January 10, 2006

Mr. David A. Christian
Senior Vice President and
Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: KEWAUNEE NUCLEAR POWER PLANT
NRC INITIAL LICENSE EXAMINATION REPORT NO. 05000305/2005301(DRS)

Dear Mr. Christian:

On November 18, 2005, the NRC completed initial operator licensing examinations at your Kewaunee Nuclear Power Plant. The enclosed report documents the results of the examination which were discussed on November 18 and December 8, 2005, with Mr. M. Gaffney and Mr. S. Johnson, respectively, and with other members of your staff.

NRC examiners administered the operating test during the week of November 14, 2005. Members of the Kewaunee Nuclear Power Plant Training Department staff administered the written examination on November 18, 2005. One Reactor Operator (RO) and six Senior Reactor Operator (SRO) applicants were administered license examinations. The results of the examinations were finalized on December 22, 2005. Five applicants passed all sections of their examinations, four of these applicants were issued respective operator or senior operator licenses. Two SRO applicants failed the written examination and will not be issued licenses. One applicant scored less than 82 percent on the written examination; and, in accordance with the guidelines of NUREG 1021, “Operator Licensing Examination Standards for Power Reactors,” ES-501.D.3.c, his license will be withheld until any appeal rights of the failed applicants are exhausted.

In accordance with 10 CFR Part 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). ADAMS is accessible from the NRC Web site at **http://www.nrc.gov/reading-rm/adams.html** (the Public Electronic Reading Room).

We will gladly discuss any questions you have concerning this examination.

Sincerely,

***/RA/***

Hironori Peterson, Chief
Operations Branch
Division of Reactor Safety

Docket No. 50-305
License No. DPR-43

Enclosures:
1. Operator Licensing Examination Report 050000255/2005301(DRS)
2. Simulation Facility Report
3. Post Examination Comments and Resolutions
4. Written Examinations and Answer Keys (RO & SRO)

cc w/encls 1 & 2:
M. Gaffney, Site Vice President
C. Funderburk, Director, Nuclear Licensing and Operations Support
T. Breene, Manager, Nuclear Licensing
L. Cuoco, Esq., Senior Counsel
D. Zellner, Chairman, Town of Carlton
J. Kitsembel, Public Service Commission of Wisconsin

cc w/encls 1, 2, 3, and 4: G. Winks, Training Manager

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

| | |
|---|---|
| Docket No.: | 50-305 |
| License No.: | DPR-43 |
| Report No.: | 05000305/2005301(DRS) |
| Licensee: | Dominion Energy Kewaunee, Inc. |
| Facility: | Kewaunee Nuclear Power Plant |
| Location: | N490 Highway 42<br>Kewaunee, WI 54216 |
| Dates: | November 14 through 18, 2005 |
| Examiners: | B. Palagi, Chief Examiner<br>C. Phillips, Examiner<br>D. Reeser, Examiner |
| Approved by: | H. Peterson, Chief<br>Operations Branch<br>Division of Reactor Safety |

Enclosure 1

# SUMMARY OF FINDINGS

ER 05000305/2005301(DRS); 12/14/2005-12/18/2005; Kewaunee Nuclear Power Plant; Initial License Examination Report.

The announced operator licensing initial examination was conducted by regional examiners in accordance with the guidance of NUREG-1021, “Operator Licensing Examination Standards for Power Reactors,” Revision 9.

<u>Examination Summary</u>:

- Seven examinations were administered (one Reactor Operator and six Senior Reactor Operator).

- Five applicants passed all sections of their examinations, four of these applicants were issued respective operator or senior operator licenses. Two SRO applicants failed the written examination and were not issued licenses. One applicant scored less than 82 percent on the written examination; and, in accordance with the guidelines of NUREG 1021, “Operator Licensing Examination Standards for Power Reactors,” ES-501.D.3.c, his license will be withheld until any appeal rights of the failed applicants are exhausted.

