UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

August 5, 2008

Mr. David A. Christian
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060

SUBJECT: NORTH ANNA POWER STATION - NRC EXAMINATION REPORT 05000338/2008301 AND 05000339/2008301

Dear Mr. Christian:

During the period of June 2-20, 2008, the Nuclear Regulatory Commission (NRC) administered operating examinations to employees of your company who had applied for licenses to operate the North Anna Power Station. At the conclusion of the examination, the examiners discussed the examination questions and preliminary findings with those members of your staff identified in the enclosed report. The written examination was administered by your staff on June 24, 2008.

Two Senior Reactor Operator (SRO) applicants and three Reactor Operator applicants passed both the written and operating examinations. Three SRO applicants and five RO applicants failed the written examination. There were eleven post examination comments. These comments and the NRC resolution of these comments are summarized in Enclosure 2. A Simulation Facility Report is included in this report as Enclosure 3.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosures 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). Should you have any questions concerning this letter, please contact me at (404) 562-4550.

Sincerely,

/RA/

Malcolm T. Widmann, Chief
Operations Branch
Division of Reactor Safety

Docket Nos.: 50-338, 50-339
License Nos.: NPF-4, NPF-7

Enclosures:
1. Report Details
2. NRC Post Examination Comment Resolution
3. Simulation Facility Report

(cc: w/encl - See page 2)

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Letter to David A. Christian from Malcolm T. Widmann dated August 5, 2008

SUBJECT: NORTH ANNA POWER STATION - NRC EXAMINATION REPORT 05000338/2008301 AND 05000339/2008301

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August 5, 2008

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Two Senior Reactor Operator (SRO) applicants and two Reactor Operator applicants passed both the written and operating examinations. Three SRO applicants and six RO applicants failed the written examination. There were eleven post examination comments. These comments and the NRC resolution of these comments are summarized in Enclosure 2. A Simulation Facility Report is included in this report as Enclosure 3.

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/RA/

Malcolm T. Widmann, Chief
Operations Branch
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Docket Nos.: 50-338, 50-339
License Nos.: NPF-4, NPF-7

Enclosures: 1. Report Details
2. NRC Post Examination Comment Resolution
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(cc: w/encl - See page 3)

See previous concurrence

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SIGNATURE	**/RA/**	**/RA/**	**/RA/**	**/RA/**			
NAME	B. Caballero	M. Bates	M. Widmann	James Dodson			
DATE	08/01/2008	08/04/2008	08/04/2008	08/04/2008	8/ /2008	8/ /2008	8/ /2008
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

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NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-338, 50-339
License No.:	NPF-4, NPF-7
Report No.:	05000338/2008301, 05000339/2008301
Licensee:	Virginia Electric and Power Company
Facility:	North Anna Power Station, Units 1 & 2
Location:	1022 Haley Drive Mineral, VA 23117
Dates:	Operating Test – June 2-20, 2008 Written Examination – June 24, 2008
Examiners:	M. Bates, Chief Examiner, Operations Engineer E. Lea, Senior Operations Examiner B. Caballero, Chief-under-instruction, Operations Engineer M. Riches, Operations Engineer Trainee
Approved by:	Malcolm T. Widmann, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000338/2008301, 05000339/2008301, 06/02-20/2008 and 06/24/2008; North Anna Power Station; Licensed Operator Examinations.

The NRC examiners conducted operator licensing initial examinations in accordance with the guidance in NUREG-1021, Revision 9, “Operator Licensing Examination Standards for Power Reactors.” This examination implemented the operator licensing requirements of 10 CFR §55.41, §55.43, and §55.45.

The NRC administered the operating tests during the period of June 2-20, 2008. Members of the North Anna Power Station training staff administered the written examination on June 24, 2008. The written examination outline was developed by the NRC. The written exam, operating test outlines and operating test details were developed by the North Anna Power Station training staff.

Two Senior Reactor Operator (SRO) applicants and three Reactor Operator applicants passed both the written and operating examinations. Three SRO applicants and five RO applicants failed the written examination. Two SRO and three RO applicants were issued operating licenses.

There were eleven post examination comments.

No findings of significance were identified.

REPORT DETAILS

4. OTHER ACTIVITIES

4OA5 Operator Licensing Initial Examinations

a. Inspection Scope

The North Anna Power Station training staff developed the written exam and operating test. NRC regional examiners reviewed the proposed examination material to determine whether it was developed in accordance with NUREG-1021, “Operator Licensing Examination Standards for Power Reactors,” Revision 9, Supplement 1. Examination changes agreed upon between the NRC and the licensee were made according to NUREG-1021 and incorporated into the final version of the examination materials.

The examiners reviewed the licensee’s examination security measures while preparing and administering the examinations to ensure examination security and integrity complied with 10 CFR 55.49, “Integrity of Examinations and Tests.”

