U.S. NUCLEAR REGULATORY COMMISSION
REGION I

OPERATOR LICENSING PROGRAM EVALUATION AND
BWR POWER OSCILLATION PROGRAM INSPECTION

Report No.:	50-410/89-12 (OL)
Facility Docket No.:	50-410
Facility License No.:	NPF-69
Licensee:	Niagara Mohawk Power Corporation 301 Plainfield Road Syracuse, New York 13212
Facility:	Nine Mile Point Nuclear Station, Unit 2
Examination Dates:	June 26 - 29, 1989, July 17 - 27, 1989 and August 2 - 3, 1989
Examiners:	Theodore Easlick, Operations Engineer Michael Spencer, EG&G Mark Parrish, EG&G

Chief Examiner: N. F. Conicella — 8/22/89
Nicola F. Conicella, Operations Engineer
BWR Section, Operations Branch, DRS
Date

Reviewed by: Richard J. Conte — 8/22/89
Richard J. Conte, Chief
BWR Section, Operations Branch, DRS
Date

Approved by: Robert M. Gallo — 8/22/89
Robert M. Gallo, Chief
Operations Branch
Division of Reactor Safety
Date

Executive Summary

Written and operating requalification examinations were administered to 12 reactor operators (ROs) and 12 senior reactor operators (SROs). These operators were divided into six crews. Each crew consisted of two SROs and two ROs. The examinations were graded concurrently by the NRC and the facility training staff. As graded by the NRC, four SROs and three ROs failed the written examination. All other operators passed the written examinations. As graded by the NRC, four SROs and one RO failed the simulator portion of the examination. All other operators passed the simulator portion of the examination. As graded by the NRC, three SROs failed the Job Performance Measures (JPMs) portion of the examination. All other operators passed the JPM portion of the examination. Finally, as graded by the NRC, two crews were determined to be unsatisfactory. All other crews were determined to be satisfactory. The requalification program was determined to be UNSATISFACTORY in accordance with Examiner Standard ES-601 based on the NRC results of nine overall individual failures out of 24 examined operators and seven individual written examination failures out of 24 examined operators. The program was marginal on several other criteria and numerous other weaknesses were exhibited as discussed herein.

As partial justification for continuing operation, two additional SROs and three ROs were administered operating examinations during the week of July 31, 1989. Only four of these operators were administered written examinations. One RO was not administered the written examination since he was involved in the validation of the written examinations. (Since this one RO has not taken a complete requalification examination, he will not be included in the total number examined as discussed herein). These operators constituted one crew. As graded by the NRC, one SRO operator failed the written examination. All other operators passed all portions of the examinations that they were administered. As graded by the NRC, the crew was determined to be satisfactory. The results of this additional crew and individual examination were not used to determine the status of the requalification program.

Based on an independent assessment, the NRC staff agreed that the licensee can continue power operations at NMP2 and meet the technical specification requirements for operator staffing with an augmented four shift rotation. However, the licensee's action plan does not address all underlying reasons for the operator performance deficiencies apparently because of an incomplete root cause analysis.

The inspection of the licensee's BWR Power Oscillation Program is documented in Section 5 of this report. No violations were identified. However, the licensee did not adequately implement the requested actions of NRC Bulletin 88-07 and Supplement 1 to the Bulletin. Inadequacies noted were: a revised procedure based on an entry condition that was not understood by all licensed operators, a weak knowledge level displayed by the operators interviewed with regard to the recent procedure revision, and several procedures that did not contain appropriate cautions. The reasons for these inadequacies appeared to be that the licensee's verification process for ensuring that licensed operators understand procedure changes was insufficient and the licensee's procedure review was not comprehensive enough.

Report Details

1. Introduction

During the examination period, NRC examiners conducted a requalification examination of 24 licensed operators (12 reactor operators (ROs) and 12 senior reactor operators (SROs)) at the Nine Mile Point Nuclear Station, Unit 2. The examiners used the process and criteria in NUREG 1021, "Operator Licensing Examiner Standards," specifically, ES-601, "Administration of NRC Requalification Program Evaluation," Revision 5, dated January 1, 1989. Additionally, as partial justification for continued operation with an UNSATISFACTORY requalification program, five additional operators were given NRC requalification examinations.

An entrance meeting was held with the licensee on May 30, 1989 in the Regional Office. The purpose of this meeting was to brief the licensee on the requirements of the new requalification program evaluation and to outline a prospective schedule for the requalification examination period.

An interim exit meeting was held on July 21, 1989 at the facility's training center. The purpose of this meeting was to notify the facility of the individual failures during the first week of examinations that had to be immediately removed from licensed duties.

The exit meeting was held with the licensee on July 31, 1989, in the NRC Regional Office. The purpose of this meeting was to inform the facility of the NRC results of the two week examination, to inform the facility that the requalification program was determined to be UNSATISFACTORY, and to have the facility describe their results of the requalification examination. The facility was also informed of the individual and programmatic weakness identified during the examination and the results of the BWR power oscillation inspection. The facility, during this exit meeting, presented their justification for continuing operation. The NRC then decided to administer NRC requalification examinations to five additional operators in order to justify the facility's continued operation.

A supplemental exit meeting was held on August 3, 1989 at the facility's training center. The purpose of this meeting was to notify the facility of the NRC results of the examinations provided to the additional five operators to justify continued operation.

All NRC and facility personnel that attended these meetings are listed in Attachment 1. The NRC-facility examination team is also listed in Attachment 1.

2. Individual Examination Results

The following is a summary of the individual examination results for the originally scheduled requalification examination. Only these results were used for the program evaluation.

NRC Grading	RO Pass / Fail	SRO Pass / Fail	Total Pass / Fail
Written	9 / 3	8 / 4	17 / 7
Simulator	11 / 1	8 / 4	19 / 5
Walk-Through	12 / 0	9 / 3	21 / 3
Overall	8 / 4	7 / 5	15 / 9

Facility Grading	RO Pass / Fail	SRO Pass / Fail	Total Pass / Fail
Written	9 / 3	8 / 4	17 / 7
Simulator	10 / 2	8 / 4	18 / 6
Walk-Through	12 / 0	9 / 3	21 / 3
Overall	7 / 5	7 / 5	14 / 10

The following is a summary of the individual examination results for the supplemental requalification examination. These results were not used for the program evaluation.

NRC Grading	RO Pass / Fail	SRO Pass / Fail	Total Pass / Fail	
Written	2 / 0	1 / 1	3 / 1	Note(1)
Simulator	3 / 0	2 / 0	5 / 0	
Walk-Through	3 / 0	2 / 0	5 / 0	
Overall	2 / 0	1 / 1	3 / 1	Note(1)

Facility Grading	RO Pass / Fail	SRO Pass / Fail	Total Pass / Fail	
Written	2 / 0	1 / 1	3 / 1	Note(1)
Simulator	3 / 0	2 / 0	5 / 0	
Walk-Through	3 / 0	2 / 0	5 / 0	
Overall	2 / 0	1 / 1	3 / 1	Note(1)

The following is a summary of the individual examination results for the combined originally scheduled and supplemental requalification examinations. These results were not used for the program evaluation.

NRC Grading	RO Pass / Fail	SRO Pass / Fail	Total Pass / Fail	
Written	11 / 3	9 / 5	20 / 8	Note(1)
Simulator	14 / 1	10 / 4	24 / 5	
Walk-Through	15 / 0	11 / 3	26 / 3	
Overall	10 / 4	8 / 6	18 / 10	Note(1)

Facility Grading	RO Pass / Fail	SRO Pass / Fail	Total Pass / Fail	
Written	11 / 3	9 / 5	20 / 8	Note(1)
Simulator	13 / 2	10 / 4	23 / 6	
Walk-Through	15 / 0	11 / 3	26 / 3	
Overall	9 / 5	8 / 6	17 / 11	Note(1)

Note (1): One RO was not administered the written examination since he was involved in the written examination validation process. He satisfactorily completed the simulator and walk-through portions but cannot be given credit for a complete examination until he satisfactorily completes the written examination. This operator is not counted in the total for individuals examined.

2.1 Individual Operator Strengths and Weaknesses

These were strengths and weaknesses observed more than once during the conduct of the examination.

a. Individual Operator Strengths

- None noted

b. Individual Operator Weaknesses

Simulator

- communications and team interactions were generally poor.
- SROs displayed an inability to properly classify events in accordance with plant Emergency Action Procedures (EAPs).
- SROs displayed an inability to prioritize operator actions during emergency situations.
- SROs did not provide specific guidance to ROs.
- SROs did not periodically brief the crew so that all operators understand what was happening and the direction that the crew was heading.
- SROs displayed weaknesses in their ability to properly use the emergency operating procedures (EOPs).
- SROs had difficulty with certain technical specification interpretations.
- SROs/STAs did not provide the required plant assessment and backup to the station shift supervisor (SSS) during the initial stages of a casualty when plant assessment and backup are needed most.
- RO actions were not always in accordance with plant procedures.
- ROs showed little initiative and required specific direction from the SROs.
- ROs displayed a weakness in their ability to perform immediate actions during an emergency without referring to the procedure.
- ROs displayed an inability to properly determine all systems injecting water into the reactor.
- ROs displayed an inability to control water level in the reactor in the normal band during emergency conditions.

- ROs did not provide feedback to the SROs and did not challenge the SROs when incorrect orders are given to the ROs.

<u>Walk-Through</u>

- Several ROs were not aware of the contents of the local Emergency Operating Procedure - Alternate Rod Insertion (EOP-RQ) tool box.

- Knowledge of the operation of the load sequencer for an emergency 4KV bus.

<u>Written</u>

- Effect of single recirculation loop operation on thermal limits.

- Operator actions for a loss of main generator stator water cooling.

- Operator actions for a loss of main generator H2 seal oil.

- Effect of loss of feedwater or feedwater heating on various plant parameters.

- Operator actions on a loss of main condenser vacuum.

- Knowledge of the operation of the EHC load limiter.

- Knowledge of containment isolation setpoints.

- Effect of a loss of instrument air on various systems which generate an RPS scram signal.

- Ability to determine core flow from recirculation loop flow indications.

- Actions required for a loss of RBCLC.

- Notifications required during emergency conditions.

- Effect of feedwater addition on reactor pressure after a scram has occurred.

- Knowledge of RRCS feedwater runback operation.

3. Program Evaluation

3.1 Overall Rating: UNSATISFACTORY

The facility program for licensed operator requalification training was rated as UNSATISFACTORY in accordance with the criteria established is ES-601, paragraph C.3.b.(1), C.3.b.(2), D.1.c.(2)(c), D.2.c.(2)(b), and D.3.c.(2)(b). Specifically, the facility did not meet criteria C.3.b(1)(b) and C.3.c.(2)(b), but the facility was marginal on several other criteria as described below:

a. At least 75% of all operators that are administered the examination must pass all portions of the examination.

   Only 63% of the operators passed all portions of the examinations. This criterion has not been met, therefore, the program is determined to be UNSATISFACTORY.

b. At least 75% of all operators must pass the written examination.

   Only 70% of the operators passed the written examination. This criterion has not been met, therefore, the program is determined to be UNSATISFACTORY.

c. The pass / fail decision agreement between the NRC and facility grading of the written and operating examinations shall be at least 90%.

   There was 100% agreement on the grading of the written examination and 95% agreement initially on the operating examination. This disagreement was a result of the NRC having failed one RO on the simulator examination that the facility did not fail. The facility subsequently agreed that the RO was a failure. This criterion is met.

d. A program may be judged UNSATISFACTORY if the NRC judges at least one crew UNSATISFACTORY and the facility evaluators judge the same crew SATISFACTORY.

   There was one crew disagreement initially in that the NRC judged the crew unsatisfactory but the facility judged the crew as satisfactory. The facility subsequently agreed that the crew was unsatisfactory. This criterion is met.

e. A program may be judged UNSATISFACTORY if there is less than 90% agreement between the NRC and facility on the individual pass / fail determinations for the simulator examination with the facility evaluating fewer individuals as unsatisfactory.

There was 95% agreement initially on the grading of the simulator examination. This disagreement was a result of the NRC having failed one RO that the facility did not fail. The facility subsequently agreed that the RO was a failure. This criterion is met.

f. If more than 1/3 of the crews are determined to be UNSATISFACTORY by the NRC regardless of individual failures, the overall program shall be judged UNSATISFACTORY.

Six crews were evaluated and two crews were determined to be unsatisfactory. Therefore, exactly 1/3 of the crews were unsatisfactory. This criterion is met.

g. The program meets the requirements of 10 CFR 55.59(c)(2), (3) and (4) or is based on a systems approach to training.

As reported by the licensee, the licensee's program meets the 10 CFR 55.59 criteria.

h. The pass / fail decision agreement between the NRC and the facility grading of the walk-through examination shall be at least 90%.

There was 95% agreement on the grading of the walk-through examination. This disagreement was a result of the NRC having failed one SRO that the facility did not fail. This criterion is met.

i. At least 75% of all operators must pass the walk-through examination.

The pass rate for the walk-through examinations was 87%. This criterion has been met.

j. The pass / fail decision agreement between the NRC and the facility grading of the written examination shall be at least 90%.

There was 100% agreement on the grading of the written examination. This criterion has been met.

Additionally, if three or more of the following are applicable to a requalification program, then that program shall be determined to be UNSATISFACTORY. If one or two of the following are applicable, the program may be determined UNSATISFACTORY.

a. The same common JPM is missed by at least 50% of the operators.

The maximum percentage of examinees missing any of the common JPMs was 33%. This criterion has been met.

b. The same question about the same common JPM is missed by at least 50% of the examinees.

One question about the same common JPM was missed by 67% of the examinees. This criterion has not been met. This constitutes a potential basis for declaring the program UNSATISFACTORY.

c. The facility failed to train and evaluate operators in all positions permitted by their individual licenses.

The facility had trained and evaluated operators as required. This criterion has been met.

d. Failure to train operators for "in-plant JPMs."

The facility had trained operators for in-plant JPMs. This criterion has been met.

e. Less than 75% of the examinees correctly answer 80% of the common JPM questions.

At least 83% of the examinees correctly answered 80% of the common JPM questions. This criterion has been met.

f. More than one facility evaluator is determined to be unsatisfactory in accordance with "Evaluation of Facility Evaluations." (ES-601)

All facility evaluators were found to be satisfactory. This criterion has been met.

In summary, the facility program did not meet all the program evaluation criteria of ES-601, therefore, the program has been rated UNSATISFACTORY.

3.2 Programmatic Strengths and Weaknesses

a. Programmatic Strengths

- Senior operations department management involvement in evaluating crew performance on the simulator during annual requalification examinations. While this is listed as a strength, operator performance deficiencies should have been identified prior to the examination.

- Sampling plan for determining the content of the written examinations.

b. Programmatic Weaknesses

- Inability of the facility's management to detect and correct requalification program weakness prior to the NRC conducting this requalification examination.

- Failure to adequately train operators on challenging simulator scenarios thereby exercising communications, prioritization, and use of emergency operating procedures.

- Failure to adequately train operators on their individual responsibilities during emergency situations.

- Failure to generate an examination bank of high quality written questions and simulator scenarios.

- Failure to properly time-validate written examination questions and simulator scenarios.

- A general lack of attention-to-detail on the part of the facility training staff, as evidenced by additional deficiencies: the lack of proper examination pre-validation by the facility; failure to promptly implement corrections to the examinations as determined by the NRC during the combined NRC-facility validation; and the administering an RO written examination to an SRO.

4. General Observations

The following is a list of potential problems and procedural inconsistencies noted by the NRC examination team during the conduct of the requalification examination. These observations are expected to be addressed by the facility.

- Procedure N2-OP-101C, (Plant Shutdown), contains a "Note" on page 10 that directs the operator to take specific actions if conditions warrant them. Procedure N2-OP-0, (Operating Procedure Writers Guide), states that "Notes" are not intended to direct operator actions. This inconsistency may be present throughout all the operating procedures.

- Procedure N2-OP-34, (Nuclear Boiler, Automatic Depressurization and Safety Relief Valves), directs the operator to refer to N2-EOP-PC Section SPT if a stuck open relief valve cannot be closed within five minutes or before suppression pool water temperature reaches 110°F. The entry condition for N2-EOP-PC is suppression pool water temperature above 90°F. There is an inconsistency between these two procedures.

- Procedure N2-OP-101C, (Plant Shutdown), section H requires the operator to arm and depress the manual scram switches along with placing the mode switch to the shutdown position if a manual scram is required. During the course of the simulator examination, only once were the manual scram switches depressed. This indicates either an operator knowledge deficiency with the requirements of N2-OP-101C or a procedural conflict with guidance given by other station procedures.

- Procedure N2-OP-78, (Remote Shutdown System), page 9 directs the operator to simply depress the RCIC manual initiation push button to start the RCIC turbine. Procedure N2-OP-35 (Reactor Core Isolation Cooling), page 10 directs the operator to "arm" and depress the push-button for a minimum of 2 seconds. There is an inconsistency between the two procedures.

- Procedure N2-OP-78, (Remote Shutdown System), page 1 describes the Appendix "R" disconnect switches as being located in panels 2CES*PNL415, 2CES*PNL416, and 2CES*PNL417. Page 7 of N2-OP-78, as written, does not consider panel 2CES*PNL417 as having Appendix "R" disconnect switches. This is an inconsistency between the system description and the actual procedure.

- Procedure N2-OP-78, (Remote Shutdown System), page 7 directs the operator to trip all feed pumps by placing the switches on panel 2CES*PNL417 to the "actuate" position. The switches on panel 2CES*PNL417 are not labeled. There is an inconsistency in the position the procedure requires and the actual switch position in the plant.

- Neither of the training key rings needed for performance of JPMs had "MH" keys which are required for entry into certain rooms needed to perform JPMs. This caused excessive delays during the conduct of several JPMs.

- All the vent valves for the charging and withdraw lines for each hydraulic control unit (HCU) have identification markings that are not authorized operator aids.

- The operator aid that is used for procedure N2-OP-36A Section 4, (Boron Injection Utilizing a Hydro Pump) is incorrect. The operator aid has the suction and discharge sides of the pump reversed. The operator aid is located in the "EOP-RQ" box in the CRD maintenance room.

- Procedure EOP-RQ directs the operator to inject boron with the hydro pump in accordance with N2-OP-36A section H.4 if boron cannot be injected with the standby liquid control system. The hydro pump utilizes an uninsulated, non-heat traced suction line from the SLC tank. The suction side of the hydro pump has a "Y" strainer. Since

the flow rate through the system would be very low, the possibility exists for boron to come out of solution in the suction line and clog the suction "Y" strainer. The NRC is concerned that an engineering analysis was not properly performed to validate this method of alternate boron injection.

5. Inspection of Implementation of NRC Bulletin 88-07 and Supplement 1, BWR Power Oscillation - TI 2515/99

5.1 Introduction

An inspection was conducted at the Nine Mile Point Nuclear Station, Unit No. 2 on June 26-29, 1989. The inspector evaluated the licensee's response to and implementation of NRC Bulletin (NRCB) 88-07 and Supplement 1 to this Bulletin. The Bulletin addressed power oscillations in boiling water reactors (BWRs). The licensee's response to the Bulletin and the supplement are contained in Niagara Mohawk Power Corporation letters NMPL 1166, dated September 19, 1988 and NMPL 1190, dated March 6, 1989. Temporary Instruction 2515/99, "Inspection of Licensee's Implementation of Requested Actions of Bulletin 88-07, Power Oscillations," was used to conduct this review.