## REPORT DETAILS

### 4. OTHER ACTIVITIES (OA)

4OA5 Other

.1 Initial Licensing Examinations

a. Examination Scope

The NRC examiners conducted an announced initial operator licensing examination during the week of November 14, 2005. The licensee used the guidance established in NUREG-1021, “Operator Licensing Examination Standards for Power Reactors,” Revision 9, to prepare the examination outline and to develop the written examination and operating test. The NRC examiners administered the operating test November 14 through 17, 2005. Members of the Kewaunee Power Station Training Department administered the written examination on November 18, 2005. One Reactor Operator (RO) and six Senior Reactor Operator (SRO) applicants were examined.

b. Findings

Written Examination

The licensee developed the written examination. During their internal review, the NRC examiners determined that the examination, as submitted, was within the range of acceptability expected for a proposed examination. Written examination comments developed during review by the NRC staff, and as a result of examination validation were incorporated into the written examination in accordance with the guidance contained in NUREG-1021.

A total of nine post-examination comments (6 RO; 3 SRO comments) were submitted by the applicants and station training department personnel on November 28, 2005. The results of the NRC’s review of the comments are documented in Attachment 3, Post Examination Comments and Resolutions.

Operating Test

The NRC examiners determined that the operating test, as originally submitted by the licensee, was within the range of acceptability for a proposed examination. The examiners validated the operating test during the validation week and replaced or modified several items in the proposed operating test. Test changes, agreed upon between the NRC and the licensee, were made in accordance with NUREG-1021 guidelines.

Examination Results

Five applicants passed all sections of their examinations, four of these applicants were issued respective operator or senior operator licenses. Two SRO applicants failed the

written examination and were not be issued a licenses. One applicant scored less than 82 percent on the written examination; and, in accordance with the guidelines of NUREG 1021, “Operator Licensing Examination Standards for Power Reactors,” ES-501.D.3.c, his licenses will be withheld until any appeal rights of the failed applicants are exhausted.

.2 Examination Security

a. Inspection Scope

The NRC examiners briefed the facility contact on the NRC’s requirements and guidelines related to examination physical security (e.g., access restrictions and simulator considerations). The examiners observed the implementation of examination security and integrity measures (e.g., security agreements) throughout the examination process.

b. Findings

No findings were noted in this area. The licensee staff was observed to be enforcing correct examination security procedures.

4OA6 Meetings

.1 Exit Meeting

The chief examiner presented the examination team's preliminary observations and findings on November 18, 2005, to Mr. M. Gaffney and other members of the Operations and Training Department staff. A subsequent exit via teleconference was held on December 8, 2005, with Mr. S. Johnson following review of the site post examination comments. The licensee acknowledged the observations and findings presented. No proprietary information was identified by the station’s staff during the exit meeting.

ATTACHMENT: SUPPLEMENTAL INFORMATION

## SUPPLEMENTAL INFORMATION

### KEY POINTS OF CONTACT

Licensee
M. Gaffney, Site Vice President
K. Hoops, Site Director
F Winks, Manager Nuclear Training
J. Ruttar, Manager Nuclear Operations
D. Fitzwater, Supervisor Nuclear Operations Training
S. Johnson, Operations Training

NRC
S. Burton, Senior Resident Inspector

### ITEMS OPENED, CLOSED, AND DISCUSSED

Opened, Closed, and Discussed

None

### LIST OF ACRONYMS USED

| | |
|---|---|
| ADAMS | Agency-Wide Document Access and Management System |
| DRS | Division of Reactor Safety |
| NRC | Nuclear Regulatory Commission |
| PARS | Publicly Available Records |
| RO | Reactor Operator |
| SRO | Senior Reactor Operator |

# SIMULATION FACILITY REPORT

Facility Licensee: Kewaunee Power Station

Facility Docket No.: 50-305

Operating Tests Administered: November 14 - 17, 2005

The following documents observations made by the NRC examination team during the initial operator license examination. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of non-compliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

During the conduct of the simulator portion of the operating tests, the following item was observed:

| ITEM | DESCRIPTION |
|---|---|
| Emergency Diesel Generator | A problem was noted with the use of the Emergency Diesel Generator Speed Control switch. When attempting to adjust frequency from 62 to 65 Hz, frequency increased uncontrollably resulting in a Diesel Generator trip. This problem occurred on both the “A” and “B” Diesel Generators. |

Question Number 32.