The examiners evaluated five SRO applicants and eight RO applicants who were being assessed under the guidelines specified in NUREG-1021. The examiners administered the operating tests during the period of June 2-20, 2008. Members of the North Anna Power Station training staff administered the written examination on June 24, 2008. The evaluations of the applicants and review of documentation were performed to determine if the applicants, who applied for licenses to operate the North Anna Power Station, met requirements specified in 10 CFR Part 55, “Operators’ Licenses.”

b. Findings

The NRC determined that the details provided by the licensee for the written exam, walkthrough, and simulator tests were within the range of acceptability expected for a proposed examination.

Two Senior Reactor Operator (SRO) applicants and three Reactor Operator applicants passed both the written and operating examinations. Three SRO applicants and five RO applicants failed the written examination.

The final RO and SRO written examinations with knowledge and abilities (K/As) question references/answers and examination references, and licensee’s post examination comments may be accessed in the ADAMS system (ADAMS Accession Numbers, ML082110224, ML082110233 and ML082110241).

Copies of all individual examination reports were sent to the facility Training Manager for evaluation and determination of appropriate remedial training.

4OA6 Meetings

Exit Meeting Summary

On June 20, 2008, the examination team discussed generic issues associated with the operating test with Mr. Sam Hughes, Operations Manager, and members of the North Anna Power Station staff. The examiners asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee personnel

E. Hendrixson, Director, Safety & Licensing
S. Hughes, Manager, Operations
J. Leberstien, Technical Consultant, Station Licensing
C. McClain, Manager, Training
J. Scott, Supervisor, Nuclear Training
W. Shura, Supervisor, Nuclear Training

NRC personnel

M. Bates, Operations Engineer
R. Clagg, Resident Inspector
J. Reece, Senior Resident Inspector
M. Riches, Operations Engineer (In-Training)

NRC Resolution to the Facility Comments

A complete text of the licensee's post examination comments can be found in ADAMS under Accession Number ML082110241.

RO QUESTION # 5

LICENSEE COMMENT:
In summary, the licensee requested that this question be graded with two correct answers. The original post exam comments submitted by the licensee can be viewed in ADAMS under ML number ML082110241.

The licensee's comment was that 1-OP-3.3, Unit Shutdown From Mode 4 to Mode 5, did not identify a specific lower limit of RCS temperature for when the accumulator discharge isolation valve breaker was *required* to be opened. Since the stem question only asked for "its required breaker position for the current plant conditions" and did not specify in accordance with 1-OP-3.3, the licensee contended that the limiting RCS temperature (with respect to the accumulator discharge isolation valve breakers) was when any RCS cold leg temperature was ≤ 280 °F in accordance with Tech Spec LCO 3.4.12 (Low Temperature Overpressure Protection System).

The licensee contended that when RCS temperature was at 325°F during a shutdown from Mode 4 to Mode 5, the accumulator discharge isolation valve breakers were allowed to be open or closed, dependent on cool down rate and other outage activities.

NRC DISCUSSION:
The question asked for the applicant to identify 1) the correct power supply to 1-SI-MOV-1865A ("A" Accumulator Discharge Isolation Valve) and 2) its ***required*** position when the unit was in Mode 4, RCS pressure was 720 psig, and RCS temperature was 325°F.

The governing plant procedure (1-OP-3.3, Rev 58, Unit Shutdown From Mode 4 to Mode 5), step 5.8.2 stated to "Verify all RCS hot leg temperatures (T_h) are less than 350°F" before the operator was directed to open the accumulator discharge isolation valve breakers. Technical Specification LCO 3.4.12 (Low Temperature Overpressure Protection System) required that the accumulators must be isolated and the power removed from the isolation valve operators *when any RCS cold leg temperature was ≤ 280 °F.* Since the 1-OP-3.3 procedure did not specifically prohibit these breakers from remaining closed when RCS temperature was 325°F, then the breaker was allowed to be either open or closed at that point in time during the plant cool down, i.e., there was no breaker position *requirement* when RCS temperature was 325°F. Therefore, there was no correct answer for this question.

NRC RESOLUTION
In accordance with NUREG 1021, Rev. 9, Supplement 1, ES-403, Section D.1.c, Question # 5 is deleted from the exam.

RO QUESTION # 15

LICENSEE COMMENT:
In summary, the licensee requested that this question be deleted from the exam. The original post exam comments submitted by the licensee can be viewed in ADAMS under ML number ML082110241. The licensee contended that:

- the question had no discriminatory value because 7 out of 8 ROs missed the question;

- determining the status of the General Warning Lights, located in the Solid State Protection System Logic Cabinets (SSPS), was not a job requirement for reactor operators at North Anna. Instead, the licensee contended that the licensed operators would dispatch Instrument Technicians to determine the cause if either of the two Safeguards Trouble annunciators (1K-G1, SFGDS PROT SYS TR A TROUBLE and/or 1K-G2, SFGDS PROT SYS TR B TROUBLE) were in an alarm condition;

- the targeted K/A for this question was not applicable to North Anna's Engineered Safety Features Actuation System (ESFAS) and SSPS because it is only associated with Programmable Logic Controller (PLC) based ESFAS/SSPS systems; and

- this question was not administered equally for the ROs and SROs because the SRO written exam question #81 provided an unfair advantage to SRO applicants when answering this question. The licensee stated that 7 out of 8 ROs missed the question whereas 4 out of 5 SROs answered the question *correctly*.