The following persons provided substantial information during this inspection:

R. Smith, Unit 2 Operations Superintendent
E. Thomlinson, Unit Supervisor Reactor Analyst
G. Weimer, Unit 2 Training Supervisor

5.2 Inspection Details

The inspector reviewed applicable procedures to verify that they provided adequate symptoms of power oscillations, cautions to avoid potentially unstable operating situations, and actions to terminate power oscillations if they do occur. The licensee had previously performed a procedural review and determined that one procedure required revision. Procedure N2-OP-29 (Reactor Recirculation System) was revised in response to NRCB 88-07 Supplement 1. The licensee developed a flow chart for N2-OP-29 that is intended to be the generic procedure for use anytime the situation arises that results in a sudden decrease in reactor recirculation flow. The licensee determined that no new procedures were required.

The inspector determined that the changes made to N2-OP-29 were adequate. However, the inspector noted deficiencies with the licensee's procedure review process as described below:

- Procedures N2-OP-8 (Feedwater Heaters and Extraction Steam Systems), N2-OP-9 (Condenser Air Removal), N2-OP-22D (Generator Hydrogen Seal Oil System), N2-OP-24 (Generator Isolated Phase

Bus Duct Cooling), N2-OP-26 (Generator Stator and Exciter Rectifier Cooling System), N2-OP-42 (Offgas System), N2-OP-68 (Main Generator, Exciter, Main Transformer 345kv Yard, and Generator/Unit Protection), and N2-OP-101C (Plant Shutdown) require power reductions, but these procedures have not been revised to include cautions regarding core power instabilities.

The licensee has committed to conduct at in-depth review of all operating procedures to determine which procedures require power changes that could place the reactor in a potentially unstable operating region of the power-to-flow map. The licensee has committed to making changes to the procedures identified by their in-depth review as needed to either include the appropriate cautions or to refer the operator to another procedure which will contain the appropriate cautions. These changes are to be completed by October 1, 1989. This item will remain unresolved pending completion of the licensee's corrective actions. (UNR 50-410/89-12-01). The inspector had no further questions in the procedural area.

The inspector interviewed eight licensed operators (two of whom were also shift technical advisors) to determine adequate knowledge level of power oscillations and the actions required to mitigate a power oscillation transient. All operators interviewed were knowledgeable of the methods to perform normal power changes to avoid regions of the power-to-flow map where the possibility exists for power oscillation to occur. However, the operators displayed a lack of familiarity with the restart requirements for recirculation pumps while in areas of potential core power instability and the majority of the operators did not know all entry conditions for the N2-OP-29 flow chart. Knowledge of all entry conditions for the N2-OP-29 flow chart is especially important since this is the only procedure the facility revised to include symptoms of power oscillations, cautions to avoid potentially unstable operating situations, and actions to terminate power oscillations if they do occur. The licensee committed to issuing an Operations Department memo for all licensed operators that fully explains when the N2-OP-29 flow chart must be entered. Also, the licensee revised the N2-OP-29 flow chart to highlight the recirculation pump restart requirements so that they are more easily visible. Prior to this revision, the caution for recirculation pump restart was hidden in the text of the N2-OP-29 flow chart and not easily seen.

The inspector reviewed lesson plans for operator training and determined that the training material properly addressed the power oscillation issue as requested by NRCB 88-07. Training was conducted for all licensed operators (including shift technical advisors) in a timely fashion. Training was conducted initially after NRCB 88-07 was issued and then later when Supplement 1 to NRCB 88-07 was issued. The training also includes

simulator training on the use of the N2-OP-29 flow chart. The inspector was concerned that even with all the training conducted, knowledge deficiencies of the operators were discovered during this inspection. As of August 14, 1989, all licensee actions were completed with respect to operator knowledge deficiencies noted.

During the conduct of this inspection, the inspector noted that procedures N2-OP-09, N2-OP-24, and N2-OP-42 direct the operator to reduce power in accordance with N2-OP-101C. N2-OP-101C is the plant shutdown procedure and it does not provide guidance for rapid power reductions. Procedures N2-OP-08, N2-OP-22D, N2-OP-26, and N2-OP-68 also require power reductions but give no guidance at all as to how to reduce reactor power. The licensee has committed to developing a new procedure that is to be used for rapid power reductions. This new procedure is to be developed and approved by October 1, 1989. This item will remain unresolved pending completion of the licensee's corrective actions. (UNR 50-410/89-12-02)

Overall, the licensee did not adequately implement the requested actions of NRC Bulletin 88-07 and Supplement 1. There appeared to be two causes for the inadequacies noted during this inspection: (1) the licensee's review of procedures effected by NRCB 88-07 and Supplement 1 was not comprehensive enough, and (2) the licensee's verification process for ensuring that licensed operators understood the procedure revisions that were made was insufficient.

6. NRC Review of Licensee Letter dated August 10, 1989

6.1 Introduction

The NRC has conducted a review of the Licensee letter from L. Burkhardt to W. Russell dated August 10, 1989. This letter, enclosed as Attachment 6 to this report, contains; (1) the Nine Mile Point Unit 2 Requalification Action Plan, (2) the Nine Mile Point Unit 2 Requalification Exam Remediation Schedule, and (3) Niagara Mohawk Power Corporation's Justification for Continued Operation.

6.2 Review Details

a. Requalification Program Action Plan

NRC has several concerns with the licensee's action plan.

- It does not analyze why NMPC was not able to detect the problems with the operator performance and knowledge deficiencies prior to the NRC conducting the requalification examination.

- It does not consider whether the simulator scenarios were realistic, manageable, and the operators were not properly trained to handle complex emergency situations.
- It does not evaluate the organization interface difficulties that led to the program deficiencies.
- It does not analyze if the training originally conducted was satisfactorily performed and properly evaluated.
- It does not analyze the quality of the training material.
- It does not analyze if there is an attitudinal problem with some operators.
- It does not address the underlying reasons why the performance and knowledge deficiencies exist.
- It does not contain provisions for independent assessment of your training program actions.
- It does not consider whether technical specification changes are required for minimum control room staffing.

b. Requalification Exam Remediation Schedule

The NRC evaluation of the remediation plan will be based on the results of the requalification examination scheduled for the week of September 18, 1989.

c. Niagara Mohawk Power Corporation's Justification for Continued Operation

The NRC staff agrees that the licensee can continue power operations and meet technical specification requirements for operator staffing with augmented four shift rotation in the short term. This conclusion is based primarily on the licensee's augmented shift staffing and their immediate corrective actions, especially with respect to crew communications and interaction. The augmented shift staffing includes operators who either passed the NRC requalification examination, a recent NRC initial licensing examination, or a licensee-administered requalification examination. No individuals, who failed the NRC-administered examination, were returned to a shift. However as detailed above, the underlying reasons for the operator performance deficiencies and knowledge deficiencies do not appear to be understood by the licensee. Additionally, the actions to correct the deficiencies do not address the full scope of the problem.

Overall, the action plan submitted by the licensee is not complete. The licensee has only addressed the problems found during the conduct of the requalification examination. The licensee failed to explain the root causes for these problems and has not included an independent review of the training program to provide an objective analysis.

ATTACHMENT 1

Persons Contacted During Requalification Program Evaluation

1. Niagara Mohawk Power Corporation (NMPC)

L. Burkhardt, Executive Vice President, Nuclear Operators	(3, 4)
J. Willis, General Superintendent, Nuclear Generation	(3, 4)
R. Abbott, Unit 2 Station Superintendent	(4)
M. Peifer, Manager Nuclear Services	(4)
R. Smith, Unit 2 Operations Superintendent	(1, 2, 3, 4)
A. Rivers, Training Superintendent	(1, 3, 4)
R. Seifried, Assistant Training Superintendent	(1, 4)
G. Weimer, Unit 2 Training Supervisor	(1, 2, 3, 4)
R. Sanaker, Unit 1 Training Supervisor	(1, 4)
M. Dooley, Supervisor, Regulatory Compliance	(1)
M. Colomb, Director, Regulatory Compliance	(3, 4)
G. Wilson, Senior Attorney	(3)
W. Davey, Unit 2 Station Shift Supervisor	(1, 3, 5)
S. Dort, Unit 2 Operations Training Instructor	(1, 3, 4, 5)
J. Kaminski, Unit 2 Operations Training Instructor	(4)
K. Cigler, Unit 2 Operations Training Instructor	(4)
R. Brown, Unit 2 Operations Training Instructor	(4)
B. Hennigan, Unit 2 Operations Training Instructor	(4)
D. Haas, Unit 2 Operations Training Instructor	(4)
K. Roenick, NYSPSC, UCMS III	(3)
P. MacEwan, NYSEG	(4)

2. Nuclear Regulatory Commission (NRC)

T. Martin, Deputy Regional Administrator, RI	(3)
B. Boger, Acting Director, DRS, RI	(3)
R. Capra, Project Director, NRR	(3)
R. Gallo, Chief, Operations Branch, DRS	(1, 3)
J. Linville, Acting Director, DRP	(3)
R. Conte, Chief, BWR Section, DRS	(1, 3)
J. Johnson, Chief, Section 2C, DRP	(1)
W. Cook, Senior Resident Inspector, NMP2, DRP	(3, 4)
N. Conicella, Operations Engineer, DRS	(1, 2, 3, 4, 5)
T. Easlick, Operations Engineer, DRS	(1, 2, 3, 4, 5)
D. Florek, Senior Operations Engineer, DRS	(3)
R. Laura, Resident Inspector, NMP2, DRP	(3, 4)
R. Barkley, Reactor Engineer, Section 1B, DRP	(3)

NOTES:

(1) present during entrance meeting on May 30, 1989
(2) present during interim exit meeting on July 21, 1989
(3) present during exit meeting on July 31, 1989
(4) present during supplement exit meeting on August 3, 1989
(5) member of the NRC examination team

ATTACHMENT 2

Simulator Examinations

(cover sheets only)

Scenario Title: SRV Fails Open

Scenario Duration: 1 hour

Scenario Number: 89-01

Revision Number: 1

Course: Licensed Operator Requal

Reviewed By: ______________________ / 7/11/89
Operations Training Supervisor — Date

Reviewed By: ______________________ / 7/11/89
~~Assistant~~ Training Superintendent — Date

Approved By: ______________________ / 7/11/89
Superintendent of Operations — Date

SCENARIO SUMMARY

SRV FAILS OPEN

The scenario begins while shutting down with power about 65%. Normal surveillances to support the shutdown are scheduled. Some equipment is out of service, the B Steam packing exhauster, as well as a typical number of bypassed LPRMs that failed during the cycle.

The A CRD pump trips on overcurrent. The CRD standby pump is quickly started. Two minutes later, flow comparator unit c will fail downscale to give a half scram which will be bypassed and reset. About ten minutes later a non-licensed operator reports an injured worker at the CRD pump station. EPP-4 is entered and an Unusual Event is declared.

A spurious opening of an SRV leads to a manual scram and excessive cool down. Some problems are encountered in establishing suppression pool cooling due to trip of an RHR pump. A small steam leak develops from the SRV tailpiece and causes increased containment pressures, leading the crew to spray the suppression chamber.

SCENARIO OBJECTIVES

The licensed Senior Reactor Operators (SSS and ASSS):

1. Direct emergency response as site emergency director.
2. Direct corrective actions to mitigate the consequences of the emergency event.
3. Direct corrective actions to mitigate the consequences of the off normal event.
4. Classify emergency events requiring emergency plan implementation.
5. Analyze indications to determine that an emergency plan event is in progress.
6. Direct the actions required for a reactor scram.
7. Direct actions for emergency medical response.
8. Direct actions for high containment pressure.

The licensed control room Reactor Operators (CSO and NAOE):

1. Attempt to close a stuck open safety/relief valve.
2. Scram the reactor manually and take immediate actions.
3. Perform actions for a safety relief valve opening.
4. Perform actions for a loss of CRD pump(s) during plant operation.
5. Perform post scram recovery actions IAW N2-OP-101C.
6. Perform the duties of the CSO when notified of an injured and contaminated person in the plant.
7. Startup suppression pool cooling mode and monitor for proper operation from the control room and shutdown.
8. Perform actions for high containment pressure.
9. Make verbal reports to immediate supervisor and other supervisors.

(*) Individual Simulator Critical Task

(**) Crew Simulator Critical Task

Scenario Title: ATWS AND SCRAM DISCHARGE VOLUME LEAK OUTSIDE PRIMARY CONTAINMENT WITH FUEL FAILURE

Scenario Duration: 1 hour

Scenario Number: 89-09

Revision Number: 1

Course: Licensed Operator Requal

Reviewed By: R. L. Weems / 7/11/89
Operations Training Supervisor / Date

Reviewed By: R. T. Seifried / 7/11/89
Assistant Training Superintendent / Date

Approved By: R. G. Smith / 7/11/89
Superintendent of Operations / Date

SCENARIO SUMMARY

ATWS AND SCRAM DICHARGE VOLUME LEAK OUTSIDE PRIMARY CONTAINMENT WITH FUEL FAILURE

While operating at 100%.power the 300A battery bus fails. Operators should take action to restore power to the 300A battery bus.

The EHC control system breaks into divergent oscillations. Fuel failure is experienced as power oscillations become more severe. A failure of the RPS to cause an immediate scram, and a scram discharge volume leak in the Reactor Building present the potential for offsite release of radioactivity.

With RCIC out of service, the operators must use SRVs and steam condensing to reduce and control reactor pressure. They should decrease pressure to reduce the coolant flow from the leak, but not exceed the cooldown rate limit.

Standby gas treatment should be initiated to process containment air before releasing it to the atmosphere. An ALERT should be declared and emergency notifications made.

SCENARIO OBJECTIVES

The Licensed Senior Reactor Operators (SSS and ASSS):

1. Direct emergency response as site emergency director.
2. Direct actions as required per EOP-RQ.
3. Direct corrective actions to mitigate the consequences of the off normal event.
4. Direct actions as required per EOP-RL.
5. Direct the actions required for a loss of electrical power.
6. Classify emergency events requiring emergency plan implementation.
7. Determine if indications of fuel failure are present.
8. Direct the actions required for a malfunction in the reactor pressure control system (EHC).
9. Perform required notifications of on-site and off-site personnel for emergency events.
10. Direct the actions required for a reactor scram.

The Licensed Control Room Operators (CSO and NAOE):

1. Startup and shutdown the steam condensing mode of RHR and monitor for proper operation from the Control Room.
2. Perform immediate actions for reactor scram.
3. Perform actions for a loss of DC power.
4. Conduct ADS/SRV manual valve operation and monitor indication.
5. Perform actions for an EHC failure.
6. Perform post scram recovery actions IAW N2-OP-101C.
7. Operate the RPS power supply transfer switch in the Control Room.
8. Make verbal reports to immediate supervisor and other supervisors.

(*) Individual simulator critical task.

(**) Crew simulator critical task.

Scenario Title: INADVERTENT START OF HPCS FOLLOWED BY TURBINE TRIP WITHOUT BYPASS

Scenario Duration: 1 hour

Scenario Number: 89-08

Revision Number: 1

Course: Licensed Operator Requal

Reviewed By: D. Werner / 7/11/89
Operations Training Supervisor / Date

Reviewed By: [signature] / 7/11/89
~~Assistant~~ Training Superintendent / Date

Approved By: R. H. Smith / 7/11/89
Superintendent of Operations / Date

SCENARIO SUMMARY

INADVERTENT HPCS FOLLOWED BY TURBINE TRIP WITH STUCK RODS

While at 100% power, a control rod drifts in, the shift must coordinate activities and seek the help of the reactor analyst.

Shortly after the rod is valved out, HPCS initiates with full rated flow to the vessel. Power and level momentarily increase, but the transient can be "ridden out". The shift should take steps to secure HPCS, place it out of service, enter appropriate LCOs and commence investigative repair efforts.

While HPCS is being secured, the feedwater master controller fails "as is" leading to a low level condition. From this point on, the operators will have to control level manually.

Later, EHC pressure regulators fail low combined with a failure of bypass valves to open. Shortly afterwards all feed pumps are lost. The Reactor scrams and the level drops quickly. The RCIC system fails to initiate on low level and must be started manually. During the scram some rods fail to insert, but power will drop below 4%. The operators will be in the EOPs to restore level and insert the control rods.

SCENARIO OBJECTIVES

The Licensed Senior Reactor Operators (SSS and ASSS):

1. Direct emergency response as site emergency director.
2. Direct corrective actions to mitigate the consequences of the emergency event.
3. Direct actions as required per EOP-RP.
4. Direct actions as required per EOP-RL.
5. Classify emergency events requiring emergency plan implementation.
6. Analyze indications to determine that an emergency plan event is in progress.
7. Direct the use of suppression pool cooling per the EOPs.
8. Perform required notifications of on-site and off-site personnel for emergency events.
9. Direct the actions required for a reactor scram.
10. Clarify Technical Specifications and application of action statement requirements.

The Licensed Control Room Operators (CSO and NAOE):

1. Scram the reactor manually and take immediate actions.
2. Control RPV pressure using the RCIC System.
3. Conduct ADS/SRV manual valve operation and monitor indication.
4. Perform actions in response to a control rod drift.
5. Return the HPCS System to standby after automatic initiation.
6. Perform post scram recovery actions IAW N2-OP-101C.
7. Startup suppression pool cooling mode and monitor for proper operation from the Control Room and shutdown.
8. Make verbal reports to immediate supervisor and other supervisors.

(*) Individual simulator critical task.
(**) Crew simulator critical task.

Scenario Title: MAIN STEAM LINE BREAK INSIDE CONTAINMENT

Scenario Duration: 1 hour

Scenario Number: 89-10

Revision Number: 1

Course: Licensed Operator Requal

Reviewed By: [signature] / 7/11/89
Operations Training Supervisor — Date

Reviewed By: R.T. Seifried / 7/11/89
Assistant Training Superintendent — Date

Approved By: R.P. Smith / 7/11/89
Superintendent of Operations — Date

3

SCENARIO SUMMARY

MAIN STEAM LINE BREAK INSIDE CONTAINMENT

The scenario begins with the shift crew maintaining 100% power when the EHC pressure regulator fails low. Reactor pressure and power increase to noticeable values and the crew should be able to quickly diagnose and correct the problem. The standby pressure regulator gains control automatically to limit the transient.

The 4B breaker for the recirculation pump opens due to human error caused by in plant maintenance.

The B recirculation pump trip causes the shift to enter the recently revised OP-29 to react to this off normal event. The pump is eventually restored to operation and allows observation of normal activities.

A sudden loss of electrical load caused by a fault on the grid provides the scram signal to start the emergency evolution. The scram is coupled with a steam line break inside the containment. Emergency actions are hampered by degraded ECCS; the LPCS injection valve fails to open, the B RHR pump trips, and the HPCS pump is marked up out of service. The shift crew is forced to make decisions about priority use of the remaining systems for Rx level control, suppression pool cooling and the spray mode.

The scenario is terminated when the reactor level is recovered and drywell pressure has been reduced.

SCENARIO OBJECTIVES

The Licensed Senior Reactor Operators (SSS and ASSS):

1. Direct the actions required for a large break LOCA inside the containment.
2. Direct emergency response as Site Emergency Director.
3. Direct actions as required per EOP-PCP.
4. Direct actions as required per EOP-SPT.
5. Direct actions as required per EOP-RL.
6. Direct the use of drywell spray per the EOPs.
7. Classify emergency events requiring emergency plan implementation.
8. Direct the use of suppression chamber spray per the EOPs.
9. Direct the use of suppression pool cooling per the EOPs.
10. Perform required notifications of on-site and off-site personnel for emergency events.

