CCS-118

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

February 5, 2009

Mr. J. R. Morris
Site Vice President
Duke Power Company, LLC
d/b/a Duke Energy Carolinas, LLC
Catawba Site
4800 Concord Road
York, SC 29745-9635

SUBJECT: CATAWBA NUCLEAR STATION – NRC OPERATOR LICENSE EXAMINATION REPORT 05000413/2008301 AND 05000414/2008301

Dear Mr. Morris:

During the period December 1 - 4, 2008 the Nuclear Regulatory Commission (NRC) administered operating tests to employees of your company who had applied for licenses to operate the Catawba Nuclear Station. At the conclusion of the tests, the examiners discussed preliminary findings related to the operating tests with those members of your staff identified in the enclosed report. The written examination was administered by your staff on December 10, 2008.

One Reactor Operator (RO) and three Senior Reactor Operator (SRO) applicants passed both the operating test and written examination. One RO and four SRO applicants failed the written examination. There were ten, (10) post-administration comments concerning the written examination. These comments, and the NRC resolution of these comments, are summarized in Enclosure 2. A Simulator Fidelity Report is included in this report as Enclosure 3.

The draft written examination submitted by your staff failed to meet the guidelines for quality contained in NUREG-1021, Operator Licensing Examination Standards for Power Reactors, Revision 9, Supplement 1, as described in the enclosed report.

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 the NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm.adams.html (the Public Electronic Reading Room).

If 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.: 50-413, 50-414
License Nos.: NPF-35, NPF-52

Enclosures: 1. Report Details
2. Facility Comments and NRC Resolution
3. Simulator Fidelity Report

cc w/encl: (See page 3)

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Letter to J. R. Morris from Malcolm T. Widmann dated February 5, 2009

SUBJECT: CATAWBA NUCLEAR STATION – NRC OPERATOR LICENSE EXAMINATION REPORT 05000413/2008301 AND 05000414/2008301

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Distribution w/encls:
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PUBLIC
J. Thompson, NRR (PM: CAT, MCG)
J. Stang, NRR

(*) See previous concurrence

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-413, 50-414
License No.:	NPF-35, NPF-52
Report No.:	05000413/2008301, 05000414/2008301
Licensee:	Duke Energy Corporation (DEC)
Facility:	Catawba Nuclear Station, Units 1 & 2
Location:	4800 Concord Road York S.C. 29745
Dates:	Operating Test – December 1 - 4, 2008 Written Examination – December 10, 2008
Examiners:	Gerard Laska, Chief Examiner, Senior Operations Examiner Frank Ehrhardt, Senior Operations Engineer Craig Kontz, Operations Engineer Michael Meeks, Operations Engineer (In-Training)
Approved by:	Malcolm T. Widmann, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000413/2008301, 05000414/2008301, 12/01-04/2008 & 12/10/2008; Catawba Nuclear Station; Operator License Examinations.

Nuclear Regulatory Commission (NRC) examiners conducted an initial examination in accordance with the guidelines in Revision 9, Supplement 1, of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors." This examination implemented the operator licensing requirements identified in 10 CFR §55.41, §55.43, and §55.45, as applicable.

Members of Catawba Nuclear Station staff developed both the operating tests and the written examination. The final written examination submittal was considered to be outside the acceptable range because it did not meet the quality guidelines contained in NUREG-1021.

The NRC administered the operating tests during the period December 1 - 4, 2008. Members of the Catawba Nuclear Station training staff administered the written examination on December 10, 2008. One Reactor Operator (RO) and three Senior Reactor Operator (SRO) applicants passed both the operating test and written examination. One RO applicant and four SRO applicants failed the written examination. Two applicants (one RO and one SRO) were issued licenses. Two SRO applicants received pass letters pending results of any appeals.

There were ten (10) post-examination comments.

REPORT DETAILS

4. OTHER ACTIVITIES

4OA5 Operator Licensing Examinations

a. Inspection Scope

Members of the Catawba Nuclear Station staff developed both the operating tests and the written examination. All examination material was developed in accordance with the guidelines contained in Revision 9, Supplement 1, of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors." The NRC examination team reviewed the proposed examination. Examination changes agreed upon between the NRC and the licensee were made per NUREG-1021 and incorporated into the final version of the examination materials.