Given the following:

- The plant startup is in progress.
- Reactor power is 1.4%.
- FW-07A & B, S/G Main Feed valves, AND FW-10A & B, S/G Bypass Feed valves, indicate closed.
- Feedwater Pump A has just been started.
- SG B level begins to slowly rise.

What action is required?

A. Direct the NAO to locally close FW-9B, Feedwater Main Control Valve 1B Bypass Valve Inlet.

B. Cycle FW-10B, S/G B Bypass, fully open and closed.

C. Close FW-12B, S/G B Feedwater Isolation Valve.

D. Stop Feedwater Pump A.

Answer: C

Applicant Comment:

An applicant commented that answer "D" should also be accepted as correct. He argued that if level was near the high level trip of 67% stopping the Feedwater Pump would be a viable answer to preclude damage to the secondary system and the turbine.

Facility Proposed Resolution:

The facility management argued that there is only one correct answer for the given conditions. Based on the given conditions, the guiding procedures are N-O-02, Plant Startup From Hot Shutdown to 35% power, and N-FW-05A, Feedwater System Normal Operation. Per N-O-02, the initial conditions are Steam Generator (SG) level is maintained between 33% and 50%. Therefore, based on the question stem a normal plant startup level band should be assumed. Furthermore, the stem states that SG B level begins to slowly rise. Thus, there is no basis for assuming SG level is near or approaching the SG Feedwater Isolation setpoint of 67%.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution it was decided to accept only the original correct answer. Based on the plant conditions given in the question stem (normal startup in progress) it is unreasonable to postulate that steam generator level is near the high level trip. A slowly rising steam generator level is anticipated by plant procedure under the conditions given this question, and procedural guidance is provided to close the feedwater isolation valve under these conditions (answer “C”). Therefore, “C” is the only correct answer.

Question Number 34.

Given the following:

- The plant is at 100% power.
- AFW Pump B is running for a surveillance test in progress.
- Annunciator 47061-M, AFW PUMP B LOW OIL PRESS, alarms.

What is the expected operator response for this condition?

A. Trip AFW Pump B, and go to N-FW-05B, Auxiliary Feedwater System.

B. Trip AFW Pump B, and go to A-FW-05B, Abnormal Auxiliary Feedwater System Operation.

C. Verify the Auxiliary Lube Oil Pump is running, and go to N-FW-05B, Auxiliary Feedwater System.

D. Verify the Auxiliary Lube Oil Pump is running, and go to, A-FW-05B, Abnormal Auxiliary Feedwater System Operation.

Answer: D

Applicant Comment:

An applicant commented that answer “B” should also be accepted as correct. He argued that per the stem of the question, a surveillance test was in progress with Annunciator 47061-M, AFW PUMP B LOW OIL PRESS, ON. Since, the stem provides no information as to whether the aux lube oil pump starts and the alarm clears, and knowing the Auxiliary Feedwater (AFW) Pump is being run only for testing, it would be correct to trip the AFW pump. The applicant sights UG-0 (Rev. F), User’s Guide For Emergency And Abnormal Procedures, section 6.1.2, which states that during abnormal conditions, an operator has the authority to take action that is clearly needed based on observed condition. With the annunciator 47061-M not clearing, there is a lube oil problem. Conservative Decision Making, would require the AFW pump be tripped.

Facility Proposed Resolution:

The facility management agreed with the applicant that, without information about the current oil pressure value and trend, an operator could not make a valid decision as to whether the Auxiliary Feedwater (AFW) Pump should be tripped or not. The question would need to be revised to make only one of the answers correct. It was recommended that two answers be accepted, “B” and “D.”

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution it was decided to accept two answers as correct. Because, the stem of the question states that the auxiliary feedwater pump low oil pressure alarm annunciates it was expected that the applicants would select answer “D” which contained the actions called out in the annunciator procedure (verify the auxiliary oil pump is running). However, neither the question stem nor answer “D” provided information as to whether oil pressure recovered. Therefore, it is equally reasonable to assume oil pressure has not recovered. Knowing that there has been a low oil pressure alarm, believing oil pressure to still be low, and from the stem knowing there is no emergency condition because it is stated that the auxiliary feedwater pump is being run for a surveillance test, it would also be correct to trip the auxiliary feedwater pump (answer “B”). Therefore both answers “B” and “D” were accepted as correct.