NRC DISCUSSION:
The question asked for the applicant to 1) identify whether there was only one Safeguards Trouble control room annunciator, common for both SSPS trains, versus two separate annunciators – one for each train, "A" and "B" and 2) recognize how a General Warning condition would affect the red light in the associated train's logic cabinet, i.e., illuminated or extinguished.

Each licensee comment for this question was addressed below.

- NUREG-1021, Rev. 9, Supplement 1, Appendix "A" (Overview of Generic Exam Concepts) outlines three principle facets of test validity as 1) content validity, 2) operational validity, and 3) discrimination validity. Furthermore, the Appendix "A" states that:

  "Test items that are so difficult that few (if any) of the examinees are expected to answer correctly do not discriminate and should not be used on an NRC examination. It is expected that every examination will contain some test items that all or most of the examinees will answer correctly or incorrectly. This does not necessarily mean that the test items or the examination are invalid."

  Prior to administration of the examination, the licensee performed reviews and conducted validation of the entire examination to assure that all test items were 1) related to the job, 2) addressed an actual or conceivable activity performed on the job, and 3) not too difficult. Based on these reviews and validation activities, this exam item was not anticipated to be so difficult that few (if any) of the applicants would be able to answer the item correctly.

Furthermore, this test item was technically correct and pertinent to the applicant's job, (see next point below). Consequently, the licensee's contention did not substantiate deleting the item from the exam.

- The North Anna Reactor Operator lesson plan (77-A, Revision 2, 05/24/2007) for the Reactor Protection System included the following reactor operator learning objective for Topic 3.6 (General Warning Reactor Trip):

  U 8966
  List the following information as it applies to the general warning reactor trip:
    - *Conditions that result in a general warning alarm*
    - *Local indications of a general warning alarm*

  The corresponding Section 3.6.2.1 of this lesson plan described that:

  "Each train's logic cabinet has lights that indicate the absence or presence of a General Warning condition, for example, the Train "A" logic cabinet has a red light which is normally off, and which will be lit if a General Warning exists on Train "A."

  The licensee's training program is based on a systems approach to training (SAT) and the training material learning objectives are directly linked to those tasks which are analyzed for the RO job. The North Anna RO lesson plan included a learning objective and the corresponding information that is required to correctly answer the test item. Therefore, the licensee's contention (that determining the status of the General Warning lights was not a part of the RO job duties) did not substantiate deleting this item from the exam.

- The NUREG 1122, Rev. 2, Supplement 1 (Knowledge and Abilities Catalog for Nuclear Power Plant Operators – Pressurized Water Reactors) K/A which this question was targeted to meet is:

  013 Engineered Safety Features Actuation System (ESFAS)
  K4.15 Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following: Continuous testing (2.6 / 3.2)

  The question was applicable at North Anna because the Safeguards Protection System Trouble alarms and General Warning Status lights were considered "design features and/or interlocks" which provide the means for continuous testing of the ESFAS and SSPS. Furthermore, the question meets the K/A and this was discussed and appropriate changes agreed on before the exam was administered.

  Because the test item was technically correct and also based on the guidance provided in NUREG 1021, Rev. 9, Supplement 1, ES-403, Section D.1.b, deletion of this test item from the exam was not warranted.

- The stem for Question #81 on the SRO written exam included the following initial plant condition statement:

  "1K-G1, SFGDS PROT SYS TR A TROUBLE annunciator was in the alarm condition."

The preceding statement (from Question #81) may have unintentionally aided the SRO applicants to eliminate distractors on question #15) because it provided cues that there were two separate annunciators – one for each train, “A” and “B” (versus only one Safeguards Trouble control room annunciator, common for both SSPS trains). This unintentional cue was not identified during the exam review process. However, the statement in the SRO Question #81 did *not* provide a cue for the second portion of the question, i.e., recognizing how a General Warning condition would affect the red light in the associated train’s logic cabinet (illuminated or extinguished). Although the examination analysis results reflected that the SRO applicants may have received unintended aid in eliminating distractors on this question, this does not invalidate this question on the RO exam.

NRC RESOLUTION
Question #15 is valid.

RO QUESTION # 20

LICENSEE COMMENT:
In summary, the licensee requested that this question be graded with two correct answers. The original post exam comments submitted by the licensee can be viewed in ADAMS under ML number ML082110241.

The licensee’s comment was that it was possible that the excess letdown system had been placed in service at the end of the previous cycle and not drained during the outage. In this case, the system would contain water with virtually zero boron concentration. Consequently, the result would be a dilution (versus boration) of the RCS when excess letdown was placed in service.