The NRC reviewed the licensee's examination security measures while preparing and administering the examinations in order to ensure compliance with 10 CFR Part 55.49, "Integrity of examinations and tests."

The NRC examiners evaluated two Reactor Operator (RO) and seven Senior Reactor Operator (SRO) applicants using the guidelines contained in NUREG-1021. The examiners administered the operating tests during the period December 1 - 4, 2008. Members of the Catawba Nuclear Station training staff administered the written examination on December 10, 2008. Evaluations of applicants and reviews of associated documentation were performed to determine if the applicants, who applied for licenses to operate the Catawba Nuclear Station, met the requirements specified in 10 CFR Part 55, "Operators' Licenses."

b. Findings

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

The NRC determined that the licensee's original written examination submittal was within the range of acceptable quality specified by NUREG-1021. However, based on post exam comments whereby five (5) additional questions were determined to be unsatisfactory the final written examination was determined to be outside of the range of acceptable quality. More than 20% (24 of 100) of questions sampled for review contained unacceptable flaws. Individual questions were evaluated as unsatisfactory for the following reasons:

- 10 questions failed to meet the K/A statement contained in the examination outline.
- 2 questions contained two or more implausible distractors.
- 9 questions on the SRO examination were not written at the SRO license level.
- 3 questions contained other unacceptable psychometric flaws.
- 4 questions contained multiple unacceptable flaws.

Future examination submittals need to incorporate lessons learned.

One RO applicant and three SRO applicants passed both the operating test and written examination. Two applicants (one RO and one SRO) were issued licenses. One RO applicant and four SRO applicants passed the operating test but did not pass the written examination.

One SRO applicant passed the operating test, but passed the written examination with an overall score between 80% and 82% and the SRO-only portion with a score between 70 and 74 %. One SRO applicant passed the operating test, but passed the SRO-only portion of the written examination with a score between 70% and 74%. Each of these applicants were issued a letter stating that they passed the examination and issuance of their license has been delayed pending any written examination appeals that may impact the licensing decision for their application.

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

The licensee submitted ten (10) post-examination comments concerning the written examination. A copy of the final SRO and RO written examination and answer key, with all changes incorporated, and the licensee's post-examination comments may be accessed in the ADAMS system (ADAMS Accession Number(s) ML090230071, ML090230075, ML090230038, and ML090230060).

4OA6 Meetings, Including Exit

Exit Meeting Summary

On December 4, 2008, the NRC examination team discussed generic issues associated with the operating test with Mr. James R. Morris, CNS Site Vice President, and members of the Catawba Nuclear Station staff. The examiners asked the licensee if any of the examination material was proprietary. No proprietary information was identified.

KEY POINTS OF CONTACT

Licensee personnel

J. Morris, CNS Site Vice President
R. Hart, CNS Manager Regulatory Compliance
R. Weatherford, Training Manager
J. McConnell, Shift Operations Manager
S. Coy, Operations Training Manager
H. Dameron, Operations Initial Training Supervisor
G. Hamilton, Operations Training
J. Suptela, Operations Training
T. Garrision, Operations Training

NRC personnel

A. Sabisch, SRI
R. Cureton, RI

FACILITY POST-EXAMINATION COMMENTS AND NRC RESOLUTIONS

A complete text of the licensee's post-examination comments can be found in ADAMS under Accession Number ML090230060.

Written Examination - Question # 5

Licensee's Comment:

Operation's administrative procedure OPM 1-7 (Emergency/Abnormal Procedure Implementation Guidelines) has a section 7.6 (Deviation From Approved Procedures) and 7.7 (Situations Not Covered by Procedure). The question developers considered this condition to fit into a situation not covered by procedure; therefore, OMP 1-7 section 7.7 would apply and paragraph B which states, "The planned course of action shall be reviewed and approved by a second SRO…" would require one additional SRO to approve the desired action.

The applicants who chose answer C believed that OMP 1-7 section 7.6 (Deviation From Approved Procedures) applied because the stem of the question stated that the OSM determined an immediate need to take action. Section 7.6 paragraph C.3 states that actions outside approved procedures can be taken when, "Actions are needed to minimize immediate personnel hazard/injury or damage to plant equipment." Section 7.6 paragraph D. 2 states that only one SRO must approve the action.

