlier.

A contingency plan was established to restroke the valve 1B33-F067B if intermediate position indication was received on the control room panel. Instrumentation at the valve locally would again be used to establish a trace on the motor current and to determine the stroke time to verify proper and complete closure. If the restroking did not clear the intermediate position indication, they would consider the valve closed provided the tracer data indicated proper closure.

This contingency, to have the electricians set up to take tracer data, had been set up on day shift in advance.

The seal P1 and P2 pressures commenced to equalize. Fully opened the RR Flow Control Valve (FCV) 1B33-F060B per 3302.01, step 8.2.1.5. The pressure across the pump seals had equalized shortly after pump shutdown, approximately 7 minutes later. The operating crew and the System Engineer were surprised when the pressure equalized. The System Engineer talked to several people about a theory that "the lower seal was not doing much such that the leakage past the lower seal was greater than what was going out the upper seal. Hence, P1 $\cong$ P2. The upper seal was carrying full pressure while the lower seal was doing nothing." Seal temperature was approximately 130°F when the pump was shut down.

Elapsed time: 2 hrs. 39 min.

<u>Time: 2009-2030</u>

Plant Status:

- Rx Power: 58%

Procedures Entered/In Use:
3302.01 (Reactor Recirculation)

- Seal Pressure P1: 985 psig (time 2013)
- Seal Pressure P2: 976 psig (time 2017)
- Seal Temperature Cavity 2: 109 °F (time 2017)
- Drywell RF Floor Leakage manual calculation sump fill time: 4.47 gpm (time 2030)
- Drywell RF Floor Leakage manual calculation pump run time:5.167 gpm (time 2030)

Commenced Loop Isolation per 3302.01, step 8.2.4.

Step 8.2.4.1	Shut 1G33-F106 RT suction from RR Loop "B"
Step 8.2.4.2	1B33-F067B Discharge valve was already shut from section on pump shutdown, Step 8.2.1
Step 8.2.4.3	Not currently in emergency condition, so did not perform this step. (i.e., did not qualify for the "IF" statement so didn't perform the "THEN" statement.)
Step 8.2.4.4	At this point, operators began to evaluate the conditions. The step does not say to wait, it just says that certain conditions must be met prior to performing the isolation.

Elapsed time: 3 hrs.

<u>Time: 2030</u>

Discussed closing 1B33-F075B (RR Pump B Seal Staging Shutoff Valve) with the system engineer and Assistant Director Plant Operations to increase the flow to the RR loop from the control rod drive (CRD) system. (This diverts flow from the seal). The RR loop temperature was around 530 °F and had only decreased about 10 °F. The STA or one of the RO's performed a quick calculation and estimated that the time it would take to reach cool down would be about 6 to 8 hours. We then had a discussion on whether we would damage anything by shutting the 1B33-F075B. The engineer indicated it would not harm the seal or the system because the valve originally closed automatically whenever the pump was shut off. (A modification to provide manual vent and drain capability had since been added and the valve no longer closed automatically on pump shutdown.)

Shut 1B33-F075B Pump B Seal Staging Shutoff Valve per 3302.01 step 8.2.4.5 in an attempt to increase the cooldown rate of the recirculation loop. At this time, operators were monitoring parameters to get the loop temperature down to less than 250°F to allow completion of steps 8.2.4.6 and 8.2.4.7, in order to complete the loop isolation.

Floor drain flow rate was approximately 4.7 gpm. RR "B" loop suction temperature was 511° F, seal cavity 2 temperature increased to ≈ 150 °F. Such an increase in seal temperature was expected since there is no forced flow through the seal cavity HX with the pump off.

Elapsed time: 3 hrs. 25 min.

<u>Time: 2055</u>

Plant Status:

• Rx Power: 58%	Procedures Entered/In Use:
• Seal Pressure P1: 985 psig (time 2013)	3302.01
• Seal Pressure P2: 976 psig (time 2017)	4001.01
• Seal Temperature Cavity 2: 109 °F (time 2017)	4008.01

- Drywell RF Leakage manual calculation sump fill time: 5.9 gpm (time 2053)
- Drywell RF Leakage manual calculation pump run time:5.75 gpm (time 2053)
- LD-27: 5.51 gpm

Floor drain flow rate increased to 5.51 gpm. Action 'A' of Technical Specification (TS) 3.4.5 was entered which required restoring leakage to less than 5 gpm within four hours. TS 3.4.5 requires that if leakage is not restored with in the 4 hour time period, the plant must be shutdown within 12 hours. 4001.01 Reactor Coolant Leakage Off-Normal procedure was entered at this time. 4008.01 Abnormal Reactor Coolant Flow was entered, conditions evaluated, and then exited.