NRC DISCUSSION:
The initial conditions of the question stated that the unit was at 100% following a refueling outage. This test item asked the applicant 1) an excess letdown lineup restriction and 2) the reason why reactor power must be monitored when placing excess letdown in service in accordance with 1-OP-8.5. The second part of this test item was targeted to Precaution & Limitation (P&L) 4.7 at the front of 1-OP-8.5 (Rev 18), which states:

4.7 WHEN Excess Letdown is placed in service, THEN monitor RCS temperature and Reactor power closely due to the possible reactivity effects. A <u>dilution</u> may be required to maintain desired RCS temperature and Reactor power level. This is due to a potentially higher boron concentration in the Excess Letdown piping.

Additionally, the instructions for shifting from normal letdown to excess letdown (Section 5.1) included the following note:

***NOTE**: When Excess Letdown is placed in service, RCS temperature and Reactor power level should monitored closely due to the possible reactivity effects. A <u>dilution</u> may be required to maintain desired RCS temperature and Reactor power level.*

If the excess letdown system was placed in service at the end of the previous fuel cycle, then the piping would contain diluted water (versus a high boron concentration). In this case, a *boration* would have been required to maintain RCS temperature and reactor power level constant. Because the P&L and the note both stated that a dilution *MAY* be required and because the licensee's postulated scenario was operationally credible, then there were two correct answers for this test item.

NRC RESOLUTION
In accordance with NUREG 1021, Rev. 9, Supplement 1, ES-403, Section D.1.c, question # 20 will be graded with two correct answers. Either "A" or "D" is correct.

RO QUESTION # 23

LICENSEE COMMENT:
In summary, the licensee requested that this question be graded with two correct answers. The original post exam comments submitted by the licensee can be viewed in ADAMS under ML number ML082110241.

The licensee's comment was that technically, there was no functional difference between the two sequences of establishing Quench Spray (QS), i.e., discharge valves actuation first and then pump(s) actuation or vice versus.

NRC DISCUSSION:
The initial conditions for this question stated that a safety injection had occurred and containment pressure was 23 psia and slowly rising. The stem question asked the applicant to identify the appropriate operator action and the sequence of the action required.

Since the Containment Depressurization Actuation (CDA) auto actuation set point (27.75 psia) had not yet been achieved, the applicant was expected to eliminate choices "B" and "D" (because they both stated that CDA was required). Additionally, the applicant was expected to recognize that the QS function was required to be manually actuated in accordance with 1-E-0, Reactor Trip or Safety Injection, Step 12, because containment pressure had exceeded 20 psia. Furthermore, the applicant was expected to recall the sequence (discharge valve first, then pump) delineated in Step 12 for initiating QS.

1-E-0 provided two sequences for verifying or initiating the same pump(s). Step 11 was applicable when an auto CDA signal failed to occur and directed the operator to start the pump first and then manually open the discharge valve. Step 12 was applicable when manual initiation of the QS function (i.e., radio-isotope control afforded by the injection of NaOH from the Chemical Addition Tank.) was required and directed the operator to first open the discharge valve and then start the pump. In both cases, the final configuration of pump(s) running with discharge valve(s) open was achieved within a very short period of time with negligible short-term consequences. Since there is no documented technical basis difference between the two approved manual actuation sequences, either sequence was an acceptable response to the conditions stated in the question stem.

NRC RESOLUTION
In accordance with NUREG 1021, Rev. 9, Supplement 1, ES-403, Section D.1.c, question # 23 will be graded with two correct answers. Either "A" or "C" is correct.

RO QUESTION # 26

LICENSEE COMMENT:
In summary, the licensee requested that this question be graded with two correct answers. The original post exam comments submitted by the licensee can be viewed in ADAMS under ML number ML082110241.

The licensee's comment was that the conditions provided in the stem of the question did not give positive confirmation of the final PORV 1-RC-PCV-1455C valve position and functionality status. Consequently, the licensee contended that there were two scenarios, each with different Tech Spec (TS) required action statements.

NRC DISCUSSION:
This question asked the applicants to recognize the one hour action statement requirement for the following plant sequence of events:

Given the following:

- *Unit 1 is operating at 100% power*
- *An instrument failure causes PORV 1-RC-PCV-1445C to lift.*
- *RCS Pressure is recovering at a much slower rate than expected.*
- *The operator closes the PORV Block Valve, 1-RC-MOV-1536, and pressure is now 2100 psig and increasing as expected.*
- *All other equipment appears to be operating normally.*

The correct answer, ("A"), was based on TS 3.4.11, Condition B:

> *"One or more PORVs inoperable for reason other than Condition A <u>and capable of being manually cycled</u>."*

The licensee contended that choice "D" was also correct based on TS 3.4.11, Condition C:

> *"One PORV inoperable <u>and not capable of being manually cycled</u>."*

The applicants were expected to interpret the third (3rd) bullet (see above) to mean that 1) RCS pressure was rising <u>*before*</u> the Block Valve was manually closed and 2) that the 1-RC-PCV-1455C PORV did not fully seat following its actuation due to the instrument failure. The licensee contended that the wording for the third (3rd) bullet (see above) implied that the PORV could have been partially or fully open (vs. only leaking by at the valve seat). The design of the PORV is such that it is never throttled, only full open or full closed. If the PORV had remained open following its lift, then the pressurizer heater capacity cannot cause RCS pressure to rise.