If the OSM's chosen actions are taken from various procedures unrelated to the current condition, then section 7.6 would apply. If the OSM's chosen actions aren't described in any procedure then section 7.7 would apply. The question didn't provide enough information for the applicants to know whether to apply section 7.6 or 7.7. Therefore, we request that both answers C and D be accepted as correct answers.

NRC Response:

The NRC agrees that the stem of the question did not provide enough information for the applicants to determine which section of OMP1-7 to apply, and that there is a basis for two possible correct answers given. However, the two answers contain conflicting information as described in NUREG 1021 Revision 9 Supplement 1, 403 D.1.c. Therefore, this question will be deleted from the examination.

Written Examination - Question # 19

Licensee's Comment:

Unwarranted continuous rod movement is an entry condition for procedure AP/1/A/5500/015 Rod Control Malfunction Case II. The immediate actions of the AP are to place the rod bank select switch in manual, verify rod motion stops and trip the reactor if the rods continue to move. The question developers considered strict procedural compliance when developing the question.

The intent of the step C.1 is to remove the CRD Bank Select switch from Auto. Any rod movement with the switch in any position other than Auto indicates a fault in the rod control system, and a reactor trip is warranted.

If the CRD Bank Select switch is in any position other than AUTO the rods can only be moved manually. The applicants who selected answer B applied NSD 705 allowance of intent met, and understood that any position other than AUTO is a position that only supports manual control of the rods; therefore, the intent of step C 1 was already met and the only required action was an immediate trip of the reactor per step C.2 RNO.

The applicants who selected answer D considered that strict procedural compliance required the rod bank select switch to be placed in the MAN position. Per strict procedural compliance answer D is correct.

Using the allowance of intent met, the first required action is to trip the reactor, and answer B is correct. Considering strict procedure compliance the first required action is to place the CRD Bank Select switch to manual, so the first require action is to place the switch in the MAN position, and answer D is correct. Therefore, we request that both answers B and D be accepted as correct.

NRC Response:

The NRC does not agree with accepting both answers B and D as correct. In this case placing the rod control select switch in manual may have stopped rod motion. Shutdown banks do not move out when the rod control select switch is in the manual position. Rod speed in the SBB position is 64 steps per minute (spm), and rod speed in the manual position is 48 spm. The path that the rod out impulse takes is different in SBB and manual positions. Therefore, it is important to take the rod control select switch to manual in this case. The first Immediate Action of AP/1/A/5500/015 Rod Control Malfunction Case II, specifically states to place the switch in Manual.

Also noteworthy was the fact that the reference stated above that the applicants applied allowing them to assume that the intent of the step was met was titled NSD 705 "Instructions for the Verification and Validation of Technical Procedures." This NSD did not contain directions on procedure use. Furthermore it was discovered that NSD 704 "Technical Procedure Use and Adherence," which was the NSD that did contain direction on when the intent of a step was met, was not applicable to abnormal and emergency procedures.

Therefore, answer D is the only answer that will be accepted as correct.

Written Examination - Question # 23

Licensee's Comment:

The question developer considered the EMFs 71-74 to be correct because their location on the steam lines makes them the first monitors to detect the change in secondary contamination. The applicants who chose answer C selected 1EMF 33 because it will be the first EMF to generate an alarm.

The question asks for the, "best indication (most sensitive and most timely)." The candidates selected different answers due to making different assumptions about what indication is being observed. Normally the operators infrequently monitor the EMF readings but are frequently monitoring the EMFs' alarm state. AP/1/A/5500/003 (Load Rejection), which would have been implemented due to the runback, does not require the operators to monitor the EMF readings. AP/1/A/5500/010 (NC System Leakage), which would be entered once a tube leak greater than 5 gpd is detected, requires monitoring of EMF readings every 15 minutes but only if the SG leak rate is greater than 40 gpd. Given the situation described in the question the operators would be monitoring the EMF alarm state not the EMF readings.

In accordance with NSD 513 (see attached) EMFs 71-74 are set to alarm at 5 GPD. 1EMF-33 readings input to a calculation that runs continuously on the Operator Aid Computer (OAC). Per NSD 513 that calculation is set to produce an OAC alarm at 5 gpd. 1EMF-33 will produce an alarm on the annunciator panel based upon a predetermined increase in count rate above the background. Consequently, EMF-33 produces an annunciator due to increasing count rate before an OAC alarm based upon the calculated leak rate.