During this time frame, the SS stayed in the MCR to provide oversight, monitor leakage, monitor seal pressure, and monitor the actions of the crew.

The SS was looking for a change in LD readings. After the sump pump cycled, the next reading was about the same as the first. Once he was satisfied that leakage was not increasing at too high of a rate (he wanted to be there if conditions deteriorated rapidly and he was comfortable with crew actions), he went to the Emergency Plan.

Elapsed time: 3 hrs. 30 min.

<u>Time: 2100</u>

The Plant Manager was informed by the Assistant Director - Plant Operations that they were conducting the loop isolation, that the pump was stopped; the discharge valve 1B33F067B was shut while waiting for the loop to coldown to 250°F, the leakage had crept up to over 5 gpm, and they were in Action "A" of TS 3.4.5.

Notification Of Unusual Event was declared per EC-02, step 4.1 Leak Rate Exceeded/Loss of Inventory.

SS went to get the Emergency Plan bag, checklist and immediately returned to the horseshoe. He began concentrating on getting notifications made in the correct amount of time.

He called security to have security to set off the pagers for the Unusual Event. The SS then asked the STA if he was comfortable with making the appropriate notifications. He responded YES and took the assignment to make the notifications.

NOTE: After the event, looking back, the STA says that if he had known then what he knows now, he would have said NO. The SS looking back agrees that he should have probably gotten an extra operator to make the notifications.

Elapsed time: 3 hrs. 46 min.

<u>Time: 2116</u>

Plant Status:

- Drywell RF Leakage manual calculation - sump fill time (6.3 gpm)

Burped the drywell using the "A" Hydrogen Mixing Compressor.

Elapsed time: 3 hrs. 55 min.

Time: 2125

Plant Status:

State IEMA and IDNS were notified of Unusual Event declaration. (Emergency Plan requires 15 minutes for State notifications, 1 hour for NRC notifications).

Elapsed time: 3 hrs. 57 min.

Time: 2127

Plant Status:

Secured the "A" Hydrogen Mixing Compressor. Drywell pressure: 0.23 psig.

Elapsed time: 4 hrs.

Time: 2130

Plant Status:

- RX power 58%
- Seal Presure P2: 982 psig.

Thoughts and considerations by the SS at this time included the following:

- Had proper indication that loop had been successfully isolated.
- The SS considered entering the Emergency step (and hence perform 3302.01 step 8.2.4.3). Inputs into this consideration involved:
  - are we putting control rod drive pressure into the seal cavity (1500 psig to 1800 psig).
  - if staging line is open, we are removing flow at 1 gpm, if closed, no flow is being removed.

- so, we have 3-5 gpm control rod drive flow going to the isolated loop which will pressurize the isolated loop lifting the injection relief valve at approximately 1250 psig.
- At this time, indications were: that the loop was isolated, that control rod drive was still injecting and that the seal staging flow valve was shut.
- The SS expected to see the following:

- If the seal leak rate was greater than CRD injection rate, we would expect to see the seal pressure start to slowly decrease.

OR

- If another drywell leak was present, which was masked earlier, and in conjunction with the seal not passing 5 gpm, would expect to see the seal start to pressurize. (If this occurred, it would mean leakage was coming from somewhere else).

This would require reopening the suction valve so pressure could be relieved to the reactor.

During this time, the SS talked to the Assistant Director Plant Operations, explained that we were at a greater than 5 gpm leakage, and told the Assistant Director Plant Operations that he felt this constituted an emergency situation and that we needed to return to step 8.2.4.3 of Procedure 3302.01, because we now met the "IF" statement.

He also judged that this qualified as an emergency because we had entered the Emergency Plan.

Additionally, he felt this qualified as an emergency because the shift had entered the Reactor Coolant Leakage procedure (CPS 4001.01), which says to isolate the leakage in step 4.6.

The SS also thought he had the means to isolate the leakage. The Assistant Director Plant Operations agreed.

° One RO was surprised that we went back to the step to close the suction valve (step 8.2.4.3)
- The SS explained the logic to the RO and after explaining the logic, the RO indicated that closing the suction made sense.
- The SS does not remember any other challenges to the procedures being used or the steps being taken or the timing of these items at any time during the evolution.

Operators proceeded to isolate the "B" RR loop by performing step 8.2.4.3a of 3302.01, shut 1B33-F023B Pump Suction Valve. This step was now performed since the "Unusual Event" was considered an "emergency

condition" as referenced in this step. 1B33-F075B in 8.2.4.3.b had been previously shut per step 8.2.4.5.

NOTE: Nothing in the procedure provides guidance on what to do if you get halfway through the evolution and the conditions change, which is the situation we were now in.

As noted earlier, the electricians had set up to monitor the loop suction valve (1B33-F023B). When the valve was closed to complete the isolation, all the correct indications were received.

