Enclosure 1

Requalification Examination Report No.: 50-397/OL-91-01

Facility Licensee: Washington Public Power Supply System
3000 George Washington Way
Mail Drop 1023
Richland, Washington 99352

Facility Docket No.: 50-397

Facility License No.: NPF-21

Requalification examinations were administered from February 25 through March 8, 1991 at WNP-2, near Richland, Washington. An Operational Evaluation was conducted on March 21 and 22, 1991.

Chief Examiner: Todd B. Sundsmo — 5-13-91 Date Signed

Approved by: J. F. Miller, Jr., Chief, Operations Section — 5-13-91 Date Signed

Summary

A. Written examinations and operating tests were administered to Senior Reactor Operators (SROs) and Reactor Operators (ROs) in accordance with Revision 6 of NUREG-1021, "Operator Licensing Examiner Standards" (Examiner Standards).

Generic weaknesses were identified in both the operating crews, and in the facility evaluators. Three operator performance concerns were identified that had safety significance: (1) the operators lacked facility with and did not consistently adhere to Emergency Operating Procedure (EOP) requirements; (2) crew teamwork was weak and caused inaccurate EOP implementation, crew uncertainties, and reduced coordination; (3) crews appeared to respond in a rehearsed fashion to scenarios they were anticipating, rather than to the actual symptoms provided by the scenario being run.

Evaluator weaknesses also focused in three main areas: (1) evaluators were sometimes not properly positioned to objectively evaluate operator decisions and performance; (2) evaluators generally did not pursue operator errors with objective post-scenario questioning; (3) evaluator administrative control of the examination process was weak, which raised performance concerns and could have impacted security.

B. Three of the four crews evaluated failed the dynamic simulator portion of the requalification examination; five of seventeen operators failed individually. Two of seventeen evaluated operators failed the written examination. Enclosure 2 identifies these crews and operators.

Overall, by the criteria established in the "Operator Licensing Examiner Standards" NUREG-1021, Revision 6, the licensed operator requalification program was evaluated as unsatisfactory. The program failed to meet all three criteria of NUREG-1021, each of which is required for the program to be satisfactory:

- two-thirds of the crews must pass the simulator examination (the passing rate was 25%),
- at least 75% of the evaluated operators must individually pass the examination (the passing rate was 59%),
- no more than a 10% non-conservative disagreement may exist between NRC and facility grading (the disagreement was 19%).

An Operational Evaluation was then conducted on March 21 and 22, 1991 to assess the abilities of the remaining operating crews to safely operate the facility. During the Operational Evaluation, one of the four crews evaluated and three individuals from two crews failed to demonstrate an adequate performance level.

C. During the examination preparation period, the Chief Examiner identified five EOP deviations from the EOP technical bases guidelines. These deviations are:

1. not allowing operators to bypass reactor trip logic to reset the reactor scram signal during ATWS conditions,

2. exiting EOPs whenever entry conditions have been mitigated, even when this was inappropriate for plant conditions,

3. not providing EOP guidance to control reactor water inventory (level) when steam cooling of the fuel is required,

4. not warning the operator that reactor water level indication may be erroneous when terminating steam cooling of the fuel, and

5. not allowing use of alternate injection systems to maintain reactor water level before the fuel is uncovered.

These EOP contradictions of the EOP technical bases are identified in Enclosure 5, and are considered unresolved items pending further inspection. The Requalification Examination and Operational Evaluation avoided any evaluation of these areas.

REPORT DETAILS

1. Personnel

Operator Licensing Examiners:

T. Sundsmo (Chief Examiner), NRC (1), (2)
L. Vick, NRC (2)
M. Morgan, PNL (1), (2)
J. Muth, PNL (2)
G. Buckley, PNL (1)

(1) Participated in Requalification Evaluation
(2) Participated in Operational Evaluation

NRC Staff:

R. Zimmerman, Director, Division of Reactor Safety and Projects
K. Perkins, Deputy Director, Division of Reactor Safety and Projects
R. Gallo, Chief, Operator Licensing Branch, NRR
L. Miller, Chief, Operations Section
C. Sorenson, Senior Resident Inspector, WNP-2
T. Meadows, Operator Licensing Examiner

Licensee Staff:

L. Oxsen, Deputy Managing Director
J. Baker, Plant Manager
S. McKay, Operations Manager
W. Shaeffer, Assistant Operations Manager
D. Kobus, Training Department Manager
B. Barmettlor, Nuclear License Training Manager
N. Hancock, Operations Liaison
G. Setser, Training Specialist
T. Messersmith, Operations Engineer
L. Monroe, Training Specialist
G. Richmond, Training Specialist
J. Perry, Training Specialist
G. Fisher, Lead Requalification Training Specialist
S. Bruce, Training Specialist
S. Hutchison, Training Specialist
S. Veitenheimer, Training Specialist

2. Examination Preparation Week, February 4 - 8, 1991

A. Prior to review of the facility proposed examination material, the structure of exam administration and scheduling was resolved jointly with the Training Department. The facility had originally submitted 12 dynamic simulator scenarios, 40 Job Performance Measures (JPMs), and two complete written examinations (each consisting of two Category A exams and one Category B exam).

After reviewing several different scheduling possibilities, the facility agreed to use "back-to-back" scheduling in which the scenarios or JPMs used in the morning would be used again in the afternoon for different

operators. Examination security would be maintained through positive control over the morning group until their exam was completed and the afternoon group was assembled and controlled. To reduce operator stress, the afternoon group would not be asked to come to work until the morning group's exam was scheduled to be completed; the morning group would then be released to go home.

This schedule reduced the number of scenarios and JPMs requiring review by 50%, and allowed the examination team to focus greater attention on the selected material.

All facility personnel who had specialized knowledge of the examinations signed security agreements as required by Examiner Standards. Positive control was maintained over all examinations and related materials to ensure examination security throughout the preparation and examination periods.

B. The written examinations consisted of 100% multiple choice format questions that were selected from the facility's examination bank, using the facility's sample plan. A major portion of the written exam review effort focused on two areas: revising negative format questions (10% of the original examinations) into a positive format, and ensuring that each question had only one correct answer. The lack of a proper qualifier (e.g., "What is the MINIMUM value that will cause.....") to ensure that only one answer was correct was a common error made in several questions.

The extent of use of negative format questions was discussed with the Licensed Operator Training Manager. The Chief Examiner identified that the "Examiners' Handbook for Developing Operator Licensing Written Examinations," NUREG/BR-0122, Revision 5 (pages 4-8 and 4-9), states that when possible, negatively stated question stems should be avoided. It further states that this type of question is often confusing, and emphasizes negative learning. The facility's position was that using 10% negative format questions in the examinations was not a concern, and fully met the intent of NUREG/BR-0122; the facility based its position upon the use of the qualifier "should" in NUREG/BR-0122. The Chief Examiner concluded that the facility's interpretation was not consistent with NUREG/BR-0122, and that this was a program weakness.

All other items concerning technical accuracy and general question format were readily resolved. The facility had made an extensive effort to revise the examination bank into multiple choice format, and had developed a sample plan following the guidance of NUREG-1021, "Examiner Standards."

C. Twenty JPMs were reviewed using the ES-603-1, "Job Performance Measure Quality Checklist." Much of the review effort focused on three areas:

- ensuring JPM critical steps were observable, and met the criteria for JPM critical steps,
- ensuring the JPMs were in agreement with facility procedures, and

- adding some type of performance anomaly to approximately 15% of the JPMs to help verify that the operator was using the applicable procedure in a questioning, rational manner rather than in a rote, unquestioning way.

The types of JPM changes considered for the latter area were, typically, automatic or remotely initiated actions that would fail to occur when the operator actuated or verified system response. For example, if an operator needed to open (or verify open) a valve from the control room in order to line up a system, either simulator indication or evaluator cue would indicate that, "the red light stayed on" (i.e., the valve did not open). An acceptable operator response to this cue would be to either inform the Shift Manager, or initiate action to investigate the situation.

The facility representatives were resistant to making any modifications to the proposed JPMs. It appeared that the facility considered that it was not appropriate to test operators using any examination material that the operators had not been previously trained and tested on, and use the results of those tests to make pass/fail decisions.