EMFs 71-74 are located on the steam line coming from each of the SGs. EMF-33 is monitoring the offgas from the condenser air ejectors. Due to their locations, EMFs 71-74 will be the first to detect an increase in secondary activity due to a tube leak.

This scenario was performed on the simulator at 100% power and again after a runback on loss of a CF pump. A 12 gpd leak in 1A SG was inserted, and in both cases 1EMF-71 count rate was the first EMF to increase, but 1EMF-33 was the first EMF to produce an alarm.
Based upon observing the EMF alarm status EMF-33 will be the timeliest indicator, which would make answer C correct.

Based upon monitoring the EMF readings EMFs 71-74 will be the timeliest because they are the first monitors to be exposed to the increase in secondary activity which makes answer D correct.

Since the question didn't clearly ask if the operators were monitoring the EMF readings or alarm state, we request that both answers C and D be accepted as correct.

NRC Response:

The NRC does not agree with accepting answers C and D. The stem of the question asked for, "Which one of the following indicators will provide the best indication (most sensitive and timely) that the S/G tube leak has increased." The question did not ask which one would alarm first. The applicants who chose distractor C assumed that the indication would have to cause an alarm first to alert the control room. NUREG 1021 appendix E, part B (7) states:

“If you have any questions concerning the intent or the initial conditions of a question, do *not* hesitate to ask them before answering the question. Note that questions asked during the examination are taken into consideration during the grading process and when reviewing applicant appeals. Ask questions of the NRC examiner or the designated facility instructor *only.* A dictionary is available if you need it.

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. For example, you should not assume that any alarm has activated unless the question so states.”

Therefore, answer D is the only correct answer.

Written Examination - Question # 42

Licensee's Comment:

The question developer considered the level required to support all ECCS and NS pumps taking suction on the containment sump. The crew enters EP/ES-1.3 when the FWST level decreases to 37%. The ND pump suctions automatically align to the containment sump, and the operators will align the remaining ECCS pumps suctions' from the FWST to the ND pump discharge per ES-1.3. When FWST level decreases to 11% the operators will align the NS pumps' suction to the containment sump per ES-1.3.

The stem states that the crew has just entered EP/ES-1.3; therefore, at that point in time the only pumps with their suction aligned to the sump are the ND pumps and all other pumps are still aligned to the FWST. EP/ES-1.3 step 2 checks for a sump level > 3.3 feet. If it isn't, the RNO verifies sump level > 2.5 feet at step 2.f. If level is > 2.5 feet then the NV and NI pumps' suctions can be aligned to the containment sump. In this situation a level of 2.5 feet will support the operations all ECCS pumps while the NS pumps are still aligned to the FWST.

When the FWST level decreases to 11% ES-1.3 directs aligning the NS pumps to the containment sump using enclosure 2. Step 2 checks for a sump level of > 3.3 feet. If it isn't then the NS pump suction isn't aligned to the containment sump. Therefore, after FWST level has decreased to 11%, 3.3 feet in the containment sump is required to support operation of all ECCS pumps.

The stem didn't provide the applicants information concerning the FWST level. That information is needed to determine which pumps are supposed to be aligned to the containment sump. If FWST level is <37% and > 11%, then answer B is correct. If the FWST level is < 11% then answer C is correct.

Since the question didn't have enough information for the applicants to know the point in time they are required to evaluate the question, we request that both answers B and C be accepted as correct.

NRC Response:

The NRC does not agree with accepting both B and C as correct. After a review of the procedure and the construction of the stem of the question, "What is the minimum containment sump level that will support operation of all ECCS pumps and the NS pumps," it is clear that the question is asking for the containment sump level as specified in ES-1.3 (Transfer to Cold Leg Recirculation) that would be sufficient to provide a net positive suction for all ECCS pumps and NS pumps. In accordance with ES-1.3 (Transfer to Cold Leg Recirculation), a containment sump level of greater than 3.3 feet is required for all pumps to take a suction on the containment sump.

Therefore, answer C is the only correct answer.

Written Examination - Question # 55

Licensee's Comment:

Valve VQ-10 gets a close signal at 0 psig. The fans are large enough to reduce containment pressure below the Tech Spec limit (See Attached). Therefore, basis for closing the valve at that 0 psig is to prevent the VQ fans from reducing containment pressure to the minimum tech spec value.