- The valve timing was right
- It had a good trace
- The Control Room indication indicated full close.

Seal pressure was observed to stay about the same and decrease just slightly. The SS watched seal pressure for about 29 minutes and indicated leakage started to go up.

Elapsed time: 4 hrs.

Time: 2130-2159

Plant Status:

Main Control Room Personnel were monitoring floor drain leakage rate and seal pressures for change. During this period of time (2144), reactor coolant leakage exceeded a 2 gpm increase in 24 hours. LD readings indicated approximately 6 gpm. Readings taken 24 hours prior to this reading were 3.82 gpm. Seal pressures during this time period were steady at 980 psig on P1 and P2. Technical Specification 3.4.5 was re-entered for the 2 gpm leakage increase.

Elapsed time: 4 hrs. 14 min.

Time: 2144

Plant Status:

- RX power 57% (2133)
- Seal Pressure P2: 979 psig. (2137)
- Drywell RF Leakage - manual calculation - sump fill time: 6.43 gpm.

NRC notification was performed for the Unusual Event declaration.

Elapsed time: 4 hrs. 29 min.

Time: 2159

Plant Status:

- RX power 57% 985 psig. (2133)
- Seal Pressure P1 & P2 979 psig. (2137)
- Seal Temperature cavity 2: 141°F (2109)
- Drywell RF Leakage - manual calculation - sump fill time: 6.43 gpm.

SS thoughts and considerations during this time frame:

- The loop is isolated and not communicating with the reactor.
- Leakage is about the rate of CRD injection flow (3-5gpm CRD injection flow with an overall indication of leakage at 6gpm).
- Logic indicated that we are currently pumping CRD water into the drywell.
- Held a discussion with the Assistant Director Plant Ops and the system engineer and discussed the following:
- Pressure is going down real slowly.
- Pressure started about 980psig when we shut the suction valve and isolated the loop.
- At this time, loop pressures were down slightly.
- The seal leak and the CRD injection were about equal at this point. This indicated we could have continued all night in this situation and not had any additional success at isolating the leakage.
- The primary focus of the overall evaluation and the SS's objective was still to try to isolate the leakage.
- The SS wanted to deviate from the normal procedural requirement to reduce loop temperature to less than 250°F or wait until the loop is depressurized to about drywell pressure in order to comply with the Reactor Coolant Leakage procedure 4001.01 to locate and isolate the leak.

Reasoning included:

- We couldn't depressurize because of CRD injection.
- We appeared to be experiencing a very slow cooldown and would have to wait 6-10 hours to shut off injection flow in order to meet the parameters specified in the procedure for RR loop temperature to be less than 250°F.
- The CRD injection appeared to be keeping the leakage high.
- Subsequently, when the SS was asked by the Assessor whether there was any discussion that taking 6 hours to 8 hours to shut the CRD injection would not meet the 4 hour criteria to get below 5 gpm. His response was that this was not a consideration. There was no discussion at this time

related to not meeting the 4 hour criteria. The primary focus was that he had leakage that he was causing and he wanted to stop it.

- As far as the SS knew at the time, no one had indicated in any way that they were concerned that we were trying to beat the clock or in any way prevent a shutdown.

The procedure also contains a Caution identifying that seal damage will occur at 250°F. The basis for the caution was discussed. The System Engineer believed that the Caution in the procedure was then mainly to protect the O-rings, because in the April event, he knew that the O-rings were not the limiting factor. The O-rings were tested at 600°F for 8 hrs. at Bingham. The SS, RR System Engineer, and the Assistant Director Plant Operations discussed the Caution in the procedure about closing 1C11-F026B, this was discussed carefully however with some urgency. They decided that catastrophic seal failure was not likely and that further degradation of an already damaged seal was acceptable. (In retrospect, the System Engineer noted, perhaps 1C11-F026B should have been <u>shut slowly.</u>) The crew was hoping for a gradual relief of the pressure, because the upper seal was expected to hold. Going to mode 4 was also discussed as the next alternative.

° The System Engineer said that we would probably have some degradation on the seal faces from thermal expansion when the CRD flow was stopped but he didn't expect a catastrophic failure.

- Even if it did fail catastrophically, there was only a finite volume of water to escape because the loop was isolated.
- Did not have signature trace on the reactor water cleanup system (RT) isolation valve. Had discussed 1G33-F106 valve isolation because there was a potential RT water could leak back into the RR loop. However, the leakage in gpm was approximately equal to the CRD injection flow in gpm and with pressure coming down slowly, he thought the isolation was holding.

° At this time we analyzed the 2 cautions contained in CPS 3302.01, Step 8.2.4.

