June 1, 2005

Mr. David A. Christian
Sr. Vice President and Chief Nuclear Officer
Dominion Resources
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE UNIT 2 REACTOR OPERATOR AND SENIOR REACTOR OPERATOR INITIAL EXAMINATION REPORT NO. 05000336/2005301

Dear Mr. Christian:

This report transmits the results of the Reactor Operator (RO) and Senior Reactor Operator (SRO) licensing examination conducted by the NRC during the period of March 18 - 21, 2005. This examination addressed areas important to public health and safety and was developed and administered using the guidelines of the "Examination Standards for Power Reactors" (NUREG-1021, Revision 9).

Based on the results of the examination, both Senior Reactor Operator and six of the seven Reactor Operator applicants passed all portions of the examination. One Reactor Operator applicant failed the site-specific written examination. The nine applicants included seven ROs, and two instant SROs. Mr. D'Antonio discussed performance insights observed during the examination with training department personnel on March 21, 2005. On April 21, 2005, final examination results, including an individual license number for the one individual who was not still pending completion of waiver requirements, were given during a telephone call between Mr. D'Antonio and Mr. Trad Horner.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). These records include the final examination and are available in ADAMS Package No. ML042440181 (RO and SRO Written - Accession Number ML051400118; RO and SRO Operating Section A - Accession Number ML051400126; RO and SRO Operating Section B - Accession Number ML051400132; and RO and SRO Operating Section C - Accession Number ML051400134), and Facility Post Examination Comments on the Written Exams - Accession No. ML051390229. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Should you have any questions regarding this examination, please contact me at (610) 337-5183, or by e-mail at RJC@NRC.GOV.

Sincerely,

/RA/

Richard J. Conte, Chief
Operations Branch
Division of Reactor Safety

Docket No. 50-336
License No. DPR-65

Enclosure: Initial Examination Report No. 05000336/2005301

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DATE	05/23/05		05/05/05		05/31/05		06/01/05	

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-336
License No.	DPR-65
Report No.	05000336/2005301
Licensee:	Dominion Nuclear Connecticut, Inc.
Facility:	Millstone Unit 2
Dates:	March 18, 2005 (Written Examination Administration) March 21-March 24, 2005 (Operating Test Administration) March 28 - April 21, 2005 (Examination Grading) April 18, 2005 (Final facility post exam comment on JPMs. End of Examination Period)
Examiners:	Joseph M. D'Antonio, Operations Engineer (Chief Examiner) Peter A. Presby, Operations Engineer (Under Instruction) Donald E. Jackson, Senior Operations Engineer
Approved by:	Richard J. Conte, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000336/2005301; March 18-24, 2005; Millstone Point Unit 2; Initial Operator Licensing Examination. Eight of nine applicants passed the examination (six of seven reactor operators, and two of two senior reactor operators).

The written examinations were administered by the facility and the operating tests were administered by three NRC region-based examiners. There were no inspection findings of significance associated with the examinations.

REPORT DETAILS

1. REACTOR SAFETY

Mitigating Systems - Reactor Operator (RO) and Senior Reactor Operator (SRO) Initial License Examination

a. Scope of Review

The NRC examination team developed the site-specific written examination; the facility developed the operating examination. Together with Millstone Unit 2 training and operations personnel, the examiners verified or ensured, as applicable, the following:

- The examination was prepared and developed in accordance with the guidelines of Revision 9 of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors." A review was conducted both in the Region I office and at the Millstone Unit 2 plant and training facility. Final resolution of comments and incorporation of test revisions were conducted during and following the onsite preparation week.
- Simulation facility operation was proper.
- A test item analysis was completed on the written examination for feedback into the systems approach to training program.
- Examination security requirements were met.

The NRC examiners administered the operating portion of the examination to all applicants from March 21 - March 24, 2005. The written examination was previously administered by the Millstone Unit 2 training staff on March 18, 2005.

b. Findings

Grading and Results

Eight of nine applicants (both SROs and six of seven ROs) passed all portions of the initial licensing examination.

Examination Administration and Performance

The following errors occurred during administration of the examinations:

In one scenario, the simulator operator inserted an incorrect malfunction. The scenario was accepted as run because the examiners observed that the incorrect malfunction was somewhat more challenging to the applicants than the intended malfunction.