Members of the Examination Team who had participated in the 1990 requalification examination noticed that substantial improvements had been made to the JPMs with regard to accuracy and detail.

D. Six dynamic simulator scenarios, and one backup scenario, were reviewed using ES-604-1, "Simulator Scenario Review Checklist." Review efforts focused mostly on four areas:

- Removing information from the shift turn-over briefing that could have allowed the operators to anticipate scenario events,
- Adding minor malfunctions to ensure that critical tasks met the requirement for safety significance,
- Adding malfunctions that would require use of the EOP contingency procedures (the NUREG-1021 guidance), and
- Revising the Individual Simulator Critical Tasks (ISCTs) to meet the standards of Revision 6 to NUREG-1021, "Examiner Standards."

The overall effect of these changes on the scenario level of difficulty was minor. Each scenario, typically, had one or two additional technical malfunctions added. These malfunctions included failure of automatic actuation portions of system logic (e.g., a pump failed to automatically start), or a component that failed to respond to manual initiation (e.g., a valve fails closed).

The NRC identified to the facility that its simulator scenario ISCTs had not been revised from the Revision 5 to the Revision 6 guidance of NUREG-1021. Initial NRC review of these scenarios did not identify the outdated ISCT methodology because the format used had the appearance of the newer standards. That is, a broad topic (ISCT), followed by smaller topics (evaluation criteria). Discussions with the Training Department staff identified that their intent was to grade both the larger and smaller topics as ISCTs using the "one may, two must" failure criterion (i.e., Revision 5 guidance). This methodology and further ISCT review confirmed that NUREG-1021, Revision 6 standards had not yet been implemented by the Training Department. The implementation date promulgated for Revision 6 was October 1, 1990.

The facility initially had not properly implemented the Revision 6 ISCT methodology and was reluctant to revise their ISCTs. After the NRC team began to draft separate ISCTs (which met the Revision 6 guidelines) to evaluate simulator performance, the facility then adopted the ISCTs proposed by the NRC.

The proposed scenarios were frequently predictable due to clues and crews built into them.

The following are examples of when shift turnover information, or other events unique to the proposed scenario, provided an indication of the scenario events:

- Scenario E-1 shift turnover discussed severe thunderstorms and high winds. The scenario involved a turbine trip and loss of power due to a lightening strike.

- Scenario E-8 shift turnover identified a small steam leak on a main turbine, and an MSIV surveillance that was in progress. The scenario involved a steam line rupture at the turbine throttle valve, and failure of the MSIVs to isolate.

- Scenario E-9 shift turnover identified an ATWS ARI surveillance that was to be performed, and had one of the TSW pumps tagged out of service. The scenario involved a loss of TSW, followed by an ATWS (failure to scram).

- Scenario E-16 included an event that had the operators identify a ground on an isolation valve for the RCIC system, which prevented the valve from closing. The scenario involved a non-isolable steam rupture in the RCIC system.

In general, the facility representatives were very resistant to make any changes to the proposed scenarios. Even cosmetic changes to reduce predictability were objected to by the facility team.

3. <u>Written Examination Results</u>

A. Two operators (1 SRO and 1 RO) failed the written examinations that were administered at WNP-2 on February 26 (for Crew B and Staff Crew), and March 4, 1991 (for Crews A and B). Both the NRC and facility agreed that these two operators failed the written examination by a narrow margin. Enclosure 2 identifies the operators that failed this portion of the requalification examination.

B. The NRC/facility grading methodologies differed on the grading of questions that had two correct answers. After grading the examination, the facility identified and documented one question (question #11.14) that had two correct answers. The facility had no formal policy on how to handle this type of question, and chose to delete it from the examination. This is the NUREG-1021, Revision 5 practice (ES-403); the Revision 6 practice is to retain the question and accept either correct answer. The facility was informed that the NRC would follow the latter practice.

Had the facility retained the question, accepting either correct answer, an additional operator would have failed the written examination as graded by the facility. As it was, this operator was passed by the facility. The NRC initially graded this examination as a borderline failure. Further review of this examination identified an additional question (#8.02) that had two correct answers. Accepting either correct answer resulted in changing this score to passing.

C. A review of the graded written examination identified that performance on the static simulator portion of the examination was poorer than on the limits and controls section. The number of incorrectly answered questions on the Category A (static simulator) portion of the written examination was five times greater than on the Category B (limits and controls) portion. This large difference is unusual, and indicates a program weakness, to the extent that many operators were not as well prepared for the Category A exam.

4. Operating Examination Results - Job Performance Measures (JPMs)

A. The JPMs were conducted using back-to-back scheduling on February 28, and March 1, 6, and 7, 1991. The JPMs were split up so that the simulator JPMs (4) were conducted on one day, and in-plant JPMs (6, two of which were done in the control room) were done on the next day. There was no examination material overlap between the two examination weeks.

The NRC passed all of the operators on the JPM portion of the operating examination; the facility evaluators failed one operator. The grading difference occurred because one JPM question was evaluated differently by the facility and the NRC. The NRC considered that the answer provided by the operator was technically correct, even though it was worded differently than the answer key. The facility disagreed.

5. Requalification Dynamic Simulator Examinations

A. The dynamic simulator examinations were conducted using back-to-back scheduling on February 27, 28, and on March 5, 6, 1991. The scenarios were split up so that two crews were examined on two scenarios the first day, and the third scenario was done on the next day. There was no examination material overlap between the two examination weeks.

B. The NRC failed three of four crews and five of seventeen individual operators on the dynamic simulator portion of the examination. The crews each failed at least one Individual Simulator Critical Task (ISCT), and one competency. Failure of either of these required a crew failure. Enclosure 2 identifies the specific crews and individuals that failed; Enclosure 3 provides a synopsis of the significant events leading to each crew failure.

The facility failed two of the three crews failed by the NRC, and failed four of seventeen individual operators. The facility passed one crew and three individual operators that the NRC failed. Thus, the NRC and facility grading differed non-conservatively by 4 out of 21 pass/fail decisions (17 individuals plus 4 crew). For this portion of the examination, only two non-conservative grading disagreements were allowed by NUREG-1021 for a satisfactory program. The facility later reversed its pass/fail decision for two ROs; however, this decision does not effect the grading agreement evaluation (see Paragraph 5.C, below).

C. The facility was reluctant to fail Crew "A" based on ISCT performance, and delayed giving their final crew grading for two days because of their objection to the safety significance validity of an ISCT. The ISCT questioned by the facility evaluators was: "Direct actions to control reactivity and shut down the reactor during ATWS conditions" (Scenario ES-9). The specific portion of this ISCT which was not performed was tripping an operating reactor recirculation pump. The facility was informed that NUREG-1021, Revision 6, provided an example that closely paralleled this ISCT in its discussion of safety significance, and that the NRC considered this a valid ISCT. However, the facility did not provide preliminary grading, on this ISCT, to the NRC until it was clear to the facility evaluators that the NRC considered this to be a valid ISCT, and that the NRC was proceeding, accordingly, with the evaluation (see Enclosure 3 for additional detail).

On April 11, 1991, the facility reversed its pass/fail grading evaluation for each of the two ROs in Crew "A" regarding their performance on Scenario ES-9. NRC management has acknowledged this grading reversal and permitted the operator (who had only been failed by the facility) to resume licensed duties. This grading reversal does not effect the requalification program evaluation criterion regarding non-conservative grading disagreements between the NRC and the facility. This evaluation is based on the final grading results provided by the facility at the Exit Meeting held on March 8, 1991. The facility stated its rationale for this grading reversal in its "Licensed Operator Requalification Training Program - Root Cause Assessment And Corrective Actions," dated April 15, 1991; and in the letter, "Request for Reinstatement of Operator License," dated April 11, 1991. These operators are identified in Enclosure 2.

D. The facility passed Crew "B" on ISCTs. The NRC failed this crew because an ISCT assigned to the Control Room Supervisor (CRS) was not satisfactorily completed.