- This first caution identifies that there is potential to damage the seal if the seal temperature is allowed to rise to 250°F. (The seal temperatures were low at this time, and we expected that they would increase).
- The second caution indicated that we would increase airborne activity in the drywell. We thought we were pumping in pressure (and leakage) and had to stop inserting flow and pressure to let it depressurize and stop the leakage. This would allow the reactor coolant pressure locked in the loop to depressurize out through the seal, increasing airborne activity in the drywell.
- We also considered that if depressurization occurred fast enough, we might get depressurization before the seal temperature got high enough to cause any damage.
- This was the only item that the SS did not feel personally comfortable with. He expected to contaminate the drywell and he discussed this with the Assistant Director Plant Operations.
- Neither the SS nor the Assistant Director - Plant Operations thought that this would be significant or should be weighed heavily as a constraint. (The SS did not ask for an opinion from RP and to his knowledge neither did the Assistant Director Plant Operations.)
- The SS did not think it was necessary to discuss with RP because stopping the leakage was more important than possibly contaminating the drywell.

- The SS discussed this with the Assistant Director Plant Operations and then made the decision to cut off the CRD flow to the seal. This was the only known item performed that was outside of the normal procedural guidance. This was outside the procedural guidance because at the time this occurred the conditions required to allow shut off of CRD flow to the seal, were not met. However, this did meet the guidance of the Off-Normal to locate and isolate the leakage.
- The SS evaluated that he could have waited 8 hours for the cool down of the loop to occur, but he saw no reason to wait.
- He knew we had a damaged seal and would be replacing the seal, at least by RF-6 and was not worried about additional minor damage to the seal.
- From discussion with the System Engineer, the key reason procedure step 8.2.4.4 was there was to prevent seal damage. Since we were going to replace the seal anyway, this was not a critical issue.

SS decided, with the concurrence of the Assistant Director Plant Operations, to shut the seal injection (1C11-F026B) per 3302.01 step 8.2.4.6. The SS gave direction to LASS to shut 1C11-F026B. At this point, the SS went back to monitoring the conditions. He expected seal pressure to go down and
leakage rate to decrease following pressure. This appeared to occur.

Elapsed time: 4 hrs. 36 min.

Time: 2206

The Assistant Director Plant Operations briefed the Plant Manager on plant conditions which were: loop isolated, CRD injection isolated; leak rate about 5.5 gpm, and seal pressures slowly falling. The Assistant Director - Plant Operations indicated loop isolations (valve closures) looked good. The Plant Manager agreed with him that it appeared the loop was isolated (based on falling seal pressures) and suggested the leak rate may take awhile to fall since it (the loop) would essentially depressurize its inventory out the seal. The Plant Manager concurred in the approach.

Elapsed time: 4 hrs. 37 min.

Time: 2207

Plant Status:

- RX power 57%
- Seal Pressure P1: 985 psig. & P2: 967 psig. (2205)
- Seal Temperature Cavity 2: 138° F
- Drywell RF Leakage manual calculation - sump fill time: 6.5 gpm

Time: 2207

Security reported that all the ERO notifications were complete for the Unusual Event.

Elapsed time: 4 hrs. 47 min.

Time: 2217

Received a seal cooler outlet temperature alarm on the "B" reactor recirculation pump. 146°F per 5003.01-1K. This was expected with CRD seal water secured.

Elapsed time: 4 hrs. 52 min.

Time: 2222

Plant Status:

- Received a seal cooler outlet alarm on our recorder (annunciator on panel "High Temp - RR pump B" which sends you to the recorder). (We were expecting to get alarms on the temperature, so this was not unexpected.)
- Doesn't remember temperature (alarms at 146°F per 5003.01-1K).
- The SS had an extra RO assisting and had him go to check the alarm. As he came back to report the temperature, sudden depressurization occurred.

When rapid depressurization occurred, the LASS directed the crew to start the hydrogen mixing compressors, which pumps drywell atmosphere into containment suppression pool.

The evacuation of containment was also directed.

At this time, seal pressure had dropped from about 900 psig to approx 280 psig.

The SS again started monitoring.

° Expected the drywell pressure would go up (was at approximately 0.23 after having burped the drywell).

° Also got a drywell air cooler drain flow alarm coming in. (One air cooler off each drywell chiller has condensate metered before going to the floor drain.) This indicates a steam leak from steam getting condensed in the cooler and going to the floor drain. This also adds to the total drywell RF leakage which was the parameter of concern.

° The drywell air cooler rate peaked at approximately 3.5 gpm.

° During this time (over about a 15 minute time period) the SS monitored the plant conditions.

° Once the SS was comfortable that drywell pressure would not be a problem he resumed evaluating the floor drain rates.

Elapsed time: 4 hrs. 53 min.