In one scenario, the simulator operator inadvertently omitted an ATWT malfunction. The remainder of the scenario was not dependent on this ATWT and was completed. However, this omission required an additional partial scenario to be run on this crew to capture the missing malfunction.

One JPM had not been corrected to incorporate comments from the validation week to run it as a “normal” rather than an “alternate path” JPM. This resulted in some confusion for the examiner and unnecessary stress for the applicant.

During the performance of one JPM, the examiners observed that attempting to synchronize an EDG by following the procedure exactly as written, would sometimes result in failure of the diesel breaker to close. The facility has determined that this is a procedural problem rather than a simulator fidelity issue. CR-05-02859

Simulator

Several simulator performance discrepancies or procedural problems were noted during the preparation and administration of the examinations as noted in Attachment 3, simulation facility report. This area is unresolved pending further review by the facility in order for the NRC staff to determine if these issues are performance deficiencies. **URI 05000336/2005301-01**

Technical Specifications

A potential issue was noted with the Technical Specifications during development of the written examination as follows:

- Wide Range Nuclear Instruments (WRNIs) are required to be operable in modes 3, 4, 5.
- Power Range Nuclear Instruments (PRNIs) are required to be operable in modes 1, 2.
- Mode 2 is defined as Keff > .99
- Power Range Nuclear Instruments come on scale at .1% power.

It thus appears possible to be in compliance with Technical Specifications with the plant in Mode 2, with power still below the range of the PRNIs, but with no requirement for operable WRNIs. This issue has been entered into the facility corrective action process as CR-05-3176, and is unresolved pending further review to determine if the NRC’s understanding is correct, if the issue is acceptable, or if the issue is a performance deficiency. **URI05000336/2005301-02.**

4OA6 Exit Meeting Summary

On April 21, 2005, the NRC provided conclusions and examination results to Millstone Unit 2 management representatives via telephone. License numbers for one of the eight applicants that passed all portions of the initial licensing examination were also provided during this time. License numbers for the remaining seven applicants were withheld pending completion of the requirements of their waivers. Mr. Trad Horner was informed that when the NRC is notified in writing that these requirements have been completed by these seven individuals, their licenses would be issued. The ninth applicant passed all sections of the operating portion, but failed the written portion of the initial licensing examination and, therefore, was denied a license at this time.

The NRC expressed appreciation for the cooperation and assistance that was provided during the preparation and administration of the examination by the licensee's training staff.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

Michael Wilson, Manager, Operator Training
Jim Kunze, Unit Two Operations
Pete Strickland, Unit 2 Shift Manager
Richard Ashey, Unit 2 Instructor
Daniel Pantalone, Unit 2 Instructor

NRC

Joseph M. D'Antonio, Operations Engineer
Donald E. Jackson, Senior Operations Engineer
Peter A. Presby, Operations Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

ITEM NUMBER	TYPE	DESCRIPTION
05000336/2005301-01	URI	Simulator deficiencies identified during NRC exam process.
05000336/2005301-02	URI	Potential gap in Technical Specification requirements for Nuclear Instrument Operability.

Attachment 2

Post Exam Comments

Facility Post Exam Comments and NRC Resolution

An NRC-developed written exam was administered to 7 RO applicants and 2 SROI applicants on Monday, March 18, 2005. The facility-developed operational exam was administered to these same applicants during the following week, March 21, 2005 - March 24, 2005).

The facility submitted post exam comments for the following 5 questions on the RO written exam and also for the following job performance measures in the admin area:

Written Exam Question Number	Admin Job Performance Measure
32	A2SRO
37	A1RO
66	A2RO
71	
72	

The facility-proposed change to Question #32 on the written exam is accepted as proposed. All other facility-proposed changes to written exam questions (#38, 66, 71 and 72) are not accepted. The facility-proposed change to the Admin JPMs are accepted. The final facility comment was received on April 18, 2005.

1. Written Exam Question #32

Question.

The plant is operating at full power with all equipment functional, except for the 'B' HPSI Pump, which is OOS for maintenance.

Then, a large break LOCA occurs combined with a loss of Bus 24D (due to an electrical fault on 24D).