The licensee also contended that by not providing a final valve position for 1-RC-PCV-1455C, that this validated an assumption that the PORV was *not* capable of being manually cycled. During the examination administration, one applicant asked the proctor whether 1-RC-PCV-1455C indicated shut. After consultation with the chief examiner, the licensee told the applicant to answer the question with the information given, i.e., no further clarification was provided to the applicant.

The following excerpt from NUREG 1021, Rev. 9, Supplement 1, Appendix E, (Policies and Guidelines for Taking NRC Examinations), Part B: Written Examination Guidelines, was read verbatim (by the chief examiner) to all the applicants before the exam was administered.

> *"When answering a question, do 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."*

The licensee's contention that choice "D" was also correct required an assumption (i.e., the PORV was not capable of being manually cycled) that was not a consequence of the conditions stated in the question.

NRC RESOLUTION
For question # 26, the only correct answer is "A."

RO QUESTION # 28

LICENSEE COMMENT:
In summary, the licensee requested that this question be graded with two correct answers. The original post exam comments submitted by the licensee can be viewed in ADAMS under ML number ML082110241.

The licensee's comment was that the applicants answered the question based on their knowledge of the background document discussions associated with FR-S.1, Response to Nuclear Power Generation/ATWS; therefore, automatic control rod insertion ("C") was also a correct choice.

NRC DISCUSSION:
This question asked the applicants to identify the action that would insert the most negative reactivity within the first 30 seconds following a 100% power ATWS at the end of core life. The choices provided were (answer key correct answer is "B"):

A. *Initiation of Emergency Boration*
B. *Manual Turbine Trip*
C. *Automatic Control Rod Insertion*
D. *Manual Control Rod Insertion*

Following a turbine trip, the moderator temperature coefficient (α_{Tmod}) reduces the core power as the coolant temperature rises. Additionally, α_{Tmod} becomes more negative at the end of core life. The reactivity effect of the turbine trip occurs at a much greater rate than the reactivity effect from automatic (or manual) rod insertion; especially within the first 30 seconds following a 100% power ATWS.

The licensee admitted in their post-exam comments that the correct answer from a transient analysis perspective was that the manual turbine trip added the most negative reactivity (vs. the automatic or manual rod insertion). However, the licensee's concern was that some applicants based their selection with the mindset that the question was asking for what the background document contained in its step discussions for FR-S.1, Step [2] (Verify Turbine Trip – Immediate Operator Action) and Step 4 (Initiate Emergency Boration of the RCS). The licensee referred to the following portions of the background document's discussion for Steps [2] and 4, respectively:

Step [2] discussion:
"The turbine is tripped to prevent an uncontrolled cool down of the RCS due to steam flow that the turbine would require. For an ATWS even where a loss of normal Feedwater has occurred, analyses have shown that a turbine trip is necessary (within 30 seconds) to maintain SG inventory."

Step 4 discussion:
"After control rod trip and insertion functions, boration is the next most direct manner of adding negative reactivity to the core."

The licensee contended that based only on these discussions provided in the FR-S.1 background document, the only information available stated that the *automatic* function of the control rods was the most direct manner of adding negative reactivity to the core (vs. *manual* rod insertion and emergency boration).

The question was soliciting the fundamental knowledge associated with the operational implications of the negative temperature coefficient as it applies to large PWR systems. The question was *not* soliciting the detailed knowledge of the licensee's background document associated with FR-S.1. Even so, the background document information associated with FR-S.1 did not conflict with the transient analysis perspective that the manual turbine trip added significantly more negative reactivity (vs. the automatic and/or manual rod insertion and emergency boration) within the first 30 seconds after the ATWS. Furthermore, the first action in FR-S.1 was to verify the reactor is tripped, which did not occur as stated in the stem. The second action was to verify that the turbine is tripped. The third action in the procedure was to verify rods inserting in AUTO at > 48 steps/minute. The FR-S.1 procedure sequence did not conflict with the transient analysis perspective that the manual turbine trip added significantly more negative reactivity within the first 30 seconds after the ATWS.

NRC RESOLUTION
For question # 28, the only correct answer is "B."

RO QUESTION # 43

LICENSEE COMMENT:
In summary, the licensee requested that this question be graded with two correct answers. The original post exam comments submitted by the licensee can be viewed in ADAMS under ML number ML082110241.