<u>Time: 2223</u>

Looking at the chart recorder, the Plant Manager and Assistant Director Plant Ops saw 7.98 gpm two times in a row following the seal failure. Believing two exact numbers to two decimal points to be bad data, informed the SS. The SS seen the Display Control System (DCS) displaying white data indicating bad input data and had the STA performing manual calculations. The LD system engineer was then called.

Elapsed time: 4 hrs. 56 min.

<u>Time: 2226</u>

Plant Status:

- RX power 57%
- Seal Pressure P1: 318 psig. & P2: 308 psig.
- Seal Temperature Cavity 2: 174° F (2221)
- Drywell RF Leakage manual calculations - sump fill time: 21 gpm

- LD-27 > pegged high 8 gpm.

Stopped B Mixing Compressor at a drywell pressure of .40 psig. Peak drywell pressure was .45 psig, total run time of 4 minutes. Drywell pressure continued to decrease to a pressure of .12 psig as steam was being condensed in the drywell. No drywell vacuum breaker opened.

This indicated to the crew that leakage was getting better and the seal depressurization was proceeding as expected.

Elapsed time: 5 hrs. 7 min.

<u>Time: 2237</u>

Plant Status:

- RX power 57%
- Seal Pressure P1: 185 psig. & P2: 187 psig.
- Seal Temperature Cavity 2 137° F (2239)
- Drywell RF Leakage manual calculation - sump fill time: 23.5 gpm
- Drywell RF Leakage manual calculation - pump run time 38.1 gpm
- LD-27 > pegged 8 gpm.

- The SS was informed by the STA that leakage was at approximately 38 gpm and a short time later the STA informed him it was going down.

- The SS expected that the leakage rate would initially increase and then start coming down.

- The Assistant Director Plant Operations said that he thought LD27 instrumentation was clamped at 8 gpm.

- About the same time, the LD system engineer called and said that LD27 did, in fact, clamp at 8 gpm.

- Simultaneously, the oncoming STA also informed the crew that LD27 clamps at 8 gpm.

- At this time the thoughts and considerations of the SS considered the following:

- The SS still felt good because everything was occurring as expected.
- His first expectation that pressure would go down was occurring.
- The second expectation was still being monitored by the SS in that he was expecting the leakage to start going down also.

° When the SS received the second set of floor drain flow data, everything continued as expected and the leakage did start going down.

° The following things were also occurring in this time frame:
- The drywell cooler drain flow alarm had cleared. (This alarm clears at approximately 1.6 to 2 gpm.)
- As noted, the floor drywell RF leakage started going down.
- Drywell pressure started coming down. (The drywell pressure had peaked at 0.45psig and the mixing compressors had been stopped at 0.40psig.)
- Drywell pressure continued to go down and reached a value of 0.12 psig.
- Because the volume contained in the isolated loop had depressurized through the seal, there was a potential that a vacuum would be pulled in the drywell. However, the drywell vacuum breakers did not activate, indicating that no vacuum was experienced.

  ° At this point the SS believed that he had accomplished all objectives and again went into the monitoring mode.

  ° When relieved, drywell RF leakage was at approximately 12 gpm, down from the 38 gpm peak.

Elapsed time: 5 hrs. 25 min.

<u>Time: 2255</u>

Plant Status:

- RX power 57%
- Seal Pressure P1: 121 psig. & P2: 120 psig. (2257)
- Seal Temperature Cavity 2: 142° F (2245)
- Drywell RF Leakage manual calculation - sump fill time: 15.07 gpm (2250)
- Drywell RF Leakage manual calculation - sump run time: 15.73 gpm (2258)

Recognized that drywell floor drain flow rate indications were inoperable. Entered actions for leak detection instrumentation per Technical Specifications.

Elapsed time: 6 hrs.

<u>Time: 2330</u>

Plant Status:

- Seal Pressure P1: 121 psig (2257) & P2: 99 psig. (2389)
- Seal Temperature Cavity 2: 150° F (2333)
- Drywell RF Leakage manual calculation - sump fill time: 11.08 gpm (2384)
- Drywell RF Leakage manual calculation - sump run time: 14.06 gpm (2329)

Calculated value for drywell floor drain flow rate was 14 gpm based on sump pump run time.

The crew felt this flow rate decrease was also an indication that the finite volume of coolant in the B recirculation loop was depressurizing and that the leakage would continue to decrease.

Note: From about 2237 until turnover at about 2322, there had been 5 dual manual calcs taken and leakage rate had decreased from a peak value of 38 gpm to between 12 and 14 gpm. This indicated that leakage was still decreasing and under control. At this time, there was about 1 hour and 20 minutes left on the 4 hour Technical Specification Action statement for isolating the leakage to within limits.