The ISCT was, "Direct actions to mitigate the effects of a non-isolable steam leak in secondary containment" (scenario ES-16). An evaluation criteria for the CRS had been pre-identified to, "Direct action to locate/isolate the steam leak." An additional evaluation criteria for a Reactor Operator (RO) of initiating action to locally isolate the valve had been pre-identified, and agreed upon by the facility. The NRC team concluded, based on a post-scenario review, that this latter evaluation criteria was most appropriate for the CRS. Since the crew did not initiate local actions to isolate the leak, the CRS' ISCT was evaluated as an individual and a crew failure. See Enclosure 3 for additional detail.

E. Additional disagreements between NRC and facility grading existed in evaluating crew competencies (ES-604, Attachment 2). The facility passed Crews "A" and "B" on competencies that the NRC determined were additional causes for crew failure. Enclosure 3 provides details of the significant events leading to each competency failure.

Both crews failed the competency of "Compliance/Use of Procedures and Technical Specifications." The crews failed to correctly implement procedures to the point that important procedural steps were not enacted correctly, which led to impeded plant recovery and unnecessary plant degradation. Both crews also had difficulty recognizing EOP entry (and re-entry) conditions.

There appeared to be two reasons for the significant grading difference between the facility and the NRC: overall reluctance by the facility to fail an operating crew, and weaknesses in facility evaluator performance. The facility's reluctance to fail an operating crew was displayed when evaluating both Crews "A" and "B". For Crew "A", the facility delayed its grading decision two days, until (as perceived by the NRC) it appeared clear to them that the NRC was going to fail the crew. For Crew "B", the facility passed the crew, even though post-scenario questioning by the facility specifically identified the error that resulted in an ISCT failure by the NRC. Paragraph 6.B below describes the facility evaluators' performance.

F. During the second week, one crew required an additional scenario to properly evaluate an operator in the CRS position. This individual's actions were overshadowed by the Shift Manager (SM) during the scenario that the operator had been assigned to be the CRS. This observation was not made by the facility evaluators and is discussed further in Paragraph 6.B, below.

6. <u>Requalification Program Evaluation</u>

A. The licensed operator training program was evaluated as unsatisfactory. The program failed to meet three NUREG-1021 criteria, each of which, individually, would have required the program to be deemed unsatisfactory:

- More than one-third of the crews failed the simulator examination. The actual failure rate was 75%.

- More than 25% of the evaluated operators failed individually. The actual failure rate was 41%.

- More than a 10% non-conservative disagreement existed between NRC and facility grading of the dynamic scenarios. The actual non-conservative disagreement was 19% (based on 17 individual operator, plus 4 crew, pass/fail decisions).

Enclosure 2 identifies specific NRC and facility grading results.

B. The NRC and facility grading significantly disagreed. Pass/fail decision disagreements included: competencies for two crews; ISCT grading for one of these crews; and non-conservative ISCT grading for three individuals. NRC and facility assignment of ISCTs also disagreed by 15%. Enclosure 2 provides grading results and shows where disagreements occurred.

The facility evaluator's performance during the dynamic simulator portion of this examination significantly contributed to NRC/facility grading disagreements. The facility's pass/fail grading reversal for each of two ROs in Crew "A" (see Paragraph 5.C) exemplifies several of the evaluator weaknesses. In general, the facility evaluators had weaknesses in these areas:

- One evaluator was assigned to observe both the CRS and an RO. Because the CRS usually stayed at the EOP flowcharts, the evaluator's attention was often diverted to the RO, who frequently performed actions at panels out of conversational hearing range to the CRS. The evaluator was often not properly positioned to have heard many CRS conversations and directions to the other RO and the SM, nor to have observed procedural errors made by the CRS.

- The evaluator assigned to observe the SM remained distant (out of conversational hearing range) during many critical periods of dynamic, post-scram EOP implementation. This evaluator was not properly positioned to have heard many SM conversations and directions to the CRS. He also was not properly positioned to have observed the procedural errors made by the CRS and SM.

- Two of the three facility evaluators did not ask sufficient post-scenario questions to clarify incorrect actions taken during the scenarios. When questions were asked, the evaluators tended to ask leading questions similar to, 'Did you recognize that level reached -XXX inches?' Questions focusing on operator performance are more effective evaluation tools.

- In general, the evaluators did not always have complete administrative control of the examination. Examples of incomplete control included three security concerns (see Enclosure 7); operators referencing procedures after scenarios were stopped, but before post-scenario questioning; forgetting to rotate an operator to ensure proper ISCT coverage, as requested by the NRC; and not observing the need to conduct an additional scenario for Crew "B" to ensure one CRS could be properly evauated.

7. Operational Evaluation

A. The Operational Evaluation was conducted on March 21 and 22, 1991 because the NRC's confidence in the ability of the remaining operating crews to safely operate the facility had been reduced by the high failure rate observed in the dynamic simulator exam. An Operational Evaluation was administered to the remaining three operating crews, and a fourth crew that the facility reconstituted from operators that had individually passed the requalification exam, to assess the ability of the operations staff to safely operate the plant. The primary goal of the Operational Evaluation was to ensure that the weaknesses seen in the crews evaluated during the requalification dynamic simulator examination were not present in the remaining operational crews.

The Operational Evaluation assessed each crew on two dynamic simulator scenarios. In general, the crews were evaluated as a crew; however, any serious individual weaknesses were addressed as well. Each crew was permitted to operate in its normal configuration as it would operate in the Control Room. There were no required operator position rotations within the crew.

B. The scenarios for this evaluation were developed during the week of March 11, 1991, using the facility simulator. These scenarios were later validated by the NRC evaluation team with NRC management and facility participation. Efforts were made to ensure that the scenarios were not overly complex, not excessively time compressed, and were within the scope of the WNP-2 Emergency Operating Procedures.

C. One of the four crews evaluated failed the evaluation. Three operators from two crews required remediation, and were not allowed to perform licensed duties until remediated. Two additional operators required remediation, but were allowed to return to licensed duties. Enclosure 2 identifies this crew and operators, Enclosure 3 provides the details of this crew failure.

D. During this evaluation, the NRC identified a significant amount of reference material maintained in the simulator that was either not in use in the Control Room, or not a controlled copy of reference material. This material was either removed from the simulator, or verified to be essentially the same material (i.e., revision number) as in the Control Room. This is a similar finding to an observation made during the previous year's evaluation that the Training Department was not consistently using the current revision of procedures. Enclosure 4 lists the documents found in the simulator that were not being maintained in a controlled manner.

8. Conclusions

During the Requalification and Operational Evaluations, three generic operator performance concerns were identified:

A. Operators did not consistently adhere to EOP requirements during simulated emergency conditions. Most of the deviations from EOP requirements were conscious decisions that were incorrectly made in order to meet other operational concerns that should have been superseded by the EOP requirements. In general, the operators did not identify to the Shift Manager (or anyone else) that they were deviating from the EOPs, and appeared to treat these deviations informally. Examples of this included:

- operating recirculation pumps during ATWS conditions,

- operating drywell sprays below 1.68 psig drywell pressure,

- a deliberate decision not to use High Pressure Core Spray (HPCS) when directed by EOPs, and

- general reluctance to use HPCS as required by the EOPs for level control.

The Chief Examiner also noted during the preparation week that members of both the operations and training department staff had incorrectly interpreted EOP requirements for use of HPCS. When presented a scenario with reactor water level stabilized below the normal band that could only be raised by using HPCS, the staff members remarked that, the use of HPCS to control reactor water level in the normal band (+13 to +54 inches) is not required. However, in fact, it is required by the "RPV Water Level" section of PPM 5.1.1, "RPV Control".

B. Weak teamwork during simulated emergency conditions caused inaccurate EOP implementation, crew uncertainties, and reduced coordination. Examples of this included:

- Generally, the CRS did not verbalize the EOP flowchart steps; decisions were made silently, then specific directions were given to the crew. This often led to bypassing required actions and caused the crew members to provide only the information solicited by the CRS. Incorrect decisions or assumptions by the CRS often were not challenged by the crew.

- Related to the above weakness, the CRS did not communicate plant recovery strategies to his team, such as routinely announcing which EOPs were entered, or the status of the plant. As a result, the team members frequently waited to respond to specific requests for particular information or actions.

- Generally, the reactor operators did not consistently verify immediate actions following a reactor trip.