Which one of the choices correctly completes the following statement regarding the impact of the loss of ECCS pumps.

Ten hours after the event, a loss of the only available ___________ adversely affect long term core cooling because the remaining ______________.

A HPSI pump would, LPSI pump does NOT have a system flowpath for boron precipitation control.
B HPSI pump would, LPSI pump could NOT be procedurally realigned for boron precipitation control via hot leg injection. **[Key answer]**
C LPSI pump would NOT, HPSI pump is preferred for boron precipitation control.

D LPSI pump would NOT, HPSI pump could be procedurally realigned for boron precipitation control via hot leg injection.

Facility Comments:

Comments:

The justification for the original correct choice 'B' states that although it is physically possible to align 'A' LPSI pump for Hot Leg Injection, there is no procedural guidance to do so. The reason there is no procedural guidance for this system alignment under the given conditions (24D deenergized), is because without 24D power it is not possible to align LPSI for Hot Leg Injection. One of the major valves in the flow path to achieve this alignment (2-SI-652) is powered by Facility Two (24D) and is located in containment (inaccessible with an LOCA). The procedure section the operators are referred to with 24D lost assumes Facility One HPSI & LPSI pumps are both available.

Therefore, as stated in Choice 'A', if the available HPSI is lost, core cooling would be in jeopardy because there is no system flowpath for Hot Leg Injection with the available LPSI pump.

Recommendation:

Based on the above explanation, we believe both Choice 'A' and Choice 'B' are technically correct for the given information in the stem.

NRC Response:

Facility recommendation for multiple correct answers is **ACCEPTED**.

The question stem states that Bus 24D is de-energized. Motor-operated gate valve, 2-SI-652, is in the flowpath for LPSI hot leg recirculation and must be opened to set up that flowpath. However, the valve operator is powered from the deenergized bus and, therefore, 2-SI-652 cannot be opened during the given event. These facts establish the conditions necessary for Choice 'A' to be a correct answer for the question. Choice 'B' remains an equally valid correct answer. Choice A and Choice B are both correct.

References:

- EOP-2541, Appendix 18, "Simultaneous Hot and Cold Leg Injection," Revision 000
- SDC-00-C, "Shutdown Cooling System" Lesson Material, Revision 3

2. Written Exam Question #38

Question:

Given the following plant conditions:

- 100% power
- SG levels at setpoint
- Steam flow and feed flow matched
- SG2 Feed Flow Transmitter FT-5269A output fails high

With NO operator actions, which of the following describes the expected plant response?

A SG level lowers, but stabilizes above the low level reactor trip. **[Key answer]**
B SG level lowers to the low level reactor trip.
C SG level rises, but stabilizes below the high level turbine trip.
D SG level rises to the high level turbine trip.

Facility Comments:

Comments:
Our original concern with this question was that it required the Candidate to go beyond the knowledge solicited by the K/A (i.e., how will the system respond to the failed instrument), and make a "quantitative" judgment as to the ***amount*** the system will respond to the failed instrument. That was the reason behind our suggested rewrite of the question (see attached) to eliminate the choice of "...low level reactor trip" and replace this distractor with one that was clearly wrong in its magnitude of response. In disallowing this suggested change, the Candidate was forced into a quantitative judgment, which depending on the specific tuning of the system for that operating cycle, could result in either Choice "A" or Choice "B" being the correct response.

Recommendation:
Based on the above explanation, we believe both Choice 'A' and Choice 'B' are acceptable answers for the given information in the stem.

NRC Response:

Facility recommendation for multiple correct answers is **NOT ACCEPTED**.

Question matches the selected K/A in that it tests an applicant's knowledge of the design response to a failure high of a feed flow input to the SG level control system, which controls feed regulating valve position.