Question Number 44.

Which of the following situations requires entry into a Technical Specification LCO Action?

A. Pressurizer Pressure Transmitter P-429 fails low while at 38% power.

B. Pressurizer level decreases to less than 17% with a reactor startup in progress.

C. Pressurizer Backup Heaters energize after a 8% load reduction.

D. Pressurizer Pressure Transmitter P-449 fails high while on RHR Cooling.

Answer: A

Applicant Comment:

An applicant commented that answer “D” should also be accepted as correct. He argued that answer “D” requires entry into a Technical Specification LCO Action to MAINTAIN HOT SHUTDOWN.

Facility Proposed Resolution:

The facility management disagreed with the applicant. They argued that with the plant on RHR Cooling, as stated in answer “D,” the reactor would be shutdown with Reactor Coolant System (RCS) temperature is less than 400ºF and RCS pressure less than 425 psig. In these conditions the plant is in INTERMEDIATE SHUTDOWN, and therefore, the LCO for PT-449 does not apply.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution it was decided to accept only the original correct answer. The applicant contends that answer “D” should be considered correct because, although no action is required, the failure would place a Mode change restriction on the plant. A review of Technical Specification 3.5, Table 3.5-2, and Table 3.5-3 confirmed the facility position that entry into a Technical Specification LCO Action was not required for the conditions listed in distractor “D.” The fact that Transmitter P-449 has a Technical Specification function in another plant Mode of operation is what makes it a good distractor. Only the conditions given in answer “A” would require Technical Specification directed actions and therefore “A” was retained as the only correct answer.

Question Number 62.

Given the following:

- Plant is at 90% power.
- Power Range Instrument N-42 fails and is removed from service per A-MI-87, Bistable Tripping for Failed Reactor Protection or Safeguards Inst.
- The surveillance for SP-47-011A, Reactor Coolant Temperature and Pressurizer Pressure Instrument Channel I (Red) Calibration, comes due.
- The decision is to bypass the failed NIS channel inputs to allow the calibration (Technical Specification 3.5.d).

What is the coincidence for the OT∆T reactor trip while the surveillance is in progress?

A. 2 out of 2

B. 2 out of 3

C. 1 out of 4

D. 1 out of 3

Answer: B

Applicant Comment:

An applicant commented that answer "D" should also be accepted as correct. He argued that, in a normal lineup, the OTDT trip coincidence is 2 out of 4 (4 channels provide input, 2 required to actuate). With N-42 input to Channel II - OTDT trip bypassed, only 3 channels provide input, 2 are still required to actuate to trip the reactor, therefore coincidence is 2 out of 3. During SP-47-011A, after initial switch lineups and verifications, the bistables associated with the RED Channel Pressure / Temperature trips are placed in TEST. After that step, 3 channels still provide input if a BLUE or a YELLOW channel actuates the reactor will trip, therefore he argued that the coincidence becomes 1/3. Therefore he believes the correct answer depends on which actions of SP-47-011A have been completed. Because, it was not stated in the question stem which actions of SP-47-011A have been completed, he contends answer "D" should also be accepted as correct.

Facility Proposed Resolution:

The facility management disagreed with the applicant. They argued that the coincidence for the OT∆T reactor trip will always remain 2 out of 3 for the conditions given in the question stem. While it is true that the number of bistables that must actuate to cause a reactor trip varies between one and two at various times during the

performance of SP-47-011A, the overall coincidence for the trip is still 2 out of 3. There will always be 3 channels, at times during the performance of SP-47-011A one of the three bistables is actuated and only one additional bistable needs to actuate to complete the 2 out of 3 logic.