The 6 applicants who chose answer D rejected answer C because the wording of the answer implied that the minimum tech spec value had been reached when the valve closed which is incorrect since the minimum tech spec value is -0.1 psig. At 0 psig the plant is in compliance with Tech Specs; therefore, the answer is not technically correct. Had the answer stated, "To prevent non-compliance…" then the answer would have been correct.
The VQ fans are sized small enough to prevent them from opening the ice condenser doors; therefore answer D is wrong. (See attached.)

We recommend that this question be deleted from the exam since there is no technically correct answer.

NRC Response:

The NRC agrees with the licensee in that there is not a technically correct answer, and the question will be deleted from the examination.

Written Examination - Question # 76

Licensee's Comment:

The stem told the applicant that the ND suction relief was leaking. The applicant was required to know that the ND relief valves discharge to the PRT. The applicant was also required to know that AP/27 will transition the operator to AP/19 if PRT level is increasing without indication that the input is from the NC system pressurizer.

The symptoms of this event would be pressurizer level and pressure decreasing and PRT level increasing. These symptoms match the entry conditions for AP/19 rather than AP/27. (See attached.) Therefore, entry into AP/27 was an incorrect diagnosis of the event. In the event of a leak step 3 of AP/27 will stop any ND pump taking suction on an NC system loop to protect the ND pump from damage.

AP/27 step 4 looks for PRT level increasing without indication of safety valve input. The intent of this step is to rule out input to the PRT from the pressurizer safety. If the safety valve is not discharging to the PRT, then the procedure assumes the input is from the ND system and the operator is directed to transition to AP/19. The only indication available for the operator to determine if the PRT input is from a pressurizer safety is safety valve tailpipe temperature and acoustic flow monitors. The question did not provide the applicant the status of those indicators. Additionally, the question didn't provide the applicant with information about the status of PRT level before or after the actions of AP/27 were performed. AP/27 step 4 doesn't specifically state <u>pressurizer</u> safety. The ND relief valve is a safety valve which discharges to the PRT.

The background document for that step doesn't clarify that the step applies to pressurizer safety valves. The stem stated that the ND relief was open can be interpreted as indication that a safety is discharging to the PRT. Procedure change request number CNS-2008-5216 has been submitted to revise AP/27 step 4a to state "pressurizer safety valve." All of the applicants correctly answered part 2 of the question. However, the applicants were not given information about pressurizer safety valve status and PRT level response which was needed adequately determine the proper procedure flowpath.

Given the ambiguity of AP/27 step 4, and the lack of information to properly evaluate the status of the pressurizer safeties and PRT level we request that question 76 be deleted.

NRC Response:

The NRC does not agree with deleting the question. After reviewing the entry conditions for AP/27, and AP/19, it appears that either procedure could be entered for the above conditions. Furthermore, the question stated that AP/27 was entered.

Step 4 of AP/27 States: Verify Leak is on ND:

a. **Plant alarms and indications – INDICATE LEAK OUTSIDE CONTAINMENT**

   OR

   PRT Level - INCREASING WITHOUT INDICATION OF SAFETY VALVE INPUT.

b. **GO TO AP/1/A/5500/019 (Loss of Residual Heat Removal System).**

The stem of the question states that the leak was from the ND via one of the ND suction relief valves that had lifted and **HAD FAILED TO RESEAT.** With the information given in the stem, and not making any new assumptions, the applicant had enough information to determine/verify that the leak is on the ND system and that a transition to AP/19 is clearly warranted.

Therefore, answer D is the only correct answer.

Written Examination - Question # 77

Licensee's Comment:

The developer considered the basis for Tech Spec 3.8.1 which states either off site or on site power is available, and in this scenario off site power is maintained. Therefore, entry into 3.0.3 was considered to be the time that the design criteria were no longer met. Tech Spec 3.8.1 action B2 requires declaring 2B NI inoperable 4 hours after 2B DG was declare inoperable. Thus, at 0700 2B NI is still considered operable; so, the ECCS design criteria for a large break LOCA was met.