The licensee's comment was that the status of the SAFE/RESET green light was irrelevant and inconsequential during the plant conditions provided in the question. The licensee's comment was that the question required a level of recall that was beyond what is required to identify and verify proper response of the Containment High Range Radiation Monitors when an actual high radiation condition exists. Consequently, the licensee's comment was that both answers, i.e., LIT and NOT LIT, were correct.

NRC DISCUSSION:
The initial conditions provided in the question were that a LOCA had occurred on 30 minutes ago on Unit and both of the Containment High Range Radiation Monitors (1-RM-RMS-165 and 1-RM-RMS-166) had amber and red lights illuminated in the control room. The question asked the applicants to recognize 1) whether or not the radiation monitors' SAFE/RESET green light was lit and 2) the location and status of the Unit 1 CONT HI RANGE RADIATION TROUBLE alarm.

The licensee contended that the level of recall required by the applicants to ascertain the status of the SAFE/RESET green light for the given plant conditions was both irrelevant and too difficult. Consequently, the licensee contended that both answers, LIT and NOT LIT, were acceptable and correct.

The reactor operator lesson plan, STUDENT GUIDE FOR RADIATION MONITORING SYSTEM (46), (Revision 3, 09/19/2007) included the following learning objectives:

> *6.3 Objective*
> *U 5250*
> *List the means provided in the control room to determine the following abnormal conditions as they apply to the CHRRMS (Victoreen) containment radiation monitors.*
> - *Loss of detector and signal cable integrity*
> - *High containment radiation*
>
> *6.4 Objective*
> *U 5251*
> *Explain how the detector and signal cable integrity of the CHRRMS (Victoreen) containment radiation monitors is normally monitored.*

The licensee's training program is based on a systems approach to training (SAT) and the training material learning objectives are directly linked to those tasks which are analyzed for the RO job. The North Anna RO lesson plan included learning objectives and the corresponding information that was required to correctly answer the test item.

According to the lesson plan, the SAFE/RESET pushbutton/light had two important functions:

1) It was energized (green) when all conditions were normal and the system was operating normally. If a loss of detector or signal cable integrity for the Containment High Range Radiation Monitors (CHRRMS) occurred, the green SAFE/RESET light *de-energized* and the audible alarm annunciated on Unit-2. **[**Note: One common annunciator alarm per unit (UNIT 1 CONTAINMENT HI RANGE RADIATION TROUBLE, Window 2A-B3, and UNIT 2 CONTAINMENT HI RANGE RADIATION TROUBLE, Window 2A-C3) actuated for alert, high radiation, and failure. Both alarms were located on the Unit 2 annunciator panel 2-EI-CB-3.**]**

2) The SAFE/RESET pushbutton/light was used for acknowledging the ALERT (amber) and HIGH (red) alarms on the CHRRMS. The lights were acknowledged and reset by depressing the SAFE/RESET button.

The SAFE/RESET light was important because it alerted the operator that a problem had occurred with the detector or signal cable. This was important information both during normal operating conditions *and* during the conditions provided in this test item, i.e., during a LOCA. For the conditions provided in the stem of the question, the SAFE/RESET light was LIT, i.e., energized, because there was not a problem with the detector or signal cable. Therefore, the licensee's contention (that ascertaining the status of this light during high radiation conditions is irrelevant and/or required too much recall) did not substantiate accepting two answers for this item.

NRC RESOLUTION
For question # 43, the only correct answer is "A."

RO QUESTION # 50

LICENSEE COMMENT:
In summary, the licensee requested that this question be deleted from the examination. The original post exam comments submitted by the licensee can be viewed in ADAMS under ML number ML082110241.

The licensee's comment was that when placing the Charcoal Filter Inlet From Decay Tanks Controller, 1-GW-FCV-101, in service; the operator was directed by procedure to adjust the controller in MANUAL first before placing the controller in AUTOMATIC. The licensee's comment was that there was no correct answer for this question.

NRC DISCUSSION:
This question required the applicant to identify the *preferred* method for *controlling* the Waste Gas Decay Tank (WGDT) release flow rate in accordance with 0-OP-23.2, WGDT and Waste Gas Diaphragm Compressors.

0-OP-23.2, Precaution & Limitation 4.13 and the Note preceding step 5.4.11 both stated the following:

> *"It is preferred to operate 1-GW-FCV-101 in AUTO control to maintain flow < 3 SCFM. IF Manual control is required, THEN it is to only be done with SRO permission. (Reference 2.4.3)"*

The procedure specifically prohibited controlling the release in MANUAL without SRO permission. When the controller was in AUTOMATIC, the flow control valve would respond to pressure changes in the WGDT to maintain flow < 3 SCFM. The stem question specifically asked the applicant to identify the *preferred* method for controlling the release *in accordance with the procedure*. The licensee's contention was that the first few procedure steps for placing the controller in service constituted "controlling" the release. The first few steps of the procedure were associated with placing the controller in service (vs. controlling); therefore, these steps were not considered "controlling in MANUAL."

NRC RESOLUTION
Question #50 is valid.