Selected K/A: 059.K4.08

Attachment

Knowledge of MFW design feature(s) and/or interlock(s) which provide for the following: Feedwater regulatory valve operation (on basis of steam flow, feed flow mismatch)

The facility concern would be understandable if whether or not the plant tripped for this malfunction a borderline situation, but it is not - this malfunction does not approach the trip setpoint. It is reasonable and appropriate to expect a license applicant to be able to have some idea of the magnitude of pant response to an instrument failure, and in this case to identify whether or not this particular SGWLC instrument failure will result in an automatic trip. In order to correctly answer the question, the applicant must understand that the outputs of two feed flow transmitters are averaged to become a single input to the steam / feed mismatch portion of the three-element level control system. Further, the applicant must realize that at 100% reactor power each feed flow transmitter output is near the high end of its instrument range (100% power reading 5.9 E6 lbs/hr, with full range reading 6.0E6 lbs/hr). With this information an applicant can easily determine that a failure high of a feed flow transmitter will cause a relatively small change to the averaged feed flow signal. The feed regulating valve will begin to throttle closed in response to the change. However, the level input signal will cause the feed regulating valve to reopen with only a small change in actual level.

The facility staff ran a failure high malfunction on one feed flow transmitter for #2 SG on the simulator from 100% power for demonstration purposes. No operator action was taken. #2 SG level was observed to drop from 70% NR level to 67.5% NR level over approximately 2 minutes before turning and beginning to trend back to setpoint. No reactor trip occurred. The low level reactor trip is set to occur when SG level lowers to 49.5% NR level (TS requires setpoint ≥48.5%).

Given that the failure only results in a 3% change in SG level and that a 20.5% to 21.5% level change would be required to initiate the reactor trip, the distractor “B” answer (that level will lower to the low level reactor trip) is clearly wrong in its magnitude of response and could not be a correct response regardless of “specific tuning for that operating cycle.”

References:
- MP2 Tech Specs, Section 2.2, “Limiting Safety System Settings”
- RPS-01-C, “Reactor Protection System” Lesson Material Revision 6
- FWC-01-C, "Feedwater Control System,” Lesson Material Revision 2

3. Written Exam Question #66

Question:

Unit 2 is conducting a reactor start up. Given the following events and conditions:

- Wide range (WR) logarithmic nuclear instrument (NI) channels C and D are out of service
- The reactor is not yet critical

- The ECP expected critical rod height is 100 steps on Regulating Group 6
- Regulating Group 4 is withdrawn to 60 steps
- WR NI Channel A failed low

WRL NI Channel A	<1.0E-1	CPS
WRL NI Channel B	6.2E2	CPS

Which one of the following statements correctly describes the required action (if any) required to comply with TECHNICAL SPECIFICATIONS?

A Immediately trip the reactor.
B Insert all control rods and shutdown the reactor.
C Stop the startup until WRL NI Channel A has been repaired. NO other actions are required.
D Immediately ensure adequate shutdown margin. **[Key answer]**

Facility Comments:

Comments:
Choice "D" (the correct answer) is directly related to ACTION 4 for verifying compliance with TS 3.1.1.1 SHUTDOWN MARGIN, which is applicable for the current mode of operation (MODE 3). Choice D is an acceptable answer but is not a complete answer, taken by itself, because it implies that startup may continue.

Choice B "Insert all control rods and shutdown the reactor" is also an acceptable response because it is a proceduralized action of OP 2202 "Reactor Startup IPTE" and is the conservative philosophy of operations in DNAP-1410 "Reactivity Management."

These actions prevent non-compliance with TS LCO 3.0.4 and its BASIS, which requires exercise of good practice in restoring systems or components to OPERABLE status before plant startup.

The Examinee may have been more compelled to place emphasis on compliance with reactor startup termination than shutdown margin verification because the startup procedure had just previously verified adequate SDM and has controls in place to continuously monitor conditions that could affect SDM (e.g., ECP). It would be unacceptable to maintain current plant conditions considering the time required to re-verify SDM by obtaining and analyzing an RCS boron sample.

Recommendation:
Accept Choice D and Choice B

NRC Response:

Facility recommendation for multiple correct answers is **NOT ACCEPTED**.