The Licensed Control Room Operators (CSO and NAOE):

1. Perform actions required for a large break LOCA inside the primary containment.
2. Scram the reactor manually and take immediate actions.
3. Manually initiate and shutdown the containment/drywell spray mode and monitor for proper operation from the Control Room.
4. Perform actions required for one recirculation pump trip.
5. Restart a tripped recirculation pump from single loop operation.
6. Perform post scram recovery actions IAW N2-OP-101C.
7. Monitor the automatic operation of the LPCS System from the Control Room.
8. Manually initiate and shutdown the suppression pool spray mode and monitor for proper operation from the Control Room.
9. Startup suppression pool cooling mode and monitor for proper operation from the Control Room and shutdown.
10. Make verbal reports to immediate supervisor and other supervisors.

(*) Individual simulator critical task.
(**) Crew simulator critical task.

ATTACHMENT 3

Job Performance Measures

(cover sheets only)

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Title: Determine Containment Water Level   Revision: 1

Task Number: 2000230501
2239410101

Operator:______________________(RO/SRO) Evaluator:__________

Directions to operators:

When I tell you to begin you are to determine containment water level as directed by EOP-Primary Containment Control. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: ✓ Perform ______ Simulate

Evaluation Location: ______ Plant ✓ Simulator ______ Control Room

Average Completion Time: 15 minutes   Actual Completion Time: ______

JPM Overall Rating: Sat/Unsat Questions: # Asked ______ # Correct ______

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:______________________ Date:______________

Approvals: 7/11/89
Training Supervisor - Unit 2

7/11/89
~~Asst.~~ Supt. - Training

Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Title: Manually Insert a Half Scram Revision: 1

Task Number: 2120020101

Operator:______________________________(RO/SRO) Evaluator:______________

Directions to operators:

When I tell you to begin you are to Manually Insert a Half Scram. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: ___✓___ Perform ~~Plant~~ _______ Simulate

Evaluation Location: _______ Plant ___✓___ Simulator _______ Control Room

Average Completion Time: 10 minutes Actual Completion Time: _______

JPM Overall Rating: Sat/Unsat Questions: # Asked _______ # Correct _______

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:_________________________ Date:_________________

Approvals: [signature] 7/11/89
Training Supervisor - Unit 2

[signature] for 7/11/89
~~Asst.~~ Supt. - Training

[signature]
Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Title: MSIV Equalizing/Main Steam Line Warmup Revision: 1

Task Number: 2390030101

Operator:______________________________(RO/SRO) Evaluator:______________

Directions to operators:

When I tell you to begin you are to perform a MSIV Equalizing/Main Steam Line Warmup through the point of opening the MSIVs and verifying proper open indications. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: ___✓___ Perform _______ Simulate

Evaluation Location: _______ Plant ___✓___ Simulator _______ Control Room

Average Completion Time: 21 Minutes Actual Completion Time: _______

JPM Overall Rating: Sat/Unsat Questions: # Asked _______ # Correct _______

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:_____________________________ Date:__________________

Approvals: [signature] 7/11/89
Training Supervisor - Unit 2

[signature] 7/11/89
~~Asst.~~ Supt. - Training

[signature]
Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Title: LPCI Shutdown Following Initiation   Revision: 1

Task Number: 2030020101

Operator:______________________________(RO/SRO) Evaluator:______________

Directions to operators:

When I tell you to begin you are to shutdown the A loop of LPCI to the Standby condition. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: ✓ Perform _______ Simulate

Evaluation Location: _______ Plant ✓ Simulator _______ Control Room

Average Completion Time: 10 minutes   Actual Completion Time: _______

JPM Overall Rating: Sat/Unsat   Questions: # Asked _______ # Correct _______

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:__________________________ Date:________________

Approvals: [signature] 7/11/89
Training Supervisor - Unit 2

[signature] 7/11/89
~~Asst.~~ Supt. - Training

[signature: R F Smith]
Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Title: Manually Initiate SLC System Revision: 1

Task Number: 2000250501

Operator:______________________________(RO/SRO) Evaluator:______________

Directions to operators:

When I tell you to begin you are to manually initiate the SLC System through the point of verifying power decreasing on APRMs. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: ✓ Perform ______ Simulate

Evaluation Location: ______ Plant ✓ ~~Simulate~~ Simulator ______ Control Room

Average Completion Time: 15 minutes Actual Completion Time: ______

JPM Overall Rating: Sat/Unsat Questions: # Asked______ # Correct______

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:__________________________ Date:________________

Approvals: [signature] 7/11/89
Training Supervisor - Unit 2

R. T. [signature] 7/11/89
Asst. Supt. - Training

[signature]
Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Title: Unload and Secure 2EGS*EG1 Revision: 1

Task Number: 2649060101

Operator:____________________(RO/SRO) Evaluator:____________

Directions to operators:

When I tell you to begin you are to unload and secure 2EGS*EG1. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: ✓ Perform ______ Simulate

Evaluation Location: ______ Plant ✓ Simulator ______ Control Room

Average Completion Time: 14 minutes Actual Completion Time: ______

JPM Overall Rating: Sat/Unsat Questions: # Asked______ # Correct______

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:____________________ Date:____________

Approvals: 7/11/89
Training Supervisor - Unit 2

7/11/89
~~Asst.~~ Supt. - Training

Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Title: Local Startup of Diesel Generator 2EGS*EG3 Revision: 1

Task Number: 2640020101
2649260401

Operator:________________________________(RO/SRO) Evaluator:______________

Directions to operators:

When I tell you to begin you are to locally start Diesel Generator 2EGS*EG3 through the point of verifying cooling water flow to the diesel. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: _______ Perform ___✓___ Simulate

Evaluation Location: ___✓___ Plant _______ Simulator _______ Control Room

Average Completion Time: 15 minutes Actual Completion Time: _______

JPM Overall Rating: Sat/Unsat Questions: # Asked_______ # Correct_______

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:___________________________ Date:_________________

Approvals: [signature] 7/11/89
Training Supervisor - Unit 2

[signature] for 7/11/89
~~Asst.~~ Supt. - Training

[signature]
Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Revision: 1

Title: Manually Insert Control Rod 30-19 By Venting the Withdraw Line

Task Number: 2000200501
2000290501R

Operator:____________________(RO/SRO) Evaluator:____________

Directions to operators:

When I tell you to begin you are to manually insert Control Rod 30-19 by venting the withdraw line until the rod is completely inserted and the lineup is isolated. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: _____ Perform ✓ Simulate

Evaluation Location: ✓ Plant _____ Simulator _____ Control Room

Average Completion Time: 8 minutes Actual Completion Time: _____

JPM Overall Rating: Sat/Unsat Questions: # Asked _____ # Correct _____

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:____________________ Date:______________

Approvals: [signature] 7/11/89
Training Supervisor - Unit 2

[signature] 7/11/89
~~Asst.~~ Supt. - Training

[signature]
Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Title: Isolate and Vent the Scram Air Header Revision: 1

Task Number: 2000200501

Operator:______________________(RO/SRO) Evaluator:______________

Directions to operators:

When I tell you to begin you are to Isolate and Vent the Scram Air Header using manual valve operation. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: _______ Perform ✓ Simulate

Evaluation Location: ✓ Plant _______ Simulate _______ Control Room

Average Completion Time: 10 minutes Actual Completion Time: _______

JPM Overall Rating: Sat/Unsat Questions: # Asked_______ # Correct_______

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:______________________ Date:______________

Approvals: [signature] 7/11/89
Training Supervisor - Unit 2

[signature] 7/11/89
~~Asst.~~ Supt. - Training

[signature]
Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Revision: 1

Title: Manual Initiation of RCIC from the Remote Shutdown Panel

Task Number: 2969040101

Operator:________________(RO/SRO) Evaluator:________

Directions to operators:

When I tell you to begin you are to manually initiate RCIC at the Remote Shutdown Panel through the point of establishing 600 gpm flow and verifying that Reactor Water level is increasing. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: ____ Perform ✓ Simulate

Evaluation Location: ✓ Plant ____ Simulator ____ Control Room

Average Completion Time: 6 minutes Actual Completion Time: ____

JPM Overall Rating: Sat/Unsat Questions: # Asked____ # Correct____

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:________________ Date:________

Approvals: [signature] 7/11/89
Training Supervisor - Unit 2

[signature] 7/11/89
~~Asst~~. Supt. - Training

[signature: R A Smith]
Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Title: Turbine Valve Testing Revision: 1

Task Number: 2129050101

Operator:______________________(RO/SRO) Evaluator:______________

Directions to operators:

When I tell you to begin you are to perform a test of Turbine Control Valve CV-1 up to the point of being ready to test control valve CV-2. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: ✓ Perform ______ Simulate

Evaluation Location: ______ Plant ✓ Simulator ______ Control Room

Average Completion Time: 10 minutes Actual Completion Time: ______

JPM Overall Rating: Sat/Unsat Questions: # Asked ______ # Correct ______

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:______________________ Date:______________

Approvals: [signature] 7/11/89
Training Supervisor - Unit 2

[signature] 7/11/89
~~Asst.~~ Supt. - Training

[signature]
Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Revision: 1

Title: Transfer the Reactor Recirculation Pumps to High Speed

Task Number: 2029090101

Operator:______________________________(RO/SRO) Evaluator:______________

Directions to operators:

When I tell you to begin you are to shift the recirculation pumps to high. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: ✓ Perform ______ Simulate

Evaluation Location: ______ Plant ✓ Simulator ______ Control Room

Average Completion Time: 20 Minutes Actual Completion Time: ______

JPM Overall Rating: Sat/Unsat Questions: # Asked ______ # Correct ______

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:_________________________ Date:________________

Approvals: RWerner 7/11/89
Training Supervisor - Unit 2

~~Asst.~~ Supt. - Training 7/11/89

Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Title: Slow Closure of MSIV 7B Revision: 1

Task Number: 2399140101

Operator:____________________(RO/SRO) Evaluator:____________

Directions to operators:

When I tell you to begin you are to slow close MSIV 7B. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: ✓ Perform ______ Simulate

Evaluation Location: ______ Plant ✓ Simulator ______ Control Room

Average Completion Time: 6 Minutes Actual Completion Time: ______

JPM Overall Rating: Sat/Unsat Questions: # Asked ______ # Correct ______

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:____________________ Date:____________

Approvals: [signature] 7/11/89
Training Supervisor - Unit 2

[signature] 7/11/89
~~Asst.~~ Supt. - Training

[signature]
Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Title: Manual Start of 2EGS*EG1 from Panel 852 Revision: 1

Task Number: 2640020101
2640030101
2649030101

Operator:______________________________(RO/SRO) Evaluator:______________

Directions to operators:

When I tell you to begin you are to perform a remote manual start of 2EGS*EG1 through the point of loading the DG to 2200 KW and 900 KVAR to the bus. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: ✓ Perform ______ Simulate

Evaluation Location: ______ Plant ✓ Simulator ______ Control Room

Average Completion Time: 20 Minutes Actual Completion Time: ______

JPM Overall Rating: Sat/Unsat Questions: # Asked ______ # Correct ______

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:__________________________ Date:________________

Approvals: 7/11/89
Training Supervisor - Unit 2

7/11/89
~~Asst~~. Supt. - Training

Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Title: Lower Suppression Pool Level Revision: 1

Task Number: 2000230501
2059360101

Operator:______________________(RO/SRO) Evaluator:______________

Directions to operators:

When I tell you to begin you are to lower the suppression pool water level via the radwaste system by approximately one-half foot. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: ✓ Perform ______ Simulate

Evaluation Location: ______ Plant ✓ Simulator ______ Control Room

Average Completion Time: 15 minutes Actual Completion Time: ______

JPM Overall Rating: Sat/Unsat Questions: # Asked______ # Correct______

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:______________________ Date:______________

Approvals: [signature] 7/11/89
Training Supervisor - Unit 2

[signature] 7/11/89
~~Asst~~. Supt. - Training

[signature]
Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Title: Recouple an Uncoupled Control Rod Revision: 1

Task Number: 2019100401

Operator:______________________(RO/SRO) Evaluator:____________

Directions to operators:

When I tell you to begin you are to recouple control rod _-_ (write in rod number) and verify the rod is recoupled. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: ✓ Perform ______ Simulate

Evaluation Location: ______ Plant ✓ Simulator ______ Control Room

Average Completion Time: 15 minutes Actual Completion Time: ______

JPM Overall Rating: Sat/Unsat Questions: # Asked ______ # Correct ______

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:____________________ Date:____________

Approvals: [signature] 7/11/89
Training Supervisor - Unit 2

[signature] 7/11/89
~~Asst.~~ Supt. - Training

[signature]
Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Revision: 1

Title: Isolate a CRD HCU 26-47 with the exception of Cooling Water Flow

Task Number: 2000360401

Operator:______________________________(RO/SRO) Evaluator:______________

Directions to operators:

When I tell you to begin you are to Isolate a CRD HCU with exception of cooling water flow. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: ______ Perform ✓ Simulate

Evaluation Location: ✓ Plant ______ Simulator ______ Control Room

Average Completion Time: 7 minutes Actual Completion Time: ______

JPM Overall Rating: Sat/Unsat Questions: # Asked ______ # Correct ______

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:__________________________ Date:________________

Approvals: DWerner 7/11/89
Training Supervisor - Unit 2

[signature] 7/11/89
~~Asst.~~ Supt. - Training

R G Smith
Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Title: Inject Boron Using the SLC Hydro Pump Revision: 1

Task Number: 2119100401

Operator:________________(RO/SRO) Evaluator:__________

Directions to operators:

When I tell you to begin you are to Inject Boron using the SLC Hydro Pump through the point where the hydro pump is running. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: _____ Perform ✓ Simulate

Evaluation Location: ✓ Plant _____ Simulator _____ Control Room

Average Completion Time: 15 minutes Actual Completion Time: _____

JPM Overall Rating: Sat/Unsat Questions: # Asked _____ # Correct _____

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:________________ Date:__________

Approvals: [signature] 7/11/89
Training Supervisor - Unit 2

R. T. [signature] 7/11/89
Asst. Supt. - Training

[signature]
Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Title: Actuation of the Appendix R Disconnects Revision: 1

Task Number: 2969040101

Operator:______________________________(RO/SRO) Evaluator:______________

Directions to operators:

When I tell you to begin you are to actuate the Appendix R disconnect switches as CSO. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: ______ Perform ✓ Simulate

Evaluation Location: ✓ Plant ______ Simulator ______ Control Room

Average Completion Time: 5 minutes Actual Completion Time: ______

JPM Overall Rating: Sat/Unsat Questions: #Asked ______ # Correct ______

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:______________________ Date:________________

Approvals: [signature] 7/11/89
Training Supervisor - Unit 2

[signature] 7/11/89
~~Asst.~~ Supt. - Training

[signature]
Supt. of Operations - Unit 2

NIAGARA MOHAWK POWER CORPORATION

LICENSED OPERATOR JOB PERFORMANCE MEASURE

Title: Local Manual Reactor Scram Revision: 1

Task Number: 2969050101

Operator:______________________(RO/SRO) Evaluator:__________

Directions to operators:

When I tell you to begin you are to perform a Local Manual Reactor Scram. I will describe general conditions and provide you access to the tools to complete this task. Before you start, I will state the task standards and initiation cues and answer any questions.

Evaluation Method: _______ Perform ✓ Simulate

Evaluation Location: ✓ Plant _______ Simulate _______ Control Room

Average Completion Time: 4 minutes Actual Completion Time: _______

JPM Overall Rating: Sat/Unsat Questions: # Asked_______ # Correct_______

Comments: (Note: Any grade of Unsat requires a comment. A JPM overall rating of UNSAT shall be given if any critical step is graded as unsat. If all critical steps are performed satisfactorily and the Task Standards met, a JPM overall rating of SAT shall be given.)

Evaluators Signature:______________________ Date:______________

Approvals: [signature] 7/11/89
Training Supervisor - Unit 2

R. T. Dupree 7/11/89
Asst. Supt. - Training

[signature]
Supt. of Operations - Unit 2

ATTACHMENT 4

Written Examination and Answer Key

Part A (RO and SRO)

STATIC SIMULATOR SCENARIO

A-5

Scenario Title: Turbine EHC Pressure Regulator Failure without Bypass Valves

Exam Duration: 45 Minutes

Scenario Number: 89-01

Revision Number: 1

Course: Licensed Operator Requal

Reviewed By: ______________________ / 7/11/89
Operations Training Supervisor — Date

Reviewed By: R. T. ______________________ / 7/11/89
Assistant Training Superintendent — Date

Approved By: ______________________ / 7/11/89
Superintendent of Operations — Date

SCENARIO DESCRIPTION

A-5

Scenario Number: 1

Type of Scenario: Abnormal Status

Title: Turbine EHC Pressure Regulator Failure Without Bypass Valves

Synopsis: The reactor is operating in a normal configuration when the main turbine trips. Following the trip the turbine bypass valves fail to open. Reactor pressure increases due to decay and residual heat causing the safety relief valves to open.

The operators should be concerned with the immediate actions for scram and turbine trip. The SROs should be planning for long term cooling of the reactor.

Malfunctions: NM10 E; Intermediate Range Monitor Channel E Detector Stuck
TC02; Pressure Regulator Fails Low
TC06; All Bypass Valves Closed

Objectives:

A. Identify a failure of the in-service pressure regulator by interpreting instrument response.

B. Locate and correctly use the following: N2-OP-23 (Electro Hydraulic Control).

C. Perform necessary system checks to ensure correct automatic system response to a pressure regulator failure.

D. Given the Operating Procedure, identify precautions, off-normal procedures and normal procedure notes and cautions which apply to system operation.

Name: ______________________________

Soc. Sec. #: ______________

Course: U-2 Annual RO Requal Exam Section A-5

Date: ______________

Grade: ______________

File Section: ______________

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e) Reviewed or attempted to review materials that are un-authorized, including the examination prior to implementation, the examination answer key, or the answers developed by any other trainee during the examination.

I have read and understand the above.

Signature: ______________________________

Date: ______________

1. QUESTION: (1.0 pts.)

1-C-1. What RPS trip signal caused the reactor to scram? (0.5)

What indication (source) did you use to determine the trip signal? (0.5)

ANSWER:

1-C-1. Reactor vessel high pressure. (0.5)

Sequence of events typer. (0.5)

2. QUESTION: (1.0 pts.)

1-C-3. Once the scram occurred, what caused reactor pressure to decrease to its low value? What caused the pressure decrease to stop?

ANSWER:

1-C-3. The addition of cold feedwater (Theoretical concept). (0.5) The pressure decrease stopped when feedwater addition stopped (concept). (0.5)

3. QUESTION: (1.5 pts.)

1-C-4. a. What signal(s) could have caused the recirculation pumps transfer to low speed? State any setpoint. (1.0)

b. In this scenario, which signal actually caused the recirc pumps to transfer to low speed? (0.5)

ANSWER:

1-C-4. a. RRCS Reactor High Pressure (0.1) >1050 psig. (0.1)
EOC-RPT (0.1) a turbine trip when the reactor power is greater than 30%. (0.1)

Feedwater low flow (0.1) < 30% (0.1)
Steam Dome to pump suction ΔT (0.1) < 10.7°F (0.1)
Low reactor water level (0.1) 159.3 inches (0.1)

b. RRCS high pressure (0.5)

4. QUESTION: (2.0 pts.)

1-C-5. What actions should be taken with regard to the Reactor Water Cleanup System? What is the purpose of these actions? (2.0)

ANSWER:

1-C-5. 1. Transfer RWCU to full reject to main condenser (per N2-OP-37, Section F.6.0) or

2. Manually trip the RWCU pumps and close pump discharge valves (2WCS-V30A and B) (either answer 1 or 2 acceptable for full credit) (1.0)

These actions are taken to prevent feedwater stratification. (1.0)

5. QUESTION: (1.0 pts.)

1-C-6. What two signals could have caused recirculation loop flow control to be in its present mode?