Shift Turnover

At this point, the SS began discussions with his relief including the current plant status and how they got there. As part of the turnover, the SS indicated that they were looking for the leakage to continue to go down and hopefully go below the 5 gpm from the Technical Specification limit and the 4.5 gpm administrative limit.

The SS indicated that at the turnover there were no specific concerns identified from the oncoming SS. The oncoming SS just wanted to know what was the plan from here. The options provided were the following:

- The first, was to see if the leakage stablized at a value below the administrative limit. If yes, then maintain single loop operation and the decision would be made the next day on what to do next. It was somewhat anticipated that if leakage was successfully reduced, we would probably maintain the plant in single loop operation until RF-6.

- The second option, was if the leakage did not stabilize below the Technical Specification limit of 5 gpm, the shift could perform an orderly shutdown of the plant.

  When questioned by the interviewer, the SS responded that there was no discussion at any time about delaying the shutdown or shutting down slowly to allow leakage to become lower.

Elapsed time: 6 hrs. 44 min.

Date: 9/6/96

Time: 0000

From R. Morgenstern Written Summary:

Approximately 0000 -- After loop suction from RT double isolated, and trend didn't change in 2 pump down cycles, I informed LASS of this fact as I passed through MCR; then Gary Mosley/Gary Setser/Dan Andrew and I met in SS office. We concluded that Drywell RF leakage would not end up being less than 5 gpm, and that S/D was to be begun. I advised Setser to ensure Nuclear Engineer had good power reduction plan (we were in a xenon transient (minor), single loop, low power) and that the crew had reviewed applicable procedures -- then proceed to Mode 3. We discussed timing, I said I didn't think we had to hurry; plant was stable, need to proceed carefully. Dan Andrew was to contact Outage Management (Bob Gruenewald) and see if they had any specific needs with regards to time at a given plant condition. We all expected Bob to want us cool as early as possible, but we'd go at a comfortable pace. Also, I mentioned to Gary Setser that if he got to Mode 3 on his shift, to ensure we were ready for DW entry, since we wanted to know the condition of Drywell RF leakage/etc. while still hot.

Time: 0014

Calculated value for drywell floor drain flow rate steady at 10.5 gpm. Drywell RF leakage manual calculation.

Elapsed time: 7 hrs. 9 min.

Time: 0039

During the turnover process at 2330, the oncoming Shift Supervisor was advised that if leakage did not get below the administrative limit of 4.5 gpm within an hour or so, that further isolation of the RWCU system from the B recirculation loop would be possible. Waiting the additional hour would give the drywell floor drain flow rate a chance to stabilize and still provide time to close the additional RWCU isolation

valves before the four hour Technical Specification time clock elapsed. CPS 3301.01, REACTOR WATER CLEANUP, was reviewed for limitations with both recirculation loop suctions isolated as would be the case with further isolation. Limitation 6.8.1 for maintaining bottom head drain flow between 63 and 200 gpm was reviewed and discussed. The crew attempted to further isolate the B RR loop from RWCU due to potential leak-by on 1G33-F106 (closed at 2009) by closing 1G33-F100, Recirc Loop A Suction and 1G33-F102, RWCU Recirc Suction Throttle. Bottom head drain flow was ~189 gpm. No change in drywell floor drain flow rate was noted. It was decided that leak-by must be occurring from the recirculation loop suction and/or discharge valves and that a reactor shutdown would be appropriate.

Elapsed time: 7 hrs. 25 min.

<u>Time: 0055</u>

Four hour Technical Specification Action 3.4.5.a elapsed for leakage greater than 5 gpm. Entered Action statement 3.4.5.c to be in Mode 3 within 12 hours (1255) and Mode 4 within 36 hours (1255 on 9/7/96).

Elapsed time: 7 hrs. 30 min.

<u>Time: 0228</u>

Completed pre-evolution brief on Reactor shutdown and commenced reducing reactor power from 85%.

Elapsed time: 9 hrs. 40 min.

<u>Time: 0310</u>

Stopped the reactor power decrease at 38%.

Elapsed time: 9 hrs. 50 min.

<u>Time: 0320</u>

Shutdown Turbine Driver Reactor Feed Pump 1B.

Elapsed time: 11 hrs. 29 min.

<u>Time: 0459</u>

Downshifted Reactor Recirculation Pump A to slow speed. Reactor power at 27%.

Elapsed time: 11 hrs. 46 min.

<u>Time: 0516</u>

Commenced rod insertion to decrease reactor power from 27%.

Elapsed time: 12 hrs.

<u>Time: 0530</u>

Completed rod insertion, reactor power 23%.

Elapsed time: 12 hrs. 20 min.

<u>Time: 0550</u>

Moisture Seperator Reheaters removed from service. Started up the Motor Driven Reactor Feed Pump. Shifted the A Turbine Driven Feed Pump from 3 element automatic to single element automatic level control.