All of the applicants that selected answers C and D understood that the 2B NI didn't have to be declared inoperable until 0900. Those who selected answer D considered design criteria to be separate from the declaration of inoperability. Declaration of inoperability is an administrative function. The Regulatory Compliance department was asked to interpret this scenario. Regulator Compliance contacted Excel Services who writes our Tech Specs. The following is their reply:

From: Dan Williamson [dan.williamson@excelservices.com]
Sent: Monday, December 15, 2008 8:32 PM
To: pwrog@excelservices.com
Subject: RE: Initial License Exam TS Question

Few comments:
>> If not yet adopted, consult TSTF-273 for "intent clarifications" related to this situation.
>> The "ECCS design criteria for a large LOCA" is different than "loss of safety function" typically used in TSpecs / SFDP. The "design criteria" was not met when the first 1A SI was inop --> loss of single failure protection.
>> The example is a bit confusing when the ending question mentions "when 2A DG becomes inoperable" -- prior to this, 2A DG was not at issue (?) Seems a typo of some kind.
>> The [A-SI + B-DG] is still is not a "loss of safety function" (see TSTF-273). The directed declaration of B-SI inop at 4 hrs due to B-DG inop (and one can wait the full 4 hours to make this declaration) can be argued to be the first time that a "loss of safety function" exists --> both A & B SI inop.

Dan Williamson
EXCEL Services Corporation
Main Offc/Cell: (904) 272-5300

Given that the design criteria were not met when 2A NI pump was declared inoperable, we request that the correct answer be changed to D.

NRC Response:

NRC agrees with the licensee's explanation. It is clear that the thought process involved in the question development was equipment operability and not actual ECCS design criteria. The exam key will be changed to make D the correct answer.

Written Examination - Question # 83

Licensee's Comment:

When plant control is aligned to the control room and a VCT Lo-Lo Level (4.3%) is detected the suction valves from the FWST open and the suction valves from the VCT close. The Design Basis Document for Loss of Control room states that all automatic NV functions are disabled when control is transferred to the Auxiliary Shutdown Complex (ASC). The DBD also states that the suction valves from the VCT open upon transfer to the ASC and are blocked from closing on Lo-Lo Level. The Loss of Control Room lesson plan states the same information found in the DBD. The DBD for the NV system doesn't discuss how the suction valves from the FWST are affected by swapping control to the ASC. Additionally, AP/1/A/5500/017 (Loss of Control Room) Enclosure 1 page 12 directs manual alignment of the NV pump suction to the FWST if VCT level is < 23%. The background document for the procedure states that, "All automatic transfer of the NV pump suction to the FWST on low VCT level is lost when control is transferred to the ASP." Based upon controlled information available to the question developers they determined that the automatic swap of the NV pump suction to the FWST on lo-lo VCT level would not occur.

During the exam review the applicants stated that they were taught that the swap to the FWST will occur automatically. The instructor who teaches the Loss of Control Room had determined that the suction valves from the FWST are unaffected by the swap to the ASC, and had included that information in the notes section of the Power Point presentation used to teach the lesson. A copy of the Power Point presentation had been provided to the applicants. The notes section of a single slide of the presentation includes the statement, "NV-252A & NV-253B will auto open on Lo-Lo VCT level, but NV-188A & NV 189B will not close." Brian Woolweber (Senior Engineer) and Nick Burgess (Engineer III) reviewed the electrical drawings and confirmed that the FWST suction valves are unaffected by a swap to the ASC and will in fact open on a VCT Lo-Lo- Level signal. (See attached note from engineering.)

Answer B is technically correct because if the suction of the NV pumps isn't manually aligned to the FWST when VCT level is < 23%, then the valves will automatically open on Lo-Lo VCT level and primary side makeup would be assured.

Answer D is technically correct because the suction supply valves from the FWST are manually opened per the requirements of procedure AP/17 to ensure primary side makeup is assured. We request that both answers B and D be accepted as correct.

NRC Response:

The NRC does not agree with accepting B and D as being correct. Per AP/1/A/5500/017 (Loss of Control Room), primary side inventory is assured by manually swapping NV pump suction to the FWST.

Step 17 of AP/1/A/5500/017 (Loss of Control Room) Enclosure 1 ASP actions directs the operator to:

> 17. **Control VCT level as follows:**
>
> a. Ensure charging and letdown flow -
> ADJUSTED TO MAINTAIN PZR LEVEL
> AT 25%.

b. **IF** NV pump suction aligned to the VCT,
THEN maintain VCT level using one of the following:
 Normal boration:

1) Start a Boric Acid Transfer pump.