ANSWER:

1-C-6. (Flow control is in Manual because) the recirculation pumps have transferred to their LFMG sets. (0.5)

or

Because the recirc flux controller output is abnormal (0.5)

6. QUESTION: (1.25 pts.)

1-C-7. Explain the impact had the "B" EHC pressure regulator <u>not</u> failed in this scenario. (1.25)

ANSWER:

1-C-7. The pressure regulator would have automatically transferred to B, (the lowest value) (.75), and plant operation would have continued (with no change in power level). (.50)

7. QUESTION: (0.75 pts.)

1-R-1. With no further operator action, describe the expected response of reactor pressure.

Include any automatic action(s) and setpoint(s) in your answer. (0.75)

ANSWER:

1-R-1. (With the bypass valves closed,) reactor pressure will increase causing the SRVs to cycle to control pressure. The lowest relief setpoint for the SRVs is 1076 psig. (0.75)

8. QUESTION: (0.75 pts.)

1-R-2. An attempt to transfer pressure regulators fails. What other action could be taken to use the turbine EHC System to control reactor pressure? (0.75)

ANSWER:

1-R-2. The BYPASS JACK could be used to open the bypass valves. (0.75)

9. QUESTION: (1.0 pts.)

1-R-3. a. One of the scram immediate actions is to insert SRM's/IRM's. State whether this has been carried out completely.

b. Support your answer with two Control Room panel indications.

ANSWER:

1-R-3. a. No, IRM E has failed to insert. (0.50)

b. Accept any two (2) of the following (0.25 points each):

1. SRM/IRM detector position lights (IRM E, P603)
2. IRM E reading lower than others (C51-R603A and B)
3. IRM E on range 1 (but still downscale)

10. QUESTION: (0.5 pts.)

1-R-4: What automatic action has occurred (other than scram) to prevent an ATWS. Give setpoints.

ANSWER:

1-R-4: Alternate Rod Insertion (0.25)
Reactor Pressure >1050 psig (0.25)

Name: ______________________

Soc. Sec. #: ______________

Course: U-2 Annual SRO Requal Exam Section A-5

Date: ______________

Grade: ______________

File Section: ______________

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e) Reviewed or attempted to review materials that are un-authorized, including the examination prior to implementation, the examination answer key, or the answers developed by any other trainee during the examination.

I have read and understand the above.

Signature: ______________________

Date: ______________

1. QUESTION: (1.0 pts.)

1-C-1. What RPS trip signal caused the reactor to scram? (0.5)

What indication (source) did you use to determine the trip signal? (0.5)

ANSWER:

1-C-1. Reactor vessel high pressure. (0.5)

Sequence of events typer. (0.5)

2. QUESTION: (1.0 pts.)

1-C-3. Once the scram occurred, what caused reactor pressure to decrease to its low value? What caused the pressure decrease to stop?

ANSWER:

1-C-3. The addition of cold feedwater (Theoretical concept). (0.5) The pressure decrease stopped when feedwater addition stopped (concept). (0.5)

3. QUESTION: (1.5 pts.)

1-C-4. a. What signal(s) could have caused the recirculation pumps transfer to low speed? State any setpoint. (1.0)

b. In this scenario, which signal actually caused the recirc pumps to transfer to low speed? (0.5)

ANSWER:

1-C-4. a. RRCS Reactor High Pressure (0.1) >1050 psig. (0.1)
EOC-RPT (0.1) a turbine trip when the reactor power is greater than 30%. (0.1)

Feedwater low flow (0.1) < 30% (0.1)
Steam Dome to pump suction ΔT (0.1) < 10.7°F (0.1)
Low reactor water level (0.1) 159.3 inches (0.1)

b. RRCS high pressure (0.5)

4. QUESTION: (2.0 pts.)

1-C-5. What actions should be taken with regard to the Reactor Water Cleanup System? What is the purpose of these actions? (2.0)

ANSWER:

1-C-5. 1. Transfer RWCU to full reject to main condenser (per N2-OP-37, Section F.6.0) or

2. Manually trip the RWCU pumps and close pump discharge valves (2WCS-V30A and B) (either answer 1 or 2 acceptable for full credit) (1.0)

These actions are taken to prevent feedwater stratification. (1.0)

5. QUESTION: (1.0 pts.)

1-C-6. What two signals could have caused recirculation loop flow control to be in its present mode?

ANSWER:

1-C-6. (Flow control is in Manual because) the recirculation pumps have transferred to their LFMG sets. (0.5)

or

Because the recirc flux controller output is abnormal (0.5)

6. QUESTION: (1.25 pts.)

1-C-7. Explain the impact had the "B" EHC pressure regulator not failed in this scenario. (1.25)

ANSWER:

1-C-7. The pressure regulator would have automatically transferred to B, (the lowest value) (.75), and plant operation would have continued (with no change in power level). (.50)

7. QUESTION: (1.0 pts.)

1-S-1. a. With no further operator action, describe the expected response of reactor pressure.

Include any automatic action(s) and setpoint(s) in your answer. (0.75)

b. By procedure, what maximum pressure should you direct the operator to maintain? (0.25)

ANSWER:

1-S-1. a. (With the bypass valves closed,) reactor pressure will increase causing the SRVs to cycle to control pressure. The lowest relief setpoint for the SRVs is 1076 psig. (0.75)

b. 1076 psig (0.25)

8. QUESTION: (1.0 pts.)

1-S-2. Assuming that sustained high pressure causes relief valve cycling, how would the SSS direct this problem be resolved?

ANSWER:

1-S-2. He would direct the reactor operator to manually open enough Safety Relief Valves to drop reactor pressure to 960 psig. (1.0)

9. QUESTION: (1.0 pts.)

1-S-3. If the control room had to be evacuated and the RSP manned, how should the SSS direct reactor pressure be controlled. (Include control band, if applicable) (1.0)

ANSWER:

1-S-3. When reactor pressure increases to greater than 1050 psig, (0.25) reduce reactor pressure to approximately 950 psig (0.25) by cycling a SRV from either Division I or Division II Remote Shutdown Panel. (0.50)

STATIC SIMULATOR SCENARIO

A-6

Scenario Title: Loss of 13.8 Kv Bus (SWG-403)

Exam Duration: 45 Minutes

Scenario Number: 89-02

Revision Number: 1

Course: Licensed Operator Requal

Reviewed By: ______ 12/11/89
Operations Training Supervisor — Date

Reviewed By: R. T. ______ 12/11/89
Assistant Training Superintendent — Date

Approved By: ______ 12/11/89
Superintendent of Operations — Date

SCENARIO DESCRIPTION

A-6

Scenario Number: 2

Type of Scenario: Abnormal Status

Title: Loss of 13.8 KV Bus (SWG-003)

Synopsis: The plant is operating at normal 100% power when the 13.8 KV Bus 3 is lost. The C TBCLC pump is prevented from automatically starting when one of the running pumps is lost with the bus. No action is taken except to silence alarms. The problem is allowed to run long enough to get a recirculation system runback signal from the loss of a feedwater pump (from the bus trip) combined with a reactor water signal (level 4).

Malfunctions: ED03C: 13.8 KV Bus Fault (2NPS-SWG-003)
CW03C: Turbine Building Closed Loop Cooling Water System Pump C Trip

Objectives:

A. Locate and correctly use NMP-2 Operating Procedures.

B. Locate and correctly use NMP-2 Technical Specifications

C. Identify normal and/or normal trends utilizing control board indications.

Name: ______________________________

Soc. Sec. #: ______________

Course: U-2 Annual RO Requal Exam Section A-6

Date: ______________

Grade: ______________

File Section: ______________

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e) Reviewed or attempted to review materials that are un-authorized, including the examination prior to implementation, the examination answer key, or the answers developed by any other trainee during the examination.

I have read and understand the above.

Signature: ______________________________

Date: ______________

1. QUESTION: (1.0 pts.)

2-C-1. How would the present condition of the Turbine Building Closed Loop Cooling System affect the plants ability to maintain 100% power.

ANSWER:

2-C-1. TBCLCS load temperatures will rise (TBCLCS heat removal capacity is reduced) (0.5). This would cause plant capacity to be reduced (concept) (0.5).

2. QUESTION: (0.75 pts.)

2-C-2. Explain how the B pilot scram valve solenoids were de-energized without an RPS trip (scram) signal.

ANSWER:

2-C-2. B RPS MG was de-energized. (0.75) (B RPS MG powers solenoids).

Note: Partial credit (0.5) allowed if 2NPS-SWG003 loss is identified without recognizing loss of B RPS MG set.

3. QUESTION: (1.0 pts.)

2-C-3. Why did the following annunciator alarm:

603143 FD WTR CONT V 10A/10B/10C 1C ACTUATOR TROUBLE

ANSWER:

2-C-3. This annunciator alarmed due to loss of power (0.50) to 2FWS-LV10B. (0.50) (Loss of 2NHS-MCC003B, due to loss of 2NPS-SWG003)

4. QUESTION: (1.0 pts.)

2-C-5. A reduction in feedwater temperature has occurred.

a. What is the current value of the actual temperature? (0.25)

b. What is the current value of the predicted temperature. (0.25)

c. Is the feedwater temperature within the limits of the Operating Procedures? (0.5)

ANSWER:

2-C-5.
a. 372.5 - 377.5°F (0.25)
b. 375 - 390°F (0.25)
c. Yes (0.50)

5. QUESTION: (1.0 pts.)

2-C-6. The plant technical specifications require three conditions to be met before starting the idle recirculation pump. Which of these conditions is not presently met? (1.00)

ANSWER:

2-C-6. The operating loop is not less than or equal to 50% of rated loop flow. (1.00)

6. QUESTION: (0.5 pts.)

2-R-1. Describe the operator action required if one of the Circulating Water Pumps now in operation were to trip.

ANSWER:

2-R-1. The Main Turbine would have to be shut down. (0.5)

7. QUESTION: (1.0 pts.)

2-R-2. a. Could 4160V Bus 15 and 600V US6 be supplied by closing breaker 15-8 without causing further degradation of the plant electrical system? (0.5)

b. What actions must be taken to complete this transfer? (0.5)

ANSWER:

2-R-2. a. Yes (0.5)

b. Place Breaker 15-3 in PTL (0.1)
Open Breakers 15-1 and 15-7 (0.1)
Close Breaker 103-8 (0.1)
Close Breaker 15-8 (0.1)
Close Breaker 15-1 and 15-7 (0.1)

8. QUESTION: (1.0 pts.)

2-R-3. Under present plant conditions, what would be the response of the reactor operator if pump CCP-P1A tripped on overload?

ANSWER:

2-R-3. 1. Attempt one restart to obtain one CCP Booster and main pump running. (0.33)

2. Perform reactor scram per N2-OP-101C, Section H. (0.33)

3. Remove RBCLC loads from service if required. (0.33)

Name: ______________________

Soc. Sec. #: ______________

Course: U-2 Annual SRO Requal Exam Section A-6

Date: ______________

Grade: ______________

File Section: ______________

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I have read and understand the above.

Signature: ______________________

Date: ______________

1. QUESTION: (1.0 pts.)

2-C-1. How would the present condition of the Turbine Building Closed Loop Cooling System affect the plants ability to maintain 100% power.

ANSWER:

2-C-1. TBCLCS load temperatures will rise (TBCLCS heat removal capacity is reduced) (0.5). This would cause plant capacity to be reduced (concept) (0.5).

2. QUESTION: (0.75 pts.)

2-C-2. Explain how the B pilot scram valve solenoids were de-energized without an RPS trip (scram) signal.

ANSWER:

2-C-2. B RPS MG was de-energized. (0.75) (B RPS MG powers solenoids)

Note: Partial credit (0.5) allowed if 2NPS-SWG003 loss is identified without recognizing loss of B RPS MG set.

3. QUESTION: (1.0 pts.)

2-C-3. Why did the following annunciator alarm:

603143 FD WTR CONT V 10A/10B/10C 1C ACTUATOR TROUBLE

ANSWER:

2-C-3. This annunciator alarmed due to loss of power (0.50) to 2FWS-LV10B. (0.50) (Loss of 2NHS-MCC003B, due to loss of 2NPS-SWG003)

4. QUESTION: (1.0 pts.)

2-C-5. A reduction in feedwater temperature has occurred.

a. What is the current value of the actual temperature? (0.25)

b. What is the current value of the predicted temperature. (0.25)

c. Is the feedwater temperature within the limits of the Operating Procedures? (0.5)

ANSWER:

2-C-5.
a. 372.5 - 377.5°F (0.25)
b. 375 - 390°F (0.25)
c. Yes (0.50)

5. QUESTION: (1.0 pts.)

2-C-6. The plant technical specifications require three conditions to be met before starting the idle recirculation pump. Which of these conditions is not presently met? (1.00)

ANSWER:

2-C-6. The operating loop is not less than or equal to 50% of rated loop flow. (1.00)

6. QUESTION: (1.0 pts.)

2-S-1. Reactor water level remained in the normal band throughout the transient even with the loss of one feed pump. Explain why vessel level was not significantly affected by the reduction in feed flow. (1.0)

ANSWER:

2-S-1. The loss of recirculation pump B reduced power (0.5) to within the capacity of one feed pump. (0.5)

7. QUESTION: (0.75 pts.)

2-S-2. Plant operation is to be continued with only recirc pump "A" in service. The current core exposure is 5 GWd/st. State the MAPLHGR limit for each of the three fuel bundle types.

ANSWER:

2-S-2. The MAPLHGR limit is 9.8. (0.25) for Type BP8CRB219
The MAPLHGR limit is 10.3. (0.25) for Type P8CRB176
The MAPLHGR limit is 9.2 (0.25) for Type P8CRB071
(± 0.2 for rounding error)

8. QUESTION: (0.75 pts.)

2-S-3. In the present feedwater and recirculation system lineups, what power level are you limited to, and which system creates that restriction?

ANSWER:

2-S-3. Single feed pump (0.25) requires that the time power is above 60% to be minimized. (0.50)

STATIC SIMULATOR SCENARIO

Scenario Title:	Single Recirculation Pump Trip	A-7
Exam Duration:	45 Minutes	
Scenario Number:	89-05	
Revision Number:	1	
Course:	Licensed Operator Requal	

Reviewed By:	[signature] Operations Training Supervisor	7/11/89 Date
Reviewed By:	R. T. [signature] Assistant Training Superintendent	7/11/89 Date
Approved By:	[signature] Superintendent of Operations	7/11/89 Date

SCENARIO DESCRIPTION

Scenario Number: 5 A-7

Type of Scenario: Abnormal Status

Title: Single Recirculation Pump Trip

Synopsis: The reactor was operating at power when the A RBCLCW pump tripped and the C Pump failed to automatically start. About the time the recirculation pumps high temperature alarm annunciated, the B recirculation pump tripped. The operators had begun to investigate the situation when the scenario was frozen.

Malfunctions:
RR10 B; Reactor Recirculation Pump Bkr Trip (B)
CW02 A; Reactor Building Closed Loop Cooling Water Pump A Trip
CW02 C; Reactor Building Closed Loop Cooling Water Pump C Trip

Objectives:

A. Identify a trip of a RBCLC Pump by interpreting instrument response.

B. Locate and use N2-OP-13 (Reactor Building Closed Loop Cooling).

C. Using Control Room indication and RBCLC Pump prints, identify the cause of the RBCLC pump trip.

D. Given the Operating Procedure, identify precautions off-normal procedures and procedure notes and cautions which apply to recirculation system operation.

Name: ______________________

Soc. Sec. #: ______________

Course: NRC/Annual RO Requal Exam Section A-7

Date: ______________

Grade: ______________

File Section: ______________

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I have read and understand the above.

Signature: ______________________

Date: ______________

1. QUESTION: (1.0 pts.)

5-C-1. a. Of the following, identify the cause for annunciator 602220, RECIRC PUMP 1A/1B MOTOR TEMP HIGH being illuminated.

Thrust Brg and Guide Brg Temp above 194°F
Stator Phase Temp above 240°F
Seal Cavity Temp above 160°F
Stator Water Temp high
Brg & Seal Water Temp high

b. What indication (source) verifies your answer?

ANSWER:

5-C-1. a. Bearing & Seal Water Temp high, (0.50)

b. Computer point RCSTC10 (on the alarm printer). (0.50)

2. QUESTION: (1.0 pts.)

5-C-2. If RBCLC flow is lost to the recirculation pumps, what is the limiting factor for continued pump operation?

State the setpoints of concern. (2 required)

ANSWER:

5-C-2. Motor winding cooling. (0.5)
248°F continuous (0.25)
266°F intermittent (0.25)

3. QUESTION: (1.0 pts.)

5-C-3. Explain the reason for the small increase in APRM levels following the initial downpower transient.

ANSWER:

5-C-3. The reduced steam flow to the turbine means less extraction steam to the feedwater heaters (0.5) (concept). This results in colder feedwater, leading to more neutron moderation (and an increase in reactor power) (0.5) (concept).

4. QUESTION: (1.0 pts.)

5-C-4. For given plant conditions, state how total core flow can be determined from the sum of the jet pump flow indicators.

ANSWER:

5-C-4. Total core flow can be determined by subtracting the indication of recirc loop 1B sum jet pump flow (B33-R611B) from the indication of recirc loop 1A sum jet pump flow (B22-R611A) on P602. (1.00)

5. QUESTION: (1.0 pts.)

5-C-5. Describe the effect on drywell parameters (pressure and temperature) due to this event. Justify your response.

ANSWER:

5-C-5. Due to the reduced RBCLC flow caused by the RBCLC pump trip/(fail to start) (.34 pts) drywell temperature should start to increase (.33 pts). As drywell temperature rises then drywell pressure will also start to rise (.33 pts). (Due to ideal gas law as T↑ P will also ↑ in a constant volume).

6. QUESTION: (1.0 pts.)

5-C-6. Why are the following annunciators in alarm:

a. 601253 RBCLC PUMP 1A/1B/1C MOTOR OVERLOAD

b. 601252 RBCLC PUMP 1A/1B/1C AUTO TRIP/FAIL TO START

ANSWER:

5-C-6. a. RBCLC Pump 1A tripped on motor overload. (0.50)

b. RBCLC Pump 1C failed to auto start in response to the loss of pump 1A. (0.50)

7. QUESTION: (1.5 pts.)

5-R-1. a. What action should be taken concerning the flow control system for recirculation loop A?

b. For recirculation loop B?

ANSWER:

5-R-1. a. Reduce the flow rate of the A recirculation loop to less than 41,000 gpm. (0.75)

b. The B recirculation loop flow control valve should be closed (to the minimum position) (0.75).