The facility is reading “implications” into the distractors which are not stated, and also appear to be reading implications of required actions (insert rods to shutdown the reactor) into their procedures when such actions are not stated. This question specifically asks the required actions to satisfy TECHNICAL SPECIFICATIONS. The facility desired second correct answer “insert all rods and shutdown the reactor” does not satisfy the Technical Specification requirement to verify shutdown margin. Inserting rods is most certainly a prudent action, but that action is not explicitly required by any procedure referenced by the licensee as a response to loss of WRNIs , does NOT satisfy the TS, and recognition of that fact was the whole point of this question. The requirement for wide range logarithmic nuclear instruments in MODE 3 per Technical Specifications is for a minimum of 2 OPERABLE channels. With less than 2 OPERABLE channels, TS 3.3.1 Action 4 directs **immediate** verification of shutdown margin requirements. While there are additional actions that may be taken as a prudent response to inadequate instrumentation during a reactor startup, these actions are not required to comply with technical specifications, and are not explicitly required by any facility procedure. Further, Choice B cannot be considered a correct answer to the question because it does not include the actions required by technical specifications.

References:
- MP2 Technical Specifications, Table 3.3-1, “Reactor Protective Instrumentation”

4. Written Exam Question #71

<u>Question</u>:

A transfer of a new fuel assembly is in progress from one location in the spent fuel pool to another using OP-2303B, "SFP Fuel Handling Operations.” The operator raises the hoist with the desired assembly grappled until upward motion is stopped by the upper limit switch interlock.

What must be done next?

A Release hoist raise switch, use the bridge/trolley controls to move to destination.
B Stop all hoist and crane movement and notify Reactor Engineering immediately. **[Key answer]**
C Lower assembly into initial location and contact Reactor Engineering for resolution.
D Slowly lower hoist until load cell indicates 250 to 290 pounds, then continue move.

<u>Facility Comments</u>:

Comments:
Choice B (the correct answer) is an acceptable answer. It correctly describes the required actions if the SFP Platform Crane Operator fails to stop when the stainless

steel hose clamp on fuel handling tool is level with the top of SFP platform crane safety rail.

Choice A is also an acceptable response. The stem of the question does not provide any information regarding a human performance error on the part of the SFP Platform Crane Operator (i.e., failure to STOP when the stainless steel hose clamp on fuel handling tool is level with the top of SFP platform crane safety rail). It is reasonable for the examinee to assume that operations are proceeding as expected and that the next action would be to move the bridge/trolley to position the fuel over its final rack location (a move that is not prevented by interlock).

Other considerations of OP 2303B “SFP Fuel Handling Operations”:

> Level of Use Reference; the procedure shall be readily available to the user, in the area where the work activity is being performed, such that the user can obtain a copy of the document as needed to perform the procedure.
>
> Prerequisite 2.1.2: All personnel participating in fuel handling have been briefed and are thoroughly familiar with this procedure and individual responsibilities.

Examinees were required to answer without the use of reference material.

Recommendation:
Accept Choice B and Choice A

Justification:
Examinees who selected Choice B exhibited knowledge of the new and spent fuel movement procedures and also knowledge of fuel handling equipment interlocks.

Examinees who selected Choice A exhibited knowledge of the new and spent fuel movement procedures by correctly identifying the next procedurally directed step, considering that they were not cued that the SFP Platform Crane Operator had incorrectly performed the previous step.

<u>NRC Response</u>:

Facility recommendation for multiple correct answers is **NOT ACCEPTED**.

Question matches the selected K/A in that it tests an applicant’s knowledge of an important operational restriction contained within spent fuel handling procedures.

Selected K/A: Generic 2.2.28
Knowledge of new and spent fuel movement procedures.

Facility comments that it is reasonable for an applicant to assume that operations are proceeding as expected since the question stem does not state that a human performance error has occurred. However, the applicant’s ability to identify that fuel movement being stopped by an interlock is an abnormal situation is central to the knowledge being tested. It is not an assumption, but a determination the applicant was expected to make based on the conditions in the stem of the question. An applicant with the expected level of knowledge regarding fuel handling operations and, in particular, knowledge of important precautions contained in OP-2303B, "SFP Fuel Handling Operations," would recognize that the described actions are prohibited and that the procedure requires **immediate** notification of Reactor Engineering.

Further, the statement "do *not* attempt to lower the assembly into the upender or storage rack" is contained in both the precaution (3.1) and in the caution prior to the step for raising the assembly (4.2.12). Step 4.2.14 directs the operator to stop all hoist and crane movement and notify reactor engineering immediately if the upper limit switch interlock stops hoist motion.