RO QUESTION # 53

LICENSEE COMMENT:
In summary, the licensee requested that this question be deleted from the examination. The original post exam comments submitted by the licensee can be viewed in ADAMS under ML number ML082110241.

The licensee's comment was that an applicant who correctly implemented 1-AP-19, Loss of Bearing Cooling Water, (using the initial conditions provided in the stem) would not be directed to start the standby bearing cooling tower fan or to shift fans to fast speed because these actions were not included in the procedure. Furthermore, the licensee's comment was that since the annunciator response (ARs) procedures were normally performed in parallel with the 1-AP-16, then the question had three potentially correct answers because three of the choices were specifically directed by the AR procedures.

NRC DISCUSSION:
The question provides the initial conditions that the crew had entered 1-AP-19 following receipt of two control room alarms, 1T-C1 and 1A-F4. The question states:

> *"Which ONE of the following describes the* <u>*initial*</u> *action required in accordance with 1-AP-19?"*

Choice "C" was identified as correct on the answer key, i.e., *Start available Bearing Cooling Tower Fans or shift fans to high speed.* However, this action was directed by the 1A-F4 AR procedure (BASIN TEMP HI/LOW) versus being directed by 1-AP-19. Choices "A" and "D" were both actions directed by the 1T-C1 annunciator procedure (HYDROGEN TEMP OR CORE MONITOR). Choice "B" was a subsequent action directed by 1-AP-19, and thus was not considered an initial action. Therefore, there was no correct answer for this test item due to the wording of the question, i.e., *"..in accordance with 1-AP-19."*

NRC RESOLUTION
In accordance with NUREG 1021, Rev. 9, Supplement 1, ES-403, Section D.1.c, question #53 is deleted from the exam.

SRO QUESTION # 92

LICENSEE COMMENT:
In summary, the licensee requested that this question be graded with two correct answers. The original post exam comments submitted by the licensee can be viewed in ADAMS under ML number ML082110241.

The licensee's comment was that under normal conditions, the Service Water (SW) system is operated with the Closed Cooling (CC) Heat Exchanger Outlet Valves throttled. Furthermore, the licensee's comment was that when removing a SW Pump from service for planned or emergent maintenance, the associated SW Pump Maintenance Operating Procedure (MOP), used to remove the pump from service, required the establishment or verification of proper SW System throttling.

NRC DISCUSSION:
The initial conditions provided in the question were that both units were at 100% power with two SW pumps running (1-SW-P-1A and 1-SW-P-1B) and 2-SW-P-1B was out of service. Subsequently, the 1-SW-P-1B SW Pump tripped and the 2-SW-P-1A was started. The question asked the applicants to identify the TS action required and also the TS bases for the action.

The configuration for both units operating at 100% power was one pump running per loop, with the following normal pump alignment:

1-SW-P-1A aligned to the "A" loop (running)
1-SW-P-1B aligned to the "B" loop (initially running but subsequently tripped)
2-SW-P-1A aligned to the "B" loop
2-SW-P-1B aligned to the "A" loop (out of service)

Given the *initial* conditions provided in the question, i.e., both units at 100% power with the 2-SW-P-1B SW Pump out of service, the CC Heat Exchanger outlet valves were *required* to be throttled for two reasons:

1. Limiting Condition for Operation (LCO) 3.7.8 required two SW System loops to be operable in Modes 1, 2, 3, and 4. The LCO 3.7.8 bases stated:

   A SW loop is considered OPERABLE during MODES 1, 2, 3, and 4 when:

   a. Either
   a.1 Two SW pumps are OPERABLE in an OPERABLE flow path; or

   a.2 One SW pump is OPERABLE in an OPERABLE flow path provided two SW pumps are OPERABLE in the other loop and SW flow to the CC heat exchangers is throttled

   Because 2-SW-P-1B was initially out of service (inoperable), the a.1 statement was not satisfied and the licensee's LCO compliance was based on satisfying the a.2 statement.

2. The Maintenance Operating Procedure (MOP) 2-MOP-49.02 used to remove the 2-SW-P-1B SW Pump from service required that the SW outlet of the CC heat exchangers so that each SW Pump discharge pressure is > 54 psig using 0-OP-49.6, Service Water System Throttling Alignment.

The following excerpt from NUREG 1021, Rev. 9, Supplement 1, Appendix E, (Policies and Guidelines for Taking NRC Examinations), Part B: Written Examination Guidelines, applied:

"When answering a question, do 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."

Given the initial conditions provided in the question, i.e., both units at 100% power with the 2-SW-P-1B SW Pump out of service, the applicant was required to assume that the CC Heat Exchanger outlet valves were throttled since this condition occurred as a consequence of the initial conditions stated in the question.