8. QUESTION: (1.0 pts.)

5-R-2. Suppose the problems with recirc pump B are corrected after its trip. What requirements must be met prior to restarting recirc pump B in order to mitigate the thermal shock effects? (3 required)

ANSWER:

5-R-2.
- Delta T between steam dome and reactor bottom head drain line coolant is less than or equal to 145 degrees F.
- Delta T between the reactor coolant within recirc loop B and recirc loop A less than or equal to 50 degrees F.
- Recirc loop A flow rate is less than or equal to 50 percent of rated loop flow.
- Delta T between reactor steam dome and recirc loop B less than or equal to 50°F.

(any 3 at .33 each)

Name: ______________________

Soc. Sec. #: ______________

Course: NRC/Annual SRO Requal Exam Section A-7

Date: ______________

Grade: ______________

File Section: ______________

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I have read and understand the above.

Signature: ______________________

Date: ______________

1. QUESTION: (1.0 pts.)

5-C-1. a. Of the following, identify the cause for annunciator 602220, RECIRC PUMP 1A/1B MOTOR TEMP HIGH being illuminated.

Thrust Brg and Guide Brg Temp above 194°F
Stator Phase Temp above 240°F
Seal Cavity Temp above 160°F
Stator Water Temp high
Brg & Seal Water Temp high

b. What indication (source) verifies your answer?

ANSWER:

5-C-1. a. Bearing & Seal Water Temp high, (0.50)

b. Computer point RCSTC10 (on the alarm printer). (0.50)

2. QUESTION: (1.0 pts.)

5-C-2. If RBCLC flow is lost to the recirculation pumps, what is the limiting factor for continued pump operation?

State the setpoints of concern. (2 required)

ANSWER:

5-C-2. Motor winding cooling. (0.5)
248°F continuous (0.25)
266°F intermittent (0.25)

3. QUESTION: (1.0 pts.)

5-C-3. Explain the reason for the small increase in APRM levels following the initial downpower transient.

ANSWER:

5-C-3. The reduced steam flow to the turbine means less extraction steam to the feedwater heaters (0.5) (concept). This results in colder feedwater, leading to more neutron moderation (and an increase in reactor power) (0.5) (concept).

4. QUESTION: (1.0 pts.)

5-C-4. For given plant conditions, state how total core flow can be determined from the sum of the jet pump flow indicators.

ANSWER:

5-C-4. Total core flow can be determined by subtracting the indication of recirc loop 1B sum jet pump flow (B33-R611B) from the indication of recirc loop 1A sum jet pump flow (B22-R611A) on P602. (1.00)

5. QUESTION: (1.0 pts.)

5-C-5. Describe the effect on drywell parameters (pressure and temperature) due to this event. Justify your response.

ANSWER:

5-C-5. Due to the reduced RBCLC flow caused by the RBCLC pump trip/(fail to start) (.34 pts) drywell temperature should start to increase (.33 pts). As drywell temperature rises then drywell pressure will also start to rise (.33 pts). (Due to ideal gas low as T↑ P will also ↑ in a constant volume).

6. QUESTION: (1.0 pts.)

5-C-6. Why are the following annunciators in alarm:

a. 601253 RBCLC PUMP 1A/1B/1C MOTOR OVERLOAD

b. 601252 RBCLC PUMP 1A/1B/1C AUTO TRIP/FAIL TO START

ANSWER:

5-C-6. a. RBCLC Pump 1A tripped on motor overload. (0.50)

b. RBCLC Pump 1C failed to auto start in response to the loss of pump 1A. (0.50)

7. QUESTION: (1.0 pts.)

5-S-1. Explain the reason for the sudden rise in reactor water level.

ANSWER:

5-S-1. The reactor water level swells immediately after the single pump trip due to more voids produced in the core since core flow is reduced. (1.00) (Theoretical concept)

8. QUESTION: (1.0 pts.)

5-S-2. Determine what action is required if reverse rotation of recirc pump B occurs after its trip (0.50) and the subsequent action required. (0.50)

ANSWER:

5-S-2. Shut the pump discharge valve RCS*MOV18B if reverse rotation does occur (0.50) and reopen RCS*MOV18B five minutes after the valve is fully closed. (0.50)

9. QUESTION: (0.75 pts.)

5-S-3. Plant operation is to be continued with only recirc pump "A" in service. The current core exposure is 5 GWd/st. State the MAPLHGR limit for each of the three fuel bundle types. (Limit ± 0.2 for rounding error)

ANSWER:

5-S-3. The MAPLHGR limit is 9.8. (0.25) for Type BP8CRB219
The MAPLHGR limit is 10.3. (0.25) for Type P8CRB176
The MAPLHGR limit is 9.2 (0.25) for Type P8CRB071

STATIC SIMULATOR SCENARIO

A-8

Scenario Title:	Loss of Condenser as Heat Sink
Exam Duration:	45 Minutes
Scenario Number:	89-06
Revision Number:	1
Course:	Licensed Operator Requal

Reviewed By:	[signature] Operations Training Supervisor	/ 7/11/89 Date
Reviewed By:	R.T. [signature] Assistant Training Superintendent	/ 7/11/89 Date
Approved By:	[signature] Superintendent of Operations	/ 7/11/89 Date

SCENARIO DESCRIPTION

Scenario Number: 6 A-8

Type of Scenario: Major Failure

Title: Loss of Condenser as Heat Sink

Synopsis: The plant was at rated conditions preparing to check a condenser water box for tube leaks. The E circ water pump had been secured and marked up for this effort. Then two of the remaining circulating water pumps trip with a short time delay between trips. The trip of the circ water pumps causes a decrease in main turbine vacuum. As vacuum decreases the operators reduce reactor power with recirculation flow. Eventually, the turbine will trip on loss of vacuum. Following the scram other circ water pumps are lost when the generator trips as NPS 13.8 KV Bus 003 fails to transfer. The operators should realize that control of reactor pressure and level with RCIC or SRVs will be necessary as the condenser is lost as a heat sink.

Malfunctions:
CW04 A; Circulating Water Pump A Trip
CW04 C; Circulating Water Pump C Trip
ED03 C; Electrical Bus Fault (Bus 003)
MC01; Main Condenser Air Inleakage

Objectives:

A. Locate and correctly use NMP-2 Operating Procedures.

B. Locate and correctly use NMP-2 Technical Specifications.

C. Identify normal and/or off normal trends utilizing control board indications.

Name: ______________________________

Soc. Sec. #: ______________

Course: NRC/Annual RO Requal Exam Section A-8

Date: ______________

Grade: ______________

File Section: ______________

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I have read and understand the above.

Signature: ______________________________

Date: ______________

1. QUESTION: (1.0 pts.)

6-C-1. a) What was the first entry condition into the EOP's

b) What will be the next entry condition if this transient continues without further operator action.

ANSWER:

6-C-1. a) Entered already when water level went below 159.3 (0.50)

b) MSIV closure (when vacuum reaches trip point) (0.50)

2. QUESTION: (1.0 pts.)

6-C-2. Why did 2NPS-SWG003 fail to transfer to Reserve Power?

ANSWER:

6-C-2. (ANN 852530) 13.8 KV Bus 003 Electrical Fault indicated.

3. QUESTION: (1.0 pts.)

6-C-3. What signal caused the reactor scram? Include setpoint.

ANSWER:

6-C-3. Reactor scram was caused by TCV Fast Closure (0.5)

(EHC) Pressure < 530 psig (0.5)

4. QUESTION: (1.0 pts.)

6-C-4. Suppose that none of the Circulating Water Pumps are likely to be recovered in the near future. List four systems that could be used under present plant conditions to effect plant cooldown.

ANSWER:

6-C-4. Any four of the following at 0.25 points each:

a. Feed and Condensate
b. RCIC system
c. RHR in Steam Condensing Mode
d. SRVs
e. RWCU System
f. MSL Drains

5. QUESTION: (1.0 pts.)

6-C-5 Explain how securing "E" circ. water pump and opening the water box could have caused the initial loss of condenser vacuum in this event.

ANSWER:

6-C-5 When the water box was drained and opened, air leaked into the condenser through the leaking condenser tubes. (1.00)

6. QUESTION: (1.0 pts.)

6-C-6. What circ water system follow-up actions are required due to the trip of the circ water pumps? (2 required)

ANSWER:

6-C-6. Shut the pump suction valve (0.5)

Place the control switches in PTL (0.5)

7. QUESTION: (1.0 pts.)

6-C-7 Why did "A" reactor recirc pump downshift to slow speed?

ANSWER:

6-C-7 Due to turbine stop or control valve closure with power >30% (Also accept EOC-RPT)

8. QUESTION: (1.0 pts.)

6-R-1. Explain, using specific plant indications, why reactor pressure drops following the turbine trip/reactor scram instead of being maintained at the setpoint of 935 psig.

ANSWER:

6-R-1. The feedwater flow and temperature indications on panel P603 show cold (1.6×10^6 lb/hr) feedwater (less than 300 °F) entering the reactor vessel. This cold feedwater suppresses reactor pressure (below the 935 psig setpoint). (Theoretical concept) (1.0)

9. QUESTION: (1.0 pts.)

6-R-2. Given the present plant conditions and trends, describe when the turbine bypass valves will no longer be effective in controlling reactor pressure? Include setpoint.

ANSWER:

6-R-2. The bypass valves will become ineffective in controlling reactor pressure when condenser vacuum decreases to the trip setpoint of the MSIV's (0.5) 8.5" Hg. (0.5)

10. QUESTION: (0.5 pts.)

6-R-3. The SSS has directed you to open the condenser vacuum breakers when procedural prerequisties are met. Given the present plant conditions, determine when the condenser vacuum breakers may be opened.

ANSWER:

6-R-3. When main turbine speed is below 1100 rpm. (0.50)

(Also acceptable: In an emergency, the SSS may direct the vacuum breakers opened at any RPM.)

Name: ______________________

Soc. Sec. #: ______________

Course: NRC/Annual SRO Requal Exam Section A-8

Date: ______________

Grade: ______________

File Section: ______________

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Signature: ______________________

Date: ______________

1. QUESTION: (1.0 pts.)

6-C-1. a) What was the first entry condition into the EOP's

b) What will be the next entry condition if this transient continues without further operator action.

ANSWER:

6-C-1. a) Entered already when water level went below 159.3 (0.50)

b) MSIV closure (when vacuum reaches trip point) (0.50)

2. QUESTION: (1.0 pts.)

6-C-2. Why did 2NPS-SWG003 fail to transfer to Reserve Power?

ANSWER:

6-C-2. (ANN 852530) 13.8 KV Bus 003 Electrical Fault indicated.

3. QUESTION: (1.0 pts.)

6-C-3. What signal caused the reactor scram? Include setpoint.

ANSWER:

6-C-3. Reactor scram was caused by TCV Fast Closure (0.5)

(EHC) Pressure < 530 psig (0.5)

4. QUESTION: (1.0 pts.)

6-C-4. Suppose that none of the Circulating Water Pumps are likely to be recovered in the near future. List four systems that could be used under present plant conditions to effect plant cooldown.

ANSWER:

6-C-4. Any four of the following at 0.25 points each:

a. Feed and Condensate
b. RCIC system
c. RHR in Steam Condensing Mode
d. SRVs
e. RWCU System
f. MSL Drains

5. QUESTION: (1.0 pts.)

6-C-5 Explain how securing "E" circ. water pump and opening the water box could have caused the initial loss of condenser vacuum in this event.

ANSWER:

6-C-5 When the water box was drained and opened, air leaked into the condenser through the leaking condenser tubes. (1.00)

6. QUESTION: (1.0 pts.)

6-C-6. What circ water system follow-up actions are required due to the trip of the circ water pumps? (2 required)

ANSWER:

6-C-6. Shut the pump suction valve (0.5)

Place the control switches in PTL (0.5)

7. QUESTION: (1.0 pts.)

6-C-7 Why did "A" reactor recirc pump downshift to slow speed?

ANSWER:

6-C-7 Due to turbine stop or control valve closure with power >30%
(Also accept EOC-RPT)

8. QUESTION: (1.25 pts.)

6-S-1. What will cause the MSIV's to close assuming no operator actions are taken? Include any setpoints in your answer as required. (1.0)

Describe the expected trend of reactor pressure due to the closure of the MSIV's.
Explain how the pressure trend will be terminated.

ANSWER:

6-S-1. When condenser vacuum decreases to 8.5" Hg, (0.25) reactor pressure will begin to rise as a result of MSIV isolation, (0.25) reactor pressure will increase slowly until the first SRV lifts (0.25) at its setpoint of 1076 psig. (0.25)

9. QUESTION: (1.0 pts.)

6-S-3. . Assume that the operator successfully transfers 13.8 KV Bus 2NPS-SWG003 to the reserve transformer as condenser vacuum decreases to 5" Hg. Reactor pressure is 900 psig.

Explain why it is not appropriate to place the condenser air removal pumps in service at this time in order to maintain condenser vacuum knowing that SJAEs are unavailable due to MSIVs closure. (1.0)

ANSWER:

6-S-3. The discharge of condenser air removal pumps are not treated by the offgas system. (1.0)

ATTACHMENT 5

Written Examination and Answer Key

Part B (RO and SRO)

Name: ______________________

Soc. Sec. #: ____________

Course: U-2 Annual RO Requal Exam Section B-3

Date: ____________

Grade: ____________

File Section: ____________

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I have read and understand the above.

Signature: ______________________

Date: ____________

1. QUESTION: (1.5 pts.)

Immediately following a primary system line break inside the drywell, a containment isolation due to high drywell pressure, and an automatic actuation of SBGT occur. An operator attempts to rapidly reduce primary containment pressure at the direction of the SSS.

Determine what actions need to be taken by the operator in order to open the following valves: (1.5)

DW Purge Inboard Outlet Isol Vlv 2CPS*AOV108 and DW Purge Outboard Outlet Isol Vlv 2CPS*AOV110.
Suppr Pool Purge Inboard Isol Vlv 2CPS*AOV109 and Suppr Pool Outboard Outlet Isol Vlv 2CPS*AOV111.

ANSWER:

1. (At Panels P873 and P875), place Purge Outboard/Inboard Valve override switches to OVERRIDE Positions. (0.75)

2. (At Panels P859 and P861), place a jumper to remove DW pressure isolation signal of 2IAS*SOV168 and 2IAS*SOV180, Instrument Air Outboard and Inboard Isolation Valves. (0.75)

NOTE: Accept valve name or number as correct answer

REFERENCE: N2-OP-61A, Primary Containment Ventilation, Purge and Nitrogen System, Rev. 3; Section H.1.0 and H.3.0
N2-EOP-PC, Primary Containment Control; Rev. 2

2. QUESTION: (1.0 pts.)

The plant is operating at steady state conditions with 60% power. Feedwater pumps 2FWS-P1A and 2FWS-P1B are in 3-elements automatic mode of operation. The standby feedwater pump, 2FWS-P1C, has been secured for preventive maintenance on its drive motor. While the maintenance activity is still in progress feedwater pump 2FWS-P1A trips. During the transient, reactor water level dips to 177 inches.

a. State whether this power level is within the capacity of the remaining feedwater pump, 2FWS-P1B. (.5)

b. State the required operator actions for this occurrence. (.5)

ANSWER:

a. Yes (.5)

b. Per N2-OP-3, H.1.0,

  a. Verify recirculation flow control valve partial closure initiated. (.2)
  b. If recirculation flow control valve fails to runback, manually close recirculation flow control valve on P602. (.05)
  c. Monitor reactor level is maintained at lower power level. (.2)
  d. Determine the cause of the feedwater pump trip. (.05)

REFERENCE: N2-OP-3, Condensate and Feedwater System, Rev. 4; Sections B, H.1.0 and H.5.0

3. QUESTION: (0.75 pts.)

The plant is operating at steady state 100% power.

What immediate actions should be taken by the operator if the air header pressure reaches 60 psig and decreasing? (0.75)

ANSWER:

Manually scram the reactor (0.25), close the outboard MSIVs (0.25), and follow N2-OP-101C (0.25).

REFERENCE: N2-OP-19, Instrument and Service Air System, Rev. 2; Sections H.2.3, I.4.0, I.5.0, I.6.0, I.7.0

4. QUESTION: (1.5 pts.)

The plant is operating at steady state 100% power when the following annunciators alarm:

851253 PROCESS GAS RADN MON ACTIVATED
603133 MAIN STEAM LINE RADIATION HIGH

An operator has confirmed that both radiation monitors, 2OFG-RE13A and RE13B, are operating properly and indicate increasing radiation levels of greater than 12 microcuries/cc. Based on the reactor water and feedwater samples taken right after the alarm actuations, the Chemistry Department reports no significant changes to the conductivities and pH values, although the specific activity of reactor water is slightly above 4 microcuries/gm dose equivalent I-131.

Based on the mentioned indications and the probable causes listed in N2-OP-42, determine the most probable cause. Justify your answer by stating why each of the other possible causes was not selected.

ANSWER:

1. Fuel damage. Probable (0.50)
2. Intrusion of reactor water cleanup or condensate resins into the reactor. Both reactor water and feedwater qualities would have significantly changed if resin intrusion occurred (0.50)
3. Introduction of organic compound (i.e. cleaning fluids, lube oil, etc.) into the reactor via building drains. Plant chemistry would degrade (low pH and high conductivity) along with increased radiation levels. (0.50)

REFERENCE: N2-OP-42, Offgas System, Rev. [illegible], Section I.46.0, Tech. Spec. Table 4.4.5-1
N2-OP-1, Main Steam System, Rev. 6; Section I.17.0

5. QUESTION: (1.0 pts.)

The plant is in the process of cooldown. RHR B is currently operating in the shutdown cooling mode with the following conditions:

- RHR B total flow is approximately 7450 gpm.
- RHR B to reactor head spray valve RHS*MOV104 is shut.
- RHR B heat exchanger service water outlet temp. is 110 degrees F.
- RHR B heat exchanger inlet bypass valve RHS*MOV8B is about 80% open.

Both recirculation pumps are inoperable. The reactor coolant temperature is 300 degrees F and decreasing at 90 degrees F per hour over the last 15 minutes.

List two actions the operator should take in order to decrease the reactor cooldown rate without causing RPV thermal stratification. (1.0)

ANSWER:

Any two (2) at .5 points each for a total of 1.0 point.

1. Open the RHR B heat exchanger bypass valve (MOV8B).

2. Throttle RHR B heat exchanger service water flow outlet (2SWP*MOV33B) (if the RHR B heat exchanger bypass valve is full open).

3. With little or no decay heat it may also be necessary to close RHR B heat exchanger outlet valve (2RHS*MOV12B).

NOTE: Either valve number or name for Full Credit

REFERENCE: N2-OP-31, Residual Heat Removal System, Rev. 5, Section D.6.0 and E.8.18

6. QUESTION: (1.0 pts.)

The plant experienced a scram caused by an inadvertent full closure of all MSIVs. With all MSIVs being closed, plant cooldown is effected with RCIC and RHR in the steam condensing mode of operation. What actions should the operator take if the RHR heat exchanger pressure decreases to less than 200 psig? (Two (2) required) (1.0)

ANSWER:

1. Throttle open RHR heat exchanger steam supply valve as required to maintain heat exchanger pressure at 200 psig. (0.5)

2. If RHR heat exchanger pressure cannot be maintained at 200 psig, reduce RPV cooldown rate to prevent exceeding RHR heat exchanger service water differential temperature of 29°F. (0.5)

REFERENCE: N2-OP-31, Residual Heat Removal System, Rev. 5; Section E.9.2

7. QUESTION: (1.25 pts.)

Considering the Reactor Building Closed Loop Cooling (RBCLC) System:

1. Determine at least three symptoms that would alert you to possible system in-leakage during normal plant operation? (0.75)

2. What immediate actions should be taken by the operator in response to a complete loss of RBCLC System? (0.50)

ANSWER:

1. Any three of the following at 0.25 point each:

   a. High radiation level indicated by the radiation monitors, (2CCP-RE131 and E115). (0.25)

   b. RBCLC Pump Suction Header Conductivity High (0.25)

   c. RBCLC Pump Suction Header PH High (0.25)

   d. RBCLC Expansion Tank (TK1) Level High (0.25)

2. a. Perform reactor scram (per N2-OP-101C, Section H). (0.25)

   b. Remove RBCLC System loads from service. (0.25)

REFERENCE: N2-OP-13, Reactor Building Closed Loop Cooling, Rev. 2; Section H.1.0 and I.7.0

8. QUESTION: (0.75 pts.)

A containment isolation occurs due to LOCA signal following a primary system line break. An operator attempts to reduce drywell temperature by operating the Drywell Cooling System.