The facility's contention that Choice A is also an acceptable response is incorrect. There is no ambiguity within the procedure regarding required actions when upward motion of a grappled assembly is stopped by the upper limit switch interlock.

The facility also challenges this question because fuel handling operations would be briefed and the applicant would have the procedure readily available. This question does not require an applicant to have the procedure memorized. This question tests understanding of an important operating limitation appropriately emphasized within a procedure. To answer this question, the applicants must recognize that actuation of the interlock was an abnormal situation, and that the response to abnormal occurrences during fuel handling is to notify RE. This is not an unreasonable level of detail to expect from an applicant on a closed reference question. While the examination standard does allow use of reference materials on a selective basis as attachments to the written exam, it cautions that the references must not improve the applicant's chances of guessing the correct answer by eliminating incorrect distractors. Use of a reference in this instance would have reduced this question level of difficulty to that of a direct lookup.

5. RO Written Exam Question #72

Question:

Refueling is in progress. A new fuel assembly has just been lowered into core location A-11 (core map attached). You are the PPO and have noted the following before and after readings on the wide range logarithmic power channels:

	BEFORE	AFTER
WR CH A	1.9E1 cps	2.0E1 cps
WR CH B	1.8E1 cps	3.2E1 cps
WR CH C	1.6E1 cps	1.9E1 cps
WR CH D	1.0E1 cps	1.2E1 cps

Based on these indications, which of the following is required?

A Suspend all core alterations and positive reactivity additions.
B Commence boration per AOP-2558, "Emergency Boration."
C Continue to monitor nuclear instruments, NO immediate action required. **[Key answer]**
D Withdraw the fuel assembly and contact Reactor Engineering for guidance.

Facility Comments:

Comments:
Choice C (the correct answer) is an acceptable answer for an anticipated count rate multiplication due to the loading of a new fuel assembly in a location adjacent to the CH B Wide Range detector. Since no information is supplied in the stem of the question as to the refueling method (e.g., Full Core Reload) it is reasonable to expect this change in some instances.

Choice B is also an acceptable response. The stem of the question provides no additional information regarding the method of refueling and status of refueling. Assumptions could be made as to the refueling method (e.g., fuel shuffle) and previous data trends of a 1/M plot and/or count rate changes.

Historical data shows that during a fuel shuffle the 1/M value rarely dips below (.8). Using the provided count rate data from the stem of this question 1/M values ($CR_{initial}/CR_{final}$) are as follows; CH A (.95), CH B (.56 almost a doubling), CH C (.84), CH D (.83). If an assumption is made that initial count rates at the start of this fuel shuffle was even lower, then its effect on a 1/M plot would be greater and a doubling may be evident. If the operator believes that an unanticipated count rate multiplication has occurred he/she is compelled by OP 2209A to commence an emergency boration.

Recommendation:
Accept Choice C and Choice B

Justification:
Choice B correctly describes the required operator action of OP 2209A and is a conservative response to a situation that required judgement.

4.5.15 IF, ant any time, unanticipated count rate multiplication, (i.e., doubling), is indicated, PERFORM the following:

a. SUSPEND refuel operations.
b. Refer To AOP 2558, “Emergency Boration” and PERFORM applicable actions to initiate boration to RCS.
c. Immediately NOTIFY Reactor Engineering and SM.
d. REQUEST evaluation be completed prior to restart of fuel handling activities.
e. INITIATE CR.

NRC Response:

Facility recommendation for multiple correct answers is **NOT ACCEPTED**.

As stated in the question and the facility comment, the procedural requirement is to commence an emergency boration if an unanticipated ***doubling*** of count rate occurs. A core map was included with this question for the applicants to use as an attached reference. Using the core map, applicants could determine from information in the question stem that the inserted assembly was adjacent to the wide range logarithmic power instrument that is showing increased counts. The count rate increase indicated on this channel has not doubled from the initial readings given. The other channels are indicating only very slight rise in counts. For the given conditions, emergency boration is not ***required***. In addition, if we were to accept the facility argument the “B” was appropriate, then answer “A” would be correct as well, a third correct answer requiring deletion of this question rather than acceptance of a second answer.