C. On several occasions, operators appeared to respond in a rehearsed fashion to exam bank scenarios, rather than to the actual symptoms provided by the scenario being run. To the extent that this did occur, operator response to unfamiliar scenarios was potentially impacted because of incorrect parallels drawn to the known scenarios. It was also noted that:

- The facility considered it inappropriate and unfair to test operators using any material which they had not been specifically trained on.

- Proposed scenarios were frequently predictable due to clues and cues built into them (see discussion, Section 2 of this Enclosure).

9. Review of Emergency Operating Procedures

The NRC made several observations regarding the Emergency Operating Procedures during the exam preparation week, and during the examination. In general, these observations are not related to the findings identified during the EOP inspection conducted in September, 1990. Enclosure 5 lists the NRC's observations regarding the EOPs, and presents the facility's verbal position on these issues. The observations identified in Enclosure 5 are Open Items and are being evaluated by the NRC.

One of the particularly significant findings in Enclosure 5 is summarized below. It is similar to a finding identified during last year's examination that is described in Notice of Violation 90-09-01. This violation was concerned with an inadequate procedure to perform the EOP actions derived from PPM 5.0.8, "WNP-2 Emergency Procedure Guidelines" step RC/Q-6.2 (alternate methods tc shut down the reactor during ATWS conditions). The present concern describes the lack of a procedure necessary to accomplish the EOP actions derived from the same step of PPM 5.0.8. This step requires that Reactor Protection System (RPS) logic trips (e.g., high drywell pressure) be overridden, if necessary, to reset the reactor scram signal. This step reads, "Reset the scram, defeating RPS logic trips if necessary,....."

However, the Chief Examiner identified that PPM 5.1.2, "Failure to Scram", did not permit operators to override RPS logic trips during ATWS conditions in order to reset the reactor scram signal; this is contrary to the guidance in PPM 5.0.8 (the EPGs). This EOP step reads,

"Can the scram be reset? Yes- Reset the scram.
No- Attempt to start both CRD pumps."

When the operator reaches this step in the flowchart, he is required to bypass steps that reset the scram, if RPS scram logic was tripped, because the tripped scram logic physically prevents resetting the scram. In this case, EOP direction is needed to defeat the RPS logic trips because the conditions that result from an ATWS usually generate a scram signal, and thus prevent resetting the scram. Resetting the scram is necessary to allow the scram accumulators to be refilled and Scram Discharge Volume to be drained so that another scram can be initiated to insert the control rods.

Resetting the scram signal and scramming again is an effective method to insert control rods during ATWS conditions; the EOPs currently do not allow this action under many ATWS conditions (i.e., when a scram signal would be present). This, and the other findings in Enclosure 5, are considered unresolved items pending further inspection.

10. Exit Meetings

The examiners met with representatives of the plant staff on March 8 and 22, 1991. The preliminary findings finalized by this report were discussed. In accordance with the licesee's Requalification Program, the facility removed from licensed duties all operators that had failed any portion of the Requalification Examination or Operational Evaluation from either NRC or facility grading; and will not return these operators to

shift without written authorization from the NRC. Enclosure 2 identifies these operators.

In addition, with respect to the need for EOP direction to bypass the RPS logic trips described in item 9 above, the facility stated that a procedure facilitating bypass of Reactor Protection System logic trips during ATWS conditions would be implemented by April 24, 1991.

ENCLOSURE 3

REQUALIFICATION AND OPERATIONAL EVALUATIONS
SIMULATOR EXAMINATION SUMMARIES

Acronyms used in this attachment:

Grading:

S	=	Satisfactory
U	=	Unsatisfactory
N/O	=	Not Observed (usually based on lack of safety significance for "as-run" scenario)

People:

SM	=	Shift Manager (SRO)
CRS	=	Control Room Supervisor (SRO)
CRO	=	Control Room Operator (RO)
STA	=	Shift Technical Advisor (not licensed)
Comms	=	Communicator (SRO - activities limited)

Systems/Procedures:

APRM	=	Average Power Range Monitor
ARI	=	Alternate Rod Insertion
ATWS	=	Anticipated Transient Without Scram
EDG	=	Emergency Diesel Generator
EOP	=	Emergency Operating Procedure
ERPVD	=	Emergency RPV Depressurization
HPCS	=	High Pressure Core Spray (6,000 gpm)
LOCA	=	Loss Of Coolant Accident
RCIC	=	Steam turbine used for RPV injection (700 gpm)
RPS	=	Reactor Protection System
RPV	=	Reactor Pressure Vessel
RSCS	=	Rod Sequence Control System
SBO	=	Station Black Out (loss of all AC power)
SLC	=	Standby Liquid Control (System)

Administrative:

E or ES	=	Evaluation Scenario (eg., E-19 or ES-19)
Crew S	=	Staff Crew
*	=	Previously evaluated by NRC during 1990 requal.

ENCLOSURE 3 (Continued - Crew "A")

REQUALIFICATION SIMULATOR EXAMINATION SUMMARY

CREW "A"
OVERALL TEAM RATING ON THE SIMULATOR EXAMINATION: UNSATISFACTORY

Comments: This crew failed the examination due to ISCT failures in scenario E-09 and failure of the "Compliance/Use of Procedures and Technical Specifications" competency.

1. Scenario E-09 (loss of all TSW / ATWS) evaluated an ISCT which read, "Direct actions to control reactivity and shut down the reactor during ATWS conditions." The crew did not properly control reactivity during ATWS conditions by failing to reduce recirculation flow to minimum, and properly trip the running Reactor Recirculation Pump (single loop operations). This pump continued to run throughout the scenario.

After the crew identified ATWS conditions, the CRS specifically directed CRO#2 (who was temporarily covering P-603 while CRO#1 was pulling the RPS fuses) to reduce recirculation flow, but not to enter the Flow Instability Region of the power/flow map, then manually insert control rods. CRO#2 lowered recirculation flow to 34,000 gpm, and proceeded to manually insert control rods. CRO#1 returned about 3 minutes later, received a turnover from CRO#2, and continued rod insertion. The recirculation pump later shifted to slow speed automatically when drywell pressure reached 1.68 psig, and continued to run throughout the scenario.

These actions violated specific direction in the EOPs to reduce recirculation flow to minimum, shift the pump to slow speed, then trip the pump. This deviation resulted in sustained 60% reactor power for 5 to 6 minutes after ATWS conditions were identified. Power was not reduced until removing the RPS fuses inserted most of the control rods. Also, the WNP-2 "Conduct of Operations," PPM 1.3.1, identifies that EOP requirements supersede all other abnormal operating procedures. The Emergency Director (Shift Manager) was not informed of this deviation from EOP requirements; the crew did not express any identification of EOP non-compliance during the scenario.

2. Under the competency of "Compliance/Use of Procedures and Technical Specifications," the crew did not correctly implement procedures to the point that important procedural steps were not enacted correctly, which led to impeded recovery and unnecessary plant degradation ("1" rating). The following examples illustrate this point:

A. E-09 - During ATWS conditions, the CRS directed control rods to be manually inserted, but failed to direct the CRO to bypass the Rod Sequence Control System (RSCS) after power was down-scale. This resulted in a rod block which prevented rod motion, and a longer period of time in which the reactor was not shut down.

ENCLOSURE 3 (Continued - Crew "A")

REQUALIFICATION SIMULATOR EXAMINATION SUMMARY

Comments:

B. E-22 - During a medium/small LOCA, the CRS initiated Drywell spray based on Drywell pressure, instead of Wetwell pressure, as required by the EOPs. During post scenario questioning, the CRS incorrectly stated that Drywell pressure (rather than Wetwell pressure) exceeding 8 psig was the parameter that should be used to determine Drywell spray actions. Proper adherence to the EOPs would have reduced the amount of time Drywell spray was required, and may have precluded the need for Drywell spray depending upon the size of the LOCA.

C. E-22 - During a medium/small LOCA, the CRS deviated from EOP requirements which call for not terminating Drywell spray when Drywell pressure fell below 1.68 psig. The CRS directed the CRO to wait until pressure fell to 1.0 psig before securing sprays. Although this action may have had technical merits, the CRS did not consult any other crew member when deciding to make this deviation. The Emergency Director (Shift Manager) was not advised of this deviation from the EOPs; the crew did not express any identification of EOP non-compliance during the scenario.