Describe what actions need to be taken to restore the Drywell Cooling System.

ANSWER:

(At P873), place Div. 1 and 2 Drywell Unit Cooling Water LOCA Override Keylock switches to "OVERRIDE". (0.25)

(At P873), open DW Cooler Inbd/Outbd valves. (0.25)

(At P873), (place the two Keylock Drywell Unit Cooler Fans GR1/2 LOCA Override switches to "OVERRIDE" then) start DRS unit coolers. (0.25)

REFERENCE: N2-OP-60, Drywell Cooling, Rev. 1, Section H.1.0
N2-OP-13, Reactor Building Closed Loop Cooling, Rev. 2, Section H.2.0

9. QUESTION: (1.0 pts.)

Plant is at 100 percent power and offgas system train B is in service when the following alarm annunciates:

851306 OFF GAS SYSTEM TROUBLE

Simultaneously, on panel 2OFG-IPNL122, system flow indication as shown on 2OFG-FI3B is oscillating and the following alarms annunciate:

122201 HIGH INLET TEMPERATURE SYSTEM A & B
122202 HIGH INLET PRESSURE SYSTEM A & B

The operator acknowledged the annunciators and was able to reset annunciators HIGH INLET TEMPERATURE SYSTEM A & B and HIGH INLET PRESSURE SYSTEM A & B. System flow returns to normal without any further operator action.

1. Based on the information given above, determine what event has occurred. (0.5)

2. Justify your answer. (0.5)

ANSWER:

1. A hydrogen explosion. (0.5)

2. Multiple system trouble alarms occurring simultaneously with the high pressure alarm. (0.5)

REFERENCE: N2-OP-42, Section H.5.0, Section I.1.0-I.2.0, I.47.0

10. QUESTION: (2.0 pts.)

The Site Emergency Plan is activated. The Site Emergency Director declares an Unusual Event.

Who must the Control Room Communications Aide attempt to notify?

ANSWER:

He must attempt to notify the following: (0.25 points each)

Nuclear Security Department
System Power Control
Oswego County
New York State
Nine Mile Unit 1 Control Room
JAFNPP Control Room
Nuclear Regulatory Commission
NRC Senior Resident Inspector

REFERENCE: EPP-20, Emergency Notifications, Rev. 11, Section 6.0 and Figure 5

11. QUESTION: (1.25 pts.)

The plant is operating at 100% power when a pressure control failure causes reactor pressure to increase to 1050 psig. The RPS scram fails to actuate and the RRCS initiates resulting in control rod insertion by the ARI initiation. APRMs reach downscale 45 seconds after the RRCS initiation signal.

One minute after the pressure spike reactor water level is 158 inches and slowly decreasing.

a. What automatic action occurred to cause this loss of level control?
b. The SSS directs you to restore level to between 159.3" and 202.3" per EOP-RL. What action do you take?

ANSWER:

a. RRCS feedwater runback (RRCS initiation pus 25 seconds with APRMS not downscale) (.5).
b. The reactor operator should take manual control of the feedwater valves to restore level (.75).

REFERENCE: N2-OP-3, Pgs. 5 & 6
N2-OP-36B, Section 3.3

12. QUESTION: (1.0 pts.)

The plant was operating normally when a scram occurred. All rods are at position 00. You note the following nuclear instrumentation values:

SRM: detectors inserted, channels indicating about 1 X 10^4 cps with negative 80 second period.

IRM: detectors inserted, all channels on Range 3 and decreasing.

APRM: All recorders downscale, front panel UPSC or INOP amber indicating lights are on and white Downscale indicating lights are on, backpanel white INOP and DNSC lights are on.

Based on the above indications only, is an EOP Entry condition met? Justify your answer.

ANSWER:

(Even though the APRMs have failed) it is not necessary to enter EOPs. (0.50) (Alternate methods of determining reactor power below 4.0% are provided for.) From the given indications, reactor power can be determined to be less than 4.0% and decreasing. (0.50)

REFERENCE: N2-EOP-RPV, RPV Control, Rev. 2, Section RQ

13. QUESTION: (1.5 pts.)

The reactor is in an accident situation with the following conditions:

Reactor pressure	950 psig
Vessel water level	-14 inches and decreasing
HPCS, Feedwater, RCIC	Out of service
LPCS and RHR pumps	Running, but not injecting
All Control Rods	Position 00

A small LOCA has occurred and the CRD system is not sufficient to return the water level to normal. State which contingencies of the Emergency Operating Procedures are being used and what action will be directed to be taken to return the reactor water level to normal. (1.5 pts.)

ANSWER:

In accordance with (RPV Control, Section RL), EOP-C1 Level Restoration (0.50), and EOP-C2 Emergency Depressurization (0.50), emergency depressurization will be performed, (allowing low pressure ECCS systems injection to return level to the normal band (159.3 to 202.3)). (0.50)

REFERENCE: N2-EOP-RPV, Rev. 2, RPV Control, Section RL
N2-EOP-C1, Level Restoration, Rev. 2
N2-EOP-C2, Emergency RPV Depressurization, Rev. 2

14. QUESTION: (1.0 pts.)

Reactor power is 9 % with the mode switch in Startup, ready to transfer to RUN, when condenser vacuum begins to drop.

a. Condenser vacuum is now 10" Hg Vac. and slowly dropping. Explain why it is not procedurally permissible to start the Condenser Air Removal Pumps?

b. Condenser vacuum falls to 7.5" Hg Vac. What Emergency Operating Procedure should be entered and why?

ANSWER:

a. Use of the vacuum pumps above 5% rated power is prohibited because: Radiolytic hydrogen and oxygen would be introduced into the plant exhaust stack in potentially explosive quantities and, (0.50)

b. N2-EOP-RPV (Control due to MSIV isolation). (0.50)

REFERENCE: N2-OP-9, Condenser Air Removal, Rev. 2, Section H.2.0 and Section D

15. QUESTION: (0.75 pts.)

The plant is in normal operation while maintenance is performing repairs on reactor building above refueling floor exhaust fan 2HVR-FN5A, which is inoperable. An error occurs, tripping fan 2HVR-FN5B. Within 15 minutes, maintenance identifies the cause of the trip and reports fan 2HVR-FN5B is ready for restart.

List the actions the operator must perform <u>prior</u> to restarting an exhaust fan.

ANSWER:

1. Bypass the Rx Bldg. low flow interlocks. (0.15)

2. Place the control switch for 2HVR*UC413B to NORMAL-AFTER-STOP. (0.15)

3. Secure one train of standby gas treatment (per section G.1.0 of N2-OP-61B). (0.15)

4. Place or verify control switches for 2HVR-FN1 (A, B, and C), 2HVR-FN2 (A and B) and 2HVR-FN5 (A and B) in pull-to-lock. (0.15)

5. Open dampers 2HVR-AOD1 A and B, 9A and B and 10A and B. (0.15)

REFERENCE: N2-OP-52, Reactor Building Ventilation, Rev. 3, Sections H.4.1-H.4.5 & I.62.0
N2-OP-61B, Standby Gas Treatment, Rev. 5, Section B

16. QUESTION: (0.75 pts.)

While operating the fuel prep machine on a backshift, a malfunction occurs and one operator is contaminated, breaks his left elbow, and receives a whole body dose estimated at 250 mrem. The situation is now under control, radiation levels are normal, and no one else has been contaminated or exposed.

a. Activation of the Emergency Plan? (yes/no)

b. Determine the level of medical emergency.

ANSWER:

a. Yes (activation of the Emergency Plan is required.) (0.25)

b. This is a Major Medical Emergency - Contaminated. (0.50)

REFERENCE: EPP-4, Personnel Injury or Illness, Rev. 12, Section 10.2
EAP-2, Classification of Emergency Conditions, Rev. 9, Attachment 2, Figure 2.A
EAP-1, Activation and Direction of the Emergency Plans, Rev. 6, Figure 1

Name: ____________________

Soc. Sec. #: ____________________

Course: U-2 Annual SRO Requal Exam Section B-3

Date: ____________________

Grade: ____________________

File Section: ____________________

Trainees will be judged to have willfully violated the integrity of an examination if they are found to have:

a) Utilized unauthorized documents during the examination.

b) Secured unauthorized documents for the purpose of accessibility during an examination.

c) Solicited examination information from other trainees or any other individuals.

d) Provided examination information to other trainees during an examination.

e) Reviewed or attempted to review materials that are un-authorized, including the examination prior to implementation, the examination answer key, or the answers developed by any other trainee during the examination.

I have read and understand the above.

Signature: ____________________

Date: ____________________

1. QUESTION: (1.0 pts.)

Reactor power	IRM range 7
Primary containment pressure	0.80 psig
Primary containment temperature	100°F and steady
Reactor containment	N_2 inerted

The Nitrogen low flow makeup to Primary Containment is in service.

State two methods available to the operator to restore containment pressure.

ANSWER:

a. Isolate the nitrogen supply (0.25) and allow primary containment pressure to decay. (0.25)

b. Vent the drywell using the Standby Gas Treatment System, to the stack (0.50)

REFERENCE: N2-OP-101A, "Plant Startup Procedure", Rev. 6, Section E.2.21
N2-OP-61A, "Primary Containment Ventilation, Purge and Nitrogen System", Rev. 3, Section H.1.2.1

2. QUESTION: (1.5 pts.)

Immediately following a primary system line break inside the drywell, a containment isolation due to high drywell pressure, and an automatic actuation of SBGT occur. An operator attempts to rapidly reduce primary containment pressure at the direction of the SSS.

Determine what actions need to be taken by the operator in order to open the following valves: (1.5)

DW Purge Inboard Outlet Isol Vlv 2CPS*AOV108 and DW Purge Outboard Outlet Isol Vlv 2CPS*AOV110.
Suppr Pool Purge Inboard Isol Vlv 2CPS*AOV109 and Suppr Pool Outboard Outlet Isol Vlv 2CPS*AOV111.

ANSWER:

1. (At Panels P873 and P875), place Purge Outboard/Inboard Valve override switches to OVERRIDE Positions. (0.75)

2. (At Panels P859 and P861), place a jumper to remove DW pressure isolation signal of 2IAS*SOV168 and 2IAS*SOV180, Instrument Air Outboard and Inboard Isolation Valves. (0.75)

NOTE: Accept valve name or number as correct answer

REFERENCE: N2-OP-61A, Primary Containment Ventilation, Purge and Nitrogen System, Rev. 3; Section H.1.0 and H.3.0
N2-EOP-PC, Primary Containment Control; Rev. 2

3. QUESTION: (1.0 pts.)

The plant is operating at steady state conditions with 60% power. Feedwater pumps 2FWS-P1A and 2FWS-P1B are in 3-elements automatic mode of operation. The standby feedwater pump, 2FWS-P1C, has been secured for preventive maintenance on its drive motor. While the maintenance activity is still in progress feedwater pump 2FWS-P1A trips. During the transient, reactor water level dips to 177 inches.

a. State whether this power level is within the capacity of the remaining feedwater pump, 2FWS-P1B. (.5)

b. State the required operator actions for this occurrence. (.5)

ANSWER:

a. Yes (.5)

b. Per N2-OP-3, H.1.0,

   a. Verify recirculation flow control valve partial closure initiated. (.2)
   b. If recirculation flow control valve fails to runback, manually close recirculation flow control valve on P602. (.05)
   c. Monitor reactor level is maintained at lower power level. (.2)
   d. Determine the cause of the feedwater pump trip. (.05)

REFERENCE: N2-OP-3, Condensate and Feedwater System, Rev. 4; Sections B, H.1.0 and H.5.0

4. QUESTION: (1.0 pts.)

The plant is operating at steady state 100% power when the following annunciators alarm:

851229	INSTR AIR SYSTEM TROUBLE
851238	INSTR AIR COMPRESSOR 1A/1B/1C AUTO START
851228	INSTR AIR CPSR 1A/1B/1C AUTO TRIP/FAIL TO START
851239	SER AIR SYS 2IAS-AOV 171 CLOSED

A reactor scram will occur if the air pressure continues to drop.

State the system responses to a loss of instrument air which results in a scram signal to the Reactor Protection System. Two (2) systems required. Any two (2) full credit.

ANSWER:

1. Individual MSIV will close due to loss of instrument air to each MSIV pilot valve and this will cause a scram because of MSIV closure. (0.5)

2. Feedwater pump minimum flow valves will fail full open due to loss of instrument air and this will cause a loss of feedwater to reactor and the resulting scram on low water level. (0.5)

3. Scram inlet and outlet valves will fail open due to loss of instrument air and this will cause the Scram Discharge Volume to fill to the RPS Scram Setpoint. (0.5)

REFERENCE: N2-OP-19, Instrument and Service Air System, Rev. 2; Sections H.2.0, I.4.0, I.5.0, I.6.0, I.7.0

5. QUESTION: (1.5 pts.)

The plant is operating at steady state 100% power when the following annunciators alarm:

851253 PROCESS GAS RADN MON ACTIVATED
603133 MAIN STEAM LINE RADIATION HIGH

An operator has confirmed that both radiation monitors, 2OFG-RE13A and RE13B, are operating properly and indicate increasing radiation levels of greater than 12 microcuries/cc. Based on the reactor water and feedwater samples taken right after the alarm actuations, the Chemistry Department reports no significant changes to the conductivities and pH values, although the specific activity of reactor water is slightly above 4 microcuries/gm dose equivalent I-131.

Based on the mentioned indications and the probable causes listed in N2-OP-42, determine the most probable cause. Justify your answer by stating why each of the other possible causes was not selected.

ANSWER:

1. Fuel damage. Probable (0.50)
2. Intrusion of reactor water cleanup or condensate resins into the reactor. Both reactor water and feedwater qualities would have significantly changed if resin intrusion occurred. (0.50)
3. Introduction of organic compound (i.e. cleaning fluids, lube oil, etc.) into the reactor via building drains. Plant chemistry would degrade (low pH and high conductivity) along with increased radiation levels. (0.50)

REFERENCE: N2-OP-42, Offgas System, Rev. 3, Section I.46.0, Tech. Spec. Table 4.4.5-1
N2-OP-1, Main Steam System, Rev. 6; Section I.17.0

6. QUESTION: (2.0 pts.)

The plant experienced a line break outside the drywell followed by a scram. The following conditions existed after the scram:

- Several control rods indicate > position 02
- Reactor water level is 165" and increasing
- Reactor pressure is 1080 psig and increasing
- Main Steam Line Pipe Tunnel = 165 degrees F
- Reactor Building Pipe Chase = 145 degrees F

The SSS directs operator to control reactor pressure below 1076 psig by opening two SRVs per N2-EOPs. Determine what other plant systems, if any, the operator can use to augment RPV pressure control in accordance with N2-EOP-RPV. (1.0) Justify your answer. (1.0)

ANSWER:

There are no other plant systems available, (1.0) since (a) both RWCU and RCIC/RHR Steam Condensing are isolated, (0.5) and (b) MSIVs and main steam line drains are isolated. (0.5).

REFERENCE: N2-EOP-RPV, RPV Control, Rev. 2
N2-OP-83, Primary Containment Isolation System, Rev. 1, Section I.1.0 and Enclosure #1, pages 28 34, and 39

7. QUESTION: (1.0 pts.)

The plant experienced a scram caused by an inadvertent full closure of all MSIVs. With all MSIVs being closed, plant cooldown is effected with RCIC and RHR in the steam condensing mode of operation. What actions should the operator take if the RHR heat exchanger pressure decreases to less than 200 psig? (Two (2) required) (1.0)

ANSWER:

1. Throttle open RHR heat exchanger steam supply valve as required to maintain heat exchanger pressure at 200 psig. (0.5)

2. If RHR heat exchanger pressure cannot be maintained at 200 psig, reduce RPV cooldown rate to prevent exceeding RHR heat exchanger service water differential temperature of 29°F. (0.5)

REFERENCE: N2-OP-31, Residual Heat Removal System, Rev. 5; Section E.9.24

8. QUESTION: (1.25 pts.)

Considering the Reactor Building Closed Loop Cooling (RBCLC) System:

1. Determine at least three symptoms that would alert you to possible system in-leakage during normal plant operation? (0.75)

2. What immediate actions should be taken by the operator in response to a complete loss of RBCLC System? (0.50)

ANSWER:

1. Any three of the following at 0.25 point each:

   a. High radiation level indicated by the radiation monitors, (2CCP-RE131 and E115). (0.25)

   b. RBCLC Pump Suction Header Conductivity High (0.25)

   c. RBCLC Pump Suction Header PH High (0.25)

   d. RBCLC Expansion Tank (TK1) Level High (0.25)

2. a. Perform reactor scram (per N2-OP-101C, Section H). (0.25)

   b. Remove RBCLC System loads from service. (0.25)

REFERENCE: N2-OP-13, Reactor Building Closed Loop Cooling, Rev. 2; Section H.1.0 and I.7.0

9. QUESTION: (0.75 pts.)

A containment isolation occurs due to LOCA signal following a primary system line break. An operator attempts to reduce drywell temperature by operating the Drywell Cooling System.

Describe what actions need to be taken to restore the Drywell Cooling System.

ANSWER:

(At P873), place Div. 1 and 2 Drywell Unit Cooling Water LOCA Override Keylock switches to "OVERRIDE". (0.25)

(At P873), open DW Cooler Inbd/Outbd valves. (0.25)

(At P873), (place the two Keylock Drywell Unit Cooler Fans GR1/2 LOCA Override switches to "OVERRIDE" then) start DRS unit coolers. (0.25)

REFERENCE: N2-OP-60, Drywell Cooling, Rev. 1, Section H.1.0
N2-OP-13, Reactor Building Closed Loop Cooling, Rev. 2, Section H.2.0

10. QUESTION: (1.0 pts.)

A normal reactor startup is in progress with both recirculation pumps at low speed and the following conditions:

- Generator output is 225 MWe.
- Generator hydrogen gas pressure is 75 psig.
- Generator Power Factor (PF) is 0.90.
- Stator cooling water conductivity is 0.2 micro mho/cm
- Alarm 851135, STTR CLG WTR PUMP 1A/1B AUTO TRIP/FAIL TO START, is received.