It was also stated that an applicant requested clarification of the status of performing the surveillance. It was answered that the surveillance had not yet been entered. This information was provided to all applicants in the room at that time. The individual providing this feedback was not in the room, as he had completed his examination.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution it was decided to accept only the original correct answer. The applicant commented that answer “D” should be considered correct because, at times in the conditions provided in the question stem, the trip of one more instrument would cause a reactor trip. While it is true that at times during the completion of SP-47-011A the trip of one more instrument would cause a reactor trip, that was not the question asked. The question asked what was the reactor trip “coincidence,” which in this case was 2 out of 3 at all times, answer “B.” The concept of what trip coincidence the plant is in is an important concept for Technical Specification compliance. Because only answer “B” provided the correct reactor trip coincidence for the conditions given in the stem it was retained as the only correct answer.

The Facility noted that a clarification to this question was provided after the applicant providing this comment had left the room having completed the examination. However, the clarification did not provide information necessary to select the correct answer. Therefore, again it was determined “B” was the only correct answer.

Question Number 63

Given the following:

- Power level is 99.9% (1771.5 MWt).
- The current UFMD and RTO OPERATING LIMITs are 1772 MWt.
- SP-87-125, Shift Instrument Channel Checks - Operating, calorimetric was completed 6 hours ago.
- The signal is lost to PPCS from Feedwater Flow channel FT-476.
- The RTO 1-minute average, PPCS point R5110G, drops to 1200 MWt.
- The Computer Group reports the signal will be restored within 10 minutes.

What is the affect on the UFMD and RTO Operating Limits, and what action is taken when the flow channel input is restored and R5110G reads normal?

A. Both the UFMD and RTO Operating limits will read 1749 Mwt.
When the signal is restored, the operator will click the APPLY UFMD LIMIT button on the PPCS UFMD Correction Factors screen to return both limits to 1772 Mwt.

B. The UFMD Operating Limit will read 1749 MWt and RTO Operating Limit will read 1772 Mwt.
Power will be reduced to less than 1749 Mwt, and when the signal is restored, the operator will click the APPLY UFMD LIMIT button on the PPCS UFMD Correction Factors screen to return the UFMD limit to 1772 Mwt.

C. Both the UFMD and RTO Operating limits will read 1769 Mwt.
When the signal is restored, the operator will click the APPLY UFMD LIMIT button on the PPCS UFMD Correction Factors screen to return both limits to 1772 Mwt.

D. The UFMD Operating Limit will read 1769 MWt and RTO Operating Limit will read 1772 Mwt.
Power will be reduced to less than 1769 Mwt, and when the signal is restored, the operator will click the APPLY UFMD LIMIT button on the PPCS UFMD Correction Factors screen to return the UFMD limit to 1772 Mwt.

Answer: A

<u>Applicant Comment:</u>

An applicant commented that there was no correct answer to this question. He argued that when the flow channel is returned, the UFMD limit will go to 1772. Then, when the “APPLY UFMD LIMIT” button is clicked <u>only</u> the RTO Limit changes, not “both” the RTO and the UFMD Limit.

Facility Proposed Resolution:

The facility management maintained that answer “A” is correct. They agree that when the flow channel is returned, the UFMD limit will go to 1772. However, to have both the the RTO and the UFMD Limit read 1772 the “APPLY UFMD LIMIT” button must be clicked. Therefore, answer “A” remains correct.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution it was decided to retain the question. This question had a two part answer. First, for the stem conditions what would the UFMD and RTO Operating Limit read, and then how to reset the limits? Only answer “A” supplied the correct answer for the first part of the question. The applicant commented that the second part of answer “A” was incorrect because the UFMD limit would automatically reset. While the second part of the answer could have been worded differently, it is correct. Both the UFMD and RTO Operating Limits would not read 1772 MWt until the APPLY UFMD LIMIT button was pushed. Because “A” is a correct answer the question was retained.

Question Number 67

Given the following:

- Waste Gas Decay Tank A is in service.
- The relief valve for that tank, WG-14A, lifts and fails to reseat.

What is the effect on the plant?

A. The release will be automatically isolated when the Waste Gas Analyzer senses the pressure drop in the Waste Gas Decay Tank.

B. The release will be automatically isolated when the Waste Gas Decay Tank pressure reducing control valve WG-201 closes.

C. The release will <u>NOT</u> be automatically isolated, but will be monitored by the Aux. Building Vent radiation monitors R-13 and R-14.

D. The release will <u>NOT</u> be automatically isolated, but will be detected by the Charging Pump Room Area Monitor R-4.