D. E-09 - As described in Paragraph 1, above, the CRS, CRO#2, and CRO#1 failed to follow EOP requirements regarding the operating Reactor Recirculation Pump. This was a deviation from EOP requirements resulting in sustained operations at 60% power for 5 to 6 minutes during ATWS conditions.

3. Under the same competency of "Compliance/Use of Procedures and Technical Specifications," the crew failed to recognize EOP entry conditions and failed to carry out appropriate actions required by these conditions ("1" rating). The following examples illustrate this point:

A. E-09 - During ATWS conditions, the CRS for this scenario failed to re-enter "RPV Control," PPM 5.1.1, when Drywell pressure exceeded 1.68 psig. When questioned after the scenario, the CRS was not aware that re-entry was required. Had re-entry been made, the CRS may have identified that the Reactor Recirculation Pump was still operating, and should have been tripped.

B. E-09 - During ATWS conditions, the CRS failed to enter "Primary Containment Control," PPM 5.1.2, when Drywell pressure exceeded 1.68 psig. This procedure would have allowed use of Wetwell spray, if needed, to control primary containment pressure.

C. E-22 - During a medium/small LOCA, the CRS failed to enter "RPV Control," PPM 5.1.1, when Drywell pressure exceeded 1.68 psig. Entry was made several minutes later on reactor water level (Level 3). Had entry been made when required, the CRS may have had better control of RPV water level. When questioned after the scenario, the CRS was not aware that RPV Control entry was required on 1.68 psig.

ENCLOSURE 3 (Continued - Crew "B")

REQUALIFICATION SIMULATOR EXAMINATION SUMMARY

CREW "B"
OVERALL TEAM RATING ON THE SIMULATOR EXAMINATION: UNSATISFACTORY

Comments: This crew failed the examination due to an ISCT failure in scenario E-16 and failure of the "Compliance/Use of Procedures and Technical Specifications" competency.

1. Scenario E-16 (non-isolable RCIC steam leak into secondary containment) evaluated an ISCT for the CRS which read, "Direct actions to mitigate the effects of a non-isolable steam leak in secondary containment." A pre-identified ISCT evaluation criteria recognized that the crew must have initiated local action to isolate the leak. This evaluation criterion was originally assigned to a CRO under a separate ISCT. However, after the scenario was conducted, the examination team determined that the responsibility for initiating this action clearly rested with the CRS, who was procedurally cued to isolate the leak.

The crew failed to initiate local action to isolate the leak, even though they correctly identified the source and location of the leak early in the scenario. This task could have been initiated by directing an operator to locally close the outboard containment isolation valve, RCIC-V-8. Access to this valve was not hampered by the leak. The CRS was prompted to take this action by the step in Secondary Containment Control, PPM 5.3.1, which reads, "Isolate all systems discharging into the area....." The crew attempted to isolate the leak by closing RCIC-V-8 from the Control Room 1 minute after indications of a RCIC steam leak were received, but the valve did not operate. CROs #1 and #2 discussed manually closing V-8, but incorrectly decided that the valve was not accessible because of the RCIC room environmental conditions. The only observed local action directed from the Control Room was to have one Equipment Operator investigate conditions in the RCIC room. No other actions or discussions regarding manual operation of V-8 were observed during the scenario. Seven minutes had elapsed from the time that the crew discovered V-8 would not close from the Control Room, until emergency depressurization was initiated.

Failure to isolate RCIC-V-8 resulted in the need for emergency RPV depressurization after the Control Room received indications of an RCIC steam leak (RCIC trip, and RCIC room high temperature annunciator). The scenario was terminated 5 minutes after emergency depressurization upon facility request due to perceived modeling problems with RPV level. Attempts to locally isolate RCIC-V-8 had not been made, and the crew did not appear to be considering action in this area.

NOTE: By design, the scenario would not have allowed successful local isolation of V-8 until after emergency depressurization was initiated.

ENCLOSURE 3 (Continued - Crew "B")

REQUALIFICATION SIMULATOR EXAMINATION SUMMARY

Comments:

2. Under the competency of "Compliance/Use of Procedures and Technical Specifications," the crew failed to correctly implement procedures on numerous occasions to the point that important procedural steps were not enacted correctly, which led to impeded recovery and unnecessary plant degradation ("1" rating). The following examples illustrate this point:

A. E-16 - During a non-isolable RCIC room steam leak, the CRS failed to initiate actions to locally isolate the leak at RCIC-V-8. This event is described in Paragraph 1 above.

B. E-16 - During a non-isolable steam leak into secondary containment, the SM classified the event as a Site Area Emergency (SAE) instead of an Alert, as required by the EPIP classification procedure (PPM 13.1.1). Post-scenario questioning identified that the SM made the SAE classification because all of the other EOPs, except Secondary Containment Control, flagged an SAE classification when emergency depressurization was required. The SM did not express any concern for degraded plant conditions that could have warranted the higher event classification.

C. E-09 - During ATWS conditions, both the Shift Manager and the CRS felt that avoiding the Flow Instability Region of the power to flow map took priority over specific EOP requirements to lower recirculation flow to minimum, shift the recirculation pumps to slow speed, then trip the recirculation pumps. During the scenario, these actions were delayed (about 3 minutes from ATWS initiation) until the STA reported that control rods were inserted far enough to reduce flow. The Shift Manager then directed recirculation flow reduction. The CRS appeared to be confused by the EOP flow chart, and simply followed the SMs directions.

During this delay period, Drywell pressure exceeded 1.68 psig, causing the recirculation flow control valve to lock-up. Also during this delay period, the MSIVs isolated, resulting in 55% reactor power dumping into the suppression pool as the crew initiated action (tripped the recirculation pump) to lower reactor power.

Although the Emergency Director (Shift Manager) had knowledge of, and directed these actions, he did not consider them to be deviations from EOP requirements. Post-scenario questioning of the SM and CRS identified that their primary concern was avoiding the flow instability region; EOP compliance did not appear to be as much of a consideration.

ENCLOSURE 3 (Continued - Crew "B")

REQUALIFICATION SIMULATOR EXAMINATION SUMMARY

Comments:

D. E-09 - During ATWS conditions, just after the events described in Paragraph 2.C. above, the SM directed that the recirculation pump be tripped from fast speed to off. The EOPs required that the pump be shifted to slow speed, then tripped. This procedural error was mitigated by the previous MSIV closure, but still caused a RPV level swell to about +55 inches (potential challenge to RPV injection systems on Level 8 trip/isolation).

E. E-09 - ARI was not initiated until 7 minutes after ATWS conditions occurred. This error did not degrade plant conditions because of the specific ATWS simulator malfunction selected (hydraulic lock).

F. E-09 - During ATWS conditions, CRO#1 only pushed 2 of the 4 manual scram push-buttons, until prompted by CRO#3 to push all 4 buttons.

G. E-22 - During a medium/small LOCA with degraded high pressure injection, the CRS failed to direct use of the CRD pumps to help control RPV level. Both CRD pumps remained off until 17 minutes into the LOCA. Use of the CRD pumps would have helped maintain RPV water level, and reduced the reliance on cycling HPCS to control level.

H. E-22 - During a medium/small LOCA, the CRS initiated Drywell spray based on Drywell pressure, instead of Wetwell pressure, as required by the EOPs. During post scenario questioning, the CRS incorrectly stated that Drywell pressure exceeding 8 psig was the parameter used to determine Drywell spray actions. Proper adherence to the EOPs would have reduced the amount of time Drywell spray was required, and may have precluded the need for Drywell spray depending upon the size of the LOCA.

3. Under the same competency of "Compliance/Use of Procedures and Technical Specifications," the crew failed to recognize EOP entry conditions and failed to carry out appropriate actions required by these conditions ("2" rating). The following examples illustrate this point:

A. E-09 - During ATWS conditions, the CRS failed to re-enter RPV Control," PPM 5.1.1, on high Drywell pressure (1.68 psig). Had re-entry been made, the errors described in Paragraphs 2.D and 2.E (above) may have been avoided.

ENCLOSURE 3 (Continued - Crew "F")

REQUALIFICATION SIMULATOR EXAMINATION SUMMARY

CREW "F"
OVERALL TEAM RATING ON THE SIMULATOR EXAMINATION: UNSATISFACTORY

Comments: This crew failed the examination due to ISCT failures in scenario E-20 and failure of the "Understanding of Plant/Systems Response" competency.