Applicants are cautioned at the start of the written exam per NUREG 1021, Appendix E, to “*not* make assumptions regarding conditions that are not specified in the question unless they occur as a consequence of other conditions that are stated in the question.” Presuming that initial counts were even lower at the start of refueling requires the applicant to make assumptions not stated in the question.

References:
- OP-2209A, "Refueling Operations,” Revision 24

6. JPM A2SRO

A step identified as critical in this SRO JPM required applicants to specify the procedure that contains the necessary post-maintenance testing guidance.

Facility Comments:

Justification for accepting 2604AO and 2308X11 as a correct answer to JPM-A2SRO “SRO AWO Acceptance”:

Millstone Unit 2 is developing a new set of procedures designed to simplify component maintenance and retesting. These new procedures, called Maintenance Operating Procedures (MOPs), are designated as 2300X. The MOPs include the steps of the 2600 procedure and additional sections for venting, draining and tagging for the component in question. When a MOP is approved, it becomes the preferred post-maintenance procedure and replaces the previously used 2600.

At the time the SRO candidates were trained and when this JPM was developed, the MOPs did not exist. At the time of the NRC Exam, a MOP was approved for the “A” HPSI Pp. I believe this is the only MOP that is approved for use at this time.

Due to the present state of transition from the 2600 procedures to the 2300X procedures, Millstone Unit 2 requests both 2604AO and 2308X11 be considered correct.

NRC Response:

Recommendation to allow both “2604AO” and “2308X11" as correct procedure references is **ACCEPTED.**

The 2300 procedure covering post-maintenance testing of HPSI Pump ‘A’ (relating to this JPM) has been issued, and is the correct procedure for the evolution. However, the license applicants were trained to use the 2600 plant equipment TS surveillance procedures as post-maintenance test procedures. An applicant, not yet trained on recent procedure changes, would be expected to identify the 2600 procedure as the correct procedure. The 2600 procedures will continue to exist and are used for conduct of periodic scheduled surveillances. The 2300 series procedures provide more comprehensive guidance, including steps for drain, refill and testing to restore operability.

References:
- 2604AO, “HPSI PUMP INSERVICE TESTING, 1,750 PSIA, FACILITY 1”, Revision 000-00
- 2308X11, “A HPSI PUMP MAINTENANCE,” Revision 000-01

7. JPM A1RO

Several steps identified as critical in this SRO JPM appeared to have incorrect error bands.

Facility Comments:

During grading of this JPM, the NRC noted several applicants who obtained the correct end result despite errors in intermediate steps identified as JPM critical steps. The NRC grader determined that there were errors in the answer key for this JPM and requested a corrected key.

NRC Response:

The NRC verified that the resubmitted key contained the corrections identified by the NRC grader.

8. JPM A2RO

Facility Comments:

Tagging of the chiller compressor breaker in this JPM should not have been a critical step. The compressor is interlocked with the associated chilled water pump, so tagging the chill water pump prevents the compressor from starting.

NRC Response:

The facility requested change to the answer key is **ACCEPTED**.

The NRC reviewed the chilled water system OP 2330C, which addresses system maintenance and does not require the chiller compressors to be tagged under the conditions in this JPM. The NRC also reviewed the tagging procedure WC-2 and noted that this procedure does not preclude dependence on interlocks for equipment protection. No procedural basis was found for requiring this breaker to be tagged.

Attachment 3

ES 501 Simulation Facility Report

Facility Licensee: Millstone Unit 2
Facility Docket No. 50-336
Operating Tests Administered on: March 21 - March 24, 2005

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

While conducting simulator scenario validation and during examination administration, examiners observed the following simulator performance discrepancies:

During validation of the scenarios, Facility 1 and 2 EBFAS could not be used cross-train with Facility 2 and 1 H2 PURGE valves to depressurize containment, although there is no reason physically why this should not work. DR# 2005-2-0013.

During validation of the scenarios, CH-501 VCT Outlet Valve control switch did not appear to operate as indicated per the schematic diagram. DR# 2005-2-0021

During the administration of one scenario, the PMW flow controller and flow totalizer indicated approximately .1 gpm with the flow control valve closed. DR#2005-2-0023