The auto start of Standby Stator Cooling Water pump does not occur. Determine the operator action required to continue generator operation. (1.0)

ANSWER:

Within an hour, (0.5) attempt to restore stator cooling water flow by starting a pump at (P851). (0.5)

REFERENCE: N2-OP-26 , Generator Stator and Exciter Rectifier Cooling System, Rev. 1; Sections I.15.0
N2-OP-68, Main Generator, Exciter, Main Transformer, 345 KV Yard, and Generator/Unit Protection, Rev. 0, Section H.3.0

11. QUESTION: (1.0 pts.)

The plant is in a normal startup from an outage. Reactor is critical with IRMs indicating 25 on Range 2. Following a rod withdrawal of one notch, the operator observes that IRM readings increase to 50 on Range 2 in 25 seconds.

1. Determine the reactor period resulting from the control rod withdrawal as stated above. (0.25) Show all calculations in your answer. (0.25)

2. What action(s) would the SSS direct the operator to take in response to a 50 second period? (0.25) State the reference. (0.25)

ANSWER:

1. Reactor Period = 1.44 X (Doubling Time), seconds
   = 1.44 X 25 seconds (0.25)

   Thus, Reactor Period is 36 seconds (0.25)

2. Stop pulling rods or insert rods as necessary (0.25) IAW OP-92, Section I.9.0 (0.25)

REFERENCE: N2-OP-101A, Plant Startup, Rev. 6, Section E.2.9
N2-OP-92, Neutron Monitoring, Rev. 1, Section I.9.0

12. QUESTION: (1.0 pts.)

Plant is at 100 percent power and offgas system train B is in service when the following alarm annunciates:

851306 OFF GAS SYSTEM TROUBLE

Simultaneously, on panel 2OFG-IPNL122, system flow indication as shown on 2OFG-FI3B is oscillating and the following alarms annunciate:

122201 HIGH INLET TEMPERATURE SYSTEM A & B
122202 HIGH INLET PRESSURE SYSTEM A & B

The operator acknowledged the annunciators and was able to reset annunciators HIGH INLET TEMPERATURE SYSTEM A & B and HIGH INLET PRESSURE SYSTEM A & B. System flow returns to normal without any further operator action.

1. Based on the information given above, determine what event has occurred. (0.5)

2. Justify your answer. (0.5)

ANSWER:

1. A hydrogen explosion. (0.5)

2. Multiple system trouble alarms occurring simultaneously with the high pressure alarm. (0.5)

REFERENCE: N2-OP-42, Section H.5.0, Section I.1.0-I.2.0, I.47.0

13. QUESTION: (1.0 pts.)

While operating the fuel prep machine on a backshift, a malfunction occurs and one operator is contaminated, breaks his left elbow, and receives a whole body dose estimated at 250 mrem. The situation is now under control, radiation levels are normal, and no one else has been contaminated or exposed.

a. Determine the level of medical emergency.

b. Determine the emergency classification.

ANSWER:

a. This is a Major Medical Emergency - Contaminated. (0.50)

b. This is classified as an Unusual Event. (0.50)

REFERENCE: EPP-4, Personnel Injury or Illness, Rev. 12, Section 10.2
EAP-2, Classification of Emergency Conditions, Rev. 9, Attachment 2, Figure 2.A
EAP-1, Activation and Direction of the Emergency Plans, Rev. 6, Figure 1

14. QUESTION: (1.0 pts.)

Reactor power is 9 % with the mode switch in Startup, ready to transfer to RUN, when condenser vacuum begins to drop.

a. Condenser vacuum is now 10" Hg Vac. and slowly dropping. Explain why it is not procedurally permissible to start the Condenser Air Removal Pumps?

b. Condenser vacuum falls to 7.5" Hg Vac. What Emergency Operating Procedure should be entered and why?

ANSWER:

a. Use of the vacuum pumps above 5% rated power is prohibited because: Radiolytic hydrogen and oxygen would be introduced into the plant exhaust stack in potentially explosive quantities and, (0.50)

b. N2-EOP-RPV (Control due to MSIV isolation). (0.50)

REFERENCE: N2-OP-9, Condenser Air Removal, Rev. 2, Section H.2.0 and Section D

15. QUESTION: (0.75 pts.)

The plant is in normal operation while maintenance is performing repairs on reactor building above refueling floor exhaust fan 2HVR-FN5A, which is inoperable. An error occurs, tripping fan 2HVR-FN5B. Within 15 minutes, maintenance identifies the cause of the trip and reports fan 2HVR-FN5B is ready for restart.

List the actions the operator must perform prior to restarting an exhaust fan.

ANSWER:

1. Bypass the Rx Bldg. low flow interlocks. (0.15)

2. Place the control switch for 2HVR*UC413B to NORMAL-AFTER-STOP. (0.15)

3. Secure one train of standby gas treatment (per section G.1.0 of N2-OP-61B). (0.15)

4. Place or verify control switches for 2HVR-FN1 (A, B, and C), 2HVR-FN2 (A and B) and 2HVR-FN5 (A and B) in pull-to-lock. (0.15)

5. Open dampers 2HVR-AOD1 A and B, 9A and B and 10A and B. (0.15)

REFERENCE: N2-OP-52, Reactor Building Ventilation, Rev. 3, Sections H.4.1-H.4.5 & I.62.0
N2-OP-61B, Standby Gas Treatment, Rev. 5, Section B

16. QUESTION: (1.25 pts.)

As the SSS/SED, you have declared a General Emergency. Using the wind direction shown in that attached figure, determine the minimum Protective Action Recommendations required. (List each PAR by ERPA)

ANSWER:

ERPA 1 - Shelter (.2)

ERPA 2 - Shelter (.2)

ERPA 3 - Shelter (.2)

ERPA 5 - Shelter (.2)

ERPA 26 - Evacuate (.2)

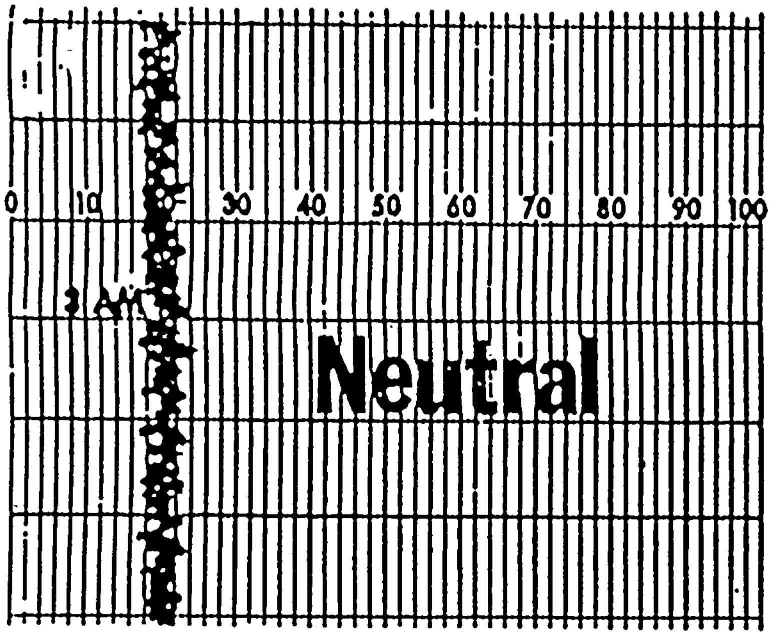
ERPA 27 - Evacuate (.25)

(NOTE: ERPA 27 is downwind, but Evacuation takes presedence over sheltering)

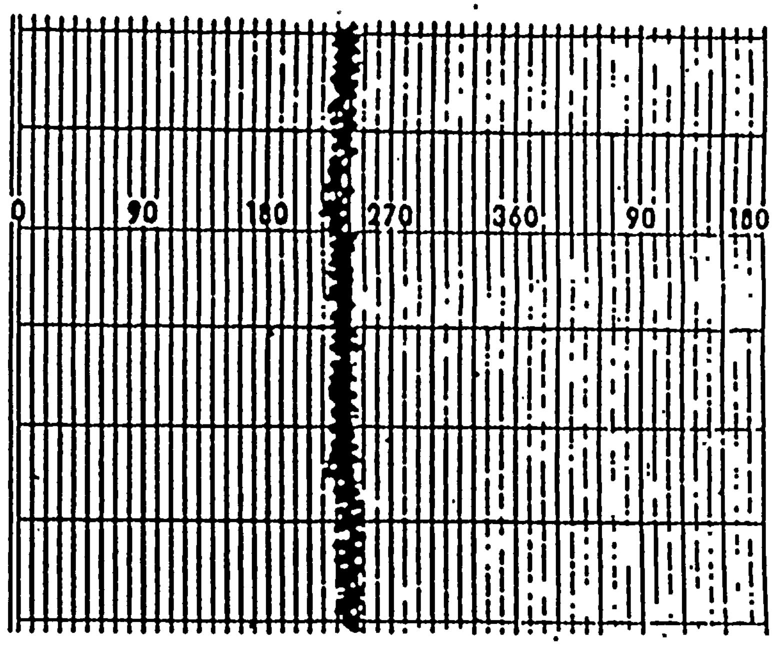
REFERENCE: EPP-8, EPP-26

Niagara Mohawk
Class 3

TYPICAL TURBULENCE ASSOCIATED WITH OVERCAST, STORMY, OR NOCTURNAL SITUATIONS HAVING RELATIVELY STRONG WINDS. MECHANICAL TURBULENCE PREDOMINATES.



Speed



Direction

Name: ______________________________

Soc. Sec. #: ______________

Course: NRC/Annual RO Requal Exam Section B-4

Date: ______________

Grade: ______________

File Section: ______________

Trainees will be judged to have willfully violated the integrity of an examination if they are found to have:

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e) Reviewed or attempted to review materials that are un-authorized, including the examination prior to implementation, the examination answer key, or the answers developed by any other trainee during the examination.

I have read and understand the above.

Signature: ______________________________

Date: ______________

1. QUESTION: (1.5 pts.)

Immediately following a primary system line break inside the drywell, a containment isolation due to high drywell pressure, and an automatic actuation of SBGT occur. An operator attempts to rapidly reduce primary containment pressure at the direction of the SSS.

Determine what actions need to be taken by the operator in order to open the following valves: (1.5)

DW Purge Inboard Outlet Isol Vlv 2CPS*AOV108 and DW Purge Outboard Outlet Isol Vlv 2CPS*AOV110.
Suppr Pool Purge Inboard Isol Vlv 2CPS*AOV109 and Suppr Pool Outboard Outlet Isol Vlv 2CPS*AOV111.

ANSWER:

1. (At Panels P873 and P875), place Purge Outboard/Inboard Valve override switches to OVERRIDE Positions. (0.75)

2. (At Panels P859 and P861), place a jumper to remove DW pressure isolation signal of 2IAS*SOV168 and 2IAS*SOV180, Instrument Air Outboard and Inboard Isolation Valves. (0.75)

NOTE: Accept valve name or number as correct answer

REFERENCE: N2-OP-61A, Primary Containment Ventilation, Purge and Nitrogen System, Rev. 3; Section H.1.0 and H.3.0
N2-EOP-PC, Primary Containment Control; Rev. 2

2. QUESTION: (1.0 pts.)

With the Reactor in Mode 1:

1. What action is required after determining that a jet pump is inoperable? (0.5)

2. Explain how an inoperable jet pump affects the core reflood capability.

ANSWER:

1. The reactor must be in at least HOT SHUTDOWN within 12 hours after a jet pump is determined to be inoperable. (0.5)

2. May not achieve 2/3 core coverage (post LOCA) (0.5)

(NOTE: Also accept drawing similar to attached diagram.)

REFERENCE: Technical Specifications 3.4.1.2
FSAR: Section 3.9.5.3.1B and Figure 3.9B-2

3. QUESTION: (1.5 pts.)

The plant is operating at 95% power when both reactor recirculation pumps trip. No scram occurs, and reactor power stabilizes at 45-50%. The reactor operator identifies the following oscillations:

APRM CHANNEL A:	43-44%	APRM CHANNEL D:	43-49%
APRM CHANNEL B:	46-48%	APRM CHANNEL E:	45-48%
APRM CHANNEL C:	42-45%	APRM CHANNEL F:	47-49%

One recirculation pump is determined to be inoperable, but the other is available for immediate restart.

What should the reactor operator do next?

ANSWER:

1. Immediately place the mode switch in Shutdown. (1.5)

REFERENCE: N2-OP-29, Reactor Recirculation System, Rev. 4, Section H.2.2
NRC Bulletin #88-07, dated 6/15/88

4. QUESTION: (1.0 pts.)

The plant is operating at steady state conditions with 60% power. Feedwater pumps 2FWS-P1A and 2FWS-P1B are in 3-elements automatic mode of operation. The standby feedwater pump, 2FWS-P1C, has been secured for preventive maintenance on its drive motor. While the maintenance activity is still in progress feedwater pump 2FWS-P1A trips. During the transient, reactor water level dips to 177 inches.

a. State whether this power level is within the capacity of the remaining feedwater pump, 2FWS-P1B. (.5)

b. State the required operator actions for this occurrence. (.5)

ANSWER:

a. Yes (.5)

b. Per N2-OP-3, H.1.0,

   a. Verify recirculation flow control valve partial closure initiated. (.2)
   b. If recirculation flow control valve fails to runback, manually close recirculation flow control valve on P602. (.05)
   c. Monitor reactor level is maintained at lower power level. (.2)
   d. Determine the cause of the feedwater pump trip. (.05)

REFERENCE: N2-OP-3, Condensate and Feedwater System, Rev. 4; Sections B, H.1.0 and H.5.0

5. QUESTION: (0.75 pts.)

The plant is operating at steady state 100% power.

What immediate actions should be taken by the operator if the air header pressure reaches 60 psig and decreasing? (0.75)

ANSWER:

Manually scram the reactor (0.25), close the outboard MSIVs (0.25), and follow N2-OP-101C (0.25).

REFERENCE: N2-OP-19, Instrument and Service Air System, Rev. 2; Sections H.2.3, I.4.0, I.5.0, I.6.0, I.7.0

6. QUESTION: (1.0 pts.)

The plant experienced a scram caused by an inadvertent full closure of all MSIVs. With all MSIVs being closed, plant cooldown is effected with RCIC and RHR in the steam condensing mode of operation. What actions should the operator take if the RHR heat exchanger pressure decreases to less than 200 psig? (Two (2) required) (1.0)

ANSWER:

1. Throttle open RHR heat exchanger steam supply valve as required to maintain heat exchanger pressure at 200 psig. (0.5)

2. If RHR heat exchanger pressure cannot be maintained at 200 psig, reduce RPV cooldown rate to prevent exceeding RHR heat exchanger service water differential temperature of 29°F. (0.5)

REFERENCE: N2-OP-31, Residual Heat Removal System, Rev. 5; Section E.9.24

7. QUESTION: (1.0 pts.)

The High Pressure Core Spray (HPCS) System initiates on low water level and injects into the reactor vessel. Reactor pressure is 850 psig when the following events occur:

* Annunciator 601718 HPCS REACTOR WATER LEVEL HIGH is received.
* HPCS pump injection valve CSH*MOV107, shuts on Level 8
* HPCS minimum flow bypass valve CSH*MOV105, opens

No malfunction has occurred in the HPCS System and the HPCS is aligned to CST. With regard to the HPCS System, what corrective actions should be taken by the operator? (4 required)

ANSWER:

a. Verify automatic response. (0.25)

b. Depress HPCS manual initiation seal-in reset pushbutton and verify white seal-in light goes out. (0.25)

c. Throttle open CSH*MOV110 and MOV112 to establish 6850 gpm HPCS system flow. (0.25

d. Verify minimum flow bypass valve CSH*MOV105 shut. (0.25)

REFERENCE: N2-OP-33, High Pressure Core Spray System, Rev. 3; Section I.6.0 & I.7.0

8. QUESTION: (0.75 pts.)

A containment isolation occurs due to LOCA signal following a primary system line break. An operator attempts to reduce drywell temperature by operating the Drywell Cooling System.

Describe what actions need to be taken to restore the Drywell Cooling System.

ANSWER:

(At P873), place Div. 1 and 2 Drywell Unit Cooling Water LOCA Override Keylock switches to "OVERRIDE". (0.25)

(At P873), open DW Cooler Inbd/Outbd valves. (0.25)

(At P873), (place the two Keylock Drywell Unit Cooler Fans GR1/2 LOCA Override switches to "OVERRIDE" then) start DRS unit coolers. (0.25)

REFERENCE: N2-OP-60, Drywell Cooling, Rev. 1, Section H.1.0
N2-OP-13, Reactor Building Closed Loop Cooling, Rev. 2, Section H.2.0

9. QUESTION: (1.0 pts.)

The reactor is operating steady state at 100% power when the following alarm annunciates:

851406 2ND PT HEATER 2A/2B/2C WATER LEVEL HI-HI

The operator observes that both blocking valves for the LP feedwater heater string, 2CNM-MOV32A and 2CNM-MOV33A shut.

1. What immediate actions should be taken by the operator to prevent fuel damage? (0.5)

2. What further action may be required? (0.5)

ANSWER:

1. Reduce recirculation flow. (0.5)
2. Insert control rods per the Reactor Analyst's instructions. (0.5)

REFERENCE: N2-OP-8, Feedwater Heaters and Extraction St^e^am Systems, Rev. 2, Section H.3.3 and I.8.0

10. QUESTION: (1.0 pts.)

The plant is operating at 100% power, when the following alarms annunciate:

851101	GENERATOR SEAL OIL PUMP 1/3/4 MOT OVERLOAD
851111	GENERATOR EMER SEAL OIL PUMP 2 RUNNING
205205	EMERG SEAL OIL PUMP RUNNING

What actions need to be taken based on the above conditions?

ANSWER:

1. Start or verify auto start of the ESOP. (0.2)

2. Verify ESOP is maintaining seal oil pressure. (0.2)

3. Place control switch for main seal oil pump (MSOP) in STOP. (0.2)

4. Shutdown the recirculation seal oil pump (RSOP) and seal oil vacuum pump (SOVP). (0.2)

5. Determine and correct the cause. (0.2)

REFERENCE: N2-OP-22D, Generator Hydrogen Seal Oil System, Rev. 1, Sections H.1.2, I.2.0, I.6.0 and I.8.0

11. QUESTION: (1.0 pts.)

The plant is in a normal startup from an outage. Reactor is critical with IRMs indicating 25 on Range 2. Following a rod withdrawal of one notch, the operator observes that IRM readings increase to 50 on Range 2 in 25 seconds.

1. Determine the reactor period resulting from the control rod withdrawal as stated above. (0.25) Show all calculations in your answer. (0.25)

2. What action(s) would the SSS direct the operator to take in response to a 50 second period? (0.25) State the reference. (0.25)

ANSWER:

1. Reactor Period = 1.44 X (Doubling Time), seconds
   = 1.44 X 25 seconds (0.25)

   Thus, Reactor Period is 36 seconds (0.25)

2. Stop pulling rods or insert rods as necessary (0.25) IAW OP-92, Section I.9.0 (0.25)

REFERENCE: N2-OP-101A, Plant Startup, Rev. 6, Section E.2.9
N2-OP-92, Neutron Monitoring, Rev. 1, Section I.9.0

12. QUESTION: (1.0 pts.)

Plant is at 100 percent power and offgas system train B is in service when the following alarm annunciates:

851306 OFF GAS SYSTEM TROUBLE

Simultaneously, on panel 2OFG-IPNL122, system flow indication as shown on 2OFG-FI3B is oscillating and the following alarms annunciate:

122201 HIGH INLET TEMPERATURE SYSTEM A & B
122202 HIGH INLET PRESSURE SYSTEM A & B

The operator acknowledged the annunciators and was able to reset annunciators HIGH INLET TEMPERATURE SYSTEM A & B and HIGH INLET PRESSURE SYSTEM A & B. System flow returns to normal without any further operator action.