Answer: C

<u>Applicant Comment:</u>

An applicant commented that the stem of the question lists the wrong valve number for the Waste Gas Decay Tank A relief valve. He argues that if the question is answered using the valve with the valve number given in the stem, answer “D” could be a possible correct answer, if a pipe failure was also assumed.

<u>Facility Proposed Resolution:</u>

The facility management maintained that this question had only a typographical that did not invalidate the question. They agreed that in the question stem “The relief valve for that tank, WG-14A, lifts and fails to reseat.” should have read “The relief valve for that tank, WG-13A, lifts and fails to reseat.” However, they argued that during the examination no individual questioned the difference between the valve number and the valve name. Additionally, they argued that even if someone attempted to answer the question for valve WG-14A, which is a valve that connects the Waste Gas Decay Tank to a vent header, answer “D” would not be correct.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution it was decided to accept only the original correct answer. This question asks the effect on the plant of the relief valve on Waste Gas Decay Tank A lifting and failing to reseat. An applicant commented that the wrong valve number was given for the relief valve (which is true), and if one attempted to answer the question using the valve with the provided valve number (and ignoring the fact that the stem clearly states it is the relief valve that opens), answer “D” could be right. In fact “D” would not be a correct answer even if the provided valve number is assumed to be correct. Additionally, none of the candidates asked a question during the examination about the difference between the valve number and the valve description. This was a minor typographical error that does not appear to have confused any applicant or invalidated the question. Therefore, “C” was retained as the only correct answer.

Question Number 81

Given the following:

- The plant is at 100% power.
- Surveillance Test SP-42-312B, Diesel Generator B Availability Test, is in progress with DG B running.
- Component Cooling Pump B trips on overcurrent.
- BRB-104, ckt 10 supplying DG B tripped open and CANNOT be closed.

What is the effect of this condition?

A. DG B is inoperable and must have its fuel supply locally isolated.
A plant shutdown must commence within one hour using the Standard Shutdown Sequence (Technical Specification 3.0.c)

B. DG B is Degraded but Operable and can be controlled locally.
A 24-hour LCO is applicable for restoration of the DC Distribution System and a 72-hour LCO is applicable for restoration of Component Cooling Pump B.

C. DG B is inoperable and must have its fuel supply locally isolated.
A 72-hour LCO is applicable for restoration of Component Cooling Pump B and a 7-day LCO is applicable for restoration of DG B.

D. DG B is Degraded but Operable and can be controlled locally.
Only the 72-hour LCO is applicable for restoration of Component Cooling Pump B.

Answer: C

<u>Applicant Comment:</u>

An applicant commented that answer “A” should also be accepted as correct. He argued that, this question has two answers dependent on the time frame. He agrees that answer “C” is correct initially. However, he argues that within 24 hours of the conditions given in the stem “A” Diesel Generator testing is required, because all “B” train emergency equipment is not OPERABLE this testing can not be performed. Therefore, at that time a plant shutdown must commence within 1 hour.

<u>Facility Proposed Resolution:</u>

The facility management maintained that there was no reason, based on the question stem, to extrapolate what the answer to this question could be 24 hours later. They argued that based on the conditions given in the question stem “C” is the only correct answer.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution it was decided to accept only the original correct answer. The applicant correctly points out that under the condition of the stem a once per 24-hour test of the "A" Diesel Generator is required and cannot be performed. Therefore he believes that if the conditions still existed 24 hours later a shutdown must be commenced in 1 hour, making answer "A" correct. While site management contends it is unreasonable to answer this question by speculating what the answer could be 24 hours later, in that the 24-hour period until Diesel Generator testing is required would be used to attempt to return equipment to service, thereby preventing the need to shutdown. Technical Specifications 3.0 c, 3.3.d.2, 3.7.b.2, and 3.7.c were reviewed and it was found that there was no requirement that "a plant shutdown must commence within 1-hour" of the conditions given in the question stem. Therefore, "C" was retained as the only correct answer.

Question Number 88

What is the reason for the Feedwater Isolation signal generated from High-High SG level?