2) Open 1NV-186A (B/A Blender Otlt To VCT Otlt).

3) Open 1NV-238A (B/A Xfer Pmp To Blender Ctrl).

OR

 Emergency boration:

1) Start a Boric Acid Transfer pump.

2) Open 1NV-236B (Boric Acid To NV Pumps Suct).

c. Verify VCT level - GREATER THAN 23%.

c. **IF** VCT level decreasing,
THEN notify
Unit 1 Aux Bldg operator to align Unit 1 NV pump suction to FWST.
REFER TO Enclosure 4 (Aux Bldg Operator Actions), Step 5.

d. **IF AT ANY TIME** VCT level decreases to less than 23%, **THEN** perform 17.c RNO.

The procedure directs the operator to **manually** align FWST suction to the NV pump if VCT level decreases to <23 %, this manual alignment also includes manual closing 1NV-188A and 1NV-189B. A note prior to step 14 of enclosure 1 states:

> **CAUTION With NV pump suction valves from the VCT (1NV-188A and 1NV-189B) and FWST (1NV-252A or 1NV-253B) open, suction supply may be lost when VCT level drops to 0% due to the H_2 pressure maintained in the VCT.**

The automatic opening of 1NV-252A or 1NV-253B when VCT level decreases to less than 4.3%, does not assure a continued NV pump suction because signal this does not close 1NV-188A and 1NV-189B, as stated in the Licensee's description above. Based on the above note this is required to assure NV pump suction. Therefore answer D is the only correct answer.

Written Examination - Question # 87

Licensee's Comment:

During the first stage of a LOCA the ice condenser is the major heat sink for cooling the containment atmosphere. After the ice has melted then NS becomes the major heat sink. RN flow rate to the NS heat exchangers is a constant value; therefore, the temperature of NS is directly related to RN temperature. Once the ice has melted containment pressure will be related to the NS temperature, and if NS temperature is higher, then containment pressure will be higher. The higher NS temperature would have little to no affect on containment pressure before the ice melts because the ice is the major heat sink, but pressure would be affected after the ice was melted. (See attached excerpt from Tech Spec 3.7.9 bases.) The developer included the word significant in the second part of the answer because the difference in NS temperature will be observable to the operator in the control room.

If the Lake Wylie temperature reaches the SLC limit, the remedial action is to align at least one train of RN to the Standby Nuclear Service Water Pond (SNSWP). The Tech Spec basis for the (SNSWP) states, "NSWS (Nuclear Service Water System) temperature influences containment pressure following a Loss of Coolant Accident and offsite dose following a Main Steam Line Break. The containment peak pressure analysis can accommodate NSWS temperatures up to 100°F. " Since the Lake Wylie temperature, thus NSWS temperature had not exceeded 100°F the applicants who chose answer D determined that the elevated RN temperature would <u>not</u> have a "significant" affect on containment temperature. Therefore, they rejected answer C and selected answer D as the most correct for the given conditions.

Question 2 asked the applicant to compare the affect of the higher lake temperature, but it doesn't ask which higher temperature to use, the last observed or the SLC limit, or what temperature it should be compared to. In reference to answer C for question 2, the first part of the answer is correct, but statement that the affect would be significant cannot be supported since question didn't imply how big a temperature difference to consider. Consequently, answer C cannot be supported as correct.

Answer D is a correct answer since all of the temperatures given for comparison are below the analyzed value of 100°F. Thus, the impact or consequence would be minimal throughout the entire sequence of the accident.

We request that the correct answer be changed to D.

NRC Response:

The NRC does not agree that answer D is a correct answer. SLC16.9.4 states that the water temperature of Lake Wylie shall be ≤ 95.5 ºF. If temperature is greater than 95.5 ºF the SLC directs the operator to align at least one NSWS loop to the Standby Nuclear Service Water Pond (SNSWP). Therefore actions must be taken to prevent exceeding any accident analysis assumptions. TS 3.7.9 "Standby Nuclear Service Water Pond" basis document discusses the effects of elevated NSWS temperatures. The basis document states in part:

> The peak containment pressure occurs when energy addition to containment (core decay heat) is balanced by energy removal from the Containment Spray and Component Cooling Water heat exchangers. This balance is reached after the transition from injection to cold leg recirculation and after ice melt. Because of the effectiveness of the

ice bed in condensing the steam which passes through it, containment pressure is insensitive to small variations in containment spray temperature prior to ice meltout.