1. Scenario E-20 (sequential loss of off-site power, eventually leading to SBO) evaluated an ISCT which read, "Direct actions maintaining maximum RPV inventory and recovering an emergency power source to prevent challenges to adequate core cooling." This ISCT was pre-assigned to the CRS, and was also assigned to the SM after the scenario was conducted. The examination team determined that the SM had taken responsibility for this task because of the personal involvement he took in investigating the status of the EDGs. The crew failed to accomplish the ISCT by not recovering an emergency power source during station blackout conditions.

During the scenario, the crew responded to the sequential loss of off-site power by tripping both Emergency Diesel Generators (EDGs) because they incorrectly concluded that the EDG cooling water system had failed. This action was taken by a CRO with concurrence of the CRS, and resulted in Station Black-Out (SBO) conditions. The crew considered using the EDGs to recover, but failed to pursue this success path because they incorrectly determined that there was a TR-B 86 Lock-out present. The crew incorrectly assumed that this lock-out prevented use of the EDGs to power the vital electrical busses: By the end of the scenario, the crew was in SBO conditions waiting for recovery of an off-site power source (of which they were cued would be "hours"), and had ruled out the only success path, using the EDGs.

NOTE: After the scenario was completed and the crew had left, the simulator was placed back into "run" and the emergency stop was reset for EDG #1. The diesel started and loaded its bus automatically.

2. Under the competency of "Understanding of Plant / Systems Response," the crew failed to interpret control room indicators correctly such that serious errors resulted that degraded plant conditions ("1" rating). The following example illustrates this point:

A. E-20 - Immediately after tripping both EDGs, CRO#1 observed the "Ready to Transfer" light next to the EDG #1 breaker extinguish. The CRS was closely observing this evolution at arms reach.

Both CRO#1 and the CRS agreed that this indication meant that there was an 86 lock-out relay present on both EDGs that prevented either EDG from loading its vital bus. A report was then made to the SM that there was a TR-B 86 Lock-out Relay present on both EDGs.

ENCLOSURE 3 (Continued - Crew "F")

REQUALIFICATION SIMULATOR EXAMINATION SUMMARY

Comments:

Actual simulator indications identified that there were no 86 lock-out relays present of any kind. The type of lockout identified by CRO#1 and the CRS would have been indicated by an annunciator and other indications on the vertical portion of the EDG panel.

This error contributed significantly to the incorrect assumption that neither EDG #1 nor #2 was a potentially available power source.

3. Under the competency of "Understanding of Plant / Systems Response," the crew failed to understand how the plant systems and components operated such that serious mistakes resulted that degraded plant conditions ("1" rating). The following examples illustrate this point:

A. E-20 - During post-scenario questioning, the SM, CRS, and CRO#1 all incorrectly stated that a transformer fault (TR-B 86 Lock-out Relay) prevented either EDG from loading its respective vital bus. In fact, even if this lock-out had been present, it would not have prevented the EDGs from loading.

This error contributed significantly to the incorrect assumption that neither EDG #1 nor #2 was a potentially available power source.

B. E-20 - Immediately after loss of off-site power, CRO#1 observed that both service water pumps had tripped and that system valves were closing. He incorrectly concluded that the service water system was not available, and proceeded to trip both EDGs because they did not have a source of cooling water. These observations and actions were reported to the CRS, who was closely supervising CRO#1; the CRS agreed with the conclusion and authorized tripping both EDGs.

The CRS and CRO#1 were not aware of the load shed feature of the 4160 Volt System which had stripped the service water pumps from their power supply, and would have automatically re-energized them. Nor were they aware of the automatic valve features that re-aligned service water upon a loss of off-site power. Inadequate knowledge of these automatic plant features caused the CRS and CRO#1 to conclude that service water was not operable.

This error contributed significantly to the incorrect assumption that neither EDG #1 nor #2 was a potentially available power source.

4. Under the competency of "Understanding of Plant / Systems Response," the crew failed to demonstrate an understanding of how their actions affected system/plant conditions such that minor misunderstandings by individuals were corrected by the team ("2" rating). The following example illustrates this point:

ENCLOSURE 3 (Continued - Crew "F")

REQUALIFICATION SIMULATOR EXAMINATION SUMMARY

Comments:

A. E-21 - During ATWS conditions, the crew had lowered reactor water level to control power, and was using the condensate booster pumps (CBP shut-off head is about 580 psig) for make-up. Reactor pressure had been lowered to about 550 psig, and the CBP flow control valve was set at 30% open. CRO#2 was directed to control reactor pressure to maintain reactor water level (within a given band). When questioned by CRO#2, the CRS explained these plant conditions and repeated his instructions. As the scenario progressed, CRO#2 continued to ask for a pressure band. During post scenario questioning, CRO#2 did not understand how his actions (pressure control) effected reactor water level.

This lack of knowledge increased the crew confusion level, and required increased supervision by the CRS to ensure reactor level and pressure were properly maintained.

ENCLOSURE 3 (Continued - re-organized Crew "B")

OPERATIONAL EVALUATION SIMULATOR EXAMINATION SUMMARY

NOTE: This crew was composed of operators who individually passed the March 1991 requalification examination while assigned to different crews.

Re-organized CREW "B"
OVERALL TEAM RATING ON THE SIMULATOR EXAMINATION: UNSATISFACTORY

Comments: This crew was evaluated as unsatisfactory based on overall crew performance during two simulator scenarios. Two individuals were evaluated as requiring remediation prior to returning to shift. Several significant crew weak areas were documented; many of these errors stemmed from non-compliance with the facility's Emergency Operating Procedures. One operator (RO license) was also evaluated as having inadequate individual knowledge of specific plant systems, especially the Reactor Protection System (RPS).

1. Scenario E-4 presented the crew with a hydraulic ATWS from about 70% reactor power. The crew took actions to reduce power. Within about 6 minutes the APRMs were downscale, recirculation pumps were off, SLC was initiated, and RSCS was bypassed. The crew had correctly focused on inserting individual control rods as the only remaining operator action needed to shut down the reactor. At this point, a medium LOCA (about 7,000 gpm, sized to be just larger than HPCS/RCIC capacity) was ramped in over 5 minutes.

From the time that these events occurred, the crew chose to control reactor water level using only RCIC (about 700 gpm capacity); HPCS (about 6,000 gpm capacity) was manually overridden. These actions were allowed by the EOPs at the time of performance, and had technical merit with regards to reactivity control. However, the facility EOPs (PPM 5.1.2) require that HPCS be used in order to maintain water level above Top of Active Fuel (TAF).

The CRS failed to direct use of HPCS, which he had manually overridden, to maintain reactor water level above TAF. Reactor water level had maintained a continuous downward trend for 13 minutes, from the time that ATWS was initiated until about 2 minutes after initially uncovering the fuel, when an inadvertent injection from low pressure systems re-covered most of the core. Failure to use HPCS resulted in ERPVD, and a core uncovery time of over 4 minutes (occurring less than 10 minutes after reactor power operations).

The SM informed the CRS and CRO#2 that reactor water level was approaching TAF (-161 inches), and announced 30 seconds later that water level was at TAF (still decreasing).

ENCLOSURE 3 (Continued - re-organized Crew "B")

OPERATIONAL EVALUATION SIMULATOR EXAMINATION SUMMARY

Comments:

The CRS failed to properly initiate ERPVD, as required by PPM 5.1.3. He directed CRO#2 to open 7 SRVs immediately upon hearing that water level was at TAF. The CRS did not refer to the EOPs (PPM 5.1.2 or 5.1.3) immediately prior to directing ERPVD.

After the SRVs were opened, the SM correctly informed the CRS that ERPVD was not required until water level reached -192 inches (29 inches below TAF). The CRS directed CRO#2 to close all SRVs. This action caused water level to shrink down to -240 inches, uncovering one-half of the core. The CRS then directed CRO#2 to again open 7 SRVs. Again, the only reference the CRS made to PPM 5.1.3 was to verify the -192 inch setpoint for ERPVD.