Elapsed time: 12 hrs. 42 min.

<u>Time: 0612</u>

Report of a fire in the Service Building basement computer room.

Elapsed time: 12 hrs. 45 min.

Time: 0615

Operator at fire scene determined no fire existed. There was light smoke. Six fire brigade members dressed out.

Elapsed time: 12 hrs. 53 min.

Time: 0623

Primary Containment Control, EOP 4402.01 entered due to Suppression Pool level greater than 19 ft. 5 in. Level increase was apparently from safety relief valve leakage, which had increased since the April 9, 1996 reactor scram.

Elapsed time: 13 hrs. 14 min.

Time: 0644

RHR B started in Suppression Pool Cooling to lower suppression pool level.

Elapsed time: 13 hrs. 23 min.

Time: 0653

Exited Primary Containment Control, EOP 4402.01. Suppression pool level less than 19 ft. 5 in.

During this period, an attempt was made to place the MDRFP on the startup level controller unsuccessfully. A decision was made to allow the on-coming shift to determine if there was a problem with these controllers, and then perform the transfer. Engineering support was obtained, but troubleshooting revealed no anomolies.

Elapsed time: 13 hrs. 48 min.

Time: 0718

Electrode Boiler 'A' supplying auxiliary steam. Main steam to auxiliary steam isolated.

Elapsed time: 14 hrs.

Time: 0730

Shift Turnover

Elapsed time: 15 hrs. 27 min.

Time: 0857

Placed the Motor Driven Feed Pump in single element automatic.

Elapsed time: 15 hrs. 34 min.

Time: 0904

'A' Turbine Driven Feed Pump in manual on low speed stop.

Elapsed time: 15 hrs. 50 min.

Time: 0920

Commenced rod insertion for shutdown from 23% reactor power.

Elapsed time: 16 hrs. 15 min.

Time: 0945

While attempting to transfer 6900v bus 1B from the UAT to the RAT, the reserve feed breaker tripped momentarily, due to operator error, deenergizing the bus. The main feed breaker automatically reclosed reenergizing 6900v bus 1B. All turbine-generator lift pumps and the 'A' GC Pump automatically started. Circulating water pump B, RWCU Pumps B and C, and the Auxiliary Boiler tripped. RPS Solenoid Inverter B trouble annunciator was also received. Entered 4200.01, LOSS OF AC OFF-NORMAL.

Elapsed time: 16 hrs. 35 min.

Time: 1005

Completed transferring all AC buses to the RAT.

Elapsed time: 16 hrs. 36 min.

Time: 1006

Restarted Circulating Water Pump B.

Elapsed time: 16 hrs. 45 min.

Time: 1015

'A' Electrode Boiler will not restart. 'B' Electrode Boiler being started.

Elapsed time: 17 hrs. 12 min.

Time: 1042

Performed pre-evolution brief on tripping the main turbine.

Elapsed time: 17 hrs. 24 min.

Time: 1054

'B' Electrode Boiler in service.

Elapsed time: 17 hrs. 30 min.

Time: 1100

Reactor power is 17%.

Elapsed time: 17 hrs. 36 min.

Time: 1106

Generator separated from the grid.

Elapsed time: 17 hrs. 38 min.

Time: 1108

Tripped the main turbine.

Elapsed time: 17 hrs. 55 min.

Time: 1125

While attempting to shift the Main Turbine steam seals to auxiliary steam, 1GS041 would not open remotely from the main control room. (MWR D61643)

Elapsed time: 18 hrs. 5 min.

Time: 1135

Turned breaker off and manually opened 1GS041.

Elapsed time: 18 hrs. 10 min.

Time: 1140

Main Turbine steam seals are on auxiliary steam.

Elapsed time: 18 hrs. 24 min.

Time: 1154

Conducted pre-evolution brief on reactor scram.

Elapsed time: 18 hrs. 29 min.

Time: 1159

Reset main turbine to minimize cooldown following the scram.

Elapsed time: 18 hrs. 36 min.

Time: 1206

Scrammed the reactor from 17% power as part of the normal shutdown per 3006.01, UNIT SHUTDOWN. Entered Mode 3.

Elapsed time: 18 hrs. 37 min.

Time: 1207

Entered EOP-1 RPV CONTROL due to a level 3 following the scram. Level recovered above level 3, exited RPV CONTROL.

Elapsed time: 19 hrs.

Time: 1230

Unisolated RWCU to commence heatup of system to return to service.

Elapsed time: 19 hrs. 25 min.

Time:1255

12 hours to Mode 3 clock elapsed for Drywell floor drain leakage rate greater than 5 gpm.

Started RWCU Pump C.