With the CC Heat Exchanger outlet valves throttled, the applicant was expected to determine the TS action required following the trip of another SW pump, i.e., 1-SW-P-1B. TS 3.7.8, Condition B was applicable when two SW pumps were inoperable. This required action was:

> *B.1 Throttle SW System flow to CC heat exchangers within 1 hour*
>
> *AND*
>
> *B.2 Restore one SW pump to OPERABLE status within 72 hours*

Because the CC Heat Exchanger outlet valves were throttled, compliance with B.1 had previously been established. Therefore, the TS action required following the trip of the 1-SW-P-1B pump was B.2, i.e., Restore one SW pump to operable within 72 hours. This eliminated choices "B" and "D."

The second part of each choice required the applicant to recognize the basis for the required TS action. According to the TS 3.7.8 bases for action B.2, with only two operable SW Pumps, the safety function of providing design SW flow to the Recirc Spray (RS) Heat Exchangers following a LOCA was still met assuming NO additional failures. The bases stated that restoring one SW pump to operable status within 72 hours together with the throttling ensured that design flow to the RS Heat Exchangers was achieved following an accident. In this configuration, a single failure disabling a SW pump would not result in loss of the SW system function. Consequently, the correct answer is "A."

NRC RESOLUTION
For question # 92, the only correct answer is "A."

SRO QUESTION # 96

LICENSEE COMMENT:
In summary, the licensee requested that this question be graded with two correct answers. The original post exam comments submitted by the licensee can be viewed in ADAMS under ML number ML082110241.

The licensee commented that the governing procedure (VPAP-1403, Temporary Modifications) required that temporary modification jumper packages received final approval by the FSRC whereas the actual jumper installation (in the plant) received final approval by the Shift Manager. The licensee contended that the stem question was not specific with respect to asking who provided final approval of the temporary modification jumper package versus asking who provided the final approval for the jumper to be installed in the plant. Consequently, the licensee contended that both choices (i.e., Shift Manager and FSRC) were acceptable and correct.

NRC DISCUSSION:
The question asked the applicant to identify 1) the governing procedure for installation of a jumper on a multiple input annunciator (1D-H5, HIGH CAPACITY S/G BLOWDOWN TROUBLE) and 2) who, by title, must provide final approval of the jumper. [This was a multiple input annunciator which required an interim jumper installation to restore functionality of the alarm.]

OP-AA-100, Conduct of Operations, Rev 0, Attachment 2 (Shift Operations) stated that "In the case of a multiple-input annunciator, for example, implementation of a temporary modification may be considered to restore functionality to unaffected circuitry." The stem of the question specifically stated that a jumper was required to restore functionality of the alarm. Consequently, the governing procedure for this jumper installation at North Anna was VPAP-1403, Temporary Modifications (versus OP-NA-200-1001, Equipment Clearance Process).

The jumper approval and installation process (in accordance with VPAP-1403, Revision 11) was as follows:

- Originator forwards the temporary mod package to the Shift Technical Advisor (Step 6.4.12)
- Shift Technical Advisor reviews the package and forwards to the Shift Supervisor (Step 6.5.3)
- Shift Supervisor approves the package (Step 6.6.10)
- Shift Supervisor obtains the Manager of Nuclear Operations or Operations Manager on call approval (Step 6.6.11)
- Shift Supervisor forwards the package to the Originator (Step 6.6.12)
- Originator obtains SNSOC Chairman's signature for SNSOC's approval of the package (Step 6.7.2)
- Originator forwards the package to the Temporary Modification Installer (Step 6.7.3)
- Installer obtains the Shift Supervisor's approval for installation (Step 6.8.1.a)
- Installer installs the approved modification (Step 6.8.1.c)

The stem question asked "Which procedure governs the *installation* of the jumper, and who, by title, must provide **FINAL** approval of the jumper?" The question stem referred to the jumper *installation* (versus jumper package) and also asked for *the very last* approval. In accordance with VPAP-1403, Temporary Modifications, Rev 11, Step 6.8.1.a, the *installer must get the Shift Supervisor's approval*. This was the FINAL approval of the jumper in accordance with the process described in the procedure.

The licensee provided documentation that the Job titles for Shift Supervisor/Nuclear Shift Supervisor and Assistant Shift Supervisor/Nuclear Assistant Shift Supervisor at the North Anna and Surry Power Stations were changed to Shift Manager and Unit Supervisor, respectively. Although the change was effective June 1, 2003, dual titles were maintained until regulatory relevant documents were revised and until the old titles were removed from Station/ISFSI Technical Specifications, Quality Assurance Topical Report, Emergency Plan, Safety Analysis Report, and Station/Corporate procedures, programs and standards when the need arises for a document revision.

NRC RESOLUTION
For question # 96, the only correct answer is "B."

SIMULATION FACILITY REPORT

Facility Licensee: North Anna Power Station

Facility Docket Nos.: 05000338/05000339

Operating Tests Administered on: June 2-20, 2008

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 in accordance with IP 71111.11, are not indicative of noncompliance with 10 CFR 55.46. No licensee action is required in response to these observations.

No simulator fidelity or configuration items were identified.