The CRS failed to direct that low pressure injection systems be overridden prior to ERPVD, as required by PPM 5.1.3. As reactor pressure fell below Low Pressure Injection (LPI) shut-off head, CRO#2 identified that LPCS and LPCI "C" were injecting. The CRS directed that these systems be manually overridden. (At this point, the core had been uncovered for over three minutes, and the inadvertent LPI injection had just recovered most of the core.) The CRS then directed that HPCS be initiated to re-cover the core.

Post scenario questioning identified that the CRS was not sure of when ERPVD was required; after reviewing the EOPs, he incorrectly stated that the direction to initiate ERPVD provided by PPM 5.1.2 conflicted with the requirements of PPM 5.1.3. The CRS was not able to identify any reason why HPCS was not initiated prior to reactor water level dropping below TAF other than the rapid progression of the transient. Neither the SM nor CRO#2 (who was directed to maintain water level above TAF, and who had earlier overridden HPCS) had identified HPCS as an available source of injection.

2. During both scenarios E-4 and E-8 (uncontrolled radioactivity release with failed fuel), the crew demonstrated weakness in the area of plant systems knowledge and diagnosis of events. The following examples illustrate this point:

A. E-8 - During an uncontrolled radioactivity release which required General Emergency classification, the SM incorrectly interpreted the control room wind direction indicator by 180 degrees. Any evacuation recommendations initiated from the control room would have sent personnel directly into the release plume.

B. E-8 - Both CRO#1 and CRO#2 failed to diagnose that a dropped control rod had occurred and was the probable cause of failed fuel. This failure had little impact on crew performance because the SM and CRS had diagnosed a high probability of this condition, but had failed to inform either CRO.

ENCLOSURE 3 (Continued - re-organized Crew "B")

OPERATIONAL EVALUATION SIMULATOR EXAMINATION SUMMARY

Comments:

C. E-8 - A malfunction at the beginning of this scenario caused an electrical failure of the manual scram system, requiring use of ARI to insert control rods. CRO#1 apparently failed to verify that the reactor scram alarms (e.g., MODE SWITCH IN SHUTDOWN) were not consistent with plant conditions (ATWS).

Post-scenario questioning identified that his knowledge of the Reactor Protection System (RPS) was deficient. Typical questions probing his failure to diagnose ATWS conditions included, 'What happens to RPS when the MODE SWITCH is placed in SHUTDOWN?' CRO#1 answered that the control rods were supposed to insert, but he could not explain how this was accomplished, nor what other indications of improper RPS operation were available.

D. E-8 - After diagnosing significant fuel failures and using the SRVs to control reactor pressure, the CRS discharged water from the Suppression Pool to Rad Waste without assessing the increased radiation levels that would be caused along this piping flow path.

E. E-4 - The CRS and CRO#2 failed to understand, during the scenario, that their actions to manually override HPCS prevented automatic system operation to keep reactor water level above TAF. This error is described in Paragraph 1, above.

F. E-4 - Near the end of the scenario, the CRS and CRO#2 attempted to initiate RCIC to help control reactor water level after ERPVD when reactor pressure was only 25 to 30 psig. After about a minute, the STA identified that there was not adequate steam pressure to run RCIC. (In fact, RCIC had isolated on low reactor pressure, and was not available.)

ENCLOSURE 4

WNP-2 SIMULATOR DOCUMENTS

The following documents were in the WNP-2 simulator prior to the Operational Evaluation. These documents had apparently been routinely kept in the simulator, but did not appear to be maintained in a controlled manner:

1. Photo Log of Reference Plant
2. Plant Systems Manual, Volumes 1 through 6
3. Simulator WNP-2 Instrument Master Data Sheets, Books 1 through 28
4. WNP-2 Floor Plans (similar to radcon survey maps)
5. 10 CFR, dated 1988
6. Graphic Display System (GDS) - contains information on Plant Data Information System Design Specifications (Rev. 2); dates back to 1985
7. Instructor Stations Instructions
8. Process Radiation Monitoring Technical Specification Setpoint Bases 1985
9. ESP (EOP Tool Kits) packages are unmarked except for a criptive note identifying EOP section. Several tool kits contain a note stating that some items are missing. None of the tool kits were controlled.
10. "Useful Information" notebook at operator console
11. Circ Water Pump House Supervisor Information notebook
12. Replacement Operator Simulator Training Schedule Phase One, Simulator and Classroom (yellow) notebook
13. Leak Detection (blue) notebook
14. Plant Process Computer Points List (blue) notebook
15. Emergency Phone Directory, Rev. 8, 1990
16. Booklet of Terminal Board and Fuse Arrangement
17. Draft, For Review Only - Improved Technical Specifications
18. Data Book, 7 volumes, (white) containing fuse lists, etc.
19. WPPSS Drawing Index and Table of Contents - no clear date of revision
20. DEH Directory (Westinghouse generator control), no revision date
21. ATWS Hydraulic (Bases)
22. Operator Aids on top of Electrical Wiring Drawing (EWD) files

ENCLOSURE 5

WNP-2 EOP ISSUES IDENTIFIED DURING THE REQUALIFICATION EXAMINATION

The following WNP-2 procedures and acronyms are referenced in this enclosure:

PPM 1.3.1 Conduct of Operations
PPM 5.0.8 WNP-2 Emergency Procedure Guidelines (EPGs)
PPM 5.1.1 RPV Control (EOP)
PPM 5.1.2 Failure To Scram (EOP)

ATWS	Anticipated Transient Without Scram
EOP	Emergency Operating Procedure
EPG	Emergency Procedure Guidelines (technical bases for EOPs)
RCIC	A steam turbine used for RPV water level control (700 gpm)
RPS	Reactor Protection System
TAF	Top of Active Fuel

1. The Chief Examiner identified that PPM 5.1.2 does not permit operators to override RPS logic trips during ATWS conditions in order to reset the reactor scram signal. Step RC/Q-6.2 of the EPGs, third item, requires that RPS logic trips (e.g., high drywell pressure) be overridden, if necessary, to reset the reactor scram signal. This step reads, "Reset the scram, defeating RPS logic trips if necessary,....."

However, the Chief Examiner identified that PPM 5.1.2, "Failure to Scram", did not permit operators to override RPS logic trips during ATWS conditions in order to reset the reactor scram signal; this is contrary to the guidance in PPM 5.0.8 (the EPGs). This EOP step reads,

"Can the scram be reset? - Yes- Reset the scram.
No- Attempt to start both CRD pumps."

When the operator reaches this step in the flowchart, he is required to bypass steps that reset the scram, if RPS scram logic was tripped because the tripped scram logic physically prevents resetting the scram. In this case, EOP direction is needed to bypass the RPS logic trips because the conditions that result from an ATWS usually generate a scram signal, and thus prevent resetting the scram. Resetting the scram is necessary to allow the scram accumulators to be refilled and Scram Discharge Volume to be drained so that another scram can be initiated to insert the control rods.

Resetting the scram signal and scramming again is an effective method to insert control rods during ATWS conditions; the EOPs do allow this action under many typical ATWS conditions (i.e., when a scram signal would be present).

ENCLOSURE 5 (Continued)

FACILITY POSITION: The facility has agreed that PPM 5.1.2 requires revision to incorporate this guidance, and that a procedure is needed to implement bypassing of the RPS logic trips. The facility stated that procedure would be issued by April 24, 1991.

2. The Chief Examiner identified that the facility had trained its operators that EOP exit (i.e., transition to a non-emergency procedure) was allowed whenever the EOP entry conditions no longer existed, without regard to other plant conditions. The following example was discussed with the facility staff:

> Assume a steam leak and fuel damage caused a non-isolable off-site radioactivity release, and PPMs 5.1.1 (RPV Control) and 5.4.1 (Radioactivity Release Control) have been entered and a Site Area Emergency has just been declared. Restoring reactor water level to the normal band, by itself, would not justify exit from PPM 5.1.1 when the uncontrolled radioactive release is in progress.

The Chief Examiner reminded the facility representatives that PPM 1.3.1 step 1.3.1.5.L.3.b.5 (page 44 of 69) requires that EOPs only be exited when an overall determination has been made that an emergency condition no longer exists. This step reads, "When the emergency no longer exists, the EOP's should be exited and the appropriate system operating procedure should be entered." Although mitigation of the entry condition symptoms is one indicator of overall plant condition, it does not provide reliable indication that an emergency condition does not exist.