Elapsed time: 22 hrs.

Time: 1530

Shift Turnover

Elapsed time: 23 hrs. 25 min.

Time: 1655

Isolated RCIC. Reactor pressure less than 150 psig.

Elapsed time: 25 hrs. 29 min.

Time: 1859

Mode 3 Checklist completed.

Elapsed time: 25 hrs. 50 min.

Time: 1920

Opened 1G33-F100, Recirc Loop A Suction and 1G33-F102, RWCU Recirc Suction Throttle.

Elapsed time: 27 hrs. 34 min.

Time: 2104

Mode 4 Checklist completed.

Elapsed time: 27 hrs. 48 min.

Time: 2118

Started RHR Pump in Shutdown Cooling.

Elapsed time: 28 hrs. 20 min.

Time: 2150

Notification of Unusual Event has been terminated. Drywell floor drain flow rate less than 5 gpm.

Elapsed time: 29 hrs. 46 min.

<u>Time: 2316</u>

Entered Mode 4

GUIDELINE FOR RR B LOOP ISOLATION AND RF LEAKAGE EVALUATION

This is to be used as a tool for decisions by Operations Management and is not intended to conflict or override CPS procedures or Tech Specs.

1. As Rx power is decreased to < 70% monitor DW RF leakage for changes.

2. As the flow control valves are manipulated monitor DW RF leakage for changes.

3. When the B RR pump is shutdown be prepared for seal leakage to rapidly increase and expedite loop isolation if required.

4. Once the B RR pump is shutdown and the loop isolated the following may be expected to happen:

    * DW RF leakage remains relatively unchanged. Then consider isolating RT from the RR loops to determine if 1G33-F106 is leaking by its seat and adding to the leakage into the RR loop. If this does not cause the DW RF leakage to decrease then CPS Management will decide what the next course of action will be.

    * DW RF leakage decreases. CPS Management will decide if continued operation in single loop is appropriate or not.

    * DW RF leakage increases. If the leakage exceeds 4.5 gpm then continue to shutdown the plant and go to Mode 4. If the leakage is less than 4.5 gpm then CPS Management will decide the next course of action.

Gary J. Mosley
Assistant Director-Plant Operations

This section has been revised to incorporate additional information and evaluation from further investigation performed by the Nuclear Assessment Department. Revisions are not marked by revision bars.

Attachment B
List of Condition Reports

The following Condition Reports were generated as a result of this assessment:

*CR-1-96-09-045, Missed sampling requirement for 15% power change in 1 hour (Page 15 of Evaluation Section)
CR-1-96-09-071, Isolation of the "B" RR Loop (1) (Page 1 of Conclusions Section)
*CR-1-96-09-072, Isolation of the "B" RR Loop (2) (Page 1 of Conclusions Section)
*CR-1-96-09-073, Isolation of the "B" RR Loop (3) (Page 2 of Conclusions Section)
CR-1-96-09-074, Isolation of the "B" RR Loop (4) (Page 2 of Conclusions Section)
CR-1-96-09-075, Isolation of the "B" RR Loop (5) (Page 3 of Conclusions Section)
CR-1-96-09-076, Isolation of the "B" RR Loop (6) (Page 3 of Conclusions Section)
*CR-1-96-09-077, Exceeding working hours limits (Page 19 of Evaluation Section)
*CR-1-96-09-078, Failure to follow procedure CPS 3302.01 (Page 4, 6 and 8 of Evaluation Section)
*CR-1-96-09-079, Procedure violation (Page 2 and 15 of Evaluation Section)
*CR-1-96-09-114, Failure to notify RP - Procedure violation (Page 5 of Evaluation Section)
CR-1-96-09-115, Failure to enter LCO when entered Suppression Pool level EOP procedure (Page 16 of Evaluation Section)
CR-1-96-09-134, Inadequate Review for impact of LD-27 Mod (Page 13 of Evaluation Section)
CR-1-96-09-137, Failure to track total Drywell leakage (Page 13 of Evaluation Section)

***Involves a procedure violation.**

Attachment C
Personnel Contacted

The following personnel were contacted during the course of the assessment:

P. D. Yocum
J. Earl
R. Morgenstern
K. Cameron
G. Mosley
R. Phares
J. Hays
W. Connell
J. Naden
B. Corley
R. Rippy
P. T. Young
J. Kaineg
K. Sheffield
J. Cunningham
R. Brixey
R. McCubbin
J. Wells
D. Andrew
J. Schottel
M. Friehofer
D. Reeser
E. Rau
D. Hodel
M. Brotherton
F. Perryman
T. Olsen
M. Baetz
T. Doerscher
J. Smith
G. Setser
R. Giuliani
J. Spencer
D. Korneman
J. Wemlinger