1. Based on the information given above, determine what event has occurred. (0.5)

2. Justify your answer. (0.5)

ANSWER:

1. A hydrogen explosion. (0.5)

2. Multiple system trouble alarms occurring simultaneously with the high pressure alarm. (0.5)

REFERENCE: N2-OP-42, Section H.5.0, Section I.1.0-I.2.0, I.47.0

13. QUESTION: (2.0 pts.)

The Site Emergency Plan is activated. The Site Emergency Director declares an Unusual Event.

Who must the Control Room Communications Aide attempt to notify?

ANSWER:

He must attempt to notify the following: (0.25 points each)

Nuclear Security Department
System Power Control
Oswego County
New York State
Nine Mile Unit 1 Control Room
JAFNPP Control Room
Nuclear Regulatory Commission
NRC Senior Resident Inspector

REFERENCE: EPP-20, Emergency Notifications, Rev. 11, Section 6.0 and Figure 5

14. QUESTION: (1.0 pts.)

The plant was operating normally when a scram occurred. All rods are at position 00. You note the following nuclear instrumentation values:

SRM: detectors inserted, channels indicating about 1 X 10^4 cps with negative 80 second period.

IRM: detectors inserted, all channels on Range 3 and decreasing.

APRM: All recorders downscale, front panel UPSC or INOP amber indicating lights are on and white Downscale indicating lights are on, backpanel white INOP and DNSC lights are on.

Based on the above indications only, is an EOP Entry condition met? Justify your answer.

ANSWER:

(Even though the APRMs have failed) it is not necessary to enter EOPs. (0.50) (Alternate methods of determining reactor power below 4.0% are provided for.) From the given indications, reactor power can be determined to be less than 4.0% and decreasing. (0.50)

REFERENCE: N2-F^P-RPV, RPV Control, Rev. 2, Section RQ

15. QUESTION: (1.75 pts.)

You are informed by Nine Mile Point Unit 1 that they have declared an alert emergency condition. What actions should the Unit 2 SSS initiate?

ANSWER:

The SSS should declare a Sympathetic Alert, (1.0) and direct the Notification Fact Sheet, Part I be completed, (0.25) accomplish the NRC initial notification (0.25) and direct Emergency Staffing level I - Control Room. (0.25)

REFERENCE: EAP-1, Activation and Direction of the Emergency Plans, Rev. 6, Sections 3.3 & 6.0
SEP, Site Emergency Plan, Rev. 20, Section 4.1.3

16. QUESTION: (0.75 pts.)

The plant is in normal operation while maintenance is performing repairs on reactor building above refueling floor exhaust fan 2HVR-FN5A, which is inoperable. An error occurs, tripping fan 2HVR-FN5B. Within 15 minutes, maintenance identifies the cause of the trip and reports fan 2HVR-FN5B is ready for restart.

List the actions the operator must perform prior to restarting an exhaust fan.

ANSWER:

1. Bypass the Rx Bldg. low flow interlocks. (0.15)

2. Place the control switch for 2HVR*UC413B to NORMAL-AFTER-STOP. (0.15)

3. Secure one train of standby gas treatment (per section G.1.0 of N2-OP-61B). (0.15)

4. Place or verify control switches for 2HVR-FN1 (A, B, and C), 2HVR-FN2 (A and B) and 2HVR-FN5 (A and B) in pull-to-lock. (0.15)

5. Open dampers 2HVR-AOD1 A and B, 9A and B and 10A and B. (0.15)

REFERENCE: N2-OP-52, Reactor Building Ventilation, Rev. 3, Sections H.4.1-H.4.5 & I.62.0
N2-OP-61B, Standby Gas Treatment, Rev. 5, Section B

Name: ____________________

Soc. Sec. #: ____________

Course: NRC/Annual SRO Requal Exam Section B-4

Date: ____________

Grade: ____________

File Section: ____________

Trainees will be judged to have willfully violated the integrity of an examination if they are found to have:

a) Utilized unauthorized documents during the examination.

b) Secured unauthorized documents for the purpose of accessibility during an examination.

c) Solicited examination information from other trainees or any other individuals.

d) Provided examination information to other trainees during an examination.

e) Reviewed or attempted to review materials that are un-authorized, including the examination prior to implementation, the examination answer key, or the answers developed by any other trainee during the examination.

I have read and understand the above.

Signature: ____________________

Date: ____________

1. QUESTION: (1.5 pts.)

Immediately following a primary system line break inside the drywell, a containment isolation due to high drywell pressure, and an automatic actuation of SBGT occur. An operator attempts to rapidly reduce primary containment pressure at the direction of the SSS.

Determine what actions need to be taken by the operator in order to open the following valves: (1.5)

DW Purge Inboard Outlet Isol Vlv 2CPS*AOV108 and DW Purge Outboard Outlet Isol Vlv 2CPS*AOV110.
Suppr Pool Purge Inboard Isol Vlv 2CPS*AOV109 and Suppr Pool Outboard Outlet Isol Vlv 2CPS*AOV111.

ANSWER:

1. (At Panels P873 and P875), place Purge Outboard/Inboard Valve override switches to OVERRIDE Positions. (0.75)

2. (At Panels P859 and P861), place a jumper to remove DW pressure isolation signal of 2IAS*SOV168 and 2IAS*SOV180, Instrument Air Outboard and Inboard Isolation Valves. (0.75)

NOTE: Accept valve name or number as correct answer

REFERENCE: N2-OP-61A, Primary Containment Ventilation, Purge and Nitrogen System, Rev. 3; Section H.1.0 and H.3.0
N2-EOP-PC, Primary Containment Control; Rev. 2

2. QUESTION: (1.0 pts.)

With the Reactor in Mode 1:

1. What action is required after determining that a jet pump is inoperable? (0.5)

2. Explain how an inoperable jet pump affects the core reflood capability.

ANSWER:

1. The reactor must be in at least HOT SHUTDOWN within 12 hours after a jet pump is determined to be inoperable. (0.5)

2. May not achieve 2/3 core coverage (post LOCA) (0.5)

(NOTE: Also accept drawing similar to attached diagram.)

REFERENCE: Technical Specifications 3.4.1.2
FSAR: Section 3.9.5.3.1B and Figure 3.9B-2

3. QUESTION: (1.0 pts.)

The plant is operating at steady state conditions with 60% power. Feedwater pumps 2FWS-P1A and 2FWS-P1B are in 3-elements automatic mode of operation. The standby feedwater pump, 2FWS-P1C, has been secured for preventive maintenance on its drive motor. While the maintenance activity is still in progress feedwater pump 2FWS-P1A trips. During the transient, reactor water level dips to 177 inches.

a. State whether this power level is within the capacity of the remaining feedwater pump, 2FWS-P1B. (.5)

b. State the required operator actions for this occurrence. (.5)

ANSWER:

a. Yes (.5)

b.. Per N2-OP-3, H.1.0,

a. Verify recirculation flow control valve partial closure initiated. (.2)
b. If recirculation flow control valve fails to runback, manually close recirculation flow control valve on P602. (.05)
c. Monitor reactor level is maintained at lower power level. (.2)
d. Determine the cause of the feedwater pump trip. (.05)

REFERENCE: N2-OP-3, Condensate and Feedwater System, Rev. 4; Sections B, H.1.0 and H.5.0

4. QUESTION: (1.0 pts.)

The plant is operating at steady state 100% power when the following annunciators alarm:

851229	INSTR AIR SYSTEM TROUBLE
851238	INSTR AIR COMPRESSOR 1A/1B/1C AUTO START
851228	INSTR AIR CPSR 1A/1B/1C AUTO TRIP/FAIL TO START
851239	SER AIR SYS 2IAS-AOV 171 CLOSED

A reactor scram will occur if the air pressure continues to drop.

State the system responses to a loss of instrument air which results in a scram signal to the Reactor Protection System. Two (2) systems required. Any two (2) full credit.

ANSWER:

1. Individual MSIV will close due to loss of instrument air to each MSIV pilot valve and this will cause a scram because of MSIV closure. (0.5)

2. Feedwater pump minimum flow valves will fail full open due to loss of instrument air and this will cause a loss of feedwater to reactor and the resulting scram on low water level. (0.5)

3. Scram inlet and outlet valves will fail open due to loss of instrument air and this will cause the Scram Discharge Volume to fill to the RPS Scram Setpoint. (0.5)

REFERENCE: N2-OP-19, Instrument and Service Air System, Rev. 2; Sections H.2.0, I.4.0, I.5.0, I.6.0, I.7.0

5. QUESTION: (1.0 pts.)

The plant is in the process of cooldown. RHR B is currently operating in the shutdown cooling mode with the following conditions:

- RHR B total flow is approximately 7450 gpm.
- RHR B to reactor head spray valve RHS*MOV104 is shut.
- RHR B heat exchanger service water outlet temp. is 110 degrees F.
- RHR B heat exchanger inlet bypass valve RHS*MOV8B is about 80% open.

Both recirculation pumps are inoperable. The reactor coolant temperature is 300 degrees F and decreasing at 90 degrees F per hour over the last 15 minutes.

List two actions the operator should take in order to decrease the reactor cooldown rate without causing RPV thermal stratification. (1.0)

ANSWER:

Any two (2) at .5 points each for a total of 1.0 point.

1. Open the RHR B heat exchanger bypass valve (MOV8B).

2. Throttle RHR B heat exchanger service water flow outlet (2SWP*MOV33B) (if the RHR B heat exchanger bypass valve is full open).

3. With little or no decay heat it may also be necessary to close RHR B heat exchanger outlet valve (2RHS*MOV12B).

<u>NOTE</u>: Either valve number or name for Full Credit

REFERENCE: N2-OP-31. Residual Heat Removal System, Rev. 5; Section D.6.0 and E.8.18

6. QUESTION: (1.0 pts.)

The High Pressure Core Spray (HPCS) System initiates on low water level and injects into the reactor vessel. Reactor pressure is 850 psig when the following events occur:

* Annunciator 601718 HPCS REACTOR WATER LEVEL HIGH is received.
* HPCS pump injection valve CSH*MOV107, shuts on Level 8
* HPCS minimum flow bypass valve CSH*MOV105, opens

No malfunction has occurred in the HPCS System and the HPCS is aligned to CST. With regard to the HPCS System, what corrective actions should be taken by the operator? (4 required)

ANSWER:

a. Verify automatic response. (0.25)

b. Depress HPCS manual initiation seal-in reset pushbutton and verify white seal-in light goes out. (0.25)

c. Throttle open CSH*MOV110 and MOV112 to establish 6850 gpm HPCS system flow. (0.25

d. Verify minimum flow bypass valve CSH*MOV105 shut. (0.25)

REFERENCE: N2-OP-33, High Pressure Core Spray System, Rev. 3; Section I.6.0 & I.7.0

7. QUESTION: (0.75 pts.)

A containment isolation occurs due to LOCA signal following a primary system line break. An operator attempts to reduce drywell temperature by operating the Drywell Cooling System.

Describe what actions need to be taken to restore the Drywell Cooling System.

ANSWER:

(At P873), place Div. 1 and 2 Drywell Unit Cooling Water LOCA Override Keylock switches to "OVERRIDE". (0.25)

(At P873), open DW Cooler Inbd/Outbd valves. (0.25)

(At P873), (place the two Keylock Drywell Unit Cooler Fans GR1/2 LOCA Override switches to "OVERRIDE" then) start DRS unit coolers. (0.25)

REFERENCE: N2-OP-60, Drywell Cooling, Rev. 1, Section H.1.0
N2-OP-13, Reactor Building Closed Loop Cooling, Rev. 2, Section H.2.0

8. QUESTION: (1.0 pts.)

The reactor is operating steady state at 100% power when the following alarm annunciates:

851406 2ND PT HEATER 2A/2B/2C WATER LEVEL HI-HI

The operator observes that both blocking valves for the LP feedwater heater string, 2CNM-MOV32A and 2CNM-MOV33A shut.

1. What immediate actions should be taken by the operator to prevent fuel damage? (0.5)

2. What further action may be required? (0.5)

ANSWER:

1. Reduce recirculation flow. (0.5)
2. Insert control rods per the Reactor Analyst's instructions. (0.5)

REFERENCE: N2-OP-8, Feedwater Heaters and Extraction Steam Systems, Rev. 2, Section H.3.3 and I.8.0

9. QUESTION: (1.0 pts.)

The plant is operating at 100% power, when the following alarms annunciate:

851101	GENERATOR SEAL OIL PUMP 1/3/4 MOT OVERLOAD
851111	GENERATOR EMER SEAL OIL PUMP 2 RUNNING
205205	EMERG SEAL OIL PUMP RUNNING

What actions need to be taken based on the above conditions?

ANSWER:

1. Start or verify auto start of the ESOP. (0.2)

2. Verify ESOP is maintaining seal oil pressure. (0.2)

3. Place control switch for main seal oil pump (MSOP) in STOP. (0.2)

4. Shutdown the recirculation seal oil pump (RSOP) and seal oil vacuum pump (SOVP). (0.2)

5. Determine and correct the cause. (0.2)

REFERENCE: N2-OP-22D, Generator Hydrogen Seal Oil System, Rev. 1, Sections H.1.2, I.2.0, I.6.0 and I.8.0

10. QUESTION: (1.0 pts.)

A normal reactor startup is in progress with both recirculation pumps at low speed and the following conditions:

- Generator output is 225 MWe.
- Generator hydrogen gas pressure is 75 psig.
- Generator Power Factor (PF) is 0.90.
- Stator cooling water conductivity is 0.2 micro mho/cm
- Alarm 851135, STTR CLG WTR PUMP 1A/1B AUTO TRIP/FAIL TO START, is received.

The auto start of Standby Stator Cooling Water pump does not occur. Determine the operator action required to continue generator operation. (1.0)

ANSWER:

Within an hour, (0.5) attempt to restore stator cooling water flow by starting a pump at (P851). (0.5)

REFERENCE: N2-OP-26 , Generator Stator and Exciter Rectifier Cooling System, Rev. 1; Sections I.15.0
N2-OP-68, Main Generator, Exciter, Main Transformer, 345 KV Yard, and Generator/Unit Protection, Rev. 0, Section H.3.0

11. QUESTION: (1.0 pts.)

The plant is in a normal startup from an outage. Reactor is critical with IRMs indicating 25 on Range 2. Following a rod withdrawal of one notch, the operator observes that IRM readings increase to 50 on Range 2 in 25 seconds.

1. Determine the reactor period resulting from the control rod withdrawal as stated above. (0.25) Show all calculations in your answer. (0.25)

2. What action(s) would the SSS direct the operator to take in response to a 50 second period? (0.25) State the reference. (0.25)

ANSWER:

1. Reactor Period = 1.44 X (Doubling Time), seconds
   = 1.44 X 25 seconds (0.25)

   Thus, Reactor Period is 36 seconds (0.25)

2. Stop pulling rods or insert rods as necessary (0.25) IAW OP-92, Section I.9.0 (0.25)

REFERENCE: N2-OP-101A, Plant Startup, Rev. 6, Section E.2.9
N2-OP-92, Neutron Monitoring, Rev. 1, Section I.9.0

12. QUESTION: (1.0 pts.)

Plant is at 100 percent power and offgas system train B is in service when the following alarm annunciates:

851306 OFF GAS SYSTEM TROUBLE

Simultaneously, on panel 2OFG-IPNL122, system flow indication as shown on 2OFG-FI3B is oscillating and the following alarms annunciate:

122201 HIGH INLET TEMPERATURE SYSTEM A & B
122202 HIGH INLET PRESSURE SYSTEM A & B

The operator acknowledged the annunciators and was able to reset annunciators HIGH INLET TEMPERATURE SYSTEM A & B and HIGH INLET PRESSURE SYSTEM A & B. System flow returns to normal without any further operator action.

1. Based on the information given above, determine what event has occurred. (0.5)

2. Justify your answer. (0.5)

ANSWER:

1. A hydrogen explosion. (0.5)

2. Multiple system trouble alarms occurring simultaneously with the high pressure alarm. (0.5)

REFERENCE: N2-OP-42, Section H.5.0, Section I.1.0-I.2.0, I.47.0

13. QUESTION: (1.75 pts.)

You are informed by Nine Mile Point Unit 1 that they have declared an alert emergency condition. What actions should the Unit 2 SSS initiate?

ANSWER:

The SSS should declare a Sympathetic Alert, (1.0) and direct the Notification Fact Sheet, Part I be completed, (0.25) accomplish the NRC initial notification (0.25) and direct Emergency Staffing level I - Control Room. (0.25)

REFERENCE: EAP-1, Activation and Direction of the Emergency Plans, Rev. 6, Sections 3.3 & 6.0
SEP, Site Emergency Plan, Rev. 20, Section 4.1.3

14. QUESTION: (1.5 pts.)

The operator is deliberately lowering the RPV water level to reduce reactor power to less than 4% per N2-EOP-C7, Level/Power Control. An MSIV isolation subsequently occurs as a result of low water level.

Is it permissible to reopen the MSIV's to reestablish the main condenser as a heat sink? Justify your answer.

ANSWER:

Yes (0.5)

1. Boron injection is required (as evidenced by use of EOP-C7). (0.25)

2. The main condenser is available because it was previously being used. (0.25)

3. There has been no indication of fuel failure because the MSIV's closed on low reactor water level. (0.25)

4. There is no evidence of a steam line break because the MSIV's closed on low reactor water level. (0.25)

REFERENCE: N2-EOP-RPV, RPV Control, Section RP and RQ, Rev. 2
N2-EOP-C7, Level/Power Control; Rev. 2

15. QUESTION: (1.0 pts.)

The N2-EOPs have been entered with the following conditions:

- Suppression pool water level is above EL. 217 ft.

A 30 psi differential pressure is reached between the Suppression chamber pressure (as indicated at CMS*PR7B on P898), and the Primary Containment Inlet N2 Pressure (as indicated at 2CPS-PI127 on P873).

What action should the SSS direct?

ANSWER:

The SSS should terminate all injection into RPV from sources external to the primary containment. (1.0)

REFERENCE: N2-EOP-PC, Primary Containment Control, Rev. 2, Section SPL
N2-OP-82, Containment Atmospheric Monitoring, Rev. 3, Section H.3.0

16. QUESTION: (1.5 pts.)

The reactor is in an accident situation with the following conditions:

Reactor pressure	950 psig
Vessel water level	-14 inches and decreasing
HPCS, Feedwater, RCIC	Out of service
LPCS and RHR pumps	Running, but not injecting
All Control Rods	Position 00

A small LOCA has occurred and the CRD system is not sufficient to return the water level to normal. State which contingencies of the Emergency Operating Procedures are being used and what action will be directed to be taken to return the reactor water level to normal. (1.5 pts.)

ANSWER:

In accordance with (RPV Control, Section RL), EOP-C1 Level Restoration (0.50), and EOP-C2 Emergency Depressurization (0.50), emergency depressurization will be performed, (allowing low pressure ECCS systems injection to return level to the normal band (159.3 to 202.3)). (0.50)

REFERENCE: N2-EOP-RPV, Rev. 2, RPV Control, Section RL
N2-EOP-C1, Level Restoration, Rev. 2
N2-EOP-C2, Emergency RPV Depressurization, Rev. 2

ATTACHMENT 6

Licensee Letter

Dated August 10, 1989