Facility management stated it had been a WNP-2 practice that any EOP being performed may be exited when the conditions that required EOP entry had been recovered, without regard to other plant conditions.

FACILITY POSITION: After further discussions, the facility representatives agreed that their initial position was incorrect, and issued an interoffice memorandum (2/8/91) to all licensed operators, including an entry in the Night Orders, to correct the EOP exit criteria practice and reinforce the applicable requirements of PPM 1.3.1.

3. The Chief Examiner identified that WNP-2 EOPs do not provide guidance for how to control reactor water level after Steam Cooling has been initiated, if an injection system is recovered prior to loss of level indication (i.e., total core uncovery).

[Steam Cooling is a last resort method to cool the core when no injection systems are available (e.g., station blackout with loss of RCIC), and water level has dropped below TAF. The exposed core is cooled by the steam surrounding it. The steam temperature and pressure increase eventually causes a Safety Relief Valve to open. When an injection system is recovered after Steam Cooling has been initiated, emergency depressurization is performed, and level restoration is initiated.]

ENCLOSURE 5 (Continued)

[PPM 5.0.8, Contingency #1, "Alternate Level Control" provides guidance for controlling reactor water level when degraded plant conditions have prevented the operator from controlling water level above TAF. WNP-2 has incorporated Contingency #1 into PPM 5.1.1.]

PPM 5.0.8 provides guidance that reactor water level control be continuously controlled during Steam Cooling using Contingency #1, "Alternate Level Control." The step in Contingency #1 which identifies when Steam Cooling is required (PPM 5.0.8, step C1-6, page 227 of 334) does not direct exit from Contingency #1, but rather, requires continued performance while Steam Cooling is in progress. Only two cases applicable to PPM 5.1.1 lead to procedural exit from Contingency #1:

- reactor water level cannot be determined, or
- Reactor water level is increasing (this would require re-entry into "RPV Water Level" section of PPM 5.1.1).

After an injection system has been recovered, and reactor water level is increasing, Contingency #1 (PPM 5.0.8 page 215 of 344); requires that RPV Level Control be re-entered and used to control reactor water level.

Contrary to the above guidance, PPM 5.1.1 directs exit from the reactor water level control section (equivalent to Contingency #1 described above) when Steam Cooling is required. After exit from the reactor water level control section, PPM 5.1.1 provides no guidance for water level control, and has no mechanism to re-enter this section.

If an operator was required to perform Steam Cooling, EOP guidance for reactor water level control would not be used when reactor water level was recovered (i.e., increasing).

FACILITY POSITION: This contingency has been incorporated into the EOP correctly. There would be no need for EOP guidance to control RPV water level after Steam Cooling was required. The operator would use a lower tier procedure (i.e., normal operating procedure).

4. The Chief Examiner identified that PPM 5.1.1 does not include a caution to the operators (PPM 5.0.0, Caution #1) that reactor water level indication may be erroneous at a step that is critical during the performance of Steam Cooling.

PPM 5.0.8 requires that PPM 5.0.0 Caution #1 be applied at all times while performing RPV Water Level Control (PPM 5.0.8, page 55 of 344) and Contingency #1 (PPM 5.0.8, page 214 of 334). The step in Contingency #1 which identifies when Steam Cooling is required (PPM 5.0.8, step C1-6, page 227 of 334) does not direct exit from Contingency #1, but rather, requires continued performance while Steam Cooling is in progress. This ensures that Caution #1 is continuously applied while monitoring reactor water level.

ENCLOSURE 5 (Continued)

Contrary to the above guidance, PPM 5.1.1 directs exit from the reactor water level control section (equivalent to Contingency #1 of PPM 5.0.8) when Steam Cooling is required. After exit from the reactor water level control section, PPM 5.1.1 provides no reference to Caution #1 of PPM 5.0.0.

[The applicable step in PPM 5.1.1, RPV Pressure Control section which should reference to Caution #1 reads, "Can RPV water level be determined?" This caution is required because of the incorrect incorporation of Steam Cooling discussed in issue #3.]

[Once initiated, Steam Cooling is continued until an injection system is recovered, or reactor water level cannot be determined. When either case occurs, emergency RPV depressurization is performed. In general, "water level cannot be determined" means the Fuel Zone water level indication has gone off-scale low (i.e., the core is totally uncovered). Caution #1 should be used to provide a critical warning to the operator that the reactor water level instruments are erroneous below a specified level.]

If Caution #1 is not followed, emergency RPV depressurization may not be performed when required by the WNP-2 EOPs because of erroneous reactor water level indication.

FACILITY POSITION: The facility agrees that Caution #1 may be applicable to the identified step of PPM 5.1.1,RPV pressure control section.

5. The Chief Examiner identified that the facility EOPs (PPM 5.1.1, RPV Level Control section) do not allow use of alternate injection systems to maintain reactor water level above TAF.

The BWR Owners Group Emergency Procedure Guidelines, Revision 4 provides guidance to use alternate injection systems to maintain reactor water level above TAF.

The facility deleted guidance from PPM 5.0.8 step RC/L-2 to use alternate injection systems to maintain RPV water level above TAF. This guidance would have allowed use of Service Water, Fire Water Standby Liquid Control, ECCS keep-fill and other non-safety grade systems to maintain reactor water level above TAF when other (ECCS) systems were not available or not adequate. Facility EOPs do allow use of alternate injection systems when water level falls below TAF.

The facility may not consider use of all available injection systems to maintain reactor water level prior to uncovering the core.

FACILITY POSITION: WNP-2 has not performed a safety analysis that would allow use of these systems for RPV injection. Use of alternate injection systems is allowed by the EOPs when the plant has degraded to beyond design bases conditions. Not using these systems while within the plant design bases is consistent with the NRC Safety Evaluation Report for the EPGs issued in August, 1988.

ENCLOSURE 6

SIMULATION FACILITY REPORT

Facility Licensee: Washington Public Power Supply System, WNP-2

Facility Docket No.: 50-397

Operating Tests Administered on: February 24 through March 8, 1991

This form is to be used only to report observations. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of non-compliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

During the conduct of the simulator portion of the operating tests, the following items were observed (if none, so state):

ITEM	DESCRIPTION
RPV water level indication (all meters).	After emergency RPV depressurization, level indication continually oscillated +/- 15 inches and did not appear to trend in a controllable manner. This forced scenarios to be terminated about 5 minutes after emergency depressurizing the RPV.
Containment temperature (all meters)	During steam line ruptures in the drywell, containment temperature would increase to about 250°F, then stepped down to 150°F, and continued increasing. Temperature readings above 250°F were not observed.

ENCLOSURE 7

EXAMINATION SECURITY CONCERNS

The following examination security concerns were identified by the NRC to the facility evaluators. Each of these concerns could have compromised examination validity had corrective actions not been promptly taken:

1. During the written examinations, the operators were controlled in holding rooms in-between examinations on opposite sides of the Plant Support Facility to prevent those who had taken only the Category A exams from discussing the exam with those who had taken only the Category B exam. However, both rooms had telephones, and usually facility evaluators were not present to monitor use of these telephones in the holding rooms. The Chief Examiner determined that communications between rooms was not possible during the brief exposure period (a facility evaluator happened to be on one of the two telephones), and discussed this concern with the facility. The telephone was removed from one of the two rooms.

2. During performance of the JPMs the facility evaluator initiated a JPM just before entering the Radiologically Controlled Area (RCA); the operator entered the RCA and proceeded by himself to the Control Room while the NRC and facility evaluators reviewed their Radiological Work Permit (RWP). This JPM was cancelled and replaced with a different JPM that had been previously reviewed and was approved by the Chief Examiner.

3. During performance of the dynamic simulator examinations, an operator who was not participating in the NRC examination joined his crew while the crew was in-between scenarios during the morning session. It appeared that the facility was not taking any administrative precautions to control this operator and prevent his informing the afternoon crews of the morning's events. Since the same scenarios were used both the morning and afternoon sessions, this could have compromised the examination. This operator was placed on a security agreement at the NRC's request.