A. Preclude excessive SG tilts due to cooler feedwater supplied to ONE SG.

B. Prevent overfill of the SG that may result in damage to secondary components.

C. Ensure containment pressure remains within maximum internal pressure limit with the affected SG faulted inside containment.

D. Protect the Feedwater Pumps from operating in runout condition with FW-7A/B, S/G A/B Main Valve, fully open.

Answer: B

Applicant Comment:

An applicant commented that answer "C" should also be accepted as correct. He argued that per the Update Safety Analysis Report, one of the 4 major factors that influence release of mass and energy following a steam line break is Steam Generator (SG) water level. Having too much inventory could challenge containment integrity, and the Feedwater Isolation signal generated from High-High SG level prevents this. Therefore, "C" should also be considered a correct answer.

Facility Proposed Resolution:

The facility management disagreed with the applicant. They argued that the specific reason for Feedwater Isolation signal on Steam Generator (SG) high level was to prevent damage to secondary components as stated in the Technical Specification Basis.

NRC Resolution:

Upon review of the question, the applicant comment, and the facility proposed resolution it was decided to accept only the original correct answer. The applicant contended that there are two correct answers given for this question. He believes that in addition to the reason given in the Technical Specification Basis, the High-High feedwater isolation is provided to limit containment pressure in the event of a faulted Steam Generator. This is a misconception on the part of the candidate. The USAR analysis for a faulted Steam Generator is based on normal level (not level at the High-High limit) and a feedwater isolation from the resulting safety injection signal. The Technical Specification 3.5 Basis states "Main feedwater isolation actuation occurs as a result of a Hi-Hi steam generator water level to prevent steam generator overfill

conditions. Steam generator overfill may result in damage to secondary components; for example, high moisture steam could erode the turbine blades at an accelerated rate." no other reason for this trip is mentioned. Therefore, "B" was retained as the only correct answer.

Question Number 90

Given the following:

- The plant has experienced a fire in the Control Room.
- The actions of E-O-06, Fire In Alternate Fire Zone, are being performed.

What are the indications of an air dryer filter failure, and what are the procedural actions performed by the Control Room Supervisor to mitigate the consequences of this failure.

A. Instrument Air Drier/Filter 1A/1B differential presure will rise above 5 psid.
   The CRS will verify SA-121, Air Drier/Filter Bypass CV, opens.

B. Instrument Air Drier/Filter 1c differential pressure will rise above 10 psid.
   The CRS will open SA-100A, Air Drier 1A Supply, and IA-300, 1½” Alt IA, and then request the Control Operator A to start Air Compressor B.

C. Instrument air header pressure will drop below 95 psig.
   The CRS will depress the Air Drier 1C RESET pushbutton to confirm problem, and then align flow through Instrument Air Drier 1B.

D. Instrument air header pressure will drop below 100 psig.
   The CRS will open SA-70 and SA-71, 1½" Dedicated Instrument Air Header Isolations, that bypass the Instrument Air Driers.

Answer: D

<u>Applicant Comment:</u>

An applicant commented that answer “A” should also be accepted as correct. He argued that high D/P across IA-121 is an early indication of an air dryer filter failure and that the normal automatic action for that condition is opening of the bypass valve. Therefore, “A” is a correct answer.

<u>Facility Proposed Resolution:</u>

The facility management disagreed with the applicant. They argued that answer “A” gives the correct actions under normal conditions. However, in the situation given in the question stem the control room is evacuated and E-O-06 is in progress. Under these conditions the Air Dryers are bypassed by opening the Dedicated Air Header isolations, which supply the air directly through a separate in-pipe air dryer, making “D” the only correct answer.

<u>NRC Resolution</u>:

Upon review of the question, the applicant comment, and the facility proposed resolution it was decided to accept only the original correct answer. The applicant contended that there are two correct answers given for this question. He argues that answer “A” contains the normal action for air dryer failure and is therefore correct. However, the question stem states that there is a fire in the control room and the actions of procedure E-O-06 are being carried out, these are the action described in answer “D” not the action described in answer “A.” Therefore, “D” was retained as the only correct answer.

## WRITTEN EXAMINATIONS AND ANSWER KEYS (RO/SRO)

RO/SRO Initial Examination ADAMS Accession # ML060060460.