Long term equipment qualification of safety related components required to mitigate the accident is based on a continuous, maximum NSWS supply temperature of 100°F or less.

To ensure that the NSWS initial temperature assumptions in the limiting analysis are met, Lake Wylie temperature is also monitored. During periods of time while Lake Wylie temperature is greater than 95.5°F, the emergency procedure for transfer of Emergency Core Cooling System (ECCS) flow paths to cold leg recirculation directs the operator to align both trains of containment spray to be cooled by loops of NSWS which are aligned to the SNSWP. Swapover to the SNSWP is required at 95.5°F rather than 95°F because Lake Wylie is not subject to subsequent heatup due to recirculation, as is the SNSWP. Therefore, the 100°F design basis maximum temperature is not approached.

ES-1.3 Transfer to Cold Leg Recirculation Enclosure 2 Step 13 Directs the operator to verify adequate heat sink by determining if the RN (NSWS) system is aligned to Lake Wylie, and if it is, to verify Lake Wylie temperature is less than 93°F. If RN is aligned to Lake Wylie, and temperature is greater than 93°F the operator is directed to align both trains of RN to the SNSWP.

The actions required by SLC16.9.4 and the statement in the basis document requiring the operator when swapping to cold leg recirculation and the actions listed in ES-1.3 indicates that the rising temperature will have an effect after the ice has been depleted.

These actions also indicate that there is more than a minimal impact during the entire accident sequence. However, it is difficult to determine if the effects of the increased Lake Temperature will be “significant.” Because the word significant is subjective in nature the NRC has determined that there is not a correct answer and this question will be deleted.

Written Examination - Question # 89

Licensee's Comment:

The developer was considering that to isolate a ruptured S/G (RSG) a level of > 11% is a precondition that must be satisfied. However, to completely isolate a RSG steps 3 – 6 within EP/E-3 must be performed, and step 6 which completes the isolation can only be performed if RSG level is > 11%.

The question didn't differentiate between initiating the isolation of an RSG and completely isolating an RSG.

Answer A is correct because, once steps 3, 4, & 5 are reached; the operator is required to perform these actions as soon as the RSG is identified. There are no preconditions to performing these steps.

Answer B is correct because it is part of the guidance which completes the isolation of the RSG by isolating the auxiliary feedwater supply when level is > 11%.

If the question had asked for the guidance to completely isolate the RSG, then there would be no correct answer to the question; however, the question asked for the procedural guidance regarding isolation which is found in both answers A & B, so both answers A and B are correct. We request that both answers A and B be accepted as correct.

NRC Response:

The NRC does not agree with accepting both A and B as correct answers. The term isolation is not defined in the procedures, and was not defined in the stem. The NRC does agree that the stem of the question did not ask for when to start the isolation, or when the isolation would be considered complete. Answer A is not correct because the steam generator would not be totally isolated, however answer B is not correct either because the NC system cooldown could begin even if the affected steam generator was not isolated, and its level was less than 11% narrow range level. Therefore, none of the answers are correct, and the question will be deleted.

SIMULATOR FIDELITY REPORT

Facility Licensee: Catawba Nuclear Station

Facility Docket No.: 05000413, & 05000414

Operating Test Administered: December, 01- 04, 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 Inspection Procedure 71111.11 are not indicative of noncompliance with 10 CFR 55.46. No licensee action is required in response to these observations.

While conducting the simulator portion of the operating test, examiners observed the following:

Item	Description
Reactivity Response differences between scenarios.	During performance of scenario # 3, (EOL boron concentration of 215 ppm) the simulator exhibited different responses to the amount of dilution. One crew diluted only 400 gallons of water to get the required temperature increase to allow the crew to commence an increase in power, and two other crews did not receive the same temperature response after diluting over 2700 gallons of water.
Simulator ANSI limits exceeded	Several times during the performance of JPMs the simulator gave the indications that it was outside of its ANSI limits. Simulator operators had to override the issue to allow the JPMs to be completed. Simulator Work Request SGB-009 submitted.
Critical Scenario/Simulator Data not captured.	The simulator parameter data collection required to be saved during the administration of the operating test scenarios was not saved during the first scenario, either due to personnel error or equipment malfunction.